

Summary of Rocky Flats Plant Waste Buried in the Subsurface Disposal Area

Edward Vejvoda

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**Los Alamos Technical Associates, Inc.
Aurora, Colorado 80014**

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ABSTRACT

This report addresses characteristics, nomenclature, and generating processes associated with radioactive and mixed waste shipped from Rocky Flats Plant to Idaho National Laboratory from 1954 to 1989. Until 1970, the waste was buried in the Subsurface Disposal Area, located in the Radioactive Waste Management Complex at Idaho National Laboratory.

The report reviews the types of waste, packaging, assay methods, transportation arrangements, waste identification labeling, and communications involved between Rocky Flats Plant and organizations at Idaho National Laboratory. An extensive set of appendixes is provided consisting mainly of Rocky Flats Plant documentation pertinent to the topics being presented.

FOREWORD

The author has 52 years of experience in the areas of plutonium and uranium operations gained primarily at Rocky Flats Plant. He was an active employee at Rocky Flats Plant from 1952 to 1987 and a consultant for 17 years. Mr. Vejvoda started employment at Rocky Flats Plant in June 1952 and retired in 1987. He returned as a consultant to Rockwell, EG&G, and Kaiser-Hill contractors until 2003. He holds bachelor's and master's degrees with a major in chemistry and a minor in physics.

His work experiences and positions include an initial assignment in 1952 to the spectroscopy laboratory for highly enriched uranium in Building 881, followed by advancement to the Plutonium Analytical Methods Development Group in 1956. From 1965 to 1974, he was the manager of the Chemical Technology Group, which carried out special projects and recovered plutonium and other actinides from scrap and residues not acceptable for the regular plutonium recovery stream. The Chemical Technology Group generated the majority of waste contaminated with Np-237, Cm-244, U-233, and other actinides used in the radiodiagnostic tracer program. The Plating Laboratory—located in Building 444 and part of the Chemical Technology Group—and the Special Recovery Group were also under his supervision.

In 1974, he became Director of Chemical Operations, which included plutonium recovery operations (Building 771) and the molten salt extraction process located in Building 776. Later, he was assigned the responsibility of director of all plutonium operations, which included pit manufacturing and assembly, plutonium recovery, pyrochemistry processing, and waste management. The waste management assignment consisted of solid waste processing, liquid waste treatment, and waste packaging and shipping.

During his operational tenure, he experienced and worked with the gradually increasing stringency of requirements of waste management practices. His experience with plutonium recovery and special recovery operations provided insight into the generation of both solid and liquid waste treatment and waste shipping requirements. Plutonium recovery activities were the major processes contributing to waste shipped to Idaho National Laboratory.

He has served as consultant to Idaho National Laboratory personnel in the areas of waste identification, waste shipping records, and Rocky Flats Plant operations and facilities. Over a 52-year period, he has experienced the Rocky Flats Plant startup, mission assignments, and decommissioning and cleanup phase.

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Personnel at Idaho National Laboratory, such as Rod Thomas, Bruce Becker, and Danny Anderson, were helpful through their questions and topics of interest. Many thanks to Vivian Schultz and staff, who produced this report and assisted with its organization, and to Ed May and Tasha Taylor for editing. This report was sponsored and encouraged by Jean Holdren, who also had the patience to see it through to completion.

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ACRONYMS

AEC	U.S. Atomic Energy Commission
CCl ₄	carbon tetrachloride
CWS	Chemical Warfare Service
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DU	depleted uranium
HEPA	high-efficiency particulate air (filter)
HEU	highly enriched uranium
INL	Idaho National Laboratory
MSE	molten salt extraction
NDA	nondestructive assay
NRTS	National Reactor Testing Station
OASIS	Organic Accelerated Solidification and Immobilization System
ORNL	Oak Ridge National Laboratory
PCE	tetrachloroethylene
PVC	polyvinyl chloride
R&D	research and development
RCRA	Resource Conservation and Recovery Act
RFP	Rocky Flats Plant
SNM	special nuclear material
TCE	trichloroethylene
TRU	transuranic

Summary of Rocky Flats Plant Waste Buried in the Subsurface Disposal Area

1. INTRODUCTION

Rocky Flats Plant (RFP)^a shipped solid, radioactive waste to Idaho National Laboratory (INL), located near Idaho Falls, Idaho. The waste was contaminated with minor amounts of weapons-grade plutonium, highly enriched uranium (HEU) (Oralloy), depleted uranium (DU), and toxic chemicals.

Shipping of waste began in April 1954 and continued into late 1989. Waste from RFP was deposited underground in a series of pits and trenches until 1970, when the U.S. Atomic Energy Commission (AEC) policy was implemented requiring segregation and retrievable storage of all solid transuranic (TRU) waste. After 1970, TRU waste received from RFP was placed in aboveground, earthen-covered retrievable storage. The aboveground stored waste was designated as TRU retrievable waste.

Because the definition of TRU waste changed in 1982, it is important to note that a large portion of the waste previously designated TRU is not TRU by today's definition. Originally, TRU waste was defined as all waste contaminated with TRU radionuclides in concentrations greater than 10 nCi/g (AEC 1973). However, in 1982, TRU waste was redefined based on a concentration of 100 nCi/g (DOE O 5820.1). Today, TRU waste is defined as waste material containing any alpha-emitting radionuclide with an atomic number greater than 92, a half-life longer than 20 years, and a concentration greater than 100 nCi/g at the end of the period of institutional control as defined in "Radioactive Waste Management Manual" (DOE M 435.1-1).

The waste from RFP was buried at the Subsurface Disposal Area, a radioactive waste landfill located in the Radioactive Waste Management Complex at INL. A baseline risk assessment and range of remedial alternatives are being developed under requirements of the Comprehensive Environmental Response, Compensation, and Liability Act (42 USC 9601 et seq., 1980) and associated *Guidance for Conducting Remedial Investigations and Feasibility Studies under CERCLA* (EPA 1988). In estimating risk and evaluating feasibility of various remedial alternatives, details of location and characteristics of radioactive waste in the landfill are essential.

The RFP shipping records, however, provided meager descriptions of waste shipped for burial. Likewise, INL receiving records for RFP buried waste were limited in characterization data. Consequently, INL personnel involved in developing the foundation for remedial decision-making initiated requests and inquiries to identify the types of RFP waste shipped and to document characterization data. These requests and inquiries were directed to Los Alamos Technical Associates, Inc., through a contractual arrangement. Los Alamos Technical Associates had several former RFP employees on their staff who were familiar with waste operations in the 1950s and 1960s and who had access to archival waste documentation that addressed shipping and characterization data. This report is a compilation of the data and information sent to various INL technical personnel. Consolidating these INL communications within this report provides a useful reference for present and future INL personnel.

a. Rocky Flats Plant is located 26 km (16 mi) northwest of Denver, Colorado. In the mid-1990s, it was renamed Rocky Flats Environmental Technology Site. In the late 1990s, it was again renamed to its present name, Rocky Flats Plant Closure Project. Most of the transuranic waste buried in the Subsurface Disposal Area originated at Rocky Flats Plant.

The report mainly concentrates on pre-1970 activities but also provides information related to post-1970 waste of interest. Appropriate background information is included to clarify the source and nature of the waste shipped. The following topics are addressed:

- Waste operations and facilities
- Waste characterization
- Waste containerization
- Waste assay
- Waste transportation and forms
- Special-order work
- Classified waste
- Special topics.

The majority of the report's text originated from documents, reports, memorandums, and other plant communications located in RFP archives. Supplemental information came from knowledgeable past RFP personnel.

Topics selected for this report were based on the author's judgment as to the data and information that INL personnel may find useful and on requests received from INL personnel. Many minor topics are not addressed that were judged of minimal importance and little consequence to remedial decision-making at INL.

Also included are a combined timeline of major occurrences at both RFP and INL, a list of correspondence from the author to Operable Unit 7-13/14 staff, a list of references cited throughout the report, and an extensive set of appendixes. The reference list contains sources that are readily available outside this report; the appendixes contain items such as reports, letters, and certificates that are not otherwise available. Each printed copy of this report will include a CD containing appendix content.

2. BACKGROUND

Construction of RFP started in 1951 and was completed by 1953. Plant operations began in 1953. Additional construction continued to meet the plant’s changing War Reserve mission. The chief War Reserve mission was to fabricate components of nuclear weapons. The main products manufactured by RFP were the so-called triggers for thermonuclear weapons. At RFP, these plutonium triggers were referred to as pits for their resemblance to a fruit pit. A secondary mission was to fabricate and assemble special nuclear devices for testing at Nevada Test Site. Rocky Flats Plant also was assigned the task of disassembling obsolete returned pits to recover and recycle special nuclear material (SNM) components. The AEC selected Dow Chemical Company to operate RFP.

2.1 Rocky Flats Plant Facilities (1953–1969)

Rocky Flats Plant was originally organized and constructed based on radioactive materials processed and handled. Radioactive materials were weapons-grade plutonium, HEU, and DU.

Production facilities were the four plants listed in Table 1. Individual buildings within each plant, their operations, and materials processed also are included in Table 1.

During the 1950s, building numbers were composed of two digits such as Building 44, Building 71, and Building 81. Later, building numbers were changed to three digits. Consequently, the original two-digit building numbers became 444, 771, and 881 by placing the additional digit at the beginning of the building number. Waste shipping records for the 1950s used the two-digit building numbering system while later records employed the three-digit system. This report will use the three-digit building numbering system.

Initially (1953–1956), all plutonium operations were carried out in Building 771. In 1957, Buildings 776 and 777 were completed to accommodate plutonium technological changes and new pit designs.

Table 1. Rocky Flats Plant production plants 1953–1970.

Plant A – Manufacture nuclear weapon components of nonspecial nuclear material	
Materials –	Depleted uranium, depleted uranium alloys, aluminum, beryllium, stainless steel, copper, and other metals in minor amounts
Buildings –	444, 447, 883A, and 441
Operations –	Foundry, machining, heat treating, and inspection (444 and 447) Rolling and forming (883A) Analytical laboratory (441)
Plant B – Manufacture nuclear weapon components of highly enriched uranium (Oralloy)	
Materials –	Highly enriched uranium
Buildings –	881 and 883
Operations –	Foundry, machining, inspection, chemical recovery, and metal recycle (881) Rolling and forming (883B) Analytical laboratory (881)

Table 1. (continued).

Plant C – Manufacture nuclear weapon components of plutonium	
Materials –	Weapons-grade plutonium
Buildings –	771, 776, 777, 779, and 774
Operations –	Foundry (771 and 776) Machining (771 and 776) Inspection and assembly (777) Chemical recovery and metal recycle (771) Pyrochemistry (776) Liquid waste treatment (774) Research and development (779) Analytical laboratory (771 and 559)
Plant D – Pit assembly and certification	
Materials –	Plutonium, highly enriched uranium, depleted uranium, and other nonradioactive materials such as beryllium, stainless steel, and aluminum
Building –	991
Operations –	Assembly, inspection, certification, packaging, and shipping assemblies

New construction caused foundry and machining operations to be transferred to Building 776. Likewise, assembly and certification operations were transferred to Building 777. Vacated foundry and machining facilities in Building 771 were taken over by the Research and Development (R&D) Metallurgy Group. Vacated facilities in Building 991 were taken over by the Physics R&D Group.

Manufacturing of HEU components in Building 881 terminated in 1964; however, cleanout of HEU material continued for several more years. Rolling and forming operations in Building 883B were terminated also. The HEU area in Building 883B was converted to DU and beryllium operations. Vacated HEU areas in Building 881 were used for R&D projects and manufacturing of nonradioactive reservoir components.

Building 889 was built in the late 1960s to allow decontamination of process and machining equipment from Building 881 that had been contaminated with HEU. Most of the decontaminated equipment was reused by the plant maintenance group and the plutonium buildings.

With the addition of plutonium Buildings 776 and 777, the capability and capacity of the plutonium analytical laboratory located in Building 771 were inadequate for fast analyses of plutonium. Consequently, Building 559 was constructed in 1967 to expand plutonium analytical services to accommodate production demand. Building 559 became the primary laboratory for plutonium analyses.

In 1963-1964, expansion and upgrade of the plutonium chemical recovery systems in Building 771 were enabled by transferring support services—such as laundry, cafeteria, and offices—to new facilities next to the building. The laundry was moved to Building 778.

The plutonium upgrades and additions described above increased the quantity of plutonium-contaminated waste shipped to INL. The termination of HEU activities in the middle and late 1960s decreased the amount of HEU-contaminated waste shipped to INL.

Plant and production support groups contributed only a few drums per year to the waste sent to INL. The health physics laboratory in Building 123 generated mainly waste from bioassay and HEU.

The medical treatment facility in Building 122 handled cases of severe contamination by plutonium and HEU. Waste was usually transferred to Building 123 for disposal.

The general plant analytical laboratory was located in Building 441. The chief contaminant was uranium from DU and DU alloys. Natural thorium samples were analyzed occasionally for special projects. Building 441 generated a few drums of waste per year.

A special project, using natural thorium and DU, took place in Building 331 using temporary facilities. Several drums of thorium and DU waste were generated and shipped to INL.

Building 995 was the plant sanitation treatment facility. Occasionally, sanitation sludge would contain plutonium concentration levels requiring shipment to INL.

2.2 Additions to Rocky Flats Plant Facilities (1970–1989)

Nuclear weapon design changes and increased production demands from the Cold War caused expansion of production facilities at RFP. In addition, plutonium facilities constructed in the 1950s were almost 20 years old.

Building 707 was completed in 1970 to meet new pit designs that could not be manufactured in Buildings 776 and 777. In 1971, an annex was added to Building 707 to accommodate all of the operations in Buildings 776 and 777 as a result of the 1969 fire in the Building 776 foundry.

After cleaning up from the 1969 fire in Building 776, production operations were limited to special projects and disassembly of returned pits. The main focus of Building 776 became waste and residue treatment. A manual size-reduction facility was established in a previous plutonium storage vault as an outgrowth of the 1969 fire recovery operation. The size-reduction facility was a large generator of TRU waste sent to INL in the 1970s and 1980s for retrievable storage.

Building 444 remained a constant from 1970 until 1989 when DU and beryllium operations ceased; however, a production plating laboratory was built in 1981 on the second floor of Building 444. The electromachining and chemical milling processes generated a variety of liquid waste with high salt content.

With the advent of nondestructive assay (NDA) drum counters, a facility for drum counting and handling was built in 1971 between Buildings 771 and 774. This addition was designated as Building 771C and handled the majority of drum counting analyses for SNM.

Terminating HEU component manufacturing freed facilities in Building 881. Therefore, the analytical laboratory in Building 881 received the workload of the general plant analytical laboratory from Building 441. Building 441 was converted to offices for plant engineering, and DU waste was no longer generated in that building.

After 1970, pyrochemical procedures expanded. Building 776, housing pyrochemical production, was expanded to accommodate molten salt extraction (MSE) of Am-241 from returned pits. This facility generated spent salt, tantalum, and plutonium- and americium-contaminated waste from magnesium oxide crucibles.

To accommodate a real-time radiography unit, a second-generation NDA crate counter, and drum counters, Building 569 was built in 1987 and was ready for use in 1988. This facility generated very little TRU or other waste that was sent to INL.

New construction of Buildings 371 and 374—completed in 1980 and 1978, respectively—was justified by the age of Buildings 771 and 774 and expansion of pyrochemical processes in Building 776. Building 374 treated liquid waste from Building 371, effluent from the first precipitation stage from Building 774, and other plant-generated waste. Building 371 experienced technical difficulties in the chemical recovery of plutonium and was remodeled to accommodate the new accountability criteria for SNM. However, electrorefining of plutonium metal to a high purity level remained routine until 1989, and Building 374 successfully handled the majority of the plant liquid waste.

2.3 Beryllium Processing

Production of beryllium components began in 1957 and consisted of machining and inspection of beryllium forms supplied by offsite vendors. A wrought beryllium process was developed at RFP in the mid-1960s to recycle beryllium metal scrap into cast beryllium forms available for machining. While beryllium is not radioactive, it was often commingled with DU and other radioactive materials shipped to INL.

In 1975, offsite vendors began supplying beryllium blanks that required a minimal effort to machine into acceptable beryllium components. Consequently, the recycling and casting of beryllium at RFP ceased. The beryllium blanks provided by offsite vendors were composed of sintered beryllium, which contained 5–6% beryllium oxide. Eliminating the wrought process after 1975 significantly reduced the beryllium waste generated by Buildings 444, 447, and 883.

Beryllium components were handled and assembled into configurations in Buildings 707, 776, and 777. Returned pits were disassembled in Building 777. These activities generated trace amounts of beryllium in some of the waste shipped to INL.

Mixtures of beryllium and plutonium were processed in Building 771. The beryllium was dissolved in a mixture of nitric and sulfuric acid. The spent acid with the soluble beryllium was transferred to Building 774 for processing into a sludge for shipment to INL.

3. WASTE OPERATIONS AND ORGANIZATION

In January 1953, a waste disposal organization was established to supervise the ultimate disposal of processed liquid and solid waste and to gather and correlate disposal data. Initially called the Waste Disposal Unit, its title was changed to Waste Disposal Coordination Group to reflect its function more closely.

3.1 Waste Disposal Unit

The original staff was one full-time chemist with oversight by an analytical laboratory manager; however, by April 1954, the full-time staff increased to two with occasional support from a third employee from the general analytical laboratory. Waste was treated and packaged by operations personnel. The actual shipping was handled by the traffic group. The Waste Disposal Coordination Group did not have any facilities under its jurisdiction.

While staffing remained at two full-time employees, the workload increased significantly. By March 1954, 2,457 (30-gal) drums of solid waste had accumulated. An arrangement to ship solid radioactive waste to INL was authorized in April 1954 (see Appendix A). The first shipment to INL was made in April 1954 and was composed of 343 drums with a gross weight of 15,829 kg (34,896 lb).

The responsibilities of the Waste Disposal Coordination Group were the chemical, radiological, and physical states of the plant collection ponds and tanks; storage and disposal of contaminated waste; collection of waste data; and coordination of waste projects. This group also authorized the release of compliant wastewater from the plant site.

The Waste Disposal Coordination Group was attached to the analytical laboratory organization since its inception in 1953. In January 1965, the group was transferred to the Health Physics Group. In September 1970, the group's title was changed to Health Physics Waste Disposal. The staffing still remained at two full-time employees. In August 1971, the group title was changed to Waste Management Waste Disposal.

3.2 Waste Operations

In the early 1970s, safe disposal of radioactive waste became a national issue. In response, the AEC established new definitions of radioactive waste based on radiation levels. Two classes of radioactive waste pertinent to RFP were TRU waste and non-TRU waste. Because the definition of TRU waste changed in 1982, it is important to note that a large portion of the waste previously designated TRU is not TRU by today's definition (see Section 1). Upgrading the importance of radioactive waste to almost a product level and establishing rigid acceptance criteria for waste disposal placed a burden on plant operations.

In the mid-1980s, radioactive waste with a hazardous component was defined as mixed waste. The hazardous component was regulated by the Resource Conservation and Recovery Act (RCRA) (42 USC § 6901 et seq., 1976), while the radioactive component remained under jurisdiction of the U.S. Department of Energy (DOE). Consequently, part of the buried radioactive waste at INL could be designated as mixed waste.

To accommodate the increased emphasis on quality and packaging of radioactive waste, the Waste Operations Group was organized that included the original Waste Disposal Coordination Group. The Waste Operations Group still reported to the Health Physics Group.

The plant operating groups still packaged the solid radioactive waste but were required to comply with standard practices issued by the Waste Operations Group. Waste inspectors were provided by the Waste Operations Group to ensure compliance. Storing and loading of radioactive waste for offsite shipment were taken over by the Waste Operations Group. The liquid waste treatment facilities, Buildings 774 and 374, were transferred to the Waste Operations Group.

Operational experience, coupled with a significant enlargement of waste-associated activities and requirements, called for an independent waste operations group, which was formed in the late 1970s. The enlarged Waste Operations Organization consisted of a Solid Waste Operations Group, a Liquid Waste Operations Group, and a Waste Management Group. The Waste Management Group was responsible for the waste quality program, waste training, waste shipments, and recordkeeping.

4. WASTE CHARACTERIZATION

Rocky Flats Plant fabricated components of nuclear weapons from plutonium, HEU (Oralloy), and DU. These fabrication operations generated both liquid and solid contaminated waste. Liquid waste was either (1) aqueous-based solutions or (2) organic-based solutions. The solid waste consisted of the following five types:

- Type I—combustibles: paper, rags, wood, and plastics
- Type II—filter paper
- Type III—Chemical Warfare Service (CWS) filters
- Type IV—sludge
- Type V—noncombustibles: glass, brick, scrap metal, ceramics, and graphite.

These five types for solid waste were used from 1954 to 1970. (Section 4.2 describes an additional five categories [1 through 5], established mainly to describe generation of plutonium-contaminated waste, but also often employed to describe HEU-, DU-, and beryllium-contaminated solid waste. The five categories are not the same as shipping Types I through V above, although they are similar in some respects.)

Later, waste to be shipped was described and identified by item description codes. Machine coolants and other process liquids were filtered using filter paper that was classified as Type II waste. The use of filter paper declined in the 1960s, which significantly reduced this type of waste. Type III CWS filters refers to CWS filters that were used in building ventilation systems. The CWS filters were eventually replaced by high-efficiency particulate air (HEPA) filters. The Type IV sludge mainly refers to the series of sludge produced by the Liquid Waste Treatment Plant (Building 774). Other sludge in minor quantities came from process and cleanout building operations such as “still bottoms” (i.e., residue from distilling processes), degreasing bath residues, tank deposits, spent vacuum pump oil, and equipment maintenance.

4.1 Liquid Waste Treatment

The treatment of liquid waste accomplished four objectives:

- Removal of radioactive constituents from aqueous waste to allow solidification for disposal offsite
- Removal of chemical constituents from aqueous waste to satisfy drinking water standards and allow discharge offsite or reuse onsite
- Solidification of nonconforming aqueous waste for disposal offsite
- Solidification of organic liquid waste for disposal offsite.

Two liquid waste treatment plants—Buildings 774 and 374—were built at RFP. Building 774 was the initial plant and began operations in the 1952–1953 timeframe. Building 374 replaced the majority of Building 774 treatment operations, although organic liquid waste treatment remained in Building 774 along with support for Building 771. Building 374 came into use in the late 1970–1980 timeframe and did

not contribute to buried INL waste. Consequently, Building 374 treatment processes will not be addressed in this report.

The Liquid Waste Treatment Plant, Building 774, was built next to Building 771 to facilitate treating aqueous solutions generated by plutonium recovery operations in Building 771. Building 774 also treated radioactively and chemically contaminated aqueous waste generated by other plant activities. Consequently, Building 774 treated aqueous solutions contaminated with HEU, DU, and plutonium. The majority of aqueous waste solutions received for treatment were nitric-acid based. Plutonium aqueous waste solutions contained trace amounts of Am-241. Neptunium-237, U-233, and Cm-244 were received occasionally in trace-to-minor amounts from special-order projects. The major cations found in the waste solutions were aluminum, calcium, iron, potassium, magnesium, sodium, and silicon. The major anions were nitrate, sulfate, and chloride.

The contaminated solutions transferred to Building 774 were analyzed before being transferred either by pipeline or container for treatment. The SNM and DU content were the basis for normal operating loss estimated by the Nuclear Materials Management Group. Solutions received by pipeline were directed to designated receiving tanks based on their acidic, radiological, and chemical contents. The solutions received through drums and other containers were siphoned into receiving tanks or treated directly.

The treatment process in Building 774 used a two-stage, ferric hydroxide carrier precipitation process for radiolytic decontamination. To make the precipitating agent, ferric sulfate, calcium chloride, and a coagulating agent were added to a specific volume of water. The precipitating agent was then added to the acidic waste solution that was made basic (pH 11) with sodium hydroxide. Ferric ions combined with hydroxide ions to form a hydroxide floc that acted as a scavenger to remove radiolytical contaminants. Calcium ion overpowers any peptizing agents in waste solutions.

The neutralized solution with the precipitated slurry was pumped to a precoated rotary drum vacuum filter, which separated liquids from solids. The collected sludge was skimmed from the rotary drum filter through a knife-blade arrangement into a prepared drum for shipment offsite. This drummed sludge containing the bulk of the radioactive constituents was identified as first-stage sludge and designated as Series 741 sludge. The first-stage effluent was collected as feed for the second-stage precipitation. Plutonium waste solutions underwent two stages of precipitation while other plant waste solutions were treated through second-stage precipitation processes. Sludge collected from second precipitation was identified as second-stage sludge and designated as Series 742 sludge.

The filtrate effluent from the second-stage through the rotary vacuum drum filter was analyzed for its radiological and chemical content. If too high in radioactivity, the effluent was reintroduced to the second-stage precipitation process. If chemical content (mainly nitrate) was too high, the effluent was pumped to solar evaporation ponds, which were next to Building 774.

4.1.1 First- and Second-Stage Sludge

Sludge removed from the rotary vacuum drum filter was about 70 wt% water based on periodic analyses. Average concentration for first-stage sludge was $1.07 \times E-05$ g/g or $3.48 \text{ 1E-}05$ Ci/g for americium and $4.71 \text{ E-}05$ g/g or $3.53 \text{ E-}05$ Ci/g for plutonium (see Appendix X).

First- and second-stage sludge was loaded into 17C or 17H steel drums of mostly 55-gal capacity, but at least 30-gal capacity. The maximum weights acceptable were 660 lb for 17H drums and 880 lb for 17C drums. Use of 17H drums was discontinued in favor of 17C drums for Building 774 sludge because of the higher maximum weight limit for 17C drums. The changeover started in the middle 1960s and was

completed by the late 1960s. A quantity of dry Portland cement was placed in the bottom of the drum. A polyethylene liner was positioned inside the drum. Additional dry Portland cement was interspersed with the filling sludge. After sealing the liner, additional dry Portland cement was placed on top of the liner. The filled drum was sealed, weighed, labeled, logged, and surveyed for surface contamination and external radiation levels.

Occasionally, first- and second-stage sludge was mixed to meet certain shipping requirements. This type of sludge was designated as Series 7412 sludge. Building 774 was expanded with an addition that was designated as 74A. Starting up the organic sludge processing unit located in the 74A addition produced experimental sludge drums that were identified as 74A sludge drums. Later, this sludge product was designated Series 743 sludge.

4.1.2 Off-Specification Waste Solutions

Aqueous waste solutions that did not meet feed specifications for first- and second-stage treatment were processed directly. These waste solutions contained objectionable constituents such as complexing agents, hazardous chemicals, and certain radioactive isotopes not normal to the plant. High chloride solutions, such as hydrochloric acid solutions, were also candidates for this type of treatment. These solutions were solidified directly with Portland cement. Acidic solutions were made basic before adding the cement.

Special solidification drums were prepared using a mixture of Portland cement and an absorbent material. The solidification drum was connected to the solidification glove box through an O-ring drum liner arrangement. The basic waste solution was added to the prepared solidification drum; the Portland cement then reacted with the added solution to form a solid. The added absorbent material aided distribution of the waste solution within the drum. A maximum of 94.6 L (25 gal) of waste solution could be solidified per drum. Waste solution received in small volumes (bottle containers) was often treated directly and placed in a prepared drum. The disposition of off-specification waste was based on quantities received and operating experience. Off-specification waste was designated as Series 744 sludge, sometimes referred to as special setups.

4.1.3 Evaporator Salts

Treated solutions high in chemical salts but meeting radioactive levels were stored in solar evaporation ponds next to Building 774. Consequently, an evaporator system was added to Building 774 composed of an evaporator, a double drum dryer, a dust scrubber system, and a steam condensate collector. A steam-heated heat exchanger was employed as the heat source for the evaporator. Water vapor generated by the evaporator was exhausted to the atmosphere through baffles and entrainment separation pads. Concentrated salt liquid from the evaporator was transferred to the steam-heated double-drum dryer. Remaining water was removed, leaving a film of dry salts baked on the rotating drum surfaces, which were then scraped using a knife-blade arrangement. The salts were collected in a catch container that was weighed and emptied into a wooden crate for shipment offsite. Evaporator salts, also called nitrate salts, were designated as Series 745 sludge.

4.1.4 Contaminated Drums

Contaminated empty drums were shipped in crates and designated as Series 746 sludge. These drums were rinsed with an appropriate solvent to reach a contamination level of <3 g of plutonium.

4.1.5 Treatment of Organic Liquid Waste

Organic liquid waste was composed mainly of a variety of oils and solvents. The types of oils received for treatment were basically cutting, lubricating, hydraulic, and vacuum pump oils. Solvents were used as degreasing and cleaning agents. Several organic liquids were employed for density measurements on machined parts. The analytical and R&D laboratories generated small volumes of contaminated organic liquids with a variety of extraction agents. The largest contributor to organic liquid waste was spent lathe coolant generated by plutonium machining operations. The majority of waste solvents were basically chlorinated and fluorinated hydrocarbons. Plutonium lathe coolant was diluted with carbon tetrachloride (CCl₄) to 30% oil and 70% CCl₄. However, the percentage of CCl₄ remaining in spent lathe coolant received in Building 774 was reduced by evaporation and probably ranged between 25 and 60%.

Organic liquid waste was treated by mixing the organic liquid with an absorbent powder to form a greaselike substance. Consequently, this operation was referred to as the “Grease Plant” or “Jelly Factory.” Microcel, manufactured by Johns-Mansville, was the absorbent material used and was mainly calcium silicate. The mixing ratio was 45 kg (99 lb) of Microcel (three bags) to 190 L (50 gal) of organic liquid. Microcel obtained from other manufacturers did not mix very well. Therefore only Microcel from Johns-Mansville was used.

Mixing was accomplished by a blender (Readco Processor) that was enclosed within a glove box. Organic liquid and absorbent powder were piped into the mixer at controlled rates. On completing the blending process, the resulting greaselike mixture was discharged into a shipping drum attached to the glove box through an O-ring attachment. Loaded drums were sealed, weighed, labeled, logged, and surveyed for surface contamination and external radiation.

In the middle 1980s, the Grease Plant treatment of contaminated organic liquid waste was replaced with an improved solidification process identified as Organic Accelerated Solidification and Immobilization System (OASIS). This process is relevant to retrievably stored TRU waste, but not to buried waste. The constituents and their respective weights required to prepare the solidification medium for a given 55-gal drum are listed in Table 2.

Table 2. Constituents and weights for a 55-gallon drum.

Constituent	Weight (lb)
Contaminated oil	170
Emulsifier	25
Envirostone	250
Water	42
Total	487

Anderson et al. (1985) provide a more thorough description of the liquid waste treatment operations in Buildings 774 and 374.

4.2 Solid Waste Treatment

The five categories listed below were established mainly to describe generation of plutonium-contaminated waste, but also were often employed to describe HEU-, DU-, and beryllium-

contaminated solid waste. (Previously, the five Types I through V of waste shipped to INL were described in Section 4, "Waste Characterization." These descriptions were used by RFP to generally describe the types of waste shipped per container. However, RFP also identified waste by generation categories for operational purposes, waste generation statistics, and cost distribution. The five categories below are not to be confused with the shipping Types I through V although they are similar in some respects.) The five categories include:

1. Line-generated
2. Sludge
3. Filters
4. Maintenance operations
5. Non-line-generated.

4.2.1 Line-Generated Waste

Line-generated waste was produced by glove-box operations. This waste usually was highly contaminated with plutonium and required a plutonium assay to determine disposition status. If above an established economic discard limit, the waste was designated as recoverable residues; however, the majority of line-generated waste was composed of items used in operating and maintaining the line.

Contaminated items were placed in a drum that was attached to the glove-box line. The drum was equipped with a drum liner. When full, the drum was disconnected, sealed, labeled, weighed, surveyed for contamination, and transferred to storage to await assay. All line-generated waste was segregated according to waste Types I through V (see Section 4).

4.2.2 Sludge Waste

Liquid waste treatment processes carried out in Building 774 produced the majority of sludge waste. However, occasionally, contaminated sludge accumulated within a piece of processing equipment and was designated as a sludge waste according to the building where it was generated, such as Buildings 771, 776 or 777, 881, and 444 or 447.

4.2.3 Filter Waste

Filter waste refers mainly to ventilation filters used to remove airborne contamination. Large filters (2 × 2 × 1 ft) were used in the exhaust plenum systems, and small filters (12 × 12 × 8 in.) were used in intake and exhaust systems on glove boxes. These filters were assayed to determine whether they were above or below economic discard limits. If above discard limits, the filter medium was removed and processed to recover SNM; however, filters were not assayed and processed until the 1960s. Therefore, filters disposed of earlier may have contained concentrations of SNM higher than the economic discard limits.

4.2.4 Maintenance Operational Waste

Most contaminated waste generated by maintenance operations consisted of contaminated equipment and ancillary electrical and piping apparatus. A significant amount of this waste was packaged in wooden crates for shipment to INL.

Glove-box maintenance required enclosure by a plastic tent to control contamination. On completion of maintenance, these plastic tents were packaged as Type I waste.

4.2.5 Non-Line-Generated Waste

Non-line-generated waste was produced in process areas outside of the glove-box lines. This waste consisted of rags, absorbent wipes, surgical gloves, and other small, routinely used items. A significant amount of this waste was generated by housekeeping activities and had trace amounts of contamination.

4.3 Isotopic Levels in Waste

The four most prevalent types of radiological elements shipped to INL were: (1) weapons-grade plutonium, (2) HEU, (3) DU, and (4) Am-241. Americium-241 is the daughter product of the beta decay of Pu-241. The isotopic content of weapons-grade plutonium varied slightly from year to year as the mixture of returned plutonium and new plutonium from the Hanford and Savannah River Site reactors was not constant (see Table 3 for variations in the plutonium isotopic concentrations from 1959 to 1976). Improved mass spectrometry instrumentation provided lower detection limits for Pu-238 and Pu-242, providing values below 0.05 wt% rather than just a minimum of 0.05 wt%. This gave a more definitive evaluation of the isotopic content in the waste. See Table 4 for typical isotopic profiles for DU, HEU, and weapons-grade plutonium.

Table 3. Rocky Flats Plant plutonium isotopic levels in waste (stream averages—plutonium wt%).

Calendar Year	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
1959–1960	<0.0500	93.714	5.593	0.5932	<0.0500
1961–1962	<0.0500	93.817	5.486	0.5979	<0.0500
1963–1964	<0.0500	94.398	4.854	0.6482	<0.0500
1965–1966	<0.0500	93.586	5.823	0.5610	<0.0500
1967–1968	<0.0500	93.451	5.953	0.5670	<0.0500
1969	<0.0500	93.538	5.953	0.4790	<0.0500
1970	<0.0500	93.450	5.965	0.4850	<0.0500
1971	<0.0500	93.533	5.929	0.4380	<0.0500
1972	<0.0500	93.513	5.939	0.4480	<0.0500
1973	<0.0500	93.596	5.918	0.4300	<0.0500
1974 (first half)	<0.0500	93.571	59.000	0.4855	<0.0500
1974 (second half)	0.0104	93.656	5.893	0.4620	0.0317
1975	0.0102	93.707	5.861	0.3940	0.0266
1976	0.0102	93.827	5.814	0.3510	0.0219

Table 4. Typical isotopic concentrations in waste.

Material	Isotope	Weight Percent
Depleted uranium	U-235	0.30
	U-238	99.70
Highly enriched uranium (Oralloy)	U-234	1.02
	U-235	93.17
	U-236	0.44
	U-238	5.37
Weapons-grade plutonium	Pu-238	0.01
	Pu-239	93.63
	Pu-240	5.97
	Pu-241	0.37
	Pu-242	0.02

4.4 Waste Constituents

Previous discussion indicated that waste shipped to INL before 1970 was characterized by five types of materials. This same characterization will be followed in discussing waste constituents.

4.4.1 Combustible Waste—Type I

Contaminated combustible waste consisted mainly of rags, paper, plastics, wood, tape, rubber, and contaminated clothing, as described below.

Rags—Cotton-based rags were used mainly to wipe up spills and for cleaning purposes.

Wood—The major wood item was contaminated forklift pallets. Wood planking used in maintenance operations also contributed to the wood category.

Tape—Yellow vinyl tape and white masking tape were used extensively for closing plastic bags, for erecting plastic working tents (houses) for maintenance, and for many other applications.

Plastics—Polyvinyl chloride (PVC) and polyplastics were the chief plastic materials. Major plastic items were bags, tubing, and sheet forms. Tygon tubing was another form of plastic discard.

Paper—Paper items included absorbent wipes, laboratory filter paper, contaminated forms, packing paper, paper cartons, and miscellaneous paper articles.

Rubber—Major rubber items were surgical gloves and glove-box gloves. Minor rubber items were rubber stoppers, gaskets, and tubing.

Contaminated clothing—The majority of protective clothing was manufactured from cotton. Routine protective clothing consisted of coveralls, t-shirts, shorts, socks, surgical caps, and an occasional outside jacket. Cotton gloves were used extensively. Discarded booties (cotton canvas tops and rubber bottoms) were included in the discarded clothing.

Radioactive and hazardous material garments—Maintenance operations and decontamination efforts often required special protective garments and supplied breathing air for contact personnel. For contamination control, plastic tents (houses) were constructed to enclose the operational area. See Table 5 for a listing of these items.

Table 5. Personal protection garments used in working with radioactive and hazardous materials.

Date	Article	Material (description)	Use
1961 to early 1980s	Body suit	Tyvek (white)	Radiation worker
1961 to early 1980s	Hood	PVC (transparent)	Radiation worker
1961 to present	Gloves	Rubber (surgeon gloves)	Radiation worker
1961 to present	Gloves	Butyl (arm length)	Radiation/hazardous environments
1961 to early 1980s	Body suit	PVC (yellow rain suit)	Radiation/hazardous environments
1961 to present	Tape	Vinyl (yellow), masking (white)	Radiation/hazardous environments
Early 1980s to 1992	Encapsulated body suit with removable hood (Level A suit, SBAG)	Vinyl Technologies Corp., vinyl suit (orange), ABS valves (black), press-polish transparent vinyl hood	Radiation/hazardous environments
Early 1980s to present	Gloves	Rubber (orange, anticontamination clothing)	Radiation/hazardous environments
1992 to mid 1990s, alternating with Rich Industries Level B garment	Encapsulated body suit and hood (Level B garment)	Vinyl Technologies Corp., two films PVC with polyvinylidene fluoride (white), ABS valves (black), transparent vinyl hood	Radiation/hazardous environments
1992 to mid-1990s, alternating with Vinyl Technologies Corp. suit currently used at Rocky Flats Environmental Technology Site since 1995	Encapsulated body suit with hood (Level B garment)	Rich Industries, Sarenex/Tyvek body (white), vinyl valves (black), transparent vinyl hood	Radiation/hazardous environments
1961 to 1997	Tent (contamination control barrier)	Polyethylene (semitransparent) 6-8-ft widths	Radiation/hazardous environments

PVC = polyvinyl chloride

4.4.2 Filter Paper—Type II

In the 1950s, filter paper was used to remove particulates from lathe coolant and cutting oils. No significant information was located addressing this type of waste; however, the amount of Type II waste was minimal compared to the other four types. Filter paper was discontinued eventually in favor of other forms of machining filters, although Type II was used occasionally when warranted.

4.4.3 Chemical Warfare Service Filters—Type III

The original ventilation and glove-box filters used in the 1950s and 1960s were basically CWS filter designs. Waste designated as Type III indicated filters employed in ventilation systems and other air-filtered streams.

4.4.4 Sludge—Type IV

The bulk of sludge shipped to INL was generated by Building 774, the Liquid Waste Treatment Facility. However, equipment cleanouts in other buildings occasionally produced sludge-based material, which was designated as Type IV waste (see Section 12.19 for a description of sludge).

4.4.5 Noncombustible Waste—Type V

Noncombustible waste consisted of scrap metal, obsolete processing equipment, broken tools, glove boxes, ventilation ducts, piping, electrical wiring, lighting fixtures, and other metallic objects. Maintenance operations generated the majority of Type V waste. Other Type V waste forms were contaminated soil, concrete, macadam pavement, ceramics, glass, and graphite molds.

4.4.6 Waste from Buildings 444, 447, and 883

Waste generated by Buildings 444 and 447 was contaminated with DU and beryllium. Typical beryllium-contaminated waste included broken obsolete graphite molds and crucibles, tool bits, chucks, coolant, filters, sweepings, absorbent wipes, and other miscellaneous items. Large amounts of sandpaper and emery cloth were discarded as combustible waste (Type I). Work tables for coating molds were covered with butcher paper, which was discarded as combustible waste.

In addition to straight DU fabrication, several DU alloys were used in fabrication operations. The two main alloys were DU-niobium and DU-titanium. Niobium and titanium concentrations ranged from 1 to 6 wt% in the alloys. Consequently, DU-contaminated waste contained a combination of DU and DU alloys.

Depleted uranium alloys employed during the 1953–1979 and 1980–1989 timeframes are arranged in descending order of use with number 1 being the largest amount of DU material used.

Depleted uranium alloys employed during 1953–1979 include:

1. Unalloyed DU
2. Uranium-titanium alloys (U-0.75 wt% titanium)
3. Uranium-niobium alloys (U-2 wt% niobium) (U-6 wt% niobium)
4. Uranium-molybdenum alloys (U-2 wt% molybdenum) (U-4 wt% molybdenum)
5. Uranium-miscellaneous; R&D efforts and special-order work, no large quantities involved (U-1 wt% molybdenum—0.75 wt% titanium—1 wt% niobium and U-1 wt% niobium—0.5 wt% titanium—1 wt% zirconium).

Depleted uranium alloys employed during 1980-1989 include:

1. Unalloyed DU
2. Uranium-niobium alloy (U-6 wt% niobium).

Aluminum, copper, and lead were used as substitutes for DU in casting and back-machining operations. Roaster oxide (i.e., uranium chips, fines, and chunks oxidized in a furnace to uranium oxide) was a large contributor to the waste stream. Graphite molds and crucibles too large to fit into a 55-gal drum were placed in 4 × 4 × 7-ft crates. Mold coatings employed were yttrium oxide, niobium oxide, and aluminum oxide suspended in water, glass, and calcium fluoride.

Buildings 444 and 447 developed their own waste identification codes to segregate the waste. These letter codes were included sometimes on shipping load lists. See Table 6 for a list of these letter codes.

Table 6. Waste codes for Buildings 444 and 447.

Code	Description	Shipping Type
A	Filter paper	II
B	Coolant still bottoms	IV
C	Metal fire brick	V
D	Paper, rags, wood	I
E	Waste oil	N/A ^a
F	Graphite	V
G	Perclene still bottoms	N/A ^a
K	Process waste filter	IV
M	Cyanide cement	V
N	Miscellaneous solids	V

a. Not applicable. Waste was processed onsite and not shipped.

Any equipment or apparatus that moved or displaced air had a CWS or a HEPA filtering system. The exhaust from furnace vacuum pumps was filtered also. These spent filters were disposed of as Type III waste.

The vacuum furnaces were lined with metal bricks. These bricks were contaminated with DU and were discarded as Type V waste.

Protective clothing was provided to operating personnel. Discarded contaminated clothing included aprons, cotton gloves, asbestos blankets and gloves, hoods, jackets, coveralls, shin guards, and underclothing. These items were discarded as Type I combustible waste.

Fabrication processes generated a variety of oil-based and organic solvent waste. The majority of the oil waste was burned on plant site. The solvent waste was distilled for reuse when applicable. Still bottoms were sent to Building 774 for disposal through the Grease Plant, resulting in Series 743 sludge.

Aqueous-based lathe coolants were used with DU machining operations for fire prevention. These spent coolants were transferred to Building 774 for disposal by the second-stage precipitation process, resulting in Series 742 sludge.

Building 883 augmented the fabrication operations in Buildings 444 and 447. The DU ingots were received from Building 444 for rolling and forming in Building 883 Section A. Ingots were placed in a furnace or in a heated eutectic salt bath to prepare them for rolling. After rolling into sheet form, the sheets were annealed in a second salt bath. Appropriate shapes were cut from the annealed sheet for forming contoured parts that were returned to Building 444 for final machining operations.

Sheet trimmings and other DU residues were returned to Buildings 444 or 447 for recasting or for conversion to roaster oxide for shipment to INL. Cleaning solvents such as trichloroethylene (TCE) and tetrachloroethylene (PCE) were used in processing operations.

Spent TCE, PCE, and chlorofluorocarbons were sent to Building 774 for disposal through the Grease Plant. Spent salt baths with uranium oxide were packaged into drums and shipped to INL as Type IV sludge. Although Dow Corning 550 fluid was occasionally used as a cooling bath after being heat treated and sent to the Grease Plant in Building 774 for disposal, such disposal was infrequent.

Spent salt baths consisted of a mixture of sodium, potassium, and lithium carbonate contaminated with uranium oxide. The original salts were a white crystalline powder, but spent salts were colored by the spent oxides and turned grayish or blackish. The spent salt baths were packaged into drums and shipped to INL.

Building 883 Section B was dedicated to rolling and forming HEU items. The process employed was the same as described for DU, except that the original ingot came from Building 881. Spent salt baths were sent to Building 881 for recovery of HEU metal fines and oxide. Resulting combustible waste also was sent to Building 881 for incineration and subsequent recovery of HEU from the ash.

In 1965, HEU operations were terminated in Building 881 and 883, except for cleanup activities. Building 883 Section B remained idle until 1983, when fabrication of DU armor plate began for the U.S. Army. This operation continued into the 1990s.

In 1962, beryllium-forming processes were established in Building 883 Section A and continued until the mid 1980s. Beryllium ingots were cast in Building 444 and encapsulated in steel cans, which were heated and rolled into a sheet form. The steel can container was then cut away to remove the rolled beryllium sheet. The sheets were etched with acid—to remove microcracks and for thinning—in a bath of combined nitric and hydrofluoric acids. After etching, the sheets were heat treated in either an acid bath or a eutectic salt bath or both. Desired shapes were then cut and formed from the sheet. The formed beryllium part was sent to Building 444 for final machining.

The steel can cuttings were disposed of as noncombustible waste (Type V) that was shipped to INL. The spent acid baths were transferred to Building 774. In the late 1970s and 1980s, the spent acid baths were transferred to Building 374, the new Liquid Waste Treatment Plant. The spent eutectic salt baths were packaged into drums and sent to INL for retrievable storage.

The use of chlorofluorocarbons was discontinued in 1988. The PCE and TCE usage also was discontinued in favor of water and Oakite in the 1985–1986 timeframe. Initially, one of the large metal presses employed an oil that contained polychlorinated biphenyls. The oil was removed from the press and deposited in the polychlorinated biphenyl storage areas; however, any oil leaks from the press were probably absorbed onto rags or absorbent wipes, which were part of the combustible waste. The amount of this type of combustible waste was minimal.

Building 883 occasionally fabricated items from tantalum, titanium, stainless steel, cadmium, and aluminum. The majority of the generated scrap was collected and sold to offsite vendors or reused on

plant site. Any combustible waste with trace amounts of the above metals was mixed with routine-generated combustible waste.

4.4.6.1 Graphite Mold Failures. Occasionally, a mold coating would fail, and the molten beryllium or DU would react with the graphite and produce a hole. The molten metal would puddle in the bottom of the furnace through the hole in the graphite mold. Cleanup would include the melt, fire brick, graphite crucible, and all of the cleanup materials. These materials were consolidated into their own drums or boxes with no dilution from other materials. The DU was written off as a normal operational loss because it was an accountable material.

4.4.6.2 Rejected Depleted Uranium Slabs. Recast DU scrap produced an ingot $20 \times 24 \times 2$ in. that was called a slab. If the slab failed to meet the impurity specifications, it was discarded as waste. The slab was placed in a drum and surrounded by soft waste to secure the slab within the drum. Although the slab was listed as Type V, substantial quantities of Type I waste were also in the drum.

4.4.6.3 Asbestos Items. Asbestos items were very common in the Building 444 foundry. The asbestos items in the following list were used in foundry operations and discarded as waste—either Type I or Type V depending on the operator's discretion—when contaminated:

- Aprons
- 4×8 -ft fire blankets
- Gloves
- Jackets
- Hoods
- Shin guards
- Tape.

4.4.6.4 Spill Cleanup. The foundry used many large cooling water systems whose cooling water contained a chromate inhibitor. Cooling water that occasionally leaked was cleaned using rags and absorbent wipes and discarded as Type I waste.

The foundry also used large vacuum systems that required significant amounts of vacuum oils. Oil changes and leaks often generated Type I waste.

4.4.6.5 Miscellaneous Waste Materials. The following is a list of materials that were discarded as waste in trace and minor quantities:

- Grinding wheels and motors
- Unclassified tooling
- Cadmium plating turnings—back machining
- Chromium plating turnings—back machining
- Lead casting residues—skull and turnings
- Aluminum chips, turnings, and casting skull
- Copper turnings and casting skull

- Spent furnace fire brick
- Contaminated furniture.

4.4.6.6 Waste Segregation. Building 444 had a beryllium machine shop, a DU machine shop, and a foundry that accommodated both beryllium and depleted uranium. The waste generated by the machine shops was segregated, but foundry waste could be commingled. Although drums and boxes generated were not marked always as to their origin within Building 444, a commingled drum was always identified as a beryllium drum.

4.4.7 Building 881 Waste

The mission of Building 881 was to fabricate weapon components of HEU (Oralloy). Building 881 had the capabilities of a foundry, of machining and inspection, and of chemical recycling. The HEU residues and metal scrap were processed to recover uranium and produce pure uranium metal. The majority of combustible waste was incinerated, and HEU was recovered from the ash. Fabricating components of HEU began in 1953 and ended in 1965. Cleaning and removal of equipment continued into the late 1960s.

The HEU waste generated from operations was mainly graphite molds and crucibles. Magnesium oxide molds and crucibles were used initially (1953–1955) but were replaced by graphite molds and crucibles. Consequently, the noncombustible Type V waste shipped in 1954–1955 to INL could contain magnesium oxide crucibles.

A secondary waste item was formed when impure materials contaminated with HEU were ground to a fine powder, leached in nitric acid, filtered, and collected as solids. These solids were called mud, which was then dried, assayed, packaged into drums, and shipped to INL as Type IV sludge. Discarded nitric acid solution with trace amounts of HEU was transferred to Building 774 for second-stage processing or to the solar evaporation ponds. Initially, these waste solutions were set in concrete that was sent to INL. Removing HEU equipment in the late 1960s and early 1970s contributed to Type V noncombustible waste.

Beginning in the early 1960s, plutonium surface contamination on HEU components was removed in Building 881. The HEU units were sprayed with nitric acid to remove the plutonium and then washed with water to remove residual acid. Plutonium-free units were cut up for introduction to the casting process. After 1965, the HEU components were sent to the Oak Ridge National Laboratory (ORNL) Y-12 plant.

The leach solution collected from sprayed nitric acid was concentrated by evaporation and uranium and plutonium precipitated by adding ammonia. The precipitate was dried and calcined to an oxide. If the oxide was very low in plutonium (<1 ppm), the oxide was shipped to the ORNL Y-12 plant. Oxides above the plutonium limit were shipped to Savannah River Site. This process continued until 1974, when it was transferred to Building 771. The oxide generated from the Building 771 process was sent to the Idaho Chemical Processing Plant (now Idaho Nuclear Technology and Engineering Center) at INL in the 1980s, provided the plutonium content was <500 ppm.

Shell Vitera oil was the machining coolant employed in Building 881 and was circulated through a centralized system. Spent oil was filtered to remove any uranium fines and then packaged in drums for disposal. A limited quantity was burned, but the majority was processed through the Grease Plant in Building 774. Other degreasing solvents such as TCE and PCE were transferred also to the Grease Plant for offsite disposal.

Several special projects were carried out in Building 881. Neptunium-237 was introduced into HEU and DU components as a bomb fraction tracer for underground testing at the Nuclear Test Site. The U-233 components were fabricated for tracing purposes and other nuclear experiments. Both Np-237 and U-233 were received as oxides and converted to metal for casting. Waste generated was mainly graphite molds and crucibles and combustible waste. Combustible waste was not incinerated but shipped to INL as Type I waste.

Thorium-containing components were fabricated for a short time in the late 1950s to early 1960s. Scrap and residues were shipped to either Savannah River Site or ORNL. The only waste sent directly to INL was commingled combustible waste.

Building 881 had an analytical laboratory for control purposes. Wet chemical analyses and emission spectrographic analyses were performed on HEU products and residues. Waste generated was mainly combustibles and a few drums per year of noncombustibles. Chief noncombustible items were emission spectrographic graphite electrodes and spent laboratory equipment.

In the late 1960s, Building 881 was converted to fabricating stainless steel reservoirs. Waste generated by this operation was not sent to INL.

4.4.8 Buildings 122 and 123 Waste

Building 122 was a medical facility, which treated industrial injuries, decontaminated personnel, and carried out routine physicals. Decontamination activities generated discarded combustibles that were shipped to INL at a level of a few drums per year.

Building 123 was a health physics laboratory, which carried out bioassays and engaged in low-level radiological studies. These efforts generated a few drums per year of combustible waste that was shipped to INL.

4.4.9 Building 991 Waste

Building 991 was originally constructed to carry out three functions: (1) shipping and receiving SNM, (2) storing SNM, and (3) assembling nuclear weapon components. Assembly operations were discontinued in 1957 and transferred to Building 777; however, a few components were assembled in the 1960s on a special basis. Building 991 continued to serve as a shipping, receiving, and storing facility for SNM into the 1990s. In addition, the building housed a metallurgical laboratory and other R&D laboratory facilities.

Functions carried out in Building 991 generated mainly combustible waste contaminated with trace amounts of HEU, DU, beryllium, and occasionally plutonium. Limited quantities of degreasing and cleaning solvents also were used. The metallurgical laboratory employed small amounts of isopropyl alcohol, carbide grinding paper, metal etching solutions, and nonhazardous polish solutions. Waste generated included cutting fines from sample preparation that were processed on plant site if the fines were HEU, DU, or plutonium.

The quantity of waste generated by Building 991 and shipped to INL was limited to less than 100 drums per year during the 1950s and less than 50 drums per year from 1960 to the end of production.

4.4.10 Building 886 Waste

Building 886 was essentially a laboratory facility constructed for nuclear criticality experiments pertinent to RFP HEU and plutonium operations. The chief fissile material was 93 wt% HEU in various forms; plutonium oxide in limited amounts was available but not used. Building 886 contributed to waste shipped to INL during the 1960s, 1970s, and 1980s. The main contributor was combustible waste contaminated with HEU. The facility generated less than 50 drums per year.

4.4.11 Sewage Treatment Plant—Building 995

Occasionally, sludge from the sewage treatment plant (Building 995) became contaminated with trace amounts of plutonium. When this occurred, the sludge was packed into crates or drums and shipped to INL in the 1950s and 1960s. The amount shipped was small in comparison to sludge shipments from Building 774.

4.4.12 Building 331 Waste

Building 331 was a temporary development facility engaged in evaluating equipment and methods applicable to casting and fabricating DU items. The facility also carried out a very limited thorium project. Waste generated was mostly DU contaminated and was limited to less than 100 drums shipped to INL.

4.4.13 Building 865 Waste

Building 865 was an R&D metallurgical facility constructed to develop and evaluate equipment and procedures associated with foundry, forming, swaging, and machining of DU, DU alloys, beryllium, and other metals of interest to the War Reserve. The facility came into use in 1970 and did not contribute to TRU waste sent to INL; however, Building 865 may have contributed a small amount to 1970 waste sent to INL that was contaminated with DU and beryllium.

4.4.14 Plutonium Waste-Generating Facilities

The facilities that generated most of the plutonium waste shipped to INL in the 1950s were Buildings 771, 774, 776, and 777. Waste from these buildings continued to be shipped until INL terminated shipping from RFP in 1989. Building 779 (R&D facility) and Building 559 (analytical laboratory) began shipping plutonium waste in the latter part of 1960 and continued until termination of shipping from RFP in 1989. These buildings in which plutonium was worked were the main contributors to buried pre-1970 waste sent to INL. In 1958, Building 778 was constructed next to Building 776 and provided laundry services to all the buildings in which plutonium was worked.

Additional contributors from 1970 until INL shipping termination in 1989 were Building 707 (fabrication and pit assembly), Building 371 (chemical recovery and pyrochemistry), and Building 374 (liquid waste treatment). Waste generated by these facilities included all waste Types I through V and, later, all kinds of waste labeled with item description codes.

Boron glass raschig rings, which resemble napkin rings, were used for nuclear criticality safety in these plutonium facilities in tanks and other containers holding liquid. The rings were replaced because of sludge buildup or failed dimensional testing. The failed rings were leached with nitric acid, rinsed in water, and packaged for shipment to INL. Leached raschig rings made up a significant amount of glass sent to INL.

Raschig rings were used also in Building 881, which processed HEU. Cleaning out Building 881 contributed to the raschig ring waste in the Subsurface Disposal Area.

4.4.15 Offsite Waste Sources

Offsite educational institutions, private companies, and other federal agencies called on RFP to assist with disposal of their radioactive waste. Offsite sources usually were under contract to the AEC for some type of work. In June 1957, the RFP AEC office granted permission for the Dow Chemical Company to accept radioactive waste at RFP that was generated by local offsite companies, institutions, and government agencies. The driving reasons for granting this permission were the capability and capacity of RFP to accept waste and ship it to federally approved facilities. Consequently, RFP accepted and shipped radioactive waste received from offsite sources from 1957 to 1971. Solid waste received was not treated or repacked at RFP, but only shipped through to INL.

The Coors Porcelain Company had a contract to produce reactor components for the TORY II-C reactor through a contract with the Lawrence Radiation Laboratory at Livermore, California, in the early 1960s. Beryllium- and uranium-contaminated liquid waste was put in solar evaporation ponds. Solids recovered from the solar evaporation ponds were eventually shipped to INL.

A summary of waste received and its corresponding shipper is shown in Table 7. Information provided in Table 7 was derived from monthly history reports issued by the Waste Disposal Coordination Group, RFP, and *Rocky Flats Toxicologic Review and Dose Reconstruction Task 3&4 Report* (ChemRisk 1992).

Descriptions of waste received from offsite sources could not be located; however, two letters were located indicating that two drums received from the Colorado School of Mines contained soil samples contaminated with plutonium (see Appendix B).

Table 7. Offsite shipments received by Rocky Flats Plant and shipped to Idaho National Laboratory.

Calendar Year	Shipper		Description
1957	Martin Aircraft	5	55-gal drums
	Lowry Air Force Base	51	cartons
	U.S. Bureau of Reclamation	2	55-gal drums
1958	Lowry Air Force Base	64	cartons
	Sunstrand	29	55-gal drums
1959	Sunstrand	10	55-gal drums
	Lowry Air Force Base	2	55-gal drums
1960	Denver Research Institute	3	Chemical Warfare Service filters
1961	G. E. Sandia	28	20-gal drums
		13	15-gal drums
	Denver Research Institution	4	55-gal drums
		2	30-gal drums
		1	carton
	Lowry Air Force Base	4	55-gal drums
	Sunstrand	2	30-gal drums

Table 7. (continued).

Calendar Year	Shipper	Description	
	Coors Porcelain Company	99,700 gal of beryllium-contaminated liquid waste to solar ponds ^a	
1962	G. E. Sandia	9	20-gal drums
	Denver Research Institute	7	cartons
	Coors Porcelain Company	137,000 gal of beryllium-contaminated liquid waste to solar ponds—first uranium-contaminated waste received ^a	
	Colorado University Medical School	First contaminated waste received	
1963	Lawrence Radiation Laboratory	42	55-gal drums
	Coors Porcelain Company	22,000 gal of beryllium-contaminated liquid waste to solar ponds ^a	
1964	Colorado University Medical School	3	55-gal drums
	Colorado School of Mines	2	55-gal drums
	Coors Porcelain Company	26	55-gal drums
1965	U.S. Geological Survey and Denver Research Institute	1	55-gal drum
1966	U.S. Geological Survey and Denver Research Institute	3	55-gal drums
		1	30-gal drum
		1	carton
1967	U.S. Geological Survey and Denver Research Institute	2	55-gal drums
	U.S. Geological Survey	1	55-gal drum
	U.S. Department of the Interior	2	55-gal drums
1968	Dow Construction, U.S. Geological Survey, Department of the Interior, and VA Hospital	8	55-gal drums
		1	30-gal drum
		2	wooden boxes
1970	TOSCO	25	55-gal drums
	VA Hospital	1	55-gal drum
	U.S. Geological Survey	16	55-gal drums
	Coors Porcelain Company	44	55-gal drums
1971	U.S. Geological Survey	21	55-gal drums
	Denver Research Institute	5	55-gal drums
	U.S. Fish and Wildlife Commission	13	55-gal drums

a. Solar pond solids shipped as evaporator salts (Series 745 sludge).

4.4.16 Waste Quality Program

The waste quality program progressed from very little control over the quality of the waste shipped in the 1950s to a fully instituted quality program in the 1970s, continuing until the termination of shipment receipt at INL in 1989.

The first quality action was taken after the first waste shipment to INL in April 1954. A letter from G. V. Beard, Chief of the Health and Safety Branch, Idaho Operations Office, to John Epp, Assistant Director, Chemical Laboratories, Dow Chemical Company, RFP, dated May 5, 1954 (see Appendix A) described liquid leakage from seven drums. The letter requested action to preclude free liquids in future shipments. This letter initiated a quality effort to eliminate free liquids and to provide for absorbing any liquids that might develop during transport to INL.

Continued correspondence between INL and RFP initiated improvements and upgrades in waste packaging through drum and crate liners, closure mechanisms, and segregation of contents.

In October 1968, the U.S. Department of Transportation (DOT) issued new regulations (hazardous materials regulations [49 CFR 171–180]) for shipping radioactive materials that were in substantial conformance with the 1967 regulations of the International Atomic Energy Agency. Shipping methods, packaging procedures, and shipping containers were upgraded to meet these new regulations. Packaging line-generated waste required using drums that met specifications of the Interstate Commerce Commission, such as the 17H and 17C drums. The drum changeover was not new to RFP as the 74 series of sludge changed from 17H to 17C drums to take advantage of the maximum weight limit from 300 kg (660 lb)/drum to 400 kg (880 lb)/drum. This changeover was instituted in 1966. The upgraded packaging was certified by AEC Albuquerque Operations Office (see Appendix C).

In the late 1960s, installing drum counters for waste assays of SNM required a more definitive matrix to apply matrix density correction factors. Consequently, the five waste types that describe the waste shipped previously were replaced by item description codes for specific materials such as graphite, fire brick, raschig rings, sand, slag, and crucibles.

In 1970, the AEC directive (AEC 1970) on the disposal of TRU waste forced RFP to establish a formal quality assurance program for waste material. This quality assurance program instituted quality control and inspections on waste containers, packaging materials and procedures, personnel training certification, improved labeling, upgraded waste descriptions, and a demand for methods of NDA. The upgrades and improvements in waste packaging at RFP in the 1970s are discussed and described by Wickland (1977).

Since this report addresses mainly pre-1970 RFP waste, continuing quality improvements that occurred during the 1970s and 1980s will not be further discussed. In general, quality and control of radioactive waste shipped to INL improved gradually from 1954 to 1969. Waste containers, packaging materials, and procedures were standardized; other improvements include descriptive labeling and a more precise description of waste categories.

5. DEVELOPMENT OF WASTE CONTAINER STANDARDIZATION AND IDENTIFICATION

The shipping containers employed to ship radioactive waste to INL evolved from any available and suitable container to standardized containers that were quality controlled and performance tested. The upgrading and improvements instituted were driven by AEC and DOE directives, DOT and Interstate Commerce Commission regulations, and INL criteria for receiving waste.

5.1 Generator Identification

From 1954 through 1969, the containers generated by RFP were identified by a prefix representing the building generator number followed by a serial number usually assigned by the building generator and coordinated by the Waste Coordination Group. These prefixes are shown in Table 8 for pre-1970 waste shipments.

In 1970 and thereafter, radioactive waste was characterized as TRU waste and non-TRU waste. Table 9 lists the sources of TRU and non-TRU waste by prefix numbers and buildings.

Table 8. Identification of waste container generator pre-1970.

Prefix Number	Building Number	Building Mission
Plutonium Waste		
122	122	Medical treatment
123	123	Health physics laboratory
59	559	Plutonium analytical laboratory
71	771	Plutonium recovery and recycle
71(596)	771, 776, 777	1969 fire waste
741	774	First-stage sludge
742	774	Second-stage sludge
743	774	Grease Plant (organic) sludge
744	774	Cemented liquid waste
745	774	Evaporator salts
746	774	Empty contaminated drums
776	776	Plutonium manufacturing
77	777	Plutonium component assembly
78	778	Plutonium laundry
79	779	Research and development laboratories
79A	779A	Research and development laboratories
81	881	HEU and plutonium waste ^a
91	991	Plutonium and HEU component assembly
95	995	Sewage treatment plant

Table 8. (continued).

Prefix Number	Building Number	Building Mission
Highly Enriched Uranium Waste		
22	122	Medical treatment
23	123	Health physics laboratory
81	881	HEU fabrication, chemical recycle ^b
83	883	HEU forming
86	886	HEU criticality exposure assembly
89	889	HEU decontamination facility
91	991	HEU component assembly
Depleted Uranium Waste		
31	331	Temporary development facility
41	441	DU analytical laboratory
44	444	DU fabrication
47	447	Roaster oxide
83	883	DU forming
Beryllium Waste		
41	441	Beryllium analyses
44	444	Beryllium fabrication
47	447	Beryllium fabrication
71	771	Beryllium component destruction
741	774	Beryllium in first-stage sludge
742	774	Beryllium in second-stage sludge
745	774	Beryllium in evaporator salts
76	776	Beryllium components handled
77	777	Beryllium components handled
79A	779A	Beryllium components processed
83	883	Beryllium forming/descaling

a. Processing of returns
b. HEU cleanout started in 1965.
DU = depleted uranium
HEU = highly enriched uranium

Table 9. Post-1970 transuranic and nontransuranic waste generators.

Prefix Number	Building Number	Building Mission
Transuranic Waste Generators		
59	559	Plutonium analyses
07	707	Plutonium fabrication
71	771	Plutonium recovery and recycle
71(596)	771	1969 fire waste
741	774	First-stage sludge
742	774	Second-stage sludge
743	774	Grease Plant (organic) sludge
744	774	Cemented liquid waste
745	774	Evaporator salts
746	774	Empty contaminated drums
76	776	Plutonium manufacturing
77	777	Plutonium component assembly
78	778	Plutonium laundry
79	779	Research and development laboratories
79A	779A	Research and development laboratories
81	881	HEU and plutonium waste ^a
95	995	Sewage treatment plant
Nontransuranic Waste Generators		
23	123	Health physics laboratory
31	331	Temporary development facility
44	444	Depleted uranium fabrication
47	447	Roaster oxide
865	865	Research and development facility
81	881	HEU cleanup and decommissioning
83	883	Depleted uranium forming
86	886	HEU criticality exposure
89	889	HEU decontamination facility

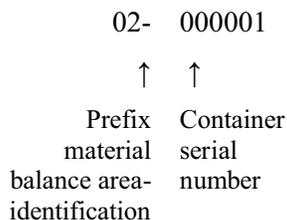
a. Processing of returns.

HEU = highly enriched uranium

Two special projects carried out in Building 881 generated TRU waste besides the processing of returns. The first used U-233, and the second used Np-237. These two projects were not routine operations and were of short duration.

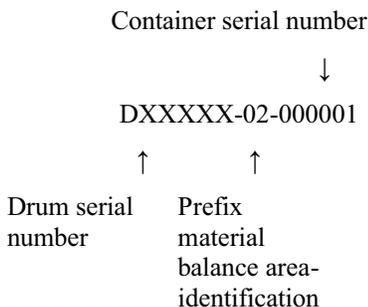
5.1.1 Post-1970 Identification of Containers

Using drum prefix numbers to identify building generators was discontinued in 1971. A new system was installed using prefix numbers from 01–99 to identify the material balance area that generated and packaged the waste container. An example is illustrated below.



A container serial number was assigned by the department responsible for the material balance area; however, Building 774 sludge still used its original prefix numbers because Building 774 did not have any material balance areas. This system was used only for TRU waste.

When Rockwell International became the contractor in 1975 for RFP, the waste container identification system was altered again. White-painted drums were used for TRU waste and were serialized by the warehouse. All white drums were issued with a metal tag with the drum serial number provided by the warehouse. The material balance area and department container serialization were retained as indicated below.



This container identification system, started in 1975, has continued to the present.

5.2 Waste Containers

The majority of waste containers shipped to INL were 55-gal drums. The second-place drum was the 30-gal drum. A small number of 40- and 45-gal drums was shipped in the 1958-1961 timeframe. A very limited number of 20-gal drums also was shipped in the 1958-1961 timeframe. In 1972, an 83-gal drum was introduced for Building 774 sludge and other plutonium-bearing waste. However, the use of 83-gal drums was discontinued in 1973. Occasionally, drums of a different capacity were received from offsite and were shipped to INL as indicated in Table 7.

The original source of drums was a mixture of vendor product drums (i.e., drums that had contained solvents or other materials and were reused to dispose of waste) and newly purchased drums. Building 774 started to use new drums for sludge in April 1958. The gradual buildup of waste from the increased plutonium manufacturing mission in the late 1950s and early 1960s increased the need to purchase new drums.

Initially, 17H drums were used but proved unsatisfactory for packaging Building 774 sludge. The 17H 55-gal drum had a weight limit of 299.4 kg (660 lb), which limited the packaging of 774 sludge. In the 1965–1966 timeframe, 17C 55-gal drums were purchased for sludge as the weight limit was 399.2 kg (880 lb) per drum. For consistency onsite, the 17C drum became the standard drum.

Shipping waste in wooden boxes began in December 1954. Wooden boxes accommodated items that were too large or too heavy for drum packaging. Originally, wooden box dimensions were tailored to fit the item to be shipped. As the use of wooden boxes (crates) increased, the box size was standardized to 4 × 4 × 7 ft to facilitate loading into trailers and railcars. At the packer's discretion, some items were wrapped in plastic before being placed in the box, such as items that were bulky, had sharp edges, had surface contamination, or had come loose or become unwrapped during shipping. Coating plywood boxes with fiberglass began in July 1971.

Other forms of waste shipped during the late 1950s were several tanks, a couple of small metal boxes, and two cylinders. Cardboard cartons were occasionally used for special items; however, in 1967, cartons were used to ship unleached spent HEPA filters. Shipment of CWS filters in drums or cartons began in March 1955.

5.3 Waste Packaging

Packaging radioactive waste evolved from a simple liner system to a maximum containment configuration. The packaging scheme was governed by the type of waste to be shipped, packaging and shipping regulations, and the receiving site's disposal criteria in effect at that time.

5.3.1 Non-Line-Generated Waste

Non-line-generated waste originated outside the glove-box lines but within a plutonium handling facility. Other sources were from the uranium processing buildings. Any sharp or cutting edges of waste were taped to prevent punctures. Waste was placed directly into a drum or wooden crate, which had a polyethylene (5- or 8-mil) liner. The liner was taped closed, closed by heat sealing, or both.

5.3.2 Line-Generated Waste

Items capable of puncturing the drum liner were taped or placed in 8-mil fiberboard sleeves with a polyethylene wrapping. Powdery items and small solid items were sealed within paint cans, plastic bottles, or other similar containers. The contaminated material was removed from the glove-box line through a bag-out procedure. The sealed polyethylene bag was placed in a 17H or 17C drum that had a 5- or 8-mil polyethylene liner. When full, the liner was taped closed, and the drum lid was secured to the drum body by a 12-gauge, bolted-ring closure system.

5.3.3 Crated Waste

Large bulky items, such as metal scrap, light fixtures, tool machines, lumber, piping, hoods, or air ducts, that could not fit in 55-gal drums were placed in wooden crates. These objects were contaminated externally to varying levels with plutonium, and in some cases with HEU. The crate was lined with an 8-mil polyethylene sheet. Heavy items were secured by bolting to the crate skids. When full, the liner was taped or heat sealed, and the lid was nailed to the crate body. The closed crate was banded with 1.25-in.-wide steel straps in at least four positions.

5.3.4 Series 774 Sludge Waste

The majority of sludge waste was generated by the Liquid Waste Treatment Plant (Building 774). Dry Portland cement was placed into a 17C or 17H 55-gal drum lined with a 5- or 8-mil polyethylene liner. Additional dry Portland cement was interspersed with the sludge during the drum-filling cycle. After completing the filling cycle, the liner was taped closed, and a second quantity of dry Portland cement was placed on top of the liner. The drum was sealed with a 12-gauge bolted ring.

5.3.5 Series 743 Grease Sludge

The organic-based (grease) sludge was produced mainly by the Liquid Waste Treatment Plant in Building 774. An oil-dry absorbent was placed in a 17C or 17H 55-gal drum lined with a 5- or 8-mil polyethylene liner. The greaselike sludge was added to the lined drum. Microcel, manufactured by Johns-Manville, was used as a hardener for the sludge. After filling, the liner was taped closed, and hardener was placed between the liner and drum lid. The lid was sealed with a 12-gauge bolted ring.

5.3.6 Neutralized Hydrochloric Acid Solutions

Waste hydrochloric acid solutions were neutralized with sodium hydroxide and solidified with magnesia cement within the drum. Lining the drum was 8-mil polyethylene. After sealing the liner with tape, magnesia cement was placed between the sealed liner and the drum lid. The lid was sealed with a 12-gauge bolted ring.

While polyethylene bag-out bags were preferred for incineration, PVC bag-out bags also were used. Waste generators were encouraged to use PVC bags when waste was not to be incinerated.

5.3.7 Post-1970 Packing

The packaging employed for the 1970–1972 timeframe for line-generated waste consisted of one or two bag-type polyethylene liners depending on the type of waste being packaged. The individual packages placed within the drum were polyethylene or PVC bags taped closed. Cardboard liners placed inside the inner bag liner were used for abrasive waste such as graphite molds or fire brick.

In the latter part of 1972, use of the 90-mil, rigid polyethylene liner began. The rigid liner was placed inside the drum; one or two polyethylene drum liner bags were placed inside the rigid liner. The polyethylene bags were taped closed, and the top of the rigid liner sealed. From 1972 to early 1982, Oil-Dri absorbent was added (0.9–1.9 L [1–2 qt]) to the top of the sealed outer drum liner. Vermiculite was substituted for Oil-Dri after February 1982. A more complete packaging description for post-1970 waste can be located in Clements (1982).

5.4 Packaging and Container Codes

Copies of load lists for trailers and atomic materials rail transfer (ATMX) railcars for the 1950s and 1960s were forwarded to INL personnel to reconcile shipper-receiver records. A variety of codes was employed on the load lists to indicate the packaged contents and the containers involved. The content-code Types I through V have been described previously in Section 4, “Waste Characterization.”

The Bureau of Explosives issued permits to RFP covering hazardous material containers for shipping. The Bureau of Explosives codes that appeared frequently on the load lists are shown in Table 10.

Table 10. Bureau of Explosives codes on the load lists.

Bureau of Explosives Permit Number	Use
2056	Plywood boxes containing machinery and glove boxes
2057	Second-hand 55-gal drums containing trash and dry waste
2058	16-gauge Interstate Commerce Commission Spec. 17C drums containing sludge
2059	Interstate Commerce Commission 12B cartons containing CWS filters
2060	Plywood boxes containing CWS filters

CWS = Chemical Warfare Service

Occasionally, the symbol “MTD” was indicated on the load lists, which indicated empty drum(s).

The label notation “Red” or “Blue” on the load lists indicated plutonium content. Containers with less than 15 g of plutonium were labeled in blue; those with 15 g or more were labeled in red.

The notation “LLD” for a given drum indicated the drum was lined with lead. Lead drum liners were used to ensure the external radiation requirement was met.

Load lists in the early 1970s used the notation “nret,” which stood for nonretrievable (i.e., not TRU) waste. Waste identified as nret was sent to INL for burial.

The designation of “Plant Waste” meant that the waste was generated outside of any process or support facility.

The labeling of radioactive containers was based on transportation and packaging regulations in effect at the time of shipment to INL; however, exemptions were granted for a few shipments, which could not meet the standard regulations.

6. WASTE ASSAY

At startup of RFP, a waste policy was in place to declare the loss of SNM through the concept of normal operating loss. The waste generator was responsible for assigning an SNM value to the waste generated. For liquid waste, a chemical or a radiometric assay for SNM was performed. Initially, an estimating procedure was used for solid waste based mainly on a “by-difference” approach coupled with operating experience.

These assay methods proved to be inadequate as the “material unaccounted for” grew to an unacceptable level. Zodtner and Rogers (1964) stressed the need for improved methods of assaying plutonium in the waste streams and identified several issues that contributed to material unaccounted for. These issues are listed below and are associated mainly with assignments to normal operating loss:

- Understatement of plutonium in graphite waste
- Understatement of plutonium on HEPA filter waste
- No available plutonium assay for in situ combustible and other solid waste forms (no representative sample available)
- Inadequate liquid assay methods.

6.1 Liquid Waste Assay

The majority of plutonium-contaminated aqueous waste was generated by the plutonium recovery facility, Building 771. See Table 11 for types of solutions transferred from Building 771 to the Liquid Waste Treatment Plant for the month of December 1961 and for aqueous solutions received from other buildings.

Determining plutonium and americium in liquid waste solutions was accomplished by radiometric and chemical titration methods. The radiometric method often required large dilutions of the original solution, which added to the inaccuracy of the assay. Obtaining a representative sample from raschig ring tanks also was difficult. Measuring liquid volume accurately within a tank was another issue. A tank calibration crew was organized to solve the tank calibration issue. A group also was formed to prepare known plutonium solutions coupled with statistical assistance to develop precise and accurate data.

Waste solutions originating from Building 771 were collected in critically safe tanks awaiting transfer to Building 774. The tanks were assayed for plutonium and Am-241 content. If below discard limits for plutonium, the solutions were transferred to Building 774 for disposal treatment. The plutonium content of the solutions was taken as a normal operating loss. On a batch basis, plutonium content was pro-rated per drum of first-stage sludge produced. The plutonium content transferred to the second-stage sludge was pro-rated also. The HEU processed into second-stage sludge was pro-rated also and taken as a normal operating loss when received by Building 774.

Because of concern for improved criticality control and for upgrading the assay, a second set of critically safe tanks was installed in the early 1970s. The sampling program was improved by having both the generator and the Liquid Waste Treatment Plant sample the same tanks. The results had to agree statistically before the second set of tanks could be transferred to Building 774. Improved analytical methods also were developed.

Table 11. Liquid waste solutions received for treatment December 1961.

First-Stage Treatment		
Source	Description	Quantity (gal)
Building 771	Ion column effluent	1,656
	Raffinate	5,077
	Cooling waste	19,800
	Nash pumps (vacuum)	4,653
	Distillate	3,750
	Miscellaneous	423
Total		35,359
Second-Stage Treatment		
441	Depleted uranium analyses	14,200
444	Depleted uranium and beryllium manufacturing	34,800
771	From first stage	39,950
771	Laundry	30,650
776	Plutonium manufacturing	13,800
881	Highly enriched uranium manufacturing	76,700
Total		210,100

6.2 Raschig Ring Tanks

The concept of employing raschig rings for nuclear criticality control was extended in the early 1960s to plutonium tanks. Raschig rings were fabricated from boron silicate glass as a hollow cylinder with a 0.72-in. outside diameter, 1.75-in. length, and wall thickness of 0.25 in. Boron acts as a neutron absorber, thereby providing a degree of criticality control for tanks holding plutonium-bearing solutions. Building 771 waste tanks were raschig ring tanks.

Several process problems were associated with raschig ring tanks. The first was the buildup of plutonium sludge on the raschig rings, thereby requiring the rings' removal and replacement with new rings. The second was the difficulty of achieving solution homogeneity within the tank. The third was ring breakage during air sparging to accomplish solution homogeneity. The fourth was thinning of the rings from fluoride ion attack. Fluoride ions were used to facilitate plutonium oxide dissolution. Calcium fluoride slag reprocessing for plutonium recovery also provided a source of fluoride ions. Ring thinning, breakage, and accumulated sludge buildup adversely affected tank calibration.

Studies of solution circulation favored air sparging over circulation from bottom to top of the tank; however, continued studies demonstrated the inconsistencies of air sparging when applied to different tanks. A lengthy air sparging cycle was necessary to obtain proper solution mixing. This problem also contributed to the requirement for two analyses of a tank before shipping waste solutions to Building 774.

The waste solution mixing procedure was a lingering nuclear accountability concern because of doubts about analytical sample representation.

Raschig ring tanks were favored for some plutonium solutions because they required less floor space than geometrically safe tanks; however, annular tanks gradually were replacing raschig ring tanks when floor space permitted the changeover. This was not the case for waste solution tanks. Raschig rings that had been removed were leached with nitric acid to remove the residual sludge, then washed in water, dried, and packaged into 55-gal drums for shipment to INL.

6.3 Solid Waste Assay

During the 1950s and early 1960s, chemical assay and radiometric analysis were the two analytical methods for SNM determinations. X-ray fluorescence methods also were developed for plutonium analyses in the early 1960s but were used mainly on production samples.

Zodtner and Rogers (1964), in addressing issues of material unaccounted for, called for developing procedures for NDA of solid waste. Consequently, an R&D effort was launched in the mid-1960s to develop gamma-neutron counting systems applicable to solid waste packaged in drums. The initial experimental R&D drum counter was activated in 1964 and was located in Building 771. The first production drum counter was installed in Building 771 in 1969.

Continuing R&D activities and electronic advancements produced sophisticated counting systems that provided helix scanning, segmented gamma scanning, low-resolution gamma assaying, alpha-neutron corrections, background corrections, matrix density adjustments, and computerization of radiometric calculations. Systems were installed to assay small containers, drums, crates, and HEPA filters. By the late 1970s and early 1980s, systems had been developed and were operating to assay all item description codes. However, these advanced systems were not available for assaying the majority of pre-1970 waste sent to INL. See Appendix D for reports and references describing these systems and their operation.

6.4 Production Nondestructive Assay Systems

In 1984, the “Handbook of the Rocky Flats Plant Production Non-Destructive Assay Systems” was compiled by Bill Ulbricht (see Appendix E). The handbook described the counting systems and the container and applicability of item description codes and listed the building location for each counting system and its operational status. In 1987, the handbook was reissued to upgrade the information on NDA (see Appendix F). These two handbooks provide an insight into the state and applicability of the NDA systems used during the 1970s and 1980s for assaying solid waste.

6.5 Plutonium Waste Discards

In 1964, the first experimental NDA system for drum counting was installed. This counting system assayed for plutonium content in waste drums shipped to INL. The advancements and improvements realized in the second-generation NDA drum counting systems and the development of correction factors for matrix density, alpha-neutron reactions, and electronic anomalies prompted a review of the data for understated plutonium assays from the 1964 counting system. Consequently, a review—covering from February 1968 to June 1971—was undertaken of the counting data for 24,000 drums assayed with the 1964 drum counting system.

Bidwell, Chanda, and Cartwright (1973) address material unaccounted for in the INL drum field, and according to Table 11 in the report, total plutonium weight was understated by about 17.6 kg (38.8 lb)

for 15,795 drums at INL (see Appendix G). These drums suspected of being understated were reviewed based on their gamma and neutron counts recorded by the 1964 drum counter. Based on new correction factors, operating experience, and recounts of drums with similar item description codes in the RFP backlog drum field, a plutonium estimate was derived for the suspect drums at INL. These plutonium estimates are summarized in Table 12 for individual item description codes. Two of the INL suspect drums were assayed at the National Reactor Testing Station. The comparison of assays is shown in Table 13.

Table 12. Suspect drums shipped to Idaho National Laboratory.

Number of Drums	Item Description Code	Description	Original Rocky Flats Plant Plutonium Weight (g)	Estimated Plutonium Weight (g)
1	300	Graphite molds	0	7
1	320	Heavy non-special-source metal (such as tantalum, tungsten, and platinum)	5	9
8	330	Dry combustibles (paper and rags)	25	2,754
1	336	Wet combustibles (paper and rags)	0	27
3	338	Filter media	49	92
1	372	Grit	45	65
2	440	Glass (except raschig rings)	1	18
3	441	Unleached raschig rings	21	1,486
33	480	Light non-special-source metal (such as aluminum, copper, and beryllium)	268	1,208
53	All	Totals	414	5,666
		Difference	5,666-414	5,252

Table 13. Suspect drum comparison.

Drum Number	Item Description Code	National Reactor Testing Station Plutonium Weight (g)	Original Rocky Flats Plant Plutonium Weight (g)	Estimated Rocky Flats Plant Plutonium Weight (g)
57493	330	500	0	476
59682	480	300	0	319

Plutonium discards to waste (i.e., normal operating losses) were tabulated by D. L. Ziegler for calendar year 1967. The total plutonium discard written off as shipped to INL for Calendar Year 1967 was 70,382 g (155.2 lb). Ziegler (see Appendix H) provides tables for Calendar Year 1967 waste covering six groups of waste listed below:

- Liquid waste from the plutonium recovery area
- Building 774 waste
- CCl₄—oil converted to grease in Building 774
- Non-line-generated hot waste
- Line-generated waste
- Crated waste.

6.6 Economic Discard Limits

6.6.1 Background

Recognition in the late 1950s and early 1960s that significant quantities of plutonium were being lost through the solid waste streams emphasized the need for plutonium assay systems tailored to assay containers of waste. Also needed were plutonium recovery systems designed to process solid and liquid waste streams.

In 1960, a project was initiated to provide the necessary plutonium recovery capability and capacity for treating low-level solid residues and liquid waste solutions. This project included the development of an NDA drum counter. Construction was completed in 1962, and the additional recovery systems were started in the mid-1960s. The first workable drum counter was installed in Building 771 in 1964. This experimental drum counter provided the necessary data and operational experience for continuing upgrades to establish a full production drum counter in 1969.

6.6.2 Establishing Economic Discard Limits

Before establishing economic discard limits, the decision to process a given residue drum for plutonium recovery was made arbitrarily by operations supervision. This inconsistent procedure showed that a better method was needed to determine the feasibility of recovery processing. In addition, the processing of all waste was not economically necessary.

The RFP approach was to establish a system of economic discard limits. As provided by the AEC and later DOE, the cost of plutonium recovered and processed to a pure metal state was compared with the value of producing new reactor weapons-grade plutonium metal. The break-even point was the economic discard limit for a given residue. These limits were ordinarily expressed in (1) gram of plutonium per gram of residue for solids and (2) gram of plutonium per liter of solution for liquids. Any residue item above the discard limit required processing, and any residue item below the discard limit was considered waste.

6.6.3 Calculating Economic Discard Limits

The economic discard limits were calculated using Equation (1):

$$D = \frac{(L)(T_{(S)})}{(C - F)(E)} \quad \text{or} \quad D = \frac{(L)(T_{(L)})}{(C - F)(E)} \quad (1)$$

The symbols of the equation are explained below.

Symbol	Definition	Units
D =	Economic discard limit concentration	kgPu/1,000 L or kgPu/kg residue
L =	Labor cost per productive man hour	\$/hour
T _(L) =	Time required to process one kiloliter of the original liquid residue through residue recovery	hour/1,000 L
T _(S) =	Time required to process 1 kg of the original solid residue through residue recovery	hour/kg
E =	Total process efficiency, fraction of plutonium in residue that is converted to metal	kgPu/kgPu
C =	DOE cost to produce new weapons grade plutonium	\$/kgPu
F =	Rocky Flats Plant cost to produce plutonium metal from concentrated plutonium nitrate feed	\$/kgPu

To determine these limits, data were collected on material accountability; processing rates and efficiency; and labor, material, and support costs. These data were collected to make the initial determination and also on a continuing basis. All plutonium-bearing residues were divided into categories with similar processing requirements. Each such category was assigned a digital item description code for tracking purposes. Similarly, each finite step of residue recovery processing was assigned an operations code number. All such costs as direct labor cost, supplies and materials, and maintenance conducted on that step were charged to that assigned code number.

For each period between inventories (usually 1 month), the gross plutonium weights of residue material charged to the operation, the net weight recovered, and the total cost of the operation were determined. The cost of processing per unit weight of residue processed, as well as the cost per gram of plutonium handled through the operations step, was calculated mathematically. The total cost per gram of plutonium was the sum of all operational steps required to convert the plutonium to metal.

Because unit recovery costs could vary significantly from month to month, for instance as a result of major maintenance work, economic discard limits were normally calculated and revised annually. In later years, when the generation rate of a particular category of residue exceeded processing capacity, a factor was added to allow for the construction of new facilities, amortized over a 10-year period.

As indicated above, economic discard limits and item description codes were coupled and driven by improved NDA procedures and recovery systems. Additional description codes and corresponding economic discard limits were added as segregation requirements became known for obtaining more residue homogeneity to achieve more accurate NDA results.

Anderson, Putzier, and Ziegler reported the discard limits for Fiscal Year 1969 (shown in Table 14) in their internal report reproduced in Appendix I.

Introduction of economic discard limits probably began with drum counting capability, which has been placed in the 1966–1967 timeframe. A letter from William F. Romine, RFP Traffic Manager, dated December 23, 1968 (see Appendix X), defines line-generated waste coupled with an economic discard limit range as: “Line-generated wastes are graphite molds, filters sludge, insulation, glass, washables, combustibles, metals, and miscellaneous residues with plutonium discard limits ranging from 7×10^{-3} g/g to 3.0×10^{-4} g/g.” This letter also indicates that economic discard limits were in place in 1968.

Table 14. Economic discard limits—Fiscal Year 1969.

Category	Discard Limit (gPu/g Total)
Sweepings	0.007
Sludge	0.007
Magnesium oxide sand	0.007
Ion exchange resin	0.007
Incinerator ash	0.007
Sweepings heels	0.007
Ash heels	0.007
Glass and ceramics	0.0005
Scarfed molds	0.00035
Graphite flow residue	0.002
Chemical Warfare Service filter 2 × 2 × 1 ft	24.0 g/filter
Dry box filters 8 × 8 × 4 in.	3.0 g/filter
Washables	0.0006
Combustibles	0.0007
Miscellaneous scrap metal	0.0003

(see Appendix I)

7. WASTE TRANSPORTATION

The RFP waste was transported to INL using tractor trailers and ATMX railcars. A combination of trailers on flatbed railcars also was employed.

7.1 Trailers

Several commercial trucking firms were contracted by RFP Traffic Group to haul waste drums and crates (boxes) from RFP to INL. The cargo consisted mainly of 55- and 30-gal drums with an occasional drum of another capacity. Crates (boxes) were the second type of container hauled. Cardboard containers also were used for CWS and HEPA filters.

Maximum drum capacity per trailer was 164, but trailer loads usually were less with a range of 62 to 154 drums per trailer. The total load weight was a governing factor.

The first shipment of drums to INL in April 1954 leaked a small amount of liquid that contaminated the trailer floor. The AEC Idaho Operations Office personnel recommended using an absorbent floor covering to absorb any leaking liquid, but this was discontinued when the policy of not putting liquids in drums was implemented.

7.2 Atomic Materials Rail Transfer Railcars

Higher levels of plutonium waste required a Type B package to meet AEC (1973) and DOT regulations, which were to take effect in the late 1960s. Rocky Flats Plant generated about 3,000 55-gal drums and 36,000 ft³ of crates having a plutonium content requiring Type B packaging. The existing containers and those under development were either inadequate or too expensive to accommodate the waste shipments to INL.

The 600 series ATMX railcar designed by Sandia Laboratory for hauling explosive materials was selected for certification as a Type B container. The ATMX-600 railcar was certified, and two were assigned to RFP by the AEC. Eventually, six railcars were assigned to haul RFP waste to INL.

The ATMX-600 railcar could accommodate 216 55-gal drums loaded individually and a significant number of crates depending on their weight. The ATMX-600 railcar also could accept two cargo carriers, which reduced the loading time significantly but reduced the total drum load to 132 per railcar. ATMX-600 railcar shipments to INL commenced in 1969 and continued until 1989.

A more thorough description of the ATMX-600 railcar is provided by Adcock (1970). The safety analysis report describing packaging for the ATMX-600 railcar was issued by Adcock and McCarthy (1974).

7.3 Piggy-Back Trailers

A special request to ship plutonium-contaminated residues to INL that had been stored at RFP was granted by AEC Albuquerque Operations Office in 1963 because RFP did not have the capacity then to process this residue backlog. Trailers containing the waste were “piggy-backed” on flatbed railcars and escorted by health physics personnel for the journey to INL. This arrangement continued until all the residue drums were shipped in 1964.

7.4 Shipping Forms

In the 1950s and 1960s, RFP began using the AEC Idaho Operations Office shipping form, “Solid Radioactive Waste Disposal Order and Shipping Data.” The form consisted of the following three sections:

- Section I identifies shipper and receiver
- Section II describes waste
- Section III verifies disposal.

Sections I and II were completed by the shipper (RFP), while Section III was completed by receiving (INL) personnel. Figure 1 is a copy of a completed form. Shipments were identified by two systems: (1) Dow serial number and (2) Rocky Flats seal number. As shown in Figure 1, the Dow serial number was 66-57-B with a Rocky Flats seal number of RF-3790. In the latter part of the 1960s, health physics numbers were used instead of Dow serial numbers. A more sophisticated radioactive waste form (ID-135) was employed in the 1970s by INL. Figure 2 shows the first page of Form ID-135.

To record trailer and ATMX-600 railcar loadings, RFP used a load-list form. The load list was completed by personnel loading the waste. Consequently, the load lists recorded for pre-1970 waste were handwritten. Figure 3 is a copy of a load list. The load lists in the late 1960s became known as waste disposal sheets. In the 1970s and 1980s, the load lists were computerized and were organized to provide data for completing Form ID-135. Figures 4 and 5 are copies of the first two pages, respectively, of a computerized load list.

FORM ID-110A (6-65) RFDOW SRL 07/08/66 1128 2 3 DISPOSAL LOCATION: PIT 4	U. S. ATOMIC ENERGY COMMISSION IDAHO OPERATIONS OFFICE SOLID RADIOACTIVE WASTE DISPOSAL ORDER AND SHIPMENT DATA	REFERENCE: IDM 0600-7 TERMS: SEE REVERSE SIDE OF FORM
ORDER NO. <u>66051 G</u>		
SECTION I		
TO: U. S. ATOMIC ENERGY COMMISSION IDAHO OPERATIONS OFFICE IDAHO FALLS, IDAHO 83401 ATTENTION: <u>R. L. Hayden</u>	FROM (NAME & ADDRESS OF AGENCY) The Dow Chemical Company Rocky Flats Division Post Office Box 888 Golden, Colorado 80401 BILL TO: (IF DIFFERENT THAN ABOVE)	
THE COMMISSION IS HEREBY REQUESTED TO DISPOSE OF THE RADIOACTIVE WASTE DESCRIBED ON THIS ORDER ACCORDING TO THE TERMS AND CONDITIONS SPECIFIED HEREON.		
ORIGINATOR <u>E. A. [Signature]</u> (SIGNATURE)	WD Coordinator (TITLE)	<u>July 8, 1966</u> (DATE)
ACCEPTED FOR THE ATOMIC ENERGY COMMISSION BY:		
<u>P. E. [Signature]</u> (SIGNATURE)	CFP EP Supervisor (TITLE)	<u>7/21/66</u> (DATE)
SECTION II DESCRIPTION OF WASTE		
Trlr. No. 800203 Dow S.N. 66-57-B Seal No. RP-3790		
1. TOTAL NO. OF PACKAGES: <u>17</u>	VOLUME IN FT. ³ : <u>2,074</u>	WEIGHT: <u>25,200</u>
2. COMPLETE DESCRIPTION OF CONTENTS AND PACKAGING <u>17 Wdn boxes of scrap metal, equipment, filter, etc.</u>		
3. CLASSIFICATION: SECRET _____ CONFIDENTIAL _____ CATEGORY: <u>Unclassified</u> US MATERIAL TYPE _____ AMOUNT: NET _____ ISOTOPE: <u>9</u>		
4. PROPOSED MEANS OF TRANSPORTATION: <u>D&R & UP railroad</u> SHIPPING DATE: <u>7-11-66</u>		
5. RADIOISOTOPES CONTAINED: <u>plutonium and uranium</u>		
6. MAXIMUM RADIATION AT SURFACE OF <u>Trailer</u> PACKAGES: <u>2.0 mR/hr</u> TOTAL CURIES: <u>not feasible</u>		
7. ASSOCIATED HAZARDS: <u>None</u>		
* These amounts and material types are estimated and provided at the end of the calendar year.		
SECTION III (TO BE COMPLETED BY PERSON WITNESSING DISPOSAL)		
DISPOSAL WAS MADE BY MEANS OF <u>Burial in Pit # 4</u>		
AT <u>190' West of the S/E Monument</u> (LOCATION)	<u>7-15-66</u> (DATE)	
<u>M.W. Breashear</u> (SIGNATURE)	<u>7-15-66</u> (DATE)	

17221 1880

Figure 1. Form ID-110A, "Solid Radioactive Waste Disposal Order and Shipment Data."

1.0
.5

WASTE DISPOSAL

No.	Drum No.	Gross Weight	Drum Size	Type	No.	Drum No.	Gross Weight	Drum Size	Type
1.	71-437	1,600	84x52x48	V	32.	71-616	1,410	84x52x48	V
2.	71-447	1,600	84x52x48	V	33.	71-497	1,200	84x52x48	V
3.	71-483	1,700	84x52x48	V	34.	71-477	1,000	84x52x48	V
4.	71-496	1,500	84x52x48	V	35.	71-493	1,100	84x52x48	V
5.	71-482	1,700	84x52x48	V	36.	71-498	1,200	84x52x48	V
6.	71-467	1,600	84x52x48	V	37.	71-92	900	84x52x48	V
7.	71-122	2,100	84x52x48	V	38.	71-134	1,800	84x52x48	V
8.	71-122	1,800	84x52x48	V	39.	81-623	1,900	84x52x48	V
9.	94-95	1,200	84x52x48	V	40.	8	10,300		
10.					41.				
11.	9	14,900			42.				
12.					43.				
13.					44.				
14.					45.				
15.					46.	40 of 8	2060		
16.	9	14,900			47.	130 of 8	2056		
17.	8	10,300			48.				
18.					49.				
19.	17	25,200	2,074 lbs		50.				
20.					51.				
21.					52.				
22.					53.	44 of 8 type V 2-WB	2,100		244 ft
23.					54.	71 of 8 type III 7-WB	4,300		488 ft
24.					55.	71 of 8 type V 5-WB	8,000		610 ft
25.					56.	71 of 8 type V 3-WB	5,700		366 ft
26.					57.	81 of 8 type V 3-WB	5,100		366 ft
27.					58.				
28.					59.	71WB	25,200		2,074 ft
29.					60.				
30.					61.				
31.					62.				

TOTALS

Date: 7-8-66

Carrier: DOW SN

Trailer No: 802203

Shipped: 17 (Boxes)
25,200

RF Seal 3790
DOW SN 665713

2.0 mtr ft

Figure 3. Handwritten Rocky Flats Plant waste disposal load list.

01/10/05 10:20 FAX 303 575 9189 LATA

FORM ID-135 (2-71)

U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
RADIOACTIVE WASTE FORM

REFERENCE
IDN: 0511

SECTION 0

0 - ON SITE;
1 - OFF SITE

DATE: MONTH 03, DAY 27, YEAR 77

NAVY TIME: 0720

AREA OF ORIGIN: R F 0

BUILDING NUMBER: 000

TO BE ON ALL CARDS

PREPARED BY: E. S. Ryan

APPROVED BY: _____ DATE: _____

APPROVED BY: _____ DATE: _____

(CONTRACTOR SS-MATERIAL OFFICER) DATE: _____

(AEC/DOE SS-MATERIAL OFFICER)

TRLR. 92-5315 DOW SN 72-658

SEAL NO. DOW HP-351

SECTION 1

DESCRIPTION OF WASTE	CUBIC FT. / POUNDS		CUBIC METERS / GALLONS		LITERS		GROSS VOLUME		GROSS WEIGHT		GROSS RADIOACTIVITY IN CURIES		DISPOSAL OR STORAGE LOCATION	
	F	P	M	K	L	G	U	U	U	U	U	U	AREA	WITHIN AREA
N. O. N. T. R. A. N. 6. 5. B.	6.2	2.0	E+0.2	F	2.7	1.0	E+0.4	P	7.8	7.4	E-0.1			

SECTION 2

CARD TYPE	TYPE OF CONTAINER	NUMBER OF CONTAINERS	UNIT VOLUME OF CONTAINER	GAL	K	UNITS	DESCRIPTION OF WASTE	SOLID WASTES		
								AREA	WITHIN AREA	MO
2	B, L, M	84	5.5	K			PLANT WASTE			

Figure 4. Page 1 of computerized Form ID-135, "Radioactive Waste Form."

LOAD LIST NUMBER 71 SINGLE LAYER DATE 3/23/72

THIS LOAD IS CLASSIFIED AS LSA

AND HAS BEEN PREPARED FOR VT

AND CONSISTS OF 84 SD AND 0 OTHERS

CONTAINER ID	CONTENTS	GROSS	MREM/HR	ELEMENT	WT.	STORAGE
PREFX SERIAL	CODE	WEIGHT	SUR 3 FT	1ST	2ND	LOCATION
224 210	950	315	-5	0.0	0.0	662
224 211	900	225	-5	0.0	0.0	662
224 212	950	265	-5	0.0	0.0	662
224 213	950	310	-5	0.0	0.0	662
224 216	950	205	-5	0.0	0.0	662
224 217	950	280	2.8	-5	0.0	662
224 218	950	380	1.0	-5	0.0	662
224 219	950	490	0.0	0.0	0.0	662
224 220	950	550	0.0	0.0	0.0	662
224 221	950	535	0.0	0.0	0.0	662
224 222	950	510	0.0	0.0	0.0	662
224 223	950	275	-5	0.0	0.0	662
224 224	950	385	-5	0.0	0.0	662
228 1	900	170	-5	0.0	0.0	662
228 101	900	145	-5	0.0	0.0	662
228 102	900	155	-5	0.0	0.0	662
228 106	900	165	-5	0.0	0.0	662
362 102	900	175	-5	0.0	0.0	662
1444 102	900	175	-5	0.0	0.0	662
1444 105	900	135	-5	0.0	0.0	662
1483 105	900	160	-5	0.0	0.0	662
1483 113	900	155	-5	0.0	0.0	662
1483 115	900	150	-5	0.0	0.0	662
1483 116	950	615	-5	0.0	0.0	662
1483 117	950	665	-5	0.0	0.0	662
1483 118	950	545	-5	0.0	0.0	662
1483 122	950	610	-5	0.0	0.0	662
1483 127	950	360	-5	0.0	0.0	662
1483 128	950	365	-5	0.0	0.0	662
1544 311	900	139	-5	0.0	0.0	662
1544 317	900	152	-5	0.0	0.0	662
1544 326	900	136	-5	0.0	0.0	662
1544 329	900	170	-5	0.0	0.0	662
1544 330	900	145	-5	0.0	0.0	662
1544 331	900	155	-5	0.0	0.0	662
1544 332	900	175	-5	0.0	0.0	662
1544 334	900	150	-5	0.0	0.0	662
1544 335	900	140	-5	0.0	0.0	662
1544 337	900	150	-5	0.0	0.0	662
1544 345	950	640	4.0	0.6	173.5	0.3 R062

92-5315
 Shipped
 3-27-72

84.5891
 27,104
 11.7 Kg-0-235
 2,334.8 Kg-0-238

S172-65B
 HP357

Figure 5. Page 2 of the computerized load list.

8. NUCLEAR CRITICALITY SAFETY

At RFP, nuclear criticality safety limits for plant operations, storage, and onsite and offsite transportation for fissile materials were established and audited by the Nuclear Criticality Group according to (1) manuals from the AEC, Energy Research and Development Administration, and DOE and (2) transportation and packing regulations from the Bureau of Explosives, Interstate Commerce Commission, and DOT. Plant operational criticality limits were set differently for wet versus dry material. The wet limits were significantly lower than the dry limits. However, adverse experiences in the 1954-1956 timeframe with liquid-leaking drums during transport to INL precluded shipment of wet waste materials.

8.1 Storage Criticality Limits

The main purpose of providing the nuclear criticality limits for plant storage is to differentiate these limits from nuclear criticality limits for transportation, which were more restrictive for fissile material content per container. A secondary purpose is to establish that plant storage criticality limits were set by the plant criticality group, while transportation criticality limits were established by government agencies.

During the 1950s and early 1960s—before construction of the waste storage and loading facility (Building 664)—waste to be shipped offsite was stored outside. During the late 1960s, indoor storage was established to preclude outdoor contamination; however, because of the lack of indoor storage facilities at that time, recoverable SNM in drums also was stored outside. Consequently, material stored outside may or may not have been shipped offsite.

The earliest documented (November 19, 1962) criticality limit discovered is in a letter, “Outside Storage for Drums of Building 71 Washables” (see Appendix J). Typical washables were rags, paper, plastics, and rubber. Washables were processed by water leaching to remove surface contamination. The water-leached washables were then dried and packaged for offsite shipment.

On March 28, 1962, a letter was issued stating the criticality limits for “Storage of 55-Gallon Drums in Buildings 71, 76, and 34” by the manager of the Nuclear Criticality Group (see Appendix K). The majority of these stored drums was processed to recover the plutonium.

Criticality limits for storage of CWS filters in Buildings 70, 80, and outside were issued in a letter by the Nuclear Criticality Group on August 12, 1965 (see Appendix L). The shipment of CWS filters with high-plutonium content in 1965 was most unlikely as plutonium recovery processes were in place.

8.2 Transportation Criticality Limits

Building 664 served as a loading facility for ATMX-600 railcars and for truck trailers. The nuclear criticality limits for storing waste drums and crates reflect the criticality limits for shipping offsite (see Appendix M.).

Offsite shipping containers are more definitively described with issuance of DOT “Special Permit No. 5948” on December 23, 1968, and as amended on January 30, 1969; February 5, 1969; and March 13, 1969.

In the 1950s and early 1960s, a crate limit for fissile material of 15 g/ft³ was generally accepted. The exact date when this criticality limit was reduced to 5 g/ft³ is unavailable. The earliest documentation for 5 g/ft³ is defined in DOT Special Permit 5948, dated March 13, 1969 (see Appendix N).

A letter of August 21, 1967, from E. H. Lee of the Dow Chemical Company (see Appendix O) requests approval for shipping waste in 55-gal steel drums and addresses average plutonium values for drums shipped in 1966 and 1967. See Table 15 for plutonium averages for the drums shipped.

Table 15. Average plutonium values for drums shipped.

Category A	Non-line-generated waste for 1,582 drums with an average plutonium content of 0.024 g per drum
Category B	Line-generated waste for 944 drums with an average plutonium content of 30.8 g per drum
Category C	Building 774 output, Series 741 first-stage sludge for 398 drums with an average of 5.6 g per drum
742	Second-stage sludge, average plutonium content <1 g per drum
743	Grease sludge, average plutonium content 1 g per drum
744	Neutralized hydrochloric acid and average plutonium content of other liquids <1 g per drum
745	Dried salts, average plutonium content <1 g per drum

9. PROCESS AND PLANT CHANGES

Characterization of the waste shipped to INL has changed because of mission, process, and plant changes and additions. The waste shipped to INL can be characterized generally based on plant operations and activities. The majority of the waste generated by plant operations and activities falls into five categories: (1) housekeeping waste, (2) maintenance waste, (3) process waste, (4) major contamination incident waste, and (5) mission changes. This chapter describes this waste and changes from the 1950s through the 1980s.

9.1 Housekeeping Waste

Housekeeping waste results from operating a facility that houses and handles radioactive materials. Examples of housekeeping waste are surgical gloves, cotton gloves, glove-box gloves, paper, plastic bags, rags, contaminated clothing, wood, tape, and other combustible materials. Over the years, housekeeping waste has remained fairly constant, with variations in quantity associated with production and R&D levels. The rising costs of waste disposal stimulated a waste reduction effort for housekeeping waste. The only notable change was an increase of polyethylene-based plastics and a reduction of PVC plastics.

9.2 Maintenance Waste

Maintenance waste was generated by repair and replace activities such as removal of obsolete equipment, installation of new equipment, upgrading safety systems, area equipment removals, and preventive maintenance requirements. These maintenance activities remained fairly constant over the active life of the plant.

9.3 Process Waste

Process waste was generated by five general operations: (1) foundry and fabrication, (2) component assembly and return disassembly, (3) production support activities such as R&D analytical and metallurgical laboratories and inspection, (4) chemical recovery and metal recycle, and (5) waste treatment.

9.3.1 Foundry and Fabrication Waste

In the 1950s and early 1960s, plutonium components were cast and then machined to final configuration. Component casting was eliminated and replaced with rolling and forming to produce components for final machining. Eliminating component casting reduced graphite mold waste but required plutonium ingot casting for rolling stock. Consequently, graphite mold waste remained constant and varied quantity-wise based on mission requirements. In the 1980s, reusable ingot molds were introduced to reduce graphite mold waste.

9.3.2 Component Assembly and Return Disassembly

The waste generated by component assembly and return disassembly activities was minimal when compared to other operations. The disassembly of returns generated classified waste, which was shipped offsite to Hanford and Nevada Test Site. The introduction of new assembly techniques had little effect on waste generation.

9.3.3 Production Support Activities

Laboratory and inspection support operations generated a small quantity of waste that was minimal compared to the total waste produced. These support operations remained fairly constant throughout the production life of the plant.

The R&D effort grew as production requirements became more complex. Waste generated by R&D activities varied according to mission requirements but did not contribute significantly to total waste produced and shipped. The R&D and special-order work generated small amounts of waste with radionuclides such as Np-237, U-233, and others; however, these activities were fairly constant from 1960 to 1989. Consequently, no significant changes occurred.

9.3.4 Chemical Recovery and Metal Recycle

Chemical recovery and metal recycling processes generated significant quantities of waste; however, these processes did not change greatly during the production life of the plant. Chemical recovery processes were nitric-acid based using stainless steel and glass equipment. The plutonium purification process used a tributyl phosphate-kerosene extraction process in the 1950s. This purification process was replaced by an ion exchange process in the early 1960s. The changeover resulted in shipping spent ion exchange resin to INL but eliminated tributyl phosphate and kerosene waste.

Introducing pyrochemical processes in the 1960s generated mixtures of chloride and fluoride salts. The MSE process for the removal of Am-241 from returned plutonium components generated large amounts of waste. Additional pyrochemical development and pilot plant operations generated additional spent pyrochemical salts in the 1970s and 1980s.

9.3.5 Waste Treatment Waste

Waste treatment operations generated two types of waste: liquid and solid. The Liquid Waste Treatment Plant, Building 774, produced sludge from aqueous solutions through a process of ferric-hydroxide-carrier precipitation. This process was used during the life of Building 774 with very little process change. This process was used also in the new Liquid Waste Treatment Plant, Building 374. Consequently, Series 741 and 742 types of sludge were practically the same over time, the only difference being less water content in the Building 374 sludge.

The treatment of organic solutions began in the mid-1960s using a calcium silicate process to produce an organic-based (Series 743) sludge, a waste product of the Building 774 Grease Plant. The calcium silicate process was replaced by an improved solidification process called OASIS. The OASIS process employed a polymeric substance called Envirostone with an appropriate emulsifier. This process was installed in the mid-1980s.

The process solvents changed with requirements of RCRA. Consequently, solvents listed in RCRA were phased out in the 1970s in favor of unlisted solvents.

Solid waste treatment remained reasonably constant in its constituents. The emphasis was on volume reduction and improved packaging. One of the improvements established in the 1970s was the water washing of leaded glove-box gloves to remove any lead-nitrate complexes. The flammability of leaded dry-box gloves in a nitric acid environment has been reported by Johnson and Lindsay (1969).

The most significant advancement in solid waste treatment was waste segregation. A simple five-type designation described the waste shipped to INL during the 1950s and 1960s. In the 1970s, the waste was described using item description codes. The NDA requirements demanded a more homogeneous matrix, which resulted in 88 description codes for waste; however, the aggregate waste shipped did not exhibit major changes, except as noted above.

9.4 Major Contamination Incidents

Rocky Flats Plant experienced two major contamination incidents resulting from two fires. The first fire occurred in 1957 in the R&D areas of Building 771. Cleanup and equipment removal generated mainly Type I combustibles, Type III CWS filters, and Type V noncombustibles. A second fire occurred in 1969 in the Building 776 plutonium foundry. This fire was much more extensive than the 1957 fire; cleanup and equipment removal generated large quantities and various types of waste.

9.5 Mission Changes

The original mission of RFP included manufacturing components of DU, HEU (Oralloy), and plutonium, coupled with pit assembly.

In 1962, the Oralloy mission was terminated, and the plutonium mission was expanded. Cleaning out the Oralloy plant (Building 881) was completed in early 1965. Consequently, significant quantities of Oralloy waste were generated from 1962 to 1965. After 1965, Oralloy waste was minimized.

Building 444 was originally assigned the mission of fabricating components of DU. In the late 1950s, fabrication of beryllium components was introduced and slowly replaced depleted production of uranium components. Consequently, straight DU waste was slowly reduced during the 1960s with only minor quantities generated during the 1970s and 1980s.

During the 1960s, uranium alloys such as uranium-niobium and uranium-titanium were introduced into Buildings 444 and 883. These alloys became the major source of DU waste in the 1960s, 1970s, and 1980s. These alloys and their respective uses are addressed in Section 4.4.6.

The demand for Am-241 by the ORNL Isotope Pool initiated the chemical recovery of Am-241 from the returned plutonium components. Rocky Flats Plant recovered Am-241 in small kilogram quantities until the early 1980s. The need for Am-241 then dropped drastically, thereby terminating recovery processing. By directive from DOE, the Am-241 was sent to the Series 741 and 742 sludge waste streams. In the late 1980s, however, the americium and plutonium in the MSE salts were removed by an aluminum alloy process and shipped to Savannah River Site. Consequently, in the early and mid-1980s, a higher level of Am-241 was present in the aqueous-based sludge from Building 774.

10. SPECIAL-ORDER WORK

In addition to plant mission assignments, requested special-order work was funded directly by purchase orders. The majority of special-order work was requested by the three design agencies: (1) Los Alamos National Laboratory, (2) Lawrence Livermore National Laboratory, and (3) the two Sandia Laboratories. Special-order work generated waste that was not typical of mission production waste.

10.1 Radionuclide Tracer Program

A variety of radionuclides was used for nuclear device diagnostics by Lawrence Livermore and Los Alamos National Laboratories. Nuclear devices fabricated at RFP for test shots at Nevada Test Site were often traced with certain radionuclides. The most prevalent radionuclide used was Np-237 followed by U-233. Plutonium-238, Pu-242, Cm-244, and Am-241 were employed occasionally but not to the extent of Np-237 and U-233.

Radionuclide tracers were added to plutonium and Oralloid metal in minor quantities to form a tracer alloy. The tracer alloy was in a feed ingot configuration, which was analyzed for tracer concentration and was then used to form the part ingot. The part ingot was rolled, formed, and machined to a final configuration.

Tracer alloy preparation procedures generated scrap and residues that could not be processed by the routine processes for plutonium recovery and metal recycling. Consequently, the plutonium-tracer scrap and residues were processed by R&D Special Recovery or sent to Savannah River Site.

10.2 Neptunium Program

The neptunium program at RFP began in 1964 and terminated in 1988. Neptunium-237 was introduced into uranium (DU and HEU) ingots in a 2–10 wt% range for shipment to the ORNL Y-12 plant. This alloy process will not be addressed, as this effort was very limited and generated a minimum of radioactive waste.

10.2.1 Neptunium Acquisition

Neptunium-237 in an oxide form was purchased by Lawrence Livermore National Laboratory from the ORNL Isotope Pool. The oxide was shipped to RFP and delivered to the Chemical Technology R&D Group. For the duration of the program, plutonium-neptunium oxide residues were shipped to Savannah River Site for plutonium and neptunium recovery. Consequently, Savannah River Site was a secondary source of neptunium.

10.2.2 Neptunium Inventory

Yearly inventory of Np-237 at RFP varied from 29 to 1,318 g from 1963 through 1988. For each fiscal year-end inventory, see Table 16. Based on the fiscal year inventories in Table 16, a 5–10% loss to waste would not represent a significant quantity of neptunium sent to INL.

Table 16. Neptunium-237 fiscal year-end inventories.

Fiscal Year	Neptunium-237 (g)	Fiscal Year	Neptunium-237 (g)	Fiscal Year	Neptunium-237 (g)
1963	29	1972	788	1980	744
1964	601	1973	768	1981	486
1965	1,292	1974	470	1982	699
1966	740	1975	485	1983	869
1967	1,215	1976	485	1984	1,040
1968	972	1976A	468	1985	931
1969	1,190	1977	458	1986	985
1970	1,105	1978	567	1987	995
1971	1,318	1979	492	1988	970

10.2.3 Neptunium Material Balance Area Accounts

The material balance area accounts that handled and processed neptunium material are listed in Table 17. Not all of the listed material balance accounts generated radioactive waste sent to INL.

Table 17. Neptunium material balance area accounts.

Material Balance Area	Building	Calendar Years	Material Balance Area	Building	Calendar Years
361-31	559	1973–1984, 1986–1988	1371-29	771	1977
383-32		1986–1988	1371-31		1974, 1976, 1977, 1985–1988
1375-10	371	1985, 1986	1371-42		1973–1988
1375-37		1986	1371-43		1985
1375-50		1984–1988	1374-31		1974, 1975–1986
1375-70		1983–1988	1374-34		1974–1978
1375-80		1985–1988	1374-50		1977–1979, 1984, 1985
1177-07	707	1984	2418-34		1974, 1975
1476-07		1973, 1974, 1977, 1978, 1985	1371-41	776	1988
1576-07		1974, 1977, 1978, 1983, 1984, 1985	1373-76		1988
214-78	771	1980, 1981, 1982	1576-76		1966–1972, 1973, 1974, 1976, 1977
215-80		1979, 1980, 1981, 1982, 1983			1982–1985

Table 17. (continued).

Material Balance Area	Building	Calendar Years	Material Balance Area	Building	Calendar Years
217-40		1983	215-76	777	1986–1988
217-53		1974–1977	223-77		1977, 1978
217-57		1974–1977	1177-38		1982, 1983
217-72		1974, 1975	1177-53		1974, 1977, 1978, 1982–1988
217-80		1974–1980, 1985–1988	1177-81		1983–1988
218-71		1963–1975	214-74	779	1985
218-72		1965–1975	214-79		1984–1987
218-75		1974, 1975	215-79		1984, 1986–1988
218-82		1965–1975	223-79		1978
223-71		1973, 1974, 1977–1982	1374-79		1988
223-73		1978	223-79	779A	1973, 1978
367-31		1963–1987	1144-18	991	1984, 1985
383-33		1973, 1976–1988	1144-40		1976, 1978, 1982, 1984–1986, 1988
1371-06		1985–1987	1144-91		1979–1981, 1983–1986, 1988
1371-27		1988			

10.2.4 Plutonium-Neptunium Alloy Preparation

Two methods of preparing a plutonium-neptunium alloy were developed. The first method was a co-reduction method in which plutonium oxide and neptunium oxide were mixed and hydrofluorinated to produce their respective fluoride compounds. The mixed fluorides were reduced with calcium metal to form an alloy button. The button was then cast with plutonium metal feed to produce a feed ingot with the required concentration of neptunium.

The second method prepared neptunium metal by hydrofluorinating neptunium oxide to produce a neptunium fluoride, which was then reduced with calcium to produce a neptunium metal button. The button or part of the button was cast with plutonium feed metal to form a feed ingot.

At times, a short ingot was cast and analyzed for neptunium content. A second feed ingot casting was prepared using the short ingot plus the proper amount of plutonium metal feed to produce the specified neptunium concentration.

The co-reduction procedure was initially employed but was discontinued in favor of the neptunium metal addition method. The co-reduction procedure generated more residues and did not consistently produce the desired concentration of neptunium.

10.2.5 Scrap and Residue Generation

Both alloy preparation methods generated scrap and residues that could not be put through the routine processes for recovering and recycling plutonium. Consequently, the plutonium-neptunium scrap and residues were processed by R&D or Special Recovery or sent to Savannah River Site.

Casting, forming, and machining activities generated plutonium-neptunium metal scrap. When feasible, alloy scrap was cast into ingot form for reuse. If the Am-241 content was >50 ppm, the ingot was subjected to MSE to reduce the americium content to <50 ppm. The alloy scrap was stored and was not sent to INL. Three types of residues were generated: (1) mixed oxide; (2) sand, slag, and crucible; and (3) molten extraction salts.

10.2.5.1 Mixed Oxide Residues. The metal alloy chips, turnings, and casting skull were burned to an oxide form. The mixed oxide was processed by nitric acid dissolution followed by ion exchange to mainly recover the plutonium and salvage as much of the neptunium as feasible. However, a small amount of the neptunium was sent to Building 774 in the form of nitrate and chloride solutions. The recovered plutonium in nitrate solution form was transferred to the War Reserve plutonium stream. The neptunium sent to Building 774 was probably in the 1–10 g range.

10.2.5.2 Sand, Slag, and Crucible Residues. The sand, slag, and crucible residue generated by the co-reduction procedure was leached with nitric acid to recover the plutonium and neptunium. The nitrate solution was transferred to the ion exchange procedure described above. The sand, slag, and crucible residue from the neptunium metal reduction procedure was processed by R&D to recover the neptunium in oxide form. The leached sand, slag, and crucible residue was sent to INL if it met the economic discard limit for plutonium only.

10.2.5.3 Molten Extraction Salt Residues. The major quantity of spent molten extraction salts, consisting of a mixture of magnesium, potassium, and sodium chloride salts, was stored for future recovery. The plutonium content as a chloride averaged about 3%; however, R&D developed a limited recovery procedure. Consequently, only milligram quantities of neptunium were sent to Building 774 in chloride and nitrate solutions. No molten extraction salts involving neptunium were sent to INL.

10.2.5.4 Graphite Molds. Casting graphite molds were processed for plutonium recovery through scarfing and nitric acid leaching of the scarfings. Scarfed graphite molds were sent to INL based on the economic discard limit for plutonium. Likewise, leached graphite scarfings were sent to INL based on the plutonium economic discard limit for graphite fines. Consequently, neptunium content in the graphite was in the microgram range.

10.2.6 Waste Generation

Liquid waste was sent to Building 774 and was a constituent in first- and second-stage sludge. Organic liquids also were sent to Building 774 to be processed by the Grease Plant. Considering the dilution factor in Building 774, neptunium concentrations in Series 741, 742, and 743 sludge were insignificant.

Combustible waste was generated but not segregated for recovery, since a significant segment of combustible waste was not line generated. The neptunium content of non-line-generated waste was probably not detectable. The majority of line-generated combustible waste was incinerated and the ashes leached for plutonium recovery. The resulting nitric acid solution was sent to ion exchange for plutonium recovery. Any neptunium present would have been sent to Building 774 in liquid form.

10.2.7 Summary

The number of events at Nevada Test Site that used Np-237 ranged from one to four per year with two per year being the norm. Consequently, the quantity of neptunium shipped to INL compared to plutonium waste can be considered insignificant. However, alpha decay of Am-241 to Np-237 also should be taken into account.

The majority of the neptunium sent to INL was in Series 741, 742, and 743 sludge. Negligible amounts were sent through graphite and sand, slag, and crucible residue.

10.3 Uranium-233 Program

Rocky Flats Plant was assigned the mission for U-233 production. Consequently, requests for fabricating U-233 items were referred to RFP. However, requests received were not mission related, but on a special-order basis.

The majority of U-233 work was associated with device-testing projects at Nevada Test Site. Uranium-233 often served as a tracing material for device-testing diagnostics. The chief requestor was Lawrence Livermore National Laboratory.

Uranium-233 was received from ORNL in oxide form. The U-232 content varied from several hundred parts per million to less than 100 ppm. The decay scheme for U-232 produces daughter products with high gamma emissions. To interrupt the decay sequences, a thorium strike was performed to remove the Th-228 daughter.

General processing consisted of dissolution of the oxide in nitric acid followed by a thorium strike using fluoride ions. Uranium-233 was precipitated as a peroxide, calcined to uranium oxide, hydrofluorinated to uranium tetrafluoride, and reduced to uranium metal with calcium metal and an iodine booster. Resulting metal was cast and machined to the required shape(s).

10.3.1 Scrap and Residues

Machining scrap was burned to uranium oxide. Metal reduction residues were leached with nitric acid for uranium removal. Uranium was precipitated with ammonium hydroxide and calcined to uranium oxide. The oxide was shipped back to ORNL.

10.3.2 Waste Generation

Items that did not contain significant quantities of U-233 were declared waste and were shipped to INL. All combustible waste was shipped. Other items that may have been shipped were glassware, small process equipment items, filters, and other miscellaneous items. Leached reduction residues and graphite casting molds were shipped to INL.

Liquid waste was transferred to Building 774, the Liquid Waste Treatment Plant. Small amounts of U-233 were blended into the Series 741 sludge. Higher levels of U-233 were processed as cemented liquids if necessary. Lead-lined drums were used if necessary to meet the requirement for contact handling.

10.3.3 Processing Facilities

The U-233 special orders were carried out in the Oralloy plant located in Building 881 in the 1950s and 1960s. Decommissioning of Building 881 moved the U-233 special-order work into the plutonium processing facilities. In the 1970s and 1980s, chemistry processing was carried out in the R&D area in Building 771. Casting and machining took place in R&D metallurgical facilities in Buildings 771 and 776.

Plutonium glove boxes were cleaned and scrubbed before introducing U-233. Oak Ridge National Laboratory placed a limit of <30 ppm of plutonium in any residue returned there; however, slightly higher part per million levels were accepted if the returned material could be blended down. If the oxide was contaminated with unacceptable levels of plutonium, it was stored for future use. Other U-233 residues contaminated with plutonium were shipped to INL.

The U-233 projects were monitored by the Health Physics Group to ensure radiation safety and that the principles of as low as reasonably achievable were followed.

10.3.4 Summary

The U-233 program was not a major effort at RFP. The frequency and scope of the projects were less than one per year with a duration of a month or two; however, discard limits were quite high for U-233 as reprocessing capability and capacity were limited.

10.4 Americium Program

The beta decay of Pu-241 to Am-241 provided a source for a single isotope of americium. The Pu-241 concentration in War Reserve plutonium was in the 0.2–0.3 wt% range. With a half-life of 14 years for the beta decay of Pu-241, returned pits could have an Am-241 concentration from 500 to 1,500 ppm. The americium was an unwanted diluent for plutonium recycled metal and did not meet metal specification for pit material. Consequently, removal of Am-241 was necessary to obtain a specification of <50 ppm at the time of casting a plutonium feed ingot.

The demand for Am-241 remained high from the late 1950s to the early 1980s. After that, the demand was greatly reduced, and Am-241 was declared a waste product by DOE.

The americium recovery process was carried out in Building 771 with liquid waste solutions treated in Building 774.

10.4.1 Americium Feed Sources

The plutonium purification process used the precipitation of plutonium peroxide from a plutonium nitrate solution using hydrogen peroxide as the precipitating agent. The impurities and Am-241 remain in the filtrate and precipitate wash solution. These two solutions were the original feed source for the americium recovery process.

In 1967, an MSE process was developed that extracted 90% of the americium content from plutonium metal. The MSE process was carried out in Building 776. Spent MSE salts were sent to Building 771 for plutonium and americium recovery.

The MSE process reduced the americium content in the plutonium peroxide filtrate. The americium content finally reached a level that was not economically feasible to recover. Consequently, filtrate processing was discontinued. The MSE salts then became the sole source of americium feed.

10.4.2 Americium Recovery Processes

The history of the americium recovery process is well presented by Beach and Perry (see Appendix P) in an internal report. This report correlates process changes with dates and indicates processing efficiencies accordingly. For detailed information, consult the Beach and Perry report in Appendix P.

10.4.3 Waste Streams

The majority of waste streams were liquids transferred to the Liquid Waste Treatment Plant Building 774. The report by Beach and Perry (see Appendix P) describes these waste streams.

Solid waste generated was spent anion and cation ion exchange resins, filter pads, and general housekeeping combustibles. Some glassware and other similar noncombustible processing equipment were discarded also. All discards were based on plutonium content regardless of their respective americium content.

10.5 Thorium Program

The thorium program was very limited in scope and was supported by special-order work. Thorium work—natural thorium does exhibit radioactivity—was carried out by the technical staff, which was considered to be an R&D organization. The thorium program was mainly a casting, rolling, and fabrication effort. Thorium scrap was returned to ORNL. A very limited amount of combustibles was generated and sent to INL.

The thorium work was carried out mainly in Buildings 331 and 881 during the early 1960s. No records were searched to identify the thorium work more precisely because the program was limited and had few progress reports available.

10.6 Curium-244 Program

The Cm-244 program was limited to a duration of about 6 months. Curium-244 was introduced into plutonium using the co-reduction technique. The curium-plutonium short ingot was blended into a feed ingot to obtain the concentration of specified parts per million for the Cm-244. The curium-plutonium scrap and residues were stored pending shipment to the Savanna River Plant. Line-generated combustibles and secondary residues were shipped to INL based on their plutonium content.

10.7 Other Special-Order Work

The special-order work reported above generated waste with unique characteristics and constituents compared to regular mission-generated waste. However, significant special-order work that did generate waste typical of regular mission-generated waste was performed throughout the productive life of RFP; this form of special-order work is not addressed in this report.

11. CLASSIFIED WASTE

Rocky Flats Plant was assigned the mission to disassemble returned pits. This mission generated non-SNM classified pit components contaminated with trace amounts of plutonium and Orallo. In addition, rejected and surplus non-SNM classified components added to the backlog of contaminated disassembled components.

Pit foundry and manufacturing processes produced contaminated classified tooling, fixturing, gauging, casting molds, plastic templates, and plastic shapes. The level of pit disassembly was governed mainly by the demand for plutonium to support new weapon builds and by obsolescence factors.

Pit production urgencies coupled with no significant declassification facilities and limited storage capacity prompted RFP to ship the contaminated classified items to Hanford and Nevada Test Site.

11.1 Shipment of Contaminated Classified Material

Rocky Flats Plant was authorized by the AEC, Energy Research and Development Administration, DOE, and sites listed in Table 18 to ship contaminated classified materials to their respective repositories.

Table 18. Classified waste shipment sites.

Site	Time Period
Idaho National Laboratory	1954–1964
Hanford	1958–1984
Nevada Test Site	1958–1964 1985–1989

In 1954, RFP received authorization from AEC Idaho Operations Office and Phillips Petroleum Company (site AEC contractor) to ship solid radioactive waste to the National Reactor Testing Station burial ground at INL. The authorization included contaminated classified waste.

Classification questions were raised by the AEC Idaho Operations Office and Phillips Petroleum that related to the isotopic composition of weapons-grade plutonium and the enrichment level of DU, which were classified at that time. Correspondence by classification personnel confirmed classification of the plutonium isotopic composition and enrichment level of DU; however, general low-level contaminated plant waste sent to INL continued as unclassified shipments. The reasoning associated with this decision was not addressed in the historical correspondence reviewed. The isotopic composition of weapons-grade plutonium was declassified in April 1964, and the enrichment level of DU was declassified shortly thereafter.

The AEC courier receipts were obtained for 1963 and early 1964 indicating the shipment of classified material to INL (see Appendix Q). Unfortunately, the courier receipts do not identify the contents of the classified material. At that time, RFP had a large backlog of low-level plutonium residues that exceeded the economic discard limit. Rocky Flats Plant requested permission from the AEC Albuquerque Operations Office to send the residues to INL for burial. Permission was granted, but RFP was required to provide health physics escorts for safety reasons.

The courier receipt in Appendix Q indicates that the radioactive waste is classified. The corresponding waste shipment data sheets for Trailers TZ-20 and TZ-21 (see Appendix R) do not indicate that classified shapes were included. The load lists for Trailers TZ-20 and TZ-21 (see Appendix S)

indicate that the types of waste were mainly combustibles (Type I) and sludge (Type IV). However, noncombustibles (scrap metal) are indicated by Type V or Type 5 for two drums originating from Buildings 776 and 771, respectively, for Trailer TZ-20. The load list for Trailer TZ-21 indicates only combustibles (Type I) and sludge (Type IV). There is the possibility that the two drums of Type 5 scrap metal were classified shapes. However, the normal operating practice for loading trailers would be to store the classified drums until a full trailer load was available. Consequently, it is highly unlikely that the two Type 5 drums contained classified shapes.

The higher quantities of plutonium may have been the reason for using the AEC courier system. Classified courier shipments were terminated in 1965. The declassification of plutonium in April 1964 also may have contributed to terminating these classified courier shipments.

The INL personnel have repeatedly inquired about the possibility that classified shapes have been sent to INL. A search of trailer load lists, ATMX-600 railcar load lists, waste shipping memos, 741 forms, and traffic correspondence provided no direct evidence that classified shapes were shipped to INL. Inquiries of former and present RFP waste and traffic personnel revealed no recollection of classified shapes being sent to INL.

A letter from the Dow Traffic Group to the Nuclear Engineering Company states that no classified waste was shipped to INL during the period from July 1963 through May 1964 and confirms that the classified waste was sent to Hanford during this period (see Appendix T). However, there is the possibility that a stray classified item may have been shipped inadvertently in an unclassified container.

12. SPECIAL TOPICS

Communications with various INL personnel revealed specific topics of interest that are addressed in this section.

12.1 Graphite Waste

Graphite waste was the result of foundry casting operations. Graphite molds were used for casting DU, HEU, and plutonium into various metal forms. The molds were fabricated in a carbon shop located in Building 445, which was attached to Building 444.

Before casting, a coating was applied to the mold surface to prevent reaction of the graphite with the molten metals. The most prevalent mold coating was calcium fluoride followed by yttrium oxide. Other oxide-based mold coatings were used experimentally.

Spent molds were often scarfed to remove any residual metal or metal that may have breached the mold coating. The HEU and plutonium scarfings were leached with nitric acid to recover the uranium and plutonium. The leached scarfings that were really graphite fines were packaged into a suitable container and placed in a drum for shipment to INL. Not all spent molds were scarfed if they were below the economic discard limit.

Scarfings from the DU molds were calcined to convert the uranium metal to an oxide form. Calcined material was packaged and placed in a drum for transport to INL. Rocky Flats Plant did not have chemical recovery facilities for recycling DU. Converted uranium oxide was often identified as "RO" on the trailer and ATMX-600 load lists.

Graphite shipped to INL had a variety of physical forms (e.g., there were whole molds, fines, pieces, chunks, and partial molds). Undoubtedly, a small amount of fines was present with any of the solid graphite mold configurations. The graphite stock employed to fabricate the molds was a high-density, nuclear-grade extruded graphite (see Appendix U).

12.1.1 Plutonium Graphite Molds

The Zodtner and Rodgers (1964) report has caused concern at INL because of the significant quantity of plutonium arbitrarily assigned to graphite molds in the 1950s and early 1960s. Since INL is concerned primarily with plutonium molds, no further discussion of DU and HEU molds will be included in this section.

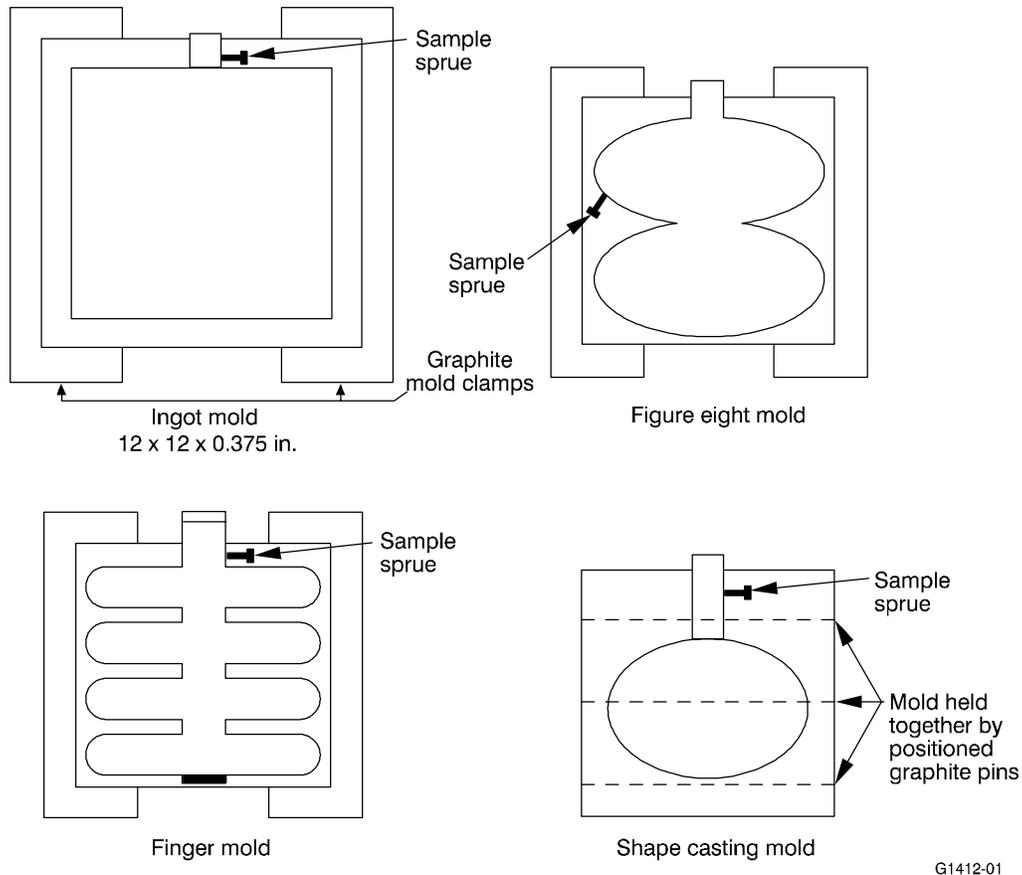
12.1.1.1 Generation Rates. The early shipments of radioactive waste to INL were defined by five designated types. Graphite molds and segments were assigned a noncombustible Type V. The noncombustible Type V included scrap metal, glass, ceramics, and graphite. A drum listed as Type V could contain a mixture of noncombustible material. Consequently, this system precludes identifying graphite mold waste through shipping records. Available waste documentation does not include shipping records for graphite molds, nor do records of building operations indicate generation rates of graphite mold waste.

This nomenclature continued until the late 1960s, when a new system of material designation was introduced using a digital system of item description codes. Introducing this system was driven by the development of economic discard limits and nondestructive radiometric instrumentation (i.e., drum and can counters). The system of item description codes precluded commingling noncombustible waste and provided a way to record shipments of graphite waste to INL.

From 1953 to 1956, plutonium foundry operations were carried out in Building 771 on a limited scale. Plutonium casting operations used a split copper mold configuration. Consequently, very little (if any) graphite waste was generated.

In 1957, Building 776 came into operation and began using graphite molds. The initial graphite molds were porous and of low density. High-density, nuclear-grade graphite later replaced the less dense graphite in foundry operations.

12.1.1.2 Types of Plutonium Graphite Molds. For War Reserve production, the four general configurations of molds are illustrated in Figure 6.



G1412-01

Figure 6. Four general configurations for graphite molds.

The above mold dimensions and the location and number of sample sprues would change somewhat for different weapon programs, although the general mold shapes would remain fairly constant.

Most of the plutonium molds were associated with War Reserve mission work; however, special-order work and R&D mold experimentation generated a minor quantity of molds with a variety of configurations. Since the molds had to conform dimensionally to the casting furnaces, the outer dimensions were reasonably constant.

12.1.1.3 Foundry Casting Operations. The graphite molds were coated with calcium fluoride to control plutonium penetration into the graphite mold. Later, yttrium oxide was used also as a mold

coating. The split molds were clamped using graphite clamps, which were discarded when defective. Shape molds varied in configuration and were clamped using graphite pins.

These molds were constructed to fit into vacuum-induction casting furnaces and to accept a molten charge of plutonium metal. On cooling, the mold was removed from the furnace and transferred to a break-out glove box. The mold was disassembled and the plutonium item removed. The sample sprue was cut off and packaged for transfer to the analytical laboratory for elemental analysis and plutonium assay. Part of the sample sprue was saved, should reanalysis be required. The surface of the plutonium item was brushed to remove any loose particles. The plutonium item was then transferred to the next scheduled operation. Any identifiable plutonium pellets and casting slag were collected and placed in a container for transfer to plutonium recovery in Building 771.

The brushings, glove-box floor sweepings, and collected graphite fines were added to the mold drum. The plutonium content added to the drum was estimated by using by-difference techniques. When the drum reached approximately 500 lb, the drum was sealed and sent to waste disposal.

In 1965, an experimental drum counter became available for measuring the plutonium content of graphite mold drums. The drum counter provided improved assay values and instituted reprocessing for plutonium recovery. These activities necessitated improvements in plutonium waste segregation. As a consequence, mold break-out fines and sweepings were placed in a gallon-sized paint can for plutonium recovery processing. Later, the development of a can counter reduced the collection container to a quart-sized, specially manufactured, stainless-steel can to improve the plutonium assay.

12.1.1.4 Plutonium Estimates on Graphite Molds. The drum counting reports describing graphite assaying indicate that graphite mold waste was given priority over other waste forms. However, other than Zodtner and Rogers (1964), no formal documentation has been located that defines plutonium estimates in graphite molds. An informal reference for the average plutonium content for line-generated waste was 30 g (see Appendix C). Graphite molds are a component of line-generated waste.

The first workable drum counter was completed in 1964 and installed in Building 771. Internal reports describing the counting efforts and results for graphite indicate that the plutonium results in graphite drums were low and required adjustment by a factor of 1.8 ± 0.18 (see Appendixes V and W).

12.1.1.5 Plutonium Recovery for Graphite Molds. As stated previously, spent graphite molds were broken up with a ball-peen hammer and placed in a 55-gal drum with an 8-mil plastic liner. When full (~226.8 kg [500 lb]), the drum was closed and assigned a by-difference plutonium value.

Starting in the mid-1960s, the mold surfaces were scarfed with a needle gun. Surface grinding was also employed. The scarfings and grinding powder were collected for plutonium recovery. The scarfed or ground graphite pieces and chunks were placed in a 55-gal drum for shipment to INL.

The leaching procedure used to recover plutonium from graphite was based on the experimental work reported by Williams and Pinamont (1965). This report also observed that plutonium assays of graphite feeds were low by a factor close to two. The collected scarfing or grinding powders were placed into dissolver pots. The original dissolver pots were glass but later were replaced with Teflon pots with a capacity of 3–4 L (3.2–4.2 qt). Appropriate amounts of nitric acid were added to the pots to dissolve the contents. The nitric acid solution was heated using an electric immersion heater. The dissolved plutonium solution was removed leaving a graphite heel in the bottom of the dissolver pot. The heel was scraped out of the pot and placed in a stainless steel pan. The heel was dried on a hot plate and was removed from the pan using a hammer to break up the solidified graphite material. The removed graphite material was

placed in a plastic-lined 55-gal drum. When the drum was full, the drum was sealed and labeled, and an estimated gram value, usually zero, was assigned to the drum.

In the late 1960s, the leached graphite was packaged into paint cans and was assayed with a can counter. Later, the volume of the leached graphite container was reduced to 0.9 L (1 qt) for an improved plutonium assay. Polyethylene bottles also were used as graphite containers.

The decision to process discarded molds was based on reducing the material unaccounted for. Consequently, scarfing, and grinding procedures were coupled with leaching the graphite removals using nitric acid to recover the plutonium. The scarfed graphite pieces contained minimal quantities of plutonium.

Breaching of the mold coating by the molten plutonium was the chief cause of plutonium loss to the mold. Using low-density graphite molds contributed to the plutonium loss. Using high-density, nuclear-grade graphite for molds helped alleviate the breaching problem.

12.1.1.6 Waste Management of Graphite Molds. Using graphite molds for plutonium casting was implemented in 1957 in Building 776. Before 1957, copper molds were used in Building 771. Plutonium mission requirements were fairly constant from 1957 to 1962 with a modest use of graphite molds. From 1963 to 1970, mission requirements of plutonium components increased significantly, which resulted in a large inventory of spent graphite molds.

The packaging of spent graphite molds from 1957 to 1962 was completed by using either a 5- or 8-mil polyethylene liner inside a 55-gal drum. The type of drum was not standardized, and vendor drums often were used. The increased demand for 55-gal drums resulted in purchasing 17-H and 17-C drums.

The early graphite mold drums were usually considered full when their weight was about 500 lb. Later, a limit of 90.7 kg (200 lb) of graphite per drum became a requirement. The letter from William F. Romine, Traffic Manager, in 1968 quotes a limit of 90.7 kg (200 lb) of graphite per drum for nuclear criticality safety reasons (see Appendix X). A certificate of approval issued in 1969 for fissile-large quantity shipping containers for ATMX-600 railcars issued by the AEC Albuquerque Operations Office quotes a limit of 45.4 kg (100 lb) of graphite for 30-gal drums and 90.7 kg (200 lb) for 55-gal drums for nuclear criticality safety reasons (see Appendix Y).

Graphite molds in the late 1960s and graphite mold segments were scarfed or ground to remove surface plutonium. The surface removals were placed in metal containers and bagged from the glove-box line into 55-gal drums. The scarfed or ground segments were placed directly into the 55-gal drums. This general drum-packaging configuration remained reasonably constant until the concept of retrievable storage—driven by the AEC directive (AEC 1970)—became the rule.

12.1.1.7 Miscellaneous Graphite Waste. Analytical laboratories in Buildings 441, 771, and 881 generated graphite electrodes contaminated with DU, plutonium, and HEU, respectively. The graphite electrodes were used in emission spectroscopy to determine elemental impurities in DU, plutonium, and HEU matrix products. The spent electrode charge was removed from the graphite electrode cup by tapping into a stainless steel vessel. The spent anode graphite cup electrode and the pointed spent cathode graphite rod electrode were placed into a second stainless steel vessel. However, the spent electrode charge was not removed from the graphite anode cup electrode for DU analysis. The collected spent electrode charges for plutonium and HEU were sent to chemical recovery to reclaim the plutonium and HEU. The collected spent electrodes were packaged into the noncombustible (Type V) waste category.

The anode cup graphite electrode was fabricated from a graphite electrode (~0.97-cm [0.38-in.] diameter), while the upper cathode graphite electrode was a pointed graphite rod (~0.33-cm [0.13-in.] diameter). Both electrodes were about 3.81 cm (1.5 in.) long.

To surpass the complicated emission spectra of uranium and plutonium, a carrier was added. The carriers employed were silver chloride, gallium oxide, and sodium fluoride. Consequently, the spent electrodes may be contaminated with these compounds. These spent graphite electrodes were not a significant contributor to the graphite waste shipped to INL.

Other graphite-based items were spent graphite bearings from chemical pumps, graphite pegs to hold molds together, and experimental casting configurations generated by R&D efforts. These items were not significant generators of graphite waste.

12.1.2 Plutonium Discard Limits

In the late 1950s and the early 1960s, RFP recognized that significant quantities of plutonium were being lost through the solid waste streams. This fact dictated the need for improved plutonium measuring systems and the installation of appropriate recovery systems to return the plutonium to the War Reserve production stream and to reduce the plutonium levels in the waste shipments.

In 1960, a project provided plutonium recovery capability and capacity for treating low-level solid waste, including graphite. The project included the development of a waste drum counter. Construction was completed in 1962, and the new recovery systems were placed online in the mid-1960s.

12.2 Roaster Oxide

Depleted uranium operations produced pyrophoric uranium fines, turnings, chips, chunks, and casting skull. Disposal of these pyrophoric items was a constant problem, as no DU recovery facilities were available at RFP. Shipment to other DU sites was not feasible because of their pyrophoric condition. In 1956, a calcining system was constructed to oxidize these pyrophoric materials to a uranium oxide form. This system was upgraded several times during its lifetime into the 1980s.

12.2.1 Roaster Oxide Description

Uranium oxide is a black, fine powder; however, added impurities, such as heat-treating salts, will alter the color from black to a black-gray depending on the uranium oxide content.

Roaster oxide disposal was often abused by including hunks of uranium metal, tooling, work gloves, and other items contaminated with DU.

12.2.2 Chip Roaster

The chip roaster feed was mainly DU and DU alloy chips, turnings, saw filings, and any other source of uranium metal collections. These metallic forms were introduced into a calcining system designated as the chip roaster. Consequently, the resulting uranium oxide powder was called roaster oxide. On shipping lists, it was listed as RO to signify DU oxide. To remove any RCRA constituents, the chips and turnings were steam-cleaned before being introduced into the chip roaster in the 1980s. The operation of the chip roaster was shut down from 1959 to 1961 because of its relocation in Building 444 (see Appendix Z).

12.2.3 Casting Skull

Uranium and uranium-alloy casting skull were burned in a foundry burn box to an oxide form. A large crucible was employed to burn the collected skull material. The burned (oxidized) skull oxide was loaded into a 30-gal drum and labeled roaster oxide.

12.2.4 House Vacuum System

The house vacuum system collected any spilled or extraneous uranium oxide or any other dry powder form. A cyclone separator was positioned in the house vacuum system to remove the powder material from the house vacuum stream. The collected material from the cyclone separator was packaged in 30-gal drums and labeled roaster oxide.

12.2.5 Sintered Metal Filter Collections

Sintered metal filters were located in the chip roaster exhaust system and in other exhaust systems including the house exhaust system. The collected particles on the metal filters were removed and composited and then placed in a 30-gal drum and labeled roaster oxide.

12.2.6 Water-Quenched Tanks

Sludge from the water-quenching tanks was created from heat treating and rolling mill operations in Buildings 444 and 883. The sludge accumulated from surface uranium oxide and residual heat-treating salts. The quenching water was drained and sent to Building 774. The wet sludge was collected, air dried, and packaged in 30-gal drums and labeled roaster oxide or sludge. At times, the sludge collected was introduced into the chip roaster. If the sludge burned, it was labeled roaster oxide. If not, it was labeled sludge.

12.2.7 Furnace Box Stubs

Uranium metal strips were bolted together and heated in an arc melting furnace. The center melt was poured into a mold configuration, while the end portions were discarded as box stubs. The box stubs were combined with the oxide from the chip roaster. The drums with box stubs were identified as roaster oxide.

12.2.8 Packaging Roaster Oxide

The roaster oxide was placed in a 30-gal drum that was overpacked with a 55-gal drum. Vermiculite was added to fill the void space between the two drums. The outer drum was wiped to remove any residual contamination. The 55-gal drum was labeled and prepared for shipment. Later, a plastic 55-gal drum liner and a cardboard disc cover were used in the packaging configuration.

12.2.9 Roaster Oxide Pyrophoricity

Stakebake and Osborn (1994) evaluated the potential pyrophoricity of the roaster oxide, concluding that there was a low probability that the few uranium chips that may be present in a drum of roaster oxide would ignite and if so, the major matrix of oxide would absorb the heat produced to preclude drum rupture.

12.2.10 Roaster Oxide Shipments

A summary of the DU waste shipped to INL from 1954 through 1970 is shown in Table 19. Table content was generated from monthly and annual history reports by the Waste Disposal Coordination Group.

Table 19. Summary of depleted uranium waste shipments to Idaho National Laboratory based on reports by the Waste Disposal Coordination Group.

Calendar Year	55-Gallon Drums	40-Gallon Drums	30-Gallon Drums	Boxes ^a	CWS Filters ^b	Tanks	Total Volume (ft ³)	Gross Weight (lb)	U-238 ^c (kg)
1954	1,217 ^d	—	—	—	—	—	—	—	738
1955	1,564	—	115	—	—	—	12,248	390,104	979
1956	1,795	—	—	—	—	2	12,347	315,727	1,174
1957	1,882	—	300	—	460	—	22,176	863,800	2,147
1958	818	37	220	—	327	—	8,055	283,938	4,209
1959	692	—	97	4	—	—	5,323	200,380	3,753
1960	839	—	28	17	—	—	6,866	230,913	4,123
1961	1,030	—	37	29	333	—	10,236	268,708	4,311
1962	839	—	4	24	—	—	6,775	208,882	4,674
1963	1,510	—	3	24	92	—	12,629	286,966	1,672
1964	2,058	—	—	42	93	—	19,381	386,931	1,339
1965	1,479	—	—	41	—	—	15,742	326,797	4,269
1966	1,488	—	—	31	—	—	14,509	420,113	53,452
1967	1,473	—	—	64	—	—	18,434	498,914	53,176
1968	1,491	—	—	44	—	—	16,216	390,470	33,373
1969	1,087	—	—	40	—	—	13,028	326,098	22,721
1970	567	—	—	63	—	—	11,252	172,383	7,084
Totals	21,829	37	804	423	1,305	2	205,217	5,571,124	203,194

a. The standard size waste box was 4 × 4 × 7 ft. Some boxes of slightly different sizes were shipped.

b. "CWS filter" was terminology used for what are now high-efficiency particulate air filters. Most were 2 × 2 × 1 ft in size, shipped in boxes in early years. In later years, some were shipped in drums.

c. Data on total weight of U-238 shipped came from a separate summary report and were not related to individual containers.

d. The report for 1954 did not break out the drum size, volume, or weight.

CWS = Chemical Warfare Service

12.3 Ion Exchange Resins

Ion exchange resins were used extensively from the late 1950s through 1989. The three major ion exchange resins employed for plutonium recovery were: (1) Dowex 1 × 4 [50-100] mesh, (2) Dowex II [20-50] mesh, and (3) Amberlite IRA-938 [20-50] mesh. These resins were used in a nitrate form. Dowex 50 × 8 cation resin was used in the americium separation process. The major resin used was Dowex 1 × 4 anion resin (nitrate form) with Dowex 50 × 8 cation resin in second place. Amberlite IDA-938 was used in special recovery processing during the 1970s and 1980s. Other ion exchange resins were employed by R&D and the analytical laboratories for evaluation and specific applications. Very limited amounts were involved in these activities.

Spent ion exchange resins were water-washed to remove residual acids. Two disposal methods were used: (1) incineration and (2) mixing with cement for shipment to INL. During the 1960s, spent ion exchange resins were mixed with cement in a 1-1 ratio. Later, the ratio was altered to 1.5 parts resin to 1 part cement as indicated in a letter from William F. Romine (see Appendix AA).

12.4 Decommissioning Building 881

As discussed previously, Building 881 discontinued producing Oralloy (HEU) components in the early 1960s. Decommissioning activities generated significant quantities of Oralloy waste, including capital equipment. Appendix BB illustrates the types of waste generated. The items marked with a date stamp were shipped to INL. Those items not marked were processed internally, stored, or both. The organic liquid waste was stored on the 903 Pad and processed later by the Grease Plant, Building 774.

12.5 Discard Mud

Impure Oralloy-bearing materials were crushed in a rod mill into pea-sized feed. The crushed feed was leached with nitric acid followed by a filtration step. The insoluble residue was called discard mud and had very little U-235 remaining. The major source of the discard mud was sand, slag, and crucible material generated from the bomb reduction of uranium tetrafluoride with calcium metal to form a uranium metal button. The major chemical constituents of the sand, slag, and crucible residue were magnesium oxide sand, magnesium oxide crucible, and calcium fluoride slag. Other materials such as furnace liners (alumina) and uranium chip-burning graphite pots were also introduced to the rod mill operation but in minor quantities. The discard mud was composed of rod mill constituents not soluble in hot nitric acid.

12.6 Plutonium-Contaminated Oralloy Waste

Several items in Appendix CC are listed as contaminated with traces of plutonium. The disassembly of returned pits generated Oralloy components contaminated with surface plutonium. Building 881 removed the surface plutonium contamination with a dilute nitric acid rinse, followed by a water rinse. The leached HEU component was dried and subjected to radiometric scan to determine any plutonium content. Plutonium-free HEU components were returned to the HEU foundry. Later, after decommissioning the HEU foundry, the HEU components were shipped to the ORNL Y-12 plant.

The leach solution was processed in a glass evaporator. The high-acid condensate was stored in a ring-packed tank and recycled to the leaching container as clean acid. The evaporator concentrate was treated with ammonia gas to precipitate the actinides. The precipitate was calcined to an oxide and shipped to the ORNL Y-12 plant if the plutonium content was very low or to Savannah River Site if the plutonium content was too high (Giebel 1964). The leaching and rinsing process generated HEU and

plutonium-contaminated solid waste such as rags, absorbent wipes, filters, and other solid waste, which were packaged and shipped to INL.

Leaching of returned Oralloy was transferred to Building 771 in the 1970s. The HEU leach solution with plutonium contamination was processed in the special recovery area using a modified plutonium and uranium recovery by extraction (commonly called PUREX) process to separate the uranium and plutonium. The plutonium fraction was transferred to the War Reserve plutonium recovery stream. The uranium fraction was precipitated with ammonia gas and calcined to uranium oxide. If the plutonium was <500 ppm, this oxide was shipped to the Idaho Chemical Processing Plant (now Idaho Nuclear Technology and Engineering Center) at INL.

12.7 Oralloy-Contaminated Equipment

Building 889 was constructed as a decontamination facility for DU- and HEU-contaminated items that could be used elsewhere in the plant. However, a significant amount of Oralloy-contaminated equipment was identified as surplus and was crated and shipped to INL (see Appendix DD). The chemical constituents in the bath salts were sodium, potassium, and lithium carbonates, which formed a eutectic salt bath. A property disposal record form is employed to write-off capital items. The D83 numbers are capital equipment numbers and not drum numbers in this case.

12.8 1969 Fire Waste

On May 9, 1969, RFP experienced a disastrous fire in the plutonium foundry, Building 776. The fire cleanup generated a significant quantity of plutonium-contaminated waste. For costing information and to differentiate fire recovery waste from routine-generated waste, a special identification number of (771-596) was used. Special numbers (A00----) were assigned to each crate shipped to INL. See Appendix EE for a list of the crated fire waste shipped, which covers shipments over a period from October 1969 to February 1972. Included in Appendix FF is a copy of the procedure employed to determine the amount of plutonium on equipment and machine tools and a copy of a letter addressing the limitations associated with wooden waste shipping crates in ATMX-600 railcars.

12.9 903 Pad Disposal

The generation of machine cutting oil and other organic solvents grew into a storage and processing dilemma. This situation forced outside storage on the 903 Pad. A treatment process was finally developed in 1966 that converted the organics into a grease and was identified as Series 743 sludge. Processing the estimated 5,230 drums began in January 1967. The chronology of processing is shown in Table 20 with a summary provided in Table 21.

Table 20. 903 Pad drum processing chronology.

Time Schedule—Processing Lathe Coolant Solutions (from M. Maas monthly progress reports)	
1/23/67	Planning started for processing lathe coolant solutions.
3/67	Start of 2-month trial period for processing lathe coolant. 191 drums processed during trial run.
5/67	Decision made to process drums of coolant from 903 storage area. Estimate 5,230 drums of organic liquid in 903 storage area.
11/27/67	Emptying of drums in the field is 70% complete.
1/25/68	3,860 drums removed to date, 77% complete.
2/27/68	Approximately 1,250 (24%) empty drums removed to date.
3/29/68	4,146 (78%) drums of organic waste removed to date.
4/26/68	2,614 (53%) empty drums removed from 903 storage area to date.
5/21/68	Last pumpable contaminated organic liquid removed from 903 storage area.
5/28/68	Final 41 drums of unpumpable greases and tars removed from 903 storage area.
6/7/68	Completed removal of all drums and pallets. Final policing of 903 storage area complete.
9/24/68	Final report on removal of all drums from 903 storage area.

Table 21. Disposal summary of plutonium-contaminated materials, 903 Pad storage area.

Total drums at start	5,230 drums
Drums sent to Building 774 for processing	4,826 drums
Drums containing plutonium contamination	3,572 drums
Empty drums (includes original empty drums plus drums emptied after solution removal)	4,672 drums ^a
Total drums processed in Building 774 with an average of 1.7 g of plutonium per drum	4,826 drums

a. All 4,672 empty drums were assayed before packaging and shipping to Idaho National Laboratory.

12.10 Asbestos and Miscellaneous Waste Items

Using asbestos items was very common in the Building 444 uranium foundry. The following is a list of the asbestos items used in foundry operations and discarded to waste when contaminated with DU:

- Aprons
- Fire blankets (4 × 8 ft)
- Gloves

- Jackets
- Hoods
- Shin guards
- Tape.

These items could have been discarded as Type I combustibles, Type V noncombustibles, or both, depending on the operator's discretion.

Other sources of asbestos in all areas were HEPA filters, filter media, and furnace insulation.

A list of materials that were discarded to waste from Building 444 is provided below. These items made up a very minor quantity of the waste generated and shipped:

- Grinding wheels and motors
- Unclassified tooling
- Cadmium turnings from back machining
- Chromium plating turnings from back machining
- Lead casting residues—skull and turnings
- Aluminum chips, turnings, and casting skull
- Copper turnings and casting skull
- Spent furnace brick
- Contaminated furniture.

12.11 High-Efficiency Particulate Air Filters

Buildings handling radioactive material at RFP used various stages of HEPA filtration on the glove box and corresponding exhaust and intake systems. The spent exhaust filters became a disposal problem because of plutonium contamination levels, acid fumes, and other aerosols. Extensive R&D and pilot experiments were carried out on a continuing basis to provide the most effective and efficient filtration systems. The most significant information will be addressed in this section. See Appendix GG for a list of reports on efforts to improve the filtration systems.

12.11.1 Background

From 1953 to 1957, all plutonium production work was carried out in Buildings 771 and 991. Chemical recovery, metal recycle, foundry, and machining operations were performed in Building 771. Final assembly of plutonium components and shipping preparation was accomplished in Building 991. Very little plutonium waste was generated by Building 991. Building 774 was constructed to process the aqueous radioactive waste generated by Building 771 and the uranium buildings on plant site into sludge for shipment offsite.

In 1957, Buildings 776 and 777 were constructed to accommodate the increased plutonium mission. Foundry and machining operations were moved to Building 776, while Building 777 handled final assembly operations. A concrete block wall separated the two facilities within the same structure.

Chemical recovery and metal recycle remained in Building 771. Foundry and machining capabilities in Building 771 were taken over by the R&D group.

Production operations in Building 776 were essentially dry processes, except for the degreasing solvents and lathe coolant, which were organic-based liquids. The recovery and recycle processes in Building 771 were mainly aqueous processes using large quantities of nitric acid. Consequently, the glove-box atmospheres were much more corrosive in Building 771 than in Buildings 776 and 777, which adversely affected the HEPA filters in Building 771 compared to Buildings 776 and 777.

12.11.2 Plutonium Ventilation Systems

The ventilation filtration systems in the plutonium buildings can be segregated into two connecting units:

- From the processing glove boxes through the HEPA filters in the booster system
- From the booster system through the final HEPA filters in the final plenum and out the exhaust stack to the atmosphere.

The booster and final plenum systems have two filter banks that are called first- and second-stage HEPA filters. The glove boxes have a small intake HEPA filter and a small exhaust HEPA filter. The intake filter is discarded normally as non-TRU waste. Glove-box operations that produce minimal amounts of dust and aerosols have exhaust filters that will satisfy criteria for non-TRU waste.

12.11.3 Before 1964

Zodtner and Rogers (1964) addressed, in part, plutonium loading onto HEPA filters for Buildings 771 and 776. Plutonium loading was reported based on the two filtration units described above.

First- and second-stage HEPA filters in the booster system for Building 776 generally load to approximately 50 g of plutonium for the lifetime of the filters as determined by radiometric counting techniques. For the final plenum filters, plutonium loading for the lifetime of the filters was negligible based on radiation counting procedures.

Booster and final plenum filters were packaged and shipped offsite. Plutonium loading for the booster filters from Building 776 was estimated by Zodtner and Rogers (1964) as 25 g per filter. The reduced plutonium loading value was to compensate for filters that were removed before full plutonium loading because of filter damage and premature plugging by foreign material.

First- and second-stage HEPA filters in the Building 771 booster system became highly contaminated with plutonium attributed to the chemical processing involved in plutonium recovery. Plutonium loading for the booster filters has been determined to average about 300 g per filter. This estimate is based on burning a full set of booster filters and assaying the resultant ash. The estimated plutonium value for shipping these filters offsite was 200 g per filter as assigned by Zodtner and Rogers (1964). Again, the plutonium loading per filter was reduced to compensate for premature removal of some filters before reaching full loading. For first- and second-stage final plenum filters, Zodtner and Rogers assigned an estimated plutonium loading value of 2 g per filter.

Plutonium loading of HEPA filters in ventilation systems for Buildings 774 and 991 was considered to be negligible. Consequently, the HEPA filters for those buildings were considered low-specific activity waste in the terminology of that time.

The Zodtner and Rogers (1964) report estimates that 70% of the 2,358.7 MT (2,600 tons) of waste sent to INL was generated as building waste. This waste was only slightly contaminated and should not contribute heavily to the burial ground contamination. About 15% of the waste sent originated from within the glove-box lines and was the main contributor to the plutonium levels in the burial pits. The remaining 15% of the waste sent consisted of obsolete and nonrepairable equipment and bulky items too large for a 55-gal drum. These contaminated items were disposed of in large wooden waste boxes and were a minor contributor to the plutonium levels.

The two major plutonium waste generators were Buildings 771 and 776 with a minor amount coming from Building 777. A waste generation distribution is shown for Buildings 771 and 776 in Tables 22 and 23, respectively.

Waste generated by the glove-box line consisted of combustibles such as rags, paper, plastics, and rubber gloves. Noncombustibles consisted of scrap metal, broken glass, heavy rubber items, and small process equipment items. A small quantity of graphite was included with the line-generated waste.

Table 22. Waste shipments to Idaho National Laboratory from Building 771.

Calendar Year	General Building Waste		Glove-Box Line Waste		Wooden Box Waste	
	Number of Drums	Weight (ton)	Number of Drums	Weight (ton)	Number of Boxes	Weight (ton)
1954	1,036	31	40	3	5	—
1955	1,383	78	60	4	6	1
1956	1,825	103	80	5	7	1
1957	3,180	217	130	9	57	16
1958	2,143	145	90	6	122	37
1959	2,609	165	110	7	105	34
1960	2,414	173	100	7	124	69
1961	2,301	144	130	9	48	18
1962	3,426	222	110	7	103	54
1963 ^a	1,183	112	50	3	73	66
Totals	21,500	1,390	900	60	650	296

a. 6-month total.

Table 23. Waste shipments to Idaho National Laboratory from Building 776.

Calendar Year	General Building Waste		Graphite Waste		Washable Waste		Miscellaneous Waste		Wooden Box Waste	
	Number of Drums	Weight (ton)	Number of Drums	Weight (ton)	Number of Drums	Weight (ton)	Number of Drums	Weight (ton)	Number of Boxes	Weight (ton)
1957	71	2	5	1	5	1	30	2	0	0
1958	447	24	100	12	25	2	100	10	7	1
1959	753	35	150	19	50	3	200	20	27	7
1960	721	44	110	14	43	3	95	9	26	11
1961	1,092	73	451	49	60	4	188	20	36	13
1962	1,409	87	521	68	83	6	169	18	26	11
1963 ^a	777	48	0	0	0	0	51	5	3	4
Totals	5,270	313	1,337	163	266	19	833	84	125	47

a. 6-month total.

The contents of the boxed waste consisted of spent process equipment, duct work, and piping, which had many inaccessible areas that could retain substantial amounts of plutonium. Sections of duct work and piping, when removed, were immediately sealed and placed in a box without cleaning. At that time, no reliable or practical method of measuring the plutonium was available. Consequently, the plutonium shipped offsite in these boxes could vary from contamination count levels to hundreds of grams.

Waste-generating streams in Building 776 can be segregated into five categories listed below:

1. General building waste
2. Graphite waste
3. Washable waste
4. Miscellaneous waste
5. Boxed waste.

General building waste was similar to that for Building 771 and was considered to be slightly contaminated with insignificant amounts of plutonium.

Graphite waste was generated by the plutonium foundry through casting procedures. Graphite molds were probably the largest contributor to plutonium sent to INL in the early years. Building 776 washable waste consisted of rubber-and plastic-based materials. These materials received a washing before being sent offsite. The miscellaneous waste generated in Building 776 was given a superficial cleaning before being placed in 55-gal drums for shipment offsite.

12.11.4 Uranium Processing Building

Building 881, which started operation in 1953, manufactured HEU (93% U-235) components coupled with a chemical recovery and metal recycle capability. Manufacturing HEU ceased in 1964, but chemical recycle and other chemical functions continued into the early 1970s. The HEPA filters were sent

offsite also. No records were located indicating the amount of HEU shipped offsite with these filters in the 1950s and 1960s.

Building 444, which started operation in 1953, manufactured DU components and, later, beryllium components. The HEPA filters from Building 444 and later from Building 447 were sent offsite also. The amount of DU contamination for the HEPA filters was not recorded.

Building 883, which started operation in 1957, rolled and formed DU and HEU. The HEPA filters from Building 883 were sent offsite also. The amounts of DU and HEU on the filters from Building 883 were not recorded but treated as low-specific activity waste for offsite shipment.

The CWS filters in the HEU area located in Building 881 became loaded with ammonium nitrate salt. The formation of ammonium nitrate salt on the plenum filters was attributed to using nitrate acid and ammonia (gas) in HEU chemical recovery processes. To extend the life of these filters, the plenum filters were subjected to steam humidification to dissolve ammonium nitrate salt, thereby permitting air passage (see Appendix HH). Consequently, traces of ammonium nitrate may be present on CWS filters from Building 881.

12.11.5 High-Efficiency Particulate Air Filter Plutonium Measurements

Before 1964, the plutonium on the HEPA filters was not measured and was not taken as a measured discard. Zodtner and Rogers (1964) initiated an effort to measure the plutonium on discarded HEPA filters and to develop recovery procedures for filters highly loaded with plutonium. Economic discard limits were developed in the late 1960s for HEPA filters. As indicated in Table 14, the economic discard limit for glove-box filters was 3 g per filter, and the economic discard limit for the larger plenum filters was 24 g per filter. The discard limits for filter media in 1992 are listed in Table 24.

Table 24. Plutonium discard limits for filter media in 1992.

Filter Media	Discard Limits (g/g)
Item Description Code 338 high-efficiency particulate air filter	0.007380
Item Description Code 335 absolute drybox filter	0.006000
Interstate Commerce Commission 376 processed filter	0.007380

The initial economic discard limits were probably in the range of the 1992 limits.

A program to develop NDA methods for drummed and boxed waste was initiated in 1963. The first production drum counter was activated in 1969 followed by several upgrades in the early 1970s; however, NDA methods were developed in the mid-1960s and employed on a pilot scale for verification studies.

A plutonium recovery procedure was developed for processing “wet” HEPA filters and “dry” HEPA filters exceeding the economic discard limit. The procedure was implemented in the mid-1960s. The filter was removed from its frame, leached with dilute nitric acid, washed with water, and subsequently dried. The wooden frames were shipped as low-specific activity waste or burned in the Building 771 incinerator if highly contaminated with plutonium.

12.12 High-Efficiency Particulate Air Filter Management

Buildings at RFP that handled radioactive materials were constructed with appropriate intake and exhaust air filtration ventilation systems. The air exhaust systems were the main concern as these systems provided the environmental protection against the release of radioactive particulates to the surrounding atmosphere. The spent exhaust filters were shipped to INL from 1954 through 1970 and designated as Type III CWS filter waste.

An internal report by Zodtner and Rogers (1964) has raised concern at INL over the large amounts of plutonium arbitrarily assigned to exhaust filters from the plutonium processing buildings. At that time, the two buildings of concern were Buildings 771 and 776, which processed and handled large quantities of plutonium-bearing materials. Consequently, this section will omit discussing the uranium processing buildings and concentrate mainly on Buildings 771 and 776. The other buildings, which came in contact with plutonium, did not have accountable amounts of plutonium on their exhaust filters.

12.12.1 Plutonium Exhaust Ventilation System

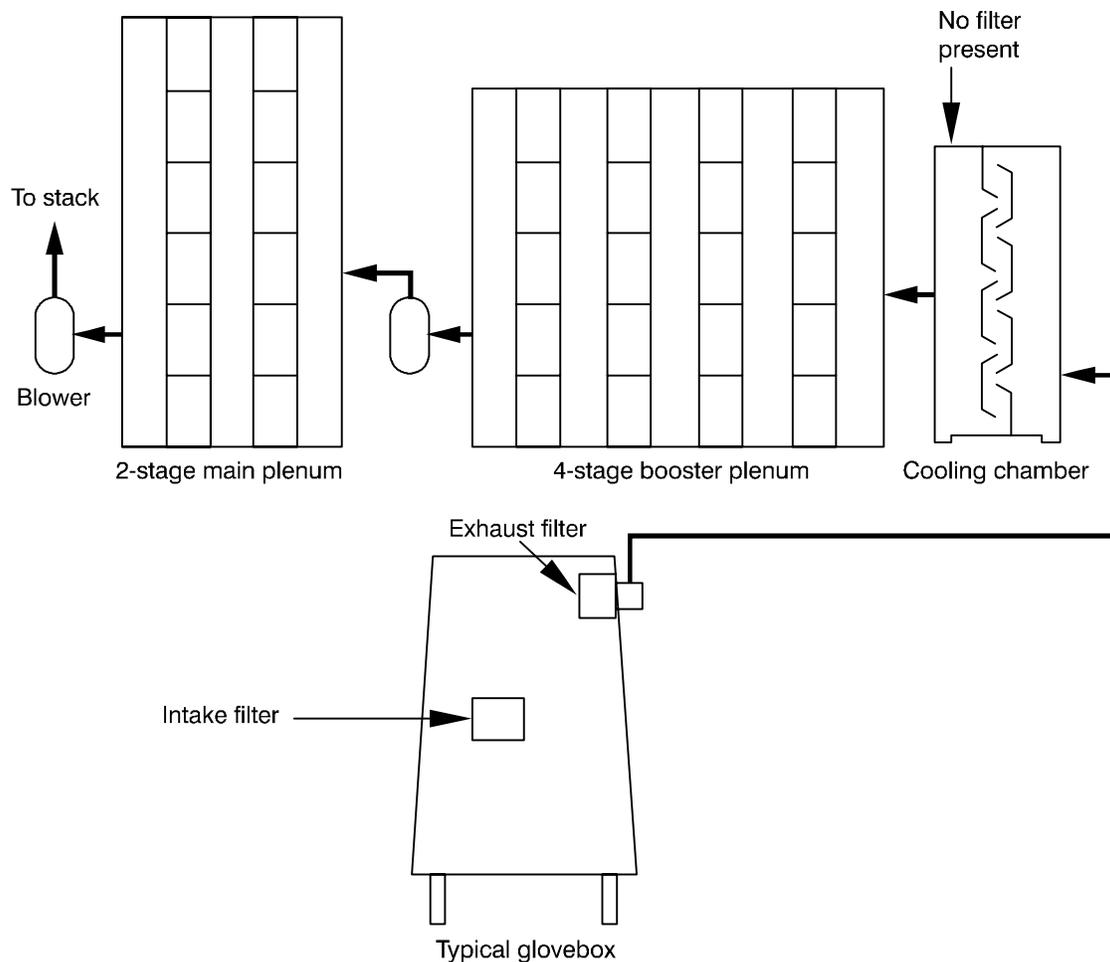
The exhaust ventilation system used in the plutonium buildings consisted of three filtration components based on the filters employed: (1) glove-box filters, (2) booster plenum filters, and (3) main building plenum filters. The air entered the glove box through an intake filter and exhausted to the booster plenum through a glove-box exhaust filter. The glove-box exhaust air transferred through the booster plenum to the main building plenum. The air passed through the main plenum and exhausted to the building stack.

The booster plenums contained two or four filter stages, while the main plenum originally had two filter stages. All booster plenums and main building plenums after 1970 were upgraded to four stages of filters coupled with automatic sprinkling systems for fire suppression. A simplified diagram of the three components and the exhaust air routing is shown in Figure 7. Not shown in Figure 7 is the exhausting of room air to the main building plenum.

The production operations in Buildings 776 and 777 were essentially dry processes except for the degreasing solvents and lathe coolants, which were organic-based liquids. The recovery and recycle processes in Building 771 were mainly aqueous processes employing large quantities of nitric acid. Consequently, the glove-box atmospheres were much more corrosive in Building 771 than in Buildings 776 and 777, which adversely affected the HEPA filters in Building 771 compared to Buildings 776 and 777.

12.12.2 Filter Background

The filters used for air filtration exhaust systems varied as necessary to improve particulate removal from exhausted air to the surrounding environment. The initial CWS filter employed impregnated paper for the filter media. The 1957 fire in Building 771 burned the paper CWS filters, which called for a fire resistant filter media. Consequently, a glass-asbestos filter was introduced in the 1959–1960 timeframe. This timeframe is somewhat substantiated by the low number of filters shipped in 1960 and the high number of filters shipped in 1959 as reported in Table 25. This shipping event indicates the possibility of a filter changeout.



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Figure 7. Plutonium ventilation system.

The carcinogenic nature of asbestos required the development of filter media that were fire resistant and eliminated the carcinogenic factor. Consequently, R&D efforts to coordinate with filter manufacturers were accelerated in the 1970s and into the 1980s. The RFP historical records provide a significant amount of documentation on studies of HEPA filter media, filter constructing, filter testing, and associated waste reduction efforts. However, this copious documentation was not directly relevant to the 1954–1970 period of interest. The present assumption is that the paper CWS filters were used up to about 1959 and were replaced with the CWS glass-asbestos filters through 1970 and beyond.

12.12.3 Filter Generation Rates

The waste identification system employed during the 1954–1970 period did not record ventilation filter generation rates by building or function. The designation used was Type III CWS filters. However, a summary report by the Waste Disposal Coordination Group listed the number of filters shipped to INL for the 1954-1970 period by year (see Table 25). Table content was obtained from monthly and annual reports of waste shipping history prepared by the Waste Disposal Coordination Group. Anderson's internal report (see Appendix II) lists the backlog of CWS filters as 463 as of June 1966 and also places the generation rate at approximately 20 filters per month.

Table 25. Summary of filters shipped to Idaho National Laboratory taken from reports prepared by the Waste Disposal Coordination Group.

Calendar Year	Number of Filters	Number of Cartons
1954	0	0
1955	1,205	0
1956	2	0
1957	1,251	101
1958	1,042	123
1959	1,679	0
1960	130	34
1961	1,592	0
1962	549	7
1963	535	0
1964	1,023	0
1965	762	0
1966	575	10
1967	990	948
1968	323	4,267
1969	209	249
1970	641	43
Total	2,508 ^a	5,782 ^b

a. Includes 24 × 24 × 14-in., 24 × 24 × 16-in., 24 × 24 × 18-in., 24 × 24 × 28-in., and 28 × 28 × 16-in. cartons of filters

b. Includes 5,496 cartons containing 55-gal drums.

12.12.4 Filter Configuration

Historical RFP documentation addressing the configuration, construction, and materials of construction was limited at best; however, Anderson (see Appendix II) includes a description of the backlog CWS filters. The CWS Type F filter was manufactured by Flanders and Cambridge; filter dimensions were 24 × 24 × 12 in. with a weight of about 22.7 kg (50 lb). Eventually, the CWS filters were replaced with HEPA filters manufactured to AEC standards.

The notes on the bottom of Table 25 list the sizes of the cartons used to ship filters to INL. From the sizes listed, the 24 × 24 in. appears constant, while the depth varies as indicated by 14 in., 16 in., 18 in., and 28 in. An exception to the 24 × 24-in. configuration is the 28 × 28 × 16-in. carton. A reasonable assumption for the 28 × 28 × 16-in. carton and the 28-in.-deep carton is to double the number of filters per carton.

12.12.5 Filter Media

From plant startup to about 1960, the CWS filters had a cellulose base for the filter media, which were combustible. The 1957 fire in Building 771 demonstrated that the cellulose-based filter media were

unacceptable because of this combustible characteristic. Consequently, a search and development program began to provide noncombustible filter media with acceptable filtering capability.

A CWS Type F filter manufactured by Flanders and Cambridge replaced the combustible CWS filter in the 1959–1960 timeframe. The materials of construction for the CWS Type F filter are listed below as reported by Anderson (see Appendix II):

- Size: 24 × 24 × 11.5 in.
- Weight: 22.7 kg (50 lb)
- Filter media: glass-asbestos
- Separator: aluminum or asbestos
- Frame: wood or cadmium-plated steel.

As stated previously, the carcinogenic nature of asbestos demanded that the Type F CWS glass-asbestos filters be phased out. Post-1970 documentation indicates that the glass-asbestos filter was employed well into the 1970s. Consequently, the filters shipped to INL from 1954 through 1960 were the cellulose-based CWS filters, followed by the glass-asbestos filters.

12.12.6 Filter Discard Limits

The plutonium material unaccounted for that became apparent in the late 1950s and early 1960s was the driving force to establish a system to determine what should be declared waste and what should be processed for plutonium recovery. A system was developed based on comparing the cost of producing a gram of reactor plutonium to the cost of recovering a gram of plutonium from residue waste. The break-even point was established as the economic discard limit for a given plutonium residue (see Section 6.6).

To establish economic discard limits for the large number of generated waste residues required developing NDA procedures for plutonium assays, instituting an appropriate cost accounting system, and staff training. The first NDA drum counter was developed in 1964, which provided estimates for plutonium content in waste residue. The first production drum counter was placed into use in 1969. The economic discard limit program became effective in the late 1960s.

For most of the plutonium-bearing waste residue, the economic discard limits were established on a gram of plutonium per gram of the matrix material. Filters were the exception, as the economic discard limit was set per filter rather than a weight-to-weight ratio. The discard limit for filters is reported by Anderson, Putzier, and Ziegler (see Appendix I); limits shown in Table 26 are for 1968.

Table 26. Filter economic discard limits for 1968.

Filter Media	Dimensions	Discard Limit
Chemical Warfare Service filter	2 × 2 × 1 ft	24 g/filter
Drybox filter	8 × 8 × 4 in.	3 g/filter

12.12.7 Plutonium Filter Content

The pre-1970 historical documentation does not address the levels of plutonium contamination on spent glove-box and plenum filters other than the estimates provided by Zodtner and Rogers (1964). Terada, Woodard, and Jensen (1985) provide plutonium levels for plenum filters. For 30 filters located in the FU-2B booster plenum in Building 771, the range of plutonium content varied from 6 g per filter to a high of 46 g per filter.

Anderson (see Appendix II) estimates that the backlog of 463 filters contains 34.6 kg (76.3 lb) of plutonium. The average plutonium content per filter would be 75 g. An additional estimate for generation placed at 20 filters per month was 300–1,000 g of plutonium. The average plutonium content per filter would range between 15–50 g of plutonium. The accumulated backlog of filters cited by Anderson indicates that highly contaminated filters were not being shipped in the mid-1960s.

12.12.8 Plutonium Recovery from Spent Filters

The plutonium recovery process described by Anderson (see Appendix II) includes a water rinse to remove any residual nitric acid. Briefly, the filter media were removed from the wood or metal frame and leached with concentrated nitric acid. Calcium fluoride was added to the leach solution as a source of fluoride ion to accelerate the dissolution of plutonium oxide. The leach solution was transferred to anion exchange processing to purify and concentrate the plutonium for further purification. The leached filter media were packaged for shipment into 55-gal drums with magnesia cement added for liquid absorption. The wood and metal frames were considered low-specific activity waste in the terminology of the time. In the 1970s, filter media processing became more sophisticated as evidenced by James (1980).

12.12.9 Filter Processing

The HEPA filters from process glove boxes and filter plenums comprise another category of residue, which required preprocessing before plutonium recovery. These filters were processed in Line 48 using saws, hammers, and screwdrivers to separate the wooden frames from the filter media. The wood was normally discardable; the filter media, if above discard, could be transferred to Line 21B for further processing. In Line 21B, the filter media were first contacted in a closed reactor vessel with anhydrous hydrofluoric acid that removed silica from the matrix. The plutonium-bearing matrix was then immersed in concentrated nitric acid in a batch dissolver. The resulting plutonium nitrate solution was filtered and transferred by vacuum to ion exchange. The remaining solids were reprocessed, if above the discard limit, or shipped to waste disposal if below the discard limit.

Glove-box filters were removed from their holding frame and tapped inside a stainless steel tray to remove any loose particulate matter. Based on visual examination and processing history for a given glove box, the operator or foreman made a decision to process or ship as waste. This procedure continued until NDA systems became available in the late 1960s. The history of the glove-box processing was important as large quantities of dust were generated in grinding, scraping, and hammer milling. In other glove boxes, the operations performed were dustless and dry, which produced very little plutonium contamination on the glove-box exhaust filter.

Certain booster plenums served processes that generated acid fumes and dusts, which contaminated the booster filters. Consequently, these booster filters were changed more frequently and were more likely to be processed for plutonium recovery.

12.12.10 Waste Filter Packaging

Spent plenum filters, which were considered waste, were packaged into cartons for shipment to INL. Those plenum filters considered to have recoverable amounts of plutonium were sent to plutonium recovery for processing. This visual examination process continued until NDA systems were developed in the mid-1960s to assay the spent filters for plutonium content.

Glove-box filters were removed from the glove-box line through a bag-out procedure. Often a double-bag procedure was followed. Bagged filters that were considered recoverable were transferred to the leaching glove-box line. filters were placed in a 55-gal drum lined with a poly liner. The operator or the foreman estimated the plutonium content. This estimate was included in the normal operating loss for a given material balance area account.

About 1967 or 1968, economic discard limits were established, which provided a nondiscretionary procedure for determining whether a given filter was waste or recoverable.

The waste-generating streams in Building 776 can be segregated into five categories, which are listed below:

- General building waste
- Graphite waste
- Washable waste
- Miscellaneous waste
- Boxed waste.

The general building waste was similar to that for Building 771 and was considered to be slightly contaminated with insignificant amounts of plutonium.

The graphite waste was generated by the plutonium foundry through casting procedures. The graphite molds were probably the largest contributor to the plutonium sent to INL in the early years. Building 776 washable waste consisted of rubber-and plastic-based materials. These materials were washed before being sent offsite. The miscellaneous waste generated in Building 776 was cleaned superficially before being placed into 55-gal drums for shipment offsite.

12.13 Waste Drums Returned to Rocky Flats Plant

Drum surveillance at INL identified drums with excessive plutonium levels and other problems. On several occasions, these selected drums were returned to RFP for interrogation and content inspection. Results were reported by RFP through internal reports. The inspection results varied widely and depended on each drum. Consequently, the inspection results are not addressed here, but the inspection reports are provided in Appendix JJ.

12.14 Mound Disposal

In the early 1950s, drums of DU, HEU, and low-level, plutonium-contaminated waste were buried in an area called Mound. The initial burial started in April 1954 and continued until September 1958. A total of 1,045 drums of oil and solid waste was buried. The majority of the radioactive contamination was DU with some HEU and possible low-level plutonium. The drums were exhumed with the oil drums transferred to the 930 pad. Completed retrieval and offsite disposal were accomplished by May 1970. Shipment to INL was noted on several trailer load lists.

12.15 Reactor Grade Plutonium

Rocky Flats Plant did not process any irradiated reactor fuel material. However, in the late 1960s, RFP fabricated fuel elements for the Zero Power Physics Reactor at the Materials and Fuels Complex (formerly Argonne National Laboratory-West). The fuel element alloy was composed of DU, plutonium, and molybdenum, with DU being the major component. The plutonium material had a Pu-240 content in the 8–10% range.

The residues and waste generated were segregated from War Reserve production and were shipped to Hanford. No packaged residue or waste was shipped to INL. However, trace amounts of Zero Power Physics Reactor material were probably commingled in the organic liquid waste and the aqueous waste sent to Building 774.

12.16 Cyanide Waste

Cyanide salts were used in Building 444 in heat-treating baths. Disposal of these spent baths was a waste disposal problem as indicated by Ryan (see Appendix KK):

The disposal of cyanide wastes which are produced in Building 44 are a potential problem. In the past, these wastes were set-up with Portland cement in Building 44. At the present time, these wastes are being sent to Building 81 for destruction. The presence of fluoride and Building 44 material in the waste makes this method undesirable for Building 81.

Because of potential safety concerns and disposal problems, cyanide heat-treating baths were eliminated in favor of carbonate-based baths.

12.17 Glove-Box Gloves

The plutonium-handling buildings at RFP used significant quantities of glove-box gloves coupled with an inspection program to detect failing gloves. Consequently, defective gloves contaminated with plutonium were shipped to INL. Giebel and Riegel (1971) provide specifications for glove-box gloves for procurement purposes. The report identifies glove materials, design parameters, and requirements.

12.18 Liquid Organic Waste

Manufacturing operations in the uranium and plutonium areas used a variety of organic liquids that became a salvage and disposal problem in the 1950s and early 1960s. Installation of a solidification process in the middle 1960s in Building 774 (known as the Grease Plant) produced disposable sludge from these organic liquids.

These organic liquids were used for machine lubricants, coolants, cutting oils, vacuum and diffusion pump oils, hydraulic oils, and a variety of degreasing and cleaning solvents. Monobromo benzene was initially used in determining the density of plutonium components. To eliminate potential health and flammability hazards associated with monobromo benzene, Freon 113 was substituted.

In the late 1950s and early 1960s, aqueous-based coolants and lubricants were substituted for organic-based oils in the uranium areas. This substitution reduced the potential fire hazard associated with pyrophoric uranium turnings and organic oils.

A variety of chlorinated hydrocarbon solvents were used based on their toxicity and their compatibility with product and equipment. The most significant solvent and lathe coolant diluent was CCl₄. Significant data from Hobbs (1982) regarding usage of CCl₄ at RFP are listed below:

- Review of purchasing data indicated about 60,566.6 L (16,000 gal) per year of CCl₄ used in the 707 plutonium manufacturing facility
- A total of 41,639.5 L (11,000 gal) of CCl₄-oil mixture sent to waste in Fiscal Year 1981
- Waste stream estimated to contain an average of 70 vol% CCl₄
- Estimated about 136.1 kg (300 lb)/day CCl₄ lost to evaporation during usage.

Hawes (see Appendix LL) reported the usage of 52,995.8 L (14,000 gal) of CCl₄ over a 12-month period. These two reports indicate a very large usage of CCl₄ per year of which about 50–70% ended up in the Grease Plant Series 743 sludge.

The total purchase of TCE for Buildings 707 and 777 was 9,274.3 L (2,450 gal) in 1989. The amount of Freon 113 used in Building 707 ranged from 3,028.3 to 3,406.9 L (800 to 900 gal) per year. Building 777 switched from using isopropyl alcohol to TCE for cleaning activities in the 1963–1964 timeframe. At this time, PCE was replaced with TCE.

In 1958, a cutting oil was used in the plutonium machining operations to facilitate machining and to reduce spontaneous combustion of the plutonium turnings. Shell Vitera cutting oil was initially used followed by a PCE washing. Later, PCE was replaced with CCl₄ as PCE attacked the glove-box gloves. Later, Shell Vitera was replaced by Texaco Regal Oil, which costs less. This replacement was in the 1970–1972 timeframe. In general, the oils used in the plutonium metal working areas were paraffin-based mineral oils with ~0.5 wt% of an antioxidant additive.

As stated previously, accumulation of this organic liquid waste was a significant disposal problem as indicated by volumes shown in Table 27 for a period in 1962. The majority of waste organic liquids was contaminated with DU, HEU, or plutonium.

Table 27. Organic waste.

Waste	Source (Building No.)	Disposition	Rate of Accumulation ^a (gal/month)
Machine coolant (Shell Vitera-CCl ₄)	776	200,000 gal in storage	4,500
Distillation still bottom	444	2,000 gal in storage (some buried in open pit)	200
Distillation still bottom (chlorinated)	444	3,500 gal in storage	50
Machine coolant	881	Buried in open pit	500
Chlorinated solvents	881	3,500 gal in storage	50
Cold oils	Miscellaneous	Buried in open pit	1,000
Trichloroethylene	777	400 gal in storage	90
Trichloroethylene	991	Will be stored	180
Miscellaneous organics	777, 771, 444	Accumulated	70

^a Rates for May, June, and July 1962.

12.19 Sludge

Sludge inevitably formed wherever liquids were used in operations and facility systems. Sludge in general can be characterized as follows:

- Organic-based
- Acidic-based
- Caustic-based
- Organic-aqueous-based.

Sludge formed in process equipment such as dissolvers, leaching vessels, degreasing vats, storage tanks, pumps, distillation bottoms, gear boxes, hydraulic presses, quenching tanks, and heat-treating baths.

12.19.1 Organic-Based Sludge

Typical organic sludge types and their origin are listed below:

- Machining coolant
- Cutting oils
- Vacuum pump oils
- Degreasing solvents

- Hydraulic fluids
- Lubricating greases and oils.

The radioactive contamination in this sludge was particulate matter. In general, contamination was low and below economic discard limits.

The disposition of organic sludge depended on its viscosity, consistency, and the availability of disposal avenues. Based on the discretion of operating personnel and waste management concurrence, organic sludge could be shipped to INL, disposed of on plant site, or stored. The installation of the Grease Plant in 1966 ultimately became the disposal route for much of the stored organic sludge. On the load lists, non-Building-774 sludge was often identified as Type IV sludge with or without the Series 743 designation.

12.19.2 Acidic-Based Sludge

Acidic-based sludge was formed in process equipment such as dissolvers, evaporators, pumps, and tanks. This sludge was characterized by its acidity and plutonium content as indicated below:

- High nitric acid with high plutonium content
- High nitric acid with low plutonium content
- Low nitric acid with low plutonium content
- Low hydrochloric acid with low plutonium content and chloride salts present.

The dissolution of this sludge produced solutions that were either sent to plutonium recovery or transferred to Building 774 based on the plutonium concentration. If sent to Building 774, the sludge solutions would become Series 741 and 742 sludge for shipment to INL or would be deposited in the solar evaporation ponds.

12.19.3 Caustic-Based Sludge

Caustic-based sludge was characterized as indicated below:

- High caustic (NaOH/KOH) with low plutonium
- Low caustic (NaOH/KOH) with low plutonium.

This sludge was usually discardable as the plutonium content was very low and in the form of particulates. The sludge was dissolved by the addition of water or low nitric acid, filtered, and transferred to Building 774 for final treatment.

12.19.4 Organic-Aqueous-Based Sludge

The organic aqueous-based sludge formed mainly in the uranium-plutonium recovery process, which employed a TBP-dodecane extractant. This sludge was handled by the Special Recovery Group to limit the introduction of organic material to the Building 774 aqueous stream and to control the quantity of aqueous solutions to the Grease Plant.

12.20 Mercury Waste

Mercury was used at RFP in diffusion vacuum pumps, instrumentation, and analytical laboratory procedures. The analytical laboratory recycled its mercury for reuse using a triple distillation procedure. The distillation bottoms were bottled and transferred to Building 774 for disposal. Other sources of spent mercury were transferred also to Building 774. The bottled mercury was discarded in a drum of solidified aqueous sludge based on operator discretion.

12.21 Excess Chemical Compounds

Excess chemical compounds accumulated and required a disposal route. Unopened containers were provided to local universities, colleges, schools, and other governmental agencies. Excess noncontaminated chemical compounds that were water soluble were added to the solar evaporation ponds next to Building 774, provided the compounds were compatible with solar pond constituents. These chemical compounds ended up in the Series 745 evaporation salts processed through Building 774 and sent to INL.

Chemical compounds not acceptable in the solar ponds were treated by the generator to fit a waste stream or sent to Building 774 for disposal. Building 774 accommodated these chemical compounds through the Series 744 sludge process or spoon feed to an acceptable waste treatment stream that produced Series 741, 742, and 743 sludge. The disposition route was governed mainly by the quantity received for disposal.

12.22 Polychlorinated Biphenyls

Polychlorinated biphenyls were used throughout RFP in electrical transformers, capacitors, hydraulic presses, and vacuum diffusion pumps. During the 1954–1970 timeframe, the polychlorinated biphenyls shipped to INL probably came through combustible waste. Leaks from hydraulic and other equipment were taken up using rags and absorbent wipes, which were sent to INL if generated in DU, HEU, and plutonium areas. Polychlorinated biphenyls were phased out in the 1970s.

12.23 Complexing Agents

Complexing agents were used by the analytical laboratories and as a constituent in decontamination solutions. The amount of complexing agents used by the analytical laboratories was minor when compared to the quantity used in decontamination efforts.

The analytical lab solutions with complexing agents were collected and treated for disposal by the generating laboratories or by chemical recovery in Building 771. If neither the laboratories nor chemical recovery could dispose of complex aqueous waste, the complexing waste was bottled and sent to Building 774. These solutions were treated by cementation techniques for disposal at INL.

The contaminated decontamination solutions were collected and transferred to Building 774. These solutions were cemented for disposal at INL. At the discretion of Building 774 operators, bottles of hot decontamination solutions would be included with Series 742 and 744 sludge.

12.24 Analytical Methods

The INL personnel have been concerned about the accuracy of plutonium and Am-241 determinations in waste sent to INL before 1971.

The RFP HEU and plutonium analytical laboratories were participants in the AEC Sample Exchange Program. Consequently, RFP analytical results were monitored by AEC for accuracy within the sample exchange results; however, these analyses were on products such as metal oxide and rich nitrate solutions. No waste samples were involved in the AEC Sample Exchange Program.

The problem with solid waste analysis was obtaining a representative sample for analysis because the waste was not homogeneous. In the 1950s and somewhat in the early 1960s, estimates for plutonium in waste items were based on a by-difference approach coupled with operating experience. The increasing plutonium material unaccounted for was a primary concern, which led to the development of NDA methods and the installation of low-level plutonium recovery processing.

As stated previously, the first NDA drum counter was an experimental model placed in service by R&D in 1964. Drum verification studies were carried out by R&D for graphite waste (see Appendix V).

The demand for NDA standards enlarged the chemical standards group and their scope. Doher and McBride (see Appendix MM) cite biases of 1–20% and variabilities ranging from 4 to 37%.

Lawless and Chanda (1970) provide an evaluation of a helix counter designed for assaying graphite and ash residues for plutonium content. Biases and precision results are reported for certified standards of graphite and ash matrices (see Appendix NN).

13. TIMELINE AND CORRESPONDENCE—WASTE MANAGEMENT

The timeline of interest for this report covers primarily 1954 to 1970. Consequently, the timeline presented below also covers primarily that period of interest.

Correspondence from Ed Vejvoda to Operable Unit 7-13/14 staff is listed below in Section 13.2.

13.1 Timeline

Date	Event
1952	INL—Original NRTS landfill, now known as the Subsurface Disposal Area, is established.
July 1952	INL—First trench opens for disposal of solid waste.
1952–1957	INL—Trenches 1 through 10 are excavated to basalt; average 1.8 m (6 ft) wide, 274.3 m (900 ft) long, and 3.7 m (13 ft) deep.
1953	INL—U.S. Atomic Energy Commission Idaho Operations Office becomes responsible for the operation of the burial grounds.
1953	RFP—Operations begin in Buildings 444, 771, 774, 881, and 991.
1953	RFP—Waste disposal coordination group forms with E. Ryan as contact.
1953	RFP—Radioactive waste accumulates with storage becoming a problem.
1953–1967	RFP—Contaminated organic waste disposal develops into major plant issue.
April 1954	RFP—First shipment of waste to INL. Several drums leak liquids; RFP-INL establish policy of no liquid shipments.
1954–1957	INL—TRU ^a -contaminated waste from RFP, packaged in drums or wooden crates, is stacked horizontally in pits and trenches with NRTS mixed fission product waste. Therefore, Trenches 1 through 10 ^b and Pit 1 contain NRTS waste interspersed with TRU-contaminated waste. Records from RFP do not accompany these shipments. Instead, an annual summary of disposals provided total radionuclide content and waste volume.
1954–1965	INL—Informal forms are used, no form number or revisions noted.
March 1955	RFP—First shipment of Chemical Warfare Service filters.
1956	RFP—Building 447 constructed attached to Building 444. Depleted uranium chip roaster placed in service. Building 447 houses waste management services.
1957	INL—Size of Radioactive Waste Management Complex expands from 5.3 ha (13 acres) to 35.6 ha (88 acres).
September 1957	INL—TRU waste buried in Pit 1.
November 1957	INL—Volume of waste from RFP increases rapidly, including items too large and bulky for trenches; pit disposal begins for TRU waste.
1957	RFP—Beryllium operation initiated in Building 444.
1957–1958	RFP—Buildings 776 and 777 begin operations. Plutonium foundry and machining transferred from Building 771.
1957–1971	RFP—Offsite waste shipments to INL.
1957	RFP—Assembly operations in Building 991 are curtailed and transferred to Building 777.
1957	RFP—Building 774 is designated to collect plutonium-contaminated organic liquid waste.

Date	Event
1957	RFP—Am-241 recovery is initiated for shipment to Oak Ridge National Laboratory Isotope Pool.
1957	RFP—Building 883 is constructed for HEU and depleted uranium rolling and forming.
September 1957	RFP—Building 771 fire curtails operations. Fire waste shipped to INL.
1957–1969	RFP—Building 776 is major user of carbon tetrachloride, while Building 777 is major user of trichloroethylene.
1958	INL—Landfill expands to 36 ha (88 acres).
1958	RFP—Building 771 resumes operations.
October 1959	INL—Pit 2 open; drums stacked in rows.
1959	INL—Procedures to accept waste standardized, including completion of disposal form.
1959–1961	RFP—Chip roaster is inoperative because of relocation within Building 444.
1960	RFP—Tributyl phosphate solvent extraction plutonium recovery process is replaced by anion exchange process.
1960	RFP—Chemical-Warfare-Service-treated cellulose media are replaced by glass-asbestos media for fire safety.
1960–1962	RFP—Line item is approved to expand Building 771 recovery capability and capacity.
1960–1963	(1) INL—NRTS accepts approved shipments of solid radioactive waste from offsite generators and continues accepting from RFP after commercial sites opened in 1963. (2) INL—Trenches 16 through 25 and Pits 2 through 5 open for disposal of waste and receive some mixture of RFP TRU-contaminated waste, NRTS waste, and offsite waste that is stacked or dumped.
December 1961	INL—Pit 3 opens. TRU and non-TRU waste is buried intermixed. Waste is stacked in rows.
1962–1965	RFP—Start of HEU cleanout in Buildings 881 and 883.
February 1962	INL—Pit 2 floods and disposal operations are moved to Pit 3 until September 1962.
July 1962	INL—Pit 3 waste is no longer stacked in rows; it's dumped at random until pit closure in January 1963.
1962	RFP—Beryllium sheet rolling begins in Building 883.
1962–1975	RFP—Beryllium wrought process implemented to recycle beryllium scrap.
January 1963	INL—Pit 4 opens for mixed low-level waste and TRU waste for 1.5 years and then is used for TRU waste only.
November 1963–1969	INL—Drums from RFP were dumped into pits instead of stacking.
1963–1964	RFP—Building 777 switches from isopropyl alcohol to trichloroethylene for cleaning parts.
1963–1964	RFP—Expansion of Building 771 chemical recovery facilities is completed.
January 1964	INL—In Pit 4, TRU drums are stacked in rows, and boxes are stacked along pit sides.
December 1964–July 1966	INL—Pit 4 closes; it reopens until final closure in September 1967.
1964–1969	INL—Environmental monitoring program at Subsurface Disposal Area is revised: 18 thermoluminescent dosimeters replace 35 perimeter film badges, collection and analysis of water samples from subsurface monitoring holes, and field investigations assess leaching.

Date	Event
1964–1970	(1) INL—Modifications to trenches: increase minimum depth to 1.5 m (5 ft), line bottom of excavations with at least 0.6 m (2 ft) of soil underburden, compact waste by dropping heavy steel plate on dumped waste in trenches, and increase soil cover over each disposal area to 0.9 m (3 ft). (2) INL—Trenches 33 through 49 are active.
1964	INL—In Pit 4, random dumping of waste begins.
1964	RFP—First research and development experimental nondestructive assay drum counter is established for waste assay.
1964–1966	RFP—HEU component manufacturing terminates and relocates to Y-12 plant at Oak Ridge National Laboratory.
January 1964	RFP—Zodtner and Rodgers (1964) report issued addressing plutonium material unaccounted for and possible understatement of plutonium in waste sent to INL.
February 1965	INL—Pit 5 opens for TRU waste only, apparently random placement of waste.
1965	RFP—Steam evaporator is installed in Building 774 to reduce liquid volumes stored in solar evaporation ponds. Produced Series 745 evaporator salts.
1965–1988	RFP—Initiation of Np-237 tracer program.
1966	INL—Pit 4 reopens; waste Form ID-110A first used.
1966	RFP—Series 742 and 744 sludge from Building 774 begins using 17C drums instead of 17H drums to take advantage of extra weight permitted.
May 1967	INL—Pit 6 opens for TRU waste only; boxes and drums generally segregated.
1967	INL—Pit 4 final closure.
1967	RFP—Molten salt extraction process for Am-241 removal from returned pits established in Building 776.
1967	RFP—Expansion of plutonium analysis laboratory by construction of Building 559.
1967	RFP—Concept of economic discard limits initiated for HEU and plutonium-bearing materials.
1967	RFP—Start using cardboard cartons for shipping high-efficiency particulate air filters.
1967	RFP—U.S. Department of Transportation shipping regulations require using specification 17H and 17C drums for shipping radioactive waste. Additional use of stronger drum liners and covers initiated.
1967	RFP—Grease Plant installed in Building 774 to process organic liquids in storage. Operations begins to process contaminated organic liquid stored on the 903 Pad.
1968	INL—Transition from waste Form ID-110-A to Form ID-125.
May 1968	INL—Pit 9 opens; drums are dumped.
August 1968	INL—Pit 10 opens; containers are dumped; fire waste, drums, and boxes are not segregated.
1968	RFP—Processing of contaminated organic liquids stored on 903 Pad completed with final shipment to INL.
1969	INL—Pits 9 and 10 flood.
1969	RFP—First production drum counter installed in Building 771.
1969	RFP—Approval to employ ATMX-600 railcars to haul waste to INL.

Date	Event
May 1969	RFP—Disastrous fire in Building 776 terminates operations in Buildings 776 and 777.
1969–1971	RFP—Cleanup of 1969 fire damage is completed with significant quantities of waste sent to INL over this period.
1969	RFP—Waste operations start in Building 776 and continue until closure.
January 1970	INL—In Pit 10, last drums are dumped; remainder of pit (about last 350 ft of the east end) is filled with boxes.
1970	(1) U.S. Atomic Energy Commission—New policy requires solid TRU waste to be segregated and stored retrievably. (2) INL—Burial of waste classified as TRU discontinues; TRU waste transferred to Transuranic Storage Area for retrievable storage.
April 1970	INL—Pit 11 ^c opens; drums are stacked in rows, and boxes are stacked along south wall of pit.
July 1970	INL—Pit 12 opens for TRU waste that was stacked.
October 1970	INL—Pit 11 closes.
November 1970	INL—All TRU waste is placed in aboveground, retrievable storage.
1970	RFP—Building 707 begins plutonium operations.
1970	RFP—Concept of TRU (retrievable) and low-level waste (nonretrievable) designations issued by U.S. Department of Energy.
1971	INL—Transition from waste Form ID-125 to Form ID-135.
1971	RFP—Drum counting facility constructed between Buildings 771 and 774 and designated as 771C.
1971	INL—Waste Management, as an organization, is formed and takes responsibility from the U.S. Atomic Energy Commission for disposal of radioactive waste.

a. In 1954, TRU waste was defined as TRU radionuclides in concentrations greater than or equal to 10 nCi/g

b. Pits 7 and 8 did not receive TRU waste

c. Drums in Pits 11 and 12 were retrieved 1974 through 1978.

HEU = highly enriched uranium

INL = Idaho National Laboratory

NRTS = National Reactor Testing Station

RFP = Rocky Flats Plant

TRU = transuranic

13.2 Correspondence from Edward Vejvoda to Operable Unit 7-13/14 Staff

Date	Recipient	Subject
December 4, 2001	Marianne Little	Rocky Flats Waste Information
December 11, 2001	Marianne Little	Rocky Flats Reports
December 18, 2001	Wendell Jolly	Video Tape Review
July 13, 2000	Bruce Becker	Distribution of Reference Material
December 22, 2000	Rod Thomas	Inventory Difference Briefing (3/7/84)
February 12, 2001	Rod Thomas	NDA Reports
February 16, 2001	Rod Thomas	Additional NDA System Information (1966-1977)
August 16, 2001	Bruce Becker	Your FAX July 23 2001
August 29, 2001	Bruce Becker	Plutonium Estimates for Rocky Flats Waste Forms
September 26, 2001	Bruce Becker	Nuclear Safety - Waste Management
January 28, 2002	Marianne Little	1964 Rocky Flats Drum Counter
February 12, 2002	K. Jean Holdren	EPA Region 10 Requests
March 4, 2002	Wendell Jolly	Videotape Cassette Review
April 29, 2002	Marianne Little	Rocky Flats Waste Information
July 31, 2002	Marianne Little	Rocky Flats Waste Report
August 15, 2002	K. Jean Holdren	Request for Rocky Flats Reports by DOE-IDO
November 7, 2002	Paul Sentieri	Your E-mail of November 6 2002
March 6, 2003	K. Jean Holdren	DRAFT Consolidated Report of Rocky Flats Wastes Shipped to INEEL
April 29, 2003	Marianne Little	Rocky Flats Sewage Sludge
June 3, 2003	Marianne Little	Rocky Flats Reports
June 13, 2003	Marianne Little	Returned RFP Waste from INEEL 1971
August 11, 2003	K. Jean Holdren	Distribution of Reference Material
August 25, 2003	K. Jean Holdren	Distribution of Reference Material
September 24, 2003	Marianne Little	Rocky Flats Shipments
September 30, 2003	K. Jean Holdren	AEC Courier Receipts - 1964
March 22, 2004	K. Jean Holdren	Photographs of Retrieved RFETS Waste
March 30, 2004	K. Jean Holdren	Draft Copy of Graphite Mold Report
March 30, 2004	K. Jean Holdren	Roaster Oxide Information
April 29, 2004	K. Jean Holdren	Draft Copy of HEPA Filter Report
May 20, 2004	K. Jean Holdren	Ingot Mold Drawing
June 29, 2004	K. Jean Holdren	Rocky Flats Waste Reports
July 26, 2004	K. Jean Holdren	Rocky Flats HEPA and Graphite Processing Reports
October 25, 2004	K. Jean Holdren	Transshipped Waste

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- 42 USC § 9601 et seq., 1980, “Comprehensive Environmental Response, Compensation and Liability Act of 1980 (CERCLA/Superfund),” United States Code.
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- 49 CFR 175, 2005, “Carriage by Aircraft,” *Code of Federal Regulations*, Office of the Federal Register.
- 49 CFR 176, 2005, “Carriage by Vessel,” *Code of Federal Regulations*, Office of the Federal Register.
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- 49 CFR 178, 2005, “Specifications for Packagings,” *Code of Federal Regulations*, Office of the Federal Register.
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Appendix A

Letter from G. V. Beard to John Epp

INTEROFFICE CORRESPONDENCE

date January 22, 1980
to J. D. McKinney
from T. L. Clements, Jr. *T. L. Clements, Jr.*
subject NON-RADIOLOGICAL HAZARDS STUDY - TLC-2-80

TYPE: Visitation - Colorado School of Mines Research Institute

File No.: RWMC-17-80

SUMMARY

A visit to the Colorado School of Mines Research Institute (CSMRI) in Golden, Colorado was made on January 15, 16, 1980. The purpose of the visit was to review available waste shipment records and interview knowledgeable personnel about waste shipments and research projects conducted during 1960-1962. Radioactive waste disposal records at the CSMRI indicate four (4) waste shipments were made to the INEL during the period of December, 1960 to October, 1962. This concurs with available INEL waste shipment disposal requests. Additional information located at the CSMRI indicates plutonium contaminated waste from a classified project was sent to the Rocky Flats Plant. Ultimate disposal of this material would have taken place at the INEL.

Available INEL Waste shipment records indicate only the October, 1962 shipment was buried in pits or trenches that may be involved in future retrieval projects. The disposal request (ID-137) identified a CSMRI purchase order. The purchase order was recovered in their archives. The purchase order identified two project numbers that absorbed the cost for disposal services. Both project files were recovered from the CSMRI archives.

Project 320311 was conducted for American Metal Climax, Inc. of Denver, Colorado. The research involved the development of rapid analytical techniques for trace elements. The identification of trace elements in ore serves as a method of determining potential ore deposits. One set of silicate rock samples and molybdenite concentrates were sent to the Argonne National Reactor in Argonne, Illinois for neutron activation. These samples were then analyzed for trace elements by various techniques, such as scintillation counting and the use of separation techniques. Chemicals used in analytical procedures would have involved acids, such as nitric, sulfuric, and hydrochloric and organic solvents such as toluene, dimethyl POPOP (1,4 bis 2,5 phenyl oxazolyl benzene), p-terphenyls, and other scintillation solutes. Complete information concerning materials used in various isotope separations and solvent extractions were not available.

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date January 22, 1980
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Information provided by F. L. Smith, past Director of Research for the CSMRI, indicated small quantities (volume is unknown) of these materials may have entered the radioactive waste stream. Chemical analysis, such as flame photometry, of unirradiated samples utilized hydrofluoric and perchloric acids. Since the samples were not radioactive, the waste chemicals from the analysis probably did not enter the radioactive waste stream.

Project 300301 was conducted for the Atomic Energy Commission, Division of Isotopes Development, under Contract No. AT(29-2)-1355, signed April 27, 1962. The title of the project was "Radioisotopes in Process Control". One aspect of the project was concerned with evaluating large volume beta and gamma detection systems for process stream analysis and automation. Initial tests for determining optimizing conditions for liquid scintillation systems utilized Cs^{137} sources. This type source allowed for comparison between liquid and plastic scintillators. A one (1) curie Co^{60} source was used to determine the effects of radiation on liquid scintillators. Waste from these projects included the radioisotope sources and general laboratory waste (paper, glassware, etc.). Scintillation solutes, such as POPOP in toluene may have entered the radioactive waste stream. The volume of this material in the waste is believed to be small. A second aspect of the project evaluated various radioisotope tags in aqueous and organic fluids, and in slurries typical of those encountered in the mining, chemical, and metallurgical industries. The evaluations were conducted in a pipeline loop that was constructed on a pilot plant level for dynamic testing of the solutions and slurries. A 6" X 6" pipe sleeve insert was neutron activated (Fe^{59}) and placed in the pipeline loop. As a particular type slurry was pumped through the pipeline, it was monitored for Fe^{59} . The values derived from the experiments were used to calculate pipeline abrasion rates for a particular type of slurry. Waste from this project generally consisted of irradiated pipe inserts, paper wipes, and broken glassware. The slurries used in the experiments were disposed of on-site at the CSMRI. It is believed no chemical waste from this aspect of the project entered the radioactive waste sent to the INEL.

During 1963 and 1964 the CSMRI conducted classified research (CSMRI Project 330412) for the Defense Atomic Support Agency (DASA), Tonopah, Nevada. The research was in support of DASA Project 2.6D, Operation ROLLER COASTER, under contract DA-49-146-XZ-225. The purpose of the project was to determine the distribution of special nuclear materials in various soil size fractions if a high explosive detonation occurs. This included possible reactions between special nuclear materials and the mineralogical constituents of the various size fractions. The CSMRI conducted petrologic and mineralogic examinations of the pre-shot soil samples. Size analysis and alpha counting procedures were conducted on post-shot soil samples.

Documents from the CSMRI indicate the plutonium contaminated wastes from the project was packaged in four (4) 55 gallon steel drums. This waste was then delivered to the Rocky Flats Plant for disposal. It can be assumed this waste was then sent to the INEL. Dry waste, such as unused portions of 25 post-shot soil samples, contaminated paper, and glassware, was placed in two (2) drums. According to F. L. Smith, there is a good possibility an unknown number of Vycor beakers were included in the waste. These beakers were used to prepare the soil samples for alpha counting procedures.

J. D. McKinney
TLC-2-80
January 22, 1980
Page 3

Each beaker may contain up to 350 ml of 4M HCl acid. The other two (2) drums contained wet wastes from solutions used to clean soil sizing screens and infrasizer cones. The solutions contained water, small amounts of Alconox (a detergent), and acetone. The solutions in both drums were treated with an unknown flocculant. The total plutonium content of all waste drums was 2.6 micrograms.

The project engineer for all of the above projects is deceased. The location of other individuals, mainly technicians, involved in the projects is unknown. Radioactive Waste disposal services were provided by the Nuclear Engineering Company, Inc., after October, 1962. Deposition of this waste occurred at Beatty, Nevada.

fg

cc: H. M. Batchelder
J. L. Clark
J. R. Fielding
K. B. McKinley
R. B. O'Brien *RS*
R. L. Silverthorne /r/ File
Central File
T. L. Clements File (2)

Colorado School of Mines Research Institute

P.O. BOX 112 • GOLDEN, COLORADO 80401
PHONE (303) 279-2581

CSMRI

December 13, 1979

Mr. Tom Clements, Jr.
United States Department of Energy
Idaho Operations Office
550 Second Street
Idaho Falls, Idaho 83401

Dear Sir:

We are returning your waste questionnaire, which has been filled out to the best of our ability.

As I mentioned to you on the phone last week, there are some files, relative to the production of this waste, here at the Institute which you are welcome to look through if you decide on a personal visit. I could find nothing in them other than what is covered in your questionnaire.

Mr. Fred L. Smith, 8795 Ralston Road, Arvada, Colorado, 80002 was responsible for these shipments and it is possible that he could tell you more about them.

If I can be of further help with this matter please let me know.

Sincerely,

Jack E. Coulson

Jack E. Coulson, Manager
Technical Services and Construction Division

/mjm

Enc.

Colorado School of Mines Research Institute

PD BOX 112 • GOLDEN, COLORADO 80401

PHONE (303) 279-2581

CSMRI

January 9, 1980

RECEIVED

J. L. Clark

JAN 15 1980

Mr. J. L. Clark, Manager
Safety Standards Branch
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, Idaho 83415

Action of
Action
Reply to
[Handwritten signature]

Dear Mr. Clark:

I will expect to see Mr. Clements in my office on January 15 - 16, 1980. He will have access to all available materials relative to the shipments of radioactive wastes from our firm to the Idaho National Engineering Laboratory.

Sincerely,

Jack E. Coulson

Jack E. Coulson, Manager
Technical Services and Construction Division

/mjm

January 18, 1980

Mr. Jack Coulson, Manager
Technical Services and Construction Division
Colorado School of Mines Research Institute
P O. Box 112
Golden, Colorado 80401

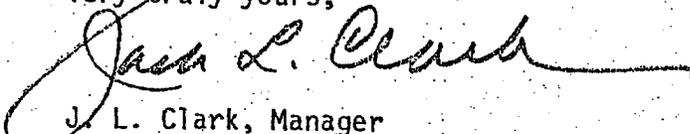
NONRADIOLOGICAL HAZARDS STUDY - JLC-21-80

Dear Mr. Coulson:

Thank you for the time and cooperation you extended to Mr. T. L. Clements on his visit to your facility on January 15-16, 1980. The waste shipment documents and project reports have been very beneficial in support of the Nonradiological Hazards Study. The help of Ms. Lola Ann Johnson in locating these documents was also appreciated.

Enclosed are copies of the waste shipment records for your file.

Very truly yours,



J. L. Clark, Manager
Safety Standards Branch

TLC:ib

Attachments:
As Stated

bcc: T. L. Clements ✓
J. R. Fielding
J. D. McKinney
R. L. Silverthorne (r) file
Central File
J. L. Clark file

PHILLIPS PETROLEUM COMPANY

Wichita Falls, Texas

November 12, 1962

Disposal of
Off-Site Radioactive Waste



NOTEGRAM

TO: Ray Fisher

FROM: A. A. Anselmo:pw *[Signature]*

Attached herewith are Waste Shipment Data forms for the burial of radioactive waste as covered by AIB-3, 11.

<u>SHIPPER</u>	<u>SERIAL NO.</u>	<u>CU FT BURIED</u>	<u>NO PIECES</u>	<u>DATE BURIED</u>
Colorado School of Mines Research Foundation, Inc P. O. Box 112 Golden, Colorado	CSM-62-1	14.6	2	11-5-62

14.6 cu ft @ 4.70 = \$10.22

Carrier: Garrett

minimum charge per shipment - \$21.00

Enc - as listed

- cc: J. W. Latchum w/o enc
- L. P. Smith w/o enc
- A. A. Anselmo file w/enc

PURCHASE REQUISITION

REQUISITIONED BY -29-62 Kent Perry	DEPARTMENT Mining	APPROVED BY <i>[Signature]</i>
ORDER TO (PERSON AND DEPT.)	CHARGE TO 50% to 320311-Act. Anal.; 50% to 300301	

ORDER FOR Radioactive waste disposal, charges for services	DATE ORDERED	PURCHASE ORDER NO. 26806
FILLED IN BY PURCHASING DEPARTMENT ONLY		

DER U.S. Atomic Energy Commission
 COM Idaho Operations Office
 c/o Phillips Petroleum Company
 P.O. Box 2067, Idaho Falls, Idaho

SHIP TO

SHIP VIA

B.	TERMS	DELIVERY DATE	CHARGE TAX	TAX EXEMPT
----	-------	---------------	------------	------------

QUANTITY	DESCRIPTION	PRICE	AMOUNT
	Disposal of radioactive waste material	21.00	21.00

OCT 29 1952
 BUSINESS OFFICE

ORDER BY	RECEIVED BY	DATE RECEIVED
----------	-------------	---------------

PHILLIPS PETROLEUM COMPANY

ATOMIC ENERGY DIVISION

P.O. BOX 2067
IDAHO FALLS, IDAHO

YOUR ORDER No.
26806

Invoice Date 12-3-62
Invoice No. 11-104
Accounting Ref. 1470-01100
J11-1-10

Charged to:

Colorado School of Mines
Research Foundation, Inc.
P. O. Box 112
Golden, Colorado

Please return a copy of this
invoice with your remittance.

DESCRIPTION

AMOUNT

For contaminated waste disposal at the NRTS burial grounds during
November, 1962 per attachment

1 shipment - minimum charge per shipment

\$21.00

C.S.M. Research Foundation
Payment Approved

Date 12-7-62
By *OW*
Account 320311 - 10.50
Project 300301 - 10.50

POSTED
OW

DO NOT TYPE BELOW THIS LINE

I certify that the above charges are correct and just
and payment therefore has not been received.

PHILLIPS PETROLEUM COMPANY

[Signature]
CHIEF ACCOUNTANT

Appendix B

Letters from Colorado School of Mines

191

UNITED STATES
ATOMIC ENERGY COMMISSION
P. O. BOX 1221
IDAHO FALLS, IDAHO

May 5, 1954

Mr. John Epp
Assistant Director, Chemical Laboratories
The Dow Chemical Company
P. O. Box 2131
Denver, Colorado

Dear Mr. Epp:

In response to your letter of April 28, requesting information relative to the condition of your van-load of contaminated waste, we are able to present the following information and observations which may be of mutual benefit in future movements of this type.

1. Truck arrived at NRTS at approximately 8:00 AM on 4/22/54, unloading and disposal were accomplished in one operation by 11:00 AM. Truck, personnel and equipment were surveyed and released by 12:00 noon.

2. An undetermined number of drums contained liquid, of these seven were leaking, five through the top and two through perforations in the bottom. At least three of these containers had leaked considerably in transit, to a degree where the liquid had soaked through the protective paper to the aluminum van floor. No detectable radioactive contamination accompanied these spills, time expended in surveying did not influence the cost of the operation. As a precautionary measure it may be advisable to use an absorbent paper on the floor of trucks carrying this type of material in the future. (Ref: - Several layers of white blotting paper, 24", B 160 caliber, .050 thickness, 25% cotton content). We are anxious to avoid disposal of liquids in our solid waste disposal ground, however, we do anticipate a certain residual amount of liquid in materials of the type that you are handling.

As a test shipment, this can be considered as exceptionally well handled by the Rocky Flats personnel. We feel that if leakage can be prevented and possible radioactive contamination of equipment and personnel is avoided that future shipments can be processed without mishap or unnecessary delay.

Yours very truly,



G. V. Beard
Chief, Health and Safety Branch
Idaho Operations Office

Appendix C

Packaging Certification by Atomic Energy Commission—ALO Contractor (August 21, 1967)

CERTIFICATION OF APPROVAL FOR FISSILE-LARGE QUANTITY SHIPPING CONTAINERS

ALBUQUERQUE OPERATIONS OFFICE, USAEC

August 21, 1967

I. ALO Contractor.

The Dow Chemical Company
Rocky Flats Division
Box 888
Golden, Colorado 80401
Contact: Traffic - W. F. Romine
Engr. - F. E. Adcock

II. Identification of Shipping Container.

Truck or trailer-on-flatcar Fissile
Class III Shipments of Radioactive Waste
in 55-gallon drums. No individual
package identification. E/E Permits
2057 and 2058 previously assigned.
ALO designation: AL-R

III. General Information Concerning Container.

Three categories of waste have been approved for shipment:

- A. Container: Used 55-gal. 18-gauge steel drums with 8-mil plastic liner.
Contents: Paper, clothing, tools and other contaminated waste generated
within the Pu fabrication area but external to the dry box system.
Pu Content: 1 gram maximum, .02 grams average.
Gross Weight: 480 pounds maximum.
- B. Container: ICC-6C or -17C (or equivalent) 55-gal. steel drum with 8-mil
plastic liner.
Contents: Tools, carbon molds, and other laboratory and production
equipment from within the Pu dry box system (line-generated waste).
Pu Content: 200 grams maximum, 30 grams average.
Gross Weight: 880 pounds maximum.
- C. Container: ICC-17C 55-gal. drum with 8-mil plastic liner.
Contents: Hardened or semi-hardened sludges, greases, neutralized
acids, and other process waste.
Pu Content: 1 to 50 grams maximum, <1 to 6 grams average, depending
on the type of residue.
Gross Weight: 880 pounds maximum.

IV. Specific Limitations and Restrictions.

1. Sole use of vehicle required.
2. Average Pu content per package must not exceed 15 grams for any vehicle.
3. Loading to be controlled by shipper's written procedure.

V. Additional Information and/or Limitations.

None.

VI. Certification of Approval.

Pursuant to Chapter AEC 0529, this container is approved subject to the
limitations described above. This certification does not relieve the
shipper of his responsibility to obtain a DOT Special Permit and to comply
with the requirements of other Federal Regulations as appropriate.

DATE: _____

Certification Official
Albuquerque Operations
U. S. Atomic Energy Commission

Appendix D
List of NDA Reports Available

NDA Reports Available

Date	Report/Speech Title	Authors	ID Number Report/Speech
5/31/66	Drum Counter – Washables	O. H. Willoughby	Internal Letter
4/28/69	Measurement of Plutonium in Process Materials and Contaminated Waste	O. H. Willoughby and D. R. Cartwright	RFP-1325 Speech
7/1/71 revised 9/30/71	Standardization and Performance of the Building 707 Drum Counter	R. N. Chanda, R. A. Harlan, R. A. Deal, J. L. Lawless and Y. Ferris	CRDL-950442-101 Internal Report
11/4/71	Building 771 Interim Drum Counter	R. A. Deal, R. N. Chanda, R. J. Nau R. A. Harlan and G. J. Cunningham	CRDL-950442-103 Internal Report
9/8/72	Operations Manual for Drum Counter (South) 771-A	R. A. Deal, L. A. Bidwell, J. L. Lawless, R. N. Chanda and H. R. Martin	CRDL-950442-109 Internal Report
4/26/73	Computer Control of Three Passive Assay Systems: Helix Counter, Can Counter II and South Drum Counter	J. L. Lawless and L. A. Bidwell	CRDL-950442-114 Internal Report
6/24/77	A Crate Counter for Normal Operating Loss	R. A. Harlan	RFP-2642 Speech
12/2/65	Drum Counter Verification Studies (Uranium in the Drum Counter)	O. H. Willoughby and L. D. Delpierre	CRDL-940232-101B Internal Report
12/30/65	Drum Counter Verification Studies (Plutonium/Graphite in the Drum)	O. H. Willoughby and L. D. Delpierre	CRDL-940232-101C Internal Report
5/27/66	Drum Counter Verification Studies (Plutonium/Graphite in the Drum Counter)	O. H. Willoughby and G. H. Cunningham	CRDL-940232-101D Internal Report
11/15/66	Drum Counter Evaluation Studies (Dead Time Counting Error Correction)	O. H. Willoughby and J. L. Lawless	CRDL-940232-101E Internal Report
3/15/67	Drum Counter Evaluation Studies (Discard Waste Evaluation Summary)	O. H. Willoughby	CRDL-940232-101F Internal Report
2/14/68	Drum Counter Status Report	O. H. Willoughby, J. L. Lawless and J. L. Martinez	CRDL-940232-101G Internal Report
2/16/68	Americium in the Can and Drum Counters	J. L. Lawless and O. H. Willoughby	CRDL-940232-101H Internal Report
2/27/68	Drum and Can Counter Computer Program	J. L. Lawless and O. H. Willoughby	CRDL-940232-101I Internal Report

Appendix E

Handbook of the Rocky Flats Plant Production Non-Destructive Assay Systems (compiled by Bill Ulbricht)

**HANDBOOK
OF THE
ROCKY FLATS PLANT
PRODUCTION
NON-DESTRUCTIVE
ASSAY SYSTEMS**



Rockwell International

**COMPILED BY:
BILL ULBRICHT
PCCO-NDA**

ROCKY FLATS PLANT
ENERGY SYSTEMS GROUP
P. O. Box 464
Golden, Colorado 80401
(303) 497-7000
Contractor to
U. S. Department of Energy



Rockwell
International

June 1984

Dear Reader:

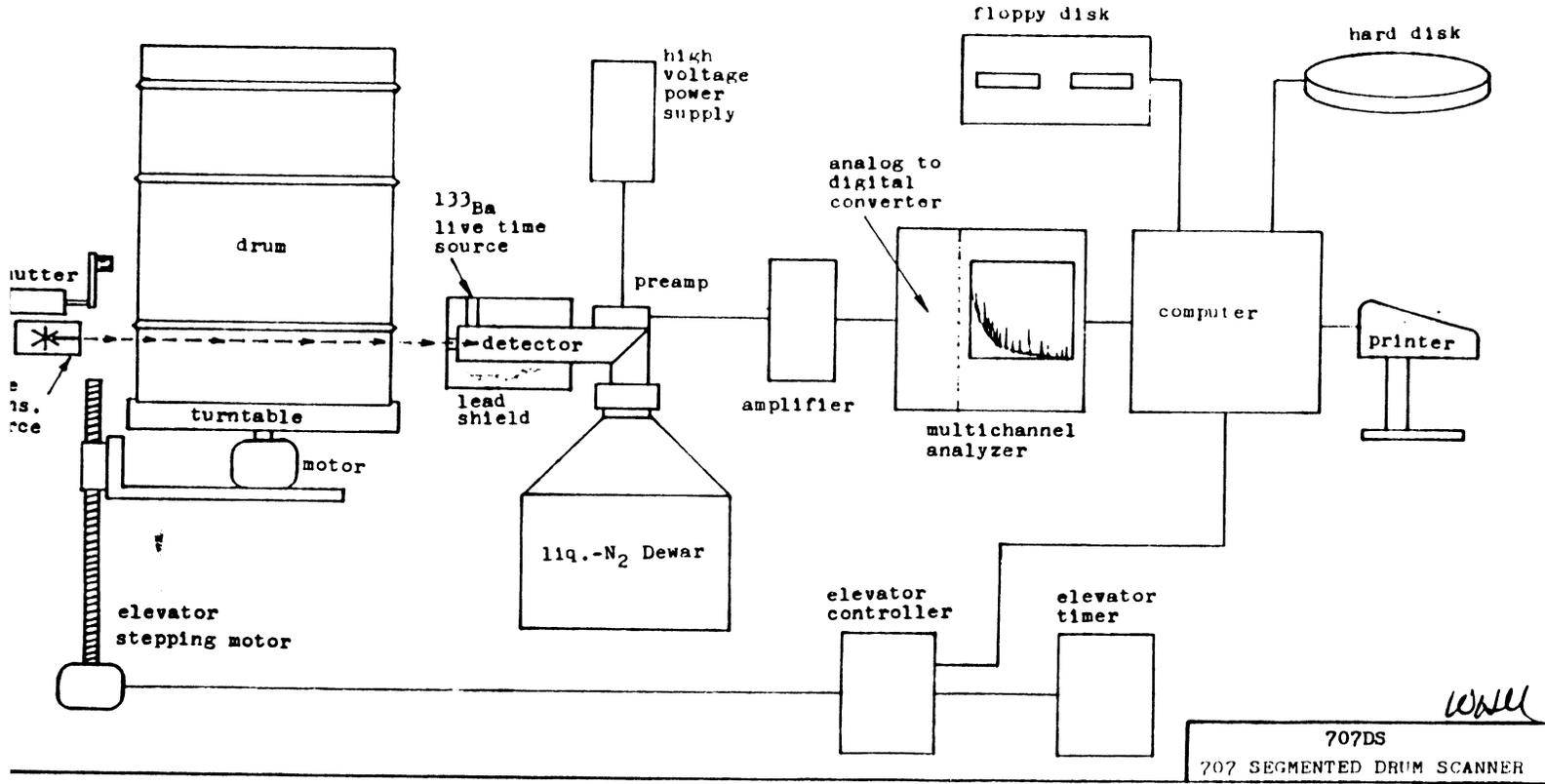
This is the first attempt at this handbook. It represents an attempt to combine information about the Production NDA Counting Systems at Rocky Flats Plant into one handbook. If you notice errors, wish additions, or have other constructive comments please direct them to Bill Ulbricht, (x7644) PCCO, Bldg. 771. This is to be an on-going effort.

Bill Ulbricht

Cordially,
Bill Ulbricht
PCCO-NDA

COMMENTS:

SIGNED: _____



June 1984

NEW NAME: 771DS OLD NAME: Segmented Drum Scanner LOCATION: bldg. 771 room 301 (annex) PHONE: 2939-annex 2601-control room

PURPOSE: Pu assay of 55 gallon drums using high resolution gamma ray spectra. Transmission corrected.

STATUS: operational

AGE: 3 years

FUTURE:

SAMPLES: TYPE(IDC): 301, 302, 320, 328, 330, 331, 336, 337, 338, 339, 374, 376, 441, 442, 480, 481

CONTAINERS: 55 gallon drums

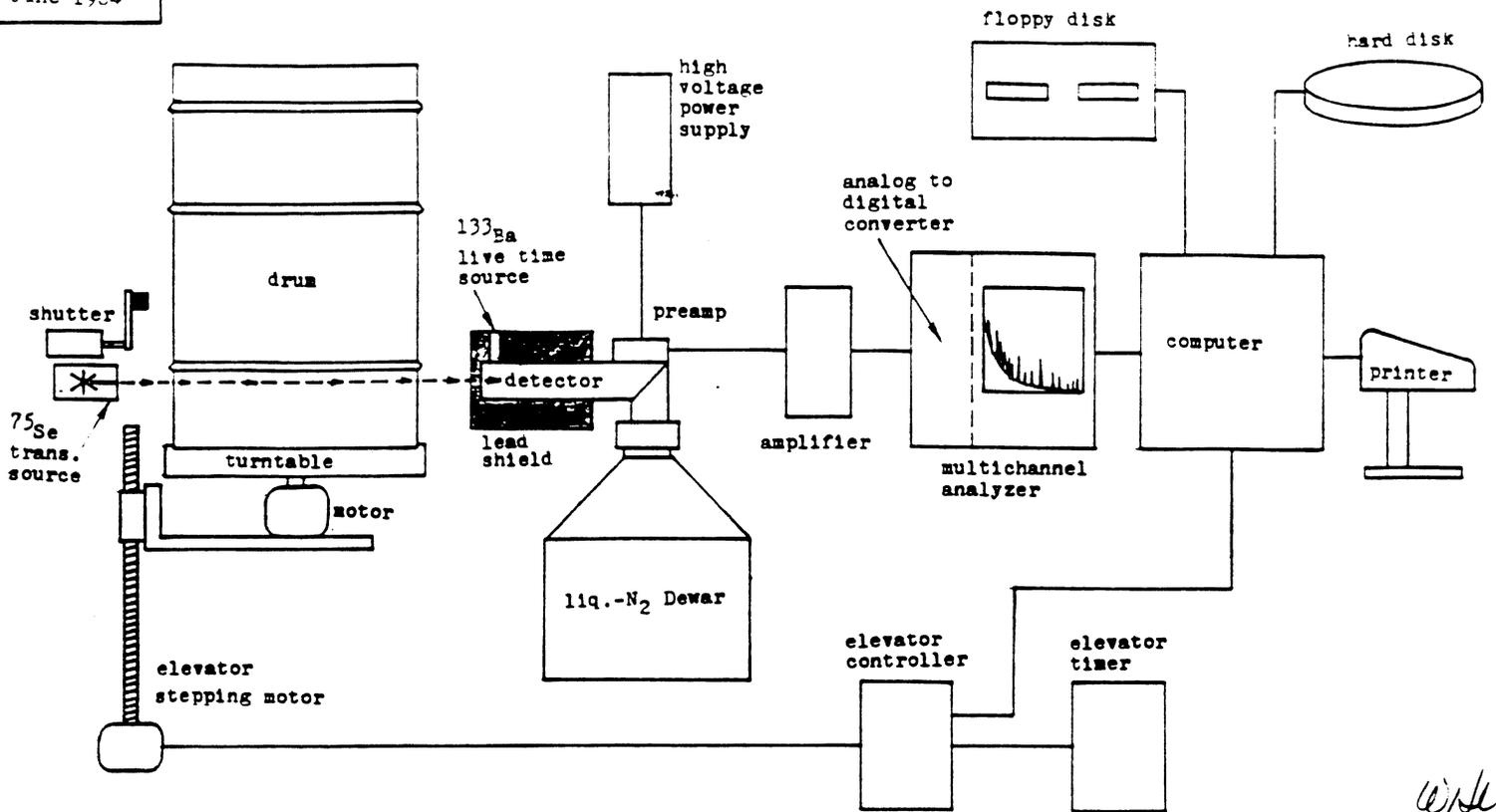
<u>SAMPLE SIZE:</u>	300 g. <u>MAX.</u>	0 g. <u>MIN.</u> Pu
	20 g. <u>MAX.</u>	0 g. <u>MIN.</u> Am
	120 g. <u>MAX.</u>	0 g. <u>MIN.</u> U

ASSAY TIME: 3 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

The elevator mechanism permits the drum to be divided into segments or slices for the gamma ray assay. Each segment is assayed individually for ^{239}Pu (414 keV) and ^{241}Am (662 keV). A transmission correction is computed for each segment based on the ^{75}Se (401 keV) transmission source peak area. A total of the quantities of Pu and Am in all the segments is computed and reported at the end of the assay.



June 1984

NEW NAME: 771CS OLD NAME: Can Scan I LOCATION: bldg. 771 room 147F PHONE: 2601

PURPOSE: Pu and Am assay of cans using high resolution gamma ray spectra. Transmission corrected.
For Molten Salts

STATUS: operational

AGE: 5 years

FUTURE:

SAMPLES: TYPE(IDC): 404, 405, 406, 407, 408, 409, 410, 411

CONTAINERS: 1 and 4 liter cans

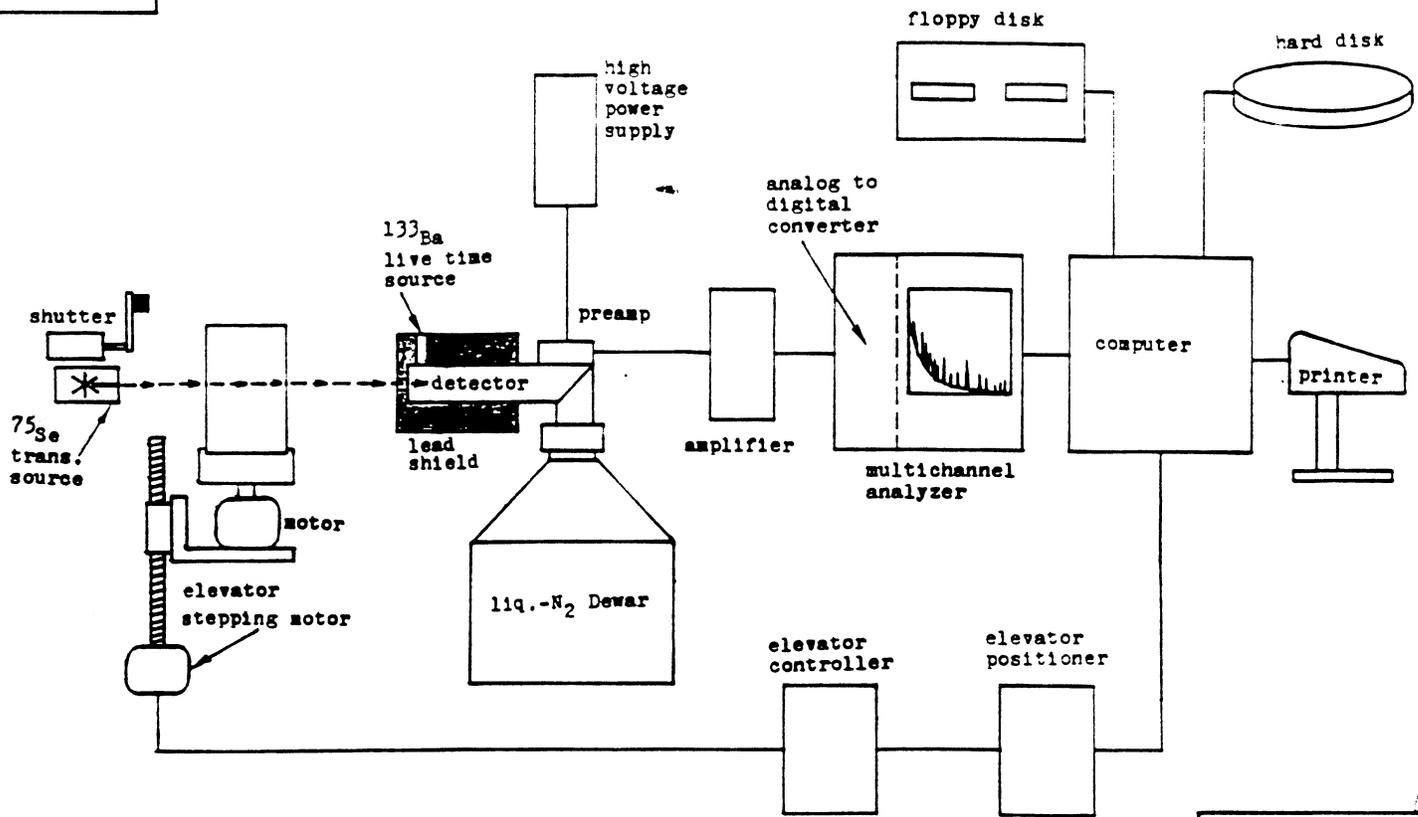
SAMPLE SIZE: 400 g. MAX. 0 g. MIN. Pu
 37 g. MAX. 0 g. MIN. Am

ASSAY TIME: 3 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

The elevator mechanism permits the can to be assayed in segments or slices. Each segment is assayed for ^{239}Pu (by examining the 414 keV peak) and ^{241}Am (662 keV). A transmission correction is computed for each segment based on the 401 keV peak (^{75}Se transmission source). A total of the quantities of Pu and Am in all the segments is computed and reported at the end of the assay.



771CS
CAN SCAN I

June 1984

NEW NAME: 776CS OLD NAME: Can Scan II LOCATION: bldg. 776 room

PHONE: 2076-process
ares
4151-control
room

PURPOSE: Pu and Am assay of Molten Salts either in line or out of line by high resolution gamma ray spectroscopy. Transmission corrected.

STATUS: operational

AGE: 2 years

FUTURE:

SAMPLES: TYPE(IDC): 404, 405, 406, 407, 408, 409, 410, 411

CONTAINERS: 1 liter cans

SAMPLE SIZE: 400 g. MAX. 0 g. MIN. Pu
 37 g. MAX. 0 g. MIN. Am

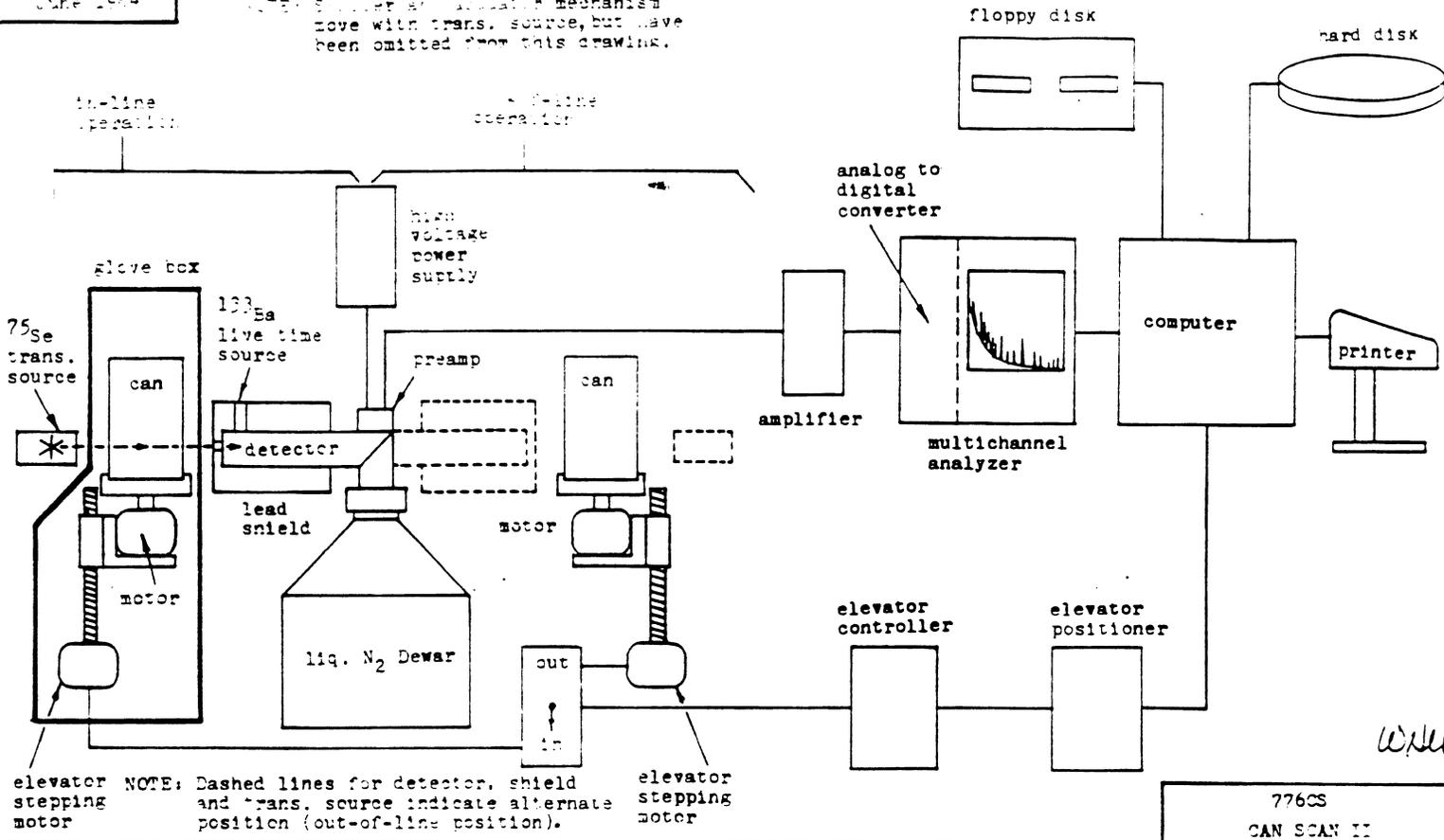
ASSAY TIME: 3 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

The elevator mechanism permits the can to be assayed in segments or slices. Each segment is assayed for ^{239}Pu by examining the 414 keV peak and ^{241}Pu with the 662 keV peak. A transmission correction is computed for each segment based on the 401 keV peak from the ^{75}Se transmission source. A total of the quantities of Pu and Am in all the segments is computed and reported at the end of the assay.

NOTE: Elevator and translation mechanism move with trans. source, but have been omitted from this drawing.



NOTE: Dashed lines for detector, shield and trans. source indicate alternate position (out-of-line position).

776CS
CAN SCAN II

WHL

June 1984

NEW NAME: 371CS1 OLD NAME: Can Scan III LOCATION: bldg. 371 room 3305(det) PHONE: 4782-control
room 3315(computer) room

PURPOSE: Pu and Am assay of Electrorefined Salts by high resolution gamma ray spectroscopy.
Transmission corrected.

STATUS: operational

AGE: 2 years

FUTURE: replace Nuclear Data data acquisition system with a DEC
computer and Canberra multichannel analyzer so that this
unit can use standardized software.

SAMPLES: TYPE(IDC): 409, 411

CONTAINERS: tall stacker/retriever cans (2.75 l.)

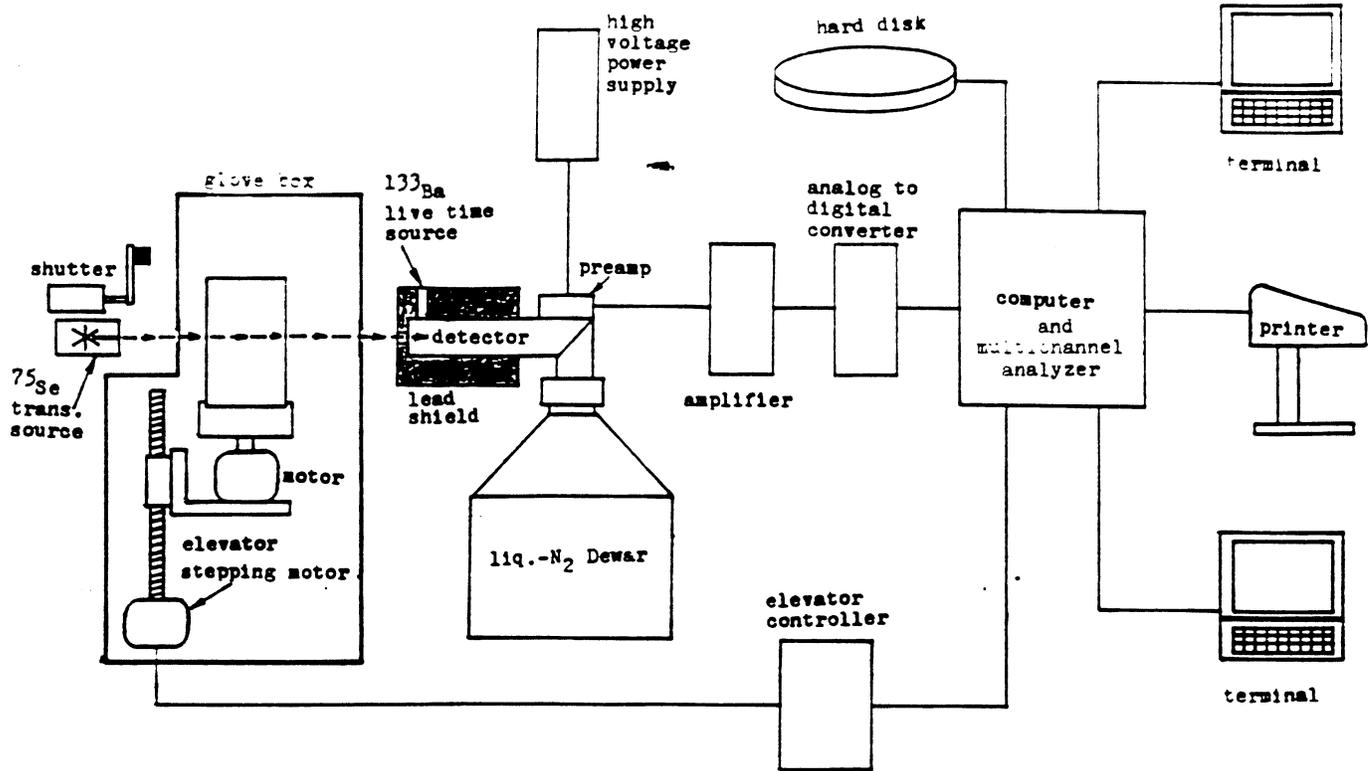
SAMPLE SIZE: 280 g. MAX. 0 g. MIN. Pu
 5 g. MAX. 0 g. MIN. Am

ASSAY TIME: 3 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

The elevator mechanism permits the can to be assayed in segments or slices. Each segment is assayed for ^{239}Pu and ^{241}Am by examining the 414 keV and 662 keV peaks respectively. A transmission correction is computed for each segment based on the area of the 401 keV peak from the ^{75}Se transmission source. A total of the quantities of Pu and Am in all the segments is computed and reported at the end of the assay.



371CS1
CAN SCAN III

W.H.H.

June 1984

NEW NAME: 371CS2 OLD NAME: Can Scan IV LOCATION: bldg. 371 room 3341 PHONE:

PURPOSE: Pu assay of residues for offsite shipment by high resolution gamma ray spectroscopy.
Transmission corrected.

STATUS: operational

AGE: 3 months FUTURE:

SAMPLES: TYPE(IDC): 392, 409, 420

CONTAINERS: 1 liter

SAMPLE SIZE: 400 g. MAX. 0 g. MIN. Pu
 37 g. MAX. 0 g. MIN. Am

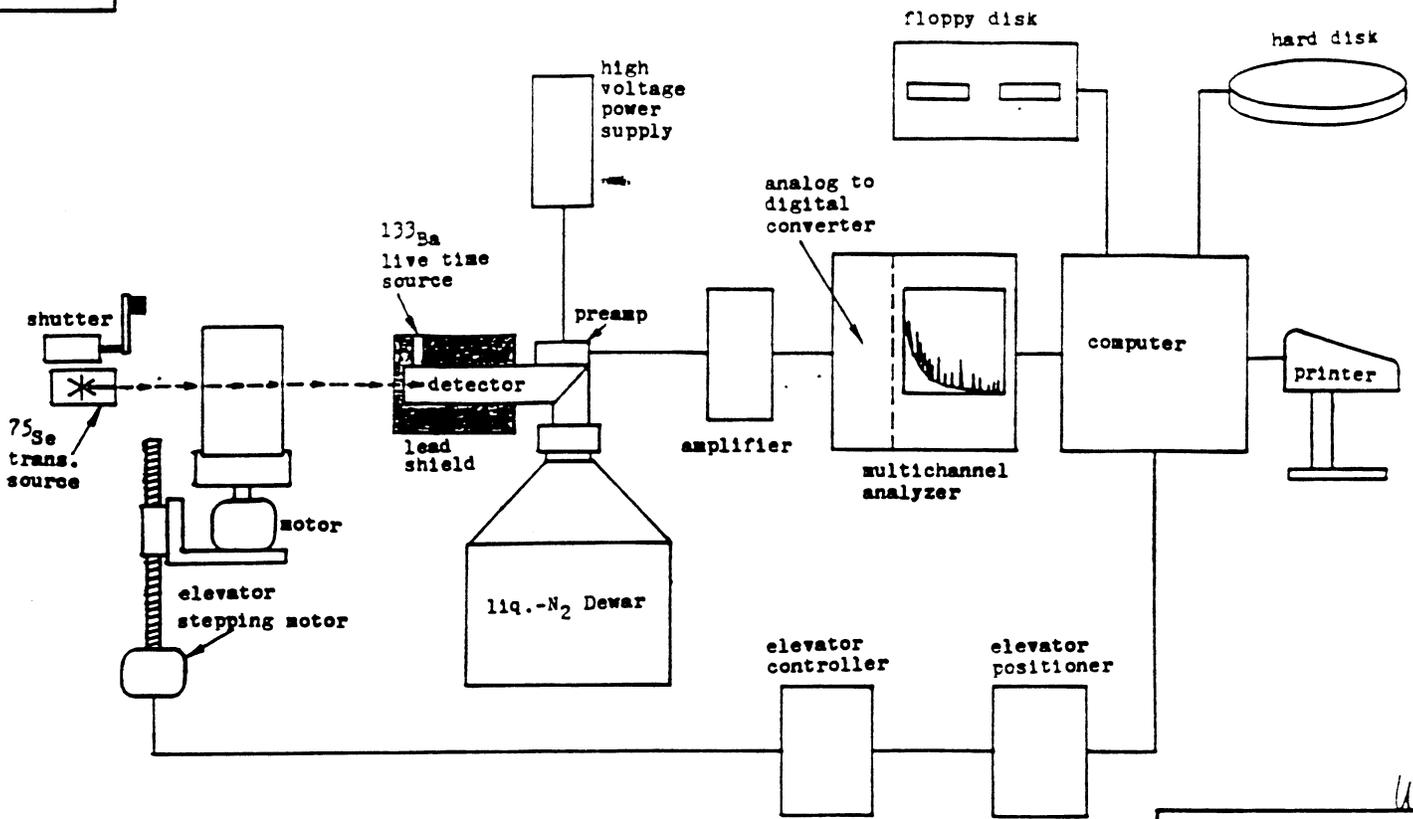
ASSAY TIME: 3 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

The elevator mechanism permits the can to be assayed in segments or slices. Each segment is assayed for ^{239}Pu and ^{241}Am by examining the 414 keV and 662 keV peaks respectively. A transmission correction is computed for each segment based on the area of the 401 keV peak from the ^{75}Se transmission source measured for each segment. A total of the quantities of Pu and Am in all the segments is computed and reported at the end of the assay.

June 1984



371CS2
CAN SCAN IV

WHL

**LOW RESOLUTION
GAMMA SYSTEMS**

June 1984

NEW NAME: 771CA1 OLD NAME: Can Counter LOCATION: bldg. 771 room 147A PHONE: 2601

PURPOSE: Pu assay by low resolution gamma ray detector (sodium iodide)

STATUS: operational

AGE: 14 years

FUTURE: to be replaced with can scan system eventually

SAMPLES: TYPE(IDC): 062, 290, 310, 311, 312, 320, 328, 330, 331, 332, 333, 334, 336, 337, 338,
339, 340, 369, 370, 371, 372, 373, 375, 376, 377, 378, 390, 391, 392, 393,
396, 397, 398, 411, 420, 421, 422, 423, 430, 431, 440, 441, 442, 480, 481

CONTAINERS: $\frac{1}{2}$ or 1 gal. poly bottle, 8801 or 8802 volrath can, 140 g. freezette

SAMPLE SIZE: 200 g. MAX. 0.5 g. MIN.

ASSAY TIME: 15 samples/hour

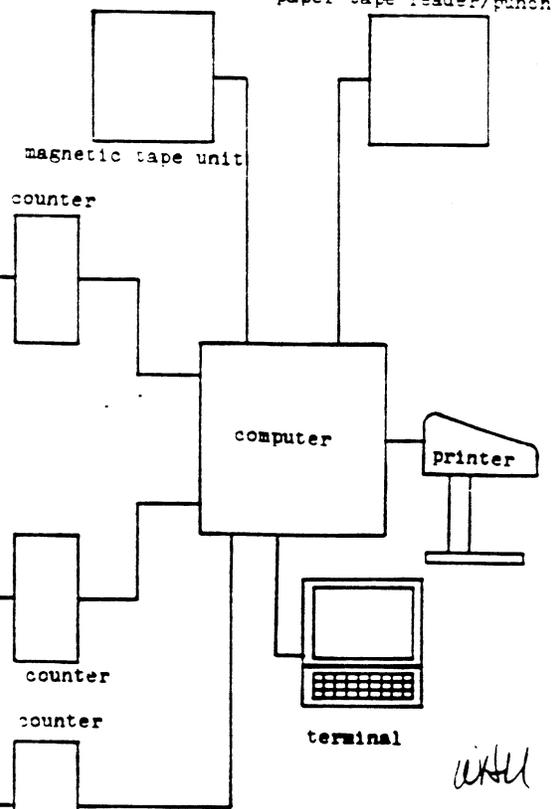
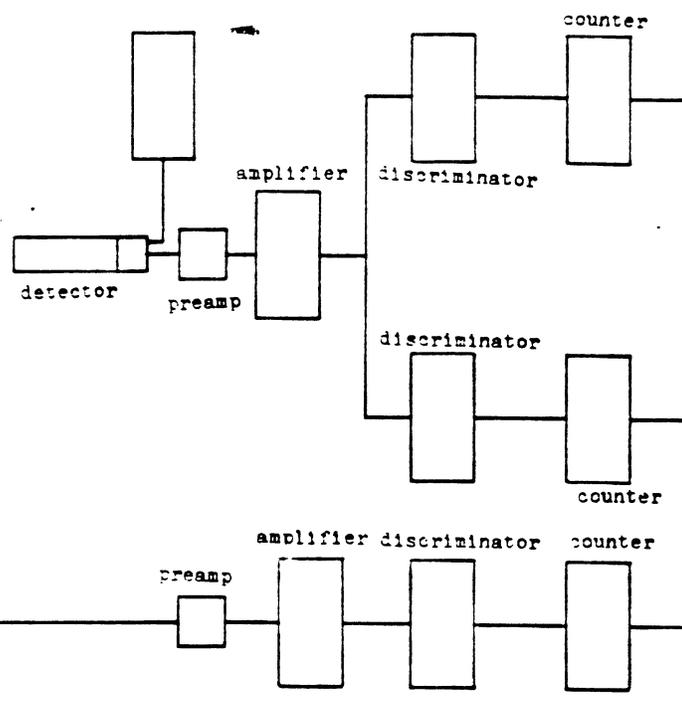
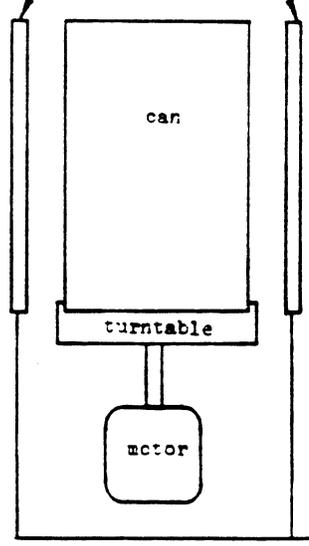
MEASUREMENT ACCURACY: \pm 20%

DESCRIPTION OF ASSAY:

The assay consists of a 100 sec. count. Two single channel analyzers are used. One detects Compton and background activity and the other Pu activity. Background and Compton counts are computed from the two values taken above and subtracted from the Pu activity counts to yield a final count. A gamma ray attenuation factor based upon prior work and assigned to the sample's IDC is applied to the final count. This product is proportional to the amount of ^{239}Pu in the sample.

The neutron activity is measured and is considered only when the gamma value and neutron value disagree by a large factor.

EP₃ neutron detectors (6 total)



771CA1
CAN COUNTER

June 1984

NEW NAME: 771CA2 OLD NAME: Helix Counter LOCATION: bldg. 771 room 147A PHONE: 2601

PURPOSE: Pu assay by low resolution gamma ray detector (sodium iodide). Transmission corrected.

STATUS: operational

AGE: 14 years

FUTURE:

SAMPLES: TYPE(IDC): 338, 396, 397, 398, 420, 421, 422, 423

CONTAINERS: ½ gal. and 1 gal. wide mouth poly bottle in clam shell

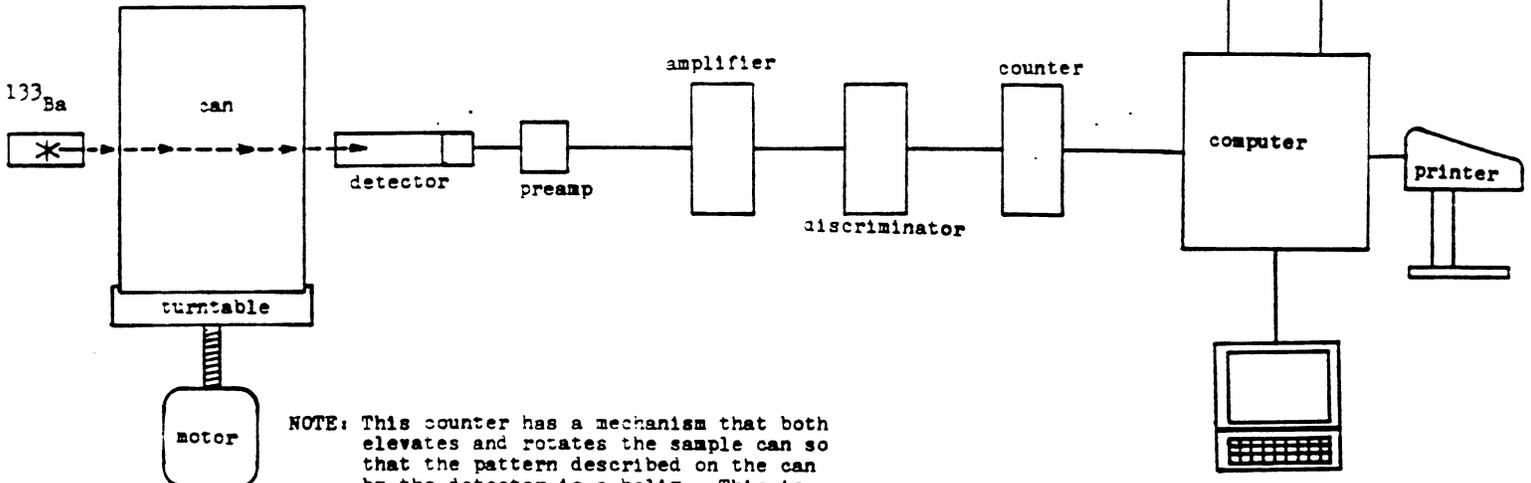
SAMPLE SIZE: 500 g. MAX. 0 g. MIN.

ASSAY TIME: 10 samples/hour

MEASUREMENT ACCURACY: 3% on IDC 420

DESCRIPTION OF ASSAY:

The helix counter is a low resolution counter that scans the sample in a helical pattern. It also utilizes a ^{133}Ba transmission source for the attenuation correction. Four 100 sec. counts are taken for each sample: (1) background, (2) background plus transmission source activity, (3) sample, and (4) sample plus transmission source. Only one counter is used and it takes data on the 384 complex of gamma rays. This assay is only applicable to full containers, otherwise the transmission correction factor will be in error.



NOTE: This counter has a mechanism that both elevates and rotates the sample can so that the pattern described on the can by the detector is a helix. This is to assure that all parts of the can are seen by the detector.

terminal

771CA2
HELIX COUNTER

WLL

June 1984

NEW NAME: 771DA OLD NAME: South Drum Counter LOCATION: bldg. 771 room 304 PHONE: 2939
(annex)

PURPOSE: Pu assay by low resolution gamma ray detector (sodium iodide) for 55 gallon drums

STATUS: operational

AGE: 14 years

FUTURE: upgrade

SAMPLES: TYPE(IDC): 292, 300, 301, 302, 320, 328, 330, 331, 334, 335, 336, 337, 338, 339, 370,
371, 372, 374, 375, 376, 377, 378, 425, 430, 431, 432, 440, 441, 442, 480,
481, 490

CONTAINERS: 55 gallon drums

SAMPLE SIZE: 300 g. MAX. g. MIN.

ASSAY TIME: 25 samples/hour

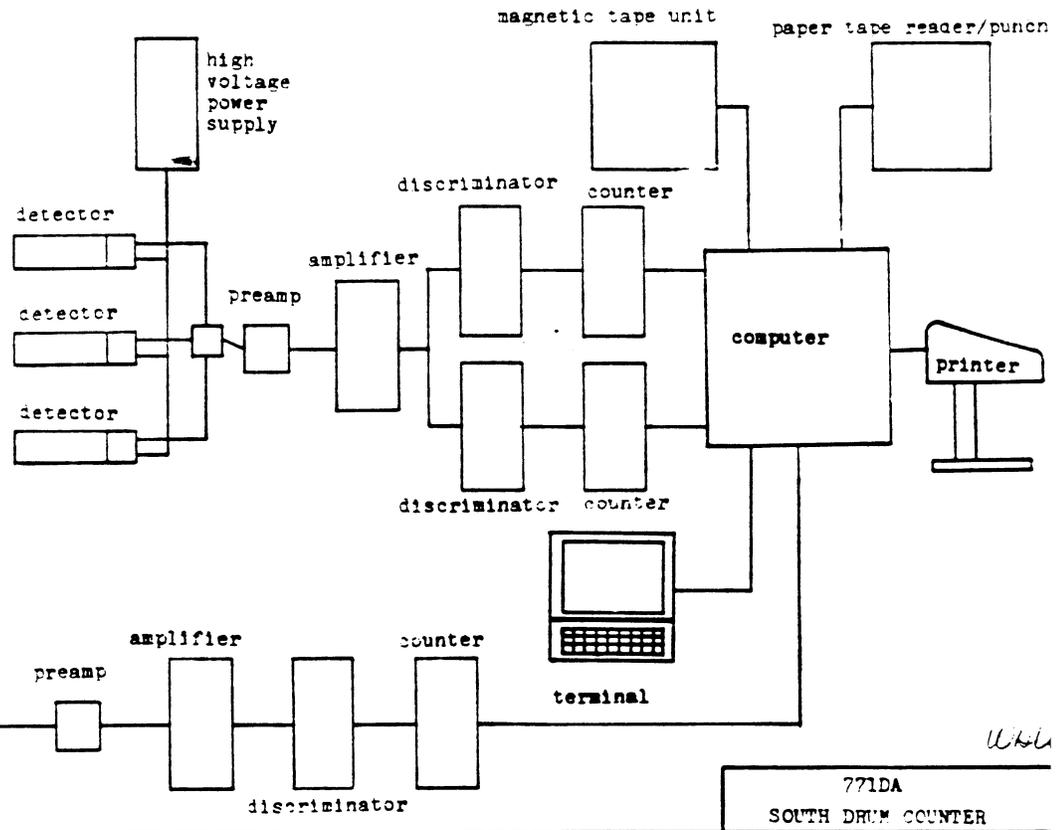
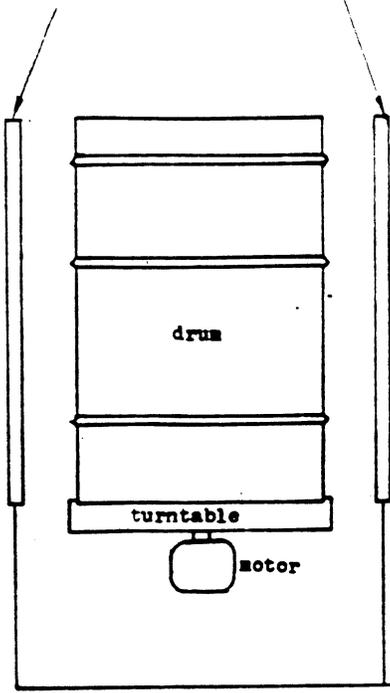
MEASUREMENT ACCURACY: 30%

DESCRIPTION OF ASSAY:

The assay consists of a 100 sec. count. Two single channel analyzers are used. One collects counts from the background and Compton regions of the spectrum while the other measures counts in the region of the 384 complex of gamma ray peaks. The Compton and background counts are subtracted from the 384 peak complex and the peak area is computed. A gamma ray attenuation factor based upon prior determinations of attenuation factors for samples of the same IDC is applied with the 384 complex peak area to compute the Pu in the sample.

The neutron activity is measured and is considered only when the gamma value for the Pu in the sample and the neutron value for the Pu disagree by a large factor.

BF₃ neutron detectors (12 total)



771DA
SOUTH DRUM COUNTER

WBL

June 1984

NEW NAME: 776DA

OLD NAME: "LCSAC"

LOCATION: bldg. 776

PHONE:

PURPOSE: Pu assay to 100 nCi/g. of sample (alpha activity) using low resolution gamma ray method

STATUS: built but not yet installed or tested

AGE: 0 years

FUTURE:

SAMPLES: TYPE(IDC):

CONTAINERS: 55 gallon drums

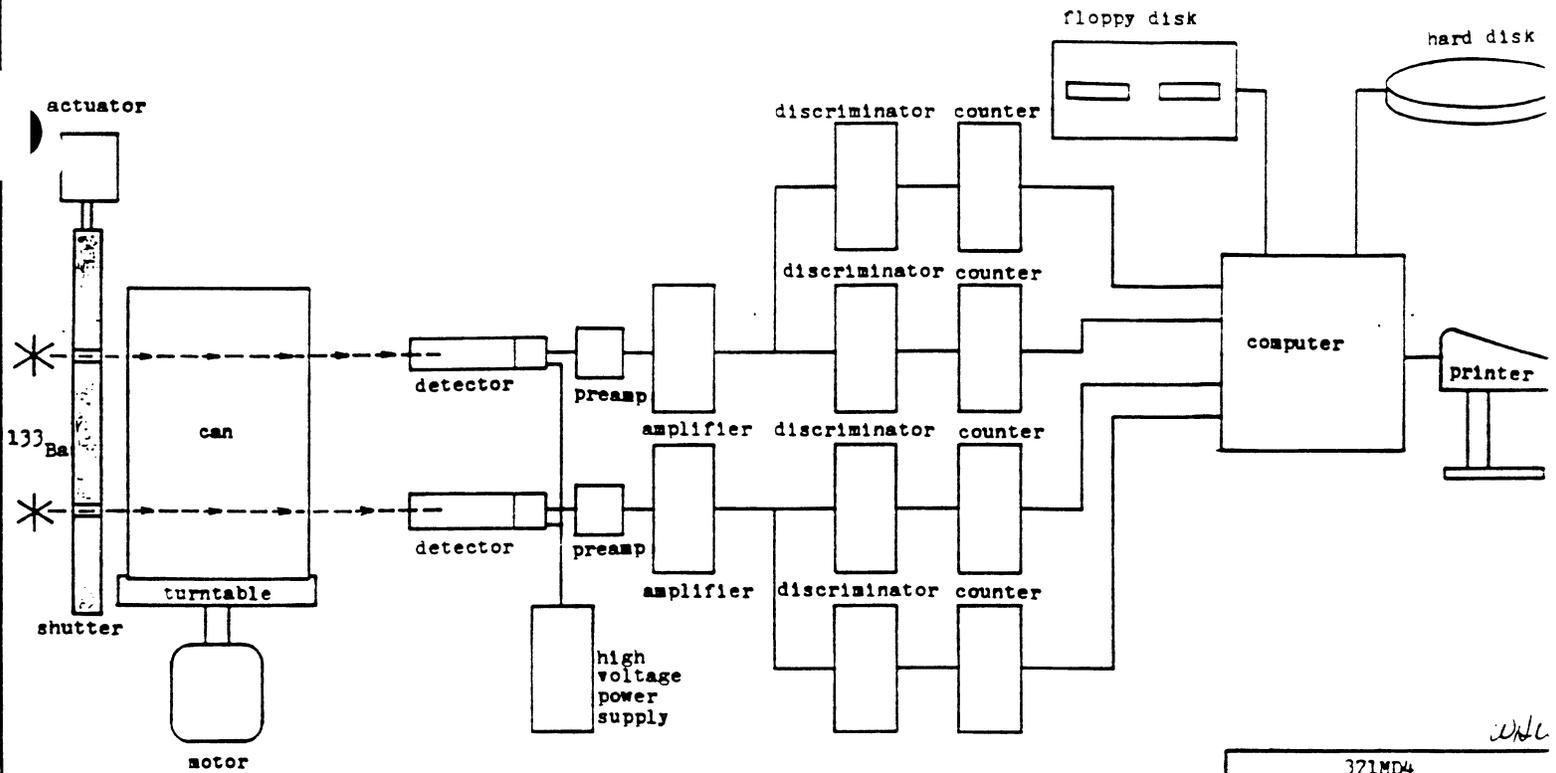
SAMPLE SIZE: g. MAX. g. MIN.

ASSAY TIME: 3-5 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

Segmented assay of drums using large area sodium iodide detector with transmission source for matrix corrections.



371MD4
ML-28

WHL

June 1984

NEW NAME: 371CA4 OLD NAME: MD-28 LOCATION: bldg. 371 room PHONE:

PURPOSE: In line Pu assay using low resolution gamma ray detector (sodium iodide). Transmission corrected.

STATUS: operational

AGE: 9 years

FUTURE: replace with a segmented gamma scanner

SAMPLES: TYPE(IDC): 062, 290, 299, 300, 310, 312, 320, 330, 331, 332, 333, 336, 337, 338, 339,
340, 371, 372, 377, 378, 390, 391, 392, 393, 394, 395, 396, 397, 398, 399,
420, 421, 422, 423, 425, 431, 440, 441, 442, 480, 481

CONTAINERS: tall stacker/retriever cans (2.75 l.)

SAMPLE SIZE: 200 g. MAX. 0 g. MIN.

ASSAY TIME: 7 samples/hour

MEASUREMENT ACCURACY: $\pm 15\%$

DESCRIPTION OF ASSAY:

This counter has two low resolution detectors (NaI) and a transmission source for matrix attenuation corrections for each detector. Both detectors have two discriminators each. One discriminator for each detector is set for Compton and background and the other is set for the 384 gamma ray complex of ^{239}Pu . From these values a 384 complex peak value is computed without the Compton and background error. A peak value is reported for both detectors.

Four 60 sec. counts are taken for each assay: (1) background, (2) background plus transmission source, (3) sample only, and (4) sample plus transmission source.

This assay is applicable only to full containers, otherwise the transmission correction will be in error.

June 1984

NEW NAME: 371CA3 OLD NAME: MD-22 LOCATION: bldg. 371 room PHONE:

PURPOSE: In line Pu assay using low resolution gamma ray detector (sodium iodide). Transmission corrected.

STATUS: not operational

AGE: 9 years

FUTURE: replace with segmented gamma scanner

SAMPLES: TYPE(IDC): 062, 290, 299, 300, 310, 312, 320, 330, 331, 332, 333, 336, 337, 338, 339,
340, 371, 372, 377, 378, 390, 391, 392, 393, 394, 395, 396, 397, 398, 399,
420, 421, 422, 423, 425, 431, 440, 441, 442, 480, 481

CONTAINERS: tall stacker/retriever cans (2.75 l.)

SAMPLE SIZE: 200 g. MAX. 0g. MIN.

ASSAY TIME: 7 samples/hour

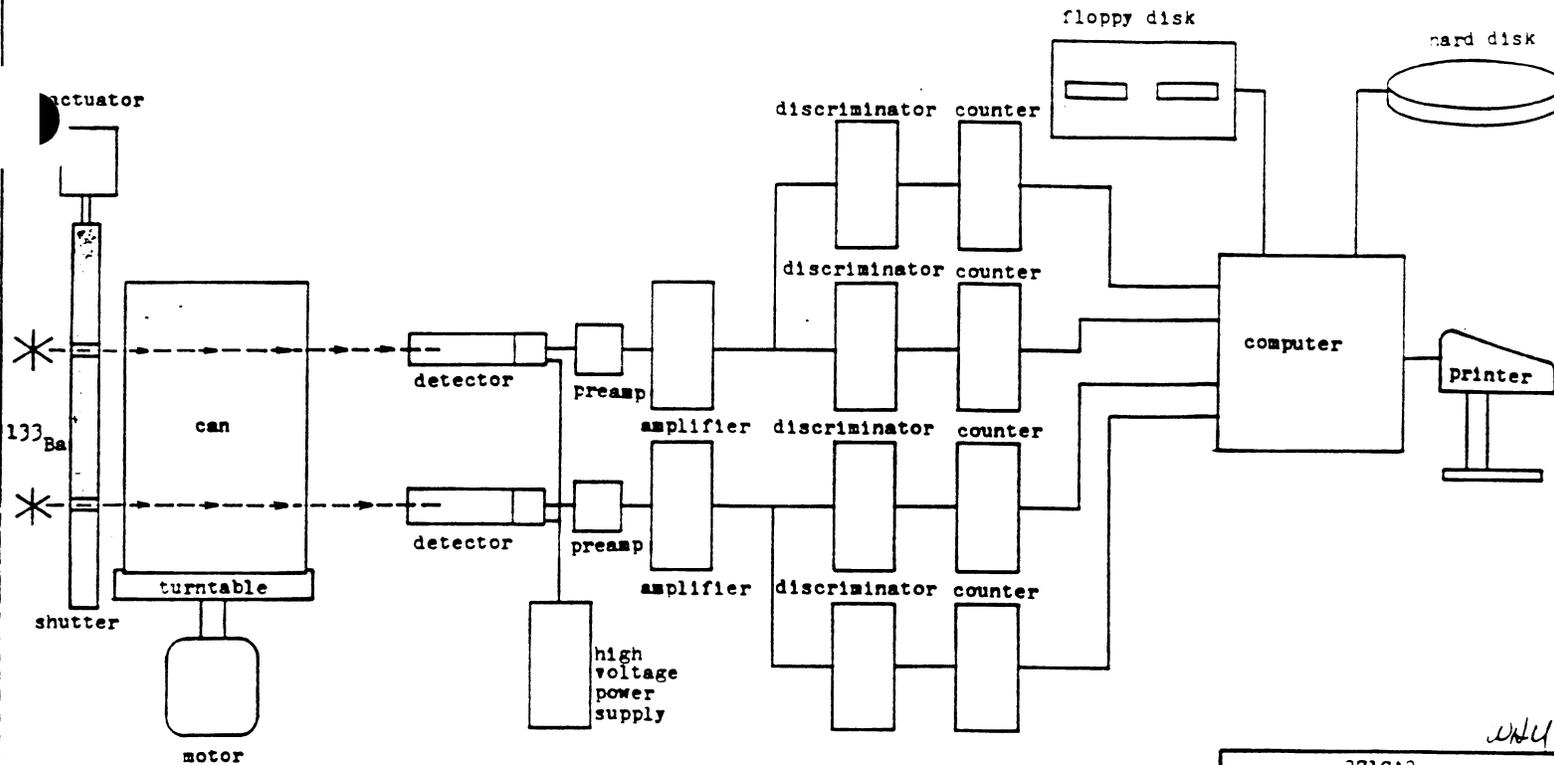
MEASUREMENT ACCURACY: \pm 15%

DESCRIPTION OF ASSAY:

This counter has two low resolution detectors (NaI) and a transmission source for matrix attenuation corrections for each detector. Both detectors have two discriminators each. One discriminator for each detector is set for Compton and background and the other is set for the 384 gamma ray complex of ²³⁹Pu. From these values a 384 complex peak value is computed without the Compton and background error. A peak value is reported for both detectors.

Four 60 sec. counts are taken for each assay: (1) background, (2) background plus transmission source, (3) sample only, and (4) sample plus transmission source.

This assay is applicable only to full containers, otherwise the transmission correction will be in error.



371CA3
MD-22

June 1984

NEW NAME: 371CA2 OLD NAME: MD-21

LOCATION: bldg 371 room

PHONE:

PURPOSE: In line Pu assay using low resolution gamma ray detector (sodium iodide). Transmission corrected.
STATUS: not operational

AGE: 9 years

FUTURE: to be replaced with segmented can scanner

SAMPLES: TYPE(IDC):

CONTAINERS: tall stacker retriever cans (2.75 l.)

SAMPLE SIZE: g. MAX. g. MIN.

ASSAY TIME: 7 samples/hour

MEASUREMENT ACCURACY: $\pm 15\%$

DESCRIPTION OF ASSAY:

This counter has two low resolution detectors (NaI) and a transmission source for matrix attenuation corrections for each detector. Both detectors have two single channel analyzers (discriminators) each. One discriminator for each detector is set for Compton and background and the other is set for the 384 gamma ray complex of ^{239}Pu . From these values a 384 complex peak value is computed without the Compton and background error. A peak value is reported for both detectors.

Four 60 sec. counts are taken for each assay: (1) background, (2) background plus transmission source, (3) sample only, and (4) sample plus transmission source.

This assay is applicable only to full containers, otherwise the transmission correction will be in error.

June 1984

NEW NAME: 371CA1 OLD NAME: MD-16 LOCATION: bldg. 371 room 3515 PHONE: 4077-control
control room 3513 room

PURPOSE: In line Pu assay using low resolution gamma ray detector (sodium iodide). Transmission
STATUS: operational corrected.

AGE: 9 years FUTURE: to be replaced with segmented can scanner

SAMPLES: TYPE(IDC): 062, 290, 299, 300, 310, 312, 320, 330, 331, 332, 333, 336, 337, 338, 339,
340, 371, 372, 377, 378, 390, 391, 392, 393, 394, 395, 396, 397, 398, 399,
420, 421, 422, 423, 425, 431, 440, 441, 442, 480, 481
CONTAINERS: tall stacker/retriever can (2.75 l.)

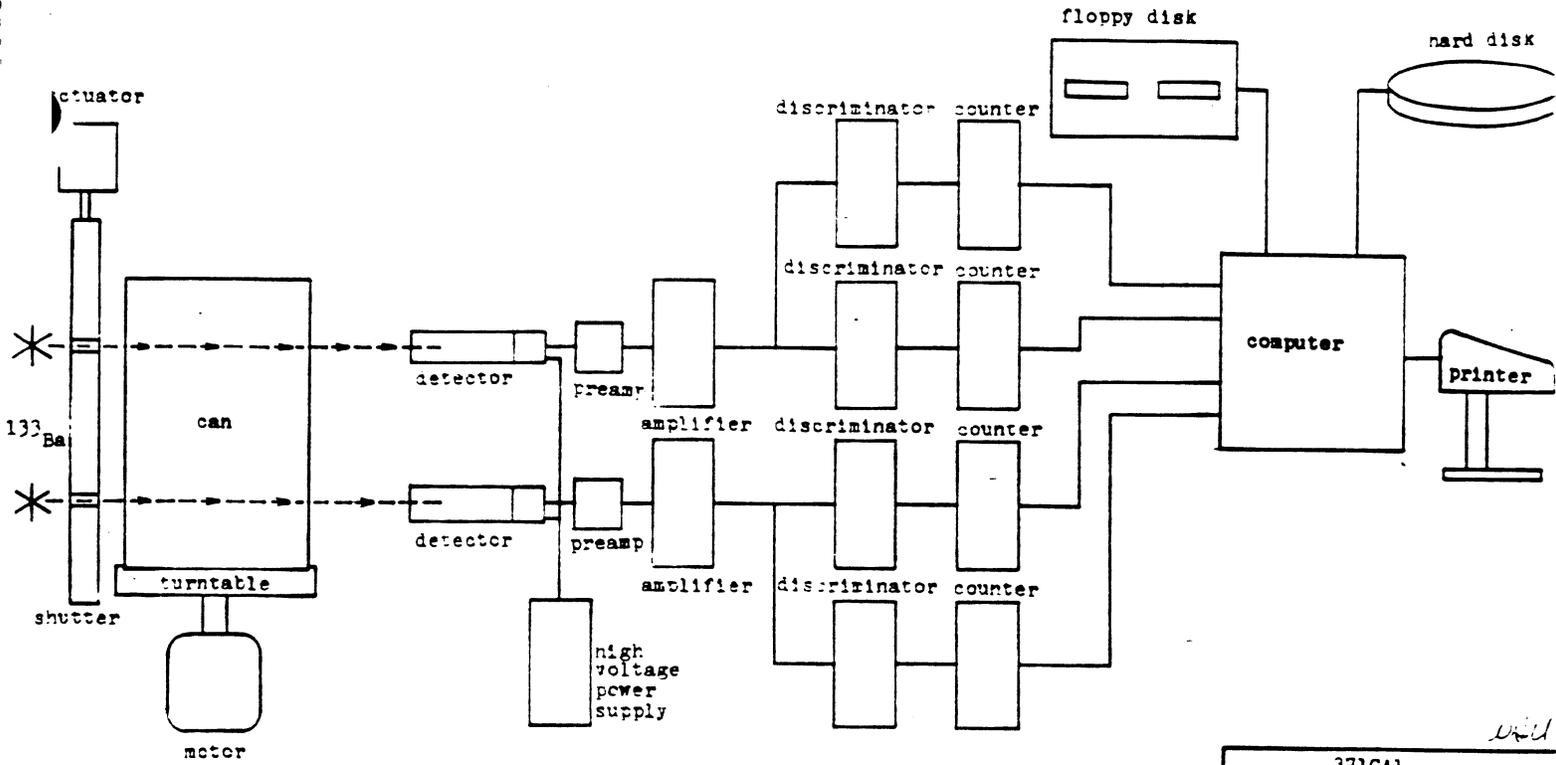
SAMPLE SIZE: 200 g. MAX. 0 g. MIN.

ASSAY TIME: 7 samples/hour

MEASUREMENT ACCURACY: \pm 15%

DESCRIPTION OF ASSAY:

This counter is for residues in the button breakout area of bldg. 371. It has two low resolution detectors (NaI) and a transmission source for matrix attenuation for each detector. Both detectors have two single channel analyzers (discriminators) each. One discriminator for each detector is set for Compton and background and the other discriminator for each detector is set for the 384 gamma ray peak complex. From these values a 384 peak complex value is computed without the Compton and background interferent. A peak value is reported for both detectors. Four 60 sec. counts are taken for each assay: (1) background, (2) background plus transmission source, (3) sample only, and (4) sample plus transmission source. This assay is applicable only to full containers, otherwise the transmission correction for the top detector will be in error.



371CA1
MD-16

WJL

**NEUTRON
ASSAY SYSTEMS**

June 1984

NEW NAME: 664CrN1 OLD NAME: Old Crate Center LOCATION: bldg. 664

PHONE: 2411-office
7368-warehouse
7643-control
room

PURPOSE: accountability by passive neutron and gamma ray assay

STATUS: operational

AGE: 5 years

FUTURE:

SAMPLES: TYPE(IDC): 330, 337, 480, 481, 490

CONTAINERS: 4' x 4' x 7' metal or wooden crates

SAMPLE SIZE: 350 g. MAX. 2 g. MIN.

ASSAY TIME: 1 samples/hour

MEASUREMENT ACCURACY: error to within a factor of 2

DESCRIPTION OF ASSAY:

Coincident neutrons from the spontaneous fissioning of ^{240}Pu are detected. The number of coincident neutrons is proportional to the amount of ^{240}Pu present.

The gamma ray assay is an independent system for determining the total amount of Pu in the crates.

Both the neutron assay and the gamma ray assay suffer from a variable background because of the location.

June 1984

NEW NAME: 664CrN2 OLD NAME: New Crate Counter LOCATION: bldg. 664
PACC

PHONE: 2411-office
7368-warehouse
7643-control
room

PURPOSE: Pu assay to separate low level waste from TRU waste

STATUS: installed, still in testing phase, not yet operational

AGE: 1 year

FUTURE:

SAMPLES: TYPE(IDC): 330, 337, 480, 481, 490

CONTAINERS: 4' x 4' x 7' metal or wooden crates

SAMPLE SIZE: g. MAX. g. MIN.

ASSAY TIME: 2 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

³He tubes (neutron detectors) are located in the walls of the assay chamber to completely surround the crate when the chamber is closed.

In the passive mode coincident neutrons from the spontaneous fissioning of ²⁴⁰Pu are detected for the analytical determination. The sensitivity for this mode is predicted to be about 10 mg of ²⁴⁰Pu.

The active portion of the detection system uses a pulsed neutron generator and detects prompt neutrons from induced fissions of ²³⁹Pu with special detectors (shielded ³He tubes).

The sensitivity of this method is predicted to be about 1 mg. ²³⁹Pu.

Both assay modes will be matrix dependent. In addition, both methods suffer from a variable neutron background because of the instrument's location.

June 1984

NEW NAME: 771DN OLD NAME: North Drum Counter LOCATION: bldg. 771 room 301 PHONE: 2939
(annex)

PURPOSE: Pu assay of drums by passive and/or active neutron methods

STATUS: being installed in bldg. 771 north drum counter well

AGE: 0 years.

FUTURE:

SAMPLES: TYPE(IDC): 320, 339, 480, 484, 485

CONTAINERS: 55 gal. drums

SAMPLE SIZE: g. MAX. g. MIN.

ASSAY TIME: 3-4 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

In the passive mode coincident neutrons from the spontaneous fissioning of ^{240}Pu are detected for the analytical determination. Sensitivity for this method is predicted to be 100 nCi/g. of sample.

The active mode uses a pulsed neutron generator and detects prompt neutrons from ^{239}Pu fissions with special detectors (shielded ^3He tubes). The sensitivity of this method is predicted to be 1 nCi/g. of sample.

Both methods will be matrix dependent.

June 1984

NEW NAME: 371DN OLD NAME: Drum Counter LOCATION: bldg 371 room PHONE:

PURPOSE: Pu assay of drums by passive and/or active neutron methods for Waste Operations

STATUS: to be ordered

AGE:

FUTURE:

SAMPLES: TYPE(IDC):

CONTAINERS: 55 gallon drums

SAMPLE SIZE: g. MAX. g. MIN.

ASSAY TIME: 3-4 samples/hour

MEASUREMENT ACCURACY:

DESCRIPTION OF ASSAY:

In the passive mode coincident neutrons from the spontaneous fissioning of ^{240}Pu are detected for the analytical determination. Sensitivities for this method are predicted to be on the order of 100 nCi/g. of sample.

The active mode uses a pulsed neutron generator and detects prompt neutrons from the induced fissioning of ^{239}Pu with special detectors (shielded ^3He tubes). The sensitivity of this method is estimated to be of the order of 1 nCi/g. of sample

Both methods will be matrix dependent.

June 1984

NEW NAME: 371CN OLD NAME: MD-30
Calcined Oxide LOCATION: bldg. 371 room 3511 PHONE:
Coincidence Counter

PURPOSE: Pu assay of PuO₂ from calciner by coincident neutron method

STATUS: not calibrated, not yet operational

AGE: 3 years

FUTURE:

SAMPLES: TYPE(IDC): 067

CONTAINERS: short stacker/retriever cans (0.7 l.)

SAMPLE SIZE: 2000 g. MAX. 0 g. MIN.

ASSAY TIME: 12 samples/hour

MEASUREMENT ACCURACY: ± 2%

DESCRIPTION OF ASSAY:

Coincident neutrons from the spontaneous fissioning of ²⁴⁰Pu are detected. The number of coincident neutrons from the sample is related to the amount of ²⁴⁰Pu present. The isotopic ratios for plutonium is needed for a total Pu determination. Matrix effects could cause errors.

APPENDIX

JUNE 1984

ASSAY METHOD

	BLDG.	NEW NAME	OLD NAME	HI RES.	LO RES.	PASS. #	ACT. #	APPROX. % UPTIME	COMMENTS
(1)	371	371CS1	CAN SCAN III	*					
(1)	371	371CS2	CAN SCAN IV	*					
(1)	371	371CA1	MD-16		*				
(2)	371	371CA2	MD-21		*				
(2)	371	371CA3	MD-22		*				
(1)	371	371CA4	MD-28		*				
(2)	371	371CN	COINCIDENCE COUNTER				*		
(5)	371	371DS	SEGMENTED DRUM SCANNER	*					
(5)	371	371DN	DRUM COUNTER			*	*		
(1)	664	664CrN1	*OLD* CRATE COUNTER	*		*			
(3)	664	664CrN2	*NEW* CRATE COUNTER			*	*		
(1)	707	707DS	SEGMENTED DRUM SCANNER	*					
(3)	707	707CN	COINCIDENCE COUNTER			*			
(1)	771	771CS	CAN SCAN I	*					
(1)	771	771DS	SEGMENTED DRUM SCANNER	*					
(1)	771	771CA1	CAN COUNTER		*				
(1)	771	771CA2	HELIX COUNTER		*				
(1)	771	771DA	SOUTH DRUM COUNTER		*				
(4)	771	771DN	NORTH DRUM COUNTER			*	*		
(1)	776	776CS	CAN SCAN II	*					
(4)	776	776DA	*LOSAC*		*				

NEW NAME: EXAMPLE: 664CrN1

- Counter number for that type counter in building.
- Type assay: S=segmented gamma scan; A=low resolution gamma assay; N=neutron method
- Type container: C=can; D=drum; Cr=crate
- Building number for detector system

'KEY'

- 1-Existing and operational
- 2-Old system. Not yet operational
- 3-New system. Not yet operational
- 4-New system. Ordered but not installed (past the planning stage)
- 5-New system. Planned

060	oxide	374	blacktop, concrete, dirt, sand	432	resin, cemented
062	oxide heel	375	oil, dry	440	glass
067	oxide in small stacker can	376	cemented insulation	441	rashing rings
290	filter sludge	377	coarse fire brick	442	rashing rings, leached
292	cemented sludge	378	pulverized fire brick	480	light non-SS metal (Al, Cu, SS, Fe)
299	miscellaneous sludge	390	unpulverized slag	481	light non-SS metal, leached
300	graphite molds	391	unpulverized slag and crucible	484	scrap metal, classified shapes
301	graphite cores	392	unpulverized sand, slag and crucible		non-SS
302	benelex and plexiglass	393	sand, slag and crucible heel	486	classified tooling for disposal
310	graphite, pulverized or fines	394	sand from button breakout (371)	490	hepa filters
311	graphite heels	395	unpulverized slag and crucible		
312	graphite, coarse	396	pulverized slag		
320	heavy, non-SS metal (Ta, W, Pb, Pt)	397	pulverized slag and crucible		
328	ful-flo filters (from incinerator)	398	pulverized slag and crucible		
330	dry combustibles	399	pulverized slag and crucible		
331	ful-flo filters (not from incinerator)	404	molten salt, Ca, Zn, K		
332	oily sludge	405	molten salt, unknown and unpulverized		
333	calcium metal	406	molten salt, unknown and pulverized		
334	fire blankets	407	molten salt 8% unpulverized		
335	filters, 8 x 8	408	molten salt 8% pulverized		
336	wet combustibles	409	molten salt 30% unpulverized		
337	plastic, washables, etc.	410	molten salt 30% pulverized		
338	insulation	411	electrorefined salt		
339	leaded dry box gloves	420	incinerator ash, virgin		
340	sludge from size reduction	421	ash heel		
369	leco heels	422	soot		
370	leco crucible	423	soot heel		
371	fire brick	425	fluid bed ash		
372	grit	430	ion column resin		
373	fire brick heels	431	resin, leached		

Appendix F

Handbook of the Rocky Flats Plant Production Nondestructive Assay Systems (1987)

Handbook of the Rocky Flats Plant

Production Nondestructive Assay Systems

1987

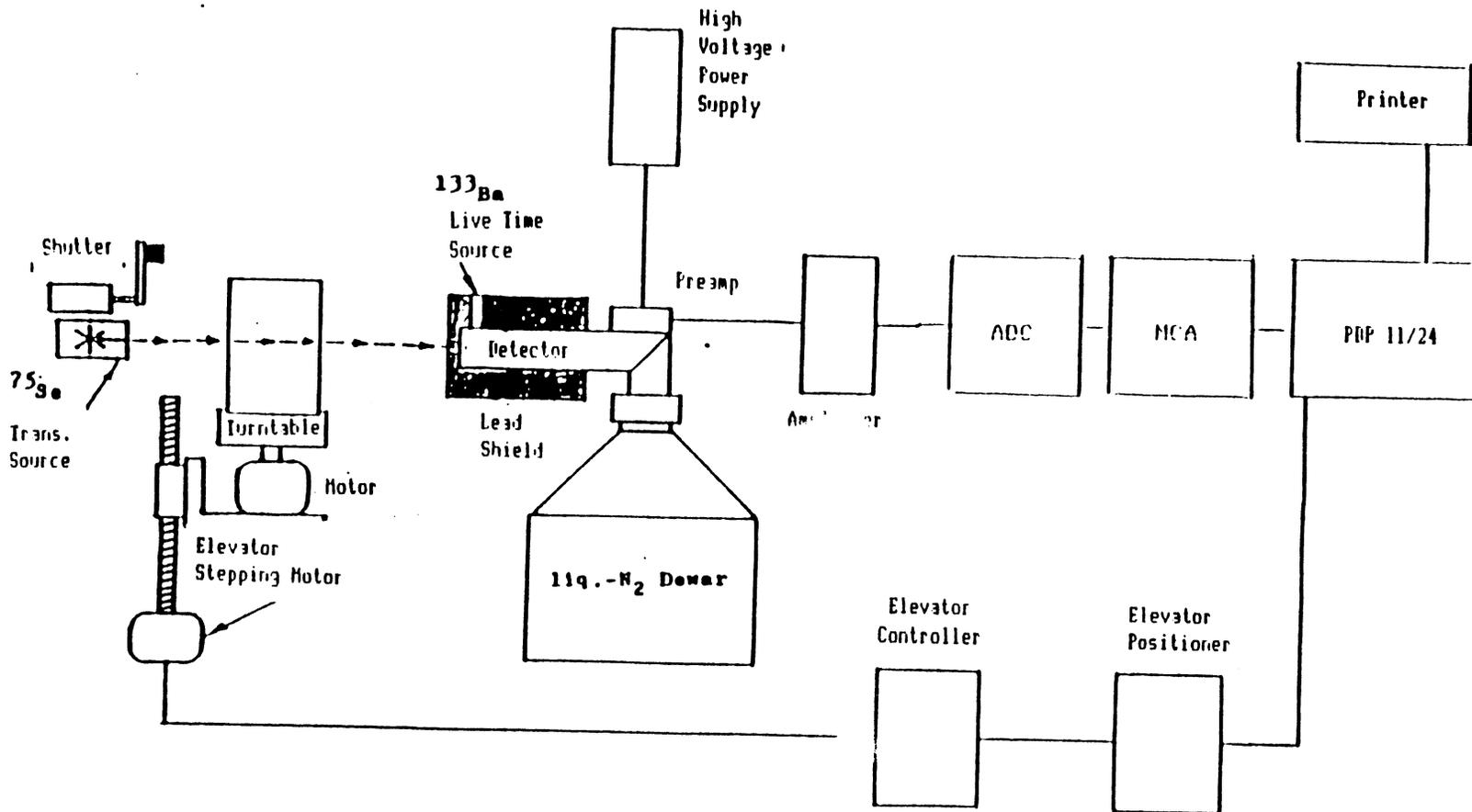


Rockwell International

Aerospace Operations
Rocky Flats Plant

HIGH RESOLUTION GAMMA CAN SCAN--SINGLE PASS

771SCSCS01



Date: 10-Nov 1987

New Name: 771SGSCS01 Old Name: CAN SCAN I
Location: BUILDIN 771 Room: 147C Phone: 2601
Yr Installed: 1981

Samples: Type (IDC): 310,312,368,387,392,398,404,405,406,407
408,409,410,411,413,414,415,420

Container Size: IDC 409 ONE LITER VOLRATH CAN INSIDE A
TWO LITER VOLRATH CAN
ALL OTHERS TWO LITER WIDE MOUTH POLY BOTTLE

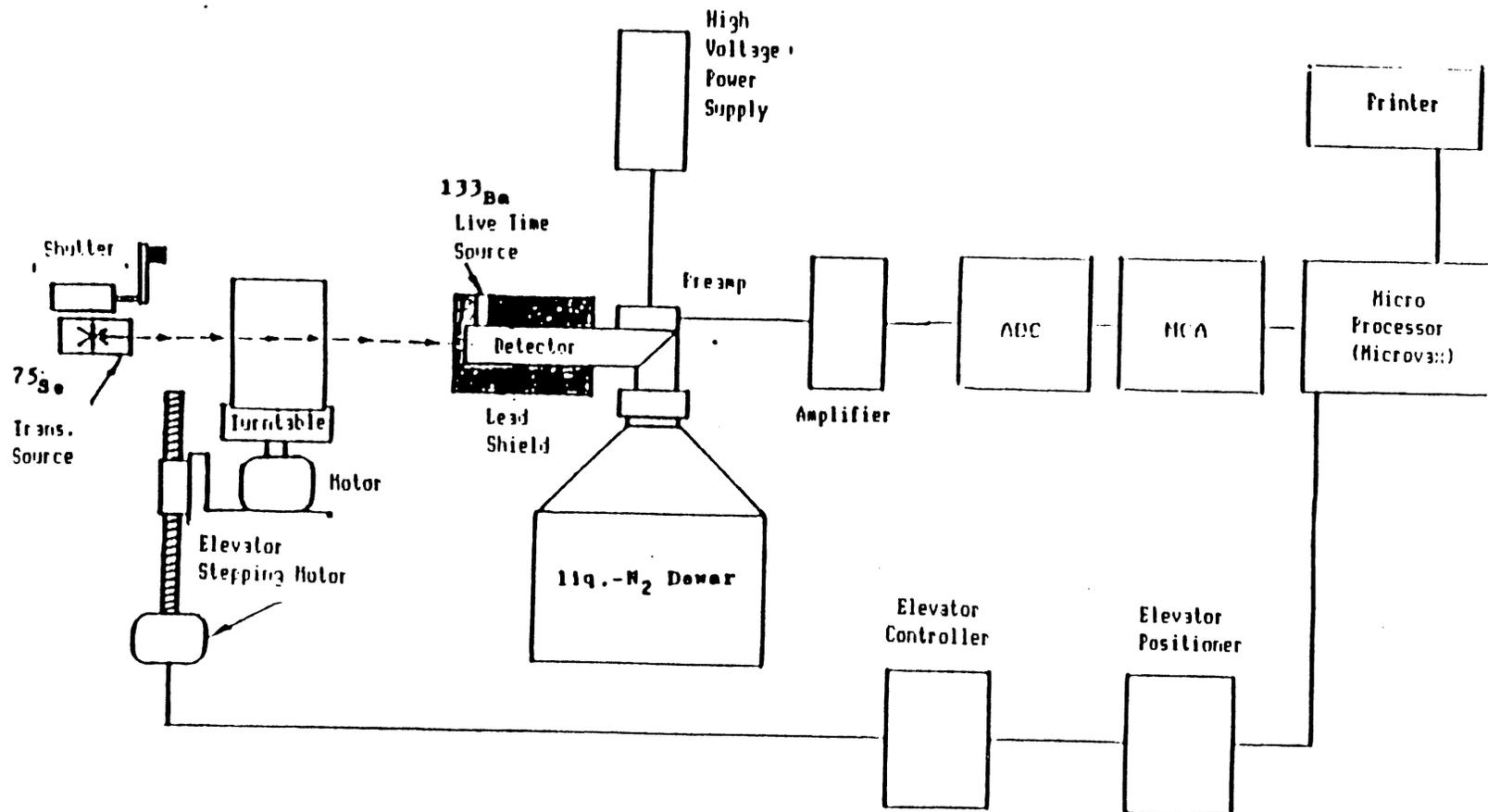
Calibration Limits	IDC 310	250 GRAMS PU
	IDC 312	125 GRAMS PU
	IDC 368	150 GRAMS PU
	IDC 387,392	400 GRAMS PU
	IDC 398	400 GRAMS PU
	IDC 409	520 GRAMS PU, 40 GRAMS AM
	IDC 420	550 GRAMS PU

Assay Time: APPROXIMATELY 20 MINUTES BUT WILL DEPEND ON
THE NUMBER OF SEGMENTS.

Future Plans for Improvements: COMMUNICATIONS VIA THE LAN TO PERMIT
SYSTEM MONITORING OF WORKING
STANDARDS AND MEASUREMENT CONTROL
STANDARD DATA TRANSMISSION. WILL
ALSO BE CALIBRATED FOR ADDITIONAL
IDC'S AFTER THE NEW MICROPROCESSOR
IS INSTALLED.

HIGH RESOLUTION GAMMA CAN SCAN--DOUBLE PASS

7715GSCS02



Date: 29-Oct 1987

New Name: 771SGSCS02 OLD NAME: NONE
Location: BVUILDING 771 Room: 147C Phone: 2601
Yr INSTALLED 1987

Samples:
Type (IDC): 368,392,398,409,420
OTHERS TO BE ADDED LATTER AS STANDARDS
BECOME AVAILABLE

Container Size: ONE LITER VOLRATH CAN INSIDE A
TWO LITER VOLRATH CAN
OR A TWO LITER WIDE MOUTH POLY BOTTLE

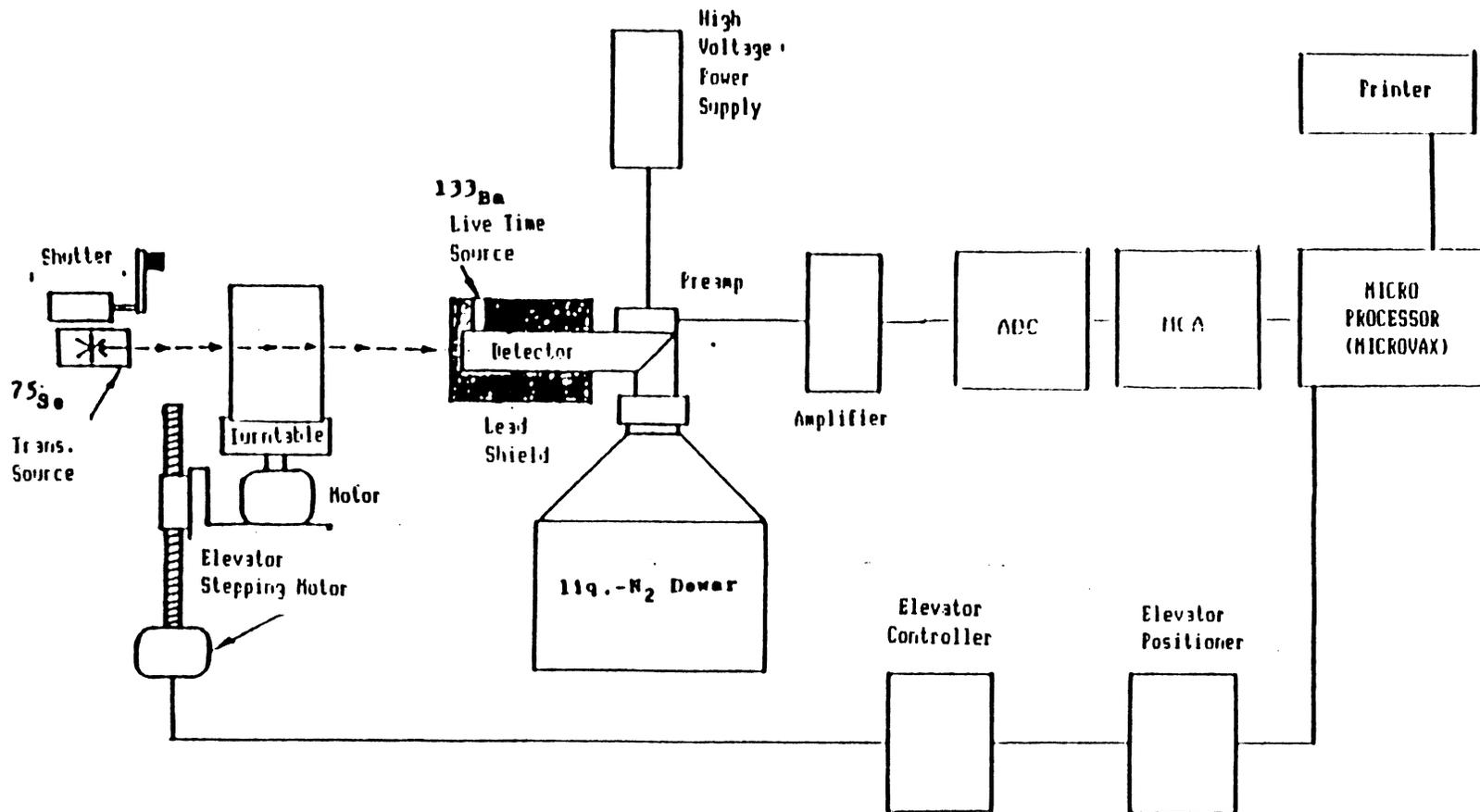
Calibration Limits:	IDC 368	150 GRAMS PU
	IDC 392	400 GRAMS PU
	IDC 398	400 GRAMS PU
	IDC 409	520 GRAMS PU
		40 GRAMS AM
	IDC 420	550 GRAMS PU

Assay Time: APPROXIMATELY 35 MINUTES

Future Plans for Improvements: COMMUNICATIONS VIA THE LAN TO
PERMIT SYSTEM MONITORING OF WORKING
STANDARDS AND MEASUREMENT CONTROL
STANDARD DATA TRANSMISSION.

HIGH RESOLUTION GAMMA CAN SCAN--DOUBLE PASS

771SG 503



Date: 29-Oct 1987

New Name: 771SGSCS03 Old Name: NONE
Location: BUILDING 771 Room: 147C Phone: 2601
Yr Installed: 1987

Type (IDC): 368,392,398,409,420
OTHERS TO BE ADDED LATTER AS STANDARDS
BECOME AVAILABLE

Container Size: ONE LITER VOLRATH CAN INSIDE A
TWO LITER VOLRATH CAN
OR A TWO LITER WIDE MOUTH POLY BOTTLE

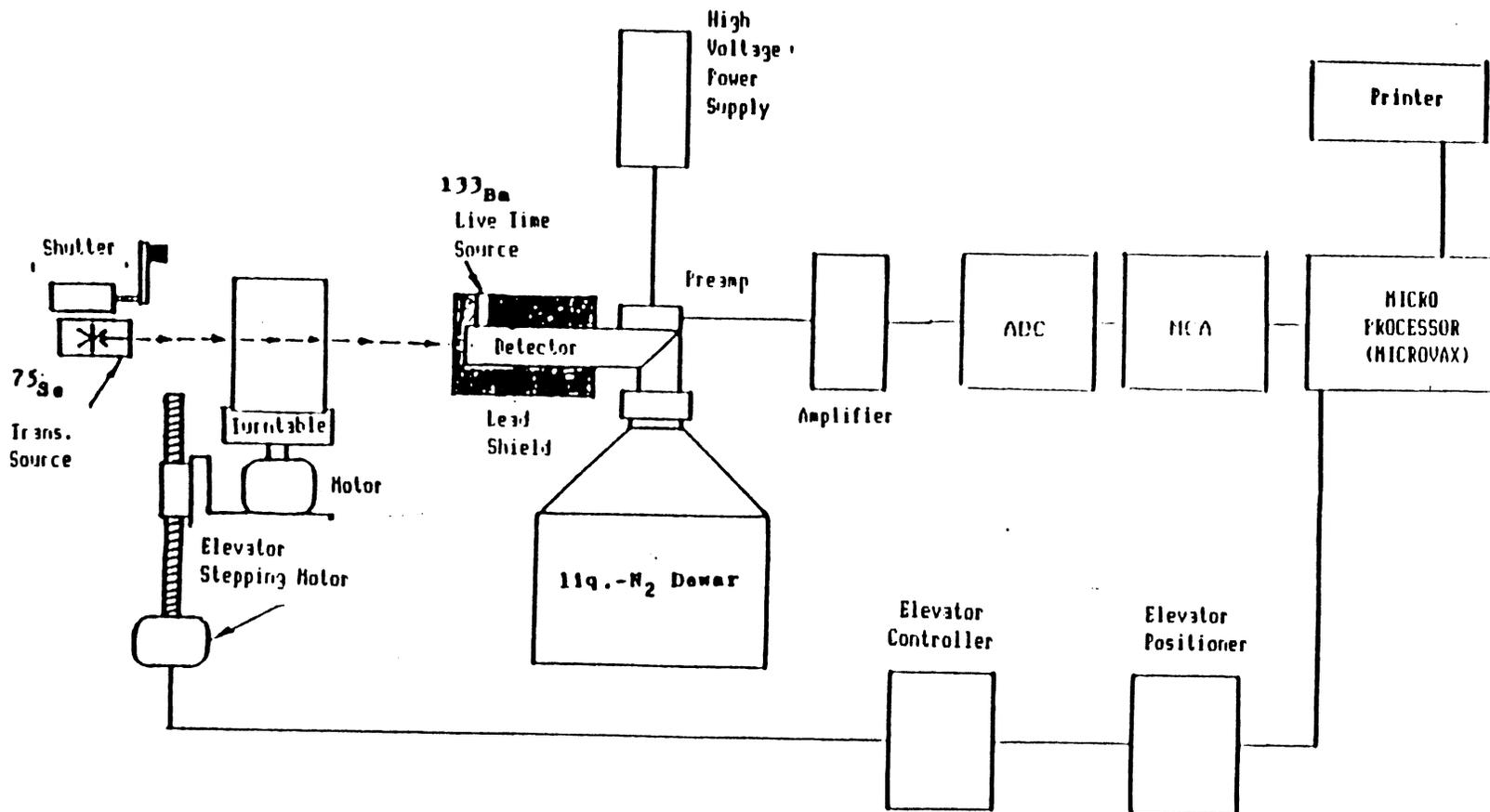
Calibration Limits:	IDC 368	150 GRAMS PU
	IDC 392	400 GRAMS PU
	IDC 398	400 GRAMS PU
	IDC 409	520 GRAMS PU
		40 GRAMS AM
	IDC 420	550 GRAMS PU

Assay Time: APPROXIMATELY 35 MINUTES

Future Plans for Improvements: COMMUNICATIONS VIA THE LAN TO
PERMIT SYSTEM MONITORING OF WORKING
STANDARD AND MEASUREMENT CONTROL
STANDARD DATA TRANSMISSION.

HIGH RESOLUTION GAMMA CAN SCAN---DOUBLE PASS

771SG5CS04



Date: 10-Nov 1987

New Name: 771SGSCS04 Old Name: NONE
Location: BUILDING 771 Room: 147C Phone: 2601
Yr Installed: 1987

Samples:

Type (IDC): 368,392,398,409,420
OTHERS TO BE ADDED LATER AS STANDARDS
BECOME AVAILABLE

Container Size: ONE LITER VOLRATH CAN INSIDE A
TWO LITER VOLRATH CAN

OR A TWO LITER WIDE MOUTH POLY BOTTLE

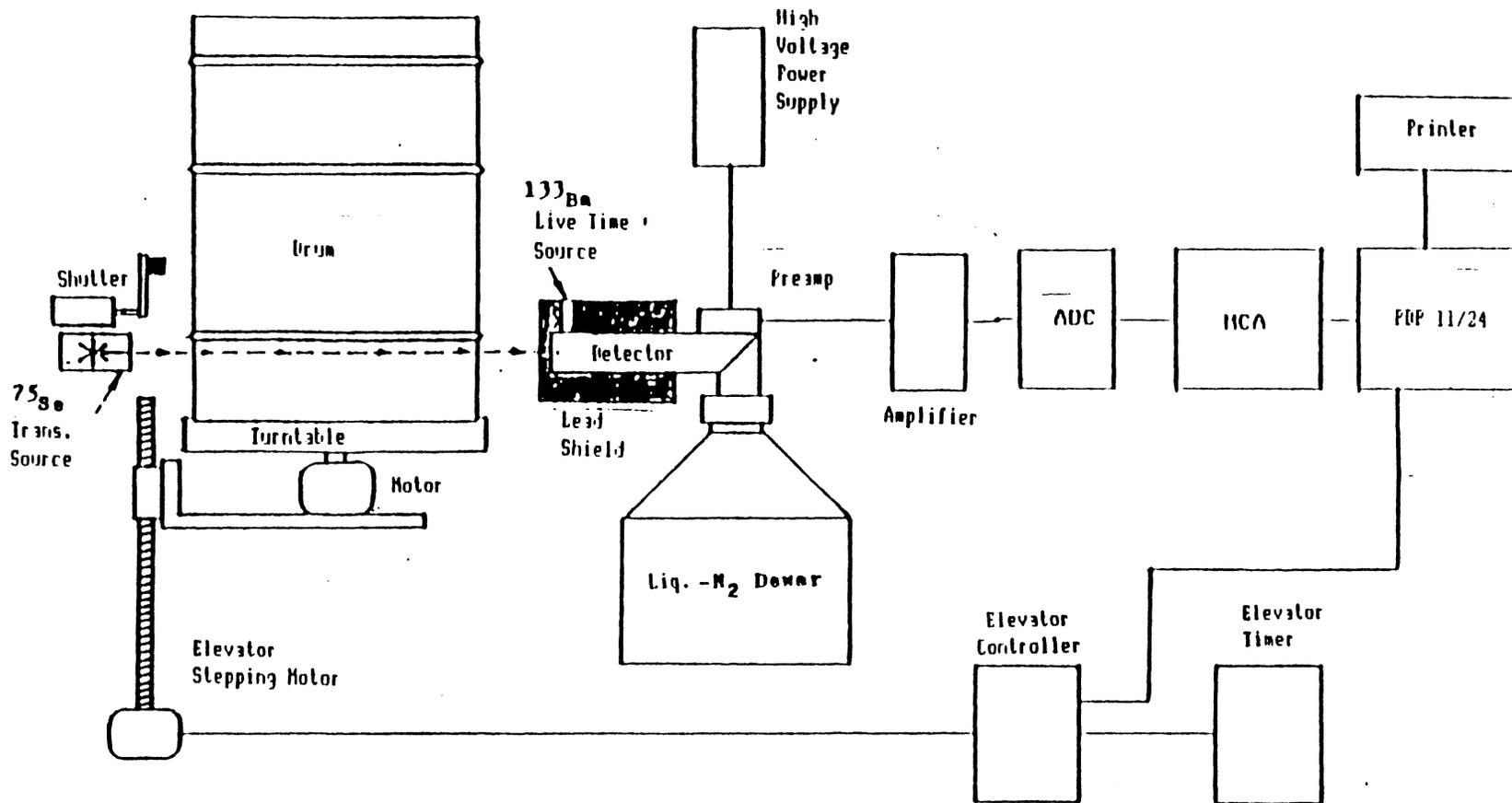
Calibration Limits:	IDC 368	150 GRAMS PU
	IDC 392	400 GRAMS PU
	IDC 398	400 GRAMS PU
	IDC 409	520 GRAMS PU
		40 GRAMS AM
	IDC 420	550 GRAMS PU

Assay Time: APPROXIMATELY 35 MINUTES

Future Plans for Improvements: COMMUNICATIONS VIA THE LAN TO
PERMIT SYSTEM MONITORING OF WORKING
STANDARD AND MEASUREMENT CONTROL
STANDARD DATA TRANSMISSION. THIS
SYSTEM WILL ALSO BE DESIGNATED FOR
MATRIX SELF ABSORPTION AND EU/PU
WORK.

HIGH RESOLUTION GAMMA DRUM SCAN--SINGLE PASS

7715GSDC01



Date: 21-Dec 1987

New Name: 771SGSDC01 Old Name: 771 SEGMENTED DRUM COUNTER
Location: BUILDING 771 Room: 301 (ANNEX) Phone: 2939
Yr Installed: 1983

Samples:

Type (IDC): 290,292,300,301,302,303,312,320,321,328
330,331,334,335,336,337,338,371,374,377,
378,379,411,425,429,430,431,432,440,441
442,479,485,487,490,491,806

320,339,341,480,481,481,484,486 FROM
ACCOUNTS 23-1576-76, 74-1371-33, 45-1374-31,
AND 55-1374-79 ONLY

Container Size: 55 GALLON DRUM

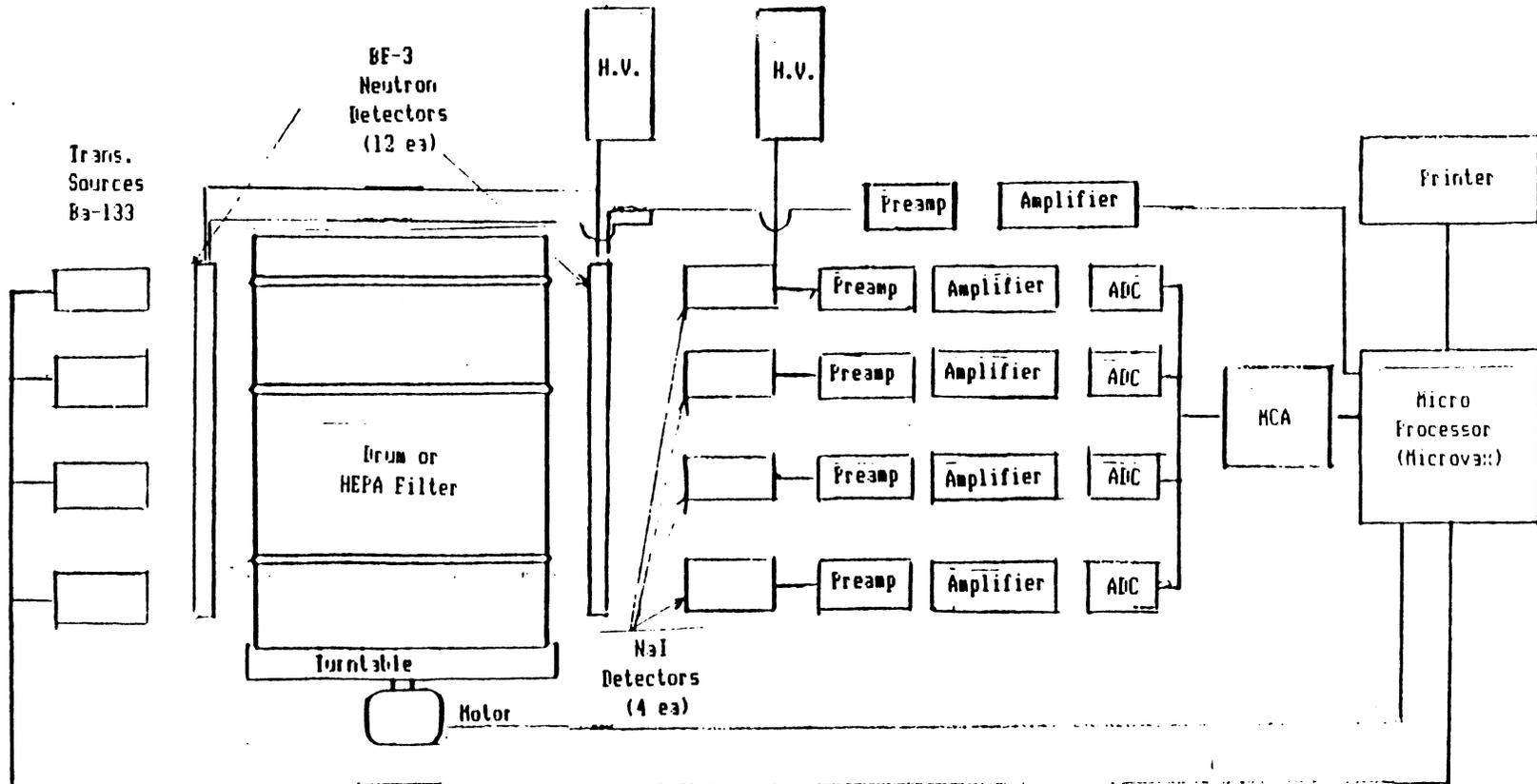
Calibration Limits: 500 GRAMS PU; 20 GRAMS AM
60 GRAMS U

Assay Time: APPROXIMATELY 20 MINUTES

Future Plans for Improvements: COMMUNICATIONS VIA THE LAN TO PERMIT
SYSTEM MONITORING OF THE WORKING
STANDARD AND MEASUREMENT CONTROL
STANDARD DATA TRANSMISSION. IT
WILL ALSO HAVE A NEW MICROPROCESSOR
INSTALLED.

LOW RESOLUTION GAMMA DRUM COUNTER PLUS NEUTRON ASSAY CAPABILITY

77156CDC01



Date: 11-Nov 1987

New Name: 771SGCDC01 Old Name: SOUTH DRUM COUNTER
Location: BUILDING 771 Room: 304 Phone: 2939
Yr Installed: 1972

Samples:

Type (IDC): 292,300,301,302,328,330,331,334,335,336
337,338,370,371,372,374,375,376,377,378
425,430,431,432,440,441,442,485,490

Container Size: IDC 490 2' X 2' X 1' CARDBOARD BOX
ALL OTHERS 55 GALLON DRUM

Calibration Limits: IDC 490 20 GRAMS
ALL OTHERS 100 GRAMS FOR NOW
MAY BE INCREASED AFTER
SYSTEM IS UPGRADED

Assay Time: 100 SECONDS

Future Plans for Improvements: THE SYSTEM WILL BE UPGRADED
WITH TRANSMISSION CORRECTION
CAPABILITY FOR GAMMA MEASUREMENTS
AND HIGH COUNT RATE NEUTRON
CAPACITY.

Date: 11-19-87

New Name: 771HEPAI01 Old Name: NONE
Location: BUILDING 771 Room: 301 Phone: 2939
Yr Installed: 1988

Samples:

Type (IDC): THIS INSTRUMENT IS NOT YET ON LINE. IT SHOULD BE OPERATIONAL IN LATE JANUARY 1988 APPROVED IDC'S WILL BE 490 AT FIRST FOLLOWED BY 330,336, AND 337.

Container Size: IDC 490 2' X 2' X 1' CARDBOARD BOX
ALL OTHERS 55 GALLON DRUM

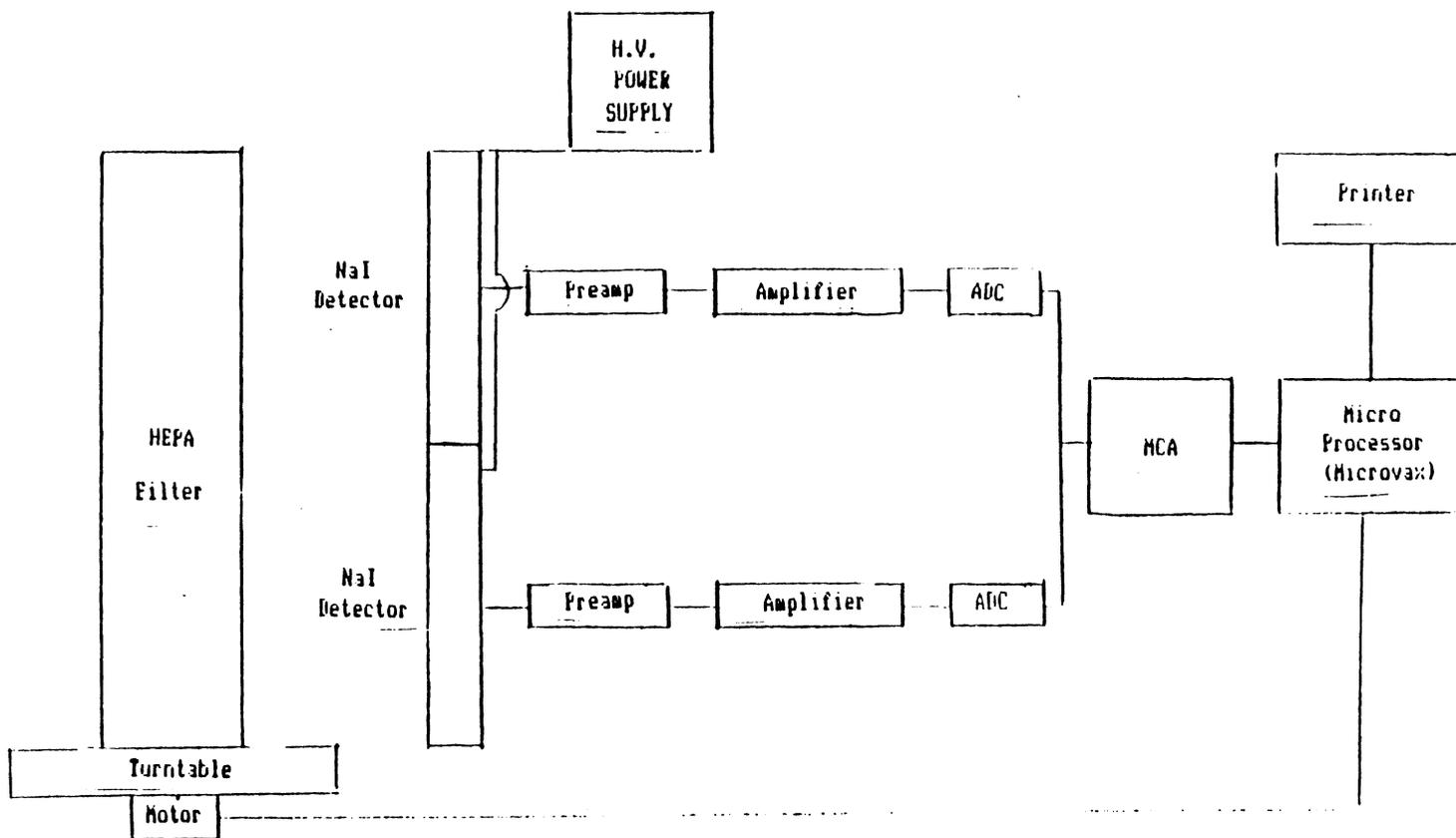
Calibration Limits: NOT YET DETERMINED

Assay Time: APPROXIMATELY 100 SECONDS

Future Plans for Improvements: COMMUNICATIONS VIA THE LAN TO PERMIT SYSTEM MONITORING OF WORKING STANDARDS AND MEASUREMENT CONTROL STANDARD DATA TRANSMISSION.

LOW RESOLUTION GAMMA HEPA FILTER COUNTER (LLW/TRU SORTING)

771HEPA101



PASSIVE/ACTIVE NEUTRON DRUM COUNTER (LLW/TRU SORTING)

771-ADC101

NO DRAWINGS AVAILABLE

Date: 11-19-87

New Name: 771PADCI01 Old Name: NONE
Location: BUILDING 771 Room: 301 Phone: 2939
Yr Installed: 1987

Samples:

Type (IDC): THIS INSTRUMENT IS NOT YET ON LINE. IT
PROBABLY WILL NOT BE IN SERVICE UNTIL LATE
1988. WHEN IT DOES BECOME OPERATIONAL IT
SHOULD MEASURE THE SAME IDC'S AS 371PADCI01.

Container Size: 55 GALLON DRUM

Calibration Limits: NOT YET DETERMINED

Assay Time: NOT YET DETERMINED

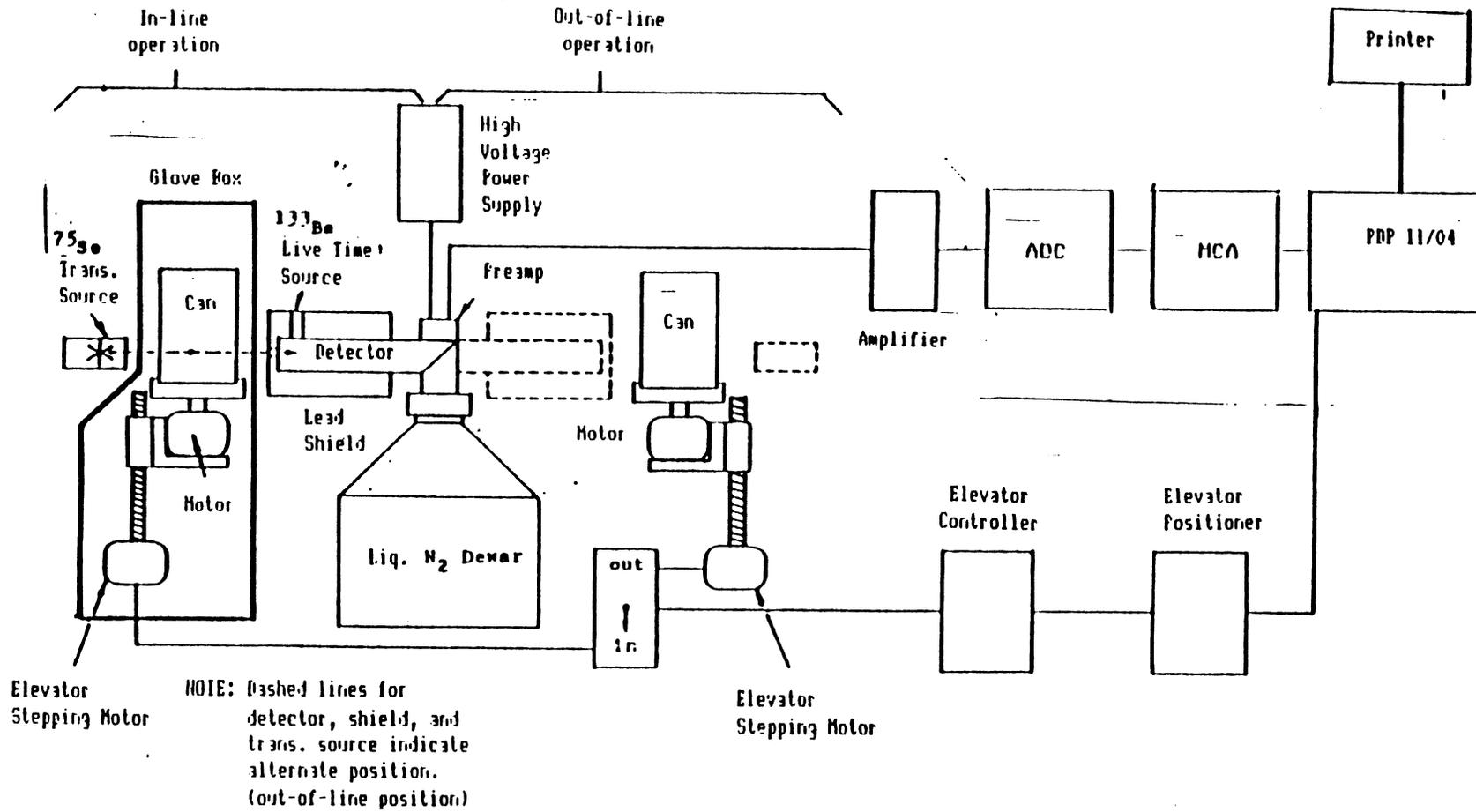
Future Plans for Improvements: PASSIVE NEUTRON COUNTING FIRST
TO BE FOLLOWED BY ACTIVE
INTERROGATION CAPABILITY.

IMPROVED COMMUNICATIONS VIA THE LAN

HIGH RESOLUTION GAMMA CAN SCAN—SINGLE PASS (OUT-OF-LINE)

77653CS01

NOTE: Shutter and actuator mechanism move with transmission source but have been omitted from this drawing



Date: 10-Nov 1987

New Name: 776SGSCS01 Old Name: CAN SCAN II OUT LINE
Location: BUILDING 776 Room: 154A Phone: 2076
Yr Installed: 1982

1987
Type(IDC) 340,368,405,409,411,414,654

Container Size:

IDC 409,411,414 ONE LITER VOLRATH CAN INSIDE
A TWO LITER VOLRATH CAN
IDC 368 HALF GALLON POLY BOTTLE
INSIDE A CLAMSHELL

Calibration Limits:

IDC 368	150 GRAMS PU, 10 GRAMS AM
IDC 409	500 GRAMS PU, 36 GRAMS AM
IDC 411	500 GRAMS PU, 36 GRAMS AM
IDC 414	500 GRAMS PU, 36 GRAMS AM

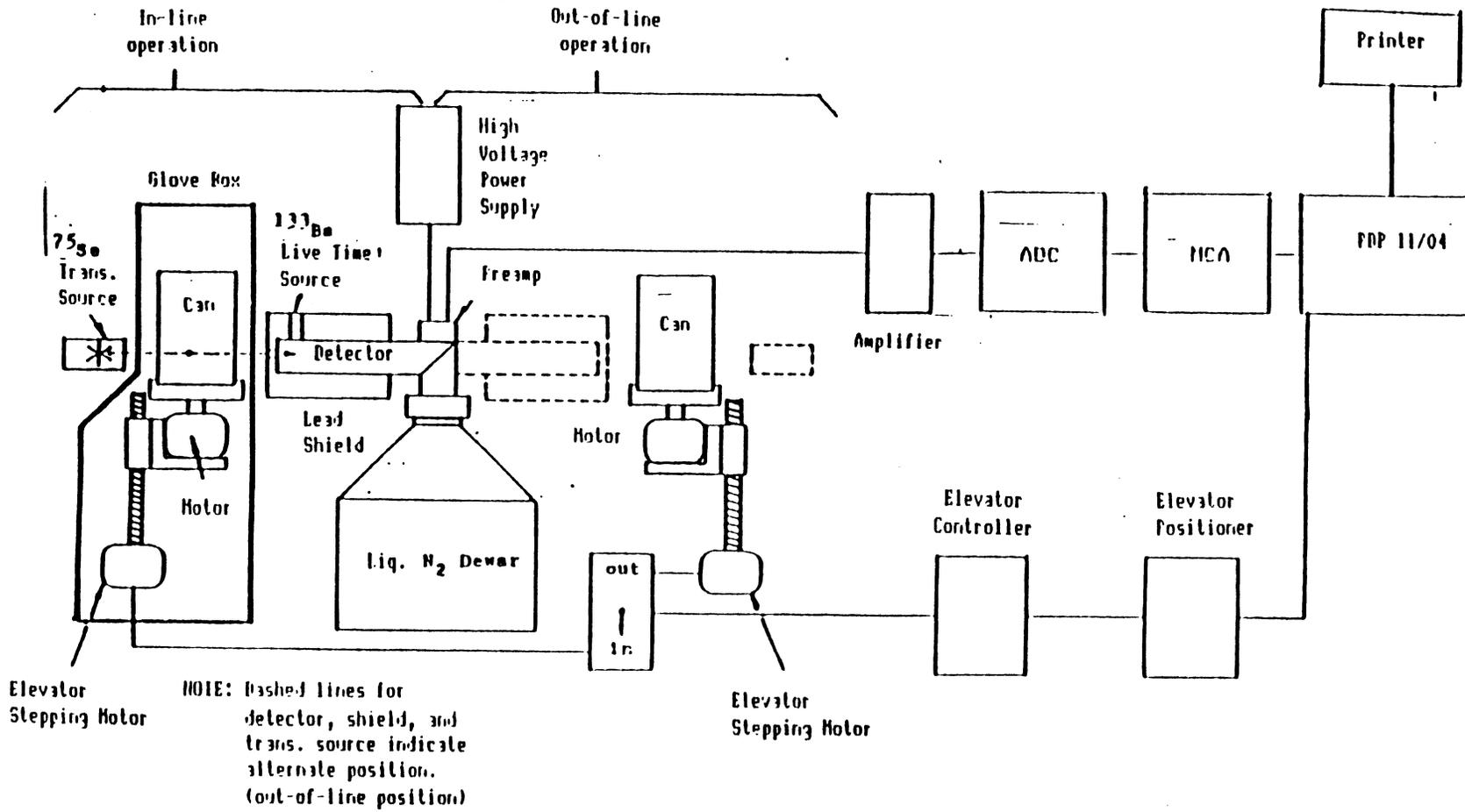
Assay Time: APPROXIMATELY 20 MINUTES

Future Plans for Improvements: UPGRADE OF MICRO PROCESSOR
TO PROVIDE FOR IMPROVED
COMMUNICATIONS VIA THE LAN

HIGH RESOLUTION GAMMA CAN SCAN--SINGLE PASS (IN-LINE)

776SGSCS02

NOTE: Shutter and actuator mechanism move with transmission source but have been omitted from this drawing



Date: 11-Nov 1987

New Name: 776SGSCS02 OLD NAME: CAN SCAN II IN-LINE
Location: BUILDING 776 Room: 154A Phone: 2076
Yr INSTALLED 1982

Samples:
Type (IDC): 368,409,413,414,429

Container Size:	IDC 368	HALF GALLON POLY BOTTLE
	IDC 409	ONE LITER VOLRATH CAN
	IDC 413	ONE LITER VOLRATH CAN
	IDC 414	ONE LITER VOLRATH CAN
	IDC 429	ONE LITER VOLRATH CAN
	IDC 414	SALT BRICK INSIDE A PLASTIC BAG

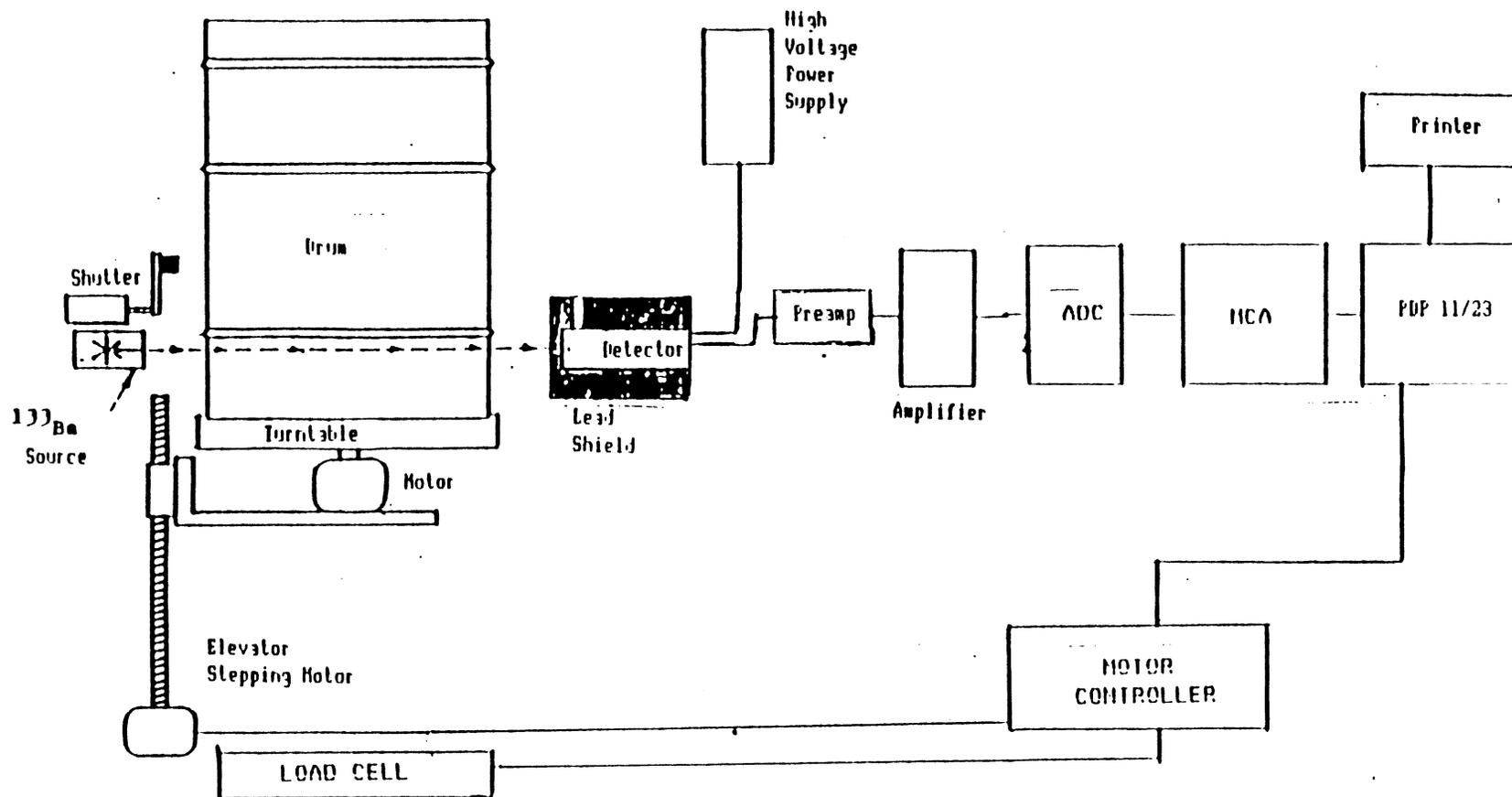
Calibration Limits:	IDC 368	150 GRAMS PU
	IDC 409	500 GRAMS PU, 36 GRAMS AM
	IDC 413	500 GRAMS PU, 36 GRAMS AM
	IDC 414	500 GRAMS PU, 36 GRAMS AM
	IDC 429	500 GRAMS PU, 36 GRAMS AM

Assay Time: APPROXIMATELY 20 MINUTES

Future Plans for Improvements: UPGRADE OF MICRO PROCESSOR SYSTEM
TO PROVIDE FOR IMPROVED
COMMUNICATIONS VIA THE LAN.

LOW RESOLUTION GAMMA DRUM SCAN (LLW/TRU SORTING)

776LO 201



Date: 29-Oct 1987

New Name: 776LOSAC01 Old Name: LOSAC DRUM COUNTER
Location: BUILDING 776 Room: 132 Phone: NONE
Yr Installed: 1984

Samples:
Type (IDC): 330,336,337,491

Container Size: 55 GALLON DRUM

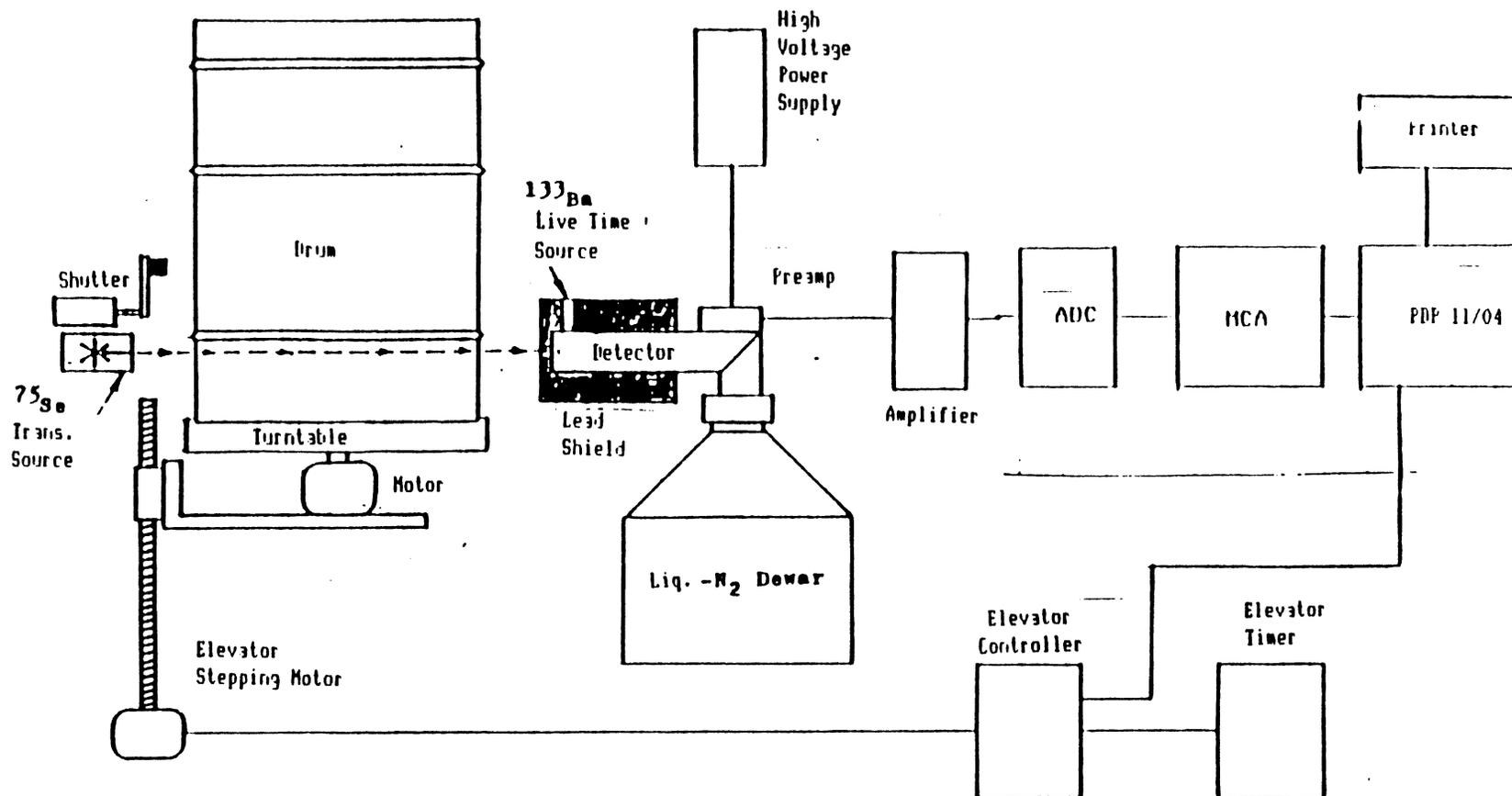
Calibration Limits:	IDC	330,336,337	20 GRAMS
	IDC	491	5 GRAMS

Assay Time: APPROXIMATELY 20 MINUTES

Future Plans for Improvements: UPGRADE OF MICROPROCESSOR
TO MICROVAX I TO PERMIT
SCREEN MENUS AND IMPROVED
COMMUNICATIONS VIA THE LAN
TO PROVIDE A DATA LINK WITH
THE SWIMS SYSTEM.

HIGH RESOLUTION GAMMA DRUM SCAN--SINGLE PASS

707SG-01



Date: 21-Dec 1987

New Name: 707SGSDCO1 Old Name: BLDG 707 SEGMENTED DRUM COUNTER
Location: BUILDING 707 Room: 196 Phone: 2966
Yr Installed: 1982

Samples:

Type (IDC): 290 292,300,301,302,303,312,320,321,328,
330,331,334,335,336,337,338,371,374,377,
378,379,411,425,429,430,431,432,440,441,
442,479,485,487,490,491,806

320,339,341,480,481,484,486 FROM
ACCOUNTS 23-1576-76, 74-1371-33, 45-1374-31
AND 55-1374-79 ONLY

Container Size: 55 GALLON DRUM

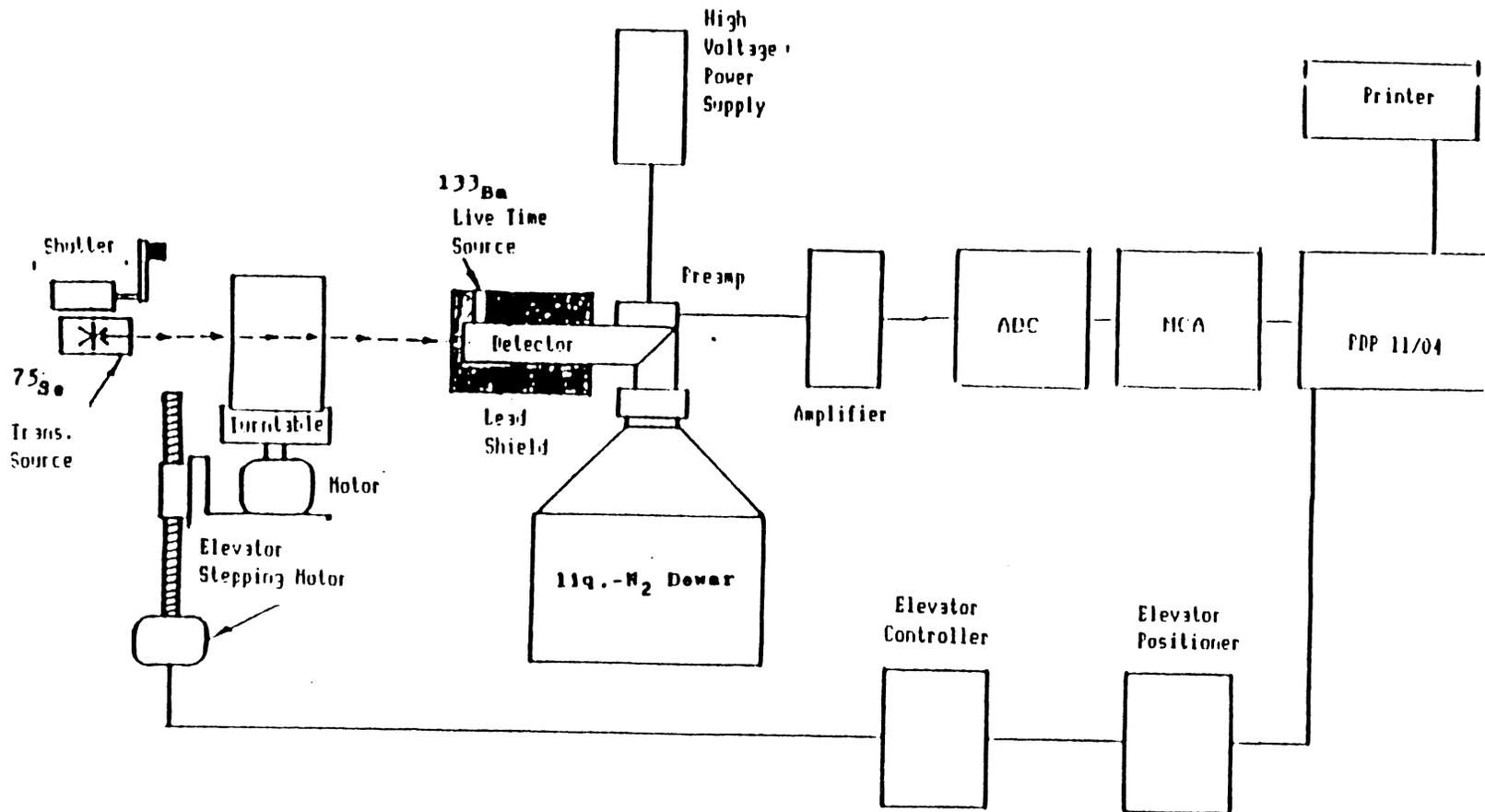
Calibration Limits: 500 GRAMS PU; 20 GRAMS AM;
60 GRAMS U

Assay Time: APPROXIMATELY 20 MINUTES

Future Plans for Improvements: UPGRADE OF MICROPROCESSOR SYSTEM
TO PROVIDE IMPROVED COMMUNICATIONS
VIA THE LAN

HIGH RESOLUTION GAMMA CAN SCANNING SINGLE PASS

371SGSCS04



Date: 11-Nov 1987

New Name: 371SGSCS04 Old Name: 371 CAN SCAN IV
Location: BUILDING 371 Room:3341 Phone: NONE
Yr Installed: 1984

Samples:
Type (IDC): 392,409,411,418,420,424,426,471,473

Container Size: IDC 392,409,420 404 PRODUCE CAN IN A 202
PRODUCE CAN
IDC 411 1 LITER VOLRATH CAN INSIDE
A 2 LITER VOLRATH CAN

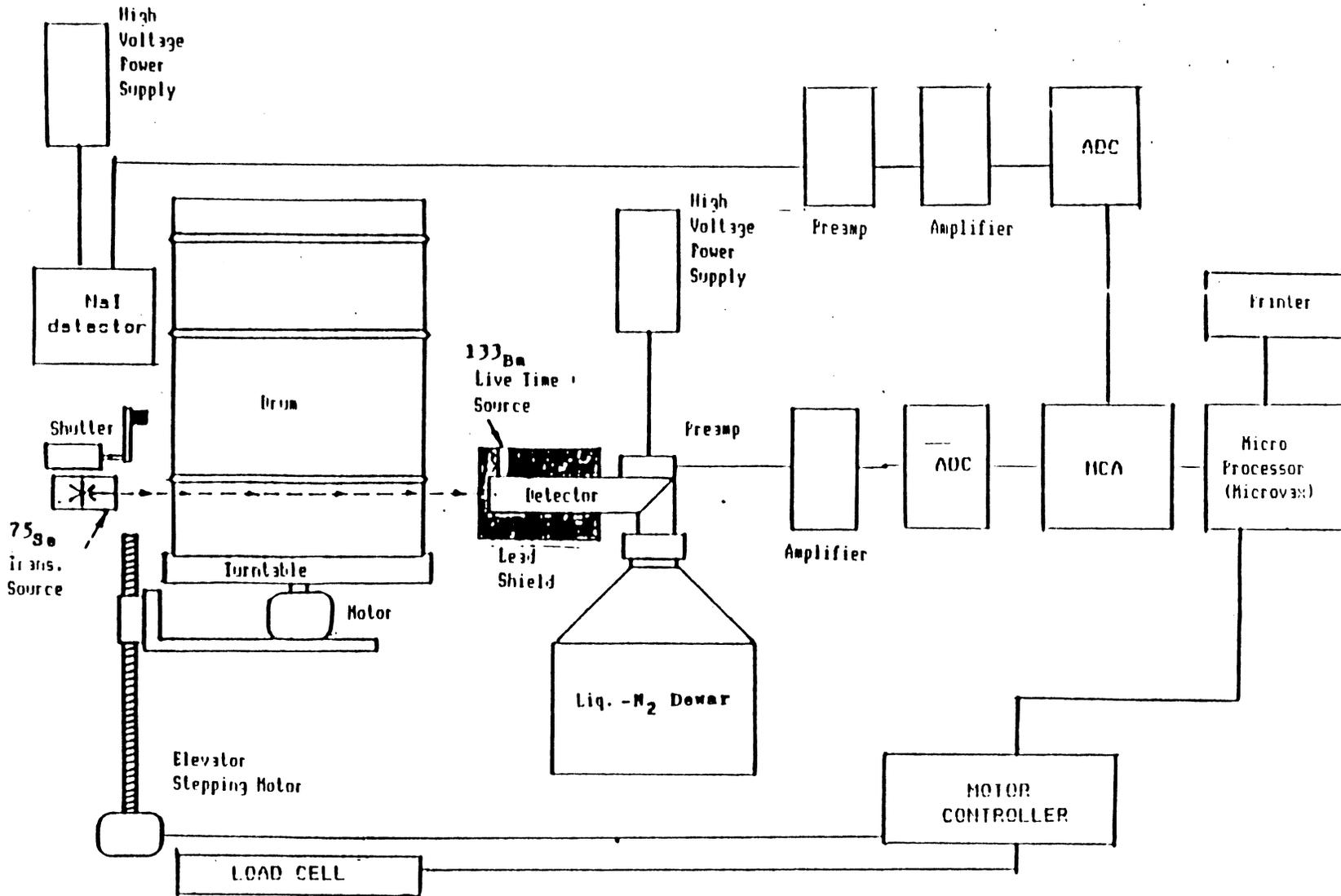
Calibration Limits: IDC 392 300 GRAMS PU
IDC 409 300 GRAMS PU AND
36 GRAMS AM
IDC 411 300 GRAMS PU AND
36 GRAMS AM
IDC 420 400 GRAMS PU

Assay Time: APPROXIMATELY 20 MINUTES

Future Plans for Improvements: REPLACEMENT WITH LINE ITEM
EQUIPMENT AS IT BECOMES
AVAILABLE.

LOW/HIGH RESOLUTION GAMMA DRUM SC¹ DOUBLE PASS (LLW/TRU SORTING)

37156SDC01



Date: 21-Dec 1987

New Name: 371SGSDC01 Old Name: NONE
Location: BUILDING 371 Room: Phone: 7304
Yr Installed: 1987

Samples:

Type (IDC): 290,292,300,301,302,303,312,320,321,328
330,331,334,335,336,337,338,371,374,377
378,379,411,425,429,430,431,432,440,441
442,479,485,487,490,491,806

320,339,341,480,481,484,486 FROM
ACCOUNTS 23-1576-76, 74-1371-33, 45-1374-31
AND 55-1374-79 ONLY

Container Size: 55 GALLON DRUM

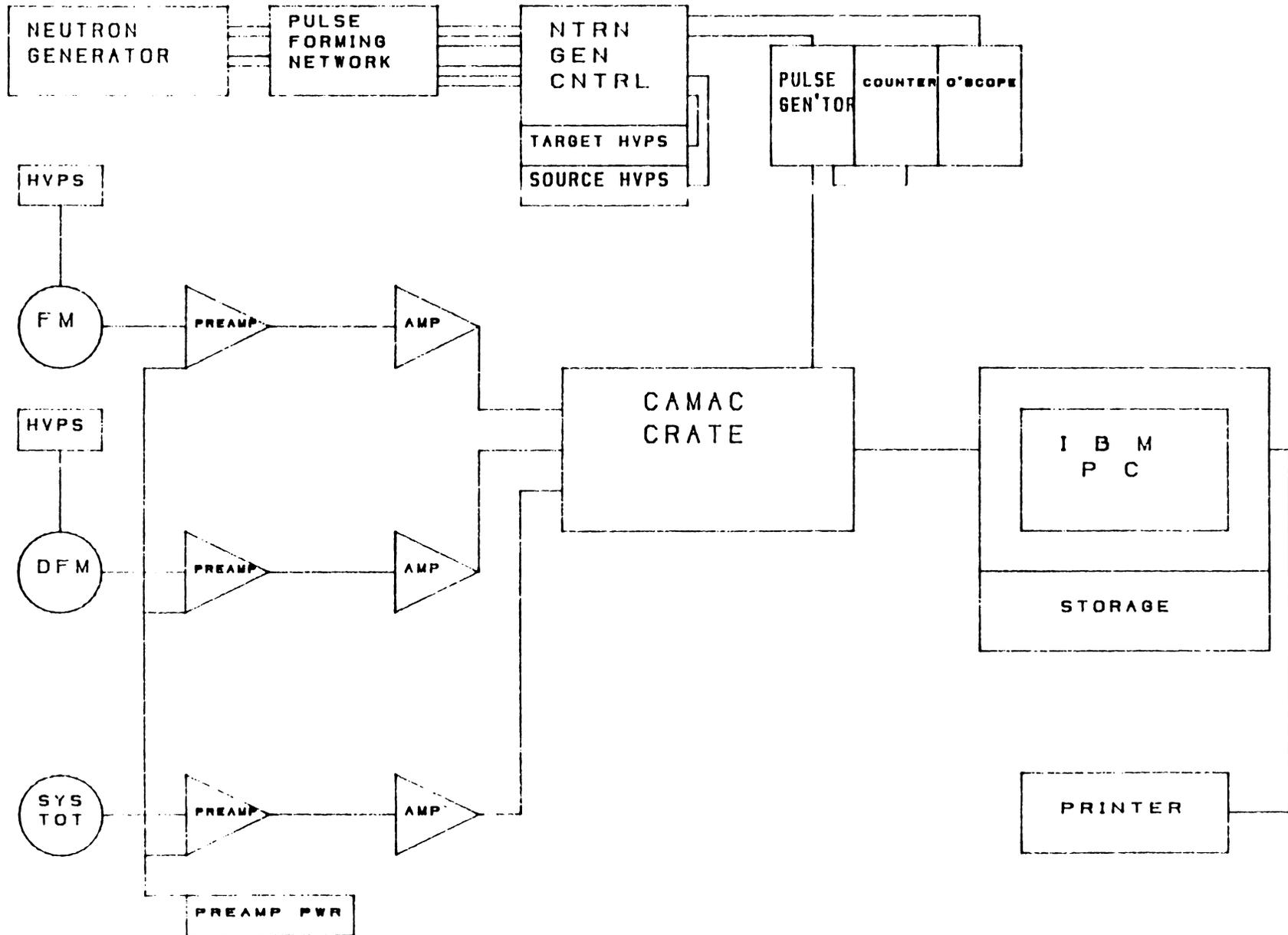
Calibration Limits: 0 TO 500 GRAMS PU

ASSAY TIME: APPROXIMATELY 35 MINUTES

Future Plans for Improvements: IMPROVED COMMUNICATIONS VIA THE LAN

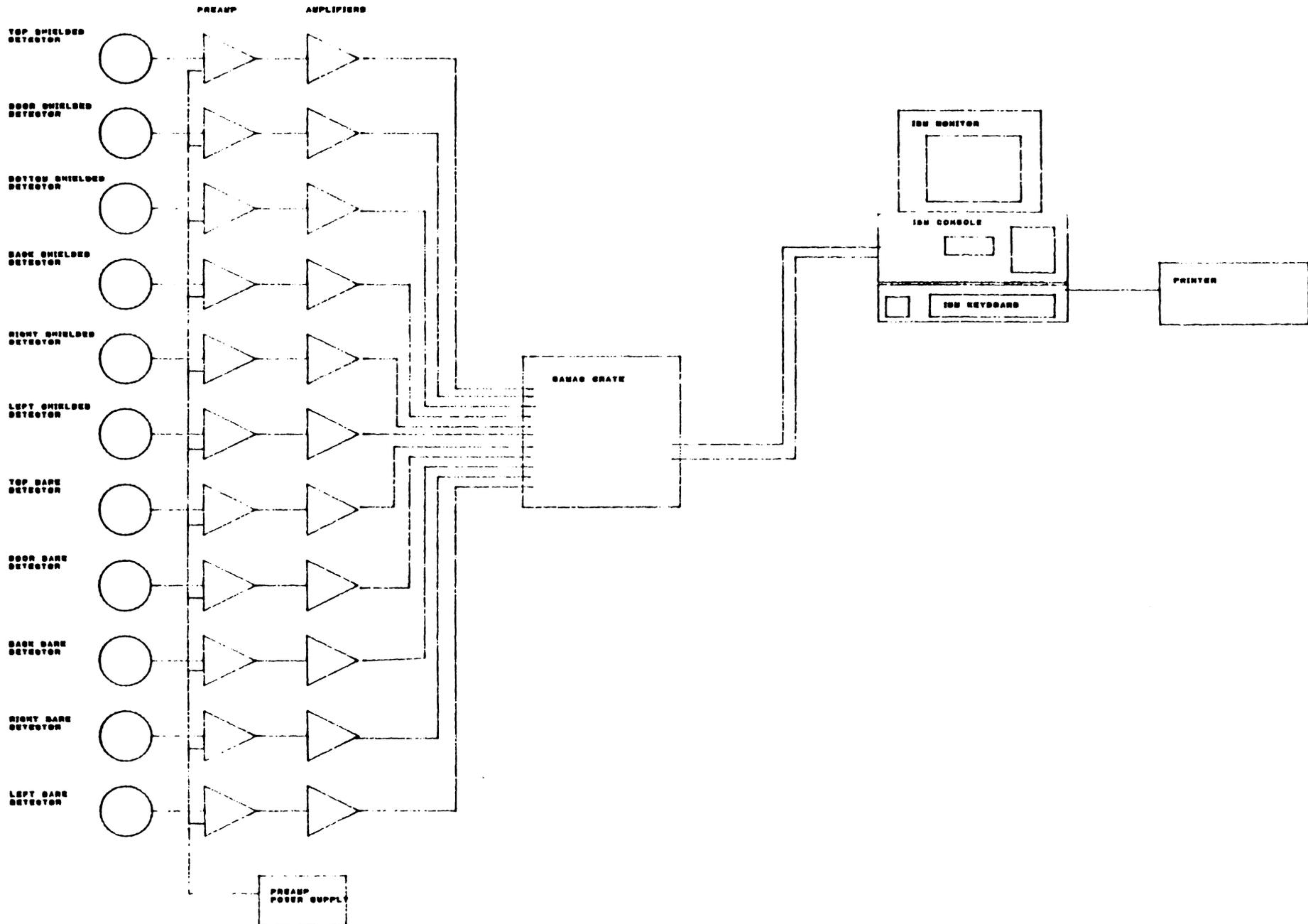
PASSIVE/ACTIVE NEUTRON DRUM C ITER (LLW/TRU SORTING)
ACTIVE ASSAY SYSTEM

371PADCI01



PASSIVE/ACTIVE NEUTRON DRUM COUNTER (LLW/TRU SORTING)
PASSIVE ASSAY SYSTEM

371PAI 01



Date: 21-Dec 1987

New Name: 371PADCI01 Old Name: NONE
Location: BUILDING 371 Room: 2202 Phone: 7465
Yr Installed: 1987

Samples:

Type (IDC): 320,330,336,339,341,480,481,484,486
803,804,807 (SLUDGES)

320,339,341,480,481,484,486 FROM
ACCOUNTS 23-1576-76, 74-1371-33,
45-1374-31, AND 55-1374-79 SHOULD
NOT BE ASSAYED ON THIS COUNTER

Container Size: 55 GALLON DRUM

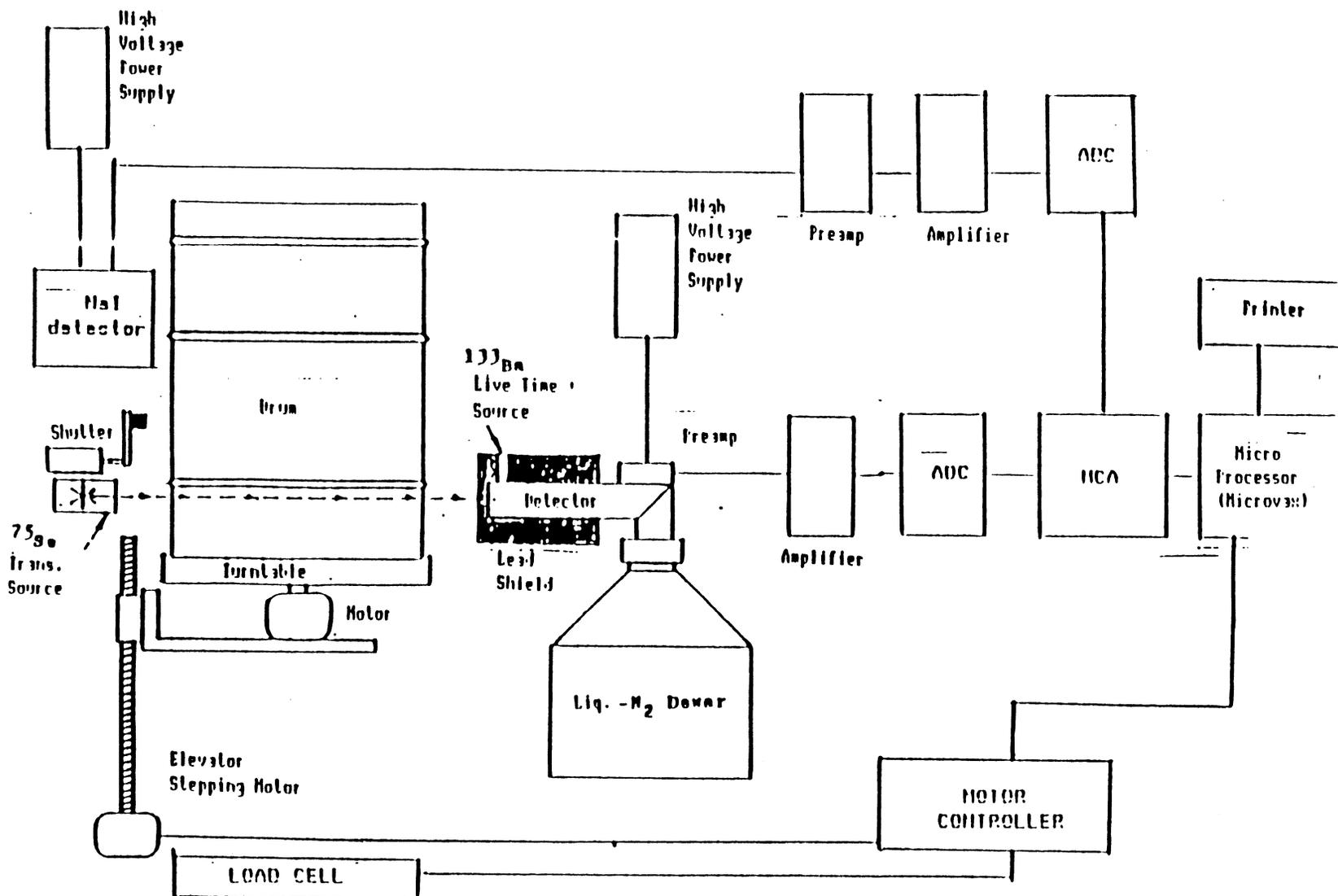
Calibration Limits: 0 TO 300 GRAMS

Assay Time: APPROXIMATELY 10 MINUTES

Future Plans for Improvements: IMPROVED COMMUNICATIONS VIA
THE LAN

LOW/HIGH RESOLUTION GAMMA DRUM SCAN--DOUBLE PASS (LLW/TRU SORTING)

569SGSDC01



Date: 21-Dec 1987

New Name: 56PSGSDCO1 Old Name: NONE
Location: BUILDING 569 Room: Phone:
yr Installed: 1987

Samples:

Type (IDC): 290,292,300,301,302,303,312,320,321,328
330,331,334,335,336,337,338,371,374,377
378,379,411,425,429,430,431,432,440,441
442,479,485,487,490,491,806

320,339,341,480,481,484,486 FROM
ACCOUNTS 23-1576-76, 74-1371-33, 45-1374-31,

AND 55-1374-79 ONLY

Container Size: 55 GALLON DRUM

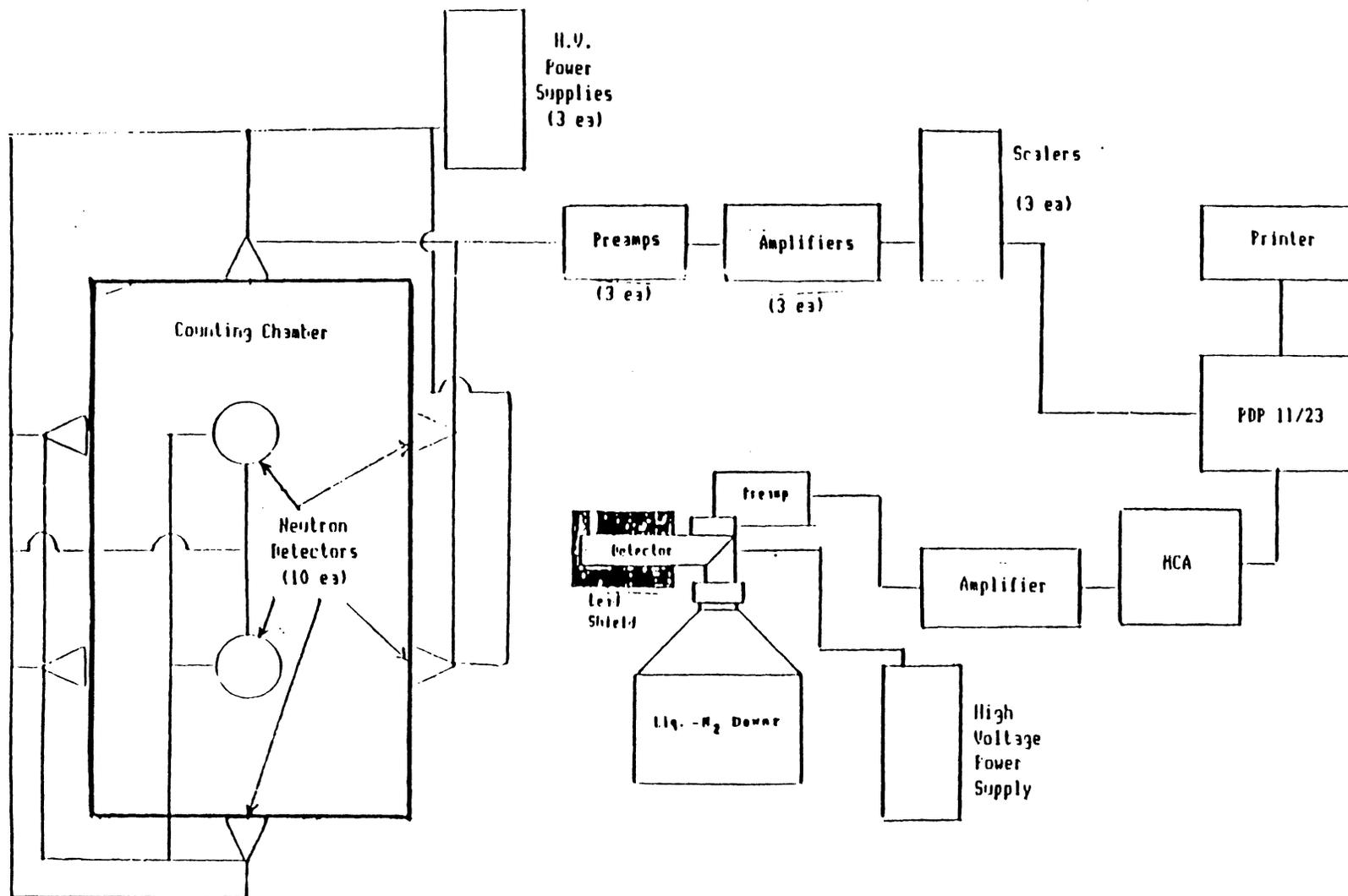
Calibration Limits: 0 TO 500 GRAMS

Assay Time: APPROXIMATELY 35 MINUTES

Future Plans for Improvements: IMPROVED COMMUNICATIONS VIA THE LAN
TO PROVIDE A DATA LINK WITH THE
SWIMS SYSTEM.

HIGH RESOLUTION GAMMA PLUS PASSIVE NEUTRON CRATE COUNTER

5690PCCI01



Date: 11-Nov 1987

New Name: 5690PCCI01 Old Name: CRATE COUNTER I
Location: BUILDING 664 Room: NONE Phone: 7643
Yr Installed: 1977

Samples:
Type(IDC) 003,004,302,330,336,337,338
374,440,441,442,480,481,490

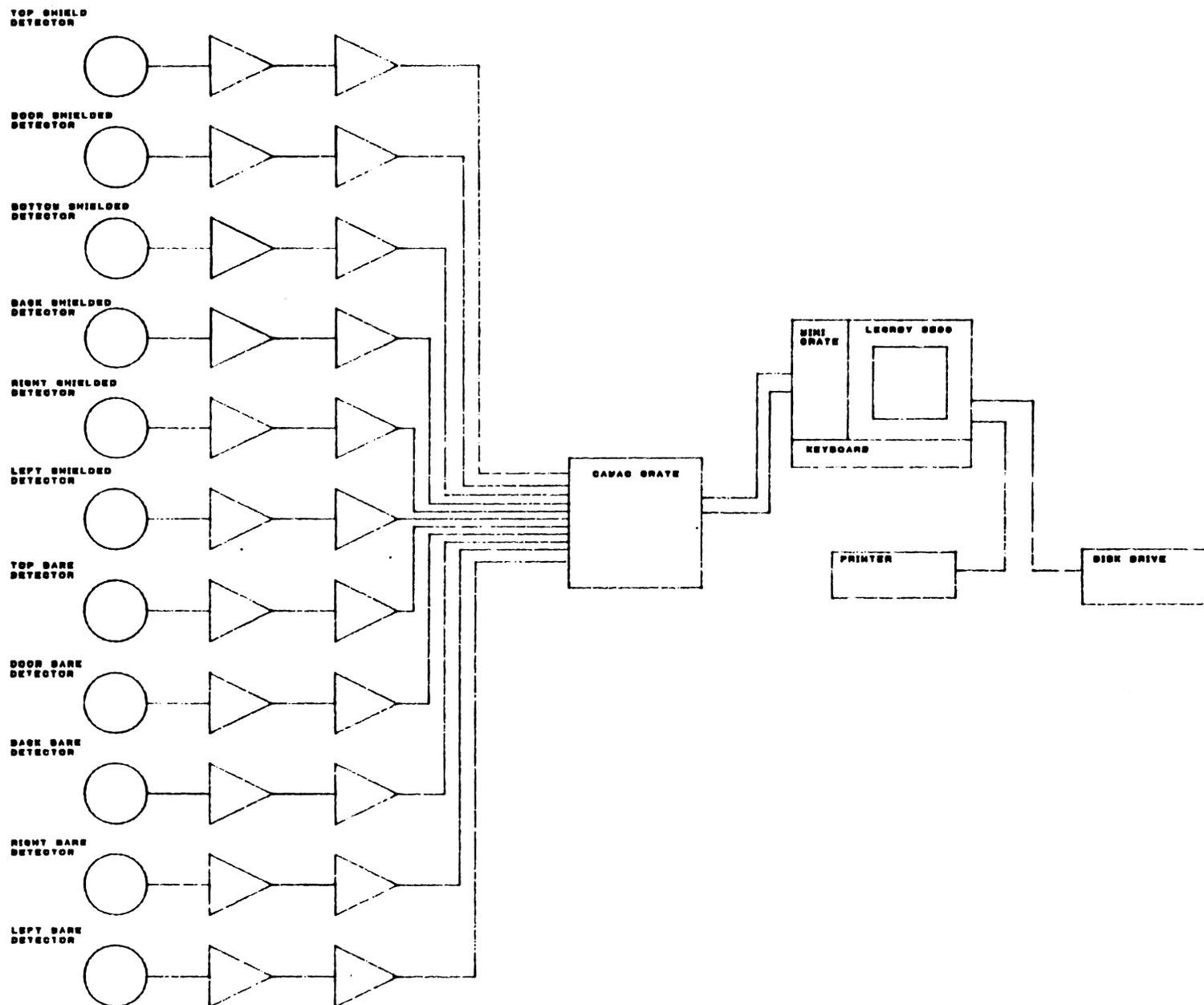
Container Size: 4' X 4' X 7' SHIPPING CRATE

Calibration Limits: 350 GRAMS

Assay Time: APPROXIMATELY 45 MINUTES

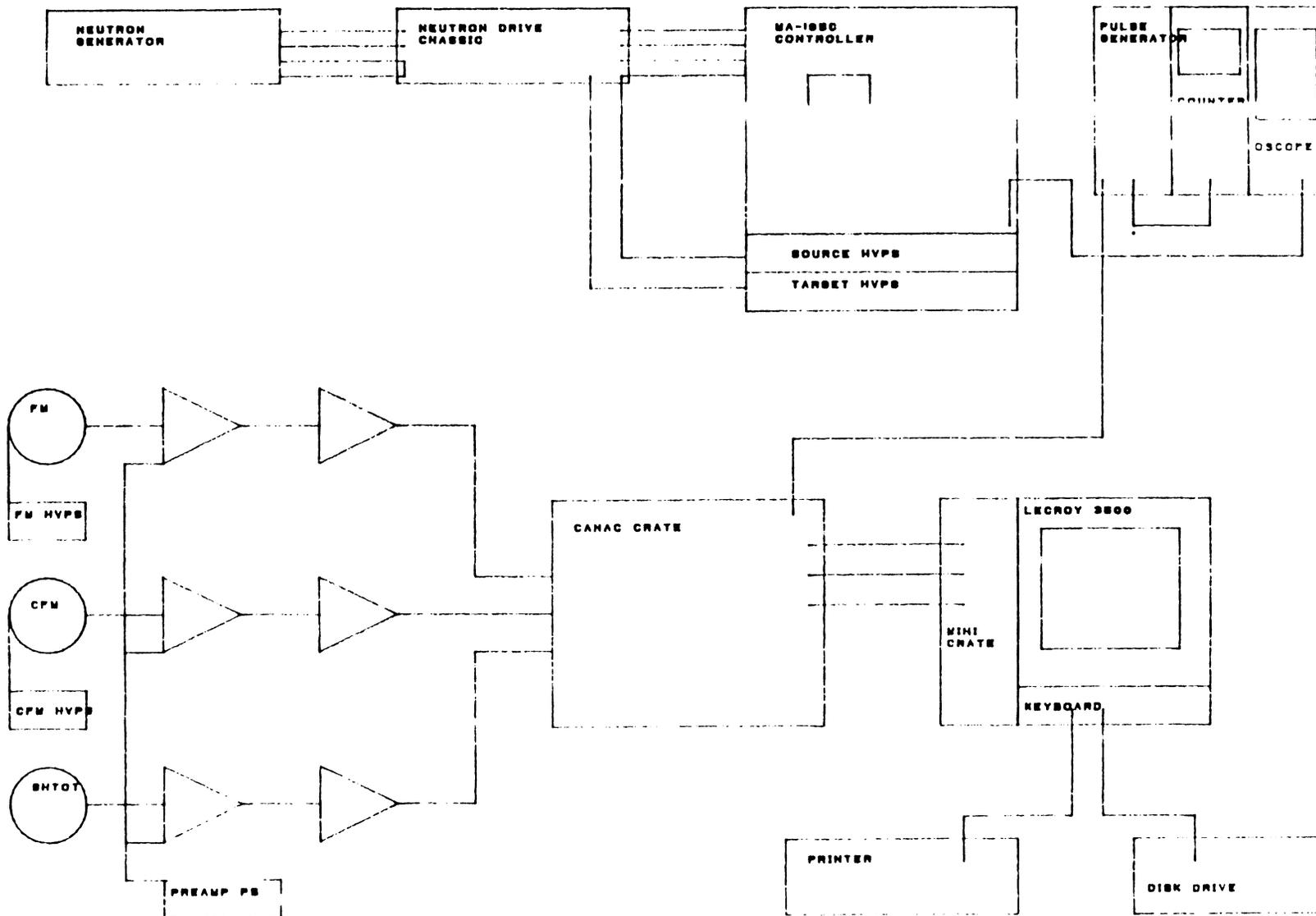
Future Plans for Improvements: IMPROVE THE GAMMA ASSAY SYSTEM BY
BY ADDING A SECOND GAMMA DETECTOR
THEREBY REDUCING THE COUNT TIME
BY 33%. POSSIBLY REBUILD THE NEUTRON
ASSAY SYSTEM. MOVE THE COUNTER
INSIDE THE PSZ TO BUILDING 569.
IMPROVED COMMUNICATIONS VIA THE
LAN TO PROVIDE A DATA LINK WITH
SWIMS SYSTEM.

PASSIVE/ACTIVE NEUTRON CRATE COUNTER (LLW/TRU SORTING)
PASSIVE ASSAY SYSTEM



PASSIVE/ACTIVE NEUTRON CRATE (CENTER (LLW/TRU SORTING)
ACTIVE ASS/ SYSTEM

569 CC101



Date: 29-Oct 1987

New Name: 569PACCI01 Old Name: CRATE COUNTER II
Location: BUILDING 664 Room: NONE Phone: 7643
Yr Installed: 1983

Samples:
Type (IDC): 330,480,481,490
ALL LLW WASTE BOXES

Container Size: 2' X 4' X 7' CRATES
4' X 4' X 7' CRATES

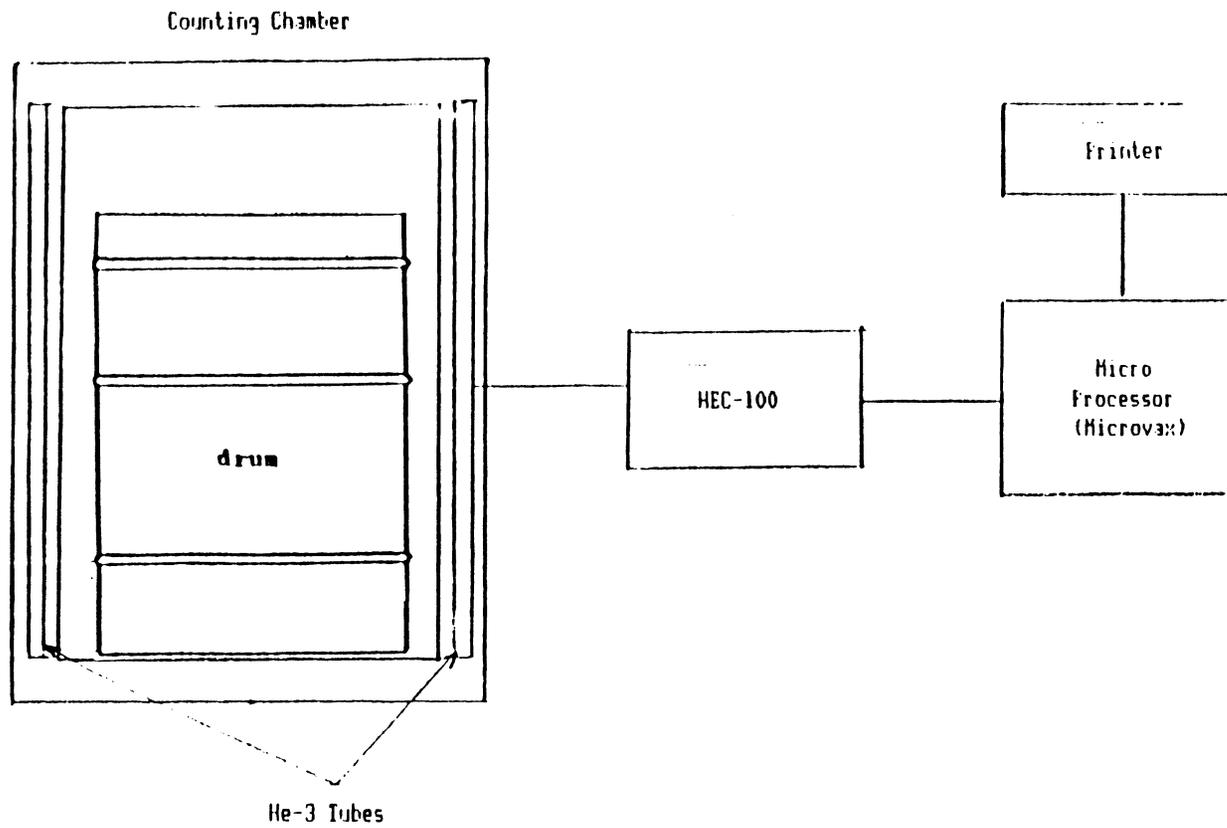
Calibration Limits: IDC 330,480,490 350 GRAMS
IDC 330,480,490 LLW/TRU SORTING
ALL OTHERS SAFEGUARDS CHECK
ONLY

Assay Time: APPROXIMATELY 25 MINUTES

Future Plans for Improvements: CALIBRATE FOR IDC'S 336 AND 337
MOVE THE COUNTER INSIDE THE PSZ TO
BUILDING 569. IMPROVED DISTRIBUTION
OF THE SHIELDED DETECTORS IF FUNDING
IS AVAILABLE. IMPROVED COMMUNICATION
VIA THE LAN TO PROVIDE A DATA LINK
WITH THE SWIMS SYSTEM.

NEUTRON COINCIDENCE COUNTER

991 1CI01



Date: 29-Oct 1987

New Name: 9910CMCI01 Old Name: NONE
Location: BUILDING 991 Room: 152 Phone: 2216
Yr Installed: 1984

Samples:
Type (IDC): VARIOUS

Container Size: 30 GALLON

Calibration Limits: FINGERPRINT MEASUREMENTS ONLY

Assay Time: LESS THAN 5 MINUTES

Future Plans for Improvements: IMPROVED COMMUNICATIONS VIA THE LAN

Date: 24-Nov 1987

New Name: CALORIMETER Old Name: CALORIMETER
Location: BUILDING 771 Room: 152,160 Phone: 2706
Installed: RF DESIGN 1970 (8 cells)
MOUND DESIGN 1987 (10 cells)

Samples:
Type (IDC): ALL

Container Size: RF DESIGN 8802 VOLRATH CAN
MOUND DESIGN 8802 VOLRATH CAN
1 GAL. WIDEMOUTH NALGENE BOTTLE
OR SMALLER
CLAM-SHELL OR SMALLER

Calibration Limits: UPPER LIMIT IS 2500 GRAMS PU
BUT COULD BE LESS DEPENDING
ON CRITICALITY LIMITS FOR A
PARTICULAR PACKAGE AND CONTENTS.

ALL CALORIMETERS ARE CERTIFIED USING
PU-238 HEAT SOURCES. CALIBRATION
RANGE IS CERTIFIED AS .2 TO 6.0
WATTS BUT EXTRAPOLATION BEYOND THIS
RANGE CAN BE MADE IF NECESSARY

Assay Time: ASSAY TIME FOR MOUND CALORIMETERS IS APPROX 6 HOURS
ASSAY TIME FOR ROCKY FLATS CALORIMETERS IS 20-24 HRS.

Plans For Improvements:

FORWARD

INSTRUMENT ASSAY METHODS DESCRIBED IN THIS SECTION INDICATE WHICH INSTRUMENTS CAN BE USED FOR SORTING LOW LEVEL WASTE (LLW) FROM TRANSURANIC WASTE (TRU). THE THRESHOLD FOR THIS SORTING FUNCTION IS 100 NANOCURIES OF SNM PER GRAM OF MATRIX MATERIAL. NDA INSTRUMENTS NOT IDENTIFIED AS BEING USEABLE FOR LLW/TRU SORTING ARE USED FOR TRU ANALYSES ONLY.

Description of NDA method

HIGH RESOLUTION GAMMA SCANNING COUNTERS--SINGLE PASS

FRAGMENTED GAMMA RAY SCANNING IS A NONDESTRUCTIVE MEANS OF ASSAYING SPECIAL NUCLEAR MATERIALS IN A LOW DENSITY WASTE PACKAGED IN CYLINDRICAL CONTAINERS.

HIGH RESOLUTION GAMMA-RAY SPECTROSCOPY IS USED TO DETECT AND MEASURE ISOTOPES OF INTEREST AND MEASURE AND CORRECT FOR AVERAGE GAMMA-RAY ATTENUATION IN DISCRETE VERTICAL SEGMENTS OF THE CONTAINER. CORRECTIONS ARE ALSO MADE FOR COUNTING LOSSES WHICH OCCUR IN THE ELECTRONIC SIGNAL PROCESSING.

THE ASSAY ITEM IS ROTATED AND TRANSLATED ABOUT ITS VERTICAL AXIS AND SCANNED SEGMENT BY SEGMENT ALONG THE AXIS, THEREBY REDUCING THE EFFECTS OF NONUNIFORMITY IN BOTH DENSITY AND NUCLIDE DISTRIBUTIONS.

TWO CONDITIONS MUST BE MADE TO OPTIMIZE THE ASSAY RESULTS. THE FIRST CONDITION IS THAT PARTICLES CONTAINING THE NUCLIDE MUST BE SMALL TO MINIMIZE SELF ABSORPTION OF THE EMITTED GAMMA RADIATION, AND SECOND, THE MIXTURE OF MATERIAL WITHIN A PACKAGE MUST BE REASONABLY UNIFORM IN ORDER TO APPLY AN ATTENUATION CORRECTION FACTOR COMPUTED FROM A MEASUREMENT OF THE TRANSMISSION SOURCE INTENSITY THROUGH THE SEGMENT. MEASURED TRANSMISSION VALUES MUST BE GREATER THAN 0.5% TO PERMIT VALID ATTENUATION CORRECTIONS.

FRAGMENTED GAMMA-RAY ASSAY IS USED TO MEASURE PLUTONIUM, AMERICIUM, AND URANIUM USING THE 414, 662, AND 185 KeV GAMMA-RAYS RESPECTIVELY. THE ASSAY OF THE NUCLIDE OF INTEREST IS ACCOMPLISHED BY MEASURING THE INTENSITY OF THE CHARACTERISTIC GAMMA-RAY. THE INTENSITY IS CORRECTED FOR COUNT RATE RELATED LOSSES AND ITEM ATTENUATION. COMPARISON TO SIMILARLY CORRECTED GAMMA-RAY INTENSITIES ON APPROPRIATE PHYSICAL STANDARDS PROVIDES THE RELATIONSHIP BETWEEN OBSERVED GAMMA-RAY INTENSITIES AND NUCLIDE CONTENT.

PHYSICAL STANDARDS ARE GENERALLY AVAILABLE FOR EACH WASTE FORM BEING ASSAYED. THESE STANDARDS HAVE CERTIFIED VALUES BASED ON CALORIMETRY AND ISOTOPIC DETERMINATION. COUNT RATE DEPENDANT LOSSES FROM PULSE PILEUP AND ANALYZER DEAD TIME IS MONITORED AND CORRECTED FOR BY ELECTRONIC MODULES AND RADIOACTIVE COUNT RATE CORRECTION SOURCE MOUNTED ON THE DETECTOR. THE AVERAGE LINEAR ATTENUATION COEFFICIENT OF EACH VERTICAL SEGMENT IS CALCULATED BY MEASUREMENT OF THE TRANSMITTED INTENSITY OF AN EXTERNAL GAMMA-RAY SOURCE. THIS SOURCE IS MOUNTED DIRECTLY OPPOSITE THE GAMMA-RAY DETECTOR ON THE FAR SIDE OF THE PACKAGE BEING ASSAYED.

INSTRUMENTS USING THE ABOVE ASSAY METHOD ARE DRUM COUNTERS 771SGSD00 AND 702SGSD001 AND PACKAGE COUNTERS 771SGSD001, 771SGSD001, 771SGSD002 AND 371SGSD004.

Description of NDA method

HIGH RESOLUTION GAMMA SCANNING COUNTERS--DOUBLE PASS

HIGH RESOLUTION GAMMA SCANNING COUNTERS--DOUBLE PASS OPERATE MUCH THE SAME AS THE SINGLE PASS COUNTERS EXCEPT THAT RATHER THAN OBTAIN THE INTENSITY OF THE TRANSMISSION SOURCE AT THE SAME TIME THE NUCLIDE INTENSITIES ARE MEASURED. A SEPARATE PASS IS MADE TO OBTAIN THE TRANSMISSION INTENSITY. THIS IS ACCOMPLISHED BY TRANSLATING THE SAMPLE PAST THE DETECTOR WITH THE TRANSMISSION SOURCE SHUTTER OPEN TO OBTAIN THE INTENSITY OF THE TRANSMISSION SOURCE AND AGAIN WITH THE SHUTTER CLOSED TO OBTAIN THE INTENSITIES OF THE NUCLIDE PEAKS.

INSTRUMENTS USING THIS ASSAY METHOD ARE PACKAGE COUNTERS 771SGSCS02, 771SGSCS03, AND 771SGSCS04.

Description of NDA method

LOW/HIGH RESOLUTION GAMMA SCANNING COUNTER-DOUBLE-PASS (TRU/LLW SORTING)

THE LOW RESOLUTION-HIGH RESOLUTION GAMMA SCANNING COUNTER WITH DOUBLE PASS CAPABILITY USES THE SAME BASIC TECHNIQUE AS THE DOUBLE PASS HIGH RESOLUTION GAMMA SCANNING COUNTERS. IN ADDITION THEY ALSO HAVE A LOW RESOLUTION SODIUM IODIDE DETECTOR WHICH SIMULTANEOUSLY PERFORMS AN ADDITIONAL ASSAY. THIS SECOND ASSAY IS UTILIZED TO PERFORM TRU/LLW SORTING AT THE 100 nCi/gram LEVEL SIMILAR TO THE LOGAC TYPE OF ASSAY.

Description of NDA method

HIGH RESOLUTION GAMMA PLUS PASSIVE NEUTRON CRATE COUNTER

THIS ROCKY FLATS DESIGNED CRATE COUNTER UTILIZES BOTH NEUTRON AND GAMMA ASSAY TECHNIQUES. NEUTRON DETECTORS ARE MOUNTED UNIFORMLY ON ALL SIDES OF THE ASSAY CHAMBER. A HP-GE GAMMA DETECTOR IS LOCATED SIX FEET AWAY FROM THE ASSAY CHAMBER AND IS COLLIMATED TO MEASURE ALL OF ONE SIDE OF A CRATE CONTAINED IN THE CHAMBER. GAMMA ASSAY OF THE ENTIRE CRATE IS ACHIEVED BY INTERRUPTING THE ASSAY AT THE SELECTED TIME MIDPOINT AND TURNING THE CRATE AROUND TO MEASURE THE SECOND SIDE.

THERMAL NEUTRONS FROM THE SPONTANEOUS FISSIONING OF PU-240 ARE DETECTED BY THE NEUTRON DETECTORS. THE NUMBER OF COINCIDENT NEUTRONS IS PROPORTIONAL TO THE AMOUNT OF PU-240 PRESENT. TOTAL PU IS COMPUTED FROM THE AMOUNT OF PU-240 EFFECTIVE.

THE GAMMA ASSAY IS PERFORMED SIMULTANEOUSLY WITH THE NEUTRON ASSAY. GAMMA-RAY INTENSITIES ARE MEASURED AT 129, 208, AND 413 keV FOR PU OXIDE DETERMINATION, 185 keV FOR U-235 DETERMINATION, 662 AND 722 keV FOR AM-241 DETERMINATION AND 1274 keV FOR PU FLUORIDE DETERMINATION. THE LATER TWO REGIONS OF INTEREST (722 AND 1274 keV) ARE ASSAYED FOR INFORMATION ONLY AND ARE USED TO RESOLVE GROSS DISCREPANCIES BETWEEN THE GAMMA AND NEUTRON ASSAY RESULTS.

Description of NDA method

PASSIVE/ACTIVE NEUTRON CRATE COUNTER (LLW/TRU SORTING)

CRATE COUNTER II IS A FIRST GENERATION PASSIVE/ACTIVE NEUTRON CRATE COUNTER (MODEL NO. 1). IT WAS DESIGNED AND CONSTRUCTED AT LOS ALAMOS, AND MOVED TO ROCKY FLATS IN 1983.

THE PASSIVE ASSAY MODE UTILIZES SHIELDED AND UNSHIELDED HE-3 NEUTRON DETECTOR TUBES LOCATED IN THE WALLS OF THE ASSAY CHAMBER. THEY COMPLETELY SURROUND THE CRATE WHEN THE CHAMBER DOOR IS CLOSED.

IN THE PASSIVE ASSAY, COINCIDENT NEUTRONS FROM THE SPONTANEOUS FISSIONING OF PU-240 ARE DETECTED AND USED FOR THE ASSAY.

THE ACTIVE PORTION OF THE ASSAY SYSTEM UTILIZES A PULSED NEUTRON GENERATOR GENERATING 14 MEV NEUTRONS THAT ARE THERMALIZED AND DETECTS PROMPT NEUTRONS FROM INDUCED FISSIONS OF FISSIONABLE SNM WITH SPECIALLY SHIELDED HE-3 NEUTRON DETECTOR TUBES.

THE ACTIVE ASSAY ANSWER IS THE ONE USED FOR LLW/TRU SORTING WHILE THE PASSIVE ASSAY ANSWER IS USED EITHER FOR COMPARISON TO THE TOTAL GRAMS ASSIGNED TO THE CRATE FROM DRUM COUNT TOTALS, OR TO ASSIGN A GRAM VALUE TO THE CRATE IF THE CRATE WAS NOT FILLED WITH DRUM COUNTED MATERIAL. GRAM VALUES ARE NORMALLY ASSIGNED ONLY TO CRATES FILLED WITH IDC 480 (MISCELLANEOUS LIGHT NON SS METAL) FROM STRIPOUT ACTIVITIES.

Description of NDA method

PASSIVE/ACTIVE NEUTRON DRUM COUNTER (LLW/TRU SORTING)

THE LOS ALAMOS DESIGNED AND CONSTRUCTED DRUM COUNTER IS A PASSIVE/ACTIVE NEUTRON DRUM COUNTING SYSTEM. IT WAS MOVED TO ROCKY FLATS IN 1986.

THE PASSIVE ASSAY MODE UTILIZES SHIELDED AND UNSHIELDED HE-3 NEUTRON DETECTOR TUBES LOCATED IN THE WALLS OF THE ASSAY CHAMBER. THEY COMPLETELY SURROUND THE DRUM WHEN THE CHAMBER DOOR IS CLOSED. THE NEUTRON DETECTOR TUBES IN THREE OF THE VERTICAL SIDES ARE POSITIONED VERTICALLY WHILE THE DETECTOR TUBES IN THE FOURTH VERTICAL SIDE ARE POSITIONED HORIZONTALLY. THIS WAS DONE IN AN ATTEMPT TO PROVIDE INFORMATION ON THE DISTRIBUTION OF THE MATERIAL PACKAGED IN THE DRUM.

IN THE PASSIVE MODE, COINCIDENT NEUTRONS FROM THE SPONTANEOUS FISSIONING OF PU-240 ARE DETECTED AND USED FOR THE ASSAY.

THE ACTIVE PORTION OF THE ASSAY SYSTEM UTILIZES A PULSED NEUTRON GENERATOR GENERATING 14 MEV NEUTRONS THAT ARE THERMALIZED AND DETECTS PROMPT NEUTRONS FROM INDUCED FISSIONS OF FISSIONABLE SNM WITH SPECIALLY SHIELDED HE-3 NEUTRON DETECTOR TUBES.

THE ACTIVE ASSAY ANSWER IS THE ONE USED FOR SORTING AT THE 100 nCi/GRAM LEVEL WHILE THE PASSIVE ASSAY ANSWER IS THE ONE USED TO ASSIGN A GRAM VALUE TO THE DRUM.

ROCKY FLATS HAS ALSO DESIGNED AND CONSTRUCTED A PASSIVE/ACTIVE NEUTRON DRUM COUNTING SYSTEM.

IT WILL OPERATE SIMILAR TO THE LOS ALAMOS SYSTEM EXCEPT THAT NONE OF THE HE-3 NEUTRON DETECTOR TUBES ARE MOUNTED HORIZONTALLY. IN ADDITION THE ROCKY FLATS SYSTEM IS PLACED IN A WELL IN THE FLOOR RATHER THAN ABOVE FLOOR LEVEL LIKE THE LOS ALAMOS SYSTEM.

Description of NDA method

LOW RESOLUTION GAMMA DRUM COUNTER PLUS NEUTRON ASSAY CAPABILITY

THE BUILDING 771 SOUTH DRUM COUNTER (771SGCDC01) IS A LOW RESOLUTION GAMMA ASSAY SYSTEM. IT UTILIZES FOUR 3 X 3 INCH SODIUM IODIDE DETECTORS WITH EACH DETECTOR MONITORING AN EIGHT INCH VERTICAL SEGMENT OF A TYPICAL 55 GALLON DRUM. ALL ASSAYS ARE CORRECTED FOR MATRIX ATTENUATION IN THE SAMPLE BY SIMULTANEOUSLY MEASURING THE INTENSITY OF BA-133 TRANSMISSION SOURCES MOUNTED DIRECTLY OPPOSITE THE DETECTORS ON THE FAR SIDE OF THE DRUM BEING ASSAYED. ANALYSIS RESULTS ARE REPORTED ON A PER SEGMENT BASIS AS WELL AS ON A TOTAL DRUM BASIS.

AS A SECONDARY VERIFICATION OF THE GAMMA ASSAY, PASSIVE NEUTRON COUNTING OF THE SAMPLE IS SIMULTANEOUSLY PERFORMED BY TWELVE BF-3 NEUTRON DETECTOR TUBES MOUNTED AROUND THE DRUM. NEUTRON ASSAY DATA IS CONSIDERED ONLY WHEN THE GAMMA ASSAY VALUE AND THE NEUTRON ASSAY VALUE DISAGREE BY A SIGNIFICANT AMOUNT.

Description of NDA method

LOW RESOLUTION GAMMA DRUM SCAN (LLW/TRU SORTING)

A LOW SPECIFIC ACTIVITY COUNTER (LOSAC) HAS BEEN DESIGNED BY ROCKY FLATS AND INSTALLED FOR THE PURPOSE OF SEGREGATING WASTE DRUMS OF NON-LINE GENERATED WASTE INTO LOW-LEVEL AND TRANSURANIC WASTE. THE SORTING POINT THRESHOLD IS 100 NANOCURIES OF PLUTONIUM PER GRAM OF MATRIX MATERIAL. THIS CORRESPONDS ROUGHLY TO 33 MILLIGRAMS (mg) OF WR PLUTONIUM IN A DRUM CONTAINING 35 KILOGRAMS (kg) OF MATRIX. THE GROSS DRUM WEIGHT IS LIMITED TO 65 kg.

GAMMA-RAYS EMITTED FROM THE MATERIAL IN THE DRUM ARE DETECTED IN A 5 INCH DIAMETER BY 3 INCH THICK SODIUM IODIDE DETECTOR. A LEAD COLLIMATOR WITH AN APERTURE 2 INCHES HIGH AND 4 INCHES DEEP DEFINES THE FIELD OF VIEW FOR THE DRUMS PLACED ON THE TURNTABLE.

ASSAY OF A DRUM IS PERFORMED BY MONITORING THE COUNTS IN THE TOTAL SPECTRUM. THE DRUM IS ROTATED ONE COMPLETE REVOLUTION DURING A SEGMENT ASSAY TIME OF 25 SECONDS. SIXTEEN ASSAY COUNTS ARE PERFORMED ON A DRUM, ONE FOR EACH OF THE 2 INCH SEGMENTS INTO WHICH THE DRUM IS DIVIDED. A ROOM BACKGROUND IS MEASURED BEFORE AND AFTER EACH DRUM. THESE BACKGROUND COUNTS ARE NORMALIZED BY THE TIME ELAPSED TO COUNT THE BACKGROUNDS AND THE TOTAL SEGMENTED ASSAY TIMES AND SUBTRACTED FROM THE SUM OF THE TOTAL COUNTS FOR THE SEGMENTS. THE RESULTING NET TOTAL COUNTS ARE COMPARED TO OUTSIDE LIMITS IMPOSED BEFORE AND AFTER . COMPARISON. IF THE DRUM IS BELOW THE PREDICTED LIMIT, THE DRUM IS LABELED LOW-LEVEL WASTE. IF THE DRUM IS EQUAL TO OR GREATER THAN THE PREDICTED LIMIT, THE DRUM IS LABELED TRANSURANIC WASTE.

TWO CHECKS ARE MADE ON THE SYSTEM TO BRING IT INTO OPERATION. AN EXTERNAL BA-133 SOURCE IS MOUNTED DIAMETRICALLY OPPOSITE TO THE DETECTOR. TWO REGIONS OF INTEREST HAVE BEEN ESTABLISHED FOR THE SPECTRUM OBTAINED FROM THIS SOURCE. BOTH OF THE TWO REGIONS MUST INITIALIZE TO THE PREDICTED COUNT RATES TO ENABLE THE SYSTEM FOR COUNTING. ONCE THESE CRITERIA HAVE BEEN MET, A WORKING STANDARD IS ASSAYED THAT MUST FALL WITHIN LIMITS OBTAINED FROM THE CALIBRATION FUNCTION AND THE STANDARD DRUM VALUE. FAILURE OF THE SYSTEM TO MEET THESE CRITERIA WILL CAUSE THE SOFTWARE TO AUTOMATICALLY PRECLUDE SUBSEQUENT ASSAYS. TWO SUCCESSIVE FAILURES OF THE SYSTEM TO INITIALIZE CORRECTLY REQUIRES INVESTIGATION BY SAFEGUARDS MEASUREMENT PERSONNEL.

Description of NDA method

LOW RESOLUTION GAMMA DRUM SCAN (LLW/TRU SORTING) CONT'D

DRUMS TO BE ASSAYED ARE WEIGHED ON THE LOAD CELL AND ARE PLACED ON THE TURNTABLE WHICH TRAVELS TO ITS FULL VERTICAL HEIGHT. THE DRUM IS THEN ROTATED ONE COMPLETE REVOLUTION AS IT IS LOWERED IN 2 INCH SEGMENTS. THE ASSAY TIME FOR EACH SEGMENT IS 25 SECONDS. SIXTEEN SEGMENT ASSAYS ARE PERFORMED ON EACH DRUM FOR A TOTAL ASSAY TIME OF 400 SECONDS. THE TWO 200 SECOND BACKGROUND COUNTS ARE ACQUIRED FOR EACH DRUM ASSAY-- ONE PRIOR TO EACH DRUM ASSAYED AND ONE AFTER. THIS SECOND BACKGROUND BECOMES THE FIRST BACKGROUND COUNT FOR THE NEXT DRUM

THE FINAL REPORT, PRINTED TO THE HARD COPY OUTPUT, CONTAINS THE OPERATOR IDENTIFICATION NUMBER, THE DRUM IDENTIFICATION NUMBER, THE DRUM MATERIAL IDENTIFICATION CODE, DRUM GROSS WEIGHT, NET WEIGHT, AND TARE WEIGHT. BACKGROUND COUNTS AND BACKGROUND ACQUISITION LIVE TIMES. THE SIXTEEN SEGMENTS OF FOREGROUND DATA AND SUMMARY DATA ARE BASED ON THE CALIBRATION MODEL.

THE SOFTWARE PROGRAM CONTROLLING THE LOSAC COUNTER HAS RECENTLY BEEN CHANGED TO INCLUDE THE ASSAY OF TRANSURANIC WASTE DRUMS BY ASSIGNING A GRAM VALUE TO DRUMS DETERMINED NOT TO BE LOW LEVEL WASTE. THE UPPER LIMIT FOR THIS CALIBRATION IS 20 GRAMS.

Description of NDA method

LOW RESOLUTION GAMMA HEPA FILTER COUNTER (LLW/TRU SORTING)

A SECOND LOW RESOLUTION GAMMA DRUM COUNTER IS THE HIGH EFFICIENCY PARTICULATE AIR (HEPA) FILTER COUNTER (771HEPA101). THIS COUNTER IS USED PRIMARILY FOR ASSAYING THE PU-239 CONTENT OF THE HEPA FILTERS. THE ASSAY DATA IS USED TO SORT THE SAMPLES INTO LOW LEVEL WASTE (LLW) AND TRANSURANIC WASTE (TRU). THE SORTING THRESHOLD IS 100 nCi/GRAM.

THE ASSAY SYSTEM CONSISTS OF TWO 4 X 4 X 16 INCH SODIUM IODIDE DETECTORS INTERFACED TO A MICROVAX CONTROLLED MULTICHANNEL ANALY. ER. ANALYSES ARE MADE BY AUTOMATICALLY WEIGHING THE SAMPLE WITH A COMPUTER CONTROLLED LOAD CELL, FOLLOWED BY ROTATING THE SAMPLE THROUGH ONE COMPLETE REVOLUTION WHILE MEASURING THE PU-239 413 KeV GAMMA-RAY PEAK. NO TRANSMISSION OR LIVE TIME CORRECTIONS ARE MADE TO THE BASIC COUNTING DATA PRIOR TO REPORTING THE FINAL ASSAY RESULTS.

Description of NDA method

NEUTRON COINCIDENCE COUNTER

THE CONFIRMATORY MEASUREMENT COUNTER (991CMC101) IS A NEUTRON COINCIDENCE COUNTER. IT UTILIZES HE-3 NEUTRON DETECTOR TUBES TO COUNT TIME COINCIDENT NEUTRONS EMITTED BY PLUTONIUM AND AMERICIUM BEARING MATERIALS. THE DETECTOR SIGNALS ARE COLLECTED BY A HEC-100 NEUTRON COINCIDENCE COUNTING INSTRUMENT. RANDOM NEUTRONS FROM ALPHA-n REACTIONS AND TIME CORRELATED NEUTRONS FROM SPONTANEOUS FISSION REACTIONS WITHIN A 30 GALLON DRUM ARE SEPARATED BY THE ELECTRONICS. BOTH ARE REPORTED ON HARDCOPY. INTERSITE SHIPMENTS ARE CONFIRMED BY COMPARISON OF TOTAL NEUTRON EMISSION RECORDS OBTAINED FROM SIMILAR CMC'S AT THE SHIPPING AND RECEIVING SITES.

Description of NDA Method

PRODUCTION CALORIMETERS BUILDING 771

Calorimeters are thermal measurement devices which determine the steady-state heat production of a sample. This ascertained specific heat is then combined with the known isotopic composition, (generally obtained by Gamma Spectrometry), to determine gram content of specific elements.

Production calorimeters in building 771 are essentially of two different designs. The first; Rocky Flats designed calorimeters, circa 1970, are single resistance type measurement devices. Eight measurement cells are housed in a right circular cylindrical water bath. The isothermal bath is electrically maintained at a constant temperature with a range of 0.01C. Hewlett Packard electronics are used for resistance measurement. An RSX operating system is used for data collection and data storage. The second; Mound designed calorimeters, circa 1987, are twin-cell, bridge resistance type measurement devices. Two measurement cells are housed in each cubic water bath. The isothermal bath is electrically maintained at a constant temperature with a range of 0.001C. Hewlett Packard electronics are used for voltage measurements. An RSX operating system is used to evaluate, process, and store acquired data.

NDA
INSTRUMENT

TYPE OF ASSAY

371S6SC304	HIGH RESOLUTION GAMMA CAN SCAN--SINGLE PASS
371S6SD001	LOW/HIGH RESOLUTION GAMMA DRUM SCAN--DOUBLE PASS (LLW/TRU SORTING)
371PADCI01	PASSIVE/ACTIVE NEUTRON DRUM COUNTER (LLW/TRU SORTING)
369S6SD001	LOW/HIGH RESOLUTION GAMMA DRUM SCAN--DOUBLE PASS (LLW/TRU SORTING)
369PDCI01	HIGH RESOLUTION GAMMA PLUS PASSIVE NEUTRON CRATE COUNTER
369PADCI01	PASSIVE/ACTIVE NEUTRON CRATE COUNTER (LLW/TRU SORTING)
707S6SD001	HIGH RESOLUTION GAMMA DRUM SCAN--SINGLE PASS
771S6SC301	HIGH RESOLUTION GAMMA CAN SCAN--SINGLE PASS
771S6SC302	HIGH RESOLUTION GAMMA CAN SCAN--DOUBLE PASS
771S6SC303	HIGH RESOLUTION GAMMA CAN SCAN--DOUBLE PASS
771S6SC304	HIGH RESOLUTION GAMMA CAN SCAN--DOUBLE PASS
771S6SD001	HIGH RESOLUTION GAMMA DRUM SCAN--SINGLE PASS
771S6DC001	LOW RESOLUTION GAMMA DRUM COUNTER PLUS NEUTRON ASSAY CAPABILITY
771HEPAI01	LOW RESOLUTION GAMMA HEPA FILTER COUNTER (LLW/TRU SORTING)
771PADCI01	PASSIVE/ACTIVE NEUTRON DRUM COUNTER (LLW/TRU SORTING)
776S6SC301	HIGH RESOLUTION GAMMA CAN SCAN--SINGLE PASS (OUT-OF-LINE)
776S6SC302	HIGH RESOLUTION GAMMA CAN SCAN--SINGLE PASS (IN-LINE)
776L0SAC01	LOW RESOLUTION GAMMA DRUM SCAN (LLW/TRU SORTING)
9910CHCI01	NEUTRON COINCIDENCE COUNTER

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IDC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
000	EMPTY CONTAINERS	N	NONE
001	FIRST STAGE SLUDGE--BLDG 774	N	NONE
003	GREASE--BLDG 774	N	NONE
004	SPECIAL SETUPS--BLDG 774	N	NONE
007	BYPASS SLUDGE --BLDG. 374	N	NONE
010	METAL BUTTON, RF, ACCEPTABLE PURITY	S	NONE
011	METAL BUTTONS, OTHER, ACCEPTABLE PURITY	S	NONE
012	METAL OF ACCEPTABLE PURITY	S	NONE
013	METAL BUTTONS AWAITING LAB ANALYSIS	N	NONE
014	ELECTROREFINED BUTTONS, ACCEPTABLE PURITY	S	NONE
015	MOLTEN SALT BUTTONS, ACCEPTABLE PURITY	S	NONE
016	LEACHED PART V METAL, ACCEPTABLE PURITY	S	NONE
017	NON-ROUTINE ER METAL UNACCEPTABLE PURITY	N	NONE
018	DEPLETED URANIUM DERBIES, ACCEPTABLE PURITY	S	NONE
019	DOR BUTTONS, UNACCEPTABLE PURITY	N	NONE
020	NON-ROUTINE METAL UNACCEPTABLE PURITY	N	NONE
021	NON-ROUTINE HYDRIDE	N	NONE
023	NON-BLENDABLE METAL	N	NONE
024	LANL ER METAL AWAITING ANALYSIS	S	NONE
025	AL ALLOYED ANODE HEEL FOR SRP	N	NONE
026	LEACHED PART V METAL, UNACCEPTABLE PURITY	N	NONE
027	ANODE FEED FOR E/R, NON-ROUTINE METAL	N	NONE
029	ANODE FEED FOR E/R, DOR REJECTS	N	NONE
030	METAL BUTTONS, ROCKY FLATS, UNACCEPTABLE PURITY	N	NONE
031	ANODE FEED FOR ELECTROREFINING-UNACCEPTABLE PURITY	N	NONE
032	METAL BUTTONS,AWAITING PROCESS FOR MOLTEN SALT	N	NONE
033	METAL BUTTONS, SKIN TURNINGS MOLTEN SALT	N	NONE
035	METAL AWAITING DISPOSITION, ANALYSIS COMPLETE	N	NONE
037	ANODE HEEL >200 GMS PU	N	NONE
038	039 PREPARED FOR E/R	N	NONE
039	ER BUTTONS FROM CELL CLEANOUT	N	NONE
040	TURNING BRIQUETTE	S	NONE
041	SOLID SCRAP BRIQUETTE	S	NONE
044	OXIDE SELECTED FOR SRP > 85%	N	NONE
045	METAL-CUT UP MOLTEN SALT - BLDG 371	N	NONE
046	OXIDE <0.20 G/G PU SELECTED FOR MMEC	N	NONE
047	OXIDE REPACKED FOR RHO <80%	N	NONE
048	OXIDE <0.50 G/G PU PACKAGED FOR LANL	N	NONE
049	ANODE HEEL PACKAGED FOR LANL	N	NONE
050	SKULLS	N	NONE
051	ANODE HEEL	N	NONE
052	OXIDE PYRO RF	N	NONE
053	OXIDE FOR CALCINING	N	NONE
054	OXIDE >0.65 G/G PU SELECTED FOR MMEC	N	NONE
056	OXIDE 0.60 - 0.65 G/G PU SELECTED FOR MMEC	N	NONE
057	OXIDE AWAITING SPEC ANALYSIS	N	NONE
058	OXIDE AWAITING PRODUCTION CATEGORIZATION	N	NONE

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IDC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
059	OXIDE 0.80 - 0.85 G/G PU PACKAGED FOR SRP	N	NONE
060	OXIDE	N	NONE
061	NON-SPEC OXIDE	N	NONE
062	HIGH PURITY OXIDE HEEL	N	NONE
063	HYDRIDES	N	NONE
064	OXIDE >0.85 G/G PU PACKAGED FOR LANL PYRO	N	NONE
065	OXIDE HEEL IN SMALL STACKER CANS	N	NONE
066	OXIDE 0.80 - 0.85 G/G PU PACKAGED FOR LANL	N	NONE
067	OXIDE IN SMALL INNER CAN--BLDG 371	N	NONE
069	ROASTER OXIDE D-38	N	NONE
070	NITRATE FEED	N	NONE
073	MIXED IDC'S OUTSIDE THE PSZ	N	NONE
080	PEROXIDE CAKE (INCLUDES GREEN CAKE)	N	NONE
081	IMPURE PEROXIDE CAKE (INCLUDES IMPURE GREEN CAKE)	N	NONE
082	GREEN CAKE IN SMALL INNER CAN - BLDG 371	N	NONE
083	HIGH FIRED OXIDE-DOR	N	NONE
084	HANFORD PUREX OXIDE	N	NONE
086	OXIDE E/R SCRAPE OUT	N	NONE
087	IMPURE GREEN CAKE IN SMALL INNER CANS--BLDG 371	N	NONE
089	GREASE OXIDE (GREEN CAKE)	N	NONE
090	TETRAFLUORIDE (PUF4)	N	NONE
091	NON-SPEC FLUORIDE	N	NONE
092	IMPURE FLUORIDE HEEL	N	NONE
093	SODIUM FLUORIDE PELLETS	N	NONE
095	TETRAFLUORIDE - 371	N	NONE
097	IMPURE FLUORIDE IN SMALL INNER CANS--BLDG 371	N	NONE
099	GREASE FLUORIDE	N	NONE
100	FILTRATE RECOVERY NITRATE FEED (EVAPORATOR BOTTOM)	N	NONE
130	AMERICIUM METAL	N	NONE
131	ANODE HEEL-READY FOR RE-RUN (NM VALUE)	N	NONE
137	ANODE HEEL-SELECTED AND CUT FOR RE-RUN (NO NM VAL)	N	NONE
140	TURNINGS (ACCEPTABLE FOR BRIQUETTING)	S	NONE
141	FABRICATION METAL FINES	N	NONE
142	TURNINGS (UNACCEPTABLE FOR BRIQUETTING)	N	NONE
145	OXIDE - FAILED FIRST L.O.I. TEST	N	NONE
146	OXIDE - L.O.I. REJECT	N	NONE
150	SOLID SCRAP OR FREE METAL, RECASTABLE	S	NONE
151	FREE METAL, FINES, UNACCEPTABLE PURITY	N	NONE
152	INGOT PIECES UNACCEPTABLE PURITY	N	NONE
153	SOLID SCRAP UNACCEPTABLE PURITY	N	NONE
154	E/R SCRAPE OUT MATERIAL	N	NONE
159	SCREENINGS FROM OXIDE	N	003200
160	REJECTED PARTS	S	NONE
161	SCRAP PART	N	NONE
170	SEMI-FABRICATED CIRCLES, SQUARES, PLATE, SHEET	S	NONE
171	RODS	S	NONE
173	SEMI-FABRICATED PARTS	S	NONE

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IOC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
180	FINISHED PARTS, NEW PRODUCTION	S	NONE
185	PARTS FROM RETIREMENTS	N	NONE
186	UNELEACHED PART V METAL	N	NONE
187	UNELEACHED METAL PARTS (116 PROGRAM)	N	NONE
189	RECYCLED BINARY-INGOTS	N	NONE
190	CASTINGS	S	NONE
191	INGOTS	S	NONE
192	FEED INGOTS	S	NONE
193	TA TARGET AND SUB-TARGET, ACCEPTABLE PURITY	S	NONE
194	BILLETS (CRITICALITY STUDY)	N	NONE
195	INGOTS OF UNACCEPTABLE PURITY	N	NONE
196	INGOTS AVAILABLE FOR BLENDING	S	NONE
197	TA TARGET AND SUB-TARGET, TO BE LEACHED	N	NONE
198	BILLETS, SPECIAL UNACCEPTABLE PURITY	N	NONE
199	SHIELDS	N	NONE
200	STANDARDS	N	NONE
201	SEALED SOURCES	N	NONE
210	METAL SAMPLES, ACCEPTABLE PURITY	S	NONE
211	RETAINED METAL SAMPLES	S	NONE
212	METAL SAMPLES, UNACCEPTABLE PURITY	N	NONE
213	MOUNTED METAL SAMPLES, UNACCEPTABLE PURITY	N	NONE
237	ANODE HEEL-SELECTED FOR OFF-SITE SHIPMENT	N	NONE
239	LOW PURITY OXIDE HEEL	N	NONE
290	FILTER SLUDGE	N	.003130
291	DRIED LAB WASTE FLUORIDE SLUDGE	N	NONE
292	INCINERATOR SLUDGE	N	.003130
295	SEWER SLUDGE	N	NONE
299	MISC. SLUDGE	N	.003130
300	GRAPHITE MOLDS	N	.000460
301	CLASSIFIED GRAPHITE SHAPES	N	.000460
302	BENELEX & PLEXIGLASS	N	.000930
303	SCARFED GRAPHITE CHUNKS	N	.003420
310	GRAPHITE SCARFINGS & FINES	N	.003220
311	GRAPHITE HEELS	N	.003220
312	GRAPHITE, COARSE	N	.003420
318	HYDRIDE-FROM TA CRUCIBLES	N	NONE
319	OXIDE-FROM TA CRUCIBLES	N	NONE
320	HEAVY NON-SS METAL (TA, W/PT)	N	.002270
321	LEAD	N	.002270
328	FILTERS, FUL-FLO, FROM INCINERATOR	N	.003130
330	COMBUSTIBLES, DRY (PAPER, RAGS, ETC.)	N	.000620
331	FILTERS, FULFLO, OTHER THAN FROM INCINERATOR	N	NONE
332	OILY SLUDGE	N	NONE
333	CALCIUM METAL	N	NONE
334	BLANKET, FIRE	N	.003130
335	ABSOLUTE DRYBOX FILTERS, NOT ACID CONTAMINATED	N	.004240
336	COMBUSTIBLES, WET	N	.000620

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IDC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
337	PLASTIC (TEFLON, PVC, POLY, ETC.)	N	.000620
338	FILTER MEDIA	N	.005310
339	LEADED DRYBOX GLOVES, NOT ACID CONTAMINATED	N	.001290
340	SLUDGE FROM SIZE REDUCTION AREA	N	.000620
341	LEADED DRYBOX GLOVES/NITRIC ACID CONTAMINATED	N	NONE
342	ABSOLUTE DRYBOX FILTERS, ACID CONTAMINATED	N	.004240
363	ELECTROREFINING SALT - FIRST USE	N	.005190
364	ELECTROREFINING SALT - SECOND USE	N	.005190
365	SALT FROM BAD DOR RUN	N	NONE
366	MED LEVEL DOR SALT	N	NONE
367	WHITE DOR SALT	N	NONE
368	MG OXIDE CERAMIC CRUCIBLES - NOT LECO	N	.002260
369	LECO HEEL	N	.005310
370	LECO CRUCIBLES	N	.004000
371	FIRE BRICK	N	.000940
372	GRIT	N	.003200
373	FIRE BRICK HEEL	N	.005310
374	BLACKTOP, CONCRETE, DIRT, & SAND	N	.001600
375	OIL DRY	N	.005310
376	PROCESSED FILTER MEDIA	N	NONE
377	FIRE BRICK, COARSE	N	.003420
378	FIRE BRICK, PULVERIZED OR FINES	N	.003200
379	FIREBRICK SCAPPED	N	.000940
380	OXALATE RESIDUES	N	NONE
386	DRIED BSO SLUDGE	N	.003130
387	REBURNED SS&C SWEEPINGS	N	.002260
388	UN-PULVERIZED SS&C TO BPP	N	NONE
389	UN-PULVERIZED SS&C TO LANL	N	NONE
390	UNPULVERIZED SLAG	N	.002260
391	UNPULVERIZED SAND & CRUCIBLE	N	.002260
392	UNPULVERIZED SAND, SLAG & CRUCIBLE	N	.002260
393	SAND, SLAG & CRUCIBLE HEEL	N	.004100
394	SAND FROM BSO	N	.002260
395	UNPULVERIZED SLAG AND CRUCIBLE	N	.002260
396	PULVERIZED SLAG	N	.002910
397	PULVERIZED SAND & CRUCIBLE	N	.002910
398	PULVERIZED SAND, SLAG & CRUCIBLE	N	.002910
399	PULVERIZED SLAG AND CRUCIBLE	N	.002910
400	ION COLUMN FEED < 5 G/L PU	N	NONE
401	ION COLUMN FEED > 5 G/L PU	N	NONE
402	SOLVENT EXTRACTION FEED	N	NONE
403	SOLVENT EXTRACTION PRODUCT	N	NONE
404	MOLTEN SALT, CA, ZN, R	N	.006150
405	MOLTEN SALT, UNKNOWN % UNPULVERIZED	N	.006630
406	MOLTEN SALT, UNKNOWN % PULVERIZED	N	.005690
407	MOLTEN SALT, 3% UNPULVERIZED	N	.006630
408	MOLTEN SALT, 6% PULVERIZED	N	.005690

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IDC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
409	MOLTEN SALT, 30% UNPULVERIZED	N	.005690
410	MOLTEN SALT, 30% PULVERIZED	N	.005690
411	ELECTROREFINING SALT - FINAL DISPOSITION	N	.005190
412	GIBSON SALT	N	.005690
413	IMPURE SALT FROM CELL CLEANOUT	N	.006630
414	DIRECT OXIDE REDUCTION SALT-UNOXIDIZED CA	N	.005690
415	PLUTONIUM CHLORIDE MIXED SALT	N	.005690
416	ZINC-MAGNESIUM ALLOY METAL	N	.005690
417	DICESIUM PLUTONIUM HEXACHLORIDE	N	NONE
418	MOLTEN SALT PACKAGED FOR LANL	N	.005690
419	UNPULVERIZED INCINERATOR ASH	N	NONE
420	PULVERIZED INCINERATOR ASH	N	.003200
421	ASH HEEL	N	.004100
422	SOOT	N	.004100
423	SOOT HEELS	N	.004100
424	ASH PACKAGED FOR LANL	N	.003200
425	FLUID-BED ASH	N	.002510
426	INCINERATOR ASH TO RHO	N	NONE
428	ASH SELECTED FOR MMEC	N	.003200
429	SCRUB ALLOY SPENT SALT	N	.005690
430	RESIN, UNLEACHED	N	NONE
431	RESIN, LEACHED	N	.001430
432	RESIN, CEMENTED	N	NONE
438	INSULATION	N	.005310
440	GLASS (EXCEPT RASCHIG RINGS)	N	.001060
441	UNLEACHED RASCHIG RINGS ONLY	N	.001060
442	LEACHED RASCHIG RINGS	N	.001060
454	DIRECT OXIDE REDUCTION SALT-OXIDIZED CA	N	.005690
470	MOLTEN SALT SELECTED FOR MMEC	N	.005690
471	MOLTEN SALT PACKAGED FOR LANL	N	.005690
472	ELECTROREFINED SALT SELECTED FOR MMEC	N	.005190
473	ELECTROREFINED SALT PACKAGED FOR LANL	N	.005190
474	LECO SELECTED FOR MMEC	N	.004000
475	LECO PACKAGED FOR LANL	N	.004000
480	LIGHT METAL	N	.000620
481	LIGHT NON-SS METAL (FE, CU, AL, SS) PREPARED FOR LEACH	N	.000620
483	SCRAP D-38 METAL (UNCLASSIFIED)	N	NONE
484	CLASSIFIED NON-VM, SCRAP METAL SHAPES (NOT BE)	N	.000620
485	SCRAP D-38 CLASSIFIED SHAPES	N	.000620
486	CLASSIFIED TOOLING FOR DISPOSAL	N	.000620
487	CLASSIFIED PLASTIC SHAPES	N	.000620
488	GLOVEBOX PARTS WITH LEAD	N	.000620
489	CLASSIFIED BE SCRAP METAL SHAPES	N	.000620
490	H.E.P.A. FILTERS (24X24) NOT ACID CONTAMINATED	N	.002390
491	PLENUM PRE FILTER	N	.005310
492	H.E.P.A. FILTERS (24X24) ACID CONTAMINATED	N	.002390
500	ENRICHED URANIUM SPECIAL SOLUTION (NON-CONFORMI B)	N	2.105000

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IDC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
501	ION COLUMN EFFLUENT	N	.011980
502	HNO3 DISTILLATE	N	.011980
503	MISC. ACID WASTE, PLUTONIUM	N	.011980
507	AM 241 OXALATE FILTRATE	N	.076660
508	ACID CHLORIDE WASTE	N	.021100
509	ACID CHLORIDE SOLUTION STANDARD	N	NONE
513	STEAM CONDENSATE &/OR COOLING WATER	N	.021100
519	STEAM (H2O) CONTAINING SS MATERIAL	N	.004290
527	MISC. BASIC WASTE SOLUTION	N	.021100
528	CAUSTIC SCRUBS &/OR FILTRATES	N	.021100
533	ORGANICS-DISC-LEVEL-COOL OIL-CAR TET-PERCHLOR ETC.	N	.030240
535	ORGANICS SOLUTION (LAB QUANTITIES)	N	.452120
541	ANALYTICAL LAB. SOLUTION	N	.212200
542	AM 241 CATION ION COLUMN EFFLUENT	N	.282710
600	AL MG METAL ALLOY (BUTTON)	N	NONE
601	AL MG OXIDE	N	NONE
620	AL ALLOY BUTTONS	N	NONE
649	CUT-UP METAL FEED FOR PU/NP	N	NONE
650	ER BUTTON FROM PU/NP	N	NONE
651	ANODE HEEL FROM PU/NP	N	NONE
652	ER SCRAPE OUT FROM PU/NP	N	NONE
653	OXIDE FROM PU/NP	N	NONE
654	ER SALT FROM PU/NP	N	.005190
655	ER CERAMICS FROM PU/NP	N	.002260
702	SLUDGE W/ WASH WATER	N	.004290
710	HEAVY WATER (D2O)	N	NONE
720	GAS (D2,HD,D2S)	N	NONE
730	DEUTERATED ORGANIC COMPOUNDS	N	NONE
777	EMPTY WASTE BOX (CRATE)	N	NONE
800	SOLIDIFIED SLUDGE - BLDG 774	N	NONE
801	SOLIDIFIED ORGANICS - BLDG 774	N	NONE
802	SOLIDIFIED LAB WASTE - BLDG 774	N	NONE
803	SOLIDIFIED SLUDGE - BLDG 774	N	NONE
804	SALTCRETE	N	NONE
805	PONDCRETE	N	NONE
806	SOLIDIFIED PROCESS SOLIDS	N	NONE
807	SOLIDIFIED BYPASS SLUDGE - BUILDING 37	N	NONE
815	CEMENTED INSULATION AND FILTER MEDIA	N	NONE
823	CEMENTED MISCELLANEOUS SLUDGE	N	NONE
838	EMPTY OPEN TOP 55 GALLON WHITE DRUM	N	NONE
910	DOE ACCEPTABLE ASSEMBLIES	S	NONE
911	SURVEILLANCE UNITS	S	NONE
912	SCRAP ENRICHED PARTS IN SHIPPING CONTAINERS	N	NONE
913	NON-WR ASSEMBLIES	S	NONE
914	RETIREMENT ASSEMBLIES	N	NONE
915	WR SUB-ASSEMBLY	S	NONE
998	NMC FOR DISCARD OF OY DRUMS AFTER APPROVAL	N	1.00000

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IDC	DESCRIPTION	SPEC/ NON-SPEC	DISCARD LIMIT
999	NMC USE ONLY	N	NONE
A60	PU/OY OXIDE 10000 PPM OY OR LESS	N	NONE
ADJ	GL ADJ NMC	N	NONE
CNV	IDC FOR CONVERSION OF COA	N	NONE
G51	ANODE HEEL 23% GALIUM	N	NONE
H35	HANFORD METAL BUTTONS	S	NONE
IPS	IN PROCESS SOLID / SOLUTION	N	NONE
U61	PU/OY OXIDE 5000 TO 10000 PPM OY	N	NONE
Y61	PU/OY OXIDE OVER 10000 PPM OY	N	NONE

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LIST OF IDC'S COUNTED ON NDA INSTRUMENTS

IDC	DESCRIPTION	IDC	DESCRIPTION
003	GREASE--BLDG 774	405	MOLTEN SALT, UNKNOWN % UNPULVERIZED
004	SPECIAL SETUPS--BLDG 774	406	MOLTEN SALT UNKNOWN % PULVERIZED
200	STANDARDS	407	MOLTEN SALT, 3% PULVERIZED
290	FILTER SLUDGE	408	MOLTEN SALT, 8% PULVERIZED
292	INCINERATOR SLUDGE	409	MOLTEN SALT, 30% UNPULVERIZED
300	GRAPHITE MOLES	410	MOLTEN SALT, 30% PULVERIZED
301	CLASSIFIED GRAPHITE SHAPES	411	ELECTROREFINING SALT-FINAL DISPOSITION
302	BENELEX AND PLEXIGLASS	413	IMPURE SALT FROM CELL CLEANOUT
303	SCARFED GRAPHITE DRUMS	414	DIRECT OXIDE REDUCTION SALT-UNOXIDIZED CA
310	GRAPHITE SCARFINGS AND FINES	415	PLUTONIUM CHLORIDE MIXED SALT
312	GRAPHITE, COARSE	418	MOLTEN SALT PACKAGED FOR LANL
320	HEAVY NON-SS METAL (TA,W,PT)	420	PULVERIZED INCINERATOR ASH
321	LEAD	424	ASH PACKAGED FOR LANL
328	FILTERS, FUL-FLO, FROM INCINERATOR	425	FLUID-BED ASH
330	COMBUSTIBLES, DRY (PAPER, RAGS, ETC.)	426	ASH PACKAGED FOR HANFORD
331	FILTERS, FUL-FLO OTHER THAN FROM INCINERATOR	429	SCRUB ALLOY SPENT SALT
334	BLANKET, FIRE	430	RESIN, UNLEACHED
335	ABSOLUTE DRYBOX FILTERS, NOT ACID CONTAMINATED	431	RESIN, LEACHED
336	COMBUSTIBLES, WET	432	RESIN, CEMENTED
337	PLASTIC (TEFLON,PVC,POLY,ETC.)	440	GLASS (EXCEPT RASCHIG RINGS)
338	FILTER MEDIA	441	UNLEACHED RASCHIG RINGS ONLY
339	LEADED DRYBOX GLOVES, NOT ACID CONTAMINATED	442	LEACHED RASCHIG RINGS
340	SLUDGE FROM SIZE REDUCTION AREA	471	MOLTEN SALT PACKAGED FOR LANL
341	LEADED DRYBOX GLOVES/NITRIC ACID CONTAMINATED	473	ER SALT PACKAGED FOR LANL
368	MG OXIDE CERAMIC CRUCIBLES-NOT LECO	479	EMPTY VOLRATH CANS
370	LECO CRUCIBLES	480	LIGHT NON-SS METAL
371	FIRE BRICK	481	LIGHT NON-SS METAL (FE,CU,AL,SS) PREPARED FOR LEAC
372	GRIT	484	CLASSIFIED NON-NH, SCRAP METAL SHAPES (NOT FE)
374	BLACKTOP, CONCRETE, DIRT, SAND	485	SCRAP D-38 CLASSIFIED SHAPES
375	OIL DRY	486	CLASSIFIED TOOLING FOR DISPOSAL
376	PROCESSED FILTER MEDIA	487	CLASSIFIED PLASTIC SHAPES
377	FIRE BRICK, COARSE	490	HEPA FILTERS (24X24) NOT ACID CONTAMINATED
378	FIRE BRICK, PULVERIZED AND FINES	491	PLENUM PRE-FILTERS
379	FIRE BRICK, SCARFED	654	ER SALT PU/NP
387	RETURNED SSC SWEEPINGS	803	SOLIDIFIED SLUDGE FROM 374
392	UNPULVERIZED SSC	804	SALT CRETE
398	PULVERIZED SSC	806	SOLIDIFIED PROCESS SOLIDS
404	MOLTEN SALT, CA K, NA, ZN	807	SOLIDIFIED BYPASS SLUDGE 374

LIST OF IDC'S VS NDA INSTRUMENT

IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)	IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)	IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)
003	5690PCCI01	350	328	771SGDC01	100	338	707SGSDC01	500
				771SGSDC01	500		371SGSDC01	500
004	5690PCCI01	350		707SGSDC01	500		569SGSDC01	500
				371SGSDC01	500			
290	771SGSDC01	500		569SGSDC01	500	339	371PADCI01	200
	707SGSDC01	500					371SGSDC01	500
	371SGSDC01	500	330	771SGDC01	100		569SGSDC01	500
	569SGSDC01	500		771SGSDC01	500		707SGSDC01	500
				707SGSDC01	500		771SGSDC01	500
292	771SGDC01	100		371SGSDC01	500			
	771SGSDC01	500		569SGSDC01	500	340	776SGSC01	300
	707SGSDC01	500		776LOSAC01	20			
	371SGSDC01	500		771HEPAI01	20	341	371PADCI01	200
	569SGSDC01	500		371PADCI01	200		371SGSD01	500
				5690PCCI01	350		569SGSDC01	500
300	771SGDC01	100		569PACCI01	350		707SGSDC01	500
	771SGSDC01	500					771SGSDC01	500
	707SGSDC01	500	331	771SGDC01	100			
	371SGSDC01	500		771SGSDC01	500	368	771SGSCS01	150
	569SGSDC01	500		707SGSDC01	500		771SGSCS02	150
				371SGSDC01	500		771SGSCS03	150
301	771SGDC01	100		569SGSDC01	500		771SGSCS04	150
	771SGSDC01	500					776SGSCS01	150
	707SGSDC01	500	334	771SGDC01	100		776SGSCS02	150
	371SGSDC01	500		771SGSDC01	500			
	569SGSDC01	500		707SGSDC01	500	370	771SGDC01	100
				371SGSDC01	500			
302	771SGDC01	100		569SGSDC01	500	371	771SGDC01	100
	771SGSDC01	500					771SGSDC01	500
	707SGSDC01	500	335	771SGDC01	100		707SGSDC01	500
	371SGSDC01	500		771SGSDC01	500		771SGSDC01	500
	569SGSDC01	500		707SGSDC01	500		569SGSDC01	500
	5690PCCI01	350		371SGSDC01	500			
				569SGSDC01	500	372	771SGDC01	100
303	771SGSDC01	500						
	707SGSDC01	500	336	771SGDC01	100	374	771SGDC01	100
	371SGSDC01	500		771SGSDC01	500		771SGSDC01	500
	569SGSDC01	500		707SGSDC01	500		707SGSDC01	500
				371SGSDC01	500		771SGSDC01	500
310	771SGSCS01	250		569SGSDC01	500		569SGSDC01	500
				776LOSAC01	20			
312	771SGSCS01	250		771HEPAI01	20	375	771SGDC01	100
				5690PCCI01	350			
				569PACCI01	350	376	771SGDC01	100
320	371PADCI01	200						
	371SGSDC01	500	337	771SGDC01	100	377	771SGDC01	100
	569SGSDC01	500		771SGSDC01	500		771SGSDC01	500
	771SGSDC01	500		707SGSDC01	500		707SGSDC01	500
	707SGSDC01	500		371SGSDC01	500		771SGSDC01	500
				569SGSDC01	500		569SGSDC01	500
321	771SGSDC01	500		5690PCCI01	350			
	707SGSDC01	500				378	771SGDC01	100
	371SGSDC01	500	338	771SGDC01	100		771SGSDC01	500
	569SGSDC01	500		771SGSDC01	500		776SGSDC01	500

LIST OF IDC'S VS NDA INSTRUMENT

IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)	IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)	IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)
378	371SGSDC01	500				441	707SGSDC01	500
	569SGSDC01	500	415	771SGSCS01	520		371SGSDC01	500
							569SGSDC01	500
379	771SGSDC01	500	418	371SGSCS04	500		5690PCCI01	350
	707SGSDC01	500				442	771SGCDC01	100
	371SGSDC01	500	420	771SGSCS01	550		771SGSDC01	500
	569SGSDC01	500		771SGSCS02	550		707SGSDC01	500
387	771SGSCS01	400		771SGSCS03	550		371SGSDC01	500
				771SGSCS04	550		569SGSDC01	500
				371SGSCS04	400		5690PCCI01	350
392	771SGSCS01	400	424	371SGSCS04	400	471	371SGSCS04	300
	771SGSCS02	400				473	371SGSCS04	300
	771SGSCS03	400	425	771SGCDC01	100	479	771SGSDC01	500
	771SGSCS04	400		771SGSDC01	500		707SGSDC01	500
	371SGSCS04	300		707SGSDC01	500		371SGSDC01	500
398	771SGSCS01	400		371SGSDC01	500		569SGSDC01	500
	771SGSCS02	400		569SGSDC01	500	426	371SGSCS04	400
	771SGSCS03	400				429	776SGSCS02	500
	771SGSCS04	400					771SGSDC01	500
404	771SGSCS01	520					707SGSDC01	500
							371SGSDC01	500
405	771SGSCS01	520					569SGSDC01	500
	776SGSCS01	500					707SGSDC01	500
							771SGSDC01	500
406	771SGSCS01	520					5690PCCI01	350
			430	771SGCDC01	100		569PACCI01	350
407	771SGSCS01	520		771SGSDC01	500	481	371PADC I01	200
				707SGSDC01	500		371SGSDC01	500
408	771SGSCS01	520		371SGSDC01	500		569SGSDC01	500
				569SGSDC01	500		707SGSDC01	500
409	771SGSCS01	520					771SGSDC01	500
	776SGSCS01	500	431	771SGCDC01	100		569PADC I01	350
	776SGSCS02	500		771SGSDC01	500		569PACCI01	350
	371SGSCS04	300		707SGSDC01	500			
				371SGSDC01	500	484	371PADC I01	200
410	771SGSCS01	520		569SGSDC01	500		371SGSDC01	500
							569SGSDC01	500
411	771SGSCS01	520	432	771SGCDC01	100		707SGSDC01	500
	776SGSCS01	500		771SGSDC01	500		771SGSDC01	500
	371SGSCS04	300		707SGSDC01	500			
	771SGCDC01	100		371SGSDC01	500	485	771SGCDC01	100
	771SGSDC01	500		569SGSDC01	500		771SGSDC01	500
	707SGSDC01	500					707SGSDC01	500
	371SGSDC01	500	440	771SGCDC01	100		371SGSDC01	500
	569SGSDC01	500		771SGSDC01	500		569SGSDC01	500
				707SGSDC01	500			
413	771SGSCS01	520		371SGSDC01	500	486	371PADC I01	200
	776SGSCS02	500		569SGSDC01	500		371SGSDC01	500
				5690PCCI01	350		569SGSDC01	500
414	771SGSCS01	520	441	771SGCDC01	100		707SGSDC01	500
	776SGSCS01	500		771SGSDC01	500		771SGSDC01	500
	776SGSCS02	500						

LIST OF IDC'S VS NDA INSTRUMENT

IDC	NDA INSTRUMENT	CALIBRATION LIMIT (GRAMS)
487	771SGSDC01	500
	707SGSDC01	500
	371SGSDC01	500
	569SGSDC01	500
490	771SGCDC01	100
	771SGSDC01	500
	707SGSDC01	500
	371SGCDC01	500
	569SGSDC01	500
	771HEPAI01	20
	5690PCCI01	350
	569FACCI01	350
491	771SGSDC01	500
	707SGSDC01	500
	371SGSDC01	500
	569SGSDC01	500
	776LOSAC01	20
	5690PCCI01	350
	569FACCI01	350
654	776SGSCS01	300
803	371PADCI01	200
804	371PADCI01	200
806	771SGSDC01	500
	707SGSDC01	500
	371SGSDC01	500
	569SGSDC01	500
807	371PADCI01	200

LIST OF NDA INSTRUMENT VS IDC

NDA INSTRUMENT	IDC	CALIBRATION LIMIT (GRAMS)	NDA INSTRUMENT	IDC	CALIBRATION LIMIT (GRAMS)		
371PADCI01	320	200	371SGSDC01	441	500		
	330	200		442	500		
	336	200		479	500		
	339	200		480	500		
	341	200		481	500		
	480	200		484	500		
	481	200		485	500		
	484	200		486	500		
	486	200		487	500		
	803	SAFEGUARDS CHECK		490	500		
	804	SAFEGUARDS CHECK		491	500		
	807	SAFEGUARDS CHECK		806	500		
	371SGSCS04	392		300	5690PCCI01	003	350
		409		300		004	350
411		300	302	350			
418		300	330	350			
420		400	336	350			
424		400	337	350			
426		400	338	350			
471		300	374	350			
473		300	440	350			
			441	350			
371SGSDC01	290	500		442	350		
	292	500		480	350		
	300	500		481	350		
	301	500		490	350		
	302	500					
	303	500	569PACCI01	330	350		
	312	500		480	350		
	320	500		481	350		
	321	500		490	350		
	328	500		OTHERS	SAFEGUARDS CHECK		
	330	500					
	331	500	569SGSDC01	290	500		
	334	500		292	500		
	335	500		300	500		
	336	500		301	500		
	337	500		302	500		
	338	500		303	500		
	339	500		312	500		
	341	500		320	500		
	371	500		321	500		
	374	500		328	500		
	377	500		330	500		
378	500	331		500			
379	500	334		500			
411	500	335		500			
425	500	336	500				
429	500	337	500				
430	500	338	500				
431	500	339	500				
432	500	341	500				
440	500	371	500				

LIST OF NDA INSTRUMENT VS IDC

NDA INSTRUMENT	IDC	CALIBRATION LIMIT (GRAMS)	NDA INSTRUMENT	IDC	CALIBRATION LIMIT (GRAMS)	
569SGSDC01	374	500	707SGSDC01	440	500	
	377	500		441	500	
	378	500		442	500	
	379	500		479	500	
	411	500		480	500	
	425	500		481	500	
	429	500		484	500	
	430	500		485	500	
	431	500		486	500	
	432	500		487	500	
	440	500		490	500	
	441	500		491	500	
	442	500		806	500	
	479	500				
	480	500				
	481	500		771SGSCS01	310	250
	484	500			312	125
	435	500			368	150
	486	500			387	400
	487	500			392	400
490	500		398	400		
491	500		404	520		
806	500		405	520		
707SGSDC01	290	500		406	520	
	292	500		407	520	
	300	500		408	520	
	301	500		409	520	
	302	500		410	520	
	303	500		411	520	
	312	500		413	520	
	320	500		414	520	
	321	500		415	520	
	328	500		420	550	
	330	500	771SGSCS02	368	150	
	331	500		392	400	
	334	500		398	400	
	335	500		409	520	
	336	500		420	550	
	337	500				
	338	500	771SGSCS03	368	150	
	339	500		392	400	
	341	500		398	400	
	371	500		409	520	
374	500		420	550		
377	500					
378	500	771SGSCS04	368	150		
379	500		392	400		
411	500		398	400		
425	500		409	520		
429	500		420	550		
430	500					
431	500	771SGSDC01	292	100		
432	500		300	100		

LIST OF NDA INSTRUMENT VS IDC

NDA INSTRUMENT	IDC	CALIBRATION LIMIT (GRAMS)	NDA INSTRUMENT	IDC	CALIBRATION LIMIT (GRAMS)	
771SGDC01	301	100	771SGDC01	425	500	
	302	100		429	500	
	328	100		430	500	
	330	100		431	500	
	331	100		432	500	
	334	100		440	500	
	335	100		441	500	
	336	100		442	500	
	337	100		479	500	
	338	100		480	500	
	370	100		481	500	
	371	100		484	500	
	372	100		485	500	
	374	100		486	500	
	375	100		487	500	
	376	100		490	500	
	377	100		491	500	
	378	100		806	500	
	425	100				
	430	100				
	431	100				
	432	100				
	440	100				
	441	100				
	442	100				
	485	100				
	490	100				
771SGDC01	290	500	771PADCI01	NONE	TO BE DET'N	
	292	500	771HEPAI01	330	TO BE DET'N	
	300	500		336	TO BE DET'N	
	301	500		337	TO BE DET'N	
	302	500		490	TO BE DET'N	
	303	500	776SGSCS01	340	300	
	312	500		368	150	
	320	500		405	500	
	321	500		409	500	
	328	500		411	500	
	330	500		414	500	
	331	500		654	100	
	334	500		776SGSCS02	368	150
	335	500			409	500
	336	500			413	500
	337	500	414		500	
	338	500	429		500	
	339	500	776LOSAC01		330	20
	341	500		336	20	
	371	500		337	20	
	374	500		491	5	
	377	500				
	378	500				
	379	500				
	411	500				
	411	500				

Appendix G

MUF Study of the Idaho Burial Field and Rocky Flats Drum Storage Area



.CRF00538959S

**DRAFT INTERNAL REPORT
NOT CLEARED FOR PUBLICATION**

Note: 05/27/1998
The RFP library does not
have a copy. I surmise
that a final version was
not released - see internally,
at least on the title
or CRDL #.

R.G. Hurlow
5/27/98

MUF Study of the Idaho Burial Field
and Rocky Flats Drum Storage Area

CRDL 950442-115 (u)

May 1, 1973

L. A. Bidwell, R. N. Chanda,
and D. R. Cartwright

CHEMISTRY RESEARCH AND DEVELOPMENT

Chemistry Instrumentation

DOW CHEMICAL U.S.A.

Rocky Flats Division

Golden, Colorado

Distribution

KWIK INDEX

DECLASSIFICATION REVIEW		DETERMINATION (CIRCLE NUMBER(S))	
1ST REVIEW DATE: <u>5/1/73</u>	AUTHORITY: <input type="checkbox"/> AOC <input type="checkbox"/> ADC <input checked="" type="checkbox"/> ADD	1. CLASSIFICATION RETAINED	
NAME: <u>2-11-02</u>	2ND REVIEW DATE: <u>2-11-02</u>	2. CLASSIFICATION CHANGED TO: _____	
AUTHORITY: <u>ADD</u>	NAME: <u>R. J. Hoffman</u>	3. CONTAINS NO DOE CLASSIFIED INFO	
		4. COORDINATE WITH: _____	
		5. CLASSIFICATION CANCELLED	
		6. CLASSIFIED INFO BRACKETED	
		7. OTHER (SPECIFY): <u>EXTRACTION</u>	

from CRDL 950442-115

INTRODUCTION

The majority of waste generated at Rocky Flats is packaged in 55-gallon drums and passively assayed for plutonium. This waste is then either reprocessed, stored for reprocessing, or shipped off-site for burial depending upon the plutonium content. If a particular drum contains an amount of plutonium which falls below preset discard limits, it is shipped to Arco, Idaho for burial. Drums containing more plutonium than the specified discard limits and not immediately reprocessed are temporarily stored at Rocky Flats.

The first drum counter was installed in 1964, and was the first drum counter at an AEC facility used for plutonium accountability. Since that time drum counter utilization and technology have rapidly advanced. Second generation systems, initially installed in 1970, are capable of more accurately analyzing drum contents. Furthermore, better standards are available which more closely simulate production generated waste. Diagnostic checks on input data; e.g., background, standards, sample data, etc., are presently computer controlled. As these were not incorporated into the 1964 system, the system was subject to possible human and procedural errors.

A study was undertaken to critically review all data on 24,000 drums assayed with the 1964 system during a three-year operational period (February, 1968 to June, 1971). It was hoped that procedural and analytical errors could be pinpointed.

The following questions were asked:

1. By recounting selected drums in the newer drum counting systems, can ~~a~~ more accurate values be derived for the material stored on-site as well as that shipped to Idaho?
2. Is the gamma-ray and/or neutron count data in reasonable agreement with the reported plutonium weight?
3. Is the gamma/neutron (γ/n) ratio of the sample drum reasonable and consistent with the reported plutonium weight?
4. Was the counter functioning properly for that day: i.e., are the count rates of the standards reasonable or close to the long term average?
5. Did a barrel get shipped off-site whose plutonium value was grossly miscalculated?

Before answering these questions, it is necessary to understand how the old drum counter operated and also what nuclear radiations were being counted.

PLUTONIUM ASSAY BY THE 1964 DRUM COUNTER

The design of the original or 1964 (old) drum counter has been described previously¹. Gamma rays were detected by eight symmetrically positioned Geiger-Mueller tubes² and neutrons by 16 boron trifluoride (BF₃) tubes. In order to calibrate or standardize the counter, 50 gram plutonium oxide and fluoride standards were placed in the center of an empty drum and the per gram response determined. All samples were then assayed using the above "standard" data and equation (1) below. However, matrices other than air altered the counter response. Therefore, additional standard drums were counted which contained varying weights of plutonium metal, oxide, and fluoride dispersed in a variety of matrices or waste categories. A straight line "correction" was thus obtained defined by an intercept and slope [equation (2)]. No matrix attenuation correction was used in the old counter which assumed that all production drums "look" the same as the standard drums.

The functional form of the equation for plutonium weight calculation was:

$$\text{grams Pu} = \bar{x}(\gamma_{\text{sample}} - \gamma_{\text{bkg}}) - y(n_{\text{sample}} - n_{\text{bkg}}) \quad (1)$$

where x, y were empirically determined constants from the 50 gram oxide and fluoride standards (after every six samples were counted, the two 50 gram working standards were recounted and appropriate adjustments made to these constants),

γ_{sample} = gamma-ray count of the sample,

γ_{bkg} = gamma-ray background.

n_{sample} = neutron count of the sample, and

n_{bkg} = neutron background.

A separate set of constants (X.Y) were determined for the oxide and fluoride standards. The set used in analyzing an unknown sample was determined from the γ/n ratio of the sample.

In order to correct for varying matrices, the plutonium weight calculated from equation 1) was corrected by:

$$P = mQ + b \quad (2)$$

where P = corrected weight of plutonium,

m = experimental slope of the calibration curve.

b = experimental intercept of the calibration curve. and

Q = observed weight of plutonium from equation (1).

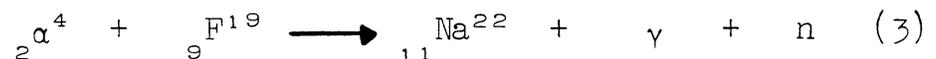
Several problem areas existed because of the method of calculation used:

- a. The γ/n ratio determined whether the sample was a fluoride or an oxide. Neutrons from (α,n) reactions other than fluoride (see following section) could cause the sample to be calculated as a fluoride with a resulting low plutonium assay.
- b. For drums containing less than 10 grams plutonium, the plutonium assay was based only upon equation (1). A small error could be introduced here, especially for the heavier matrices.

NUCLEAR REACTIONS OF PuO₂ and PuF₄

To facilitate an understanding of the count rates observed for a given drum, a brief summary of the nuclear reactions occurring within a drum is essential. All remarks specifically apply to WR plutonium processed at Rocky Flats.

The major sources of gamma rays (γ) and neutrons (n) for both PuO₂ and PuF₄ are shown in Table I. For both chemical forms of plutonium, gamma rays originate from the alpha particle decay of ²³⁹Pu, and neutrons come from the spontaneous fission of ²⁴⁰Pu. However, a nuclear reaction occurs with fluorine if the fluorine atom is chemically bonded to a plutonium atom as in PuF₄. This reaction yields high energy gamma rays (>1 MeV) and neutrons:



The Geiger-Mueller tubes utilized in the original drum counter did not discriminate between the 384 keV gamma rays from ²³⁹Pu and the high energy gamma rays from the (α, n) reaction on fluorine. Therefore, extra PuF₄ gamma rays, both full energy and Compton scattered, would be detected. As a result, the gamma-ray count for a drum containing PuF₄ was approximately twice that for a drum containing PuO₂. The additional neutrons from PuF₄ amount to about 40 times the number from PuO₂. It should be noted that an ($\alpha, n\gamma$) reaction occurs with oxygen in PuO₂, but the cross-section is so low that the neutrons and gamma rays produced are insignificant for this discussion.

The Geiger-Mueller tubes were count-rate sensitive; i.e.,

Table I

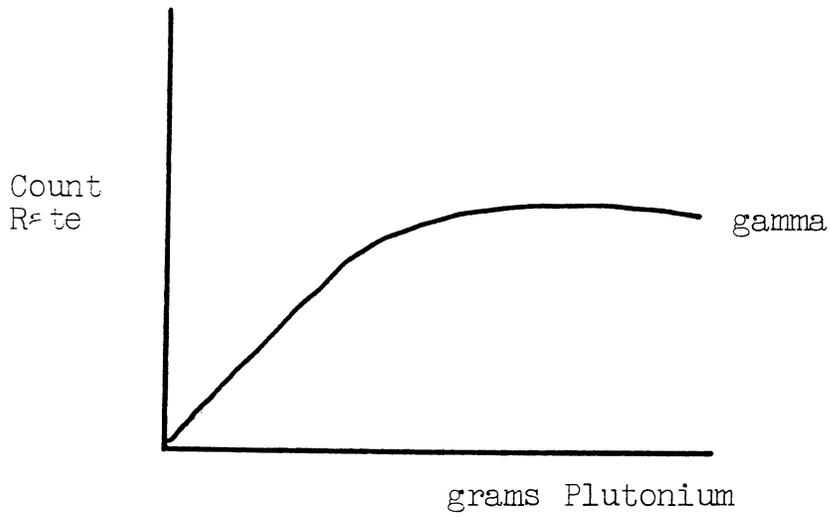
Sources of Gamma rays and Neutrons from PuO_2 and PuF_4

Chemical Form	Gamma rays	Neutrons
PuO_2	Pu-239 decay	Pu-240 fission
PuF_4	Pu-239 decay + $\alpha, n\gamma$ reactions	Pu-240 fission + α, n reactions

above a particular count rate (about 1,000 counts/second) the detector response was not rapid enough to record individual counts (Figure 1). An experimentally determined factor (dead time correction (τ)) must therefore be applied to the total count to correct for this dead time loss. This factor increased with increasing count rate and was incorporated into the plutonium weight calculations.

Figure 1

Schematic Representation of Plutonium
Geiger-Mueller Tube Response versus Grams ~~Pa~~



8

COMPUTER PROCESSING OF DRUM COUNTER DATA

All old drum counter data obtained during the interval February, 1968 through June, 1971 were reviewed (approximately 24,000 drums). For ease of handling, the following data were keypunched onto cards for processing by the IBM-360 computer:

1. fluoride standard gamma and neutron counts.
2. sample gamma and neutron counts.
3. waste category (MS Code).
4. ~~final~~ calculated plutonium weight, and
5. identifying information (sample and document numbers).

All keypunched cards were listed and the data verified with the material transfer forms. Lack of space on the cards prevented the inclusion of the oxide standard data.

Several FORTRAN programs were written to sort, calculate, and evaluate the information contained on these data cards. Initial sorting was by identification number to determine whether any material may have been counted on the Can Counter. Material so identified was immediately discarded, as primary interest was focused on drum counted material.

Secondary sorting was on the basis of the Chemical Operations plutonium inventory account. A cross-verification by sample number identified all barrels greater than discard in the Rocky Flats storage area. Of these, some 300 barrels that had been "split" were eliminated, because a split barrel was reassigned an identification number and the old number voided on the inventory.

Since a specific list ~~by number~~ of drums shipped to Idaho was

unavailable, those drums located in the burial field at Arco, Idaho were determined by the process of elimination. This was accomplished by making the assumption that only those drums whose material transfer form states "LESS THAN AEC DISCARD LIMITS" were shipped off-site.

The pertinent data had now been categorized into two groups:

1. those drums located in Rocky Flats storage, and
2. those barrels shipped to Idaho for burial.

Subsequently, these two data sets were handled in a similar manner.

To determine calculational errors in drum contents, further sorting of both data sets was done by MS Code and plutonium weight. Within a given MS Code the data were arranged according to increasing plutonium weight. When arranged in this manner, the count rates (gamma and neutron) and plutonium weight should show the same rate of increase. Exceptions were noted.

The γ/n ratio was also calculated. An average count rate of the 50 gram standards, PuO_2 and PuF_4 , was used to obtain theoretical limits on the γ/n ratios of 3.20 and 0.18, respectively. Exceptions, or those drums with a γ/n ratio outside this range, were noted. An estimate of the plutonium content of these drums was obtained by considering the two extremes; i.e., the drum contained all PuO_2 , or all PuF_4 . The gamma per gram (γ/g) factors were obtained from an averaged count rate of the 50 gram standards. The gamma/gram factors

were 1100 and 1800 for PuO_2 and PuF_4 , respectively.

Results of these calculational efforts are presented in the following two sections.

ARCO BURIAL FIELD

Of the 24,000 drums counted during the selected time interval by the process of elimination approximately 16,000 were determined to have been sent to Arco, Idaho. The data for these drums were examined in the manner discussed in the previous section; i.e., "suspect" drums were calculated separately and a recount correction factor applied to several other categories.

A computer listing by category and increasing plutonium weight was searched for "suspect" drums. The resulting 60 "suspect" or erroneously calculated (EC) drums are presented in Table VIII. Original count data, identification, MS Code, and reported plutonium contents are given.

Several factors must be considered in an attempt to estimate the contents of the 60 EC drums based on the original count data. They are:

1. No matrix attenuation correction was made in the "old" drum counter calculation. Attenuation of the gamma-ray counts by the matrix tends to bias the plutonium weight calculation low, especially for densely packaged materials such as fire brick (371), sand, slag, and crucible (392), and scrap metal (480).
2. Drums containing more than 100 grams of plutonium typically have a total count in excess of 100,000 for the 100-second counting interval. The Geiger-Mueller tubes utilized in the "old" drum counter are particularly insensitive at these high count rates, especially compared to the more recent systems utilizing sodium iodide detectors. The

- experimentally determined dead time (τ) of 200 microseconds appears to be too low at count rates above 100,000 counts/100 seconds. Therefore, those drums with counts \geq 100,000 would calculate low.
3. The assumption that sample and standard drums are packaged in an analogous manner is assumed true during the standardization and calibration of the drum counting systems. This assumption has been proven invalid for the new sodium iodide gamma-ray detection systems for at least three categories of waste/scrap material. Lead gloves (339) were repackaged in 1972 to show a previous low bias of 40%. Graphite (300) had been biased at least 10% low prior to 1972. Washables (460) had been packaged to include lead gloves which would attenuate the emitted gamma rays and thus bias the calculated plutonium weights low. Furthermore, the "old" drum counter was more sensitive to these differences than the newer systems. On this basis, one must conclude that at least three categories of waste counted during the selected time interval for this study must be biased low for reasons not covered in items 1 and 2 above.
 4. Although corrections have been made to the plutonium calculations for electronic drift and noise, the electronics utilized in the "old" drum counter are not the state-of-the-art electronics compared to those used in the newer systems, particularly with respect to repetition rates and electronic stability.
 5. Diagnostic checks on uncorrected count data had not been

incorporated into the computer program for the "old" drum counter during the time interval selected for this study. Therefore, if the count rates were too high for either the sample or working standard, the only means for flagging them was through visual inspection, which has a high probability of being subject to human error.

The actual calculation of the individual drums was performed taking into account the factors mentioned in items 1 through 5. Since it was not possible to recount these 60 drums on newer counting systems, the estimation of plutonium contents and application of appropriate numerical factors was subject to the discretion of the authors. Due to variability of the count data a uniform correction was not applied to all drums. Rather, a certain amount of "experience" was incorporated into the final determination of the estimated plutonium contents.

The estimates for these 60 EC drums are presented in Table VIII. As a check on the validity of these estimates, particularly for those drums containing more than 100 grams of plutonium, two previously flagged drums are included in the table (57493: 476 g estimated, 500 g assayed; 59682: 319 g estimated, 300 g assayed). The assay values are those independently measured by J. E. Cline at the NRTS in Idaho.

Six barrels (48804, 48458, 52427, 48622, 48651, and 51662) were flagged by the process of elimination and thus determined to have been shipped to Arco, Idaho. However, further investigation

showed that there is no record of these six barrels having been shipped off-site. Therefore, one must assume that they were flagged at the time they were counted and consequently returned to the production facility for either reprocessing or repackaging. If they were repackaged, a new identification number was assigned to the barrel.

These six barrels have been included in Table VIII, but their reported and estimated plutonium contents have not been incorporated into the subtotals nor the total MUF figure reported for Arco.

The total MUF at Arco due to the 54 remaining barrels is estimated at 5.2 kg.

The remainder of the drums at Arco are summarized by category in Table IX. The correction factors that were used in the "old" drum counter are tabulated. Currently assigned correction factors are those shown in Table VI (recount data).

The correction factor for densely packaged material; i.e., Code 392 was the same as for Code 480 for the reasons described in the previous section.

Application of these factors to the totaled plutonium weights for all categories showed a 22% increase (15.4 kg) in the amount of material shipped to Idaho.

Table IX

Remaining Drums at Arco

MS Code	Number of Drums	SS Weight (grams)	Old Drum Counter Factors		New Correction Factor	Corrected SS Weight (grams)
			A	B		
290	48	1652	0	1	1.0	1652
300,310	428	4881	13.34	1.57	1.1	5369
320	32	204	0	1	1.88	384
330,331, 335	6031	8943	25.67	0.81	1.0	8943
336,337	2347	2330	0	1.4	1.0	2330
338	26	1905	0	1	1.0	1905
339	224	4232	0	1	2.44	10326
371,372	381	11022	0	1	1.48 ^P	16313
391	5	1	0	1	1.48	1.5
410	18	3249	0	1	1.0	3249
432	142	5763	0	1	1.0	5763
440,441	571	2109	7.37-0 (0-200g)	1-2.05	1.0	2109
422	3	234	0	1	1.0	234
460,461, 464(discontinued)	753	6035	3.72	1.21	1.0	6035
480,481	1996	11666	0	1	1.48	17266
490	2788	6021	1.25	1.16	1.0	6021
Totals:	15795	70249				87897

Appendix H
Plutonium Waste Discards CY67

PLUTONIUM WASTE DISCARDS CY67

CRDL - 940561-102

D. L. Ziegler

July 18, 1968

Chemistry R and D
Chemical Technology

THE DOW CHEMICAL COMPANY
Rocky Flats Division
Golden, Colorado

Distribution

✓ E. A. Putzier
M. A. Thompson
L. D. Hazelton
K. W. Calkins
A. K. Williams
R. W. Woodard
E. S. Ryan
IRF (Record)

REVIEWED FOR CLASSIFICATION/UCM
By *JAT*
Date *07-29-02* U/NU

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INTRODUCTION

All unclassified plutonium contaminated waste generated by the Rocky Flats Plant are shipped to Idaho for burial. Because of the changes in the shipping regulations of radioactive wastes, it is desirable to have an updated compilation of the wastes generated by this plant. This report contains information about the quantity and the plutonium concentration of the various types of wastes generated during the calendar year 1967.

SUMMARY

Nuclear Materials Management recorded a discard of 70,382g of plutonium in CY67. The wastes tabulated in this report amount to 67,588g of plutonium. A total of 957g of plutonium were charged off due to decay to americium. The difference of 1,837g of Pu ($70,382 - 67,588 - 957 = 1,837$) is due to the time when the material is shipped from the various buildings versus when it is actually charged off by Nuclear Materials Management.

The following table shows the amounts of the various types of wastes:

	Volume		Weight (lb)		Plutonium (g)
	<u>gal.</u>	<u>ft³</u>	<u>Gross</u>	<u>Net</u>	
Liquid Waste* from 771	270,895	--	--	--	8,125
Carbon Tetrachloride* - cutting oil	16,840	--	--	--	829
Line Generated	--	13,282	306,578	216,228	54,878
Hot Waste	--	58,303	1,196,155	799,455	--
Crated Waste	--	57,326	--	--	3,756
Bldg. 774 Waste	--	30,780	1,865,880	--	--
Grease	--	40,558	3,178,314	--	--
Empty Oil Drums	--	<u>14,317</u>	<u>128,403</u>	<u>--</u>	<u>--</u>
Total	--	214,566	6,675,330	--	67,538
Pu Decay					<u>957</u>
Total Charged Off					68,545

*The Pu is charged off in these streams but the material is shipped from the plant site as Building 774 waste (Stage 1 sludge, Stage 2 sludge, cemented liquid and evaporator salts) and as grease.

DISCUSSION

The wastes are broken down into the following six categories:

1. Liquid waste from the plutonium recovery area.
(Bldg. 771).
2. Wastes from the waste treatment plant (Bldg. 774).
3. Carbon tetrachloride - cutting oil from the fabrication area (Bldg. 776).
4. Non-line generated wastes from all areas of the plant site.
5. Line-generated waste from all areas where plutonium is handled in glove box lines.
6. Crated waste from all areas of the plant.

The total amount of plutonium in the tabulated data is 67,588g. Nuclear Materials Management recorded 70,382g of plutonium discarded during this same period of time (CY67). This 70,382g included a total of 957g of plutonium charged off due to plutonium decay to Americium. Adding the plutonium decay to that tabulated gives a value of 68,545g which is less by $(70,382 - 68,545 = 1,837\text{g})$ 1,837g than that recorded by NMM. This discrepancy is due mainly to the difference between when the material leaves the various buildings and when it is actually charged off by NMM.

Liquid Waste from Building 771

This data was obtained from the Waste Treatment Facility where the volume and Pu content of each incoming stream from Building 771 is totaled by the month. This information is tabulated in Table 1. The liquid waste amounted to 780,895 gal. of solution containing 8,125g of plutonium for CY67.

Waste from the Waste Treatment Plant (Bldg. 774)

The liquid wastes from Building 771 are converted to solid form in the waste treatment plant. The liquid wastes either go through a precipitation and evaporation process or are solidified with cement. Three products came from the precipitation and evaporator salts. The majority of the plutonium entering the process goes to stage 1 sludge and very little goes to the stage 2 sludge and the evaporator salts. The evaporation step consists of two parts: solar evaporation and forced evaporation. Table 2 contains the information about the Building 774 wastes. The miscellaneous items under the headings Stage 2 sludge and cemented liquid are the empty bottles which are used as containers for transporting same solutions to Building 774. The gross weight of the waste from Building 774 was 526,984 lb and the volume was 8,923 ft³ for CY67.

Carbon Tetrachloride - Cutting Oil Waste

The amount and plutonium content of the carbon tetrachloride - cutting oil waste was obtained from Nuclear Materials Management. The data for this type of waste is tabulated in Table 5. The amount of grease and number of empty drums was obtained from the Waste Treatment Facility (Bldg. 774). Two drums of CCl₄-oil will produce about three drums of grease. The amount of grease shipped from the plant site (40,558 gal.) is much larger than the CCl₄-oil generated (16,240 gal.) because back-log material was being worked off during this period. The amount of plutonium discarded in the CCl₄-oil generated was 829g for CY67.

Non-Line Generated Waste

This material is such things as rags and paper which are generated outside of the glove box lines. No measurable amount of plutonium was discarded in this category. The total gross weight of hot waste generated on the plant site in CY67 was 1,196,155 lb. and the volume was 58,303 ft³. This information is shown in Table 4.

Line Generated Waste

This category of waste is material which is generated in the glove box lines. The majority of this comes from the plutonium recovery line in Building 771. Some of it is residues like graphite (generated in other buildings) which

has had the plutonium removed prior to discard. Some of it is generated in the recovery process such as ion exchange resins, discard equipment, and glass ware. The laboratories and R and D areas also generate material which contribute to this category of waste. Table 5 contains the data on this type of waste discarded in CY67. Table 5 gives a break down on the amount of waste that is $> 15g$ Pu per drum and that $\leq 15g$ Pu per drum. In CY67, 54,378g plutonium was discarded in this category. This represents about 80% of the total plutonium discarded from the plant. Most of it (about 49,000g Pu) is in the $> 15g/\text{drum}$ group. The total gross weight of ~~line~~ generated waste was 306,578 lb. in CY67 and the volume was 13,202 ft³. Information for this category was obtained from log books where the information is recorded as the material leaves the various buildings.

Crated Waste

A total of 159 crates which contained estimated amounts of plutonium were shipped from the plant in CY67. The other 279 crates contained nil plutonium or came from non plutonium areas. This information was obtained from log books where information is recorded as it leaves the various buildings (771, 776, 777, and 779) and from a record of crates which have been shipped from the plant site (non plutonium areas).

The data for crated waste are tabulated in Table 6. A total of 57,326 ft³ of crated waste were generated in CY87. This is such things as process equipment and building materials which are too large to be put in a 55 gal. drum.

TABLE 1

Liquid Waste from the Plutonium Recovery Area CY67

Date (Qtr. CY 67)	Caustic Scrub Solution			Ion Column Effluent			Distillate			Americium Ion Column Effluent		
	Volume (gal.)	Pu (g)	Concentration (g/l)	Volume (gal.)	Pu (g)	Concentration (g/l)	Volume (gal.)	Pu (g)	Concentration (g/l)	Volume (gal.)	Pu (g)	Concentration (g/l)
1st	20,127	510.2	6.7×10^{-3}	75,339	1,162.3	4.1×10^{-3}	19,245	188.6	2.6×10^{-3}	8,753	64.2	1.9×10^{-3}
2nd	27,233	538.0	5.2×10^{-3}	67,786	800.9	3.1×10^{-3}	29,025	281.5	2.6×10^{-3}	3,966	37.3	2.5×10^{-3}
3rd	31,907	529.4	4.4×10^{-3}	58,412	395.0	1.8×10^{-3}	22,282	171.9	2.0×10^{-3}	4,004	70.6	2.7×10^{-3}
4th	26,209	345.2	3.5×10^{-3}	69,499	432.3	1.6×10^{-3}	24,667	371.8	4.1×10^{-3}	5,758	27.3	1.3×10^{-3}
TOTAL	105,476	1,922.8	4.8×10^{-3}	271,136	2,790.5	2.7×10^{-3}	95,219	1,013.8	2.8×10^{-3}	22,481	169.4	2.0×10^{-3}

Date (Qtr. CY 67)	HCL Solutions			Misc. Solutions			Cooling Water			Total of all Solutions		
	Volume (gal.)	Pu (g)	Concentration (g/l)	Volume (gal.)	Pu (g)	Concentration (g/l)	Volume (gal.)	Pu (g)	Concentration (g/l)	Volume (gal.)	Pu (g)	Concentration (g/l)
1st	2,244	95.1	1.1×10^{-2}	3,397	44.3	3.4×10^{-3}	61,417	8.5	3.7×10^{-5}	190,522	2073.2	2.9×10^{-3}
2nd	793	32.7	1.1×10^{-2}	5,940	203.3	9.0×10^{-3}	65,198	33.6	1.4×10^{-4}	199,941	1927.3	2.5×10^{-3}
3rd	—	—	—	7,605	356.1	1.2×10^{-2}	62,858	14.2	5.9×10^{-5}	187,068	1507.2	2.1×10^{-3}
4th	40	1.2	7.9×10^{-3}	11,625	1,408.7	3.2×10^{-2}	65,566	30.5	1.2×10^{-4}	203,364	2617.0	3.4×10^{-3}
TOTAL	3,077	129.0	1.1×10^{-2}	28,567	2,012.4	2.1×10^{-2}	255,039	86.8	9.0×10^{-5}	780,895	8124.7	2.7×10^{-3}

TABLE 2
Building 774 Waste CY67

Date (Qtr. CY 67)	<u>Stage 1 Sludge</u>			<u>Stage 2 Sludge</u>					
	Drums (No.)	Vol. ³ (ft.)	Gross wt. (lb.)	<u>Sludge</u>			<u>Misc.</u>		
				Drums (No.)	Vol. ³ (ft.)	Gross wt. (lb.)	Drums (No.)	Vol. ³ (ft.)	Gross wt. (lb.)
1st	393	2,492	177,366	221	1,624	111,603	110	808	24,469
2nd	235	1,727	114,220	348	2,558	171,832	33	243	4,764
3rd	225	1,654	110,839	341	2,506	159,552	36	265	4,829
4th	<u>239</u>	<u>1,757</u>	<u>120,708</u>	<u>229</u>	<u>2,193</u>	<u>152,721</u>	<u>18</u>	<u>132</u>	<u>2,002</u>
TOTAL	1,038	7,630	523,133	1,209	8,886	595,713	197	1,448	36,064

Date (Qtr. CY 67)	<u>Cemented Liquid</u>						<u>Evaporator Salts</u>		
	<u>Cemented Liquid</u>			<u>Misc.</u>			Drums (No.)	Vol. ³ (ft.)	Gross wt. (lb.)
	Drums (No.)	Vol. ³ (ft.)	Gross wt. (lb.)	Drums (No.)	Vol. ³ (ft.)	Gross wt. (lb.)			
1st	161	1,183	76,447	48	353	4,033	6	44	2,666
2nd	107	786	44,995	33	243	2,956	16	118	6,335
3rd	27	198	12,090	30	220	2,743	350	2,572	160,031
4th	<u>80</u>	<u>588</u>	<u>36,773</u>	<u>37</u>	<u>272</u>	<u>3,949</u>	<u>842</u>	<u>6,189</u>	<u>357,952</u>
TOTAL	375	2,755	170,305	148	1,088	13,681	1,214	8,923	526,984

TABLE 3

CCl₄-Oil Converted to Grease in Building 774 CY67

Date (Qtr. CY 67)	CCl ₄ -Oil Generated		Grease Produced from CCl ₄ -Oil (Predominantly Backlog Material)			Empty Grease Drums		
	Volume (gal.)	Pu (g)	No. Drums	Volume (ft ³)	Gross Weight (lb.)	No. Drums	Volume (ft ³)	Gross Weight (lb.)
1st	5,070	263	979	7,196	516,410	627	4,608	37,156
2nd	5,740	199	1,625	11,944	955,398	440	3,234	26,210
3rd	2,800	75	1,538	11,304	899,929	547	4,020	39,347
4th	<u>3,230</u>	<u>292</u>	<u>1,376</u>	<u>10,114</u>	<u>806,579</u>	<u>334</u>	<u>2,455</u>	<u>25,690</u>
TOTAL	16,840	829	5,518	40,558	3,178,314	1,948	14,317	128,403

TABLE 4

Non-Line Generated Hot Waste CY67
(Buildings 771, 776, 777, and 779 only)

<u>Date (Qtr. CY 67)</u>	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gross Weight (lb.)</u>	<u>Net Weight (lb.)</u>
1st	1,485	10,915	183,064	108,814
2nd	1,349	9,915	161,252	93,802
3rd	1,357	9,974	126,237	58,387
4th	<u>1,565</u>	<u>11,503</u>	<u>162,056</u>	<u>83,806</u>
TOTAL	5,756	42,307	632,609	344,809

Other Non-Line Generated Hot Waste
(Buildings 222, 223, 331, 444, 881, 883, 886 and 991)

<u>Date (Qtr. CY 67)</u>	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gross Weight (lb.)</u>	<u>Net Weight (lb.)</u>
Jan-June 67	1,096	8,046	285,007	230,207
July-Dec. 67	<u>1,082</u>	<u>7,950</u>	<u>278,539</u>	<u>224,439</u>
TOTAL	2,178	15,996	563,546	454,646

TABL

Line Generated Waste for CY 67
(Building 771, 776 and 779)

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gross wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concentration (g/g)</u>
<u>Graphite</u>						
<u>>15 g/drum</u>						
1st Qtr.	8	59	2,341	1,941	994	1.1x10 ⁻³
2nd Qtr.	36	265	10,446	8,646	5,893	1.5x10 ⁻³
3rd Qtr.	87	639	15,619	11,269	6,120	1.2x10 ⁻³
4th Qtr.	<u>66</u>	<u>485</u>	<u>17,274</u>	<u>13,974</u>	<u>6,195</u>	1.0x10 ⁻³
TOTAL	197	1,448	45,680	35,830	19,202	1.2x10 ⁻³
<u>Graphite</u>						
<u>≤ 15 g/drum</u>						
1st Qtr.	35	257	8,087	6,337	228	7.9x10 ⁻⁵
2nd Qtr.	15	110	3,851	3,101	112	8.0x10 ⁻⁵
3rd Qtr.	93	684	14,569	9,919	836	1.9x10 ⁻⁴
4th Qtr.	<u>85</u>	<u>625</u>	<u>25,111</u>	<u>20,861</u>	<u>833</u>	8.8x10 ⁻⁵
TOTAL	228	1,676	51,618	40,218	2,009	1.1x10 ⁻⁴
<u>Sand Slag and Crucible</u>						
<u>>15 g/drum</u>						
1st Qtr.	6	44	1,523	1,223	1,955	3.5x10 ⁻³
2nd Qtr.	9	66	2,342	1,892	3,229	3.8x10 ⁻³
3rd Qtr.	2	15	526	426	572	3.0x10 ⁻³
4th Qtr.	<u>4</u>	<u>29</u>	<u>1,154</u>	<u>954</u>	<u>1,192</u>	2.8x10 ⁻³
TOTAL	21	154	5,545	4,495	6,948	3.4x10 ⁻³

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gr. wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Fu (g)</u>	<u>Concentration (g/g)</u>
Sand Slag and Crucible <u>≤ 15 g/drum</u>	None					
Insulation <u>> 15 g/drum</u>						
1st Qtr.	1	7	123	73	37	1.1x10 ⁻³
2nd Qtr.	60	441	6,316	3,316	5,474	3.6x10 ⁻³
3rd Qtr.	71	522	6,826	3,276	3,745	2.5x10 ⁻³
4th Qtr.	<u>76</u>	<u>559</u>	<u>8,315</u>	<u>4,515</u>	<u>2,877</u>	1.5x10 ⁻³
TOTAL	208	1,529	21,580	11,180	12,233	2.4x10 ⁻³
Insulation <u>≤ 15 g/drum</u>						
1st Qtr.	--	--	--	--	--	--
2nd Qtr.	47	345	5,432	3,082	309	2.2x10 ⁻⁴
3rd Qtr.	51	375	6,059	3,509	365	2.3x10 ⁻⁴
4th Qtr.	<u>10</u>	<u>74</u>	<u>919</u>	<u>419</u>	<u>105</u>	5.5x10 ⁻⁴
TOTAL	108	794	12,410	7,010	779	2.4x10 ⁻⁴
Fire Brick <u>> 15 g/drum</u>						
1st Qtr.	--	--	--	--	--	--
2nd Qtr.	1	7	122	72	224	6.9x10 ⁻³
3rd Qtr.	--	--	--	--	--	--
4th Qtr.	<u>2</u>	<u>15</u>	<u>412</u>	<u>312</u>	<u>131</u>	9.3x10 ⁻⁴
TOTAL	3	22	534	384	355	2.0x10 ⁻³
Fire Brick <u>≤ 15 g/drum</u>	None					

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>G wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concen. - tion (g/g)</u>
<u>Resin</u> <u>> 15 g/drum</u>						
1st Qtr.	3	22	390	240	514	4.7x10 ⁻³
2nd Qtr.	4	29	626	426	345	1.8x10 ⁻³
3rd Qtr.	2	15	256	156	34	4.8x10 ⁻⁴
4th Qtr.	<u>6</u>	<u>44</u>	<u>1,022</u>	<u>722</u>	<u>244</u>	7.5x10 ⁻⁴
TOTAL	15	110	2,294	1,544	1,137	1.6x10 ⁻³
<u>Resin</u> <u>≤ 15 g/drum</u>						
1st Qtr.	--	--	--	--	--	--
2nd Qtr.	4	29	342	142	31	4.8x10 ⁻⁴
3rd Qtr.	5	37	734	484	39	1.8x10 ⁻⁴
4th Qtr.	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	<u>--</u>	--
TOTAL	9	66	1,076	626	70	2.5x10 ⁻⁴
<u>Scrap Metal</u> <u>> 15 g/drum</u>						
1st Qtr.	9	66	2,041	1,591	213	3.0x10 ⁻⁴
2nd Qtr.	5	37	1,240	990	104	2.3x10 ⁻⁴
3rd Qtr.	8	59	2,057	1,657	170	2.3x10 ⁻⁴
4th Qtr.	<u>5</u>	<u>37</u>	<u>1,028</u>	<u>778</u>	<u>90</u>	2.6x10 ⁻⁴
TOTAL	27	199	6,366	5,016	577	2.5x10 ⁻⁴
<u>Scrap Metal</u> <u>≤ 15 g/drum</u>						
1st Qtr.	92	676	14,469	9,869	504	1.1x10 ⁻⁴
2nd Qtr.	63	463	10,759	7,609	327	9.5x10 ⁻⁵
3rd Qtr.	44	323	8,314	6,114	132	4.8x10 ⁻⁵
4th Qtr.	<u>47</u>	<u>345</u>	<u>7,532</u>	<u>5,182</u>	<u>233</u>	9.9x10 ⁻⁵
TOTAL	246	1,807	41,074	28,774	1,196	9.2x10 ⁻⁵

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gr wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concent ion (g/g)</u>
<u>Glass</u> <u>> 15 g/drum</u>						
1st Qtr.	8	59	1,617	1,217	326	5.9x10 ⁻⁴
2nd Qtr.	2	15	500	400	39	2.1x10 ⁻⁴
3rd Qtr.	4	29	890	690	74	2.4x10 ⁻⁴
4th Qtr.	<u>11</u>	<u>81</u>	<u>2,423</u>	<u>1,873</u>	<u>341</u>	<u>4.0x10⁻⁴</u>
TOTAL	25	184	5,430	4,180	780	4.1x10 ⁻⁴
<u>Glass</u> <u>≤ 15 g/drum</u>						
1st Qtr.	12	88	2,085	1,485	103	1.5x10 ⁻⁴
2nd Qtr.	34	250	7,809	6,109	155	5.6x10 ⁻⁵
3rd Qtr.	10	74	1,985	1,485	77	1.1x10 ⁻⁴
4th Qtr.	<u>20</u>	<u>147</u>	<u>4,314</u>	<u>3,314</u>	<u>120</u>	<u>7.8x10⁻⁵</u>
TOTAL	76	559	16,193	12,393	455	8.1x10 ⁻⁵
<u>Washables</u> <u>> 15 g/drum</u>						
1st Qtr.	5	37	1,007	757	105	3.1x10 ⁻⁴
2nd Qtr.	22	162	4,012	2,912	516	3.9x10 ⁻⁴
3rd Qtr.	26	191	5,123	3,823	760	4.4x10 ⁻⁴
4th Qtr.	<u>70</u>	<u>515</u>	<u>16,552</u>	<u>13,052</u>	<u>2,006</u>	<u>3.4x10⁻⁴</u>
TOTAL	123	905	26,694	20,544	3,387	3.6x10 ⁻⁴
<u>Washables</u> <u>≤ 15 g/drum</u>						
1st Qtr.	38	279	5,996	4,096	321	1.7x10 ⁻⁴
2nd Qtr.	57	419	7,782	4,932	458	2.0x10 ⁻⁴
3rd Qtr.	44	323	5,357	3,157	241	1.7x10 ⁻⁴
4th Qtr.	<u>56</u>	<u>412</u>	<u>7,503</u>	<u>4,703</u>	<u>292</u>	<u>1.4x10⁻⁴</u>
TOTAL	195	1,433	26,638	16,888	1,312	1.7x10 ⁻⁴

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>G wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concentration (g/g)</u>
<u>Combustibles > 15 g/drum</u>						
1st Qtr.	1	7	212	162	40	5.4x10 ⁻⁴
2nd Qtr.	1	7	110	60	16	5.9x10 ⁻⁴
3rd Qtr.	2	15	192	92	41	9.8x10 ⁻⁴
4th Qtr.	2	15	240	140	43	6.8x10 ⁻⁴
TOTAL	6	44	754	454	140	6.2x10 ⁻⁴
<u>Combustibles ≤ 15 g/drum</u>						
1st Qtr.	2	15	156	56	1	3.9x10 ⁻⁵
2nd Qtr.	34	250	2,892	1,192	178	3.3x10 ⁻⁴
3rd Qtr.	19	140	2,052	1,102	75	1.5x10 ⁻⁵
4th Qtr.	52	382	6,357	3,757	254	1.5x10 ⁻⁴
TOTAL	107	787	11,457	6,107	508	1.8x10 ⁻⁴
<u>Line Generated > 15 g/drum</u>						
1st Qtr.	1	7	191	141	25	3.9x10 ⁻⁴
2nd Qtr.	2	15	345	245	67	6.0x10 ⁻⁴
3rd Qtr.	--	--	--	--	--	--
4th Qtr.	3	22	360	210	95	1.0x10 ⁻³
TOTAL	6	44	896	596	187	6.9x10 ⁻⁴
<u>Line Generated ≤ 15 g/drum</u>						
1st Qtr.	14	103	1,928	1,228	88	1.6x10 ⁻⁴
2nd Qtr.	10	74	1,142	642	23	7.9x10 ⁻⁵
3rd Qtr.	11	81	1,260	710	30	9.3x10 ⁻⁵
4th Qtr.	2	15	262	162	16	2.2x10 ⁻⁴
TOTAL	37	273	4,592	2,742	157	1.3x10 ⁻⁴

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>C wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concentration (g/g)</u>
<u>Filters > 15 g/drum</u>						
1st Qtr.	--	--	--	--	--	--
2nd Qtr.	10	74	783	283	331	2.6x10 ⁻³
3rd Qtr.	4	29	438	238	172	1.6x10 ⁻³
4th Qtr.	--	--	--	--	--	--
TOTAL	14	103	1,221	521	503	2.1x10 ⁻³
<u>Filters ≤ 15 g/drum</u>						
1st Qtr.	1	7	109	59	0	--
2nd Qtr.	7	51	907	557	43	1.7x10 ⁻⁴
3rd Qtr.	4	29	451	251	38	3.3x10 ⁻⁴
4th Qtr.	3	22	252	102	8	1.7x10 ⁻⁴
TOTAL	15	109	1,719	969	89	2.0x10 ⁻⁴
<u>Tantalium > 15 g/drum</u>						
1st Qtr.	1	7	246	196	29	3.3x10 ⁻⁴
2nd Qtr.	5	37	1,464	1,214	118	2.1x10 ⁻⁴
3rd Qtr.	--	--	--	--	--	--
4th Qtr.	--	--	--	--	--	--
TOTAL	6	44	1,710	1,410	147	2.3x10 ⁻⁴
<u>Tantalium ≤ 15 g/drum</u>						
1st Qtr.	--	--	--	--	--	--
2nd Qtr.	--	--	--	--	--	--
3rd Qtr.	1	7	203	153	4	5.8x10 ⁻⁵
4th Qtr.	2	15	438	338	18	1.2x10 ⁻⁴
TOTAL	3	22	641	491	22	9.9x10 ⁻⁵

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gr wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concent on (g/g)</u>
<u>Miscellaneous >15 g/drum</u>						
1st Qtr.	2	15	338	238	36	3.3×10^{-4}
2nd Qtr.	6	44	1,028	728	1,060	3.2×10^{-3}
3rd Qtr.	1	7	232	182	225	2.7×10^{-3}
4th Qtr.	2	15	436	336	599	3.9×10^{-3}
TOTAL	11	81	2,034	1,484	1,920	2.9×10^{-3}
<u>Miscellaneous ≤15 g/drum</u>						
1st Qtr.	14	103	1,364	664	53	1.8×10^{-4}
2nd Qtr.	18	132	2,522	1,622	101	1.4×10^{-4}
3rd Qtr.	14	103	2,495	1,795	29	3.6×10^{-5}
4th Qtr.	25	184	3,721	2,471	61	5.4×10^{-5}
TOTAL	71	522	10,102	6,552	244	8.2×10^{-5}
<u>Am Waste >15 g/drum</u>						
1st Qtr.	--	--	--	--	--	--
2nd Qtr.	2	15	274	174	154	2.0×10^{-3}
3rd Qtr.	1	7	134	84	143	3.8×10^{-3}
4th Qtr.	--	--	--	--	--	--
TOTAL	3	22	408	258	297	2.5×10^{-3}
<u>Am Waste ≤15 g/drum</u>						
1st Qtr.	13	96	2,118	1,468	8	1.2×10^{-5}
2nd Qtr.	6	44	1,140	840	20	5.2×10^{-5}
3rd Qtr.	8	59	1,482	1,082	42	8.6×10^{-5}
4th Qtr.	11	81	1,998	1,448	41	6.2×10^{-5}
TOTAL	38	280	6,738	4,838	111	5.1×10^{-5}

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gr wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concentration (g/l)</u>
<u>Cm 244 Waste > 15 g/drum</u>						
1st Qtr.	1	7	128	78	51	1.4x10 ⁻³
2nd Qtr.	--	--	--	--	--	--
3rd Qtr.	--	--	--	--	--	--
4th Qtr.	<u>1</u>	<u>7</u>	<u>200</u>	<u>150</u>	<u>41</u>	6.0x10 ⁻⁴
TOTAL	2	14	328	228	92	8.9x10 ⁻⁴
<u>Cm 244 Waste ≤ 15 g/drum</u>						
1st Qtr.	3	22	374	224	0	--
2nd Qtr.	3	22	334	184	20	2.4x10 ⁻⁴
3rd Qtr.	--	--	--	--	--	--
4th Qtr.	<u>1</u>	<u>7</u>	<u>138</u>	<u>88</u>	<u>1</u>	2.5x10 ⁻⁵
TOTAL	7	51	846	496	21	9.3x10 ⁻⁵
<u>Grand Total of Line Generated Waste > 15 g/drum</u>						
1st Qtr.	46	337	10,157	7,857	4,325	1.2x10 ⁻³
2nd Qtr.	165	1,214	29,608	21,358	17,570	1.8x10 ⁻³
3rd Qtr.	208	1,528	32,293	21,893	12,056	1.2x10 ⁻³
4th Qtr.	<u>248</u>	<u>1,824</u>	<u>49,416</u>	<u>37,016</u>	<u>13,954</u>	8.3x10 ⁻⁴
TOTAL	667	4,903	121,474	88,124	47,905	1.2x10 ⁻³

	<u>No. of Drums</u>	<u>Volume (ft³)</u>	<u>Gr wt. (lb.)</u>	<u>Net wt. (lb.)</u>	<u>Pu (g)</u>	<u>Concent Ion (g/g)</u>
<u>Grand Total of Line Generated Waste ≤15 g/drum</u>						
1st Qtr.	224	1,646	36,686	25,486	1,306	1.1x10 ⁻⁴
2nd Qtr.	298	2,189	44,912	30,012	1,777	1.3x10 ⁻⁴
3rd Qtr.	304	2,235	44,961	29,761	1,908	1.4x10 ⁻⁴
4th Qtr.	<u>314</u>	<u>2,309</u>	<u>58,545</u>	<u>42,845</u>	<u>1,982</u>	<u>1.0x10⁻⁴</u>
TOTAL	1,140	8,379	185,104	128,104	6,973	1.2x10 ⁻⁴
<u>Grand Total of All Line Generated Waste</u>						
1st Qtr.	270	1,983	46,843	33,343	5,631	3.7x10 ⁻⁴
2nd Qtr.	463	3,403	74,520	51,370	19,347	8.3x10 ⁻⁴
3rd Qtr.	512	3,763	77,254	51,654	13,964	6.0x10 ⁻⁴
4th Qtr.	<u>562</u>	<u>4,133</u>	<u>107,961</u>	<u>79,861</u>	<u>15,936</u>	<u>4.4x10⁻⁴</u>
TOTAL	1,807	13,282	306,578	216,228	54,878	5.6x10 ⁻⁴

BLE 6

Crated Waste CY 67

Date (Qtr. CY 67)	<u>Building 771 Incinerator Filters</u>			<u>Misc. Waste (from Buildings 771, 776 and 777)</u>			<u>Misc. Waste with Nil Pu (from Buildings 771, 774, 776* and 777)</u>			<u>Total Crated Wastes (from Buildings 771, 776 and 777)</u>		
	No.	Volume (ft ³)	Pu (g)	No.	Volume (ft ³)	Pu (g)	No.	Volume (ft ³)	No.	Volume (ft ³)	Pu (g)	
1st	6	826	510	24	3,302	313	17	2,206	47	6,334	823	
2nd	17	2,342	482	20	2,734	264	12	1,385	49	6,461	746	
3rd	16	2,205	671	21	2,871	230	36	4,950	73	10,026	901	
4th	<u>10</u>	<u>1,375</u>	<u>988</u>	<u>45</u>	<u>6,195</u>	<u>298</u>	<u>40</u>	<u>5,500</u>	<u>95</u>	<u>13,070</u>	<u>1,286</u>	
TOTAL	49	6,748	2,651	110	15,102	1,105	105	14,041	264	35,891	3,756	

Other Crated Waste CY 67
(Non-Plutonium Waste)

Date	<u>Building 444</u>			<u>Building 881**</u>			<u>Building 883</u>			<u>Building 886</u>			<u>Building 991</u>			<u>Total Crated</u>		
	No	Volume (ft ³)	Gross Weight (lb)	No.	Volume (ft ³)	Gross Weight (lb)	No.	Volume (ft ³)	Gross Weight (lb)	No	Volume (ft ³)	Gross Weight (lb)	No	Volume (ft ³)	Gross Weight (lb)	No	Volume (ft ³)	Gross Weight (lb)
Jan-Jun 67	3	345	5,120	23	690	42,400	21	2,449	75,500	—	—	—	2	248	3,000	49	3,732	126,020
Jul-Dec 67	<u>49</u>	<u>5,748</u>	<u>84,280</u>	<u>50</u>	<u>3,620</u>	<u>92,000</u>	<u>25</u>	<u>3,299</u>	<u>79,050</u>	<u>1</u>	<u>36</u>	<u>1,400</u>	—	—	—	<u>125</u>	<u>17,703</u>	<u>256,730</u>
TOTAL	52	6,093	89,400	73	9,310	134,400	46	5,748	154,550	1	36	1,400	2	248	3,000	174	21,435	382,750

* from Jan-June 1967 - 222 Cartons (1,280ft² and 12,700 lb with nil plutonium) were discarded

** from Jan-June 1967 - 718 cartons (3,375 ft³ and 33,450 lb, with nil plutonium) were discarded

Appendix I

Sorting of Radioactive Waste at Rocky Flats

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SORTING OF RADIOACTIVE WASTE
AT ROCKY FLATS

D. M. Anderson
E. A. Putzier
D. L. Ziegler

February 11, 1970

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SUMMARY

The sorting or segregation of solid radioactive residues or waste as to type of matrix is done for the purposes of radioactive material measurement and processing. The sorting is basically done at the point of generation. The only segregation of waste as to type of radionuclide is because the radionuclides are, in general, processed in different buildings. Appendix A is a list of specific questions and answers as they relate to sorting of radioactive waste.

The liquid wastes are segregated at the point of generation to facilitate processing of the waste streams. The water meeting the State and Federal regulations for off site release is discharged into Walnut Creek which traverses the plant site. About 78,000,000 gallon per year of water, meeting the regulations for off site release, was estimated for the plant for FY 1969 assuming normal operation for the full year. Aqueous liquids that are not compatible with the decontamination process are converted to solids by setting them up in cement. The organic liquids are converted to grease (oil- CCl_4 + calcium silicate). The other liquid wastes are converted to solids and water vapor. The amount of waste generated in FY 1969 by Rocky Flats plant annualized to reflect normal generation for the full year is given:

	Volume (cu. ft.)	Contamination Level
First stage sludge	7,840	Non-LSA
Second stage sludge	10,140	Non-LSA
Evaporator salts	18,280	LSA
Cemented liquids	3,740	Non-LSA
Grease	5,300	Non-LSA
Line-generated waste	10,500	Non-LSA
Nonline-generated waste	38,930	LSA
Crated waste	<u>40,100</u>	Non-LSA
 Total Plutonium Waste	 134,830	
 Non-plutonium hot waste-crated	 29,400	Non-LSA
Non-plutonium hot waste-drummed	<u>15,050</u>	LSA
	 179,280	

DISCUSSION

Segregation of types of waste material begins at the point of generation. The four general categories of waste or residues are liquid waste, nonline-generated solids, line-generated solids, and crated waste. The sorting or segregation of radioactive contaminated waste at Rocky Flats is done mainly to facilitate measurement of radioactive content and to facilitate processing of the material. Some segregation as to type of contaminate, plutonium, americium, uranium, or beryllium is accomplished because the processing is done in different buildings. How each of the four basic waste categories are handled will be discussed separately.

Liquid Waste

The liquid waste is divided into six subcategories:

1. Low radioactive-low chemical*
2. High radioactive-low chemical*
3. Low radioactive-high chemical*
4. High radioactive-high chemical*
5. Organic waste
6. Complex waste

*Low radioactive	- below drinking water standards for radioactivity
Low chemical	- below drinking water standards for chemical contaminants
High radioactive	- above drinking water standards for radioactivity
High chemical	- above drinking water standards for chemical contaminants

An attempt is made in each processing area to segregate the liquid waste into the six subcategories with the objective of decreasing the waste treatment costs in Building 774.

No liquid waste is shipped to Idaho for burial. The only liquid leaving the plant site is water that meets the state and federal regulations for off site release. The effluent from the sewage treatment plant and the low radioactive-low chemical waste are the two streams which meet the requirements. The effluent from the sewage treatment plant is analyzed daily for compliance with State and Federal requirements.

Each batch of low radioactive-low chemical waste is also sampled and analyzed for compliance prior to release. These streams are discharged into Walnut Creek which traverses the plant site. All liquids which do not meet the requirements for off site release are sent to Building 774 for treatment.

The high radioactive-low chemical waste is processed in the second stage of the waste treatment plant (Building 774) where the stream is decontaminated, utilizing a ferric hydroxide carrier precipitation process. After treatment, the effluent is analyzed for compliance with State and Federal regulation. After verification of compliance, the water is released off site via Walnut Creek.

The low radioactive-high chemical waste is sent either to the solar evaporation ponds or to the waste evaporator in Building 774. This material cannot be released off site because of the high chemical content. The vapor from the evaporator is continuously monitored for contamination by means of a conductivity cell, and daily samples are taken by Health Physics personnel. The evaporator bottoms is sent to a double drum dryer where the remaining water is removed from the dissolved salts and goes up the evaporator stack. The dried solids are drummed for shipment to Idaho as evaporator salts.

The high radioactive-high chemical waste is processed through two stages of decontamination in Building 774, utilizing ferric hydroxide carrier precipitation process. The decontaminated aqueous waste is combined with the low radioactive-high chemical waste and is processed in the waste evaporator as described above. The precipitate is removed from the supernate liquid via vacuum filtration. The filter cake is drummed for shipment to Idaho as first and second stage sludge.

The organic waste (cutting oil-carbontetrachloride) is shipped to Building 774 where it is mixed with calcium silicate to produce a semi-solid product called "grease." About 37 gallons of the organic waste is converted to a 55-gallon drum of "grease." Future plans call for recovery of the carbontetrachloride by distillation. The recovered carbontetrachloride will be reused and the oil will be converted to "grease" with calcium silicate.

The complex waste is a stream produced in the analytical laboratory. Because these complexing agents will prevent effective decontamination of the other aqueous waste, it cannot be treated in the ferric

hydroxide carrier precipitation process. These complex solutions are set up with a mixture of magnesia and Portland cement to form the cemented waste category.

The total volume of waste which would have been generated in FY 1969, assuming normal operation for the full year, was estimated at 179,280 ft³. Of this quantity, 45,300 ft³ was estimated as first stage sludge, second stage sludge, evaporator salts, cemented liquids and "grease." In addition to these solid wastes, about 78,000,000 gallons of water which meet the State and Federal regulations for off site release were approximated for FY 1969, assuming normal operation for the full year. It should be pointed out that these liquid wastes come from all areas of the plant site and after treatment cannot be segregated as to type of radionuclide. If another waste evaporator were available, the nonradioactive chemical wastes could be converted to nonradioactive contaminated salts. The primary segregation of waste at the point of origin is obtained by supervisory control as well as the segregation obtained by the various tankage and piping systems.

Nonline-Generated Solid Waste

All waste removed from a radioactive material processing area is considered as contaminated and is packaged and shipped as such. Nonline-generated waste contaminated by beryllium is also presently packaged and shipped to Idaho for burial. Non-contaminated beryllium chips are returned to the supplier for reclaiming. The nonline-generated uranium wastes are segregated from the plutonium-ameridium wastes because of being generated in separate buildings. However, no provisions are made to keep them segregated during shipment and burial.

All the drums of nonline-generated waste are surveyed by Health Physics personnel before removal from the area of generation. All drums indicating greater than 0.5 mr/hr at the surface are sent to Building 771 for drum counting and possible plutonium recovery. The non-line generated waste which shows less than 0.5 mr/hr at the surface of the drum is considered as low specific activity (LSA) waste and is shipped as such.

The volume of nonline-generated waste for FY 1969 (assuming normal operation for the full year) was estimated to be about 36,930 ft³ as plutonium contaminated and about 15,050 ft³ as nonplutonium contaminated waste. The nonplutonium wastes include that contaminated with uranium or beryllium.

Line-Generated Waste

The line-generated waste contains the bulk of the plutonium discarded and shipped to Idaho for burial. This waste category is segregated at the point of generation according to type of waste material. A list of the line-generated waste subcategories with the corresponding discard limits are presented in Table I. The wastes or residues are segregated to aid in counting for nuclear materials accounting and to facilitate processing when necessary. The type of matrix has to be known to obtain a valid determination of plutonium content by the drum counting technique. The residues which contain more than the discard concentration limit of plutonium are processed in Building 771 for recovery. The material which contains less than the discard concentration limit of plutonium is discarded as line-generated waste.

Because the residues are processed and counted in a segregated fashion, they are packaged in 55-gallon drums segregated as to matrix type. Because these wastes do not meet the criterion for LSA, they have to be shipped in the ATMX rail car. These drums are mixed with other non-LSA drums in shipment and burial. Other non-LSA drummed wastes include the first stage sludge, second stage sludge, grease, and cemented liquid. The annualized line-generated waste volume for FY 1969 was estimated to be 10,500 ft³, assuming normal operating conditions for the full year.

TABLE I

<u>Category</u>	<u>Discard Limit (g Pu/g Total)</u>
Sweepings	0.007
Sludge	0.007
MgO Sand	0.007
Ion Exchange Resin	0.007
Incinerator Ash	0.007
Sweepings Heels	0.007
Ash Heels	0.007
Glass and Ceramics	0.0005
Scarfed Molds	0.00035
Graphite Flow Residue	0.002
CWS Filter 2' x 2' x 1'	24.0 grams/filter
Dry Box Filters 8" x 8" x 4"	3.0 grams/filter
Washables	0.0006
Combustibles	0.0007
Miscellaneous Scrap Metal	0.0003

Crated Waste

The crated waste is composed of large bulky equipment which cannot be conveniently placed in a 55-gallon drum. The equipment which is generated in a glovebox line is cleaned prior to removal. The plutonium content is estimated by smear analysis. Each crate produced in a radioactive material processing area is surveyed by Health Physics personnel prior to shipment.

The crated waste is, in general, segregated by radionuclide type due to its point of generation. Assuming a full year of normal generation (no fire May 11), the annualized estimated crated waste generation for FY 1969 is 40,100 ft³ for the processing area and 29,400 ft³ for the nonplutonium processing area. The crates are not segregated during shipment or burial.

APPENDIX

QUESTIONS AND ANSWERS ON SORTING
OF SOLID RADIOACTIVE WASTES

1. Q. Are solid wastes from radiation areas sorted into contaminated and uncontaminated wastes?
A. In general wastes from plutonium processing and laboratory areas are not sorted as to contaminated and uncontaminated waste. Any waste items originating in a glovebox line are definitely contaminated. Waste generated from maintenance operations, bagging out procedures, and glove changes are either measured as contaminated or assumed to be, and probably at all times are. Some waste generated outside the glovebox line in the process area is probably not contaminated. These wastes are treated as contaminated and are shipped as LSA waste to Idaho for burial.

2. Q. Are radioactive solid wastes sorted as to combustible/noncombustible nature? If so, is this to feed an incinerator? To give special storage or surveillance to combustibles? Other reasons?
A. The combustible material from within the glovebox line is segregated so that it can be incinerated. The incineration is a part of the plutonium recovery process. Only the waste which contains significant quantities of plutonium are incinerated. Nonline-generated wastes (combustibles) are not incinerated. The wastes are sorted into various categories (graphite, scrap metal, glass combustibles, etc.) mainly for two reasons:
 1. Each type of waste requires different processing techniques and therefore must be segregated prior to processing.
 2. The accuracy in determining the amount of radioactive material in a drum by neutron-gamma counting techniques is increased when the material

is segregated. It should be noted, however, that most drums will contain small amounts of other categories due to processing e. g., plastic bags used to remove noncombustibles from the glove-box line.

3. Q. Are the wastes sorted as to compressible/noncompressible nature? If so, is this by visual inspection by a sorter? Identification supplied by person discarding? Point of origin (i. e., assumption all wastes from a particular building are compressible)? Other?

A. The sorting which is done for processing and counting reasons results in segregation where some categories could be compacted and others which would not be compactable. The nonline-generated waste would be considered compressible and the line-generated combustible category would be considered compressible. The segregation is done by visual inspection when they are generated. However, the sorting methods may have to be improved to provide effective compression. All other categories contain considerable quantities of noncompressible waste.

4. Q. Are any special sortings required for wastes containing animal carcasses, pyrophoric materials, alkali metals, explosives, or other unusual hazards? If so, please describe briefly.

A. Only small quantities of waste which present special hazards are generated. These are such things as lithium metal waste and cyanide solutions. In general these wastes are converted to a nonreactive form prior to shipment. Better methods of handling lithium metal waste and cyanide solutions are under consideration.

5. Q. Are wastes sorted according to penetrating radiation level? If so, is the purpose to limit exposure to workers by special shielding and handling? To bury at a separate part of plant burial ground? To meet DOT packaging and shipping regulations? Other?

A. Plutonium contaminated wastes are sorted relative to DOT packaging and shipping regulations. Specifically those which meet the low specific activity definition and those which don't. No segregation of waste is done for the purpose of limiting personnel radiation exposure. All of Rocky Flats wastes are buried at Idaho in the same location.

Pl & non-Pl wastes are segregated at R.F. Idaho

6. Q. Are wastes sorted according to identity of radionuclides? If so, is it by plutonium, americium, and transuranium nuclides? Strontium-90 and cesium-137? Mixed fission products (as distinct from induced activity)? Other?

A. In general, no. Since americium in our waste is a result of its growth from plutonium, the two exist together in varying proportion. On a broad scale, uranium wastes originate from different areas and are handled separate from americium and plutonium wastes.

7. Q. How is sorting in No. 6 done? Identification provided by person discarding the waste? Point of origin (i. e., assuming all waste from a particular area to be plutonium contaminated)? Direct radiation measurement on contaminated surfaces? External radiation measurement on package? Other?

A. The sorting according to identity of radionuclide is accomplished as a function of the point of origin. Uranium wastes are generated in a separate area from the plutonium and americium.

8. Q. If waste packages are identified as containing significant quantities of plutonium by an external radiation measurement, indicate limit of detection.

A. Waste packages are either 55-gallon drums or wooden boxes. Drums are identified as containing significant quantities of plutonium by counting in a specially designed neutron-³ drum counter. This device has the capability of detecting quantities down to 1.0 gm of plutonium in a 55-gallon drum. No external counting methods have been developed for determination of plutonium in boxes. Such a determination is made by estimates of plutonium on individual items which go into the box.

9. Q. What is purpose of sorting in No. 6? Protect workers by special handling and contamination control requirements? Comply with DOT packaging and shipping regulations? Bury in separate part of plant burial ground or in special containers? Please describe briefly any special on-site burials.

A. Sorting is partly for compliances with DOT regulations and more recently to comply with AEC policy statement to separate plutonium wastes from others.

There are three on site burial locations described as follows:

1. Sanitary landfill dump for uncontaminated trash disposal located in the northwest quadrant of the plant site outside the limited area fence in current use.
2. A number of empty steel drums were crushed and buried in the east central portion of the plant site outside the limited area fence. These barrels had been used to store uranium-contaminated liquids and were emptied prior to crushing. The burial of these drums was started in 1957. No additional burial has been done at this site since 1955.
3. A number of steel drums containing depleted uranium machined chips, uranium-contaminated liquid waste, plutonium-contaminated liquid waste, and other contaminated dry waste are buried in the east central portion of the plant site, inside the limited area fence. The burial of these drums was started in 1954. No additional burial has been done at this site since 1958.

*Should
get rid
of these.*

10. Q. Do you believe significant reductions in volume of solid radioactive waste could be achieved by sorting actually contaminated from uncontaminated materials if this is not done presently? What do you estimate the economics of such sorting would be? What volumes could be saved?

A. Most waste originates under situations where contamination is highly probable. Better administrative control of what goes into a processing area may have some influence on volume of waste generation without a significant change in economics. This could result in a reduction in the cost of handling contaminated waste. The probable small reduction in radioactive waste would not amount to more than a few thousand dollars a year. Any attempt at sorting within areas would require additional manpower and a special hooded handling area. It would be expected that to cover all plutonium buildings would require an operating budget of \$100,000/year plus a capital investment to provide a sorting area.

Sorting of uncontaminated wastes requires very careful procedure tends to allow mistakes to occur relative to plutonium getting into uncontrolled areas.

11. Q. If you presently identify wastes which you believe to be contaminated with significant quantities of plutonium, do you believe the volume of such wastes could be significantly reduced by additional sorting? What do you estimate the economics of such sorting would be? What volumes could be reduced?

A. No, for reasons stated in 10. Volume reduction would probably be much less than 10 percent. Economics are considered under 10.

12. Q. What is your estimate of the volume of waste at your sites that might be amenable to incineration? Compression?

A. Of the waste generated at Rocky Flats, assuming normal operation for the full year of FY 1969, approximately 54,000 ft³ of nonline-generated waste and approximately 7,000 ft³ of line-generated waste could be incinerated. The line-generated waste includes graphite and resin

which will require an improved incinerator design. The incineration is estimated to give approximately 93 percent reduction in volume or equivalent to 56,700 ft³.

If suitable compaction equipment were available, approximately 54,000 ft³ of nonline-generated waste could be compressed by about 75 percent resulting in a volume reduction of about 40,500 ft³ of waste. If a compactor suitable for compressing metal such as drums and gloveboxes were available, approximately 41,700 ft³ (estimated) of crated waste could be compacted by about 75 percent resulting in a volume reduction of about 31,300 ft³ of waste. The total for both types of compaction on about 91,700 ft³ of waste would result in a volume reduction of about 41,800 ft³ of waste.

Appendix J

Outside Storage for Drums of Building 71 Washables

Criticality Letter
Building 771 Washables

November 19, 1962

H. W. Vaughan

cc:

— C. L. Schuske (Record)

OUTSIDE STORAGE FOR DRUMS OF BUILDING 71 WASHABLES

Typical materials contained in these drums are rags, paper, plastics, rubber etc. The maximum water content in a drum is 20 w/o; the maximum graphite per drum is 25 lbs. Each drum is limited to 500 g of C plant material.

For the drums of washables a planar array, 2 drums high, with drums in contact is permissible outside the building provided:

1. The drums are elevated above the ground to allow for water drainage.
2. The drums are covered with tarpaulins to prevent in-leakage of water.

HWK
HWK:mjd

CLS
C. L. Schuske

Appendix K

Storage of 55-Gallon Drums in Buildings 71, 76, and 34

Criticality Letter for Storage
of 55-Gallon Drums

March 28, 1963

L. L. Zedner

cc:

D. G. Hesterlein (2)
W. H. Wright (2)
D. F. Smith
C. L. Schulte (Record)

STORAGE OF 55 GALLON DRUMS IN BUILDINGS 72, 76, AND 86

A. General Rules:

1. Each system must be posted with criticality limits.
2. Each system must be isolated from all other material by at least 3-feet.
3. Any system involving an edge-to-edge separation must have this spacing fixed mechanically to assure the determined separations.

B. Types of Waste Materials:

1. Miscellaneous Wastes---typical material includes the following items: glass, rags, paper, plastics, rubber, MgO etc. The maximum water content of any part of the load is 20 weight percent. The maximum graphite per drum is 25 lbs.

Criticality Limits

load limit

a. 500 g/drum

b. 1000 g/drum

ARRAY

planar array
2 high in contact

double line in
contact array
1 high

Storage of 55 Gallon Drums in Buildings 71, 76, 84 (Cont)

2. Graphite---this material contains no hydrogenous material except for the plastic bags containing the material. Maximum plutonium content is 30% by weight.

Criticality Limits

load limit

ARRAY

a. 2000 g/drum

3-foot edge-to-edge
planar array 1 high

b. 500 g/drum

planar array in
contact 1 high

3. Residues---this material has a maximum water content of 50% by weight. They are cakes having a solid appearance and are contained in quart cartons which in turn are stored in the 55 gallon drums. Maximum graphite per drum is 25 lbs.

Criticality Limits

load limit

ARRAY

400 g/drum

in contact planar
array 1 high

4. Incinerator Ash and HgO ---essentially dry, less than 10% by weight water. The maximum plutonium content is 33-1/3% by weight. The material is contained in #10 cans having a total weight limit per can of 3 1/2 lbs and a plutonium content limit of 1 kg.

Criticality Limits

load

array

3 kg per drum

planar in contact
array 1 high

5. Bottled Aqueous Wastes---the material must be analyzed for each container. Maximum per container is 100 g. Cadmium poisoning is suggested for all concentrations over 10 g/liter. (1g for each 3 g of plutonium)

Storage of 55 Gallon Drums in Buildings 71, 76, and 84 (cont)

Criticality Limits

load

a. 200 g/drum

b. 400 g/drum

ARRAY

planar in contact
array 1 high

planar array 1 high
3-foot edge-to-edge

C. L. S.
C. L. Schube

DVS:mjd

Appendix L

Letter to R. F. Rogers from J. D. McCarthy

Criticality Letter for
Storage of C.W.S. Filters

August 12, 1965

F. Rogers

cc:

C. L. Schuske (Record)

**CRITICALITY RECOMMENDATION: STORAGE OF CWS FILTERS IN
BUILDINGS 70 AND 80**

CWS filters, each containing a maximum of 300 g, may be stored in boxes (26" x 26" x 14"), in the following arrangement:

1. Filters arranged in rows, each row two filters wide by three filters high with no restriction on length.
2. A minimum of 3 feet between adjacent rows.

All other material must be kept at least 3 feet edge-to-edge from the array.

HCH
JDM:mjd

J. D. McCarthy
J. D. McCarthy

Appendix M

Nuclear Materials Safety Limits for Building 664

NUCLEAR MATERIALS SAFETY LIMITS FOR BUILDING 664

The following limits are to be used for storing material in Building 664.

Array Descriptions

1. Storage Room (including counter)
 - a. 55-Gallon Drums - A two-high array of drums with each drum loaded as specified below.
 - b. Crates - Crates loaded as below may be stacked with no height restrictions or spacing.
 - c. Sources - A maximum of eight (8) sources at 30 g dry Pu per source. No spacing restrictions.
2. Railcar Loading Dock (including ground level)
 - a. 55-Gallon Drums - A two-high array of drums with each drum loaded as specified below.
 - b. Crates - Crates loaded as below may be stacked with no height restrictions or spacing.
 - c. Cargo Containers - A one-high array of cargo containers containing a two-high array of drums or crates loaded as specified below.

Cargo containers may be stacked two-high provided each cargo container is limited to a maximum total of 2000 g Pu.

Material Descriptions

1. 55-Gallon Drums - Each drum is limited to a maximum of 200 g Pu plus nonfissile material (provided the nonfissile material does not exceed 200 lbs of graphite).
2. Crates - Each crate is limited to a maximum of 5 g Pu per ft³ (but is not to exceed 350 g Pu per crate) and contains lathes, pipes, lumber, soil, etc.

Approved by Richard B. Lane Approved by R. G. Kutz
 Approved by G. A. Quinn-Jones Approved by T. G. Borsky
 Approved by L. B. Trinson Approved by ST & Allen

Appendix N
Special Permit No. 5948



DEPARTMENT OF TRANSPORTATION
HAZARDOUS MATERIALS REGULATIONS BOARD
WASHINGTON, D.C. 20590

SPECIAL PERMIT NO. 5948

This special permit is issued pursuant to 49 CFR 170.13 of the Department of Transportation (DOT) Hazardous Materials Regulations, as amended, and on the basis of the December 23, 1968, petition by the Dow Chemical Company, Golden, Colorado, as amended January 30, 1969, February 5, 1969, and March 13, 1969.

1. The DOW CHEMICAL COMPANY is hereby authorized to ship fissile and large quantities of radioactive materials, n.o.s. in accordance with the provisions of the U. S. Atomic Energy Commission (USAEC), Albuquerque Operations Office, approval dated December 20, 1968, as amended March 11, 1969, and as further provided for herein. This permit is issued only to authorize the shipment of large quantities of normal form radioactive waste materials containing fissile material, in packaging which does not completely conform to the requirements of §173.398(c).

2. The authorized packaging consists of either DOT Specification 17C or 17H, or equivalent, steel drums, or wooden crates (DOT Specification 19A or 19B, or equivalent). Each drum or crate must be lined with 5-mil or 8-mil polyethylene respectively. These packages, with the contents authorized herein are authorized for shipment only in a specially modified ATMX Series 600 rail car. The methods of loading, shoring, and positioning of the packages within the rail car must be in conformance with the procedures as outlined in Dow Chemical Company's report No. 2257-68-PE, as revised January 28, 1969 and March 7, 1969. These procedures generally involve the following:

Drums - Drums are loaded in USASI MH 5.1 standard group I demountable cargo containers (8' by 8' by 20'). Two such containers are placed in the rail car. The end doors, of each container with four double-action door locks, are positioned towards the center of the rail car. Each container is tied by cross-corner guys to the rail car frame.

Crates - Crates are loaded in a similar cargo container, except that the sides and top are open. Crates will be blocked, as necessary, within the container frame using timbers or inflatable dunnage.

When cargo containers are unavailable, or the size of crates precludes their usage, the modified loading procedures for a "three-bay" configuration as described on Page 10A (Section VC) of Report No. 2257-68PE may be used. These procedures include repositioning of the car center supports to form bays of the required length, reinforcement of the car ends and center supports, facing of the end members and sides of the supports plus the load with 1" plywood sheets, and removal of all "play" by blocking and shoring with timbers and inflatable dunnage as required, plus fastening of large items to the floor mounts.

3. The authorized contents of each package consists of large quantities of normal form plutonium-239, -240, -241, and/or americium-241, or mixtures of the foregoing; in the form of radioactive wastes. The radioactive contents of each package, which must be in a form that is not readily dispersible, are further described and limited as follows:

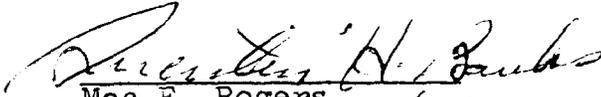
a. Drums - not more than 100 grams (for drums of less than 55 gallon size) or 200 grams (for 55 gallon or larger size) of fissile material per drum, with a maximum thermal decay energy of 2 watts per drum contents and not more than 200 pounds of graphite per drum. The contents consist of either process wastes or line-generated wastes which are further described as follows:

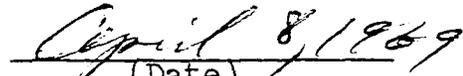
Process Wastes - greases or sludges, hardened with "oil-dry" or cement. In the grease, americium and plutonium are present as fine solids of metal or oxide, or dissolved in organic matter an average concentration of 5×10^{-10} g/g or 1.62×10^{-9} ci/g for the americium and 2×10^{-6} ci/g for the plutonium. In the sludges, americium and plutonium exist as the hydroxide, at an average concentration of 1.07×10^{-5} g/g or 3.48×10^{-5} ci/g for the americium and 4.71×10^{-5} g/g or 3.53×10^{-5} ci/g for the plutonium.

Line-Generated Waste - includes graphite molds, filter sludge, insulation, glass, washables, combustibles, metals, and miscellaneous residues with plutonium discard limits ranging from 7×10^{-3} g/g to 3×10^{-4} g/g.

- b. Crates - The contents are similiar to the line generated wastes described above, except that the size or bulkiness of the items precludes the use of drums. This would include such items as pipe, lumber, equipment, hoods, lathes, etc. The fissile material content of each crate shall not exceed 5 grams per cubic foot.
4. The authorized packaging meets the requirements for shipment as Fissile Class I. The transport index must be assigned based on external radiation levels. Crates, 30-gallon and 55-gallon drums, with the contents as set forth above, may be intermixed in any manner within the ATMX-600 rail car.
5. Prior to each shipment authorized by this permit, the shipper shall notify the consignee of the dates of shipment and expected arrival. The shipper shall notify each consignee of any special loading/unloading instructions prior to his first shipment.
6. The outside of each rail car must be plainly and durably marked "DOT SP 5948", in connection with and in addition to the other markings and labels prescribed by the DOT regulations. Each shipping paper issued in connection with shipments made under this permit must bear the notation "DOT SPECIAL PERMIT NO. 5948", in connection with the commodity description thereon.
7. Shipments are authorized only by rail, in modified ATMX Series 600 Rail Cars which have been assigned for the sole use of the consignor, with instructions to that effect issued with the shipping papers.
8. The shipper is required to furnish an experience report to this Board before expiration of the permit and when any amendment is requested. This report must include the approximate number of packages shipped, and the number of packages involved in any loss of contents.
9. Any incident involving loss of contents of the package must be promptly reported to this Board at the earliest feasible moment following the incident.
10. This permit does not relieve the shipper or carrier from compliance with any requirement of the DOT regulations, except as specifically provided for herein.
11. This permit expires March 31, 1971.

Issued at Washington, D.C.:


Mac E. Rogers
For the Administrator
Federal Railroad Administration


(Date)

Address all inquiries to: Secretary, Hazardous Materials
Regulations Board, U.S. Department of Transportation,
Washington, D.C. 20590. Attention: Special Permits.

Dist: a, e, h, i

Appendix O

Letter to V. C. Vespe from E. H. Lee

August 21, 1967

1191-67PE

Mr. Vincent C. Vespe, Director
Operational Safety Division
U. S. Atomic Energy Commission
Albuquerque Operations Office
P. O. Box 5400
Albuquerque, New Mexico 87115

Thru: Seth R. Woodruff, Jr., Manager
RFAO, USAEC

**REQUEST FOR APPROVAL -
SHIPMENTS OF RADIOACTIVE WASTE IN 55-GALLON STEEL DRUMS**

GENERAL.

Three categories of radioactive waste, as described below, are currently being shipped from Rocky Flats to a burial site. We request your approval of these shipments as a whole, considering the special controls listed, rather than approval of each type of package.

All shipments will be by truck, or by trailer-on-flatcar assigned to our sole use. Identification of packages, placarding of vehicles, and control of external radiation will comply with all AEC and DOT regulations.

CATEGORY A.

Container. Used, 55-gallon metal drums, open-head, 18-gauge (.0478) with bolt type locking ring and 3/8" diameter bolt. These drums have been cleaned inside but have not been re-conditioned or leak tested.

Contents. Paper, rags, clothing, tools, and other items - all non-radioactive material externally contaminated with plutonium. This material is known as non-line-generated waste and includes all types of solid refuse from within the plutonium fabrication buildings, but outside of the drybox system.

Material Limits. Package gross weight will be limited to 480 pounds. Plutonium content will be limited to one gram. The six-month summary for the period ending January 1, 1967, shows 1582 drums shipped with an estimated total plutonium content of 37 grams, or .024 grams per drum.

Packing. The drum is lined with an 8-mil polyethylene bag. Soft bulky items are placed directly in this liner. Heavy items, or items with sharp edges are wrapped in 8-mil fiberboard sleeves and individually sealed in 8-mil polyethylene before being placed in the liner. The liner is sealed and the drum lid fixed with the lock ring and bolt.

General Notes. Current shipments are being made under Bureau of Explosives Permit Number 2057.

CATEGORY B.

Container. 1CC-6C or -17C (or equivalent) 55-gallon open-head metal drum.

Contents. Paper, tools, carbon molds, and other production and laboratory equipment from within the plutonium-drybox system (line-generated waste). All material is non-radioactive but is externally contaminated with plutonium.

Material Limits. Package gross weight will be limited to 880 pounds. Plutonium content will not exceed 200 grams per drum.

The six-month summary for the period ending January 1, 1967, shows the following distribution of plutonium per drum:

<u>Grams per Drum</u>	<u>No. of Drums</u>	
0-10	241	* These would be
10-20	314	repackaged to
20-30	200	comply with the
30-40	100	200-gram limit
40-50	13	and the average
50-100	29	content per drum
100-200	21	would be lower
200-300 *	17	than the 30.8
300-400 *	9	value shown below.

Estimated plutonium content is 29,100 grams for these 944 drums, or an average of 30.8 grams per drum. Figures are based on current sampling and/or neutron-counting techniques.

Packing. Same as Category A above.

General Notes. We are currently shipping this material under Bureau of Explosives Permit Number 2057.

CATEGORY C.

Container. 1CC Specification 17C, 55-gallon metal drum, open-head, 16-gauge, with bolt-type locking ring and 5/8 inch diameter bolt.

Contents. Plutonium-contaminated greases, oils, and salts hardened to a solid or semi-solid state as noted below:

<u>DOW MATERIAL IDENTIFICATION</u>	<u>AVERAGE GROSS WEIGHT (POUNDS)</u>	<u>AVERAGE PU CONTENT (GRAMS)</u>	<u>MAXIMUM PU CONTENT (GRAMS)</u>	<u>MATERIAL DESCRIPTION</u>
#741 first-stage sludge	500	5.6 *	50	Consistency of hardened concrete. Hardener: Portland cement.
#742 second-stage sludge	500	< 1	1	-- do --
#743 grease	600	1	5	Consistency of soft putty. Hardener: Johns-Manville Micro-Cel (E)
#744 neutralized HCL & other liquids	450	< 1	7	Consistency of wet boiler cement. Hardener: 85% magnesia cement.
#745 dried salts	650 **	< 1	< 10 **	Dry, granular substance. No hardener required.

* Based on the 398 drums shipped during the first quarter of CY-67.

** Estimate based on pilot production run of 10 drums.

Material Limits. The shipper's operating procedure will limit gross weight to 880 pounds, since ICC Specification 17C does not give an authorized maximum limit. Plutonium content will be limited per the above table.

Packing. Each drum is lined with an 8-mil polyethylene bag. A quantity of the proper hardener (when required) is placed in the bottom of the liner and the drum is filled. Hardener is added during the filling process and then the filled drum is topped off with hardener. The plastic liner is sealed with tape and then additional hardener, or an absorbent material, is placed between the liner and the drum lid. The lid is then locked in place.

General Notes. Bureau of Explosives Permit Number 2058 has been issued to cover shipments of Portland cement-hardened sludges.

TESTING.

Two packages were chosen for testing, both Category C materials. Grease, #743, was selected because of its density and its plastic consistency. And #744 was tested since it was the most fluid-like of any of the materials being shipped.

Both of these packages met the normal conditions of transportation as described in the attached report. The free drop and puncture accident tests were also applied and each package showed some small loss of contents.

The packages listed in Categories A and B, as well as the remaining drum/material combinations in Category C, will meet all normal handling tests.

CRITICALITY.

There should be no question about nuclear safety for these shipments; however, a complete evaluation is being prepared and will be available at a later date.

SPECIAL CONTROLS.

In addition to sole use of vehicles, we will control the loading of these vehicles by an operating procedure that will be submitted for your review. This procedure will specify the following minimum requirements:

1. The average plutonium content in any vehicle must not exceed 15 grams per package (to be accomplished by intermixing Category A, B and C materials as necessary). Data is to be maintained as required by Chapter 0529, Paragraph III, E.4.

2. Placing of packages within the vehicle to control penetrating radiation.
3. Methods to control loading and blocking so packages will not shift or otherwise be damaged in transit.

SUMMARY.

We feel that these Fissile Class III shipments can be safely made without risk to public life or property.

Each package will meet the normal handling tests listed in Chapter 0529. The trailer or closed truck will provide additional protection from fire or penetration, and will limit the dispersal of packages in the event of an accident.

These shipments could well be considered as equivalent to a shipment of packages each containing 15 grams of fissile material. As such, the amount of protection being provided is more than adequate.

A Certification of Approval Form is included for your signature.

W. H. Lee, Manager
Product Engineering

FEA:bfh
Enc.

cc:

Seth R. Woodruff, Jr. - AEC, Rocky Flats, w/enc.
W. F. Romine - Dow, Rocky Flats, w/enc.
C. L. Schuske - Dow, Rocky Flats, w/enc.
F. J. Trapp (Record) - Dow, Rocky Flats, w/O enc.

Appendix P

Production Scale Americium Recovery at Rocky Flats 1953–1980

INTERNAL REPORT - NOT CLEARED FOR PUBLICATION

PRODUCTION SCALE
AMERICIUM RECOVERY AT ROCKY-FLATS
1953-1980

CRD 8-023 CRD-000090-023

March 3, 1980

REVIEWED FOR

SEP 3 1980

PATENTABLE
MATERIAL

*None found
D. G. Heberlein*

L. J. Beach
C. C. Perry

Process Chemistry Support

ROCKWELL INTERNATIONAL
Energy Systems Group
Rocky Flats Plant
Golden, Colorado

Distribution:

J. W. Berry
K. W. Calkins
L. W. Doher
D. A. Dunn
E. D. Erickson
L. C. Farrell
P. G. Hagan
R. A. Hilbig
J. B. Knighton
F. J. Miner
G. H. Thompson
E. Vejvoda
Attn: J. L. Holst
C. W. Tesitor

KWIX Index

- Americium
- Plutonium
- Material Balance
- Molten Salt Residues
Accountability
- Materials
recovery
Nuclear
materials
management
Potassium
hydroxides
precipitation
ion exchange
Plutonium-241
Americium-241
Thiocyanates
Oxalates
Dissolution

ABSTRACT

This report contains a brief history of production scale americium recovery processes that have been employed at Rocky Flats. Material balances were made for the current process and the previous KOH precipitation-thiocyanate ion exchange process. Comparison of the material balances shows that the current process recovers more americium and generates less liquid volume than did the KOH precipitation-thiocyanate ion exchange process. The present process contributes 35 - 64% of the americium in the Plutonium Recovery discard streams. Deficiencies in the current process unit operations are discussed and appropriate process changes are recommended. This information is useful in evaluating alternative americium recovery process schemes.

ACKNOWLEDGMENTS

The authors wish to thank E. J. Koehler in the Plutonium Recovery Operations group for providing the bulk of the liquid waste data used in this report. We also wish to thank D. R. Schlepp in the Process Chemistry Support group for writing a computer program used to tabulate input and output data for the material balances. R. A. Borgmann and R. E. Clark in the Process Chemistry Support group compiled the solid waste output data used in the material balances. P. G. Hagan of the Chemical Research group developed process schemes for the improvement of the cation exchange system and oxalate precipitation.

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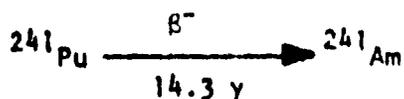
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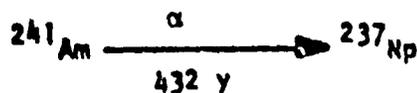
INTRODUCTION

Americium Growth in Plutonium

Americium appears in all plutonium bearing materials processed at Rocky Flats as the result of the beta decay of plutonium-241 to americium-241 by the following reaction:



Significant radiation emissions are the 20.8 KeV beta emission from the above reaction and the 60 KeV gamma emission accompanying the alpha decay of americium-241 by the following reaction:



Neptunium-237 has a half life of 2.14×10^6 y with alpha and gamma emissions. The radiations from this nuclide are not significant when plutonium process streams are evaluated for radiation emission.

The weight percent of plutonium-241 in plutonium process streams at Rocky Flats ranges from 0.75 for plutonium recently separated from spent reactor fuel elements to less than 0.5 for aged material.

The concentration of americium-241 in plutonium-241 bearing materials can be calculated by:

$$C_A = C_P (1.031 \times 10^4) (e^{-1.6 \times 10^{-3}t} - e^{-5.33 \times 10^{-3}t}) + C_A^0 e^{-1.6 \times 10^{-3}t}$$

where:

C_A = Concentration of ^{241}Am in ppm

C_P = Concentration of ^{241}Pu in weight percent

C_A^0 = Initial ($t=0$) concentration of ^{241}Am in ppm

t = Elapsed time in years

Since site return plutonium metal contains greater than 1000 ppm of americium, a process scheme has always been necessary to limit the americium content of recycle plutonium. It follows that the need to minimize personnel exposure to this gamma radiation is a major factor in the selection of americium recovery processes.

History

Plutonium recovery operations at Rocky Flats were started in mid-1953. Precipitation of plutonium peroxide from plutonium nitrate feed solution was included in the first plutonium recovery flowsheet. This process scheme provides an effective partition of americium into the filtrate and wash solutions, and has been included in all subsequent flowsheets.

Process equipment was not available for americium purification until 1954. During this interim period, americium rich plutonium peroxide filtrate was not discharged to waste processing. The plutonium peroxide filtrate was contacted with ammonia gas to precipitate the americium and to neutralize the excess peroxide. The resulting sludge accumulated in filtrate storage tanks and carboys. The sludge was transferred to plastic bottles and stored for future processing. Recovery of americium from these sludge residues was not completed until 1967. Figure 1 is an abbreviated flowsheet of plutonium peroxide precipitation at Rocky Flats in 1953.

During 1954 a glove-box was received from the plutonium recovery facility located at Chalk River, Canada. Equipment for laboratory scale americium process development was installed in this glove-box and then abandoned after six months of operation. Less than twenty grams of americium as americium chloride solution was produced during this operation.

In 1956 a new facility became operational that was dedicated to americium recovery from plutonium peroxide filtrate solutions. The design capacity of this system was 0.5 g Am/day. The feed solution throughput averaged 100 l/day of peroxide filtrate. This process scheme included a cation resin column used primarily to partition iron into the 1 M NH_4SCN cation wash solution. The absorbed actinides were eluted with 5 M NH_4SCN . Figure 2 is a flowsheet of this process.

The total americium produced at Rocky Flats from 1953 to 1956 was less than 150 grams as AmO_2 and americium chloride solutions.

In 1959 the process was modified to expand the use of the strong complexes of plutonium, americium, and iron to partition the iron thiocyanate onto an anion resin. Ryan describes this process in RFP 130.⁽¹⁾ This process scheme was the basis for the design criteria for a new americium recovery

FIGURE 1 - PLUTONIUM PEROXIDE PRECIPITATION - 1953

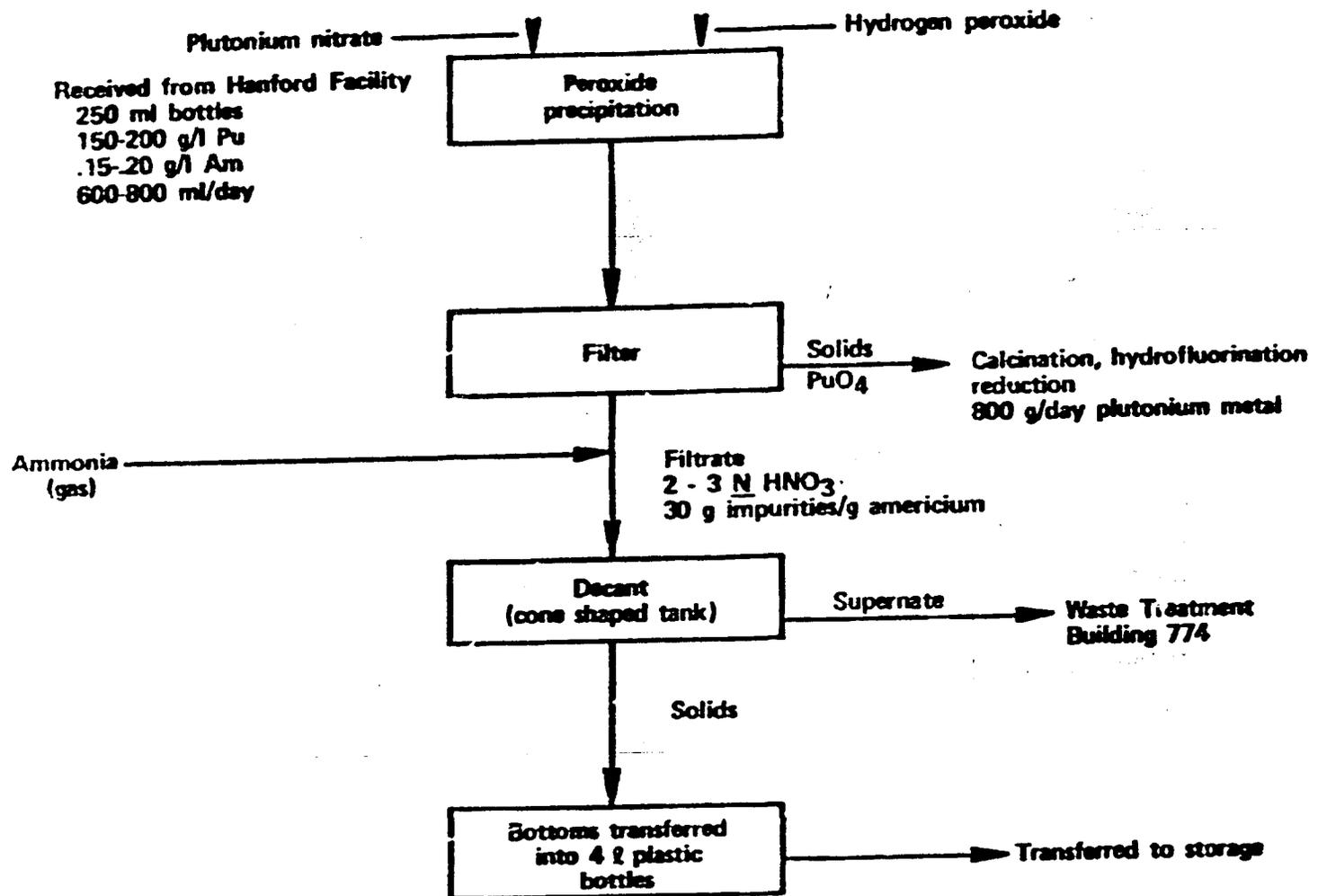
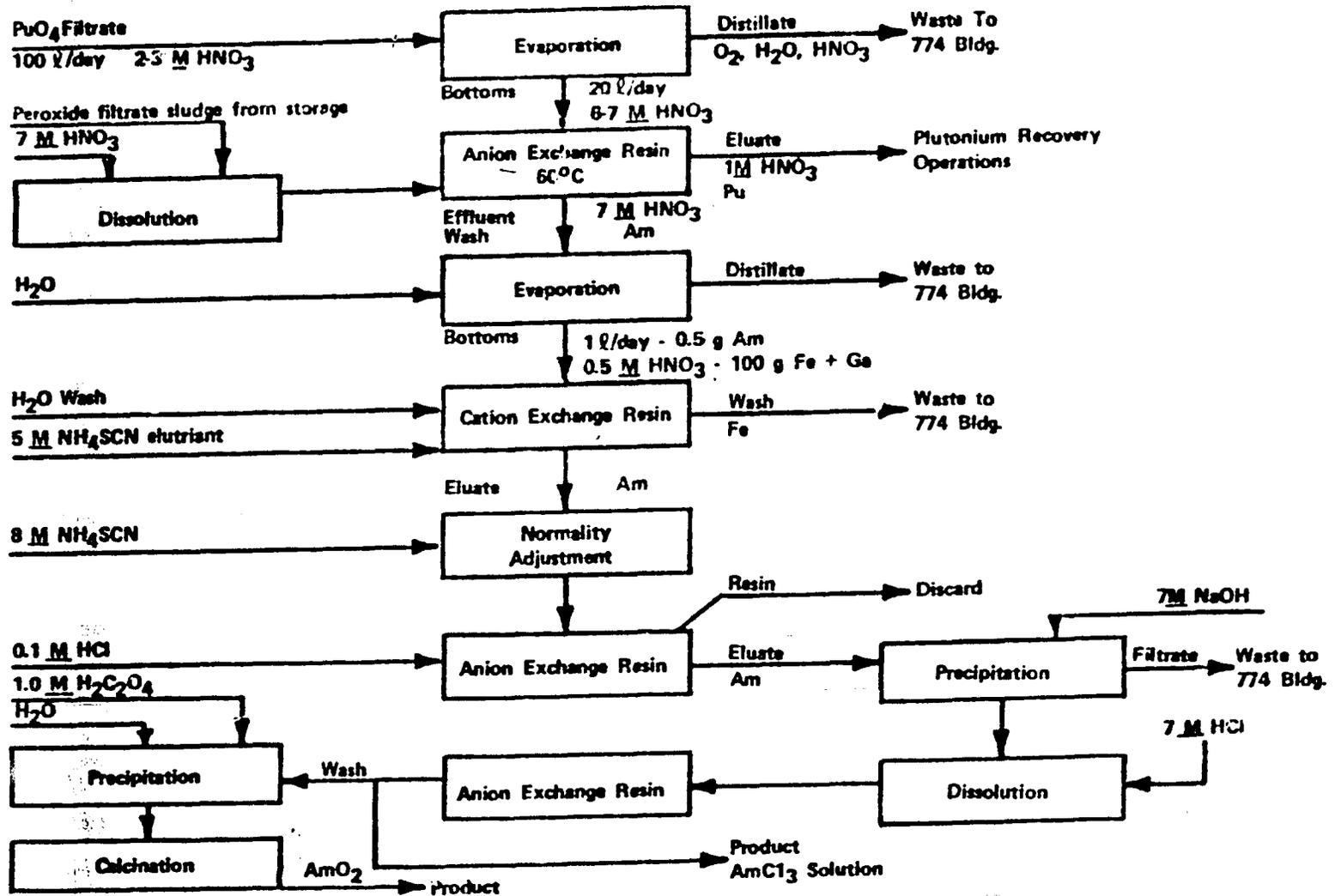


Figure 2. AMERICIUM RECOVERY FLOW SHEET - 1955 THROUGH 1958



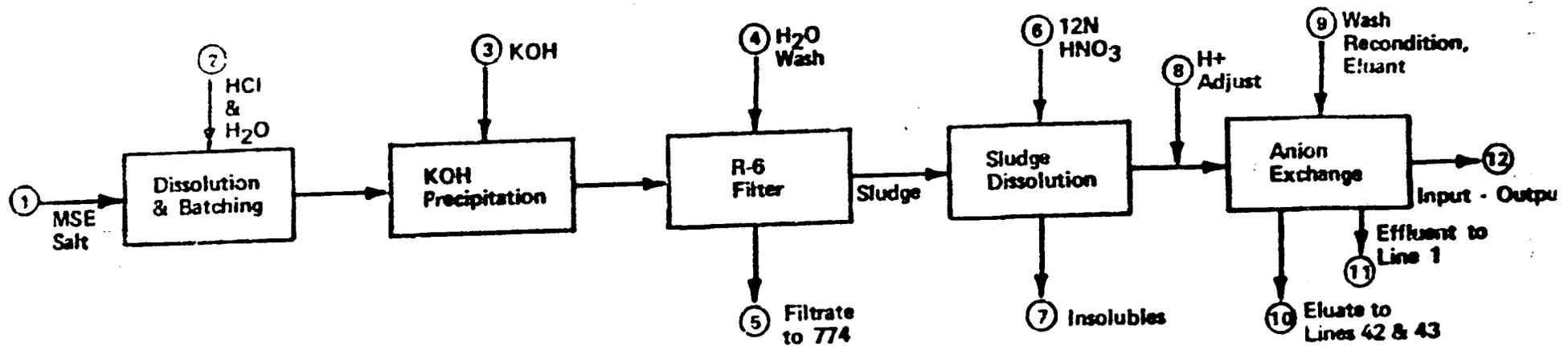
system installed during 1963-1964. This process facility is designated as Line 1. The americium bearing nitrate feed solutions for Line 1 processing were generated from plutonium peroxide filtrate and dissolution of previously discussed sludge residues. Figure 4 is a flowsheet of this process.

A molten salt extraction process (M.S.E.) was developed in 1967 which extracted 90% of the americium content of plutonium metal. This process resulted in a lower americium content in the plutonium peroxide filtrate. Americium concentration steadily diminished until recovery of americium from plutonium peroxide filtrate was no longer justified economically. At this time, salt residues from the M.S.E. process became the only significant feed material for the americium recovery process. M.S.E. salt dissolution, KOH precipitation of the americium and plutonium, and dissolution of the KOH precipitate in nitric acid were the process operations performed in Line 30. Figure 3 is a flowsheet of this process. This glove-box and attached process tanks are lined with plastic for protection against chloride corrosion. In 1974 Proctor developed a process to absorb plutonium and americium on cation resin and partition chloride ions into the cation effluent,⁽²⁾ based on previous work done by Kudera and Guyer.⁽³⁾ This process replaced the caustic precipitation in Line 30, greatly decreasing radiation exposure. The thiocyanate ion exchange units of operation in Line 1 were eliminated in 1975. Elimination of these complex operations greatly decreased radiation exposure. Figures 5 and 6 are flowsheets of the new process. Americium losses in output streams are excessive with either the KOH precipitation-thiocyanate ion exchange process or the new process.

Early americium recovery process schemes at Rocky Flats were based on throughputs of less than 30 g Am/year. The design capacity of Chemical Operations recovery process in Building 771 is now 100 g Am/month. These increased throughputs prevent the use of process options that require temporary shielding of extensive "hands-on" glove-box operations.

The AmO_2 product from the thiocyanate process consistently met purity specifications. Lead and plutonium contamination causes frequent reprocessing of the AmO_2 product from the current process. Experiments indicated that the addition of chloride ion to the oxalate precipitation feed solution

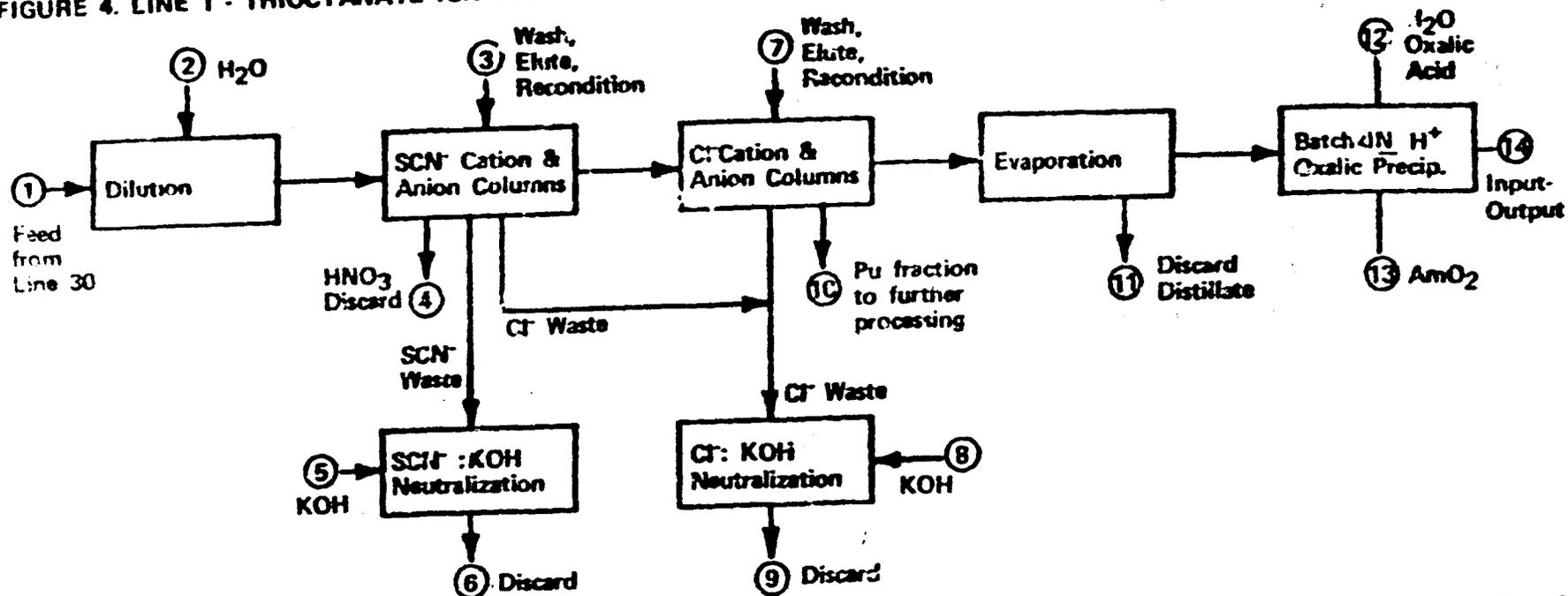
FIGURE 3. LINE 30 - CAUSTIC PRECIPITATION MATERIAL BALANCE FOR THE PERIOD OF MARCH '73 - OCTOBER '73



Stream No.	1	2	3	4	5	6	7	8	9	10	11	12
8wt % Mg Cl ₂ Salt-K ₂	*1059.6											
Pu (g)	17,100				*30		740			*16,027	*131	172
Am (g)	2,200				*6		100			*83	*1404	607
10.5 N HCl (ℓ)		351										
H ₂ O (ℓ)		4050		5198				2740				
7N HNO ₃ (ℓ)									2400		*6060	
0.35 N HNO ₃ (ℓ)									4639	*5839		
12 N HNO ₃ (ℓ)						2120						
6 M KOH			1785		*10,784							

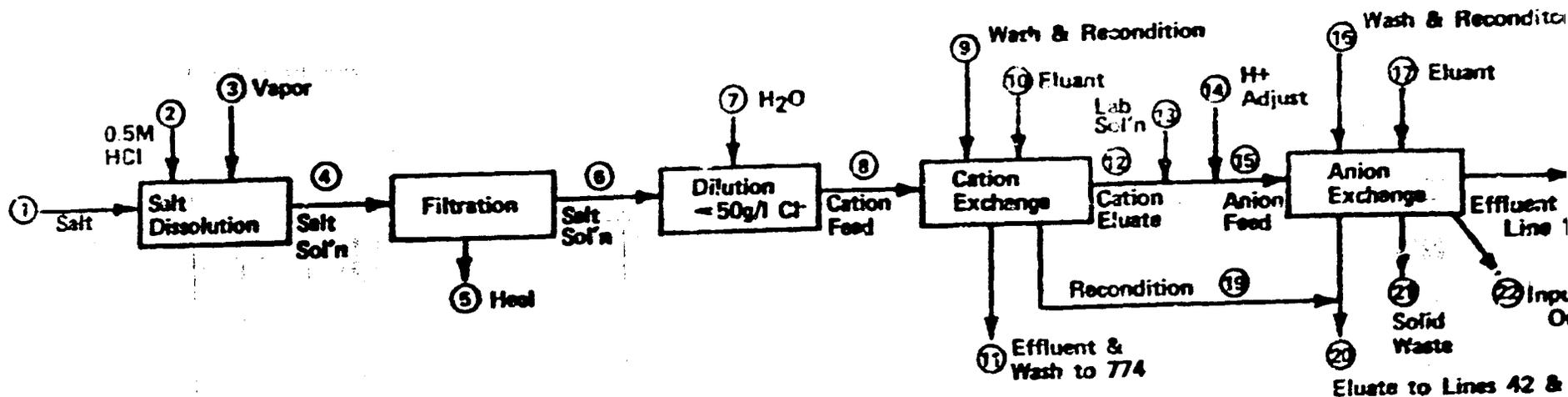
*Measured values. All other values estimated from production data.

FIGURE 4. LINE 1 - THIOCYANATE ION EXCHANGE MATERIAL BALANCE FOR THE PERIOD OF MARCH 73 - OCTOBER 73



Stream No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14
Pu (g)	131					No Data			No Data	No Data	No Data			131
Am (g)	1,404					No Data			No Data	No Data	No Data		836	568
7 N HNO ₃ (l)	6,060		1,520	172,090										
0.35N HNO ₃ (l)			6,545											
H ₂ O (l) H ₂ C ₂ O ₄ (l)		115,140	8623				340					654 106		
0.5 M SCN ⁻ (l)			11,106			10,549								
3.0 M SCN ⁻ (l)			20,096											
0.1N HCl (l)			6,115				180		3,695	40	836			
7 N HCl (l)							1,100							
6M KOH (l)					4,440	4,440		2,398	2,398					

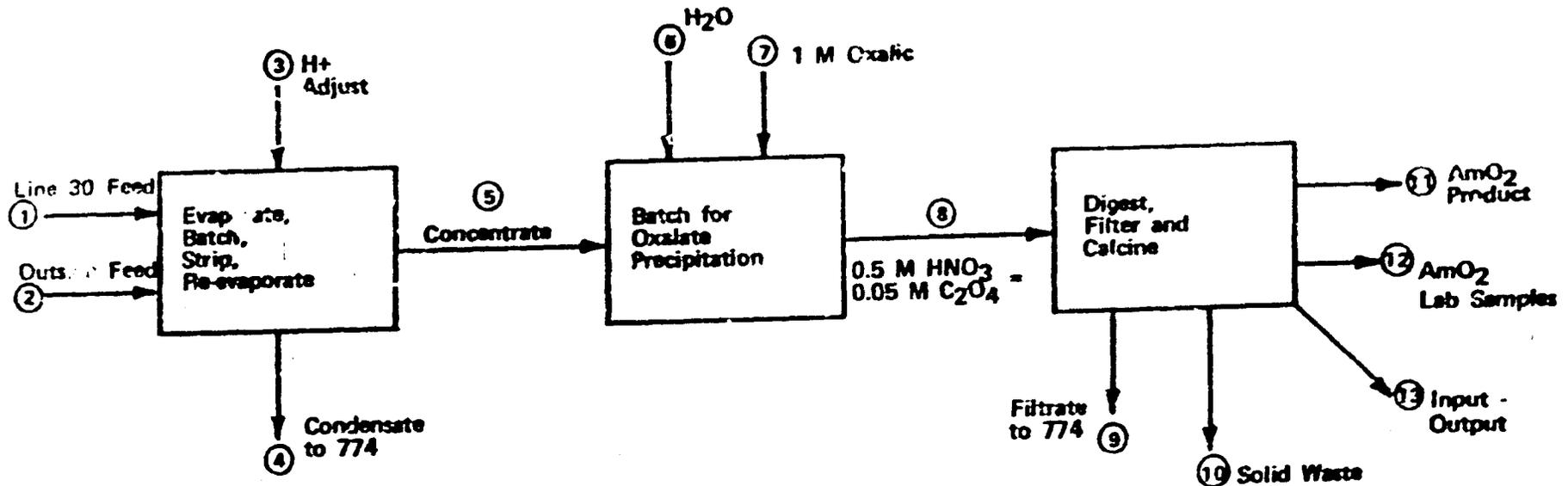
FIGURE 5. LINE 30 - PRESENT PROCESS MATERIAL BALANCE FOR THE PERIOD OF MARCH '76 - OCTOBER '79



Stream No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	
8 Wt % Mg Cl ₂ Salt-Kg	1658.6																						
33 Wt % NaCl ₂ Salt-Kg	*214.6																						
Pu-g	83,084			88,094	*2,437	88,647		88,647			*231	88,388			87,245				*151	*67,788	*8,388	13,047	
Am-g	7,127			7,127	*52	7,875		7,875			*39	7,837	120		7,157				*4594	*937	*380	1,246	
H ₂ O-l							8,370																
0.5 M HCl-l		4,240	2,120	2,120		2,120		8,480			*54,200												
7.0 M HNO ₃ -l										10,280		11,340		4,583	18,410	22,831							
0.35 M HNO ₃ -l										53,890			507										
Am/Pu	0.080			0.030	0.021	0.082		0.082			0.135	0.082	0.122		0.082					39.42	0.614	0.080	0.054
(Pu) Stream X100%	98.91			98.91	2.1	98.20		98.20			0.21	98.89	1.09		98.96					0.17	75.27	7.87	14.47
(Am) Stream X100%	98.34			98.34	0.72	97.82		97.82			0.82	97.16	1.86		98.76					63.39	12.93	8.24	17.20

*Measured values. Other values estimated.

FIGURE 6. LINE 1 - PRESENT PROCESS MATERIAL BALANCE FOR THE PERIOD OF MARCH 76 - OCTOBER 79



Stream No.	1	2	3	4	5	6	7	8	9	10	11	12	13
Pu (g)	*151	*3		*54	100			100	*99		*1		
Am (g)	*4,594	*106		*133	4,567			4,567	*389	*142	*3,935	98	3
7-1. N HNO ₃ (ℓ)	*29,041	*297	56,741	*79,829	6,250								
H ₂ O (ℓ)						73,112							
1M H ₂ C ₂ O ₄ (ℓ)							3,970						
0.5 N HNO ₃ (ℓ)								83,332	*83,332				
Pu/Am	0.033	0.028		0.406	0.022			0.022	0.254		0.001		
(Pu) Stream / (Pu) Total Feed X 100%	98.05	1.95		35.06	64.94			64.94	64.29		0.60		
(Am) Stream / (Am) Total Feed X 100%	97.74	2.26		2.83	97.17			97.17	8.28	3.02	83.72	2.09	0.06

*Measured values. Other values estimated.

decreases the lead contamination. This effort is not a total success on a production scale because the lead concentration of the product AmO_2 does not always meet the one percent purity specification. Plutonium contamination in the AmO_2 product is due to inefficient anion resin column performance.

Potassium hydroxide neutralization was investigated as an alternative to the massive dilutions presently used to adjust the normality of the americium oxalate precipitation feed solution. The product AmO_2 contained greater than one weight percent potassium and did not meet purity specifications.

Efforts are being made to improve the operation of the production cation resin system. Dilution of the feed solution improves the plutonium and americium loading capacity. A final wash with 1.5 M HNO_3 improves impurity separation. These changes have not been incorporated into the production cation exchange system at this time.

Americium recovery processing in Lines 30 and 1 was stopped in May, 1979 for equipment repair. This process is scheduled to resume operations in October, 1980.

SUMMARY OF DATA

Figures 3 through 6 are material balances for Lines 1 and 30 for the two production americium recovery processes. Actual volumes and quantities of americium and plutonium for solutions shipped between material balance areas and discard level solutions were available for the current processes. Data was not available for all of the KOH precipitation-thiocyanate ion exchange discard streams. These streams are identified in the material balances.

The quantities of americium and plutonium in the M.S.E. residue salt feed for the current Line 30 process (Figure 5) are calculated values. M.S.E. residue salts are received at Line 30 with an assigned plutonium value. In the past, the assigned plutonium value for the residue salt was calculated by subtracting the weight of the relatively pure M.S.E. plutonium button from the weight of the impure plutonium feed. This is known as a "by-difference" plutonium value. Available production control records did not include any correction to the by-difference value for impurities in the feed and for plutonium losses to gloves, filters, crucibles, etc. Inspection of production records for feed impurity levels and the level of plutonium losses to other residues indicated that approximately 2% of the by-difference plutonium value is neither plutonium or americium.

The plutonium and americium contained in a salt may be estimated by measuring the neutron flux with a germanium-lithium (GeLi) detector. Forty-two packages of M.S.E. residue salts were GeLi counted and the results compared with the by-difference plutonium values. The results are shown in Table 1. The two methods do not agree well on the contents of individual packages, however the by-difference value is only 2.7% higher than the GeLi count value for the total actinide content of the forty-two packages. If 2% of the by-difference value is neither americium or plutonium, then the two methods differ by only 0.7%. The amount of americium and plutonium removed from Line 30 during March '76 through October '79, and GeLi count results from recently generated M.S.E. residue salts both indicate that the americium value is 8% of the plutonium value. From this data the quantities of americium and plutonium in the feed salt were estimated. The plutonium

and americium values for the feed salt to the KOH precipitation process (Figure 3) were estimated from production records.

Table II is a comparison of volume generation and americium recovery efficiency of the present process and the KOH precipitation-thiocyanate ion exchange process. The current process generates 23% less volume overall and recovers 45% more americium from Line 1 feed than did the KOH precipitation-thiocyanate ion exchange process. The current process recovers 54% of the americium in the M.S.E. residue salt in the form of high purity americium oxide, while the KOH precipitation-thiocyanate ion exchange system recovered only 38% of the americium.

Table III shows the quantity of actinides in various plutonium recovery waste streams for March, 1976.

TABLE 1: TABULATION OF PLUTONIUM AND AMERICIUM VALUES
ASSIGNED TO M.S.E. RESIDUE SALTS

DATE	SAMPLE ID'S	Ge-Li Count				By-Difference
		Pu (g)	Am (g)	Am/Pu	Am+Pu	
9-21-78	28919 + 28917	776.06	117.42	0.151	893.48	958.0
3-21-78	28909 + 28908	816.68	99.10	0.121	915.78	995.0
3-10-78	28955 + 28956	742.31	87.0	0.117	829.91	998.0
8-9-78	28937 + 28938	916.0	70.93	0.077	986.93	922.0
7-20-78	28936 + 28935	620.71	64.02	0.103	684.73	932.0
7-20-78	28963 + 28961	804.08	74.91	0.093	878.99	963.0
7-14-78	28934 + 28931	846.7	88.9	0.105	935.60	928.0
7-14-78	28926 + 28925	720.88	83.21	0.115	804.09	928.0
7-12-78	28953 + 28952	789.75	72.02	0.091	861.77	895.0
7-12-73	28928 + 28927	919.2	83.38	0.091	1002.58	924.0
7-6-78	28940 + 28939	828.45	59.06	0.071	887.51	930.0
7-6-78	28916 + 28915	781.09	126.77	0.162	907.86	959.0
4-12-78	28976 + 28947	875.75	35.65	0.041	911.40	998.0
6-9-78	28984 + 28982	914.98	36.56	0.040	951.54	832.0
6-14-78	28968 + 28964	920.42	68.0	0.074	988.42	927.0
6-14-78	28930 + 28929	973.5	83.96	0.086	1057.46	949.0
6-8-73	28941 + 28942	798.98	58.0	0.073	856.98	958.0
6-2-78	28959 + 28957	812.83	77.94	0.096	890.77	893.0
6-2-78	28949 + 28948	964.21	100.78	0.104	1064.99	947.0
4-11-78	28975 + 28969	682.78	89.84	0.132	772.62	984.0
4-17-78	None (drum 11327) N/A	879.01	12.55	0.014	891.56	736.0
4-17-78	11221 (?) N/A	824.69	7.67	0.009	832.36	807.0
Total			1,597.67		19,807.33	
Total		18,209.66				20,363.0

**TABLE II: COMPARISON OF CURRENT PROCESS AND
KOH PRECIPITATION-THIOCYANATE ION EXCHANGE PROCESS**

		<u>CURRENT PROCESS</u>	<u>CAUSTIC PRECIPITATION AND THIOCYANATE ION EXCHANGE</u>
Volume Generated	- Line 30	82.5 ℓ /Kg Salt	17.5 ℓ /Kg Salt
	- Line 1	71.4 "	184.1 "
	- Overall	154.6 "	199.8 "
Am Recovered From Feed	- Line 30	63%	64%
	- Line 1	86%	59.5%
	- Overall	54%	38%
Unaccounted For Am		17%	27%
Volume Generated in Line 1 per Gram Am Product		34.0 ℓ /g Am Product	*230 ℓ /g Am Product

* Concentration of Pu-Am in 8% salt is approximately one-half the Pu-Am concentration in 30% salt. However, doubling the concentration of Am in the Line 1 feed only reduces the volume per gram Am product to 160 ℓ /g.

TABLE III: PLUTONIUM RECOVERY SOLUTIONS TO WASTE TREATMENT
DURING MARCH 1 - MARCH 31, 1976

<u>Source</u>	<u>g Pu</u>	<u>g Am</u>
Caustic Scrub	91	0.4
Anion Effluent	49	128*
Cooling Water	21	0.4
Pu Distillate	29	0.1
Line 30 HCl Waste	14	3.0
Line 1 Distillate	<u>3</u>	<u>10.8</u>
Total	207 g	142.7 g

* 28-64 g of this material came from Line 30 anion eluate.
Note that the americium loss is more than twice the plutonium loss.

DISCUSSION

Current Process

The material balance for the current process, Figures 5 and 6, identifies operational deficiencies in almost every unit operation. Each unit operation was analyzed and possible simple operational changes discussed.

The dissolution operation releases excessive quantities of chloride vapors which accelerates glove-box and equipment corrosion. Incomplete dissolution of americium and plutonium oxides leads to actinide losses in the one micron cartridge filters and in the sludge on the raschig rings in the receiving tanks. Replacement of the current steam coil in an open pot by a simple enclosed reflux dissolver would decrease vapor generation and improve dissolution. A decanting operation followed by a second dissolution with concentrated acid solutions would probably be required to significantly decrease the quantity of insoluble americium and plutonium oxides.

The cation exchange operation presently converts the actinides from a chloride to a nitrate system with partial decontamination of mono and divalent cations. The data from the demonstration runs for the current cation exchange process is detailed in report RFP-2347 by S. G. Proctor.⁽²⁾ Proctor loaded a solution of americium, plutonium, potassium, sodium, and magnesium in 0.5M HCl on Dowex[®] 50x8 H⁺ form cation exchange resin at a rate of 1.4 ml/cm²-min. The loaded columns were washed with water and eluted with 7.2N HNO₃. Proctor reported the following quantities of feed material were recovered in the eluate:

<u>Element</u>	<u>% Recovered in Eluate</u>
Pu	99.8
Am	99.9
Mg	13.5
K	26.6
Na	18.8
Cl	0

[®] Trademark of the Dow Chemical Company, Midland, Michigan.

The resin capacity was 473 g M.S.E. salt/litre resin or 14.0 g actinide/litre resin. The current production process is almost identical to that used by Proctor. Production data substantiates Proctor's data. The material balance shows a resin capacity of 17 g actinide/litre resin with plutonium and americium recovery rates of 99.7% and 99.5% respectively. Chloride carry over is less than one gram per litre. No data is available for recovery rates of potassium, sodium, magnesium and other cation impurities.

The anion exchange system is not operating efficiently. In the eluate stream the Am/Pu ratio is 0.014 to 0.027, which represents a loss of 13% to 30% of the americium fed to Line 30. In the eluate stream, the Pu/Am ratio is 0.033, almost a factor of two higher than the desired 0.017 ratio. The excessive quantities of plutonium in the americium stream necessitated the installation of an anion stripper column in Line 1 which increased waste solution generation and americium losses. A suggestion has been made to use a macroporous resin to allow easier absorption of the bulky plutonium-hexanitrate complex on the resin. Studies should be conducted on various resins using production cation eluates as the anion feed before a new resin could be recommended for production use. A second option would employ the selective citrate elution of americium from a cation column. The americium rich anion effluent would be diluted to 0.4 M H^+ and loaded onto Dowex 50x8 cation exchange resin. The plutonium would be removed with a 0.1M oxalic acid wash and americium eluted with a 0.2M citrate solution. The eluate would then be adjusted to 0.35M H^+ with 7 N HNO_3 and made 0.1M in oxalic acid to precipitate the americium. The dilution of anion effluent to 0.4M is similar to the current dilution of evaporator condensate to 0.5M, so overall solution generation of the citrate elution process would be comparable to present operations.

This process discards excessive amounts of americium, as shown by the material balances, Figures 5 and 6, and Tables II and III. Twenty-four to thirty-eight percent (24-38%) of the americium in Line 30 residue salt feed is discarded to waste treatment by Lines 1 and 30. As a result, Lines 1 and 30 are major contributors of americium in plutonium recovery waste streams. Table III shows that 29-54% of the americium in plutonium recovery waste streams during March 1976 can be attributed to Lines 1 and 30. During March 1976 through October 1979, 4,270 grams of americium were

sent to waste treatment in plutonium recovery streams. Lines 1 and 30 contributed 35 - 64% of this americium. Americium losses in Line 30 anion eluate alone accounts for 22 - 51% of the americium in the discard streams. Possible improvements to the anion system have been discussed. The current process is marginal in terms of personnel radiation exposure. When possible, automatic valves have been installed to decrease operator time in the gloves. A hot cell would be required to significantly decrease radiation exposure received when performing such manual operations as loading residue salts into the dissolver, scraping the americium oxalate from the filter, and weighing and packaging the americium oxide. These manual operations were primarily responsible for the occasional removal of an operator from Line 1 or Line 30 in order to keep their radiation exposure level within guidelines. If the maximum personnel exposure limits are decreased, extensive modifications will have to be made to Lines 1 and 30 or the material throughput decreased.

Proposed Improvements to the Cation Exchange System

It has been demonstrated at Rocky Flats that the capacity for absorption of Pu^{+3} and Am^{+3} on Dowex 50x8 (H^+ form) cation resin can be approximately doubled by decreasing the total molar concentration of the cation feed solution from 2.7 M to less than 1.0 M. This concept is substantiated by Selke who determined that the equilibrium parameter for the exchange of univalent ions for divalent ions on Amberlite^(R) IR-120 is shifted to favor pickup of univalent ions by an increase in concentration. (4)

The nitric acid concentration of the wash was varied during experimental tests in an attempt to increase the decontamination of mono and divalent cations. It was found that if the feed concentration was less than one molar, a water wash followed by a 1.5N HNO_3 wash resulted in the recovery in the eluate of only 9.3% of the calcium fed, 0.7% of the magnesium, 0.1% of the potassium, and 2.5% of the lead. Essentially none of the americium and 0.4% of the plutonium were removed in the 1.5N HNO_3 wash. Further experimentation should be done to confirm these results. If no americium is removed in the acid wash, this plutonium bearing solution can be sent to the anion columns in the main plutonium recovery process.

(R) Trademark of Rohm and Haas Company, Philadelphia, Pennsylvania.

However, if americium is removed by the acid wash a back-up column would be needed to control actinide losses to the waste stream.

The proposed changes appear promising, but the dilution of the salt solution with 0.5M HCl poses severe problems for the present production equipment. One kilogram of 30 wt % MgCl_2 salt residue, bearing an average of 50 grams of plutonium and americium, contains approximately 30 moles of cations and chlorides. The dissolved salt solution would be batched to 50 grams chloride per litre solution with 0.5M HCl to the one total mole per litre level as the solution is fed to the cation columns. Approximately 35 litres solution would be generated per kilogram salt. The cation loading process would have to be interrupted periodically to sample and ship the 225 litre capacity chloride effluent tank. The new process would load 12 Kg salt on 12.3 litres resin compared to the 12-14 Kg salt currently being loaded on 39 litres resin. As a result of the large dilution and the limited capacity of the chloride effluent tank, the proposed operation would require twice the time of the current operation to process a fixed amount of material. However, the eluate from the proposed process would be half the present eluate volume and contain considerably fewer cation impurities. The overall volume generated through Line 30 for the proposed process would be equal to or less than the present volume as a result of decreased downstream volumes.

Implementation of the proposed changes would double the resin capacity but cut the cation exchange throughput in half. The dilution concept might be feasible for production if:

- 1) A lesser dilution could be employed.
- 2) A water and 0.5M HCl solution could be used as the diluent.
- 3) The loading rate could be increased.

The three suggested modifications would decrease the required time for a cation exchange run. As presently proposed, the dilution concept is not a viable production alternative unless the increased purity of the cation eluate drastically improves the anion exchange operation and decreases americium oxide recycle.

Mixing Liquids In Raschig Ring Filled Tanks

The major Line 30 material balance discrepancy is a result of the inadequate mixing of the anion eluate-cation recondition receiving tank, Tank 219. Tank 219 frequently is stratified by layers of differing density and plutonium concentration. The first solution in the tank is often cation reconditioning solution, consisting of 7N HNO_3 and little americium or plutonium. The anion exchange eluate is then added. The first eluate solution contains 5-6N HNO_3 and little plutonium, followed by a smaller quantity of 0.35N HNO_3 containing 200-1000 g plutonium. Prior to sampling, the tank is "vacuum sparged". Vacuum sparging consists of connecting the top of a partially filled tank to a vacuum source, opening the drain line and allowing air to bubble up from the drain line through the solution. When vacuum sparging in a raschig ring tank the bubbles typically channel up the center of the tank, leaving fluid around the sides of the tank not agitated. Mixing studies conducted by M. A. Fredrickson⁽⁵⁾ indicate that it is unlikely that vacuum sparging will mix layers of differing density solutions in a raschig ring filled tank. The tank sample is taken from the drain line at the bottom of the tank where the low concentration plutonium solution is located. Consequently there is high probability that samples from Tank 219 are consistently not representative of the whole tank. Biased Tank 219 samples could easily account for the 13 Kg plutonium and 1.2 Kg americium material balance discrepancy. To improve the accuracy of Line 30 material balance data and to properly assess the americium in the anion eluate, Tank 219 should be replaced with an annular tank designed to facilitate mixing. If an annular tank is not available, a pump to circulate Tank 219 should be obtained. Tank 218, which contains anion effluent solutions of constant density, is repeatedly circulated through an anion exchange stripper column to lower the plutonium concentration. The excellent material balance around Line 1 indicates that Tank 218 was well mixed. Studies need to be made to determine the time and flow rates required to mix layers of differing density solution in a raschig ring filled tank.

CONCLUSIONS

Changes in the current process which do not require extensive equipment modifications and do not increase the risk of higher radiation exposure have been discussed. These changes improve the efficiency of the process but have a limited affect on the design capacity. The current process is state of the art technology for production scale recovery of americium and plutonium from M.S.E. residue salts. If future production schedules are to be met, high capacity process schemes must be developed and the appropriate remote operation facilities obtained to limit personnel exposure.

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2. S. G. Proctor, "Cation Exchange Process for Molten Salt Extraction Residues," RFP-2347, March 10, 1975.
3. D. E. Kudera, R. H. Guyer, "Plutonium and Americium Recovery From a Chloride-Salt Matrix," RFP-1642, June 4, 1971.
4. Nachod Schubert, Ion Exchange Technology, Academic Press, Inc., New York, 1956. p. 60.
5. M. A. Fredrickson, "A Quantitative Study of Mixing in Annular and Raschig Ring Filled Tanks," RFP-2998

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**DATE
FILMED**

1-22-87

Appendix Q
Courier Receipt

ATOMIC ENERGY COMMISSION
Form AEC-48 (2-63)

COURIER RECEIPT

N- 2622

I have received: No. of envelopes 2 No. of packages 1 No. of containers 1

Transmitted by DAN CHEMICAL CO. BURY FRANKLIN
(Name and address of sender)

Addressed to POU. URTS - BURIAL GROUNDS
(Name and address of addressee)

Identified as follows: TWO 5-GAL. CONTAINERS OF WASTE

T7-21 & T7-20 - UNLABLED

Clifford H. Thomas #460
(Name of person from whom received (if courier, include Courier Card Number))

L. J. Schmitt
(Signature of recipient (if courier, include Courier Card Number))

BURIAL GROUNDS 11-23-63
(Place of transfer) (Date)

Appendix R
Waste Shipment Data

TRAILER NO. T2-20

SEAL NO. 2121

SERIAL NO. DOW 624
63-72-

WASTE SHIPMENT DATA

(FILL IN ALL BLANK SPACES. IF INFORMATION NOT APPLICABLE OR NOT KNOWN, SO STATE. ATTACH EXTR SHEETS AND DRAWINGS SECURELY IF NEEDED.)

1. NUMBER OF PACKAGES IN PROPOSED SHIPMENT (1) 158-55 Gal Drums
2. TOTAL WEIGHT OF PROPOSED SHIPMENT 26,132
3. VOLUME IN CUBIC FEET OF PROPOSED SHIPMENT 1,161 Cu. ft.
4. COMPLETE DESCRIPTION OF CONTENTS AND PACKAGING METHODS (1) 17-55 Gal Drums of 74 ser sludge. (2) 141-55 Gal Drums of paper, rags, scrap metal, etc.
5. HIGHEST RADIATION LEVELS AT OUTSIDE SURFACE OF PACKAGES 4 MR/HR
6. CHEMICAL FORM OF MATERIAL COMPRISING BULK OF WASTE Combustible Type I
7. CORROSIVE ACIDS OR BASES CONTAINED IN WASTE None
STRENGTH OF SUCH CORROSIVE ACIDS OR BASES _____
8. ANY PORTION OF THE CONTENTS ARE () ARE NOT (X) POTENTIALLY EXPLOSIVE. GIVE DETAILS IF POTENTIALLY EXPLOSIVE _____
9. ANY PORTION OF THE CONTENTS WILL () WILL NOT (X) REACT VIOLENTLY WHEN EXPOSED TO AIR () WATER. GIVE DETAILS IF CONTENTS WILL REACT VIOLENTLY WHEN EXPOSED TO AIR OR WATER. _____
10. RADIOISOTOPES CONTAINED Uranium and Plutonium
11. ESTIMATED QUANTITY IN CURIES Not Feasible
12. SPECIAL PRECAUTIONS THAT MAY BE NECESSARY BECAUSE OF BACTERIAL CONTENT None
13. OTHER INFORMATION THAT MAY BE USEFUL IN REDUCING SHIPPING AND HANDLING HAZARDS None
14. PROPOSED MEANS OF TRANSPORTATION D. and R.G.W.R.R. (Motor Trailer and Rail)
15. APPROXIMATE DATE OF SHIPMENT October 31, 1963
Based 11-23-63 JH

TRAILER NO. TZ-21

SEAL # 2122

SERIAL NO. DOW 614
63-73-B

WASTE SHIPMENT DATA

FILL IN ALL BLANK SPACES. IF INFORMATION NOT APPLICABLE OR NOT KNOWN, SO STATE. ATTACH EXTRA SHEETS AND DRAWINGS SECURELY IF NEEDED.)

1. NUMBER OF PACKAGES IN PROPOSED SHIPMENT (1) 155-55 Gal Drums

2. TOTAL WEIGHT OF PROPOSED SHIPMENT 27,962

3. VOLUME IN CUBIC FEET OF PROPOSED SHIPMENT 1,139 cu. ft.

4. COMPLETE DESCRIPTION OF CONTENTS AND PACKAGING METHODS (1) 25-55 Gal Drum of 74 series sludge. (2) 130-55 Gal Drums of paper, rags, scrap metal, etc.

5. HIGHEST RADIATION LEVELS AT OUTSIDE SURFACE OF PACKAGES 1.5 MR/HR

6. CHEMICAL FORM OF MATERIAL COMPRISING BULK OF WASTE Combustible Type I

CORROSIVE ACIDS OR BASES CONTAINED IN WASTE None
STRENGTH OF SUCH CORROSIVE ACIDS OR BASES _____

ANY PORTION OF THE CONTENTS ARE () ARE NOT () POTENTIALLY EXPLOSIVE. GIVE DETAILS IF POTENTIALLY EXPLOSIVE _____

ANY PORTION OF THE CONTENTS WILL () WILL NOT () REACT VIOLENTLY WHEN EXPOSED TO AIR OR WATER. GIVE DETAILS IF CONTENTS WILL REACT VIOLENTLY WHEN EXPOSED TO AIR OR WATER.

RADIOISOTOPES CONTAINED ~~BY~~ Uranium and Plutonium

ESTIMATED QUANTITY IN CURIES Not Feasible

SPECIAL PRECAUTIONS THAT MAY BE NECESSARY BECAUSE OF BACTERIAL CONTENT None

OTHER INFORMATION THAT MAY BE USEFUL IN REDUCING SHIPPING AND HANDLING HAZARDS None

PROPOSED MEANS OF TRANSPORTATION D. and R.G.W.R.R. (Motor and Rail)

APPROXIMATE DATE OF SHIPMENT October 31, 1963

3. Serial. 11-22-63 AH

TRAILER NO. TZ-21

SEAL # 2122

SERIAL NO. DOW 61

63-73-

WASTE SHIPMENT DATA

FILL IN ALL BLANK SPACES. IF INFORMATION NOT APPLICABLE OR NOT KNOWN, SO STATE. ATTACH EXT SHEETS AND DRAWINGS SECURELY IF NEEDED.)

1. NUMBER OF PACKAGES IN PROPOSED SHIPMENT (1) 155-55 Gal Drums

2. TOTAL WEIGHT OF PROPOSED SHIPMENT 27,962

3. VOLUME IN CUBIC FEET OF PROPOSED SHIPMENT 1,139 cu. ft.

4. COMPLETE DESCRIPTION OF CONTENTS AND PACKAGING METHODS (1) 25-55 Gal Drum of 74 seri sludge. (2) 130-55 Gal Drums of paper, rags, scrap metal, etc.

5. HIGHEST RADIATION LEVELS AT OUTSIDE SURFACE OF PACKAGES 1.5 MR/HR

6. CHEMICAL FORM OF MATERIAL COMPRISING BULK OF WASTE Combustible Type I

7. CORROSIVE ACIDS OR BASES CONTAINED IN WASTE None
STRENGTH OF SUCH CORROSIVE ACIDS OR BASES _____

ANY PORTION OF THE CONTENTS ARE () ARE NOT () POTENTIALLY EXPLOSIVE. GIVE DETAILS IF POTENTIALLY EXPLOSIVE _____

ANY PORTION OF THE CONTENTS WILL () WILL NOT () REACT VIOLENTLY WHEN EXPOSED TO AIR OR WATER. GIVE DETAILS IF CONTENTS WILL REACT VIOLENTLY WHEN EXPOSED TO AIR OR WATER.

RADIOISOTOPES CONTAINED ~~By~~ Uranium and Plutonium

ESTIMATED QUANTITY IN CURIES Not Feasible

SPECIAL PRECAUTIONS THAT MAY BE NECESSARY BECAUSE OF BACTERIAL CONTENT None

8. OTHER INFORMATION THAT MAY BE USEFUL IN REDUCING SHIPPING AND HANDLING HAZARDS None

PROPOSED MEANS OF TRANSPORTATION D. and R.G.W.R.R. (Motor and Rail)

APPROXIMATE DATE OF SHIPMENT October 31, 1963

rec'd. 11-27-63 AH

Appendix S

Load Lists

RF. 2121

Dow 63-72-

1963

No.	Drum No.	Gross Weight	Drum Size	Type	No.	Drum No.	Gross Weight	Drum Size	Type
	✓ 71-4841	146		I	32.	✓ 71-4977	106		I
	✓ 71-4855	110		I	33.	✓ 76-899	131		I
	✓ 71-4662	102		I	34.	✓ 71-4716	96		I
	✓ 76-894	104		I	35.	✓ 71-9915	116		I
5.	✓ 71-4920	106		I	36.	✓ 71-4719	142		I
	✓ 76-902	100		I	37.	✓ 71-4908	126		I
7.	✓ 71-4919	88		I	38.	✓ 71-4757	132		I
	✓ 76-892	88		I	39.	✓ 71-4924	82		I
9.	✓ 71-4901	124		I	40.	✓ 71-4769	108	7154	I
10.	✓ 76-904	104		I	41.	✓ 76-1050	95		I
11.	✓ 76-1027	129		I	42.	✓ 71-4856	124		I
	✓ 71-4762	132		I	43.	✓ 76-1048	65		I
13.	✓ 742-7826	577		IV	44.	✓ 71-4852	136		I
14.	✓ 741-3525	570		IV	45.	741-3522	546		IV
15.	✓ 742-7864	547		IV	46.	✓ 742-7877	552		IV
16.	✓ 742-7875	541		IV	47.	✓ 742-7863	550		IV
17.	✓ 76-895	77		I	48.	✓ 742-7870	563		IV
18.	✓ 76-893	70		I	49.	✓ 71-4766	98		I
19.	✓ 742-7867	516		IV	50.	✓ 71-4907	100		I
20.	✓ 741-3523	586		IV	51.	✓ 71-4914	134		I
21.	✓ 71-4918	86		I	52.	✓ 71-4913	84		I
22.	✓ 76-900	146		I	53.	✓ 742-7885	550		IV
23.	✓ 71-4909	144		I	54.	✓ 742-7872	581		IV
24.	✓ 71-4720	144		I	55.	✓ 742-7878	548		IV
25.	✓ 71-4916	106		I	56.	✓ 742-7861	562		IV
26.	✓ 76-1026	100		I	57.	✓ 76-1016	94		I
27.	✓ 71-4763	102		I	58.	✓ 71-4777	116		I
28.	✓ 76-1042	93		I	59.	✓ 71-4765	130		I
29.	✓ 71-4756	132		I	60.	✓ 71-4718	126		I
30.	✓ 71-4848	140		I	61.	742-7883	559		IV
31.	✓ 76-905	105		I	62.	✓ 742-7886	555		IV

TOTALS

6,115

7,907

Date: 31 Oct. 63

Carrier: Rex Shingleton

Trailer No: 17

Bottom
Shipped 11-19-63

158-55 Dallas Pans
4 MR/HR
1.161 cu ft.

1137

RF 2122

DOW 23-73-B

1963

No.	Drum No.	Gross Weight	Drum Size	Type	Drum No.	Gross Weight	Drum Size	Type
1.	✓ 76-1057	104		I	✓ 71-4850	136		I
2.	✓ 76-1049	120		I	✓ 71-4843	138		I
3.	✓ 71-4972	118		I	✓ 71-4847	124		I
4.	✓ 71-4967	108		I	✓ 71-4661	112		I
5.	✓ 76-1059	98		I	✓ 71-4844	134		I
6.	✓ 76-1065	106		I	✓ 76-901	140		I
7.	✓ 71-4946	96		I	✓ 71-4951	60		I
8.	✓ 71-4887	92		I	✓ 71-4947	94		I
9.	✓ 71-4999	108		I	✓ 71-4853	126		I
10.	✓ 71-4891	96		I	✓ 71-4954	130		I
11.	✓ 71-4900	136		I	✓ 71-4948	132		I
12.	✓ 76-1007	109		I	✓ 76-860	126		I
13.	✓ 742-7859	570		IV	✓ 71-4945	82		I
14.	✓ 742-7824	606		IV	✓ 742-7823	585		IV
15.	✓ 742-7808	581		IV	✓ 742-7869	546		IV
16.	✓ 742-7848	555		IV	✓ 742-7880	560		IV
17.	✓ 76-859	103		I	✓ 742-7871	560		IV
18.	✓ 76-1001	99		I	✓ 71-4664	116		I
19.	✓ 76-889	132		I	✓ 76-1002	94		I
20.	✓ 76-994	107		I	✓ 76-861	139		I
21.	✓ 741-3488	625		IV	✓ 76-1008	85		I
22.	✓ 741-3469	551		IV	✓ 742-7884	547		IV
23.	✓ 742-7887	609		IV	✓ 741-3527	574		IV
24.	✓ 741-3464	538		IV	✓ 742-7862	552		IV
25.	✓ 71-4681	128		I	✓ 742-7857	543		IV
26.	✓ 76-993	84		I	57. ✓ 76-858	92		I
27.	✓ 742-7879	551		IV	58. ✓ 76-1004	84		I
28.	✓ 742-7825	581		IV	59. ✓ 71-4678	120		I
29.	✓ 71-4668	128		I	60. ✓ 76-1006	65		I
30.	✓ 71-4949	80		I	61. ✓ 741-3526	563		IV
31.	✓ 71-4948	90		I	62. ✓ 741-3528	536		IV
TOTALS		7,909				7,893		

Date: 3/10/63

Carrier:

Trailer No: TZ-6

Bottom

Shipped 11-19-62

1.5 mms/hr

155-55 GAL DRUM

1.5 MR/HR

1.70 FL

RF 2122

5,802

DOW 63-7:

No.	Drum No.	Gross Weight	Drum Size	Type	No.	Drum No.	Gross Weight	Drum Size	Type
1.	✓71-4922	140		/	32.	✓71-4701	102		/
2.	✓71-4883	126		/	33.	✓71-4790	92		/
3.	✓71-4884	130		/	34.	✓71-4709	90		/
4.	✓71-4954	108		/	35.	✓71-4803	94		/
5.	✓76-1110	77		/	36.	✓71-4708	86		/
6.	✓71-4658	126		/	37.	✓71-4715	122		/
7.	✓76-991	70		/	38.	✓71-4800	96		/
8.	✓71-4663	86		/	39.	✓71-4802	112		/
9.	✓71-11764	98		/	40.	✓71-4679	142	9.36 ⁴¹⁶⁴	/
10.	✓76-875	84		/	41.	✓71-4798	138		/
11.	✓76-896	64		/	42.	✓71-4740	120		/
12.	✓71-4659	106		/	43.	✓71-4712	96		/
13.	✓76-863	105		/	44.	✓71-4742	150		/
14.	✓76-864	116		/	45.	✓71-4801	130		/
15.	✓71-4748	106		/	46.	✓71-4746	124		/
16.	✓71-4682	136		/	47.	✓71-4714	76		/
17.	✓76-978	106		/	48.	✓71-4753	122		/
18.	✓71-4676	148		/	49.	✓71-4710	84		/
19.	✓71-4705	94		/	50.	✓71-4736	120		/
20.	✓71-4677	108		/	51.	✓71-4788	102		/
21.	✓71-4703	112		/	52.	✓71-4888	108		/
22.	✓71-4685	92		/	53.	✓71-4797	104		/
23.	✓71-4686	84		/	54.	✓71-4738	122		/
24.	✓71-4693	98		/	55.	✓71-4799	68		/
25.	✓76-890	86		/	56.	✓71-4752	118		/
26.	✓71-4690	122		/	57.	✓71-4792	124		/
27.	✓71-4875	112		/	58.	✓71-4739	124		/
28.	✓71-4688	96		/	59.	✓71-4796	120		/
29.	✓71-4669	94		/	60.	✓71-4749	84		/
30.	✓71-4691	120		/	61.	✓71-4794	148		/
31.	✓76-891	98		/	62.	✓71-4745	92		/

TOTALS 3,248

3,412

Date: 31 Oct '63

Carrier: Rio Grande

Trailer No: 72-21

No.	Drum No.	Gross Weight	Drum Size	Type	No.	Drum No.	Gross Weight	Drum Size	Type
1.	✓742-7881	554		IV					
2.	✓741-3508	496		IV					
3.	✓71-4687	110		I					
4.	✓76-979	90		I					
5.	✓71-4695	142		I					
6.	✓76-1003	100		I					
7.	✓742-7856	587		IV					
8.	✓742-7843	537		IV					
9.	✓742-7873	593		IV					
10.	✓76-865	101		I					
11.	✓76-995	110		I					
12.	✓76-876	65		I					
13.	✓76-980	104		I					
14.	✓71-4674	99		I					
15.	✓76-965	118		I					
16.	✓71-4684	92		I					
17.									
18.									
19.									
20.									
21.									
22.									
23.									
24.									
25.									
26.									
27.									
28.									
29.									
30.									
31.									

TOTALS

3,897

Date: 31 Oct '63

Carrier: Rio Grande

Trailer No: T2-21

Bottom

19,699

No.	Drum No.	Gross Weight	Drum Size	Type	No.	Drum No.	Gross Weight	Drum Size	Type
1.	✓ 71-4986	138		/	31.				
2.	✓ 71-4966	88		/	32.				
3.	✓ 76-963	124		/	33.				
4.	✓ 71-4889	124		/	34.				
5.	✓ 71-4963	106		/	35.				
6.	✓ 76-998	76		/	36.				
7.	✓ 71-4665	118		/	37.				
8.	✓ 71-4912	90		/	38.				
9.	✓ 71-4667	124		/	39.				
10.	✓ 71-4922	134		/	40.				
11.	✓ 76-897	78		/	41.				
12.	✓ 71-4911	72		/	42.				
13.	✓ 76-903	101		/	43.				
14.	✓ 71-4859	108		/	44.				
15.	✓ 71-4758	122		/	45.				
16.					46.				
17.		2476			47.				
18.					48.				
19.					49.				
20.					50.				
21.					51.				
22.					52.				
23.					53.				
24.					54.				
25.					55.				
26.					56.				
27.					57.				
28.					58.				
29.					59.				
30.					60.				
31.					61.				
					62.				

FRONT
 Top 4,184 = 47%
 Bottom 8,971
 13,155

REAR
 Top 4,079 = 53%
 Bottom 10,728
 14,807

27,962

TOTALS 1,603

Date: 31 Oct '63 Carrier: Rio Grande Trailer No: T2-2

2nd

Appendix T
Letter to J. S. Corbett

Mr. J. S. Corbett
Nuclear Engineering Company, Inc.
P. O. Box 218
Pleasanton, California

Dear Mr. Corbett:

Subject: Wastes Shipped Off-Site To Idaho Falls, Idaho,
During The Eleven Month Period - July, 1963,
Thru May, 1964 - From Rocky Flats Plant

The following information is to fulfill your request made during
your visit of May 8, 1964.

The tabulation does not include classified wastes shipped to
Hanford, Washington, nor the sludges produced in the decon-
tamination of liquid wastes. The plutonium content of the
sludges would prevent their disposal by Nuclear Engineering
under present regulations.

Also enclosed is a tabulation of the wooden boxes shipped as
to size and number.

<u>Type of Container</u>	<u>Number</u>	<u>Volume Cu. Ft.</u>	<u>Weight Lbs.</u>
30 gal. drums	4	16	594
55 gal. drums	10,450	76,656	1,732,828

Appendix U
Graphite Specifications

GRAPHITE SPECIALTY PRODUCTS

UCAR

TECHNICAL INFORMATION

Since 1896, UCAR Carbon Company has been supplying quality carbon and graphite products, skilled technical service, and experienced application assistance worldwide.

GRADE CS EXTRUDED GRAPHITE

Densified prior to graphitization, Grade CS is an extruded graphite 25-30% higher in strength than CBY. Grade CS is also a low ash graphite with an oxidation resistance similar to grade CBY. It is the material of choice for applications requiring higher densities, better mechanical strength, or less porosity than grade CBY. Grade CS is the most popular multi-purpose grade.

Typical Applications

- Casting molds and furnace parts for hot metal
- Powder metallurgy
- Boats and trays for sintering applications
- Resistance heating
- Crucibles for melting and alloying

Sizes

CS Extruded: 3 - 30 x 72" diameter rounds
 15 x 30 x 72", 17 x 17 x 72", 17 x 17 x 88"
 19 x 19 x 72", 24 x 24 x 72" rectangles

Typical Properties @ Room Temperature

	ENGLISH	W.G.	A.G.	METRIC	W.G.	A.G.
Density	lbs/ft ³	108		g/cm ³	1.74	
Maximum Particle Size	inches	0.03		mm	0.76	
Specific Resistance	10 ⁻⁴ Ω in	2.8	3.7	μΩm	7.1	9.4
Flexural Strength	psi	3060	2260	kg/cm ²	216	160
Young's Modulus	10 ⁶ psi	1.58	1.20	kg/mm ²	1120	840
Tensile Strength	psi	2150	1600	kg/cm ²	150	110
Compressive Strength	psi	6600	6600	kg/cm ²	470	470
Permeability	Darcy's	0.05	0.04	Darcy's	0.05	0.04
Hardness	Rockwell "R"	83		Rockwell "R"	83	
C.T.E. (to 100°C)	10 ⁻⁶ /°F	1.4	2.0	10 ⁻⁶ /°C	2.5	3.6
Thermal Conductivity	BTU-ft/hr ft ² °F	80	85	W/mK	160	145
Ash Content	ppm	800		ppm	800	

THINKING IN GRAPHITE

UCAR CARBON COMPANY, INC.

UCAR

This information is not to be taken as a warranty or representation for which we assume legal responsibility nor as permission or recommendation to practice any patented invention without a license.



Appendix V

Drum Counter Verification Studies (CRDL 940232-101C)

DRUM COUNTER VERIFICATION STUDIES
Plutonium/Graphite in the Drum Counter

CRDL-940232-1010

December 30, 1965

O. H. Willoughby

L. D. Delpierre

Analytical Instrumentation Group

Chemistry Research and Development Laboratories

THE DOW CHEMICAL COMPANY

Rocky Flats Division

Distribution:

K. V. Best

J. T. Byrne

D. R. Cartwright

L. C. Farrell

Attn: E. D. Erickson

R. A. Frohreich

J. N. Vance

L. V. Grill

L. D. Hazelton (2)

M. E. Maas

J. R. Mann

E. A. Putzier

R. F. Rogers

E. S. Ryan

R. W. Sorenson

E. H. Stowers

A. K. Williams

INTRODUCTION

In the production of certain plutonium parts, graphite is used as a mold material. The waste products of the casting operation are graphite fragments, sweepings, and scarfings containing plutonium in varying concentrations. Accurate plutonium assay of these waste products is difficult due to inhomogeneity. Plutonium content of the waste is assigned in a rather arbitrary manner, agreeable to both the AEC and Dow. Although the arbitrary assignment procedure satisfies accountability requirements, an accurate assay method is needed before the graphite waste materials can be transferred to another facility for recovery or disposal. Experiments were performed using plutonium in graphite in the drum counter to determine if the counter will provide an accurate assay of the material in question.

EXPERIMENTAL

Since no convenient method could be devised for quantitatively distributing plutonium oxide (PuO_2) throughout massive fragments of graphite, an experimental technique was devised to simulate the expected distribution as closely as possible. PuO_2 was chosen as an experimental material, because it was believed that most of the plutonium associated with graphite wastes is in the oxide form.

Fourteen samples were prepared by sandwiching about 4 grams of PuO_2 between graphite blocks measuring 2 in. by 2 in. by 3/8 in. Shallow depressions to hold the PuO_2 were drilled in the mating surfaces of each pair of blocks. The blocks were filled with PuO_2 , taped together and transferred from the "glove-box line" into paper cartons.

The samples were drum counted individually and then in groups of 3, 7, 11, and 14. A comparison of drum-counter results with those obtained by multiplying the weight of PuO_2 per sample by the gram per gram assay² is shown in Table 1.

* Plutonium assay was performed by the Plutonium Control Laboratory using two different methods. The gravimetric factor obtained from these methods is accurate to within 1%. An average value is used in calculating results in this report.

Next, three samples of PuO₂ in graphite were placed in a drum and surrounded by uncontaminated graphite fragments. Drum-counting followed. This operation was repeated with 7, 11, and 14 samples in the drum surrounded by graphite. In each instance an even distribution of samples throughout the graphite-filled drum was achieved.

RESULTS

Table 1 illustrates the comparison of drum counter results with those computed on the basis of laboratory analyses of the PuO₂. The ratio shown is a comparison of the apparent plutonium content of the drum of graphite fragments with the actual plutonium content.

TABLE 1
Plutonium-Graphite in the Drum Counter

Weight Pu (Grams) Lab Analyses	Wt. (Grams) Pu - Drum Counter Without Graphite Fragments	Wt. (Grams) Pu - Drum Counter With Graphite Fragments	Ratio
8.99 Group of 3	9.22	3.94	2.3:1
22.64 Group of 7	23.94	11.70	1.9:1
37.80 Group of 11	42.02	19.36	2.0:1
48.81 Group of 14	55.73	27.95	1.7:1

CONCLUSIONS

Graphite surrounding plutonium will attenuate the emitted gamma flux and hence create an error in drum counter measurements. In this experiment the reduction was in a 2 to 1 ratio, i.e., one gram of plutonium appears as one-half gram to the drum counter. The 2 to 1 ratio is constant throughout the range of from 0 to 50 grams Pu. In this range, the reduction ratio depends on the graphite content of a drum, i.e., the quantity of attenuating material present. The 2 to 1 reduction ratio has not been proven for all forms of graphite wastes and at this time should only be used as a guide in evaluating the actual plutonium content of a drum of waste graphite.

Future experiments will evaluate other forms of Pu-graphite wastes.

Appendix W

Drum Counter Verification Studies
(CRDL 940232-101D)

DRUM COUNTER VERIFICATION STUDIES
Plutonium/Graphite in the Drum Counter

CRDL 940232-101D

May 27, 1966

O. H. Willoughby

G. J. Cunningham

Analytical Development Group
Chemistry Research and Development Laboratories

THE DOW CHEMICAL COMPANY
Rocky Flats Division

DISTRIBUTION:

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R. A. Frohsich	R. W. Sorenson
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ABSTRACT

The work regarding plutonium in graphite fragments described in CRDL 940232-101C was extended using standards prepared by the Chemical Standards Laboratory. A factor of 1.85 ± 0.18 was derived for use in correcting apparent plutonium content of graphite filled drums to actual content.

INTRODUCTION

There are approximately 800 drums of waste graphite mold materials on the plant site to which a sizable quantity of plutonium has been somewhat arbitrarily assigned. For purposes of improved accountability procedures and recovery decisions, it was desired to have a more accurate estimate of the actual quantity of plutonium contained in the graphite. Although the drum counter offered the only readily available means of measuring the plutonium content of these drums, previous experiments had shown that gamma attenuation by graphite lowers the apparent plutonium content of waste graphite fragments.

At the request of K. V. Best, Metal Production--Pu Chemistry, the work with Pu in graphite (CRDL 940232-101C) was extended using standards prepared by the Chemical Standards Laboratory. Through this experiment, a factor was derived for converting the indicated Pu content of previously counted drums to actual content. A standard source was prepared for use in future counting of graphite filled drums.

EXPERIMENTAL

Sixty-five medical capsules were filled with a total of 50 grams of plutonium as the oxide and encapsulated in one dram plastic vials. Preparation of these oxide standards was performed by the Chemical Standards Laboratory. The chemical and isotopic composition, weight and agreement integrity of the standards are tabulated in the following table.

TABLE I

Measurement ARN	S-00181	
Direct Assay	0.8834	g/g
100%-impurity assay	0.8816	g/g
Theoretical purity	0.8830	g/g
Chem SL purity	0.8817	g/g
Light elements	<5	ppm
Pu-240	5.76	atom %
Am-241 (3/23/66)	14	ppm
X-ray diffraction parameter	0.594	
Theoretical X-ray diffraction parameter	0.596	
Total PuO ₂	56.786	g
Average PuO ₂ per vial	0.874	g/g
Total Pu SV	50.068	g
Average Pu SV per vial	0.770	g/g
SD between vials (Pu)	0.018	g/g
Relative SD between vials (Pu)	2.33	%

The vials were placed in slots machined in fragments representing both ingot and part molds. The slots were covered with paper cushions to prevent breakage of the vials in handling. Five of the fragments were evenly distributed throughout a drum of uncontaminated graphite fragments (280 lbs. net). The drum was then counted using usual drum counter techniques. Next a group of 10 then 20, 30, 40, 50, 60, and 65 standard fragments were distributed throughout the graphite and counted between each grouping. The experiment was extended above the 50 gram level using previously prepared oxide standards. The apparent plutonium content of the graphite filled drum was computed after each counting using the standard drum-counter formula.

* Although the exact plutonium content of the individual standards was unknown, statistical analysis of the data supplied by the Standards Lab showed that there was a 99.3% chance of the samples weighing within $\pm 5\%$ of calculated value at the 3.9 gram level and greater than a 99.3% chance at higher levels. In other words, the calculated plutonium content of the drum containing 5, 10, 20, 30, 40, 50, and 60 vials was well within the detectability limits of the drum counter. When 65 vials were used, the actual plutonium content was accurately known.

RESULTS

Table II illustrates the comparison of actual plutonium content to apparent content based on drum counter calculations. The ratio was calculated by dividing actual content by apparent content.

TABLE II

<u>Number of Vials in Drum</u>	<u>Actual Weight (grams)</u>	<u>Apparent Weight (grams)</u>	<u>Ratio</u>
5	3.85	1.65	2.33:1
10	7.69	4.02	1.91:1
20	15.38	9.05	1.70:1
30	23.08	12.07	1.91:1
40	30.77	17.39	1.77:1
50	38.46	23.15/21.45*	1.74/1.79:1
60	46.15	25.80	1.79:1
65	50.00	30.19	1.66:1
Additional Stds.	62.32	34.18	1.82:1
Additional Stds.	80.63	42.97	1.87:1

* Standard vials redistributed and recounted as a check on distribution technique.

Statistical evaluation of the data showed an average apparent loss of 1.85. The standard deviation about the mean is 0.18, and the relative standard deviation is 9.66%.

DISCUSSION - CONCLUSIONS

In drum counting contaminated graphite fragments, an apparent plutonium loss in the ratio of 1.85:1 is encountered. That is 1.85 grams of plutonium will appear as 1 gram, hence, when counting drums of graphite fragments, the apparent plutonium content must be multiplied by 1.85 to achieve the true content. Application of this correction factor does not apply to graphite sweepings or to other forms of plutonium waste materials.

- 5 -

There can be no certainty in correctness of applying the correction factor to drums of graphite fragments containing more than an apparent 200 grams of plutonium. In fact, little reliance can be placed on any drum counter reading above the 200 gram level. Work is underway to extend the drum counter range to 1 Kg plutonium. However, until this work is completed, drum counter readings above 200 grams plutonium can only be used as a relative indication of actual content.

Appendix X

**Letter to Hazardous Materials Regulation Board,
Department of Transportation,
from William F. Romine**

December 23, 1968

T-3-4 #4

Hazardous Materials Regulations Board
Department of Transportation
Washington, D.C. 20590

Gentlemen:

REQUEST FOR DOT SPECIAL PERMIT

We request that The Department of Transportation issue The Dow Chemical Company, Rocky Flats Division, a special permit for shipments of packages of radioactive and fissile materials transported in ATMX 600 series rail cars.

1. Specific portion(s) of the regulation for which the special permit is being requested.

Sections 173.395(e)(2) and 173.396(c)(3).

2. Justification for the permit, reasons why the regulations are not appropriate, why the public interest would be served and the basis upon which the proposal would provide at least an equivalent degree of safety to that provided by the regulation concerned.

We produce approximately 2000 drums and 60 crates of fissile material and radioactive contaminated equipment annually. Economic waste discard limits are established for our various processes; however, these limits far exceed the amounts of radioactive and fissile materials permitted in one container as specified in the current transportation regulations. Because of the volume of waste produced at Rocky Flats it is necessary that we maximize the use of a container that will give us the greatest carrying capacity. We have examined the use of a 30/55-gallon combination container and the Brookhaven concrete vault. Neither appear to be economically adaptable to the volume of waste generated at Rocky Flats. Neither of the above are specification containers and would require special permits. It seems to us that the use of the ATMX 600 series car would provide the most economical method of transporting our waste. The present transportation regulations are not appropriate in that the specification containers do not have sufficient volume for waste products.

Using a DOT Spec 6M container for our processed waste in lieu of a 55-gallon drum would increase the number of containers shipped by over a factor of twenty. Container costs would be increased by a factor of over two-hundred-and-fifty. The DOT Spec 6M container is a reusable container; however, since the waste is buried in the shipping container there is no opportunity to reduce container costs by reusing the containers. Also we must face the problem that in many instances the line-generated waste could not be packaged in the DOT Spec 6M container. There are no specific regulations that authorize the use of wooden crates for equipment contaminated with plutonium in the amounts that are released by Rocky Flats. We believe that the public interest would be served by our spending the least amount of government money to provide a safe waste package. It is our conclusion that the proposed ATMX 600 series car would provide a safe package arrangement that would offer an equivalent degree of safety during transportation to that provided by the present regulations.

3. Detailed description of the ATMX 600 series rail car.

See the attached container description, evaluation and U.S. Atomic Energy Commission, Albuquerque Operation's certification.

4. The type, form, quantity, properties and characteristics of the material being transported.

The material involved is americium 241 and a mixture of several plutonium isotopes predominately Pu-239, Pu-240 and Pu-241 contained in sludges, greases, line-generated wastes or fixed on equipment, machinery, and miscellaneous process equipment. Specific activity of the plutonium when adjusted for isotopic ratios, is approximately 0.75 curies per gram plutonium.

In the grease, americium and plutonium may exist as either fine solids of metal or oxide, or dissolved in the organic materials at an average concentration of 5×10^{-10} g/g or 1.62×10^{-9} c/g for the americium, and 2×10^{-6} g/g or 1.5×10^{-6} c/g for the plutonium.

In the sludge, americium and plutonium exist as the hydroxide along with those of iron, aluminum, etc., at an average concentration of 1.07×10^{-5} g/g or 3.48×10^{-5} c/g for the americium and 4.71×10^{-5} g/g or $3/53 \times 10^{-5}$ c/g for the plutonium.

Line-generated wastes are graphite molds, filter sludge, insulation, glass, washables, combustibles, metals and miscellaneous residues with plutonium discard limits ranging from 7×10^{-3} g/g to 3×10^{-4} g/g. Washables consist of such things as plastics and rubber gloves

impregnated with plutonium oxide. Combustibles include paper, rags and plastic bags generated in the plutonium processing areas. Miscellaneous residues consist of line-generated solids which cannot be ground for leaching and other small quantities of such things as metal, solid americium waste, filters and miscellaneous process equipment.

Crated wastes are large or bulky items that would not fit in a 55-gallon drum and include such things as pipe, lumber, equipment, plywood, windows, light fixtures, metal scrap, lathes, machines, hoods and air ducts. These objects are externally contaminated with plutonium to varying degrees. The contamination is fixed and is not readily dispersible.

The waste materials will be packaged as follows:

Sludge: A Quantity of dry portland cement is placed in a Specification 17-C or 17-H steel drum of at least 30-gallon capacity which is then lined with a 5-mil polyethylene liner. Additional dry portland cement is interspersed as the sludge fills the container. After filling, the liner is sealed and a quantity of dry portland cement is added. The drum is then closed with a 12-gauge bolted ring closure.

Grease: A quantity of oil-dry is placed in a Specification 17-C or 17-H steel drum of at least 30-gallon capacity which is then lined with a 5-mil polyethylene liner. After filling, the liner is sealed and the drum is closed with a 12-gauge bolted ring closure.

Line-Generated Waste: The material is bagged in a polyethylene bag and removed from the line into a Specification 17C or 17-H steel drum of at least 30-gallon capacity which has been lined with a 5-mil polyethylene liner. After filling, the liner is sealed and the drum is closed with a 12-gauge bolted ring closure. No more than 200-pounds of graphite is loaded in any one drum.

Crated Waste: The crate is constructed of at least 3/4" plywood according to Specification 19-A or 19-B, and is lined with an 8-mil polyethylene sheet. 6" x 6" or larger skids are used when the contents exceed 5,000 pounds and a 1" plywood floor is used. Heavier pieces of equipment is secured in the crate by being bolted to the skid. After filling, the crate is closed. After closing, the crate is banded with 1 1/4" steel strapping in at least four places.

Loading Limitations: In addition to the 200 pound per drum limitation on the graphite there is a 200 gram limitation on fissile material

loaded in a 55-gallon drum or a 100 gram limitation on fissile material loaded in drums of less than 55-gallon capacity. Each drum is also limited to a thermal decay energy of 2-watts.

5. Shipping and accident experience with the container type being proposed.

None as this is a new concept of using the rail car as the outer container.

6. Name the proposed mode of transportation, and describe the transport controls needed.

We propose to ship in carload lots, exclusive use, via rail freight as Fissile Class II where the number of 30-gallon drums does not exceed 200 or the number of 55-gallon drums does not exceed 286. Intermixing of 30 and 55-gallon drums will be permitted provided that $X/200$ plus $Y/286$ is less than one where X equals the number of 30-gallon drums and Y equals the number of 55-gallon drums.

When the number of drums exceed the above stated number the shipment will move as Fissile Class III on an exclusive use basis. We are proposing to use one ATMX 600 series car and the car will be assigned to the sole use of The Dow Chemical Company, Rocky Flats Division.

The rail car will be sealed with a U.S. Government Red-Ball seal with an attached tag bearing the following inscription:

*Notice to Carriers
Government Seals
DO NOT BREAK!

If broken make immediate contact with:

Traffic Department
The Dow Chemical Company
Denver, Colorado
PHONE: 444-3311 (COLLECT)

The attached card (Form RF-26540) is attached to the shipping memo of each bill of lading.

The consignee is notified of the departure time of the car, its estimated time of arrival and the seal number applied to the car.

The carrier notifies us of the arrival at destination.

The rail car must be transported near the center of the train not closer than the sixteenth car from either the headend or rearend of the train whenever possible.

During switching the rail car must be shoved to rest and no other car(s) must be "humped" into it.

The rail car must not be transported next to cars of explosives or film or open-top cars loaded with lading such as poles or pipe that might act as projectiles during an accident.

7. State the name, address, and telephone number of the applicant.

The Dow Chemical Company, Rocky Flats Division, Post Office Box 888, Golden, Colorado, 80401, Telephone, 444-3311, Extension 2377. All inquiries should be directed to William F. Romine, Traffic Manager.

8. Loading Procedures: The ATMX 600 series car is divided into three bays. Drums and/or crates will be loaded on pallets which are secured within the car. Inflatable dunnage shall be used to prevent shifting. Crates over 3,000 pounds shall be loaded on the bottom tier. All loading and unloading will be done on AEC or AEC Contractor's property. The load limit per car will be 90,000 pounds.

9. Labeling and Radiation Levels: The containers will be labeled according to the provisions of 173.399. The rail car will be placarded according to the provisions of 174.541. Container radiation levels will comply with the provisions of 173.393.

10. Recommendations regarding any changes to the regulations which would be desirable to obviate the need for similar special permits.

None. Very few installations are capable of shipping or receiving radioactive waste in rail car shipments. The ATMX 600 series cars are owned by the U S Atomic Energy Commission, Albuquerque Operations, are limited in number (14), and are used in other applications. It is doubtful that all of these cars could be released for waste loading. If these cars had not been designed and built for explosive ordinance transport we probably would not have progressed with the idea to use the rail car as the outer package.

Your approval of the ATMX 600 series rail car as a special permit container will be appreciated.

ORIGINAE SIGNED BY
WILLIAM F. ROMINE
William F. Romine
Traffic Manager

WFR:
Orig. and 1 cc - Hazardous Materials Regulations Board
Encs.

Appendix Y

Packaging Certification by Atomic Energy Commission—ALO Contractor (Rev. 3-7-69)

CERTIFICATION OF APPROVAL FOR FISSILE-LARGE QUANTITY SHIPPING CONTAINERS

ALBUQUERQUE OPERATIONS OFFICE, USAEC

Rev. 3-7-69

I. ALO Contractor
The Dow Chemical Company
Rocky Flats Division
Post Office Box 888
Golden, Colorado 80401
Contact: Engr. - Frank E. Adcock
Traffic - W. F. Romine

II. Identification of Shipping Container
ATMX-600 Series Railcar
Serial No. ATMX-600 thru-614

ALO Designation: AL-R9
DOT Special Permit;

III. General Information Concerning Container :

A. Packaging shall consist of an ATMX-600 Series railcar loaded with drums or crates as described below. Inside dimensions of the car are 9'-4" width, 9'-2" height, and 50'-0" length. Useful load is 90,000 pounds.

B. Authorized contents shall consist of non-radioactive material contaminated with radioactive materials that are in a form such that they are not readily dispersible.

Process waste (hardened oils, sludges, greases, etc.) and line-generated waste (plastic, glass, ceramics, metals, etc.) shall be packaged in ICC-17C or-17H or equivalent steel drums of at least 30-gallon capacity.

Machinery, process apparatus, and other large items of plant equipment shall be packaged in nailed and glued crates of at least 3/4" plywood. Minimum size shall be 48"x48"x84".

C. Fissile Class I shipments are authorized provided the following restrictions are met:

Drums: Graphite shall be limited to 200-pounds and thermal decay energy to 2-watts per drum. Fissile material shall be limited to 100-grams for 30-gallon and 200-grams for 55-gallon drums.

Crates: Fissile material shall be limited to 5-grams in any cubic foot.

IV. Specific Limitations and Restrictions.

Car(s) to be positioned, when possible, not nearer than the 16th car from both engine or occupied caboose. Requirements of DOT paragraph 174.589 for placarded cars to be observed.

V. Additional Information. None.

VI. Certification of Approval.

Pursuant to Chapter AEC 0529, this container is approved subject to the limitations described above. This certification does not relieve the shipper of his responsibility to obtain DOT Special Permit and to comply with the requirements of other Federal Regulations as appropriate.

Date: _____

Certification Official
Albuquerque Operations
U. S. Atomic Energy Commission

Appendix Z

Building 44 Discarded Waste Report— Sources of Roaster Oxide

**BUILDING 44
DISCARDED WASTE REPORT**

Sources of Roaster
Oxide

Date February, 1962

Doc. No. 654

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description
			Gross	Tare	Net	SS Net	
780496 ✓✓✓			273	17	256		Mach. Shop Oxide
780497 ✓✓			209	18	191		Mach. Shop Oxide
780498 ✓✓✓	Sampled		224	16	208		Mach. Shop Oxide
600735 ✓✓✓			261	15	246		Crucible Oxide
780499 ✓✓✓			244	16	228		Mach. Shop Oxide
780500 ✓✓✓			248	16	232		Mach. Shop Oxide
600736 ✓✓✓			218	15	203		Crucible Oxide
600737 ✓✓✓	Sampled		335	15	320		Crucible Oxide
780501 ✓✓✓			215	16	199		Mach. Shop Oxide
780502 ✓✓✓			215	16	199		Mach. Shop Oxide
7503 ✓✓✓			244	16	228		Mach. Shop Oxide
780504 ✓✓✓			238	16	222		Mach. Shop Oxide
780505 ✓✓✓			274	16	258		Mach. Shop Oxide
780506 ✓✓✓			276	16	260		Mach. Shop Oxide
610195 ✓✓✓	Sampled		219	15	204		Vacuum Oxide
600738 ✓✓✓			332	15	317		Crucible Oxide
✓780507 ✓✓			280	16	264		Mach. Shop Oxide
780508 ✓✓✓			272	16	256		Mach. Shop Oxide
780509 ✓✓✓			250	14	236		Mach. Shop Oxide
600739 ✓✓✓			397	15	382		Crucible Oxide
780510 ✓✓✓			207	16	191		Mach. Shop Oxide
780511 ✓✓✓			252	16	236		Mach. Shop Oxide

APPROVED

APC Department

RAM Date 3-5-62

A. W. ...
Shipped by

E. J. ...

Received by

BUILDING 44
DISCARDED WASTE REPORT

Date May, 1962

Doc. No. 667

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description	
			Gross	Tare	Net	SS Net		
780575 ✓	Sampled 416509		227	16	211		Mach. Shop Oxide	
780576 ✓✓			290	16	274		Mach. Shop Oxide	
780577 ✓✓			256	16	240		Mach. Shop Oxide	
780578 ✓✓			240	16	224		Mach. Shop Oxide	
600756 ✓✓			361	15	346		Cruc. Oxide	
780579 ✓✓			282	16	266		Mach. Shop Oxide	
780580 ✓✓			219	16	203		Mach. Shop Oxide	
780581 ✓✓			242	16	226		Mach. Shop Oxide	
780582 ✓✓			242	18	224		Mach. Shop Oxide	
600757 ✓✓✓		Sampled		365	15	350		Cruc. Oxide
780583 ✓				221	17	204		Mach. Shop Oxide
780584 ✓				233	17	216		Mach. Shop Oxide
600758 ✓				348	15	333		Cruc. Oxide
780585 ✓			263	17	246		Mach. Shop Oxide	
780586 ✓			250	17	233		Mach. Shop Oxide	
780587 ✓			269	17	252		Mach. Shop Oxide	
780588 ✓			250	17	233		Mach. Shop Oxide	
600759 ✓			361	15	346		Cruc. Oxide	
780589 ✓			258	17	241		Mach. Shop Oxide	
780590 ✓			290	17	273		Mach. Shop Oxide	
780591 ✓			294	16	278		Mach. Shop Oxide	
780592 ✓✓			274	16	258		Mach. Shop Oxide	
780593 ✓✓		315	17	298		Mach. Shop Oxide		
780594 ✓		270	17	253		Mach. Shop Oxide		
600760 ✓✓		364	15	349		Cruc. Oxide		
600761 ✓		357	15	342		Cruc. Oxide		
780595 ✓		280	17	263		Mach. Shop Oxide		
780596 ✓		313	17	296		Mach. Shop Oxide		
780597 ✓		271	16	255		Mach. Shop Oxide		
780598 ✓		252	16	236		Mach. Shop Oxide		
610200 ✓	Sampled		220	15	205		Vac. Oxide	
			8677.00	503.00	8174.00			

E. E. Sutton Jr.

Shipped by

E. E. Sutton Jr.

[Signature]

BUILDING 44
DISCARDED WASTE REPORT

ite March, 1962

Doc. No. 460

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description
			Gross	Tare	Net	SS Net	
780518 ✓✓	Sampled		232	16	216	Machine Shop Oxide	
780519 ✓✓✓			246	16	230	Machine Shop Oxide	
780520 ✓✓✓			265	16	249	Machine Shop Oxide	
780521 ✓✓			254	16	238	Machine Shop Oxide	
600742 ✓✓	Sampled		410	15	395	Crucible Oxide	
780522 ✓✓✓			273	16	257	Machine Shop Oxide	
780523 ✓✓✓			253	16	237	Machine Shop Oxide	
780524 ✓✓✓			302	16	286	Machine Shop Oxide	
780525 ✓✓		267	16	251	Machine Shop Oxide		
780526 ✓✓		295	16	279	Machine Shop Oxide		
600743 ✓✓✓✓		358	15	343	Crucible Oxide		
780527 ✓✓✓✓		281	16	265	Machine Shop Oxide		
780528 ✓✓✓✓		290	18	272	Machine Shop Oxide		
780529 ✓✓✓✓		298	17	281	Machine Shop Oxide		
780530 ✓✓✓✓		257	19	238	Machine Shop Oxide		
780531 ✓✓		266	17	249	Machine Shop Oxide		
600744 ✓✓		352	15	337	Crucible Oxide		
780532 #2		291	17	274	Machine Shop Oxide		
533 ✓✓✓✓		257	16	241	Machine Shop Oxide		
534 ✓✓✓		329	18	311	Machine Shop Oxide		
535 ✓✓		295	16	279	Machine Shop Oxide		
780536 ✓✓✓		273	16	257	Machine Shop Oxide		
600745 ✓✓		366	14	352	Crucible Oxide		
780537 ✓✓		293	16	277	Machine Shop Oxide		
780538 ✓✓✓		245	16	229	Machine Shop Oxide		
780539 ✓✓		257	16	241	Machine Shop Oxide		
780540 ✓✓		245	18	227	Machine Shop Oxide		
780541 ✓✓✓		284	16	268	Machine Shop Oxide		
780542 ✓✓✓		261	16	245	Machine Shop Oxide		
600746 ✓✓✓		383	15	368	Crucible Oxide		
610197 ✓✓✓	Sampled		273	15	258	Vacuum Oxide	
780543 ✓✓			286	16	270	Machine Shop Oxide	
780544 ✓✓			320	16	304	Machine Shop Oxide	
780545 ✓✓			299	16	283	Machine Shop Oxide	
780546 ✓✓			313	16	297	Machine Shop Oxide	
600747 ✓✓✓			362	15	347	Crucible Oxide	
610198 ✓✓			272	15	257	Vacuum Oxide	
600748 ✓✓			352	15	337	Crucible Oxide	
780547 ✓✓			284	15	269	Machine Shop Oxide	
780548 ✓✓			271	14	257	Machine Shop Oxide	
780549 ✓✓			266	16	250	Machine Shop Oxide	
TOTALS				16976	655	11321	

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Department

March 1962

[Signature]
Shipped by

[Signature]

BUILDING 44
DISCARDED WASTE
PLANT SITE

Date August 1962

Destination _____

677

Batch No.	Analytical Req. No.	M* T C	A* M C	Gross Weight Kgs.	Tare Weight Kgs.	Net Weight Kgs.	SF Net* Weight Kgs.	Scale No.	Material Description
780617 ✓✓✓✓				244	14	230			Machine Shop Oxide
780618 ✓✓✓✓				220 230	14	216			" "
780619 ✓✓✓✓	417085			233	14	219			" "
600768 * ✓	417116			254	14	240			Cruc Oxide
780620 ✓✓✓✓				222	14	208			Machine Shop Oxide
780621 ✓✓✓✓				247	14	233			" "
780622 * 2				238	14	224			" "
610202 * 2	417117			228	20	202 208			House Vac Oxide
780623 ✓				209	14	195			Machine Shop Oxide
780624 ✓				242	14	228			" "
780625 ✓				244	14	230			" "
780626 ✓				241	14	227			" "
780627 ✓				239	14	225			" "
780628 ✓				239	14	225			" "
600769 ✓				215	14	201			Cruc Oxide
				220	212	208			

to be filled in by APC

APPROVED
APC Department

[Signature]
Shipped by
E.A.R.

BUILDING 44
DISCARDED WASTE REPORT

September, 1962

Doc. No. 683

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description
			Gross	Tare	Net	SS Net	
780629 ✓			226	17	209		Machine Shop Oxide
780630 ✓			235	14	221		" " "
780631 ✓			240	14	226		" " "
600770 ✓✓✓	Sampled 9-6-62		250	15	235		Cruc. Oxide
780632 ✓			270	20	250		Machine Shop Oxide
780633 ✓✓✓	Sampled 9-13-62		225	15	210		" " "
780634 ✓			240	17	223		Machine Shop Oxide
780635 ✓✓✓			235	17	218		" " "
780636 ✓✓✓			230	17	213		" " "
780637 ✓✓✓			258	17	241		" " "
780638 ✓✓✓			235	17	218		" " "
600771 ✓✓✓			314	14	300		Cruc. Oxide
600772 ✓✓			342	15	327		" "
780639 ✓✓✓			240	17	223		Machine Shop Oxide
780640 ✓✓✓			237	14	223		" " "
780641 ✓✓✓			231	14	217		" " "
780642 ✓✓			236	14	222		" " "
780643 ✓✓			231	14	217		" " "
780644 ✓✓✓			345	15	330		Cruc. Oxide
780645 ✓✓✓			235	14	221		Machine Shop Oxide
620024 ✓✓✓			258	17	241		" " "
780646 ✓✓✓✓			237	15	222		Fab. Oxide
780647 ✓✓✓✓			227	14	213		Machine Shop Oxide
600774 ✓			231	14	217		" " "
610203 ✓	Sampled 10-1-62		340	15	325		Cruc. Oxide
780652 ✓			317	15	302		House Vac. Oxide
780653 ✓	Sampled 10-1-62		227	14	213		Machine Shop Oxide
780651 ✓			228	14	214		" " "
780648 ✓			240	15	225		" " "
780649 ✓			231	14	217		" " "
780650 ✓			225	17	208		" " "
780654 ✓✓			233	14	219		" " "
780655 ✓✓			277	14	263		Machine Shop Oxide
780656 ✓✓			244	14	230		" " "
			273	14	259		Machine Shop Oxide

Total 8312

Shipped by *R. L. Hanson*

Received by *E. J. R.*

RECEIVED
DATE 10/17/62

BUILDING 44
DISCARDED WASTE REPORT

Date November, 1962

Doc. No. 692

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description
			Gross	Tare	Net	SS Net	
✓ 600780 #1	417260		304	15	289		Crucible Oxide
✓ 610204 #1	417261		250	10	240		Vacuum Oxide
✓ 780682 #1	417293		246	15	231		Mach. Shop Oxide
✓ 600783 #1			318	15	303		Crucible Oxide
✓ 600781 #1			339	15	324		" "
✓ 780683 #1			241	15	226		Mach. Shop Oxide
✓ 780684 #1			289	15	274		" " "
✓ 780685 #4			221	15	206		" " "
✓ 780686 #2			250	15	235		" " "
✓ 780687 #2			273	15	258		" " "
✓ 780688 #2			238	15	223		" " "
✓ 600783 #2			299	15	284		Crucible Oxide
✓ 780689 #2			245	15	230		Mach. Shop Oxide
✓ 780690 #2			259	15	244		" " "
					Total 3567		

Shipped by

ELR

Received by

BUILDING 44
DISCARDED WASTE REPORT

December, 1962

Doc. No. 697

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description
			Gross	Tare	Net	SS Net	
✓780691 #1	417352		235	15	220	Mach. Shop Oxide	
✓780692 #3			241	15	226	" " "	
✓600784 #1	417373		350	22	328	Crucible Oxide	
✓780693 #3			254	15	239	Mach. Shop Oxide	
✓620025 #4	417375		264	15	249	Fabrication Oxide	
✓610205 #1		417381		302	15	287	House Oxide
✓600785 #4			318	16	302	Crucible Oxide	
✓780694 #4			230	15	215	Mach. Shop Oxide	
✓780695 #4			237	15	222	" " "	
✓780696 #4			235	15	220	" " "	
✓780697 #4			240	15	225	" " "	
✓600786 #2			268	15	253	Crucible Oxide	
✓600787 #2			342	15	327	" "	
✓780698 #2			240	15	225	Mach. shop Oxide	
✓780699 #2			263	15	248	" " "	
✓780700 #1			244	15	229	Roaster Oxide	
✓780701 #1			251	15	236	" "	
✓780702 #1			270	15	255	" "	
✓780703 #1			258	15	243	" "	
			2042	293	Total 1749		

APPROVED
A. C. Department
Date: Dec. 9, 1962

R. L. Murray
Shipped by

E. A. R.
Received by

BUILDING 44
DISCARDED WASTE REPORT

Date 3-29-66

Doc. No. _____

Batch or Drum No.	Analytical Req. No.	M T C	Weight in Kilograms				Material Description
			Gross	Tare	Net	SS Net	
Drum no.4			197	14	183	165	U + 10% MO.Scrap
Drum no.6			198	14	184	166	U + 10% MO.Scrap
Drum no.3			199	14	185	166	U + 10% MO.Scrap
Drum no.5			188	14	174	157	U + 10% MO.Scrap
Drum no.2			196	14	182	164	U + 10% MO.Scrap
Drum no.1			203	14	189	170	U + 10% MO.Scrap
<p><i>This is Joe Bolando scrap.</i></p> <p><i>From Ken Gallahan to 4/2/66</i></p>							

Ed Mastella
Shipped by

Received by

Appendix AA

**Letter to R. D. Gaskins from
William F. Romine**

T-3-6.3.1

December 19, 1969

R. D. Gaskins

cc:

M. A. Vaas

E. A. Putzier

SHIPPING ION EXCHANGE RESIN

F. E. Adcock's letter dated December 12th on the above subject was in response to my query regarding the advisability of shipping ion exchange resins on a 1 to 1 basis.

Please follow his suggestion in mixing up to $1\frac{1}{2}$ parts of resin to one part of cement for shipment in 55 gallon drums. These drums must be then shipped in the ATIX Cars.

William F. Romine

WFR:q

cc:

✓ file 3-4#4

Appendix BB

Non-Salvable Contaminated Waste for Disposal, “B” Plant (July 1964–September 1965)

Building 881
Decommissioning Waste



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

July 17, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - JUNE

The following list of non-salvabel waste from "B" Plant is ready for disposal. All drums are of 55 gallon capacity.

<u>Drum No.</u>	<u>Material</u>	<u>Gross Wt. Lbs.</u>	<u>Serial No.</u>
78-S	Discard Oil	476	60-8829
79-S		483	"
80-S		314	"
81-S		418	"
82-S	Discard Perclene	768	60-8414
84-S		767	8413
83-S	Discard Triclene	695	60-8817
85-S AUG 15 1964	Discard Mud	300	60-8844
86-S AUG 15 1964		354	8830
89-S AUG 15 1964		274	8816
90-S AUG 15 1964		495	8815
91-S AUG 15 1964		500	8802
92-S AUG 15 1964		418	8802
87-S AUG 15 1964	Graphite	371	60-8827
88-S AUG 15 1964		349	8828

A. M. McNeill

A. M. McNeill
Metal Production-81

AMc:mr



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

July 17, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - JUNE

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

<u>Drum No.</u>	<u>Material</u>	<u>Gross Wt. Lbs.</u>
81-74 AUG 15 1964	Miscellaneous Dry	72
75 AUG 15 1964	" "	76
76 AUG 15 1964	" "	Box ?
77 SEP 31 1965	" "	Box ?
78 SEP 31 1965	" "	Box ?
79 AUG 15 1964	" "	85

A. M. McNeill
A. M. McNeill
Metal Production-81

AMc:mr



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

August 7, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - JULY

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums are of 55 gallon capacity.

Table with 4 columns: Drum No., Material, Gross Wt. Lbs., Serial No. Rows include Discard Oil (drums 93-S to 111-S), Graphite (drums 104-S to 109-S), Discard Mud (drums 106-S to 108-S), and Triclene (drum 112-S). Total weight is 14,759 lbs.

A. M. McNeill
A. M. McNeill
Metal Production-81

AMc:mr



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

August 7, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - JULY

The following list of non-salvable waste from "B" plant is ready for disposal. All drums listed are of 55 gallon capacity.

<u>Drum No.</u>	<u>Material</u>	<u>Gross Wt. Lbs.</u>
81-80 AUG 1 1964	Glass	239
81 AUG 15 1964	Noncombustibles	
82 AUG 15 1964	"	
84	"	
DEC 1 1964 87 thru 102	"	5,944
DEC 15 1964	"	6,183
DEC 15 1964	"	
DEC 15 1964	"	
DEC 15 1964	"	
81-83 NOV 1 1964	Miscellaneous Dry 8-7x48x52	Unknown 5000 #
85 OCT 1 1964	"	"
86	" 7-5x48x52"	" 1950 #
81-98 JUN 28 1965	" Bot	

A. M. McNeill
A. M. McNeill
Metal Production-81

AMc: mr

DEC 1 1964 -94
DEC 1 1964 -94
DEC 1 1964 -95



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

September 21, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - AUGUST

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums are of 55 gallon capacity.

<u>Material Type</u>	<u>No. of Drums</u>	<u>Drum No.</u>	<u>Serial No.</u>	<u>Gross Wt. Lb.</u>
Misc. Dry	1	103	DEC 10 1964	77
Broken Glass	3	106 - 108		673
Misc Dry	2	NOV 14 1964 104 - 105	Wooden Boxes NOV 14 1964	Unknor
Triclene		113-S	60-8955	
		114-S	8954	
		116-S	8929	
		119-S	8960	
		120-S	8961	
				3,333
Perclene		115-S	60-8941	
		117-S	8910	
		121-S	8959	
				2,284
Mud		118-S	60-8930	
		122-S	8956	
		123-S	DEC 10 1964 8958	
		124-S	8957	
				1,040

A. M. McNeill
A. M. McNeill
Metal Production-81

AMc:mr

DEC 10 1964 /68
DEC 10 1964 /67
DEC 10 1964 /66



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

October 15, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - SEPTEMBER

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with columns: Material Type, No. of Drums, Drum No., Serial No., Gross Wt. Lb. Includes entries for Miscellaneous Dry, Glass, Plastics, Graphite, Mud, Percelene, Tricelene, and Oil.

A.M. McNeill/R. Miller

A. M. McNeill
Metal Production-81

AMc:mr



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

November 18, 1964

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - OCTOBER

The following list of non-salvable waste from "B" plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with 5 columns: Material Type, No. of drums, Drum No., Serial No., Gross Wt. Lb. Includes entries for Triclene, Mud, Graphite, Misc. Dry, and Pu Waste Dry with various drum and serial numbers and dates.

EPR-273

MAR 27 1965 559-2

A. M. McNeill
Metal Production-81

AMc/mr



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

December 8, 1964

16
8
11/11/64

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - NOVEMBER

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

<u>Drum No.</u>	<u>Material Type</u>	<u>Gross Wt. Lb.</u>	<u>Serial No.</u>
81-136	Misc. Dry Pu	112	
137	" "	107	
138	" "	123	
139	" "	106	
140	" " Pu	104	
141	" "	240	
142	" "	Box ?	
143	" " <i>21R-123C</i>	Box ?	
144	" "	Box ?	
145	" "	Box ?	
142-S	Discard Mud	315	60-9133
143-S	Graphite	306	60-9110
144-S	Discard Perclene	807	60-9106
145-S	Discard Triclene	734	60-9105
146-S	Discard Perclene	722	60-9108
147-S	Discard Perclene	807	60-9107
148-S	Discard Oil	410	60-9109
149-S	Discard Oil	459	60-9109
150-S	Discard Oil	448	60-9109
151-S	Oil from boiler room	421	
152-S	Discard Mud	264	60-9088

A. M. McNeill
A. M. McNeill
Metal Production-81

AMc/mr



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

January 18, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - DECEMBER

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

<u>Drum No.</u>	<u>Material Type</u>	<u>Gross Wt. Lb.</u>	<u>Serial No.</u>
153-S APR 10 1965	Discard Mud	301	60-9203
154-S APR 10 1965	Discard Mud	286	60-9204
155-S APR 10 1965	Discard Mud	215	60-9190
156-S APR 10 1965	Discard Mud	288	60-9152
157-S APR 10 1965	Discard Mud	284	60-9151
158-S APR 10 1965	Discard Mud	305	60-9150
159-S	Discard Oil	311	60-9149
160-S	Discard Oil	413	60-9149
161-S	Discard Oil	373	60-9149
162-S	Triclene	641	60-9189
163-S	Discard Oil	390	60-9226
164-S	Triclene	720	60-9253
165-S	Discard Oil	441	60-9205
166-S	Discard Oil	449	60-9206
167-S	Discard Oil	442	60-9206
168-S	Discard Oil	437	60-9226
169-S	Discard Oil	368	60-9208
170-S	Discard Oil	436	60-9208
171-S	Discard Oil	434	60-9208
172-S	Discard Oil	468	60-9207
173-S 60-9207 1965	Discard Oil	461	60-9207
174-S	Discard Oil	370	60-9206
175-S	Discard Oil	225	60-9207
176-S	Discard Oil	403	60-9205
177-S	Discard Oil	425	60-9226
178-S	Discard Oil	297	60-9188
179-S	Discard Oil	470	60-9188
180-S 60-9188 1965	Discard Oil	476	60-9188
181-S	Discard Perclene	775	60-9218
182-S	Discard Oil	453	60-9205

<u>Drum No.</u>	<u>Material Type</u>	<u>Gross Wt. Lb.</u>	<u>Serial No.</u>
146 APR 10 1965	Misc. Dry	Box ?	
147 APR 10 1965	Misc. Dry	Box ?	
148 APR 10 1965	Misc. Dry	Box ?	
149 APR 10 1965	Misc. Dry	Box ?	
150 APR 10 1965	Misc. Dry	Box ?	
151 APR 10 1965	Misc. Dry	Box ?	
152 APR 10 1965	Misc. Dry	Box ?	
153 APR 10 1965	P. U. Contaminated Waste	88	
154 APR 10 1965	P.U. Contaminated Waste	94	
155 APR 10 1965	P.U. Contaminated Waste	113	
156 APR 10 1965	P.U. Contaminated Waste	79	
157 APR 10 1965	P.U. Contaminated Waste	57	

A. M. McNeill

A. M. McNeill
Metal Production - 81

AMc/ms



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

April 7, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - MARCH

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with 4 columns: Drum No., Material Type, Gross Wt. Lb., Serial No. Includes entries for Discard Mud, Discard Oil, Perclene, Triclene, Paint and Oil, and Miscellaneous Dry.



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

May 10, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - APRIL

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with 4 columns: Drum No., Material Type, Gross Wt. Lb., Serial No. Includes entries for Discard Mud, Graphite, Discard Perc., Discard Triclene, Misc. Liquid, and P.U. Dry Waste.

A. M. McNeill
Metal Production - 81



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

June 9, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - MAY

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with 4 columns: Drum No., Material, Gross Wt. Lbs., and Serial No. The table lists various waste materials such as Graphite, Mud, Discard Oil, and Triclene, along with their respective weights and serial numbers.

Non-Salvable Contaminated Waste for Disposal,
 "B" Plant - May

-2-

June 9, 1965

<u>Drum No.</u>	<u>Material</u>	<u>Gross Wt.</u> <u>Lbs.</u>	<u>Serial No.</u>
81-240	MAY 22 1965 U ²³³ Waste	96	
241	"	94	
242	"	127	
243	"	89	
244	"	115	
245	"	136	
246	"	85	
247	"	83	
248	"	230	
249	"	170	
250	"	138	
251	"	112	
252	"	112	
253	"	111	
254	"	100	
255	"	216	
256	"	101	
257	"	108	
258	"	106	
259	"	156	
260	"	106	
261	"	111	
262	"	101	
263	"	94	
264	")	
265	")	
266	") Bldg. 71 Drums	
267	") of (233) Waste -	
268	") No weight	
269	") available.	
270	")	
271	JUL 27 1965 Pu Waste	98	
272	JUL 27 1965 "	85	
273	JUL 27 1965 "	77	
274	JUL 27 1965 Dry Waste	270	
275	U ²³³ Waste	64	
276	OCT 1 1965 "	Box?	
277	DEC 13 1965 "	Box? 1600 #	
278	JAN 17 1965 Misc. Dry Waste	Box?	
279	DEC 13 1965 "	Box? 2100 #	
280	"	Box?	
281	DEC 13 1965 "	Box? 1300 #	
282	DEC 13 1965 "	Box? 850 #	

U-233

A. M. McNeill



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

July 9, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT -JUNE

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with 4 columns: Drum No., Material, Gross Wt. Lbs., and Serial No. It lists various waste items such as 'Discard perclene', 'Discard oil', and 'Filter mud' with their respective weights and serial numbers.

A. M. McNeill
Metal Production-81



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

August 5, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - JULY

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

<u>Drum No.</u>	<u>Material Type</u>	<u>Gross Wt. Lb.</u>	<u>Serial No.</u>
263-S	Discard Perclene	759	60-9583
264-S	Discard Perclene	767	60-9582
065-S	Discard Perclene	796	60-9537
266-S	Discard Oil	425	60-9565
267-S	Discard Oil	468	60-9550
268-S	Discard Oil	424	60-9565
269-S	Discard Oil	434	60-9565
270-S	Discard Oil	464	60-9550
271-S	Discard Oil	445	60-9550
272-S OCT 11 1965	Graphite	327	60-9554
273-S OCT 11 1965	Graphite	343	60-9558
274-S OCT 11 1965	Graphite	394	60-9560
275-S OCT 11 1965	Graphite	363	60-9557
276-S OCT 11 1965	Graphite	343	60-9555
277-S OCT 11 1965	Graphite	375	60-9556
278-S OCT 11 1965	Graphite	384	60-9559
279-S OCT 11 1965	Discard Mud	327	60-9584
280-S	Discard Mud Special	103	60-9548
281-S OCT 11 1965	Discard Mud Special	97	60-9548
282-S JUL 27 1965	Discard Mud Special	92	60-9548
283-S JUL 27 1965	Discard Mud Special	148	60-9554
284-S	Oil And Paint Thinner	524	
285-S OCT 11 1965	Discard Mud	420	60-9613
286-S	Discard Mud	395	60-9586
287-S	Discard Oil	397	60-9586
288-S	Discard Oil	398	60-9586
289-S	Discard Oil	436	60-9585

August 5, 1965

<u>Drum No.</u>	<u>Material Type</u>	<u>Gross Wt. Lb.</u>	<u>Serial No.</u>
290-S	Discard Oil	469	60-9585
291-S	Discard Oil	426	60-9585
292-S	Discard Oil	400	60-9586
293-S	Discard Mud	375	60-9614
81-296	OCT 11 1965 Miscellaneous Dry	137	
297	NOV 15 1965 Miscellaneous Dry	Box ? 1500 #	
298	NOV 15 1965 Miscellaneous Dry	Box ? 1400 #	
299	JAN 1 1966 Miscellaneous Dry	Box ?	
300	NOV 15 1965 Miscellaneous Dry	Box ? 1500 #	
301	OCT 11 1965 Pu Waste	125	
302	OCT 11 1965 Pu Waste	82	
303	OCT 11 1965 Pu Waste	85	
304	OCT 11 1965 Pu Waste	100	
305	OCT 11 1965 Pu Waste	132	
306	OCT 11 1965 Pu Waste	90	
307	OCT 11 1965 Pu Waste	67	
308	Used Oil	456	

A. M. McNeill

A. M. McNeill

Metal Production - 81



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

September 14, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - AUGUST

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with columns: Drum No., Material Type, Gross Wt. Lb., Serial No. Includes entries for P. U. Waste, Misc. Dry Waste, and Discard Mud with various dates and weights.

A. M. McNeill
Metal Production - 81

AMM:jal

Appendix CC

Non-Salvable Contaminated Waste for Disposal, “B” Plant (February)



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO

March 5, 1965

E. S. Ryan

cc:
File

NON-SALVABLE CONTAMINATED WASTE FOR DISPOSAL, "B" PLANT - FEBRUARY

The following list of non-salvable waste from "B" Plant is ready for disposal. All drums listed are of 55 gallon capacity.

Table with 4 columns: Drum No., Material Type, Gross Wt. Lb., Serial No. Lists items 193-S through 204-S and 81-177 through 190.

A. M. McNeill
Metal Production - 81

Appendix DD

Building 883 Crate Number 8716215

Numbers for Crates

8716 222 -
8716 223 -

PDR 3859

HOIST & TRACTOR MONORAIL, 1-TON

D-83-0350-000000

D-83-0351-000000

2 EA

EXCESS

MONORAIL, 1-TON WINTERER

D-83-0347-000000

CLEANER, OIL, ULTRASONIC

D-83-0133-000000

PDR 3826

SALT BATH

D-83-0145-000000

SALT BATH HOOD

D-83-0305-000000

PDR 3719

TABLE, DAB SPRAY BOOTH

D-83-0387-000000

BOOTH, SPRAY DAB DIVILBLISS

D-83-0384-000000

INSTRUMENT PROTECTORANCE

V D-83-0064-032072

PDR 3724

FILTER, SALT S/S 6 EA

D83-0535-000000

D83-0536-000000

D83-0537-000000

D83-0538-000000

D83-0539-000000

D83-0540-000000

2 crates

Crates Numbers { 8716213 }
 { 8716214 }

Bldg 84

PDR - 3555

77 - 1604 - 00000

77 - 1606 - 00000

77 - 876 - 96799

77 - 877 - 96799

CAPITAL EQUIPMENT IN WASTE BOXES

BOX NO. 883-24

PDR-4798

Burner Gas

D-83-0631-000

D-83-0632-000

PDR-4816

Pump

D-83-0219-034731

PDR-4972

Scaffold

D-83-285-45029

Cart

D-81- -7534

PDR-4976

Lincoln Welder

D-34-741-29231

BOX NO. 883-25 ✓

Salt Filters

PDR-4798

D-83-0541-000

D-83-0542-000

D-83-0602-000

D-83-0603-000

D-83-0604-000

D-83-0605-000

D-83-0606-000

D-83-0607-000

D-83-0608-000

Oil Filters

~~XXXXXXXXXX~~
D-83-0553-000

D-83-0554-000

D-83-0612-000

D-83-0613-000

D-83-0614-000

D-83-0615-000

D-34-741-29231 - Lincoln Welder PDR 4976

D-83-0794-075029 S/N 105044

D-83-0674-047911 S/N 6843

D-83-285-45029 - scaffold parts PDR 4972

D-81- -7534 - Cart parts PDR 4976

BOX NO. 883-33 - PDR 4972

D-83-285-45029 - Scaffold parts PDR 4972

Appendix EE
Crated Wastes

THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO 80401

November 5, 1969

O. L. Burton
D. R. Gammel
P. T. Godesiabois
M. L. Manrahan
A. L. Luman
W. L. Ramer

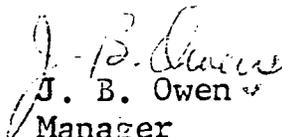
CRATED WASTES

Recent discussions have clarified the handling and labeling of crated wastes resulting from the decontamination of Buildings 776 and 777. The following pertains to the area south of between column lines F & G and east of column line 12 only.

1. A conscientious effort has been made and must continue to package all wastes in 55 gallon drums whenever reasonably possible.
2. All items which will not fit into 55 gallon drums are to be packaged in wooden crates. The crates are to be surveyed for gamma radiation before they are sealed.
 - a. If the gamma radiation is less than 0.5 mr/hr the label on the crate is to be marked 0 (zero) plutonium and the crate is to be stenciled LSA for normal shipment.

- b. If the gamma radiation is greater than 0.5 mr/hr an attempt is made to locate the source of the radiation. If possible, the source of the radiation is to be removed from the crate and placed into a 55 gallon drum. If the gamma radiation remains greater than 0.5 mr/hr the label on the crate is to be marked 0 (zero) plutonium and the actual gamma reading is to be stenciled on the crate. Waste Disposal Coordination will attach a yellow Group III label to the crate. The crate will be shipped by ATMX car.

You will be provided with additional information when the decontamination of the area north of between column lines F & G and west of column line 12 is started.


J. B. Owen
Manager
Area Decontamination

JBO:bts

cc:

L. P. Ferris II
L. F. Grill
L. D. Hazelton
W. D. Kittinger
P. F. Kreizenbeck
E. A. Putzier

CRATED FIRE WASTE SHIPPED DURING OCTOBER, 1969

Shipped 10-10-69

776 - 1 (A000968)
 - 5 (A000972)
 - 11 (A001072)
 - 14 (A001036)
 - 19 (A001246)
 - 20 (A001247)
 - 21 (A001307)
 - 28 (A001455)
 - 36 (A001651)
 - 48 (A001905)
 - 61 (A002052)
 - 65 (A002060)
 - 71 (A002080)
 - 72 (A002081)
 - 87 (A002212)
 - 97 (A002383)
 -109 (A002340)
 -141 (A002632)
 -175 (A003101)
 777 - 22 (A005741)
 - 23 (A005757)
 - 33 (A005838)
 - 36 (A005919)
 - 55 (A006334)
 - 56 (A006335)
 - 57 (A006336)
 - 63 (A006367)
 - 64 (A006374)
 - 66 (A006378)
 - 83 (A006474)
 -101 (A006566)
 -102 (A006572)
 -103 (A006590)
 -104 (A006591)
 -116 (A006673)
 -121 (A006720)

Shipped 10-21-69

776 - 8 (A001069)
 - 9 (A001070)
 - 10 (A001071)
 - 22 (A001308)
 - 25 (A001387)
 - 26 (A001380)
 - 27 (A001454)
 - 29 (A001613)
 - 34 (A001648)
 - 66 (A001654)
 - 83 (A002163)
 - 84 (A002169)
 - 90 (A002235)
 - 95 (A002268)
 - 96 (A002278)
 - 98 (A002284)
 -100 (A002298)
 -101 (A002299)
 -108 (A002339)
 -111 (A002343)
 -112 (A002349)
 -113 (A002350)
 -125 (A002400)
 -140 (A002482)
 -145 (A002730)
 -148 (A002733)
 -150 (A002738)
 -168 (A003019)
 -209 (A003266)
 777 - 51 (A006278)
 - 79 (A006452)
 - 86 (A006477)
 - 91 (A006510)
 - 93 (A006512)
 -120 (A006704)
 -137 (A006945)

Shipped 10-28-69

776 - 31 (A001631)
 - 33 (A001618)
 - 58 (A002027)
 - 73 (A002032)
 - 82 (A002162)
 - 85 (A002192)
 - 93 (A002244)
 - 94 (A002247)
 - 99 (A002218)
 -103 (A002329)
 -104 (A002330)
 -107 (A002338)
 -132 (A002430)
 -133 (A002431)
 -144 (A002640)
 -154 (A002784)
 -155 (A002794)
 -166 (A002996)
 -182 (A003142)
 -190 (A003167)
 -199 (A003211)
 777 - 39 (A006053)
 - 41 (A006115)
 - 45 (A006230)
 - 48 (A006263)
 - 52 (A006299)
 - 53 (A006312)
 - 59 (A006345)
 - 61 (A006355)
 - 73 (A006393)
 - 76 (A006416)
 - 77 (A006428)
 - 95 (A006549)
 - 98 (A006553)
 -125 (A006766)
 -131 (A006831)

Distribution:

L.F. Grill

L.D. Hazelton

~~L.B.~~ Owen NOV 17 1969

CRATED FIRE WASTES SHIPPED DECEMBER 1969

Shipped 12 3 69*

A000707
A005371
A005395

776 16 (A001139)
776 17 (A001140)
777 26 (A005764)
777 27 (A005786)
777 28 (A005804)
777 29 (A005805)
777 30 (A005828)
777 31 (A005816)

Shipped 12 10-69*

A000699

Shipped 12 16 69

A000880

776 18 (A001245)
776 37 (A001660)
776 45 (A001690)
776 53 (A001978)
776 60 (A002036)
776 79 (A002143)
776 81 (A002161)
776 88 (A002215)
776 92 (A002237)
776 106 (A002337)
776 114 (A002351)
776 117 (A002373)
776 121 (A002391)
776 124 (A002395)
776 127 (A002412)
776 142 (A002633)
776 149 (A002737)
776 152 (A002754)
776 160 (A002837)
776 165 (A002880)
776 167 (A002995)
776 172 (A003064)
776 179 (A003018)
776 180 (A003140)
776 181 (A003141)

777 40 (A006096)
777 43 (A006170)
777 65 (A006377)
777 72 (A006402)
777 80 (A006465)
777 99 (A006554)
777 100 (A006565)
777 114 (A006654)
777 127 (A006783)
777 135 (A006920)

Copies

L F Gr II

L D Haz It

~~L B Owen~~

CRATED FIRE WASTES SHIPPED

January, 1970

January 9, 1970

776- 23 (A001315)
 776- 30 (A001619)
 776- 33 (A001667)
 776- 42 (A001684)
 776- 43 (A001685)
 776- 46 (A001691)
 776- 50 (A001920)
 776- 51 (A001659)
 776- 52 (A001664)
 776-105 (A002333)
 776-110 (A002341)
 776-113 (A002374)
 776-120 (A002370)
 776-137 (A002462)
 776-171 (A003060)
 776-176 (A003111)
 776-186 (A003147)
 777- 11 (A005619)
 777- 33 (A005933)
 777- 42 (A006143)
 777- 44 (A006169)
 777- 53 (A006339)
 777- 69 (A006388)
 777- 75 (A006415)
 777- 78 (A006434)
 777- 84 (A006475)
 777-107 (A006597)
 777-122 (A006721)
 777-123 (A006747)
 777-126 (A006782)
 777-128 (A006784)
 777-129 (A006790)
 777-139 (A007003)
 777-141 (A007018)
 779- 5
 779- 6

January 16, 1970

776- 41 (A001673)
 776- 54 (A002017)
 776- 55 (A002018)
 776- 56 (A002019)
 776- 62 (A002053)
 776- 67 (A002064)
 776-68-2(A002065)
 776- 69 (A002066)
 776- 74 (A002083)
 776- 75 (A002120)
 776- 76 (A002123)
 776- 78 (A002131)
 776- 80 (A002156)
 776- 91 (A002336)
 776-126 (A002411)
 776-128 (A002413)
 776-139 (A002481)
 776-143 (A002639)
 776-170 (A003032)
 776-177 (A003112)
 776-188 (A003156)
 777- 85 (A006476)
 777- 87 (A006505)
 777- 88 (A006506)
 777- 90 (A006509)
 777- 92 (A006511)
 777- 94 (A006548)
 777- 96 (A006551)
 777-117 (A006679)
 777-124 (A006748)
 777-130 (A006818)
 777-133 (A006891)
 777-138 (A006990)
 777-140 (A007016)
 777-142 (A007017)
 779- 7

January 21, 1970

776- 24 (A001386)
 776- 40 (A001672)
 776- 44 (A001686)
 776- 47 (A001904)
 776-68-1(A002051)
 776- 70 (A002072)
 776- 77 (A002127)
 776-119 (A002375)
 776-122 (A002392)
 776-123 (A002393)
 776-130 (A002416)
 776-134 (A002432)
 776-135 (A002433)
 776-169 (A003020)
 776-184 (A003144)
 776-185 (A003145)
 776-187 (A003148)
 776-214 (A003329)
 776-227 (A003455)
 777- 46 (A006240)
 777- 47 (A006247)
 777- 49 (A006276)
 777- 70 (A006389)
 777- 82 (A006473)
 777- 89 (A006508)
 777-109 (A006614)
 777-110 (A006618)
 777-115 (A006655)
 777-118 (A006680)
 777-119 (A006703)
 777-121 (A006719)
 777-132 (A006836)
 777-134 (A006908)
 777-136 (A006831)
 779- 4
 779- 8

January 23, 1970

776-129 (A002415)
 776-136 (A002445)
 776-147 (A002732)
 776-153 (A002764)
 776-156 (A002810)
 776-157 (A002813)
 776-158 (A002814)
 776-159 (A002819)
 776-161 (A002862)
 776-162 (A002872)
 776-163 (A002934)
 776-203 (A003234)
 776-204 (A003246)
 776-208 (A003257)
 776-211 (A003284)
 776-212 (A003303)
 776-217 (A003351)
 776-218 (A003352)
 776-221 (A003385)
 776-222 (A003392)
 776-225 (A003411)
 776-231-1 (A003463)
 776-231-2 (A003465)
 776-232 (A003469)
 777- 50 (A006277)
 777- 54 (A006301)
 777- 60 (A006343)
 777- 62 (A006358)
 777- 68 (A006380)
 777- 71 (A006401)
 777- 97 (A006552)
 777-105 (A006592)
 777-108 (A006612)
 777-111 (A006626)
 777-112 (A006627)
 777-113 (A006631)

L.F. Grill
 J.S. Owen
 L.D. Hazelton

2/20

CRATED FIRE WASTES SHIPPED -- FEBRUARY, 1970

February 5, 1970

776-348 (A004351)
 776-434 (A011124)
 A000359
 A000360
 A000381
 A000382
 A000383
 A005140

February 11, 1970

776-151 (A002744)
 776-173 (A003095)
 776-174 (A003096)
 776-183 (A003143)
 776-210 (A003132)
 776-215 (A003332)
 776-228 (A003460)
 776-233 (A003476)
 776-330 (A004241)
 776-337 (A004305)
 776-338 (A004316)
 776-345 (A004340)
 776-349 (A004352)
 776-352 (A004375)
 776-354 (A004377)
 776-355 (A004378)
 776-356 (A004398)
 776-380 (A004856)
 776-381 (A004857)
 776-386 (A004950)
 776-388 (A004999)
 776-389 (A010008)
 776-390 (A010048)
 776-392 (A010266)
 776-393 (A010267)
 776-395 (A010351)
 776-402 (A010493)
 776-404 (A010538)
 776-405 (A010550)
 776-406 (A010617)

777-311 (A007774)
 777-312 (A007775)
 777-319 (A007839)
 777-321 (A007860)

February 18, 1970

776-198 (A003210)
 776-201 (A003218)
 776-205 (A003247)
 776-213 (A003306)
 776-216 (A003350)
 776-219 (A003370)
 776-220 (A003377)
 776-226 (A003452)
 776-230 (A003465)
 776-235 (A004035)
 776-262 (A003643)
 776-323 (A004192)
 776-324 (A004193)
 776-482 (A003963)

 777-152 (A007106)
 777-156 (A007143)
 777-157 (A007149)
 777-158 (A007150)
 777-159 (A007166)
 777-160 (A007191)
 777-161 (A007197)
 777-164 (A007295)
 777-165 (A007309)
 777-166 (A007310)
 777-167 (A007311)
 777-168 (A007315)
 777-169 (A007316)
 777-170 (A007317)
 777-171 (A007320)
 777-172 (A007321)
 777-177 (A007340)
 777-178 (A007341)
 777-183 (A007387)
 777-190 (A007394)
 777-249 (A007592)
 777-259 (A007609)
 777-266 (A007633)
 777-271 (A007644)
 777-272 (A007645)
 777-273 (A007646)

February 25, 1970

776-223 (A003395)
 776-234 (A004003)
 776-239 (A004071)
 776-245 (A004101)
 776-246 (A004102)
 776-247 (A003532)
 776-249 (A003541)
 776-251 (A003555)
 776-252 (A003572)
 776-283 (A003784)
 776-298 (A003902)
 776-299 (A003903)
 776-303 (A003932)
 776-310 (A003974)
 776-311 (A003975)
 776-315 (A004116)
 776-317 (A004134)
 776-321 (A004160)
 776-391 (A010057)
 776-394 (A010268)
 776-399 (A010392)

 777-184 (A007388)
 777-185 (A007389)
 777-188 (A007392)
 777-209 (A007477)
 777-211 (A007484)
 777-233 (A007534)
 777-239 (A007548)
 777-241 (A007550)
 777-246 (A007577)
 777-248 (A007591)
 777-250 (A007593)
 777-255 (A007598)
 777-256 (A007599)
 777-264 (A007621)
 777-265 (A007622)
 777-267 (A007634)
 777-268 (A007635)
 777-275 (A007648)
 777-316 (A007799)

Copies:

L.F. Grill
 L.D. Hazelton
~~J.B. Owen~~
 E.S. Ryan

RECEIVED

MAR 19 1970

CRATED FIRE WASTES SHIPPED TO IDAHO FALLS, IDAHO -- MARCH, 1970

March 5, 1970	March 13, 1970	March 18, 1970	
776-255 (A006505)	776-178 (A003115)	776- 59 (A002234)	777-144 (A007328)
776-254 (A006550)	776-193 (A003177)	776-164 (A002985)	777-154 (A007115)
776-265 (A006551)	776-224 (A003404)	776-194 (A003185)	777-182 (A007336)
776-284 (A007866)	776-240 (A004077)	776-236 (A004022)	777-194 (A007420)
776-289 (A006907)	776-271 (A006996)	776-238 (A004051)	777-198 (A007499)
776-294 (A006934)	776-273 (A006714)	776-241 (A004076)	777-206 (A007471)
776-296 (A006891)	776-275 (A003729)	776-242 (A004082)	777-207 (A007472)
776-307 (A006963)	776-276 (A003732)	776-253 (A003524)	777-210 (A007480)
776-308 (A006964)	776-281 (A003759)	776-256 (A003606)	777-212 (A007493)
776-312 (A006979)	776-292 (A003825)	776-261 (A003640)	777-214 (A007498)
776-320 (A004161)	776-306 (A003954)	776-267 (A003659)	777-218 (A007512)
776-326 (A004208)	776-329 (A004233)	776-274 (A003725)	777-222 (A007516)
776-342 (A004367)	776-335 (A004257)	776-279 (A003742)	777-225 (A007519)
776-350 (A004365)	776-343 (A004338)	776-280 (A003758)	777-226 (A007520)
776-374 (A004731)	776-351 (A004367)	776-285 (A003794)	777-227 (A007521)
776-375 (A004751)	776-365 (A004639)	776-287 (A003796)	777-230 (A007531)
776-379 (A004655)	776-377 (A004839)	776-288 (A003797)	777-240 (A007543)
776-431 (A004721)	776-385 (A004943)	776-291 (A006318)	777-252 (A007510)
776- (A004935)	776-397 (A004952)	776-295 (A003892)	777-262 (A007513)
777-143 (A007019)	776-397 (A010335)	776-292 (A004163)	777-263 (A007613)
777-147 (A007066)	776-401 (A010444)	776-295 (A004196)	777-270 (A007641)
777-150 (A007056)	776-403 (A010535)	776-297 (A004206)	777-274 (A007647)
777-153 (A007239)	777-162 (A007249)	776-323 (A004207)	777-279 (A007669)
777-175 (A007338)	777-174 (A007335)	776-331 (A004243)	777-281 (A007671)
777-181 (A007334)	777-180 (A007381)	776-333 (A004248)	777-283 (A007684)
777-186 (A007390)	777-192 (A007412)	776-339 (A004317)	777-286 (A007697)
777-191 (A007405)	777-205 (A007453)	776-341 (A004335)	777-289 (A007701)
777-217 (A007504)	777-223 (A007517)	776-346 (A004341)	777-295 (A007710)
777-223 (A007522)	777-224 (A007516)	776-360 (A004422)	777-299 (A007739)
777-229 (A007526)	777-235 (A007544)	776-361 (A004533)	777-301 (A007747)
777-231 (A007532)	777-280 (A007670)	776-363 (A004556)	777-302 (A007748)
777-232 (A007533)	777-282 (A007673)	776-366 (A004640)	777-305 (A007759)
777-233 (A007547)	777-291 (A007706)	776-368 (A004658)	777-306 (A007760)
777-243 (A007569)	777-300 (A007741)	776-369 (A004666)	777-310 (A007770)
777-244 (A007570)	777-315 (A007797)	776-373 (A004730)	
777-245 (A007576)	777-319 (A007839)	776-384 (A004928)	
777-257 (A007605)		776-396 (A010384)	
777-268 (A007700)		776-398 (A010386)	
		776-400 (A010413)	

March 25, 1970

776-189 (A003166)
 776-200 (A003217)
 776-244 (A004093)
 776-248 (A003533)
 776-250 (A003544)
 776-270 (A003591)
 776-282 (A003777)
 776-302 (A003924)
 776-304 (A003937)
 776-318 (A004143)
 776-347 (A004346)
 776-357 (A004389)
 776-358 (A004390)
 776-359 (A004407)
 776-357 (A004657)
 776-370 (A004717)
 776-372 (A004729)
 776-382 (A004859)
 776-383 (A004883)
 776- (A003993)
 777-149 (A007051)
 777-175 (A007322)
 777-195 (A007424)
 777-197 (A007429)
 777-199 (A007431)
 777-202 (A007441)
 777-211 (A007514)
 777-236 (A007545)
 777-251 (A007594)
 777-253 (A007595)
 777-260 (A007610)
 777-273 (A007662)
 777-284 (A007695)
 777-292 (A007709)
 777-293 (A007734)
 777-304 (A007758)
 777-308 (A007771)

March 30, 1970

776-116 (A002372)
 776-191 (A003168)
 776-192 (A003176)
 776-207 (A003256)
 776-243 (A004083)
 776-254 (A003591)
 776-263 (A003645)
 776-266 (A003652)
 776-268 (A003668)
 776-272 (A003697)
 776-286 (A003795)
 776-297 (A003892)
 776-301 (A003918)
 776-305 (A003940)
 776-336 (A004258)
 776-340 (A004319)
 776-344 (A004339)
 776-353 (A004376)
 776-376 (A004824)
 776-378 (A004854)
 777-145 (A007027)
 777-151 (A007172)
 777-153 (A007107)
 777-187 (A007391)
 777-213 (A007494)
 777-215 (A007499)
 777-219 (A007513)
 777-242 (A007554)
 777-247 (A007586)
 777-261 (A007615)
 777-277 (A007653)
 777-290 (A007703)
 777-294 (A007711)
 777-295 (A007714)
 777-296 (A007715)
 777-297 (A007715)
 777-303 (A007749)
 777-307 (A007769)

Copies:

L.F. Grill
 L.D. Hazelton
~~L.S. Owen~~
 E.S. Ryan

CRATED FIRE WASTES SHIPPED TO IDAHO FALLS, IDAHO -- APRIL, 1970

Shipped 4- 8-70

776-197 (A003195)
 776-198 (A003210)
 776-202 (A003222)
 776-257 (A003608)
 776-258 (A003627)
 776-259 (A003630)
 776-260 (A003638)
 776-269 (A003683)
 776-277 (A003737)
 776-278 (A003738)
 776-290 (A003817)
 776-293-2 (A003833)
 776-293-1 (A003826)
 776-300 (A003906)
 776-314 (A004118)
 776-316 (A004117)
 776-319 (A004156)
 776-362 (A004546)
 776-364 (A004638)
 776-407 (A010678)
 776-409 (A010717)
 776-414 (A010921)
 776-415 (A010922)
 776-418 (A010923)
 776-421 (A010966)
 776-422 (A010997)
 776-423 (A011023)
 776-425 (A011046)
 776-426 (A011047)
 776-428 (A011073)
 776-431 (A011105)
 776-432 (A011108)
 776-435 (A011125)
 776-438 (A011170)
 776-442 (A011223)
 776-444 (A011266)
 776-445 (A011267)
 776-446 (A011275)
 776-450 (A011310)
 776-452 (A011357)
 776-455 (A011360)
 776-456 (A011383)
 776-457 (A011384)
 776-459 (A011396)
 776-460 (A011412)
 776-A003602
 776-A010965

777-148 (A007048)
 777-155 (A007119)
 777-176 (A007339)
 777-179 (A007366)
 777-189 (A007393)
 777-193 (A007419)
 777-196 (A007428)
 777-200 (A007438)
 777-201 (A007440)
 777-203 (A007445)
 777-204 (A007452)
 777-208 (A007474)
 777-216 (A007500)
 777-221 (A007515)
 777-234 (A007528)
 777-237 (A007546)
 777-254 (A007597)
 777-258 (A007603)
 777-269 (A007638)
 777-276 (A007652)
 777-285 (A007696)
 777-287-1 (A007698)
 777-287-2 (A007699)
 777-309 (A007772)
 777-322 (A007888)

Shipped 4-15-70

776-408 (A010688)
 776-410 (A010782)
 776-411 (A010783)
 776-412 (A010864)
 776-413 (A010920)
 776-416 (A010929)
 776-417 (A010930)
 776-419 (A010947)
 776-424 (A011037)
 776-429 (A011095)
 776-430 (A011096)
 776-433 (A011120)
 776-436 (A011127)
 776-437 (A011128)
 776-440 (A011201)
 776-441 (A011222)
 776-448 (A011298)
 776-449 (A011309)
 776-454 (A011359)
 776-461 (A011421)
 776-462 (A011442)
 776-465 (A011446)
 776-467 (A011476)
 776-474 (A011522)
 776-475 (A011523)
 776-480 (A011627)
 776-485 (A011690)
 776-487 (A011727)
 776-489 (A011851)
 776-492 (A011896)
 776-493 (A011897)
 776-509 (A012176)

777-317 (A007805)
 777-318 (A007820)
 777-320 (A007849)
 777-325 (A008018)
 777-326 (A008057)
 777-327 (A008058)
 777-328 (A008077)
 777-329 (A008078)
 777-330 (A008085)
 777-331 (A008099)
 777-332 (A008118)
 777-333 (A008119)
 777-335 (A008132)
 777-336 (A008146)
 777-337 (A008170)
 777-338 (A008173)
 777-339 (A008174)
 777-340 (A008181)
 777-341 (A008182)
 777-342 (A008191)
 777-343 (A008192)
 777-344 (A008195)
 777-345 (A008196)
 777-346 (A008197)
 777-347 (A008211)
 777-348 (A008213)
 777-349 (A008216)
 777-351 (A008225)
 777-352 (A008228)
 777-354 (A008233)
 777-355 (A008234)
 777-360 (A008266)
 777-361 (A008351)
 777-364 (A008390)
 777-366 (A008402)
 777-369 (A008468)
 777-371 (A008473)
 777-373 (A008500)
 777-374 (A008514)
 777-375 (A008518)
 777-378 (A008575)

**Distribution: L.F. Grill
 L.D. Hazelton
 J.B. Owen
 E.S. Ryan**

Shipped 4-23-70

776-453 (A011358) 777-324 (A007953)
 776-464 (A011445) 777-350 (A008224)
 776-466 (A011447) 777-353 (A008232)
 776-468 (A011477) 777-356 (A008235)
 776-469 (A011495) 777-357 (A008247)
 776-470 (A011498) 777-362 (A008369)
 776-471 (A011502) 777-363 (A008370)
 776-472 (A011506) 777-365 (A008400)
 776-473 (A011511) 777-367 (A008436)
 776-476 (A011524) 777-368 (A008450)
 776-477 (A011525) 777-370 (A008498)
 776-480 (A011628) 777-372 (A008499)
 776-481 (A011629) 777-376 (A008519)
 776-483 (A011688) 777-377 (A008560)
 776-484 (A011689)
 776-486 (A011726)
 776-488 (A011805)
 776-491 (A011895)
 776-494 (A011905)
 776-495 (A011957)
 776-496 (A011978)
 776-497 (A011993)
 776-498 (A011994)
 776-499 (A012005)
 776-500 (A012076)
 776-501 (A012077)
 776-502 (A012107)
 776-503 (A012108)
 776-504 (A012109)
 776-505 (A012110)
 776-506 (A012113)
 776-507 (A012157)
 776-508 (A012158)
 776-510 (A012196)
 776-511 (A012216)
 776-522 (A012268)
 776-529 (A012372)
 776-533 (A012443)
 776-534 (A012444)
 776-535 (A012481)
 776-536 (A012483)
 776-538 (A012525)
 776-541 (A012640)
 776-543 (A012642)
 776-545 (A012732)
 776-551 (A012819)
 776-561 (A013266)
 776-5000322

Shipped 4-30-70

776-332 (A004246)
 776-334 (A004251)
 776-427 (A011070)
 776-443 (A011265)
 776-458 (A011389)
 776-478 (A011537)
 776-490 (A011894)
 776-512 (Filters)
 776-513 (Filters)
 776-514 (Filters)
 776-515 (Filters)
 776-516 (Filters)
 776-518 (Filters)
 776-519 (Filters)
 776-520 (Filters)
 776-521 (Filters)
 776-540 (A012561)
 776-553 (A012939)
 776-555 (A012996)
 776-563 (A013418)
 776-564 (A013426)
 777-379 (A008576)
 777-382 (A008616)
 777-386 (A008655)
 777-389 (A008659)
 777-390 (A008633)
 777-393 (A003670)
 777-394 (A003671)
 777-401 (A008733)
 777-488 (A008840)
 777-410 (A008926)

CRATED FIRE WASTES SHIPPED TO IDAHO FALLS, IDAHO -- MAY, 1970

May 7, 1970

776-528 (A012955)
776-539 (A012956)
776-552 (A012958)
776-554 (A012959)
776-557 (A013056)
776-559 (A013211)
776-562 (A013374)

777-313 (A007775)
777-314 (A007779)
777-393 (A008617)
777-384 (A003636)
777-387 (A008656)
777-398 (A008657)
777-391 (A008664)
777-397 (A008702)
777-398 (A008711)
777-399 (A003721)
777-405 (A008768)
777-409 (A008924)
777-411 (A003966)

May 14, 1970

776-500 (A012670)
776-507 (A012642)
776-542 (A012641)
776-545 (A013598)
776-566 (A013594)
776-567 (A013605)
776-568 (A012372)
776-569 (A013652)
776-570 (A013653)
776-571 (A013654)
776-572 (A013655)
776-573 (A013656)
776-575 (A013670)
776-576 (A013671)
776-577 (A013672)
776-578 (A013673)
776-579 (A013674)
776-580 (A013683)
776-581 (A013689)
776-582 (A013690)

May 20, 1970

777-402 (A003751)
777-403 (A003752)

May 26, 1970

776-498 (A003777)
776-495 (A003107)
776-468 (A011444)
776-524 (A012933)
776-525 (A012937)
776-526 (A012316)
776-527 (A012343)
776-531 (A012374)
776-549 (A012734)
776-550 (A012735)
776-556 (A013355)
776-558 (A013339)
776-593 (A013694)
776-584 (A013695)
776-585 (A013696)
776-586 (A013747)
776-587 (A013751)
776-588 (A013752)
776-589 (A013753)
776-590 (A013754)
776-591 (A013827)
776-592 (A013851)
776-593 (A013863)
776-594 (A013867)
776-595 (A013871)
776-604 (A013935)
776-605 (A013936)
776-606 (A013972)
776-608 (A014041)
776-614 (A014119)

777-392 (A003557)
777-395 (A003672)
777-396 (A003673)
777-400 (A003727)
777-404 (A003761)
777-406 (A003784)
777-407 (A003798)
777-412 (A003967)
777-413 (A009018)
777-415 (A009035)
777-416 (A009036)
777-417 (A009057)

ADDITION TO FEBRUARY, 1970 LIST:

February 12, 1970

776-196 (A003192)

CRATED FIRE WASTES SHIPPED TO ARCO, IDAHO

June, 1970

June 16, 1970

776-206 (A003251)
 776-237 (A004034)
 776-544 (A012712)
 776-548 (A012783)
 776-560 (A013265)
 776-574 (A013657)
 776-596 (A013908)
 776-597 (A013927)
 776-598 (A013928)
 776-599 (A013929)
 776-600 (A013930)
 776-601 (A013931)
 776-602 (A013932)
 776-609 (A014042)
 776-610 (A014043)
 776-611 (A014044)
 776-612 (A014045)
 776-617 (A014153)
 776-618 (A014156)
 776-619 (A014157)
 776-620 (A014158)
 776-621 (A014159)
 776-622 (A014160)
 776-623 (A014167)
 776-624 (A014168)
 776-625 (A014169)
 776-627 (A014196)
 776-628 (A014197)
 776-632 (A014227)
 776-633 (A014234)

777-146 (A007035)
 777-359 (A008256)
 777-380 (A008599)
 777-414 (A005030)
 777-418 (A009090)

June 23, 1970

776-630 (A014199)
 776-631 (A014200)
 777-419 (A009122)
 777-421 (A009149)
 777-422 (A009150)
 777-423 (A009170)
 777-424 (A009173)
 777-425 (A009186)
 777-427 (A009268)
 777-428 (A009269)
 777-429 (A009270)
 777-430 (A009271)
 777-431 (A009307)
 777-432 (A009322)
 777-433 (A009323)
 777-434 (A009416)
 777-435 (A009461)
 777-437 (A009513)
 777-439 (A009527)
 777-440 (A009543)
 777-442 (A009588)
 777-443 (A009613)

June 25, 1970

776-309 (A003973)
 776-479 (A011572)
 776-528 (A012344)
 776-532 (A012409)
 776-546 (A012751)
 776-547 (A012752)
 776-603 (A013933)
 776-626 (A014195)
 776-640 (A014367)
 776-642 (A014369)
 776-643 (A014370)
 776-644 (A014371)
 776-646 (A014404)
 776-647 (A014405)
 776-650 (A014449)
 776-653 (A014517)
 776-654 (A014518)
 776-660 (A014570)
 776-664 (A014575)
 776-665 (A014584)
 777-381 (A008612)
 777-436 (A009490)
 777-438 (A009523)

cc:

L.D. Hazelton

~~J.B. Owen~~

E.S. Ryan

CRATED FIRE WASTES SHIPPED -- AUGUST, 1970

August 31, 1970

776-634 (A014235)
776-635 (A014275)
776-638 (A014365)
776-639 (A014366)
776-652 (A014451)
776-655 (A014522)
776-657 (A014524)
776-658 (A014555)
776-661 (A014571)
776-667 (A014586)
776-668 (A014590)
776-670 (A014622)
776-672 (A014668)
776-673 (A014673)
776-674 (A014674)
776-675 (A014684)
776-676 (A014709)
776-677 (A014717)
776-678 (A014727)

777-447 (A009701)
777-448 (A009708)
777-450 (A009723)
777-452 (A009744)
777-453 (A009771)
777-454 (A009803)
777-455 (A009802)
777-457 (A009839)
777-459 (A009857)
777-461 (A009885)
777-462 (A009922)

cc:

L.D. Hazelton

 J.B. Owen

ESR:db 9-2-70

CRATED FIRE WASTE SHIPPED DURING OCTOBER, 1970

October 9, 1970

776-659 (A014556)
776-680 Filters
776-682 Filters
776-686 Filters
776-687 Filters
776-696 (A014821)
776-699 (A014832)
776-700 (A014833)
776-701 (A014834)
776-703 (A014838)
776-704 (A014848)
776-705 (A014849)
776-707 (A014870)
776-708 (A014871)
776-709 (A014894)
776-711 (A014911)
776-712 (A014925)
776-715 (A014929)
776-716 (A014932)
776-717 (A014958)

777-445 (A009651)
777-463 (A009923)
777-465 (A009933)
777-466 (A009934)
777-467 (A009962)
777-468 (A009971)
777-469 (A009972)
777-473 (A019010)
777-474 (A019032)
777-475 (A019033)
777-476 (A019034)
777-477 (A019035)
777-478 (A019036)
777-479 (A019062)
777-480 (A019071)
777-481 (A019072)

October 21, 1970

776-697 (A014823)
776-698 (A014824)
776-702 (A014835)
776-706 (A014850)
776-713 (A014927)
776-714 (A014928)
776-718 (A014978)
776-719 (A014979)
776-720 (A014994)
776-721 (A014995)
776-722 (A015029)
776-723 (A015030)
776-724 (A015042)
776-725 (A015043)
776-726 (A015049)
776-727 (A015069)
776-728 (A015070)
776-729 (A015087)
776-731 (A015112)
776-732 (A015124)
776-733 (A015130)
776-734 (A015131)
776-735 (A015143)
776-737 (A015145)
776-741 (A015198)
776-742 (A015199)
776-743 (A015224)
776-744 (A015241)
776-745 (A015276)

777-470 (A009995)
777-471 (A019003)
777-472 (A019009)

cc:
L.D. Hazelton
J.B. Owen

CRATED FIRE WASTES SHIPPED TO IDAHO FALLS, IDAHO

November, 1970

November 4, 1970

776-607 (A014023)
776-613 (A014113)
776-629 (A014198)
776-636 (A014289)
776-648 (A014424)
776-666 (A014585)
776-679 (A014756)
776-730 (A015088)
776-738 (A015169)

777-464 (A009932)
777-489 (A019116)
777-494 (A019169)
777-499 (A019200)

November 12, 1970

776-694 (A014799)
776-739 (A015192)
776-740 (A015197)
776-752 (A015343)
776-763 (A015615)
776-765 (A015638)
776-769 (A015686)
776-770 (A015692)
776-771 (A015708)

777-483 (A019077)
777-485 (A019096)
777-486 (A019097)
777-487 (A019098)
777-488 (A019115)
777-490 (A019128)
777-491 (A019131)
777-492 (A019152)
777-504 (A019252)
777-505 (A019253)
777-506 (A019254)
777-507 (A019255)
777-508 (A019264)
777-511 (A019280)
777-512 (A019281)
777-513 (A019285)
777-515 (A019293)
777-516 (A019299)
777-517 (A019300)
777-518 (A019310)
777-519 (A019311)
777-524 (A019335)
777-525 (A019338)

L.D. Hazelton

L.B. Owen

CRATED FIRE WASTE SHIPPED TO IDAHO FALLS, IDAHO

December, 1970

December 15, 1970

776-615 (A014144)
776-616 (A014146)
776-753 (A015355)
776-754 (A015411)

777-495 (A019170)
777-496 (A019182)
777-497 (A019192)
777-529 (A019353)
777-530 (A019361)
777-531 (A019363)
777-547 (A019507)
777-548 (A019508)

December 22, 1970

776-695 (A014808)
776-746 (A015294)
776-748 (A015301)
776-756 (A015442)
776-759 (A015504)
776-760 (A015505)
776-761 (A015539)
776-764 (A015624)
776-766 (A015643)
776-767 (A015669)
776-785 (A015899)
776-804 (A015983)

777-493 (A019153)
777-503 (A019224)
777-510 (A019279)
777-527 (A019351)
777-528 (A019352)
777-535 (A019388)
777-536 (A019389)
777-537 (A019397)
777-538 (A019403)
777-539 (A019404)
777-540 (A019410)
777-541 (A019450)
777-543 (A019459)
777-544 (A019474)
777-552 (A019565)
777-562 (A019663)
777-563 (A019668)
777-564 (A019676)
777-565 (A019677)

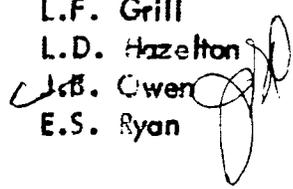
Distribution

L.F. Grill

L.D. Hazelton

J.B. Owen

E.S. Ryan



CRATED FIRE WASTE SHIPPED TO IDAHO FALLS, IDAHO

January, 1971

January 19, 1971

776-762 (A015591)
776-772 (A015709)
776-773 (A015725)
776-774 (A015747)
776-775 (A015802)
776-776 (A015821)

777-542 (A015591)
777-551 (A019558)
777-553 (A019580)
777-555 (A019614)
777-567 (A019638)

January 28, 1971

776-777 (A015839)
776-778 (A015863)
776-779 (A015870)
776-780 (A015871)
776-781 (A015874)
776-782 (A015886)
776-783 (A015888)
776-784 (A015889)
776-786 (A015900)
776-789 (A015916)
776-790 (A015926)
776-791 (A015927)
776-792 (A015934)
776-793 (A015935)
776-794 (A015952)
776-795 (A015955)
776-796 (A015956)
776-797 (A015967)
776-798 (A015975)
776-799 (A015976)
776-805 (A016008)
776-806 (A016009)
776-808 (A016022)
776-809 (A016025)

Distribution:

L.F. Grill
L.D. Hazelton
J.B. Owen
E.S. Ryan

Feb. 3, 1971

CRATED FIRE WASTE SHIPPED

February, 1971

February 22, 1971

776-692 (filters)
776-693 (filters)
776-694 (filters)
776-695 (filters)
776-800 (A015970)
776-802 (A015981)
776-820 (A016055)
776-821 (A016067)
776-822 (A016068)
776-823 (A016069)
776-824 (A016070)
776-830 (A016086)
776-837 (A016105)
776-854 (A016233)

777-558 (A019639)
777-559 (A019647)
777-572 (A019726)

February 23, 1971

776-781 (A015914)
776-811 (A016027)
776-812 (A016028)
776-813 (A016029)
776-815 (A016040)
776-833 (A016098)
776-834 (A016099)
776-835 (A016103)
776-839 (A016107)

777-566 (A019691)
777-568 (A019704)
777-576 (A019779)
777-580 (A019803)
777-582 (A019809)
777-586 (A019821)
777-589 (A019836)
777-595 (A019879)
777-600 (A019890)
777-601 (A019912)
777-609 (A019954)

L.F. Grill
L.D. Hazelton
✓ J.B. Owen
E.S. Ryan

3- 5-71

CRATED FIRE WASTE SHIPPED

March, 1971

March 3, 1971

776-816 (A016041)
776-817 (A016042)
776-818 (A016043)
776-819 (A016044)
776-828 (A016074)
776-831 (A016095)
776-832 (A016097)

777-561 (A019662)
777-567 (A019703)
777-577 (A019785)
777-584 (A019813)
777-590 (A019837)
777-591 (A019863)
777-592 (A019869)
777-593 (A019877)
777-596 (A019883)
777-605 (A019947)
777-606 (A019951)
777-610 (A019955)
777-611 (A019963)
777-615 (A019980)

March 31, 1971

776-801 (A015980)
776-803 (A015982)
776-807 (A016010)
776-826 (A016072)
776-827 (A016073)
776-855 (A016234)
776-857 (A016236)
776-871 (A016358)

777-587 (A019834)
777-588 (A019835)
777-608 (A019953)
777-636 (A018088)
777-637 (A018089)
777-638 (A018100)
777-639 (A018103)
777-646 (A018132)
777-648 (A018134)
777-656 (A018194)
777-678 (A018288)
777-681 (A018300)
777-685 (A018316)
777-687 (A018318)

L. F. Grill
L. D. Hazelton
✓ J. B. Owen
E. S. Ryan

April 12, 1971

Recor File

CRATED FIRE WASTE SHIPPED - APRIL, 1971

April 2, 1971

776-836 (A016104)
848 (A016207)
850 (A016214)
858 (A016237)
860 (A016279)
864 (A016292)
865 (A016319)
868 (A016342)
872 (A016359)
875 (A016381)
893 (A016457)
900 (A016481)
901 (A016482)
929 (A016547)
933 (A016554)
937 (A016561)
938 (A016562)

777-554 (A019601)
556 (A019625)
602 (A019915)
604 (A019940)
621 (A018015)
626 (A018035)
631 (A018054)
652 (A018168)
657 (A018206)
663 (A018227)
668 (A018250)
669 (A018258)
688 (A018319)
690 (A018334)
694 (A018346)
695 (A018358)
696 (A018359)
698 (A018361)

April 6, 1971

776-859 (A016265)
867 (A016329)
898 (A016474)
914 (A016522)
918 (A016526)
919 (A016527)
922 (A016530)
925 (A016536)
927 (A016545)

April 6, 1971

777-616 (A019981)
617 (A019982)
632 (A018055)
640 (A018112)
649 (A018144)
682 (A018301)
684 (A018308)

April 8, 1971

776- 870 (A016357)
877 (A016386)
896 (A016460)
931 (A016552)
942 (A016569)
950 (A016577)
961 (A016607)
971 (A016634)
979 (A016644)
990 (A016659)
1001 (A016681)
1002 (A016682)
1004 (A016684)
1007 (A016687)

777-619 (A018003)
654 (A018176)

April 16, 1971

776- 788 (A015915)
810 (A016026)
825 (A016071)
846 (A016192)
849 (A016213)
852 (A016231)
853 (A016232)
863 (A016291)
897 (A016461)
904 (A016486)
912 (A016509)
932 (A016553)
960 (A016606)
964 (A016615)
966 (A016617)
1012 (A016697)
1020 (A016708)

April 16, 1971

777-560 (A019661)
571 (A019716)
573 (A019735)
574 (A019762)
575 (A019763)
578 (A019795)
579 (A019796)
583 (A019801)
585 (A019820)
594 (A019878)
597 (A019886)
599 (A019888)
614 (A019979)
625 (A018034)
627 (A018036)
679 (A018291)
683 (A018305)
699 (A018376)

April 21, 1971

776-838 (A016106)
851 (A016215)
880 (A016396)
881 (A016397)
882 (A016398)
886 (A016418)
889 (A016439)
892 (A016445)
906 (A016498)
907 (A016499)
909 (A016501)
911 (A016508)
965 (A016616)

777-629 (A018038)
642 (A018114)
644 (A018116)
645 (A018131)
650 (A018145)
651 (A018150)
667 (A018249)
670 (A018263)
671 (A016264)
672 (A018265)
673 (A018268)
674 (A018269)
675 (A018270)
680 (A018293)

April 29, 1971

776- 814 (A016039)
895 (A016459)
903 (A016485)
934 (A016558)
935 (A016559)
936 (A016560)
939 (A016563)
940 (A016564)
941 (A016565)
943 (A016570)
967 (A016618)
975 (A016638)
977 (A016642)
978 (A016643)
981 (A016646)
985 (A016650)
987 (A016652)
989 (A016658)
1003 (A016683)

777-570 (A019714)
665 (A018240)

NOTE: On March, 1971,
list delete 777-656 (A018194)
and add 777-686 (A018317).

cc:
L. F. Grill
L. D. Hazelton
✓ J. B. Owen
E. S. Ryan

CRATED FIRE WASTE SHIPPED - MAY, 1971

May 3, 1970

776- 866 (A016328)
 876 (A016385)
 878 (A016387)
 885 (A016413)
 887 (A016437)
 888 (A016438)
 891 (A016444)
 899 (A016480)
 905 (A016487)
 928 (A016546)
 962 (A016608)
 973 (A016636)
 986 (A016651)
 996 (A016676)
 997 (A016677)
 1000 (A016680)
 1008 (A016688)
 777- 620 (A018014)
 635 (A018085)
 653 (A018174)
 655 (A018190)
 664 (A018239)
 677 (A018276)
 700 (A018377)

May 20, 1971

776- 842 (A016149)
 844 (A016171)
 847 (A016200)
 902 (A016484)
 908 (A016500)
 910 (A016502)
 913 (A016521)
 915 (A016523)
 917 (A016525)
 921 (A016529)
 926 (A016544)
 930 (A016548)
 949 (A016576)
 972 (A016635)
 976 (A016641)
 980 (A016645)
 988 (A016657)
 992 (A016661)
 998 (A016678)

May 20, 1971

776-1006 (A016686)
 1009 (A016694)
 1011 (A016696)
 1014 (A016699)
 1015 (A016700)
 1016 (A016701)
 1017 (A016705)
 1019 (A016707)
 1025 (A016716)
 1026 (A016717)
 1032 (A016723)
 1033 (A016728)
 1034 (A016729)
 1038 (A016733)
 1039 (A016734)
 1040 (A016735)
 1041 (A016736)
 1042 (A016737)
 1043 (A016738)
 1044 (A016739)
 1045 (A016740)
 1055 (A016753)
 1056 (A016754)
 1062 (A016760)
 1078 (A016779)
 1081 (A016788)
 777- 628 (A018037)
 641 (A018113)
 647 (A018133)
 658 (A018207)
 659 (A018208)
 691 (A018341)
 701 (A018378)

May 25, 1971

776- 840 (A016136)
 841 (A016148)
 856 (A016235)
 862 (A016290)
 879 (A016395)
 883 (A016399)
 884 (A016406)
 890 (A016440)
 923 (A016533)
 924 (A016534)

May 25, 1971

776- 958 (A016604)
959 (A016605)
982 (A016647)
983 (A016648)
993 (A016662)
1010 (A016695)
1021 (A016709)
777- 603 (A019939)
643 (A018115)
656 (A018194)
660 (A018214)
666 (A018245)
676 (A018275)

cc:

L. F. Grill
L. D. Hazelton
✓ J. B. Owen
E. S. Ryan

August 24, 1971

Building 776 Cont.

776-1119 (A016829)
776-1123 (A016833)
776-1129 (A016839)
776-1133 (A016843)
776-1137 (A016847)
776-1140 (A016850)
776-1153 (A016863)
776-1155 (A016865)
776-1159 (A016869)
776-1170 (A016830)
776-1178 (A016889)
776-1182 (A016903)
776-1190 (A031312)
776-1193 (A031430)
776-1197 (A030487)

Building 777

777-744 (A018458)
777-751 (A031546)
777-761 (A031582)
777-774 (A031660)

August 27, 1971

Building 776

776-994 (A016663)
776-1028 (A016719)
776-1029 (A016720)
776-1053 (A016751)
776-1054 (A016752)
776-1063 (A016761)
776-1064 (A016762)

August 27, 1971

Building 776 Cont.

776-1094 (A016801)
776-1096 (A016803)
776-1116 (A016826)
776-1147 (A016857)
776-1168 (A016878)

Building 777

777-630 (A018046)
777-633 (A018064)
777-737 (A018451)
777-738 (A018452)
777-748 (A031536)

cc:

L. F. Grill

L. D. Hazelton

✓ J. B. Owen

E. S. Ryan

CRATED FIRE WASTE SHIPPED - AUGUST, 1971

August 3, 1971

Building 776

776-874 (A016372)
776-991 (A016660)
776-1027 (A016718)
776-1030 (A016721)
776-1036 (A016731)
776-1051 (A016749)
776-1059 (A016757)
776-1061 (A016759)
776-1080 (A016781)
776-1082 (A016789)
776-1084 (A016791)
776-1086 (A016793)
776-1088 (A016795)
776-1089 (A016796)
776-1122 (A016832)
776-1126 (A016836)

Building 777

777-693 (A018343)

August 6, 1971

Building 776

776-843 (A016168)
776-845 (A016191)
776-920 (A016528)
776-946 (A016573)
776-974 (A016637)
776-1006 (A016685)
776-1018 (A016706)
776-1023 (A016713)
776-1046 (A016741)

August 6, 1971

Building 776 Cont.

776-1047 (A016743)
776-1052 (A016750)
776-1057 (A016755)
776-1058 (A016756)
776-1110 (A016820)

Building 777

777-624 (A018033)
777-662 (A018224)

August 18, 1971

Building 776

776-894 (A016458)
776-1024 (A016715)
776-1120 (A016830)
776-1134 (A016844)
776-1144 (A016854)
776-1145 (A016855)
776-1148 (A016858)
776-1167 (A016877)
776-1174 (A016884)
776-1184 (A031088)

Building 777

777-770 (A031632)

August 24, 1971

Building 776

776-1093 (A016800)
776-1102 (A016809)
776-1117 (A016827)

F 1
JM

CRATED FIRE WASTES SHIPPED - FEBRUARY, 1972

February 21, 1972

15-728 (A018436)

15-743 (A018457) X

15-752 (A031553) X

15-754 (A031560)

15-762 (A031582) X

15-1013 (A016698)

15-1180 (A016890)

15-1204 (A032154)

cc:
✓ B. A. Bowman
L. F. Grill
J. B. Owen
E. S. Ryan



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 188
GOLDEN, COLORADO 80401

June 2, 1969

E. S. Ryan ✓

DRUM COUNTED 776 FIRE WASTE FOR BURIAL

Please be informed that 628 drums from the 776 fire cleanup have been drum counted to date. All 628 drums were found to be less than the AEC discard limit. The drum numbers range from A00001 through A00630, excluding A00361 and A00366 that were voided. Of the 628 measured, 617 contain zero grams Pu. The 11 drums that contain quantities of Pu are as follows:

<u>Drum Count No.</u>	<u>Gms Pu</u>
A00002	1
A00017	1
A00080	2
A00158	1
A00226	2
A00269	1
A00305	6
A00334	1
A00414	1
A00515	1
A00527	2
TOTAL	<u>19</u>

All 628 drums may be shipped off plant site to the burial grounds.

R. D. Mullet

RDM:ejw

APPROVED:

L. F. Grill
MANAGER CHEMICAL OPERATIONS

cc:

L. F. Grill
L. D. Hazelton
H. E. Bowman

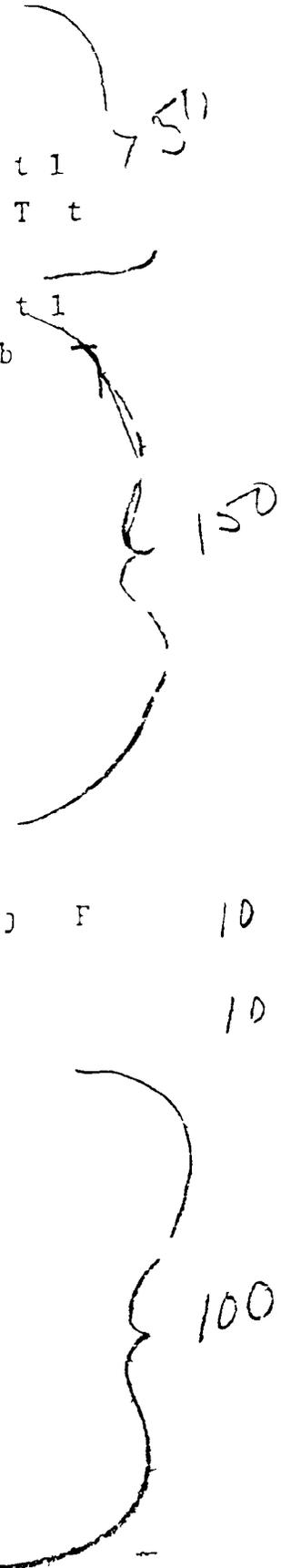
* ~~Director~~ in field (444) List is from Waste Mgt Fld East of

Verification of WSC in 54676
By R Taylor

Created

R Taylor 12-1-7

	B	I	A	N	S	P	S	C	t
1	771	41			48	81	7		
2	77	612	019	6	63	52	78		
	777	5 1	A019808		5	1	6	E1	
4	770	692	016742		4	7	83	H t t l	
5		514	PDP 1 6984		9	87	75	C t T t	
6	777	718	A019990		6	77	51	D y l	
7	776	692	A01674		45	78	83	H t t l	
8	777	613	A01 978		87	49	57	Gl b	
9	771	69			84	60	48		
10	771	7			84	60	48		
11	777	687	PD 777 1485		48	57	87		
12	7 7	607	A019952		55	8	87		
13	777	598	A019 7		56	51	84		
14	771	404			42	72	53		
15	7 1	403			42	2	53		
16	771	02			42	72	53		
17	771	87			42	7	53		
18	771	01			42	2	53		
19	776	1143			75	59	56	D y	
20	7 6	951	A016 78		64	50	75	B 11 j F	10
21	77	752	031553		49	62	83	M t l	
22	551	10	N P		90	50	94		10
23	771	52			66	20	65		
24	76	3 4			53	18	77	B 1	
25	776	995	A016675		65	60	61		
26	771	21			48	60	4		
27	771	68			3	4	110		
28	771	90			84	49	72		
29	776	1125	A016835		55	43	66		
30	77	716	A018124		63	63	54		
31	771	389			72	48	84		
32	777	54			49	81	56		



	<u>P</u>					<u>D</u>	<u>I</u>	
3	771	0		4	60			
34		441		3	11	4	Sh	ld g g 150
5	71	405		48	84			26
36	776	417	A011276	190	11	64	M	h L th 100
37		957	A016 5	64	6	75	D y l	
8	776	945	A016 2	66	57	75	M t l	20
39	7	6074	A01 457	49	4	84	M t l	
40	776	94	A016 75	64	5		M	ll
41	777	703	A018 0	87	8	5	D y b	
42	776	44	016 1	5	56	66	B ll j F	50
43	776	956	016585	75	61	56		5
44	776	952	A016 79	75	56	64	B ll j F	50
45	551	166		96		63	Gl l	10
46	776	968	A016 19	4	65	53	D y l	
47	777	622	A018016	8	49	56	Gl b	20
48	771	391		75	9	57		
49	777	5	A00 2	35	51	49	v	L I D t t 10
50	771	414		48	7	84	H d B l	
51	551	16	A	8	1	60		
52	551	168	A P	96	56	60		
53	776	769	A0316 1	48	72	84		100
54	776	953	A016 82	65	75	56		
55	88	8716 14	PD 55	63	52	9		
56	777	762	A0 1 83	48	66	84	M t l	
57	777	782	A0 16 0	51	51	6		
58	776	51	A0113 5	184	116	63		100
59	77	1606		6	5	98		
60	776	768	A01 671	71	6	93		
61	7	9 014		9	68	42	W ll	
62	771	357		48	48	96		
63	889	178		87	56	15		
64	776	1	A011200	16	112	6	M	ll th 100

2000

	<u>F</u>	<u>AO I 31</u>	<u>S</u>	<u>D</u>
65	77 9		4 84	
66	7 1 58		0 44 100	
67	771 91		48 57 79	
68	711 21		48 60 84	
69	716 571		66 94 122	
70	776 6 5		4 105 57	B q g P 50
71			64 93 155	
72	70 07		51 78 124	U
73	776 762		71 88 189	
74	20 5 61		67 73 79	
75	776 3		75 11 0	B B (1)
76	771 771		51 8 87	B d 11
77	777 765		58 73 103	
78	176 1192		72 87 189	
79	776 7 2		71 87 188	
80	776 513		74 112 111	
81	776 525 ✓		72 114 189	
82	A 0		64 4 172	
83	100 ✓		71 76 189	
84	207 21		44 0 117	E t F d C y
85			63 76 116	
86	770 6 5		64 49 114	
87	777 75		84 86 130	
88	776 6 7 ✓		71 104	
89	71 72		60 86 141	
90	717 705		51 8 112	
91	207 168		49 12 118	
92				

250

[Handwritten signature] 2/24/70

February 24, 1970

C. A. Noble

LIMITATION ON SIZE OF WOODEN CRATES FOR WASTE SHIPMENTS

We have a problem with three wooden crates containing Monarch lathes from Building 776. The crates will not fit into ATMX cars. Ed Ryan's measurements indicate dimensions as follows:

15' 7" x 5' 3" x 9' 4"
15' 9" x 5' 3" x 9' 4"
15' 3" x 5' 3" x 9' 4"

Since these crates contain gram quantities of plutonium we cannot ship them in trailers.

You will recall my letter on September 6, 1968 which placed a limitation of 14' 10" x 8' x 8' on crates for ATMX cars. This assumed use of three bays per car. Two of our cars are altered for taking wooden crates and for proper use of dunnage and cables, the dimensions should now be held to less than 14' 2" by 8' 4" wide by 8' 9" high.

On January 14, 1969, I amended the September 6 letter to allow dimensions up to 22' by 8' by 8'. This assumed placing crates in the cars altered for cargo crates which have two bays. While we can accommodate the larger size, we prefer, if at all possible, to stay with the smaller since some alterations may be necessary in using the 2-bay cars for wooden crates.

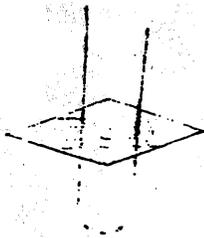
[Handwritten signature]
E. A. Puzler
Health Physics

EAP:ab

cc: *[Handwritten signature]*
J.B. Owen
E.S. Ryan

Appendix FF

Measuring Plutonium on Equipment and Machine Tools Removed from the Fire Area



THE DOW CHEMICAL COMPANY

ROCKY PLATE DIVISION
P. O. BOX 333
GOLDEN, COLORADO 80401

June 10, 1969

H. E. Bowman	A. T. Schutten
L. F. Grill	E. J. Walko
A. R. Konecny	J. F. Willging
✓ J. B. Owen	E. R. Young
C. H. Partington	

MEASURING PLUTONIUM ON EQUIPMENT AND MACHINE TOOLS REMOVED FROM THE FIRE AREA

Attached is a copy of a procedure for determining the Plutonium on equipment removed from the fire area that has been accepted by Albuquerque Operations Office (Reference attached acceptance letter).

This procedure must be followed to measure the Plutonium content on machine tools and equipment (includes construction materials) removed from the immediate fire area. If the material cannot be measured by the approved procedure, it must be broken or cut into pieces and drummed. Drums will be assigned a fire control number and transferred to Building 771 for a Plutonium determination in the gamma/neutron drum counter.

L. F. Grill and J. B. Owen have agreed to the use of gamma counting all materials (including construction materials) outside the immediate fire area to establish that no Plutonium is present. The fire area for this purpose is defined as follows:

That portion of Buildings 776/777 bordered on the south by column line D, on the west by column line 3, on the north by column line L and on the east by column line 25.

Materials disposed of from outside the immediate fire area containing no Plutonium will be boxed. The boxes will be moved by area decontamination crews in accordance with the procedure established by B. A. Bowman dated June 6, 1969. In addition, the label on each box noting "0" Plutonium will be signed as approved by either responsible supervision of the decontamination crew or a member of the Safeguards Committee.

If there are any questions concerning the procedures for determining Plutonium content on machine tools, equipment and construction materials, please contact L. F. Grill or J. B. Owen.


L. D. Hazelton
Accountability Representative

LDH/LFG:jb
Enclosures

cc:
C. H. Dompierre

DETERMINING PLUTONIUM ON EQUIPMENT REMOVED FROM THE FIRE AREA

There will be a large quantity of equipment and machine tools too large for measurement by drum counter taken out of the fire area in Buildings 776/777. The procedure for measuring the Plutonium remaining on this type material after extensive cleaning is as follows:

1. R&D personnel will calculate the quantity of Plutonium remaining on the equipment by the following techniques:
 - a. Using Kem Wipes saturated with KW solution smear one or more square feet of the surface. (Number of square feet in sample determined by size of equipment.)
 - b. Using Kem Wipes saturated with a OSPHO solution, smear the same square feet area wiped with the KW solution.
 - c. Combine the Kem Wipes from both smears and send to the laboratory for a total Plutonium analysis.
 - d. Calculate the total Plutonium remaining on the equipment by multiplying the total Plutonium content per square foot determined by laboratory analysis times the total number of square feet of surface.
2. Prepare a record listing type of equipment, number of square feet, Plutonium per square foot and total Plutonium. Send this record to the Manager of Chemical Operations (L. F. Grill) Building 771.
3. The Manager of Chemical Operations, the Accountability Representative (L. D. Hazelton) and a Senior Research Engineer (B. L. Kelchner) will review each item to determine whether additional cleaning would be economical. Their decision will be based upon past experience dealing with the following factors:
 - a. Cost of further cleaning using actual direct labor costs (hourly wage plus variable burden).

- b. Cost of cleaning materials.
 - c. Cost of recovering the Plutonium in cleaning residues.
 - d. Quantity of Plutonium expected to be removed by additional cleaning.
 - e. Estimated Plutonium recovery processing yield of 3.d quantity.
 - f. Value of Plutonium (current incremental value standard).
4. Those items on which further cleaning is determined uneconomical will be crated for disposal with the measured Plutonium content identified as the value to be reported as a Normal Operational Loss.
 5. The listings of material discarded will be the source documents for entry of the Normal Operational Loss data into the Accountability control records.

Approved:



L. F. Grill, Manager
Chemical Operations



L. D. Hazelton
Accountability Representative

Appendix GG
Ventilation Filter Reports Available

Ventilation Filter Reports Available

Date	Report/Speech Title	Authors	ID Number Report/Speech
2/24/61	Evaluation of Filter Flammability and Filter Bank Fire Detection Systems	P. D. Erickson, J. A Geer and F. J. Linck	RFP-222 Speech
3/17/58	Absolute Air Filters Flammability and Fire Control Studies	P. D. Erickson and F. J. Linck	RFP-97 Speech
8/31/81	Volume Reduction of Used High Efficiency Particulate Air (HEPA) Filters	O. J. Butterdahl	RFP-3132 Speech
8/76	Exhaust Filtration on Gloveboxes used for Aqueous Processing of Plutonium	R. W. Woodard, K. J. Grossaint and T. L. McFeters	RFP-2495 Speech
10/20/76	Analysis of HEPA Filters from Six Stages of Filtration – 771 Building Plenum	K. J. Grossaint	MSL-76-684 Lab. Report
No date available	Disposal of HEPA Filters by Fluid Bed Incineration	D. L. Ziegler and A. J. Johnson	RFP-2769 Speech
7/25/78	Flander's Binder (for HEPA Filters)	R. S. Cichorz	PSL-78-482 Lab. Report
8/86	Characterization of Spent HEPA Filters from Rocky Flats Plant	P. M. Arnold and J. L. Blakeslee	PSD-86-056 Lab. Report
Unknown	IDC Description of Absolute Drybox (HEPA) Filters	Unknown	IDC 335 Description

Appendix HH

Extension of Useful Life of Exhaust Air Filters after Ammonium Nitrate Loading

REFERENCE

UNCLASSIFIED

R00049URFP

~~RFP~~

RFP-00049
AECU-3417

PHYSICS

OK/NEH

UNITED STATES ATOMIC ENERGY COMMISSION

EXTENSION OF USEFUL LIFE OF EXHAUST
AIR FILTERS AFTER AMMONIUM NITRATE
LOADING. Report.

By

- R. P. Craig ✓
- P. D. Erickson ✓
- J. A. Geer ✓
- F. J. Linck, Jr. ✓

Descriptors

Air filters
 Filters
 Radioactive
 aerosols
 Particles

October, 1956

Rocky Flats Plant
 Dow Chemical Company
 Denver, Colorado

Dissolution

Steam

Humidity

Technical Information Service Extension, Oak Ridge, Tenn.

Glass

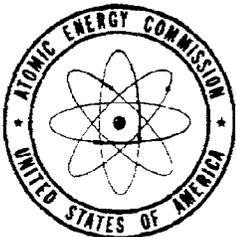
Cellulose

Asbestos

Ammonium

compounds

Nitrate



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AECU-3417

THE DOW CHEMICAL COMPANY

ROCKY FLATS PLANT

DENVER, COLORADO

U. S. ATOMIC ENERGY COMMISSION CONTRACT AT(29-1)-1106

EXTENSION OF USEFUL LIFE OF EXHAUST AIR FILTERS

AFTER AMMONIUM NITRATE LOADING

By

R. P. Craig
P. D. Erickson
J. A. Geer
F. J. Linck, Jr.

L. L. Zodtner - Section Superintendent

October, 1956

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ABSTRACT:

CWS-6 type air filters, designed to filter radioactive particles from exhaust air streams, were found to plug rapidly with ammonium nitrate under conditions of operation. Humidification of the filter bank with steam has been found to remove the ammonium nitrate from the filter pores and reduce the pressure drop across the filter to near its original value. Microscopic observation showed dissolution of the ammonium nitrate and subsequent recrystallization along the filter fibers, leaving the filter pores open to air passage. Useful life of these filters has been extended by occasional humidification to more than four times the service life initially indicated. Glass filters have not been found amenable to rejuvenation in similar manner.

ACKNOWLEDGMENTS:

The urgency of finding a means of extending the useful life of filters in service prompted the cooperative efforts of numerous individuals in solving this problem. F. R. Niles and D. M. Bassler were responsible for direction of early stages of this work. J. A. McGlone also assisted in accumulating and recording data.

INTRODUCTION:

Exhaust air from an AEC facility passes through a bank of CWS-6 type filters (Cambridge Absolute and Mine Safety Appliances type E6) before release to the atmosphere. This type filter is made of cellulose and asbestos, 0.035 to 0.045 inches in thickness, folded around pleated separators to give approximately 64 square feet of filtering area per square foot of filter face area. The filter medium is water repellent to the extent that a single ply will support a column of water not less than 20 inches high without immediate penetration.

The filters remove small quantities of radioactive particles carried by the exhaust air stream. Under conditions of plant operation, the filters were found to plug more readily than expected, raising the pressure drop across the filter bank to the maximum tolerable value within a few months. Since considerable expense is involved in replacing the filters in such a bank, a means of extending the useful life of these filters for even a few weeks would result in a cost saving of several thousand dollars.

Chemical analysis showed the material plugging the filters to be mostly ammonium nitrate. Laboratory investigations then were undertaken in an effort to find some means of extending the useful life of the filters by prevention of

ammonium nitrate plugging or by removal of the plugging material from the filter pores. These investigations brought out the possibility of removing the ammonium nitrate from the pores, and thereby reducing the pressure drop across the filters to near its original value, by a simple humidification procedure.

On the strength of the success of initial laboratory tests, the entire bank of filters in service was humidified immediately, since the need for reduction of pressure drop across the bank was immediate. This treatment was eminently successful.

Results of the first humidification of the filter bank, as well as the earlier laboratory investigations, dictated construction of a test duct in which single air filters could be subjected to repeated humidification and other treatments as desired to test the long range effects of such treatment on pressure drop and filter life. A filter in the test duct could be loaded with ammonium nitrate in a few hours to the extent that it would be loaded after several months service in the exhaust filter bank. The test duct, therefore, allowed observation of effects of repeated loading and humidification treatments within a short time, and made possible the prediction of similar effects on the exhaust filter bank over a relatively long period of service.

Observations of the effects of repeated humidification of filters in service are not yet complete, and additional single filter tests are continuing. However, considering the tremendous savings involved in extending filter life by such simple means, it is felt this work should be reported at this time so others might possibly benefit immediately from this information.

It should be noted that the work herein reported is merely a production trouble-shooting type of investigation, and was not meant to be an exhaustive study of the field.

EXPERIMENTAL EQUIPMENT:

Preliminary Experiments:

Initial laboratory tests were made with a small filter chamber about four inches square, in which samples of filter media could be clamped into position and be studied under controlled filtering conditions. The pressure drop across the sample was measured as air from various fume or smoke-producing trains was passed through the sample at a controlled rate.

Single Filter Test Duct:

The duct used in testing single complete filters is shown in Figures 1 and 2. The duct is approximately two feet square in cross section, and exhausts directly into the building exhaust air plenum. Since air is pulled through the duct by the building exhaust fans, control of air flow could be achieved only through pneumatically operated dampers at the duct exit. Air flows ranged from 30 to 130 feet per minute, and were measured with an Alnor thermo-anemometer. Anemometer readings were recorded continuously on a Foxboro recorder. A Sola constant voltage transformer held instrument voltages constant.

Humidity of inlet air was determined with wet and dry bulb thermometers, the temperatures being recorded on a Brown Portable recorder. These instruments are visible in the photograph, Figure 1.

Figure 2 shows the position of a test filter in the duct. Gasketing prevented air passage around the filter. Pressure drops across the filter were measured with manometers.

The main purpose of the test duct was to allow loading of a filter in a few hours to the extent it would be loaded in service after several months usage. Loading with ammonium nitrate was accomplished by passing ammonia gas over a tray of nitric acid in the duct entrance. These loading rates were not absolutely constant nor reproducible. Pressure drops across the filter were used as a measure of the degree of loading.

RESULTS AND DISCUSSION:

Analysis of the material plugging the filters in the building exhaust filter bank had shown the material to be mostly ammonium nitrate. Microscopic examination of plugged filter paper from the exhaust bank showed the ammonium nitrate to be present as clusters of beads, each about two microns in diameter. The beads appeared to penetrate only about ten percent of the filter paper thickness.

Laboratory investigations, during which both plugged filter media from the exhaust bank and media plugged under controlled conditions in the laboratory apparatus were studied, showed that exposure to a humid atmosphere, after plugging by either method, caused droplets of solution to coalesce on the surface of the filter medium. Microscopic observation during

humidification of a test sample showed dissolution of the ammonium nitrate upon humidification, followed by coalescence of solution droplets and subsequent recrystallization of the ammonium nitrate. Rapid drying of the droplets left large crystals of ammonium nitrate adhering to the filter surface. Longer contact of the droplets with the filter medium seemed to allow localized wetting of filter fibers, causing the ammonium nitrate to crystallize along the fibers upon subsequent drying. In either case, the pressure drop upon passage of air through the filter medium after humidification and recrystallization was practically as low as the drop across unused medium.

These effects are shown in photomicrographs of figures 3 through 8, all of which were taken under 30X magnification, of CWS-6 type filter paper. Figure 3 shows a new filter paper before exposure to ammonium nitrate. Figure 4 shows the same type paper after ammonium nitrate loading. Figure 5 shows large droplets of solution formed upon humidification of the loaded filter paper. It can be seen here that most of the salt has been dissolved from the filter pores. It appears that condensing water, wetting the ammonium nitrate surface, deposits in a thin film until the ammonium nitrate dissolves. The solution does not immediately wet the paper and asbestos surface beneath, and surface tension then causes the liquid film to condense into droplets. A single

large droplet of solution is shown in Figure 6. The difference in appearance of Figure 5 and Figure 6 arises, of course, from a difference in location of focal plane; in Figure 5, the surface of the filter paper is shown in focus; in Figure 6, focus is upon the droplet itself.

Figure 7 shows a typical heavily loaded paper after humidification and partial drying. Some surface droplets remain as clear solution; others have clouded and are seen in varying stages of crystal formation. Figure 8 shows a completely dried sample after such treatment. Comparison of Figure 8 with Figures 3 and 4 indicates the extent to which the filter pores have actually been freed of plugging material by this treatment. Indeed, the pressure drop across paper such as shown in Figure 8 was found to be very little different from paper such as shown in Figure 3.

The effects shown in these figures were achieved merely by exposing the paper samples to a humid atmosphere. After using passage of damp air to reduce the pressure drop across a filter sample in the experimental laboratory apparatus, it was confirmed that subsequent passage of clean dried air did not cause the pressure drop to rise again.

Observation of the above effects suggested the possibility of reducing the pressure drop across the bank of filters in service by adding steam to the exhaust air before its passage

through the filters. Even a slight reduction in pressure drop would have brought it back to a tolerable level, and promised to extend the useful life of the filters perhaps by several weeks. Initial efforts, adding steam to an air duct leading into the exhaust air plenum, met with some success. The improvement, however, was limited and was restricted to particular sections of the filter bank. The next approach was to add steam directly into the filter plenum with all exhaust fans temporarily shut off, operating individual fans at low speed for short intervals to pull the steam into each section of the filter bank. In this treatment, steam was added through a single 80-pound steam line, directly into the exhaust plenum, at a rate of approximately two-thirds of a pound of steam per hour per square foot of filter face area. Steam was applied for an hour and 20 minutes, and then allowed to stand in the plenum, with fans off, for an additional 40 minutes. When normal operation was resumed, it was found that this treatment had reduced the pressure drop across the filter bank to the original value of new filters.

The practicality of reducing pressure drop by humidification is hinged upon the assumption that the filters are not damaged mechanically and filtration efficiency is not reduced appreciably by such treatment. Checks on a single filter in the test duct indicated no damage to the filter medium upon prolonged contact with much larger quantities of

steam than are used in a normal humidification treatment. A qualitative check with a dye smoke showed no evidence of passage of any dye particles through the dried filter after such treatment. A more quantitative efficiency check was obtained by measurements of air-borne radioactivity on both sides of the filter bank after several humidifications, as indicated below.

An indication of the number of times a filter which had been plugged with ammonium nitrate could be rejuvenated repeatedly was obtained from experiments in the test duct. Figure 11 shows the pressure drops observed during repeated loading and humidification of a CWS-6 type filter. Seven complete cycles of loading and humidification were achieved successfully, with little indication of reduced capacity of the filter for subsequent use. In evaluating the data of Figure 11, loading times cannot be correlated exactly with amounts of salt collected on the filter because the loading rate was not constant. However, loading times can be used as a rough comparative indication of the amounts of salt collected.

Marked decreases in pressure drop during loading cycles, as shown in Figure 11 during the fifth and sixth loading, were real and could be correlated directly in each case with rainy weather. The humidity of the air passing through the building and, therefore, that passing through the test duct

during the loading cycles was dependent upon atmospheric humidity. Rainfall outside the building, therefore, effected a certain degree of natural humidification of the filters, with corresponding pressure drop reductions. Intentional humidifications shown in Figure 11 were accomplished by steaming the filter for eight to ten minutes each time.

Figure 12 shows the service history of the filters in the exhaust bank after the initial humidification described above. Before the initial humidification it appeared necessary to replace these filters after only twenty-two weeks service. As seen in Figure 12, the initial treatment extended the useful life of the filters an additional thirty-four weeks. Subsequent humidification treatments have extended the life of these filters, at the time of this writing, to a total of eighty weeks. The degree of humidification achieved was not the same during each humidification treatment, for the amount of steam added and conditions prevailing were governed by convenience at the time treatment became necessary. Humidification treatments numbered 5 and 6 were completed in 25 minutes each using a total of one-half pound of steam per square foot of filter face area each time.

Again in Figure 12 as in Figure 11, effects of periods of rainy weather are evident as apparently spontaneous decreases in the pressure drop.

Also shown in Figure 12 is a record of the amount of radioactive material passing through the filter bank. The fact that the air count behind the filters maintained its extremely low value is good evidence that the filtration efficiency of the filters was not diminished by the humidification treatments. The radioactivity measurements indicated that a filtration efficiency in excess of 99% was maintained. The filter media also had suffered no apparent loss of mechanical strength.

An experiment has been conducted on the filter bank to see if a water spray would reduce the pressure drop as effectively as steam humidification. A fine spray from a garden nozzle was passed over the filters three times in rapid succession. This treatment did bring about a reduction in pressure drop of 0.3 inch of water. However, the front edges of the filter medium appeared mechanically damaged by the spray, and the pressure drop returned to its original value within a days time. These results indicated that only a small percentage of the area of the filter medium, perhaps just the front edges, had been affected by the spray treatment.

Although several filters with CWS-6 type filter media have been checked in the test duct, no difference in behavior of the filters upon humidification and reloading has been observed. Filters of this type tested were the Cambridge Absolute model A-1000 and MSA Type E-6. MSA filters made

with Bolivian asbestos were tested, as well as those made with African asbestos. (A different adhesive had been used with the Bolivian asbestos.)

Non-combustible filters made of glass fibers also have been tested briefly in the test duct. Interest in this type filter arose from concern over possibilities of fire in the present filter bank, particularly in view of the presence of an increasing load of ammonium nitrate on the filters. Experience at other installations gave dramatic proof of fire possibilities. Tests showed, however, that glass filters could not be rejuvenated effectively by humidification in the same manner as were the CWS-6 type filters. The Cambridge Absolute Non-combustible filter with metal frame, aluminum separators, and glass fiber filter medium, and the MSA Ultra Air Space filter with wooden frames and glass fiber filter medium were tested. The behavior of a Cambridge Non-combustible filter, shown in Figure 13, was rather typical of this type. The pressure drop could be reduced by humidification after the filter was plugged with ammonium nitrate. However, the subsequent rise in pressure drop on resumption of salt loading was so rapid that rejuvenation by such means for continued use was entirely impractical. Comparison of Figures 11 and 13 points out strikingly the difference in behavior of these types of filters.

For purposes of comparison with photomicrographs of the CWS-6 type filter medium, Figures 9 and 10, which are photomicrographs of type 1106B glass fiber medium from an MSA Ultra Air Space filter, are shown also at 30X magnification. This sample received treatment identical to that of the CWS-6 sample shown in Figures 3 through 8. Figure 9 shows new glass fiber medium at the top beside a sample which had been loaded with ammonium nitrate. The ammonium nitrate mat can be seen, although somewhat indistinctly, beneath the outermost loose glass fibers (bottom). Figure 10 shows the loaded sample after thorough humidification and subsequent drying. Apparently, wetting of the glass surface and/or capillary effects have not allowed extensive agglomeration of solution into droplets as occurred with the paper and asbestos filter media. The rapid replugging of humidified glass media, as shown in Figure 13, indicated that only a small fraction of the filter pore area had actually been freed of its ammonium nitrate content.

CONCLUSIONS:

Simple steam humidification can be used to extend the service life of CWS-6 type filters to a remarkable degree under conditions of ammonium nitrate loading. In areas of dry climate, or plants in which low humidity is maintained intentionally, such treatment could result in great monetary

savings by reduction of exhaust filter replacement costs.

Glass filters of types currently available are not amenable to rejuvenation by humidification under similar conditions.

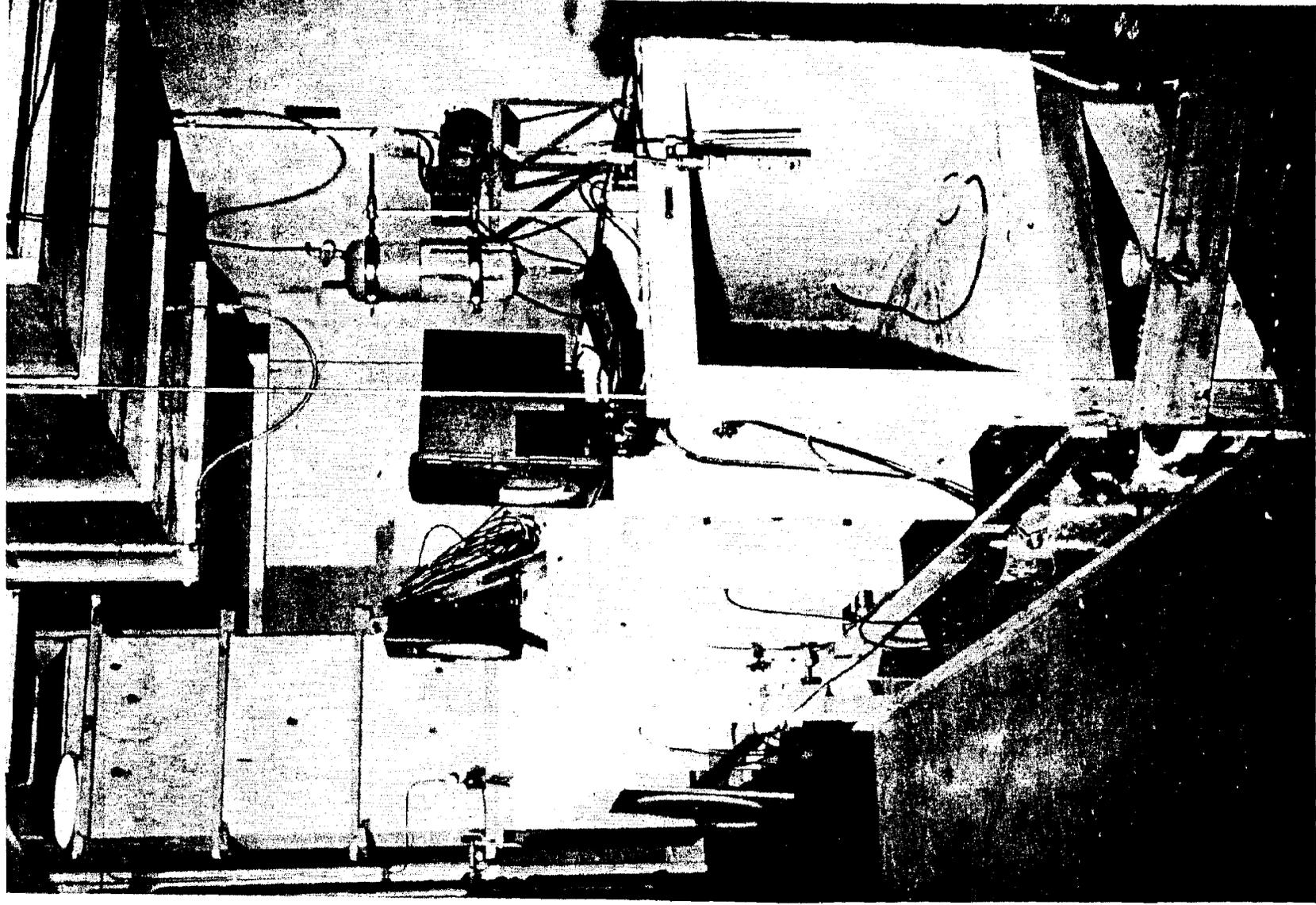


Figure 1
TEST DUCT

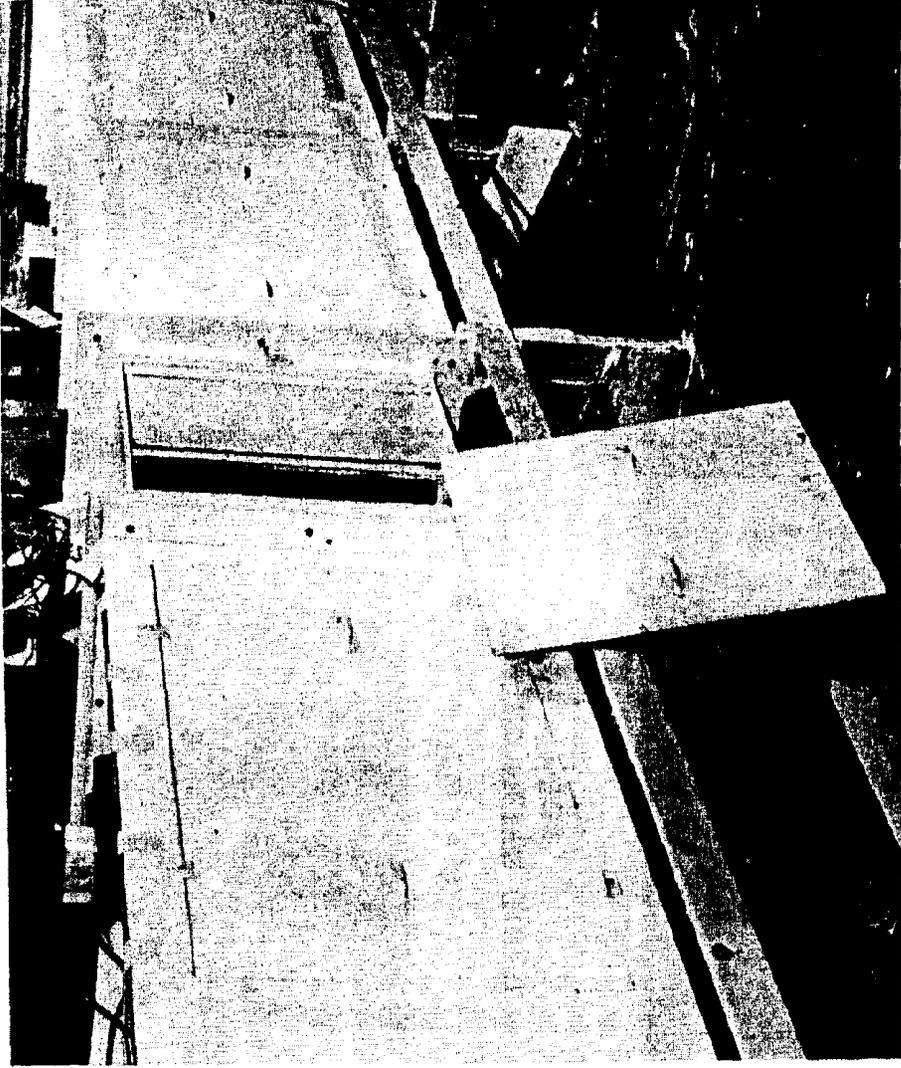


Figure 2
FILTER POSITION IN TEST DUCT

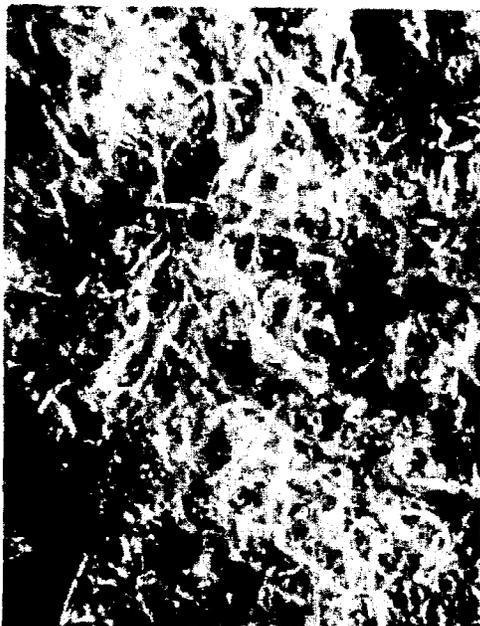


Figure 3
NEW FILTER PAPER
30X



Figure 4
COATED FILTER PAPER
30X

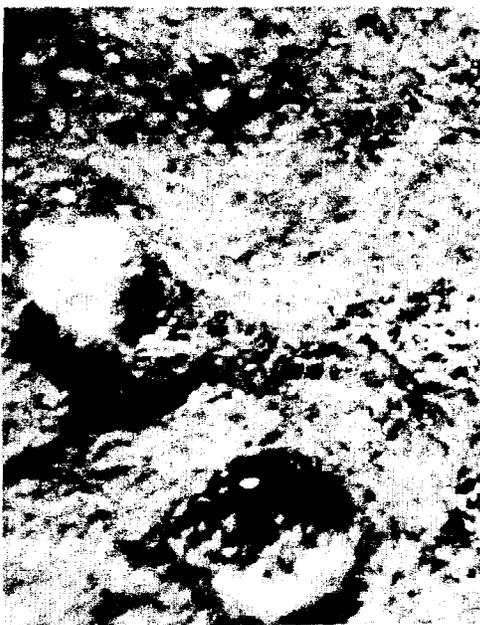


Figure 5
HUMIDIFIED FILTER PAPER
30X



Figure 6
HUMIDIFIED FILTER PAPER
30X



Figure 7
PARTIALLY DRIED
FILTER PAPER
30X

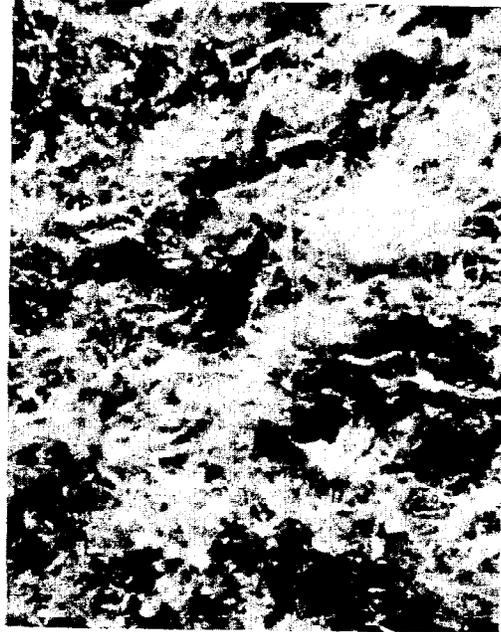


Figure 8
DRIED FILTER PAPER
30X



Figure 9
COATED (Lower) AND UNCOATED
(Upper) GLASS FILTER
30X



Figure 10
HUMIDIFIED AND DRIED
GLASS FILTER
30X

Figure II
EFFECT OF REPEATED NH_4NO_3 LOADING
AND STEAM HUMIDIFICATION
ON COMBUSTIBLE FILTER

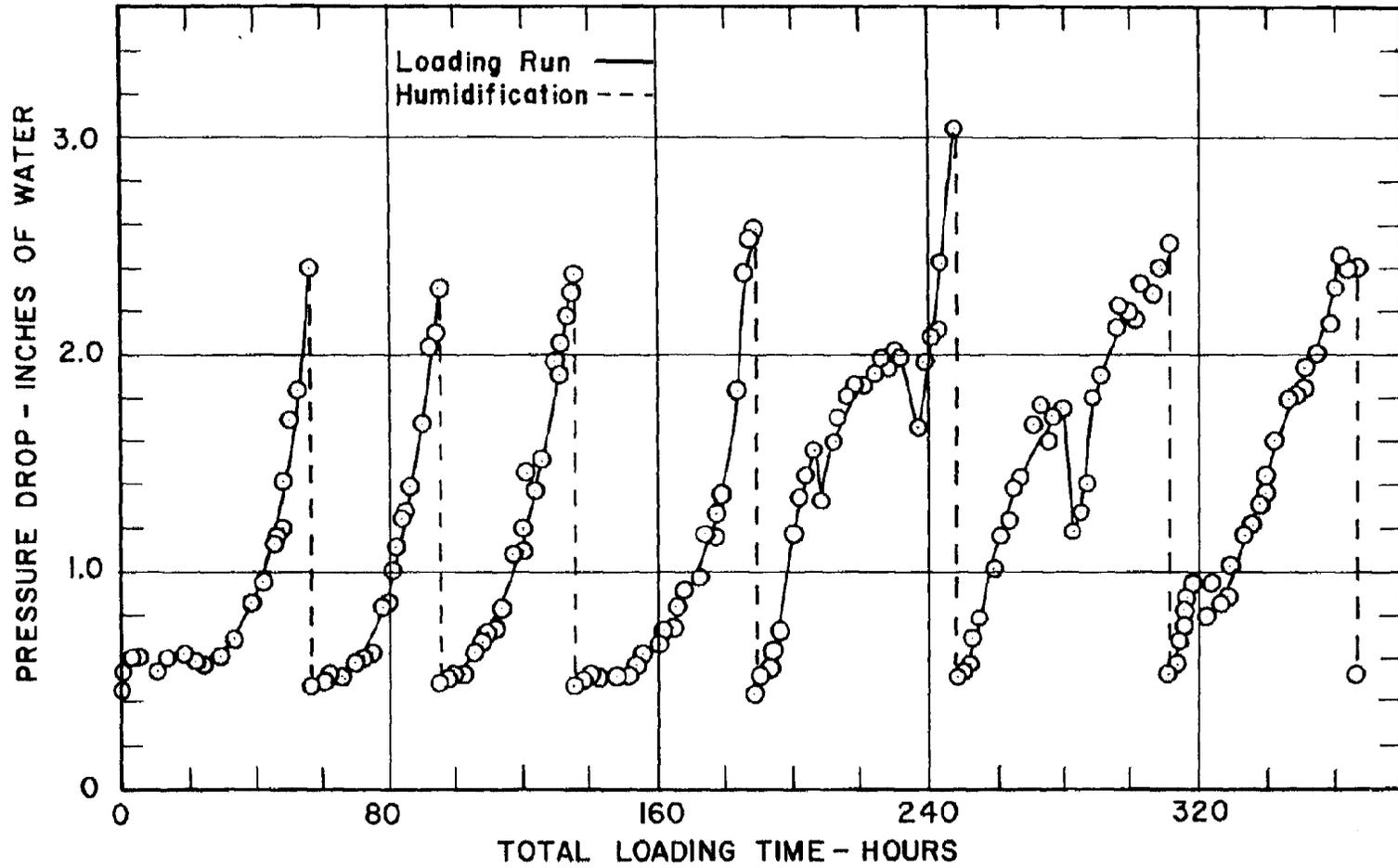


Figure 12

EFFECT OF REPEATED HUMIDIFICATIONS
ON FILTERS IN SERVICE

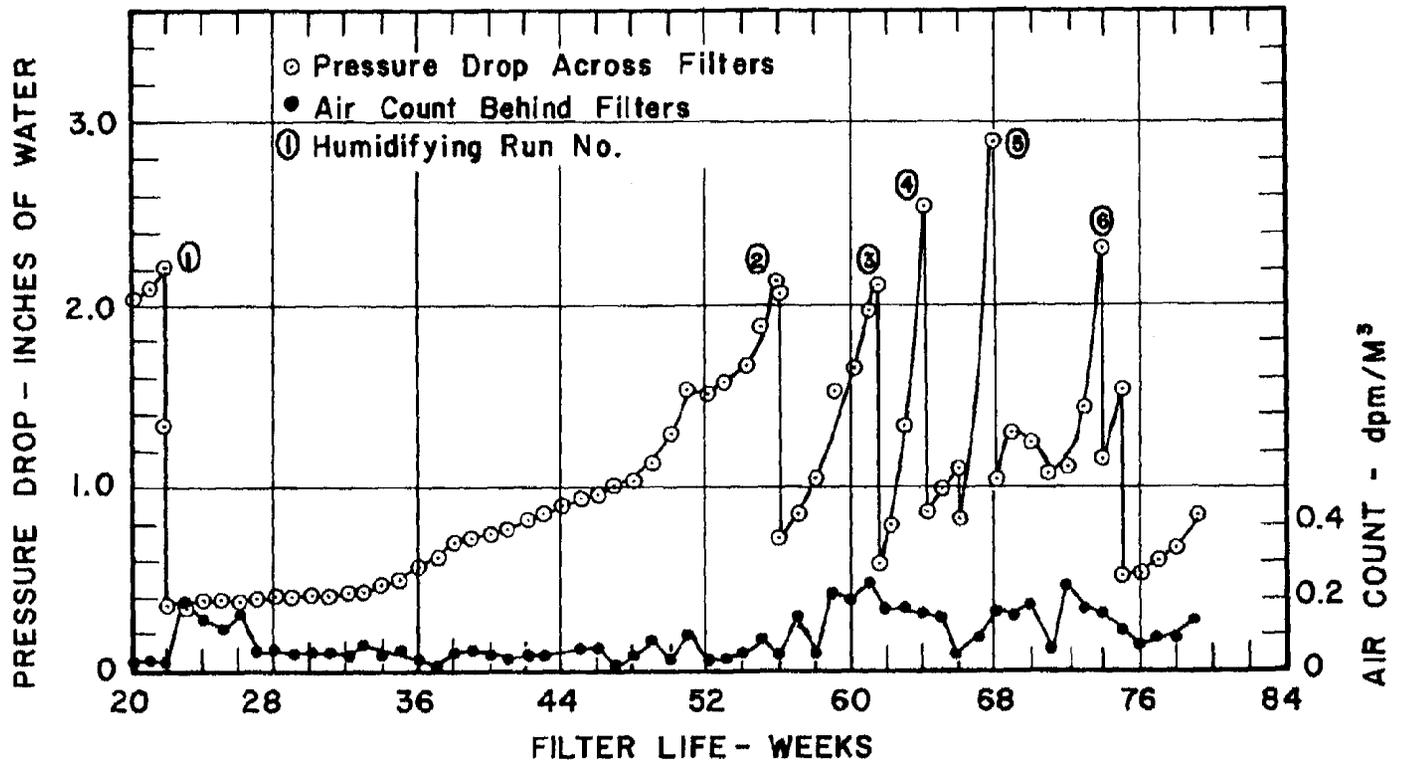
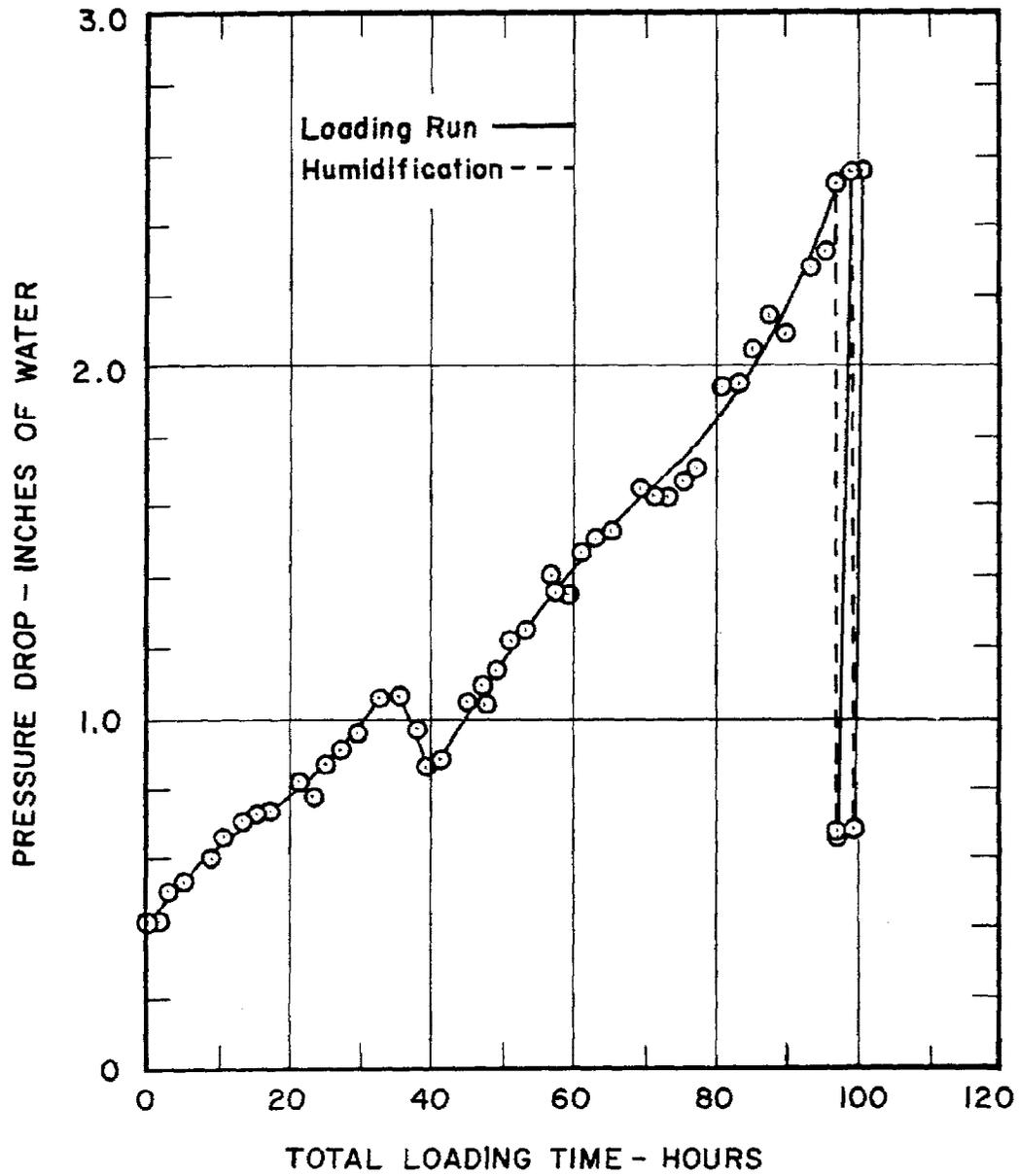


Figure 13
EFFECT OF REPEATED NH_4NO_3 LOADING
AND STEAM HUMIDIFICATION
ON CAMBRIDGE NON-COMBUSTIBLE FILTER



Appendix II

Preliminary Design Criteria for CWS Filter Processing

ATTACHMENT A

PRELIMINARY DESIGN CRITERIA
FOR
CWS FILTER PROCESSING

A-957

by
U. M. Anderson

WF-25-66

July 15, 1966

THE DOW CHEMICAL COMPANY
Rocky Flats Division
Golden, Colorado

U. S. Atomic Energy Commission Contract AF(29-1)-1106

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1.0 INTRODUCTION

Currently there is no process or equipment on hand to process the backlog of current generated CMS filters from Buildings 71 and 76. This report proposes a process and the necessary equipment to work off this backlog.

2.0 SUMMARY

The proposed process for recovering plutonium from CMS filters consists of the following:

- A. Remove frame from filter media using hand tools and tear filter media apart.
- B. Place filter media in a geometrically unfavorable vessel containing 13 M HNO_3 , CaF_2 and CaH_2 at 90° C.
- C. Leach material from 1 to 4 hours using mechanical agitation.
- D. Dump material out of vessel onto a screen backed by a roughing filter.
- E. Spray wash the remaining filter media with .55 M HNO_3 while on the aforementioned screen.
- F. Pull solid as dry as possible, with vacuum source on filter.
- G. Barrel solids with magnesia cement for burial.
- H. Transfer liquid to storage tank for further processing by ion exchange.
- I. (1) Bag out metal frames for burial.
(2) Bag out wooden frames for incineration.

The proposed C/S filter processing system consists of the following: (Figure 1)

- A. Drybox, air lock, bag and drum take out, etc.
- B. Disassembly area
- C. Steam jacketed, agitated kettle, with removable top.
- D. Fume removal system.
- E. Screens, and filters
- F. Spraying system
- G. Hopper
- H. Holding tanks
- I. Batch make-up tanks

3.0 GENERAL

3.1 Description

Flanders & Cambridge C.N.S. Type F Filter.

Size: 24" x 24" x 11-1/2"

Weight: ~ 0 lbs.

Filter media: glass-astbestos

Separator: aluminum or asbestos

Frame: wood or cadmium plated steel

3.2 Generation and Backlog

As of June, 1966, 463 filters containing an estimated 34.6 kg are on hand awaiting processing. Current generation is approximately 20 filters per month (300-1000 gm Pu.)

3 Processing

3.3.1 Leaching

To promote dissolution of the Pu contained on CWS filters it is necessary to both leach and agitate. While leaching some filters present a foaming problem, therefore the leaching vessel must have ample capacity to handle this situation.

Those filters coming from the 71 Building incinerator off-gas system are of the aluminum separator type. This aluminum when dissolved in HNO_3 liberates large quantities of NO , HF , and H_2 fumes. The system must therefore be capable of handling said vapors.

3.3.2 Filtering

The filter media in the leach solution do not settle out sufficiently to allow decantation of liquid. Therefore it is necessary to filter the slurry on a large enough surface area that reasonable flow rates can be maintained. The presence of cake slows filtering down considerably on this material.

3.3.3 Drying

Excess liquid must be removed from the material to prevent leakage during shipping. Some leakage is however tolerable if the barrel is properly packaged with magnesia cement.

Therefore the solid will be pulled dry by vacuum and packaged directly with magnesia cement.

4.0 DESIGN BASIS

The processing facility should be capable of leaching 1 filter per 8-hour period. Processing vessel should have capacity to leach a minimum of 1/2 CWS filters per cycle, with ample over-capacity for foaming. Assuming 30 to 75 gallons of liquid are needed per filter, a 20 to 25 gallon vessel should be acceptable. The vessel must have heat capacity to raise the temperature of the acid from 23° C to 100° C in less than 30 minutes and instrumentation to control the temperature in the 50° C to 100° C range.

The filter system should be of sufficient capacity to hold contents of leaching vessel and so designed to filter contents in less than one hour.

Makeup tanks and storage tanks must be of sufficient size to provide uninterrupted service on a three-shift basis.

5.0 EQUIPMENT

5.1 Drybox

A drybox is required to contain the system. An airlock is required to introduce a whole filter including outer package into the line. A smaller airlock is needed to put supplies into the line. Bag and drum take outs are required to remove filter frames and filter media.

5.2 Disassembly Area

A table is necessary to place filter on while removing the filter media from the frame. Electric hand tools will be used for this task. The filter media will be torn apart and placed in the kettle from this point.

5.3 Leaching Vessel

The leaching vessel should be an open top, steam jacketed, baffled, tilting kettle. It must be capable of heating 1.5 M HNO₃ from 20° C to 100° C in less than 30 minutes. A removable top complete with agitator and fume removal connection are necessary. The kettle should have a capacity of approximately 25 gallons.

5.4 Solid-Liquid Separation

The filtering system should be so constructed that a screen basket can be placed over the filter to allow for easy removal of the solid material. The filter should have storage capacity to hold the bulk of the slurry.

5.5 Fume Removal

The system should include a wet scrubber to remove nitrous fumes, HF, H₂, and SiF₄ from the reactant mixture.

Appendix JJ

Analysis of Waste Drums No. 771-7959 and 771-7961 Returned from the National Reactor Test Site

Reports on Drums Returned
to Rocky Flats from INEEL



THE DOW CHEMICAL COMPANY

ROCKY FLATS DIVISION
P. O. BOX 888
GOLDEN, COLORADO 80401

December 16, 1971

Mr. F. E. Abbott
Manager, RFAO, USAEC

ANALYSIS OF WASTE DRUMS NO. 771-7959 AND 771-7961,
RETURNED FROM THE NATIONAL REACTOR TEST SITE.

Transmitted herewith are two copies of the report, "Analysis
Of Rocky Flats Waste Barrels," prepared by A. K. Williams.


H. E. Bowman
Assistant General Manager
for Operations

HEB:mma
Orig. and 1cc - Mr. Abbott
Enc.

cc:
R. E. Hayne - Dow, Rocky Flats
M. E. Hughes - Dow, Rocky Flats
J. B. Owen - Dow, Rocky Flats
✓ M. A. Thompson - Dow, Rocky Flats
J. F. Willging - Dow, Rocky Flats
A. K. Williams - Dow, Rocky Flats

ANALYSIS OF ROCKY FLATS WASTE BARRELS

A. K. Williams

December 14, 1971

Distribution:

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A. K. Williams	-	Dow, Rocky Flats

ABSTRACT

This report covers the activity at both Rocky Flats and the National Reactor Testing Station (NRTS) in returning drums numbered 771-7959 and 771-7961 to Rocky Flats. The total quantity of plutonium found in these two drums was 1589 grams. Comparisons are made between various counting systems used to assay the drums and the final Rocky Flats' analysis.

INTRODUCTION

In January 1971, two Rocky Flats waste drums, Nos. 771-7959 and 771-7961, were shipped to the Idaho Nuclear Corporation (now Aerojet Nuclear Corporation [ANC]) for storage. At that time, it was indicated that each drum contained zero grams plutonium. A subsequent review of Rocky Flats (RF) drum counting data (March 1971) indicated that these barrels could contain appreciable quantities of plutonium. The two drums were located by ANC and isolated. Gamma measurements were made by J. E. Cline of ANC, and he reported that drum 771-7959 contained a minimum of 600 grams plutonium and a maximum of 1700 grams plutonium; and, drum 771-7961 a minimum of 370 grams plutonium and a maximum of 1100 grams plutonium. This information was received at the Rocky Flats Area Office on June 7, 1971 in a memo from W. L. Ginkel (IDO) to F. E. Abbott (RFAO). A team consisting of Dow, RFAO, and ALO personnel then visited IDO on June 30, through July 1, 1971 to put together a plan to repackage the drums to allow their return to RF using existing transportation permits. This report summarizes the activities which transpired in repackaging the drums for return. It also presents comparisons in data between counting systems used at NRTS by Gulf-Atomic Mobile Assay System (GAMAS) and ANC, and RF. These are compared with analytical data from RF. The official values reported are the RF analytical results.

CONCLUSIONS

In reviewing the contents of these drums and correlating dates and RF practices and procedures for loading and counting drums of waste and recoverable residues, the following conclusions were derived.

1. The plutonium scrap recovery area had been started up on a limited basis (dissolution had been allowed to start) when the substantial loss in plutonium was reported in the December 1970 inventory. The area was shut down and cleanup of the glovebox lines started. The major portion of materials in these drums were residues cut out from the dissolution lines, 2 and 3.
2. The sludge and filter media found in the drums should have been can counted and assigned a value before placing in a drum. There was a violation of procedures for those packages containing sludge and FulFlo[®] filters.
3. The drum counter produced count ratios which could not be handled by the computer program; thus, a negative number was printed.
4. The drum count sheets were not reviewed by a technical person as they were generated. The sheets showing zero grams plutonium allowed for shipping these drums to Idaho as waste.

5. Gamma survey results on these drums did not alert anyone to thinking that these drums contained large quantities of plutonium. This is probably due to some drums showing high gamma readings from americium.
6. Splitting of drums can be done safely at Idaho with experienced personnel following a predetermined plan.
7. Analysis by the GAMAS and ANC systems compare favorably with the final analysis at RF.

DISCUSSION

1. Preliminary Analysis

Because of anomalies in drum count results involving negative numbers and inventory problems in the plutonium scrap recovery plant in December 1970 and January 1971, a search was made by RF to determine if there were any drums sent to Idaho which may have contained substantial quantities of plutonium. In reviewing 27,800 barrel count sheets (for both RF fire waste and regular production waste) generated in the period June 1969 through February 1971, five drums were identified as being suspicious. Three of these drums were generated a number of months earlier and were buried in trenches. Two drums were generated in January 1971 and these were located by ANC and isolated from the waste storage area. These drums were identified as combustibles by RF. The drum count sheets for these two drums are shown in Figures 1 and 2. In reviewing these sheets, we found at RF that the gamma to neutron ratio by calculation could lead to a print of negative numbers, and that when the data was fed to the computer, the plutonium content would be reported as zero. In the past, small negative numbers (such as -1.0) could be expected in some drums with low counts. These two drums had high counts and large negative numbers. It was also noted that the scaler used in the counting system could exceed its highest number, flip over, and start again at zero. We concluded that this

occurred when these two drums were counted. Recognizing the problem, corrective action was taken to prevent the scaler from exceeding its limits. The computer was reprogrammed so a plutonium value would not be printed if the count ratios were outside present limits. These corrective actions were completed in February 1971. In addition to the above, a qualified technical man now reviews each drum count sheet to ensure that any strange count ratios are detected immediately. Besides preventing additional accidental shipments to waste, this technical review will also minimize assigning improper plutonium values to residues awaiting processing in the RF scrap recovery plant.

The gamma survey on these drums showed 32 mr/hr for 771-7959 and 17 mr/hr for 771-7961. These values indicated that the waste was active and could contain measurable quantities of plutonium. We surmise that at the time, the gamma activity was thought to be from americium.

Analysis by J. E. Cline of ANC showed that drum 771-7959 contained from 600 to 1700 grams plutonium and that drum 771-7961 contained from 370 to 1100 grams plutonium. Radiographs indicated an undefined high density area in drum 771-7959 which appeared to contain a large quantity of plutonium. Based on these results, it was decided to split

the drums at NRTS so that they could be returned to RF. The contents of each drum appeared to be too high to allow return to RF under existing transportation permits. A team consisting of Dow, RFAO, and ALO personnel visited IDO on June 30 and July 1, 1971, to devise a plan with IDO and ANC personnel to split the drums and return them to RF.

2. Plan for Opening RF Drums 771-7959 and 771-7961

Personnel involved in the planning at the June meeting were as follows:

Dow-RF	A. K. Williams, M. A. Thompson, J. B. Owen
RFAO	T. C. Jones
ALO	W. B. Johnston
IDO	G. Wehman, T. W. Asbury, B. D. Johnson, P. G. Voilleque, M. Hankins, B. Estes, B. L. Schmalz
ANC	B. R. Baldwin, J. W. McCaslin, R. B. O'Brien, D. P. Halls, W. W. Hickman, L. D. Hanson, J. Hanny

After preliminary discussions regarding operational safety and possible methods and locations for opening the drums, a plan was developed by contractor and AEC personnel. This plan was approved by all parties concerned on July 1, 1971. The plan called for the work to be performed in the ARA-1 hot cell, since this cell had one stage of HEPA filtration.

Rocky Flats personnel would construct a plastic tent inside the cell which would contain an additional stage of HEPA filtration. The active packages which were thought to be small, would be placed in either a modified 6M container (DOT permit 5791) or a 1518 container, when the drum was split. The purpose of the splitting was to reduce the plutonium content of each drum to less than 200 grams since this was the limit of DOT permit 5948. The modified 6M container (DOT permit 5791) can contain up to 4.5 kg plutonium for compounds having an H:X ratio ≤ 3 for Class II shipment. Upon returning to RF, it was found that this permit could not be used, since the plutonium must not decompose at temperatures up to 750°F. Because of this, a special exemption was requested for the 2030 container (DOT permit 5332) for up to 1500 grams per container.

Splitting of the drums was scheduled for the week of July 19, 1971. Upon completing the plan and obtaining agreement from everyone concerned, RF personnel began making arrangements to collect and ship the necessary materials to do the drum splitting at NRTS. A copy of the detailed plan is on file at Rocky Flats. Copies of this plan are also available at ALO, IDO, and ANC.

3. Drum Splitting

The Rocky Flats' team arrived in Idaho on July 19, 1971. Previous to their arrival, all needed supplies had been shipped by RF and received by ANC. The Rocky Flats' team consisted of the following personnel:

A. K. Williams - Chemical Operations
J. B. Owen - Waste Management
M. E. Hughes - Chemical Operations
R. E. Hayne - Health Physics

Upon arrival at the facility (ARA), a safety review was conducted by L. D. Hanson. A health physics indoctrination was then held at Central Facilities. Upon completion of the indoctrination, a health physics survey of alpha and gamma activity in ARA-1 was conducted by RF. The survey results were then compared with previous surveys taken by ANC health physics personnel. After comparing and agreeing on the activity levels, RF personnel then constructed a plastic house inside the cell. Four filters (2 feet by 2 feet by 1 foot) were needed to provide adequate airflow from the house. The house was completed the evening of July 19, and plans were made to open and repack the drums on July 20.

On July 20, the procedures to be used were again reviewed by RF personnel with L. D. Hanson at ANC. At that time, it was agreed that a dry run was not required. The house in ARA-1 was constructed so that the entire operation could be observed from the operating side. Besides those from ANC, observers from AEC-Washington - G. L. Daly, ALO - W. Holmes, and IDO - G. Wehman, observed all operations. An ANC health physics technician provided backup support, and L. D. Hanson remained in charge of all emergency equipment.

All Dow personnel were dressed to prevent skin contamination and wore full face mask respiratory protection during the drum splitting and cleanup operations.

After agreement was reached between ANC and Dow personnel on the adequacy of the procedures to be used, the first drum was transferred to Dow and placed inside the plastic house.

J. B. Owen and M. E. Hughes performed the drum splitting operation, R. E. Hayne performed the necessary health physics services, and A. K. Williams coordinated the effort and mapped the drums.

In the first drum (771-7959), the following observations were made.

- a. Upon opening the lid, alpha contamination was found on the inside surfaces of the drum. Because of this, all packages, as they were removed, were resealed in an additional bag. In some instances, packages were double bagged.
- b. The top 12 inches of the drum was empty.
- c. The top package consisted of tygon tubing which read 20 mr/hr and occupied the area to approximately a 6 inch depth.
- d. The next third of the drum contained nine packages which varied from 4 mr/hr to 250 mr/hr in gamma activity. Two of the packages were found which indicated high gamma levels. One package showed 250 mr/hr and appeared to consist of wet wiping papers. Some free liquid was observed in the bag. This bag was found at the center of the drum, directly under the top package of tygon tubing. The ticket on the package indicated that its contents came from Line 2, which is a dissolution line at RF. The second package was located 10 inches from the side of the drum about halfway down. This package showed 190 mr/hr gamma and contained about 1/2 inch of sludge in a quart plastic carton, and wiping papers.

- e. The bottom third of the drum contained packages which varied from 5 mr/hr to 22 mr/hr gamma.
- f. Material was identified as coming from Lines 2, 3, 5, and 21.

Based on these observations, each of the two high level gamma packages was placed in a 2030 container. The first was placed in container No. 771-7959B and the second package in container No. 771-7959A. The containers were sealed, surveyed, and removed from the cell. Outside, they were again surveyed and transferred back to ANC. The containers were then sent to the GAMAS trailer for analyses. The remaining packages were placed in a new drum and inspected and sealed following the waste management procedures used at RF. The empty contaminated drum was enclosed in plastic bags, sealed, and placed in a RF crate for disposal.

After removing all material from the first drum from the building, the second drum, 771-7961, was brought in and transferred to Dow personnel. Upon opening this drum, the following observations were made.

- a. No contamination was found when the drum was opened.

- b. Two active packages were found. One was found on the side about halfway down and reading 125 mr/hr gamma. The second package was on the opposite side and 6-10 inches lower, and read 110 mr/hr gamma. No free liquids were visible in these two packages and the material appeared to be plastic and paper. No wet items were found in this drum. Each of the two active packages were placed in a 2030 container. The 125 mr package was put in container 771-7961A, and the 110 mr package was put in container 771-7961B.
- c. No contamination was found inside this drum. The contents were split by removing half of the remaining packages and placing them in a new drum (771-7961A). The original drum was then numbered 771-7961B.

These drums were then packed and inspected using RF waste management procedures. Each of the 2030 containers and drums were monitored and removed from the cell. The containers and drums were then resurveyed and transferred back to ANC. Each of the above were then sent to the GAMAS for analysis.

4. Cleanup Operations

Upon completion of splitting operations, the plastic house was dismantled, and all items which were contaminated or

suspected of being contaminated, were placed in the RF crate. The crate was then sealed and prepared for disposal at the Idaho waste storage area. The remaining items were packaged for return to RF. After cleanup was complete, a health physics survey was taken of the ARA-1 cell and surrounding area. The survey results were compared with the initial survey and it was agreed between ANC and Dow that the operation did not further contaminate any area.

Upon completing the cleanup, Dow personnel returned to RF. Preliminary analysis by the GAMAS indicated that the drums each contained less than 200 grams and that each of the 2030 containers contained less than 1500 grams. The containers were then put in storage awaiting DOT approval for shipment to RF.

5. Shipment to Rocky Flats

Approval to ship these containers to RF was received on July 27. Rocky Flats personnel then made arrangements to pack the containers into the approved outer container and seal them for shipment to RF. In the meantime, ANC had informed RF that three of the containers showed a pressure on the gages. (The inner container of the 2030 shipping package is essentially a pressure cooker equipped with a pressure gage.) Container 771-7959B showed a pressure of

7 psig, 771-7961A showed 7 psig, and 771-7961B showed 1 psig. Preparations were then made to vent the containers before preparing for shipment to RF. Since observations were not made on gage pressure at the time the containers were closed, there was some uncertainty on the meaning of the observed values. With these particular containers, problems are occasionally experienced with faulty pressure gages. Dow personnel vented the containers on August 15. The only one showing pressure relief was 771-7959B. Container 771-7961A remained the same, and 771-7961B was not vented. Health physics surveys during the venting operation showed no release of alpha contamination. The four containers were received at Rocky Flats on August 18, 1971. Each of the containers was analyzed in the RF drum counter to provide an estimate of the plutonium values and then given to the R&D Chemical Processing group for processing and further analysis. Details of this work is found in CRDL-950345-15-1.¹ A summary of the data in this report is as follows:

Container No. 771-7959A

The contents were sludge and rags. These were separated, analyzed by can counter and then

¹R. E. Giebel, R. G. Leeb1, and J. A. Battaglino, "Plutonium Measurement in Idaho Waste Return," CRDL-950345-15-1, dated November 12, 1971 (internal report).

calcined and analyzed by X-ray. The analyses for this container were as follows:

GAMAS	- 270 grams
RF Can Counter	- 262 grams
RF X-ray on Ash	- 264 grams
RF Drum Counter	- 167 grams (estimate)

Container No. 771-7959B

The inner container showed 1 psig. Because of the history of pressure buildup at Idaho, the entire container was placed in the glovebox line. When the container was opened, no evidence of pressurization was found and the sealed plastic bags inside the container were not pressurized. The contents were found to be three filter pads containing plutonium bearing sludge, a FulFlo filter, a pint polyethylene bottle half full of wet sludge, and a second pint bottle 1/4 full of wet sludge, plastic and paper. The material was identified as coming from Line 2, and was packaged on January 18, 1971. A total of 70 ml of liquid was found which assayed 0.21 g/l Pu, 5.3×10^{-3} g/l Am, and 2.2 N H^+ . The contents of the container were split into five separate packages and can counted. Each package was then calcined

and the ash sampled for X-ray analysis. The analyses of this container were as follows:

GAMAS	- 600-800 grams
RF Can Counter	- 715 grams
RF X-ray on Ash	- 974 grams
RF Drum Counter	- 628 grams (estimate)

Container No. 771-7961A

Although the gage on this container showed greater than 15 psig, attempts to vent the container resulted in no apparent pressure relief. The contents were found to be plastic, a towel, paper, sludge, plastic tubing, and some scrap metal. The analyses for this container were as follows:

GAMAS	- 243 grams
RF Can Counter	- 243 grams
RF X-ray	- 252 grams
RF Drum Counter	- 225 grams (estimate)

Container No. 7961B

The pressure gage was under 7 psig. The plastic bag inside the container was pressurized and contained filter sludge. Analyses for this container were as follows:

GAMAS	-	98 grams
RF Can Counter	-	106 grams
RF X-ray	-	99 grams
RF Drum Counter	-	85 grams (estimate)

The RF drum counter results on the 2030 containers are low in comparison to the others reported. These results must be considered estimates, since the containers did not approximate the standards and the plutonium was highly localized.

Although we originally intended to leave the three drums of remaining material at Idaho, these were returned to RF on October 18. This was done at the request of Roy Crouch, ALO, and agreed to by A. K. Williams, Dow. Analyses of these drums were as follows:

<u>Drum No.</u>	<u>GAMAS</u>	<u>RF</u>
771-7959	155	163
771-7961A	217	165
771-7961B	50	61

Comparisons of RF, GAMAS, and ANC analyses for the total drum contents are found in Table I.

TABLE I

Comparison of Analyses Between RF, GAMAS, and ANC

<u>Barrel No.</u>	<u>g Pu GAMAS</u>	<u>g Pu ANC</u>	<u>g Pu RF</u>
771-7959	1060 ± 180	600-1700	--
771-7961	560 ± 90	370-1100	--
771-7959 Divided Totals	1125 ± 120	---	1401
771-7961 Divided Totals	608 ± 60	---	577

CORRECTIVE ACTION

Many of the corrective actions were implemented within a month of these two drums being shipped. These were instituted because of problems involving drum splitting at RF which, in turn, led to searching for anomalies in drum counting as outlined previously.

1. A new high level drum counter has been installed and is currently in operation. This counter contains many improvements which should eliminate the chances for shipping additional drums containing significant quantities of plutonium.
2. The original drum counter now has mechanical stops to prevent the registers on the scaler from flipping over. This counter is now used for non-line generated waste and as a backup system for the new counter.
3. The computer program has been changed so that count ratios outside the prescribed limit cannot be calculated. The print-out on the count sheet shows that the drum must be re-examined.
4. All count sheets (for drum, can, and Helix counters) are now reviewed by a knowledgeable technical person before releasing a drum to waste.

5. All bags cut from the production line are now surveyed using gamma and neutron survey instruments. From these surveys, a plutonium value is assigned and a running inventory is kept on each drum. Any package higher than 125 mr/hr in gamma plus neutron must be placed back in the line to determine the source of the radiation. The source must be removed before the waste or residue can go into a drum.
6. First line supervision reviews drum contents and certifies total plutonium content before the drum is sealed and sent to the drum counter. Gross differences between the counter and the estimate are reviewed. Gross differences may be defined as drums shown containing zero grams by supervision, but found containing 30 or more grams by the counter.
7. Waste Management inspection procedures have been implemented which would have probably detected these two drums prior to shipping to Idaho.

FIGURE 1

BARREL COUNTER

BACKGROUND GAMMA	2941	SAMPLE #	T064313
OXIDE STD GAMMA	49937	BACKGROUND NEUTRON	247
FLUORIDE STD GAMMA	92825	OXIDE STD NEUTRON	19862
		FLUORIDE STD NEUTRON	472776
SAMPLE GAMMA	358134	SAMPLE NEUTRON	378573
SAMPLE ATTENUATION	100	GROSS WT IN LBS	137
TARE WT IN LBS	64	NET WT IN LBS	73
GRAMS PU	-5149.86		
UNCORRECTED GRAMS PU	-5149.86		
MS CODE	330		
I. D. #	000		

DOCUMENT*****
 * 64313 *

 * FROM - ACCOUNTABILITY CODE * TO - ACCOUNTABILITY CODE * DATE *

 * * W/R * MATL BAL * * * W/R * MATL BAL * * * * *
 * CTL * ALOT * AREA * FUNCT * CTL * ALOT * AREA * FUNCT *MO*DAY* YR*

 2 * 01 * 137131 * 30 * 2 * 01 * 037432 * 20 *01* 19* 1 *

 * NEW ITEM * ELEMENT * ITEM DESC * NET WT * ASSAY * SS NET *
 * IDENT # * CODE * CODE * K-GMS * G/G * WT *

 * * 53 * 330 * 33.1 * 0.0 * 0.0 *

LESS THAN AEC DISCARD LIMITS

 *ISSUED BY **RECEIVED BY *
 *AUTHORIZED **AUTHORIZED *

KEYPUNCH
 FIRST CARD-----COL. 79-80, 10
 SECOND CARD-----COL. 79-80, 20

FIGURE 2

BARREL COUNTER

BACKGROUND GAMMA	2941	SAMPLE #	T064315
OXIDE STD GAMMA	49937	BACKGROUND NEUTRON	287
FLUORIDE STD GAMMA	92825	OXIDE STD NEUTRON	19862
		FLUORIDE STD NEUTRON	472776
SAMPLE GAMMA	395667	SAMPLE NEUTRON	514572
SAMPLE ATTENUATION	100	GROSS WT IN LBS	150
TARE WT IN LBS	64	NET WT IN LBS	86
GRAMS PU	-2136.79		
UNCORRECTED GRAMS PU	-2136.79		
MS CODE	330		
I. D. #	000		

DOCUMENT*****
 * 64315 *

 * FROM - ACCOUNTABILITY CODE * TO - ACCOUNTABILITY CODE * DATE *

 * * W/R * MATL BAL * * * W/R * MATL BAL * * * * *
 * CTL * ALOT * AREA * FUNCT * CTL * ALOT * AREA * FUNCT *MO*DAY* YR*

 2 * 01 * 137131 * 30 * 2 * 01 * 037432 * 20 *01* 19* 1 *

 * NEW ITEM * ELEMENT * ITEM DESC * NET WT * ASSAY * SS NET *
 * IDENT # * CODE * CODE * K-GMS * G/G * WT *

 * * 53 * 330 * 39.0 * 0.0 * 0.0 *

LESS THAN AEC DISCARD LIMITS

 *ISSUED BY **RECEIVED BY *
 *AUTHORIZED **AUTHORIZED *

KEYPUNCH
 FIRST CARD-----COL. 79-80, 10
 SECOND CARD-----COL. 79-80, 20

PLUTONIUM MEASUREMENT IN IDAHO

WASTE RETURN

R. E. Giebel
R. G. Leebl
J. A. Battaglino.

CRDL 950345-15-1

November 12, 1971

Chemical Processing
CHEMISTRY RESEARCH AND DEVELOPMENT
DOW CHEMICAL U.S.A
Rocky Flats Division
Golden, Colorado

Distribution:

D. R. Cartwright - w/o enclosures ✓
E. D. Erickson - w/o enclosures ✓
Attn: C. R. Forrey
M. A. Thompson - w/enclosures
E. Vejvoda - w/enclosures
A. K. Williams - w/enclosures
IRF

PLUTONIUM MEASUREMENT IN IDAHO

WASTE RETURN

INTRODUCTION

Rocky Flats routinely transfers plutonium waste to Idaho for interim storage. The barreled waste is randomly spot-checked for plutonium content by Gulf-Atomics using radiometric counting technique. Several barrels were observed to contain plutonium above the discard limits. The contents from these drums were placed into four pressure cookers by Rocky Flats personnel for return to the Rocky Flats plant. The pressure cookers were transported in DOT approved 20/30 shipping containers. The process waste material was radiometrically gamma-neutron counted upon return to Rocky Flats. To arrive at an accurate plutonium value for each pressure cooker, the contents were converted to an ash. The analyzed ash was returned to Chemical Operations for recovery. This report describes the processing of the waste material to a form applicable for plutonium measurement and summarizes the plutonium values obtained.

RESULTS

A detailed description of the processing scheme followed for each pressure cooker is presented in the Appendix. The contents of pressure cooker 7959B were subdivided into

five categories for processing because of its high plutonium content. Each category was gamma-neutron counted prior to being ashed for x-ray analysis. After being sampled, the ash from the five waste categories was combined and was analyzed as a single batch. The plutonium values obtained are listed in Table I.

Table I

Plutonium Results from Pressure Cooker 7959B

<u>Waste Category</u>	<u>Gamma-neutron Pu Determinations (g)</u>	<u>X-Ray Pu Determinations (g)</u>
Paper and Plastic	220 (-1-)	320
Fulflo Filter Element	47	35
R-6 Filter Pads	244 (-1-)	207
Bottle with Sludge (1st)	104	221
Bottle with Sludge (2nd)	<u>100</u>	<u>191</u>
Total (s)	715	974 ⁽⁹¹⁹⁾ (-2-)

(-1-) Estimated plutonium values. Can counter is not capable of assigning a value over 200 gms.

(-2-) Composite plutonium value.

The contents of each remaining pressure cooker were measured prior to ashing by radiometric (gamma-neutron) can counting. These radiometric plutonium values, and the plutonium values obtained by x-ray analysis on the ashed materials are compared in Table II.

Table II

<u>Cooker Number</u>	Comparison of Plutonium Results			DRUM COUNTER	
	<u>Can Counter (g)</u>	<i>ARCO</i>	<u>X-ray (g)</u>	OLD	NEW
7959A	262	267	264	166	157
7959B	715	800	974	2200	628
7961A	243	234	252	267	225
7961B	<u>106</u>	103	<u>99</u>	86	61
Total (s)	1326		1589		

APPENDIX

General Process Procedure

Each of the four pressure cookers (Type DPV-B) was shipped to Dow Rocky Flats in Type 20/30 shipping containers.

Each pressure cooker, containing process waste, was gamma-neutron counted at Idaho by Gulf Atomics. After receipt at Dow Rocky Flats, each pressure cooker was gamma-neutron counted within the 20/30 shipping container using both the old and new drum counters. Then each cooker was placed in a 55-gallon drum and again counted using the old and new drum counters.

Each cooker was then opened and the contents were examined. One cooker showed pressure on the gage and the entire cooker was placed in a glovebox to minimize accidental spread of plutonium contamination. The gage on another cooker indicated pressure but no pressure was released

when the relief valve was vented. However, when the cooker seal was broken, a gas was released and the gage returned to zero. The relief valve was defective. A third pressure cooker gage showed several pounds of pressure. The gases were released to the area ventilation system after sampling showed no radioactivity. The contents of each cooker were photographed, repackaged into one-gallon containers and gamma-neutron counted for plutonium determinations. The one-gallon containers were placed in a glovebox and separately calcined to an ash which was blended, weighed, and sampled for plutonium and americium.

Cooker 7959A

Cooker 7959A contained a one-gallon tin can (paint-type) which contained a polyvinyl chloride (PVC) cut-out bag. (See Figure 1, left side). The bag was placed in a glovebox and opened. It contained sludge and rags in a one-quart plastic container (See Figure 2). The contents were can-counted, calcined, weighed, and sampled (See Table II for results).

Cooker 7959B

The pressure gage on this cooker indicated one pound pressure. The entire cooker was placed in a

glovebox and opened. A sealed plastic outer bag and an inner bag were not pressurized; however, the inner bag was wet with droplets of a liquid.

Figure 3 shows these cut-out bags after opening.

The contents of the inner bag, as shown in Figure 4, contained three R-6 filter pads, one Fulflo[®] filter cartridge, a 500-ml poly bottle one-half full of wet sludge, a 500-ml poly bottle one-quarter full of sludge, loose wet sludge, plastic, and paper. A material identification ticket indicated the wet material was packaged on January 18, 1971 from

Line 2. A total of 70 ml of acid solution (2.2 N H^+) was collected from the plastic bags. Laboratory analyses indicated 0.21 g/l Pu and $5.3 \times 10^{-3} \text{ g/l americium-241}$. The contents were can-counted in five packages as follows:

1. Plastic, paper, wet sludge
2. Fulflo filter cartridge
3. R-6 polypropylene filter pads
4. 500-ml poly bottle with sludge (1st)
5. 500-ml poly bottle with sludge (2nd)

Each of the five packages was individually calcined and the resulting ash was sampled. The ash was combined, blended, and a composite sample was taken for plutonium assay. Table I summarizes the plutonium values determined for each package within cooker 7959B.

Cooker 7961A

The pressure gage on this cooker indicated greater than 15 psig. The relief valve did not function. The cooker contained a large plastic bag (Figure 1, right side) which was placed in a glovebox and opened. The attached material control ticket contained no information. Two smaller plastic bags were within the large bag. One bag contained a dirty towel, the other contained some scrap metal, paper, sludge, and a large plastic tubing (Figure 5). The two packages were can-counted separately. The metal was leached and the solution was combined with the sludge, calcined to dryness, weighed, and sampled. (See Table II for plutonium results).

Cooker 7961B

The pressure gage on this cooker was under 7 psig. The cooker contained a plastic bag which appeared pressurized and was immediately placed in a glovebox. The inner plastic bag contained filter sludge (See Figure 6). The contents were calcined, weighed, and sampled (See Table II for plutonium results).



FULL SIZE OF Figure 1
IS IN FILE A- 548

FIG. 1

Fig 1



FULL SIZE OF Figure 2
IS IN FILE A- 548

FIG. 2.



FULL SIZE OF Figure 3
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FIG 3

Fig 3



FULL SIZE OF Figure 4
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FIG 4

Fig 12



FULL SIZE OF Figure 5
IS IN FILE A- 548



FULL SIZE OF Figure 6
IS IN FILE A- 548

FIG 6

1064

A06913

A0 6913

Lab. No. MSL-74-133

Req. No. 52467

Date: 11-27-73

1 P

DOW CHEMICAL U.S.A.
Rocky Flats Division
Mass Spec Laboratory

ANALYSIS REPORT

To R. E. Nelson Bldg. 771 Charge 1371

cc: File Dept. _____

Ext. _____

Sample Description:

Sludge from Arco shipping drum.

Identify.

Results of Analyses or Tests:

Infrared, mass spectral, and thermal analytical results indicated the sludge was composed of minor amounts of organic residue, major amounts of inorganic and water.

- IR - Inorganic residue as: ammonium ion (NH_4^+)
- water
- siliceous material (SiO_2)
- metal oxides, halides (as PuO_2 , CaF_2 , etc.)
- possible graphitic material?

- mass spec outgas - major: water
- minor: HCl, HF, hydrocarbon, CO_2 , ammonia
- trace: dioctylphthalate (bag plasticizer?), POE (polyoxyethylene surfactant?)

TGA-DTA - The only thermal event recorded was a large water volatilization endotherm peaking at 100°C . The thermogravimetric trace indicated the loss of water from 25 to 180°C (equivalent to 37 wt. %), and gradual loss of residual water and other volatiles (equivalent to 27 wt. %) from 180°C to 770°C .

Analysis By MLR, DEA, JRG, RSC.

DIH, KJG

File or Plate No. IR 927

Reported By K. J. Gossaint

K. J. Gossaint

Approved By J. R. Turbett

J. R. Turbett



A0 7273 2
Rockwell International

ANALYTICAL REPORT

Atmospheric International Division
Rocky Flats Plant
P.O. Box 494
Golden, Colorado 80401

Organic Mass Spec Lab

A07273UINF

To	C. E. Wickland K. Terada File J. Hold 9.24.79	Account No.	389	Date	12-1-75	Lab. No.	MSL-76-188
				Reported by	D. I. Hunter and K. J. Grossaint		
				Approved	J. R. Turbett		

Sample Description
 Fourteen drums of laundry sludge returned to Rocky Flats from Idaho storage area.

Analysis Results
 These drums of laundry sludge were returned to Rocky Flats after a drum of the sludge was accidentally punctured and an audible pressure release was noted. Further examination revealed bulged tops on some of these drums.
 A special punch fixture was manufactured and used at Rocky Flats to determine the drum pressures and obtain a gas sample for Mass Spectrometric analysis.
 Drums no. 17-03909 and 17-03932 had the metal seal broken when received. The most severely deformed drum was 17-03919. A picture of this drum is included in this report.
 All 14 drums were marked content code 374 (blacktop, concrete, dirt and sand). Drum 17-03919 was opened and found to contain oil dry around the 55 gallon polyethylene bag of material. These bags were not opened and examined. No free liquid was observed.
 The source of pressure in all 14 drums was methane (CH₄) and carbon monoxide (CO).
 Details appear in the following table.

Drum No.	Pressure ⁽¹⁾ (torr)	Volume %			Light Hydrocarbons	O ₂	Ar	CO ₂	NO _x
		CH ₄	CO	N ₂					
17-03925	665	20	8	63	0.3	8	0.8	0.2	
17-03919	1396(27psia)	85	10	4	0.1	---	0.2	---	
17-03933	1190	55	17	24	0.3	3.5	0.5	---	
17-03907	896	49	11	38	0.2	1	0.6	---	
17-03898	928	62	11	26	0.3	0.2	0.4	---	
17-03909	613	3	3	73	trace	19	0.9	0.2	
17-03920	878	51	11	31	0.2	6.0	0.5	---	
17-03899	730	51	14	26	0.2	7.5	0.5	0.9	trace
17-03905	828	22	11	62	0.4	3	0.9	---	
17-03914 ⁽²⁾	1000	18	5	62	0.3	13	0.8	0.2	
17-03927	650	41	14	43	0.4	0.7	0.7	---	
17-03906	1066	14	7	67	0.4	11	0.8	---	
17-03932 ⁽²⁾	612	3	2	78	0.1	16	1	---	trace
17-03911	618	48	10	33	0.2	8	0.5	---	trace

(1) atmospheric pressure on analysis date = 612 torr.

(2) high N₂ and O₂ values indicate possible leak when sampled.

Note: trace hydrogen chloride (HCl), Chloroethene (C₂H₃Cl₃), carbon tetrachloride (CCl₄) and hydrogen sulfide (H₂S) were noted in all 14 drums.

Appendix KK

Letter from E. S. Ryan to G. E. White

August 19, 1963

G. E. White

cc:

J. G. Epp

M. E. Mass

File (Record)

HISTORY REPORT - PROCESS WASTE DISPOSAL GROUP - JULY, 1963

1. Personnel: Salary 3; Hourly 6
2. Facilities: The repair of the defects in the filter bank and the relocation of the plenum in the Building 74 addition were started by the Sheet Metal Shop.

The transfer of the liquid wastes in Pond No. 2A to Pond No. 2B was made by the Pipe Shop. A small heel of acid wastes remaining in Pond No. 2A was neutralized by pumping the basic liquid wastes in the south section of Pond 2B into Pond No. 2A. The liquid waste was then pumped from Pond 2A to the north section of Pond No. 2B.

Men from the Service Group resumed drumming of the salts and sand in Pond No. 2A.

A small scale experiment was made to test the burning capabilities of the lining of Pond No. 2A. The planking would not support combustion of its own and requires an additional fuel for rapid destruction.

3. Safety and Security: A safety and security meeting was held on July 29, 1963.
4. Trips: E. S. Ryan attended discussion meetings on July 8 and 9, 1963 at Oak Ridge National Laboratories, Oak Ridge, Tennessee. These discussions dealt with methods for disposal of contaminated liquid wastes by evaporation and deep well.
5. Visitors: Ray Miller, A.E.C., Health and Safety, ALO, Albuquerque, New Mexico, visited the Rocky Flats Division on July 25, and 26, 1963.

August 19, 1963

6. Operations: Table A is a summary of the liquid wastes, treated and untreated, which were released under the supervision of the Process Waste Disposal Group.

Table B lists the high and low of Fonds No. 1, No. 5 and No. 9.

Table C lists the drums of contaminated wastes in storage by the various buildings. Released drums of wastes were moved to Building 63 from the production areas.

Table D is a breakdown of 6 trailer shipments to Idaho Falls, Idaho.

Table E is a summary of the aqueous liquid wastes received, processed and released from Building 74.

The following radioassay results are on weekly composites made from daily samples of the effluent from the drainage tile.

Week of 7-1-63 - 7-5-63	200 d/m/l
Week of 7-8-63 - 7-12-63	150 d/m/l
Week of 7-15-63 - 7-19-63	170 d/m/l
Week of 7-22-63 - 7-26-63	150 d/m/l

Analysis of Well Waters Sampled July 25, 1963

Well	pH	d/m/l	NO ₃ ppm	Sp Gr @ 27°C	Depth of Water	
					7-25-63	6-21-6
No. 1	7.6	40	280	1.003	7' 2"	7' 6"
No. 2	8.1	10	1,100	1.000	8' 7"	6' 9"
No. 3	8.1	20	440	1.001	15' 4"	12' 2"
No. 4	7.7	5	3,400	1.003	12' 4"	11' 6"
No. 6	8.1	15	10	1.000	3' 10"	5' 7"

The wells were not bailed during the month.

Well No. 5 remains dry.

No oils or still bottoms were burned during the month.

7. Future Problems: A new method for the sampling of the laundry waters released by Building 42 is needed. The present method of grab samples by the laundry operators, whenever other operations permit, does not supply a representative sample. A sampler controlled by the cycling mechanism of the washers would give a sample which would have more meaning.

The disposal of cyanide wastes which are produced in Building 44 are a potential problem. In the past, these wastes were set-up with Portland Cement in Building 44. At the present time, these wastes are being sent to Building 81 for destruction. The presence of flouride and Building 44 material in the waste makes this method undesirable for Building 81.

8. Development Work: F. E. Butler, Process Chem, CRML, reports no work on Waste Disposal projects because of a rush project for Midland.

K. Fry, Analytical Labs, Building 71, began a study on the feasibility of concentrating radioactive material with the iron hydroxide floc. This study covers the repeated use of the floc in successive batch treatments of aqueous wastes. If successful, the process could lead to the recovery of the material from the iron floc.

E. S. Ryan

E. S. Ryan

ESR:bls

Appendix LL

Investigation CCl₄ Carbon Tetrachloride by R. W. Hawes

INVESTIGATION CCL4 CARBON TETRACHLORIDE

11-27-85 R.W. HAWES

CONTACTS PHIL SHOEMAKER Foundry 707 x-7959 page 4000-108
KIETH GROSSAINT Labs 559 x-2154
RICK GETTY Labs 559 x-4791
WENDY HENDERSON Manufacturing 707 x-4705
PAUL KING Foundry 707 page 4000-280
FARREL HOBBS 779 x-7431
ROCKY PETRACCHI Industrial Hygiene

Shoemaker confirmed the figures supplied by Setlock on the CCl4 usage. Hobbs figures usage at 14,000 gallons in the last 12 months

The other substance used to clean chips is Freon TF. Estimated at 55 gallons per month in B module, 55 gallons per month in C module, 25 to gallons per month in 776 and small amounts used in A module. This will combine with the carbon tet and be disposed of in the same facility. Estimated total usage 150 gallons per month. Hobbs says that the Freon is used for density determinations more than cleaning and is discarded frequently but the usage is probably right.

The evaporation rate of this Freon is about one third the rate of evaporation of carbon-tet.

Other substances used;

Texas Regal cooling oil known as Regal R&O or Regal 645
Mobile 643 hydraulic oil
Tra-Bon lubricating oils for the machines.

Total usage of these oils discarded to the same tank as the carbon-tet is about 28 gallons per month.

The waste tanks used are the pencil tank V-32 in the pit of module C, of building 707 and tank 1103 in building 1103. Currently tank V-32 empties into tank 1103 where it is sampled for plutonium before it is turned over to building 774 Liquid Waste Processing.

The primary usage of the carbon tetrachloride is to wash the oils off the metal chips. This is done prior to briquetting the chips for the foundry.

Paul King is collecting 500 ml samples from the waste tanks and will send them to 559 labs (11 a.m. 11-27-85).

^{TCA}
Hobbs says that 1,1,1 Tri chloroethane is used in a closed cleaning system and only trace amounts would be found in the waste liquids. Trade name "Chlorthane V6" usage estimated at 500 gallons per month. (W.A.G.)

Appendix MM

A Control Design for Plutonium Counting Systems

REFERENCE

A CONTROL DESIGN
FOR
PLUTONIUM COUNTING SYSTEMS. *SPEECH.*

by
L. W. Doherty and J. D. McBride

For Presentation at the 11th Annual Meeting
of the
Institute of Nuclear Materials Management

Descriptors

- Nuclear materials management
- Plutonium
- Radiochemical analysis
- Safeguard

May ~~24~~ 26, 1970

Gatlinburg, Tennessee

ABSTRACT

This paper describes a design which assures control of the plutonium counting systems used at RF to measure solid plutonium in waste and process materials, while establishing measurement bias and variability information for process control and inventory management.

INTRODUCTION

Chemical processing of plutonium offers many challenges because of this element's unique characteristics and because of the special administrative controls that surround it. The challenges include (1) the physical containment of plutonium for health reasons, (2) the danger of nuclear excursions, and (3) the accountability required of plutonium. This paper deals with the control of one type of measurement of plutonium for safeguard and accountability reasons. The method of measurement is radiometric, which is non-destructive.

At the 10th Annual Meeting of the Institute of Nuclear Materials Management, O. H. Willoughby and D. R. Cartwright of the Dow Rocky Flats Research and Development Department described the development of a non-destructive method for assaying plutonium in waste and process residue.¹ This method is radiometric and is classed as "passive"; that is, it uses the inherent radiation from the plutonium as a basis for measurement.

For a description of the measurement systems, two paragraphs from the Willoughby-Cartwright paper are quoted, as follows:

"The first radiometric system, the drum counter, was

Slide 1

installed in 1964. The drum counter geometry consists of an annular array of eight 30-inch halogen-quenched Geiger-Müller detectors and 16 boron trifluoride neutron detectors within a shielded cavity. Fifty-five-gallon drums of contaminated waste are lowered into the cavity with a hydraulic hoist. Cadmium shielding is wrapped around

¹ RFP-1325, "Measurement of Plutonium in Process Materials and Contaminated Waste," O. H. Willoughby and D. R. Cartwright; a speech presented to the 10th Annual Meeting of the Institute of Nuclear Materials Management, April 28 - 30, 1969, at the Stardust Hotel, Las Vegas, Nevada.

the gamma detector to reduce the effects of low-energy radiation from americium on plutonium measurements. Electronic pulses resulting from radiation passing through the detectors are amplified, shaped, and scaled using standard nuclear instrumentation.

The early successes with the drum counter led to the development of the can counter, which is similar in design to the drum counter but smaller. The can counter performs plutonium

Slide 2

assays of gallon-size waste or residue packages in the process area where the wastes or residues are generated. The speed of the analysis and the easy acquisition of data have been of great benefit to production personnel in making accountability and management decisions for further waste and residue processing."

Existing data indicate that the can and barrel method of measurement is superior to the classic method of sampling and subsequent analysis for plutonium by laboratory methods, primarily because sampling errors are removed by measuring entire batches with the counting geometries.

The measurement systems described above provide the instrumentation necessary to detect plutonium, but the raw hardware by itself cannot assay the amount of plutonium present without the use of standards for comparison.

Consideration of standards for use with the geometries presents a two-fold problem: (1) the sample medium and the geometry affect the response of the detection systems, and (2) the amounts of plutonium required to provide multipoint, standard curves derived from individual standards are so great as to be economically impractical. Control of routine measurements at RF depends upon comparison with absolute standards, which must overcome the problems just mentioned.

In order to solve these problems, the concept of a modular standard was adopted in which each standard consists of modules appropriate to the geometry. For the drum counter, the modules are 1-gallon plastic bottles within a 55-gallon barrel; for the can counter, the modules are plastic-bagged material in a 1-gallon plastic bottle. Each module for each category of residue contains either (1) dry plutonium dioxide plus the average medium for that category, mixed until they are homogeneous, or (2) the medium only. Such a concept permits wide variability in both the amount of plutonium contained in the standard and the plutonium per unit weight.

The standard modules are prepared by mass measurement of Rocky Flats stream plutonium in synthetic, average media, rather than by sampling and destructive testing of actual residue.

Slides 3 and 4

The plutonium for these standards is processed as follows: (1) plutonium peroxide is received from Chemical Operations; (2) the plutonium peroxide is calcined to the stoichiometric ratio of plutonium dioxide ($\text{PuO}_{1.98}$) at 850°C . for 100 hours; (3) the oxide is then analyzed and the plutonium content corrected for metallic impurities, such as Am^{241} ; and (4) isotopic ratios are determined by mass spectrometry, for calculating the atomic weight.

Both the media and the dry plutonium dioxide are measured and loaded into the modules so that the outer surfaces of the modules are free of alpha radioactivity. The modules can then be safely manipulated inside their 55-gallon and 1-gallon containers.

A standard for the drum counter consists of 27 wide-mouth plastic bottles. To simulate the heterogeneous characteristics of process plutonium residue in the barrels, some of these bottles are standard modules containing plutonium dioxide, some are empty bottles, and some are bottles containing residues free of radioactivity. (Slide 5 demonstrates the concept and versatility of the standard for the drum counter.)

Slide 5

A standard for the can counter consists of nine standard modules which are plastic bags that may either contain plutonium dioxide or may be "blank". Thus, a can standard possesses the same characteristics as a drum standard.

Slide 6

RF categorizes process residues according to their origin, and therefore their physical and chemical characteristics. Because the medium of these residues affects the radiometric response in the two counting systems, the Standards Laboratory is obliged to synthesize standards representing these same categories.

Standardization for category of residue requires multiple working standards that permit variations in standard values. This is possible, using a minimum amount of plutonium, because the working standards are made up of modular standards. With these standards, R and D personnel can provide standard curves for each category. The curves are normalized using both ideal (homogeneous) standards and non-ideal (heterogeneous) standards for each datum point. Therefore, a certain amount of variability is represented in the standard algebraic expressions because of the heterogeneous nature of RF solid residues.

The standards are also used in the RF measurement control program.
Why a measurement control program?

The former RFD procedure for the measurement of amounts of nuclear material involved the following separate operations:

(1) each batch of material was weighed or measured by volume, and sampled; and (2) the sample was analyzed for SS material content.

Because the analysis can never be exact, and because the sample cannot be considered truly representative, the measurement may be inaccurate. Therefore the quantities of plutonium used in computing a material balance are inexact. It becomes important, then, to determine just how inexact or biased the measurements are.

The RF measurement control program evaluates the systematic errors and their variabilities. Measurements of a given characteristic (plutonium assay) are compared with other measurements of the same characteristic made with an accepted standard. The second, or standard, measurement may be considered absolute because as a weight measurement it is vastly superior to the first, or observed (counting) measurement. The average difference between the observed and the standard values is the bias of the measurement. The standard deviation of the bias is a good measure of its variability.

The measurement control program is divided into categories of process materials and residues which are assayed for plutonium content in the counting geometries. Each program is assigned a Nuclear Materials Management (NMM) description code number. The Chemistry Standards Laboratory changes the values of the plutonium inside each standard by manipulating the modules, records these changes, and submits at least five such standards per month to be measured. The counting personnel measure the plutonium

Slides 7 and 8

in each standard and transmit the observed value to the Chemistry Standards group, which compares the values by subtracting the standard from the observed value. Thus, the validity of the measurement system for a particular residue category is established.

At the end of the month, the measurement control program numbers, the standard values, and the observed values are summarized by computer according to program and category as follows:

Slides 9, 10, and 11

(1) Each standard value is subtracted from its corresponding observed value; (2) an average of these differences is determined and (3) the standard deviation of the differences is calculated.

The average difference determines the mean bias per type of residue and the standard deviation around this average estimates the expected variability of each bias. The variability establishes realistic limits for a given bias, which in turn helps determine the disposition of residues, the validity of counter measurements, and the accuracy of inventories. The limits of error may then be calculated using these statistics.

In a recent six-month period, the measurement control program reflected biases from 1 percent to 20 percent and variabilities ranging from 4 percent to 37 percent. The measurement category showing the 37 percent variability was considered out-of-control and the production material surrounding the standard evaluation was remeasured following the appropriate remedial action.

CASE HISTORY

A portion of the RF plutonium chemical processing involves a purification process using ion exchange resin. The spent resin becomes a residue which is either leached to a discard level or incinerated for plutonium recovery. Disposition of this material must be decided after it has been packaged, removed from the dry-box system, and stored in 55-gallon drums. The spent resin is measured by the drum counter. The Standards Laboratory establishes a control for this drum counting measurement, as previously described.

During one period, the measurement control showed a bias as great as 37 percent. Such a bias in any measurement make it impossible to make a decision. If the bias is unknown, a decision could be made based upon an assay that is assumed to be valid and the decision would probably be wrong.

Fortunately, the bias was known from the measurement control program and remedial action was taken. All measurement was halted. The instrumentation and counting procedure were investigated and revised as necessary. All drums assayed from the last acceptable control count to the unacceptable one were recounted and the records adjusted. Disposition was made based on the recounted values. Control was re-established and measurements continued.

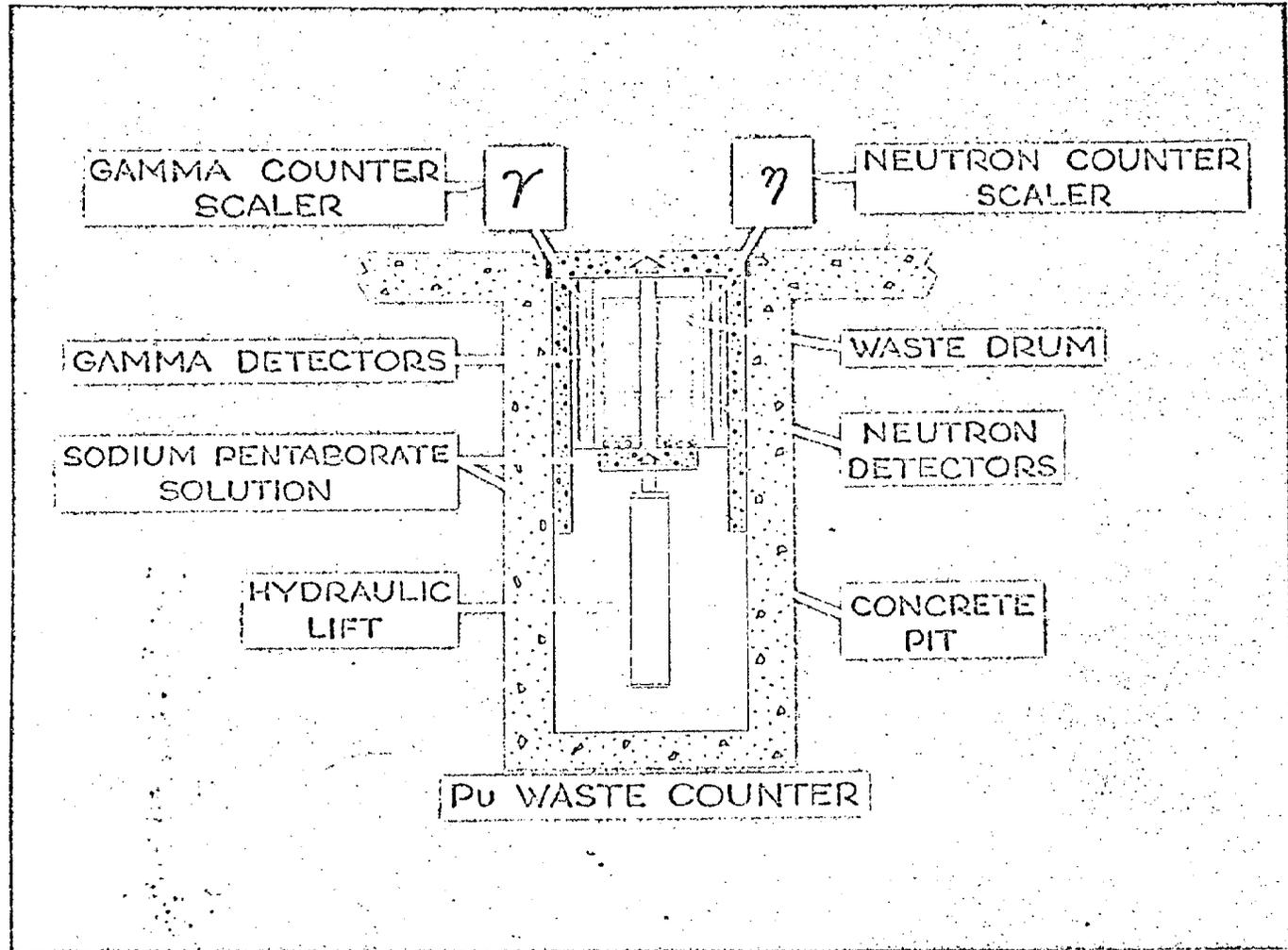
SUMMARY

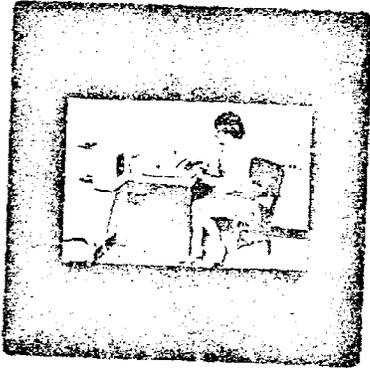
The Rocky Flats Division of The Dow Chemical Company assays plutonium in solid chemical process materials and contaminated wastes by non-destructive, radiometric techniques. Such working measurements were new to RF a few years ago and, therefore, the Chemistry Standards Laboratory was obliged to design a system that both standardized the instrumentation and placed measurement controls around it. Both calibration and working standards were designed and tested. These simulated those plutonium-bearing, solid, process materials and residues measured by the two types of counting geometries developed and currently in use at RFD. These standards have two constant external configurations, but they are highly variable in their internal configurations and their ranges of standard values. Thus, the standards are able to evaluate (1) the calibration curves, (2) the geometries and electronics, and (3) routine process measurements, under true working circumstances. The measurement control programs, which developed from the design, supply bias and variability information from (1) interplant shipments, (2) in-process items, and (3) discards. The controls are therefore useful management tools and proceed on a continuing basis.

ACKNOWLEDGEMENTS

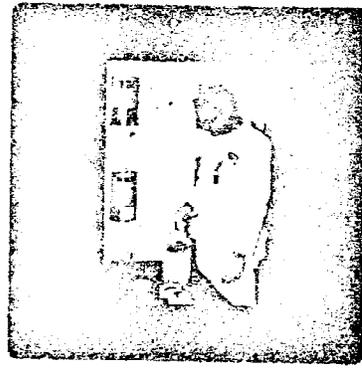
The authors wish to thank O. H. Willoughby and D. R. Cartwright for their technical assistance, W. R. Meininger and the Chemical Operations Staff for the process-related information, and the Technical Writing group which assisted in preparing this paper.

SLIDE 1 : DRUM COUNTER SCHEMATIC

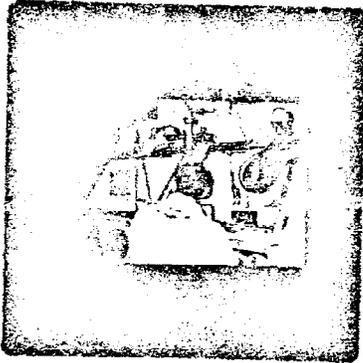




SLIDE 9



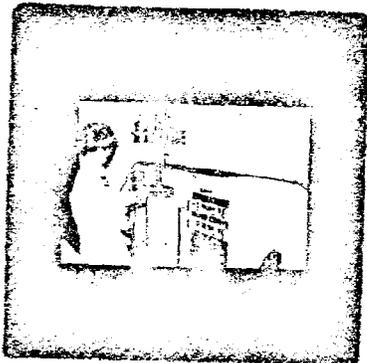
SLIDE 8



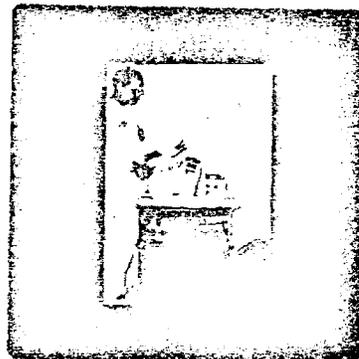
SLIDE 5



SLIDE 4



SLIDE 2



SLIDE 7

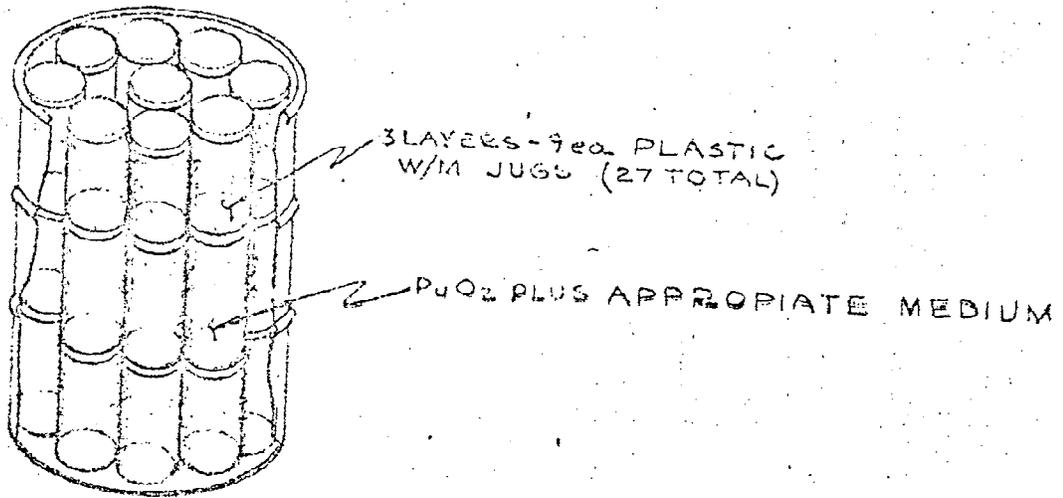
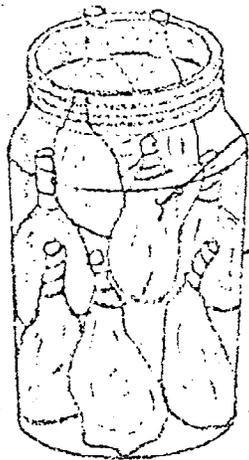


FIG. 2. COMBUSTIBLES AND WASHABLES BARREL COUNTING STD.

SLIDERS

DRAFT



9 BAGS COMP. FULL

1 GAL. W/M NALGENE
CONTAINER

FIG. 3 TYPICAL CAN COUNTING STD.

SLIDE 6,

DRAFT

STATISTICS - CURRENT MONTH

<u>N</u>	<u>Bias</u>	<u>S.D.</u>	<u>Avg.S.V.</u>
5	-3	+4	102

MEASUREMENT CONTROL REPORT

<u>Prog.No.</u>	<u>Description</u>	<u>Range</u>
C400	Contained Pu in Resin by Barrel Count	0 - 175 g

Appendix NN
A Plutonium Waste Counter

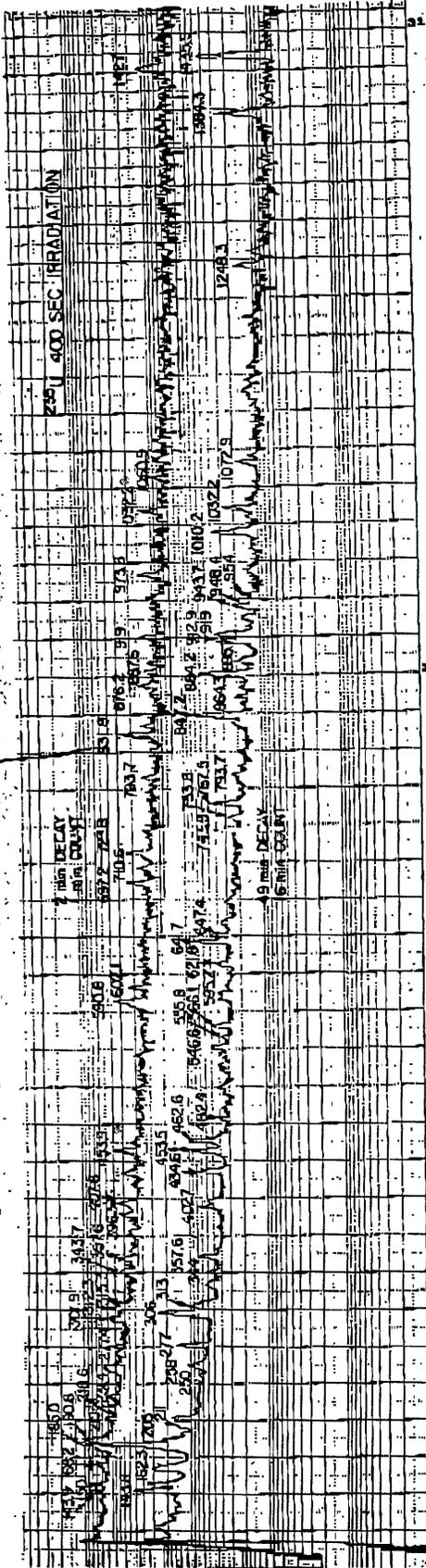


Fig. 1. Portions of fission product gamma-ray spectra following a 400-second bremsstrahlung irradiation of a ²³⁵U sample. End-point energy of the bremsstrahlung beam 8 MeV.

line can be improved by optimizing the irradiation, decay, and count times. However, the complex gamma spectrum requires high resolution spectrometers and analysis procedures to accurately extract the gamma-line intensities.

An isotopic assay has been performed on some composite samples of uranium, thorium, and carbon. Using the 12 most prominent gamma lines, the amounts of ²³⁵U and ²³²Th were determined with an overall accuracy of better than 8%. The ²³⁵U content was measured to be within 2% of the known amounts.

1. N. C. RASMUSSEN, J. A. SOVKA, and S. A. MAYMAN, "The Non-Destructive Measurement of Burnup by Gamma-Ray Spectroscopy," *Proc. Symp. Nuclear Materials Management*, pp. 828-48, IAEA (September 1965).

2. S. KALCOFF, "Fission-Product Yields from Neutron-Induced Fission," *Nucleonics*, 18, 11, 201 (1960).

4. A Plutonium Waste Counter,* J. L. Lawless, R. N. Chanda (Dow)

An operational system designed for the measurement of large quantities of plutonium contained in gallon-sized packages is described. The system performs a passive measurement employing both the 384-keV gamma-ray complex of ²³⁹Pu and spontaneous fission neutrons from ²⁴⁰Pu. Also included is a provision for measuring and correcting for the gamma-ray attenuation in the sample.

A block diagram of the system is shown in Fig. 1. Several unique construction features are contained in the design, including the selection and arrangement of shielding and the scanning mechanism. The sample rests upon a turntable which causes the sample to describe a helical

TABLE I
 Evaluation of Helix Counter for Graphite and Ash Standards

Standard Value (grams)	Bias (%)	Precision (%)	Bias at 95% Confidence Level (%)	Precision at 95% Confidence Level (%)
Graphite				
10	6.3	1.1	6.5	2.3
25	9.8	1.2	12.3	2.6
50	5.1	1.0	7.1	2.1
100	2.9	0.8	4.5	1.7
200	-0.4	2.0	4.3	4.1
350	2.2	2.5	7.2	5.3
500	-2.7	5.8	14.2	12.1
Ash				
10	6.0	1.6	10.0	7.0
25	2.6	2.0	6.8	7.6
50	-2.0	1.4	3.4	5.2
100	-0.1	1.8	3.7	6.7
200	-0.5	1.6	3.8	9.0
350	2.3	4.4	11.1	16.4
500	-4.2	4.0	12.1	14.8
698.8	-4.1	3.6	12.1	14.2

*Sponsor: L. A. Matheson.

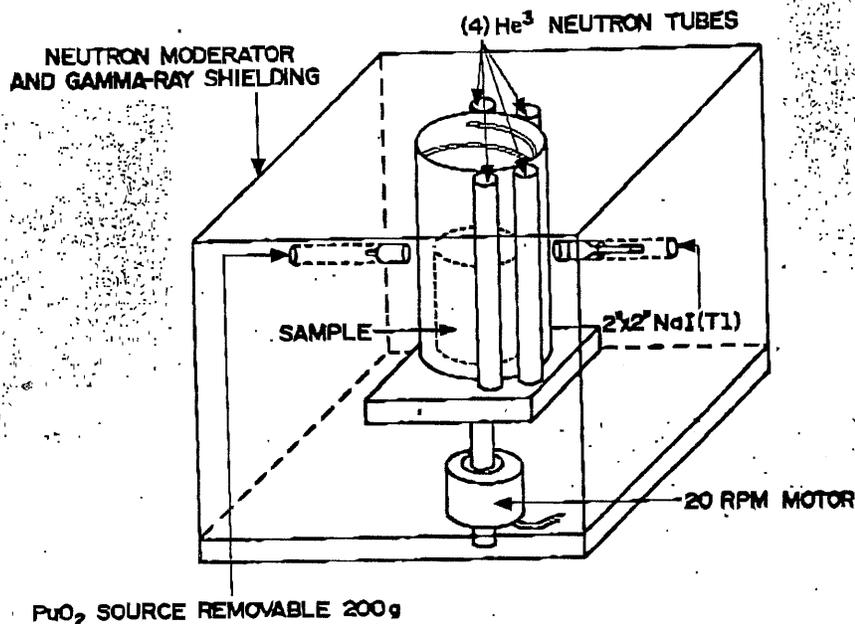


Fig. 1. Helix counter.

path between a 2- x 2-in. NaI(Tl) detector and a 200-g PuO_2 gamma-ray attenuation measurement source. Two 100-sec scans are performed, one of which is made with the external plutonium source for the sample attenuation measurement. The NaI(Tl) gamma-ray response is recorded on a single-channel analyzer set to effectively bracket the 384-keV complex. A simultaneous gross neutron count is performed using four ^3He neutron detectors. Conventional commercial electronics are used throughout. The entire system including electronics, moderator, and shielding rests upon a movable cart.

The precision and accuracy of the counter has been measured using certified standards of graphite and ash matrices.^{1,2} Each standard was counted 5 times and the data subjected to statistical analysis. Over a range of 0 to 700 g of contained plutonium, a precision was obtained of $\approx 6\%$ with a bias of $\leq 10\%$. At the 95% confidence level, this corresponds to a bias of $<14\%$ with a precision of $<16\%$. This system has been in operation at Rocky Flats for the past year.

1. J. L. LAWLESS, Y. M. FERRIS, and J. D. McBRIDE, "Evaluation of the Helix Counter Incinerator Ash Study," CRD-940232-104-C, Rocky Flats (April 1970).
2. J. L. LAWLESS, Y. M. FERRIS, and J. D. McBRIDE, "Evaluation of the Helix Counter Virgin Graphite Study," CRD-940232-104-D, Rocky Flats (April 1970).

5. Nondestructive Gamma Assay of Plutonium-Contaminated Solid Waste in Drums; Alexander J. Dukat, John Gonser, Dean B. James (NUMEC)

Nuclear materials management and safeguards as well as operations control require that drums of plutonium-contaminated wastes be assayed for their plutonium content. Destructive analytical techniques are not practical because of the difficulty of obtaining representative samples. A nondestructive analysis should provide a simple and rapid analysis of total plutonium content of the drum, with no bias resulting from physical and geometrical configurations within the drum.

The measurements were made with a 55-cm³ Ge(Li) detector and a Nuclear Data 2200 series multichannel analyzer as described previously.¹ The drum was rotated at 4 rpm on a turntable while being counted by the stationary detector. A previously determined environmental background was subtracted, and an approximately 85-keV portion of the "384-keV" complex was integrated. A horizontal Compton correction was based on the activity in a 20-keV portion of the spectrum just higher in energy than the integrated portion of the spectrum. A correction was made for the self absorption of the solid material within the drum by comparing the activity due to a planar plutonium source shining through the unknown drum to that activity produced by the same source shining through a dummy drum.¹

With an operationally reasonable scan time of 1000 sec of live time and a detector-face-to-barrel-center distance of 54 in., the calibration coefficient measured with a drum containing varying numbers of known plutonium packages with PVC-film matrices was determined to be 1.1019 (g sec)/count.

Using twice the standard deviation of the background as a criterion for detection, 0.07 g of ^{239}Pu may be determined. Experimental one-sigma precision for 1, 10, and 50 g of ^{239}Pu is ± 0.12 , ± 0.33 , and ± 0.87 g, respectively, as determined by repetitive measurements.

The extent of the loss in accuracy with package location in the drum was determined by filling a drum with three layers of seven, 1.5-gal cardboard packages each, all of the same matrix material, and then alternately substituting one package with a plutonium-containing package. With reference to a drum containing plutonium uniformly distributed throughout, a package in the center of the drum showed about an 11% reduction in activity; packages on the bottom-layer edge showed a 3% reduction in activity; while packages on the top layer showed a 6% increase in activity.

The necessity of segregating materials into drums with similar gamma absorption density was demonstrated. Packages of real waste were assayed by the package-assay system¹ and randomly placed in drums without regard to plutonium content or self-absorption correction.