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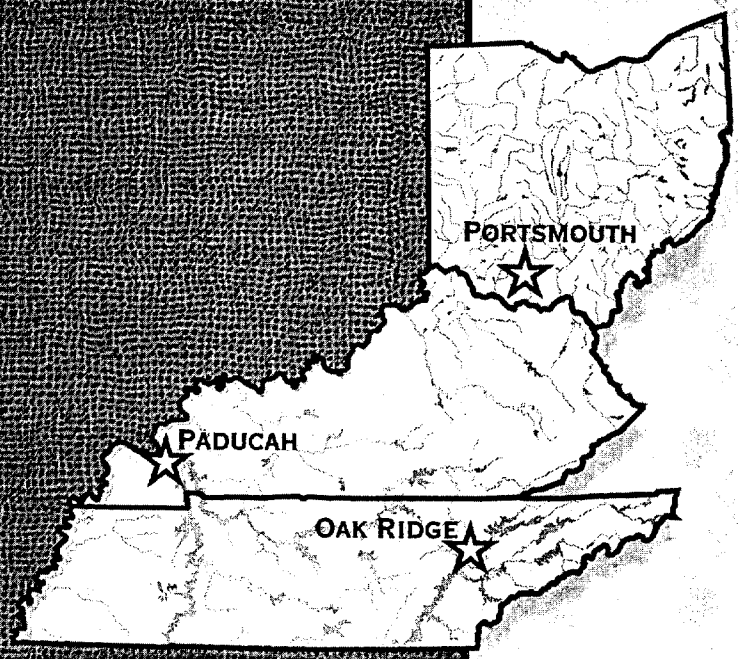
RECYCLED URANIUM MASS BALANCE PROJECT PORTSMOUTH, OHIO SITE REPORT



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RECYCLED URANIUM MASS BALANCE PROJECT PORTSMOUTH, OHIO SITE REPORT

June 19, 2000

Prepared by
TetraTech NUS and Theta Technologies, Inc.
under subcontract 23900-BA-ES008

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BECHTEL JACOBS COMPANY LLC
managing the
Environmental Management Activities at the
Portsmouth Gaseous Diffusion Plant
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ACRONYMS

^{237}Np, Np	Neptunium-237
^{239}Pu, Pu	Plutonium-239
^{99}Tc	Technetium-99
AEC	Atomic Energy Commission
ASTM	American Society for Testing and Materials
B&W	Babcock and Wilcox
Ci	Curies
CIP	Cascade Improvement Program
CUP	Cascade Uprating Program
DAC	Derived Air Concentration
D&D	Decontamination and Decommissioning
DIA	Division of International Affairs
DOE	Department of Energy
dpm	Disintegrations per Minute
DU	Depleted Uranium
EH	DOE Office of Environmental Safety and Health
EH-3	DOE Office of Nuclear Safety
ERP	Extended Range Product
ES&H	Environmental, Safety, and Health
F_2	Fluorine
FP	Fission Product
FY	Fiscal Year
GAT	Goodyear Atomic Corporation
GCEP	Gas Centrifuge Enrichment Plant
GDP	Gaseous Diffusion Plant
gU	Grams of Uranium
HEU	Highly Enriched Uranium
HF	Hydrogen Fluoride
HRT	Hanford Reactor Tails
HT_CO_4	Pertechnetic Acid
ICP	Inductively Coupled Plasma Spectroscopy
ICPP	Idaho Chemical Processing Plant
kg	Kilogram
kgU	Kilograms of Uranium
LAW	Low Assay Withdrawal
LEU	Low Enriched Uranium
LLW	Low Level Waste
M&O	Maintenance and Operating
MDL	Minimum Detectable Limit
MgF_2	Magnesium Fluoride
MPC	Maximum Permissible Concentration
MTU	Metric Tons Uranium
NaF	Sodium Fluoride
NFS	Nuclear Fuel Services, Inc.
NLO	National Lead of Ohio (Fernald Plant)
NMMSS	Nuclear Materials Management and Safeguards System
NRC	Nuclear Regulatory Commission
ORGDP	Oak Ridge Gaseous Diffusion Plant
PGDP	Paducah Gaseous Diffusion Plant
PPF	Paducah Product Feed
PORTS	Portsmouth Gaseous Diffusion Plant

ACRONYMS (CONT'D)

PUREX	Plutonium – Uranium Extraction
RSD	Relative Standard Deviation
RU	Recycled Uranium
SNM	Special Nuclear Materials
SRT	Savannah Reactor Tails (Savannah Recycle Tails)
SS	Source and Special
S&S	Safeguards and Security
TRU	Transuranic
U	Uranium
UO_2	Uranium Dioxide
UO_3	Uranium Trioxide
U_3O_8	Triuranium Octoxide
UF_4	Uranium Tetrafluoride; Green Salt
UF_6	Uranium Hexafluoride
UNH	Uranyl Nitrate Hexahydrate
USEC	United States Enrichment Corporation
VHE	Very Highly Enriched

Executive Summary

This report has been prepared as the Portsmouth Gaseous Diffusion Plant (PORTS) response to Deputy Secretary of Energy, T.J. Glauthier's memorandum of September 15, 1999. It is presented to address:

1. Shipments and receipts of recycled uranium (RU);

2. Levels of transuranic (TRU) and fission product (FP) contaminants in the PORTS flows and processes that had the potential to expose workers; and

3. Information on mass balances for the RU, TRU, and FP to identify potential Environmental, Safety and Health (ES&H) concerns.

The PORTS site has received and dealt with RU and its legacy from startup through current modern day operations. Sources and amounts of RU received were:

1. Uranium hexafluoride (UF_6) feed manufactured at the Paducah Gaseous Diffusion Plant (PGDP) or the Oak Ridge Gaseous Diffusion Plant (ORGDP) from recycled uranium trioxide (UO_3) – 1,095.1 metric tons uranium (MTU);

2. UF_6 feed supplied from other miscellaneous foreign and domestic sources – 5.0 MTU; and

3. Oxides and other non-UF_6 forms of uranium containing TRU/FP from miscellaneous foreign and domestic sources for conversion to UF_6 at the PORTS Oxide Conversion Facility – 19.0 MTU.

4. A total of 4.6 MTU of non-UF_6 may have been utilized for development activities.

Additionally, an estimated 60 to 90 kilograms (kg) of the fission product technetium-99 (^{99}Tc) were received through FY 1997 in some 121,485 MTU of enriched UF_6 withdrawn from the PGDP cascade and supplied to PORTS as feed.

Most of the RU-UF_6 was used as gaseous diffusion (cascade) feed with the last sizeable amount (400 MTU) fed in January 1974. Subsequently, a smaller amount, 1.4 MTU of highly enriched uranium (HEU) RU-UF_6 was fed to the X-326 Process Building cascade as late as FY 1997 - FY 1998. Of the total 23.6 MTU received as non-UF_6, 5.6 MTU was converted to UF_6 in the Oxide Conversion Facility. Approximately 1.9 MTU of the 5.6 MTU of UF_6 this was used as cascade feed. The disposition of the 4.6 MTU of non-UF_6 that was potentially used for development activities is uncertain.

Shipments of RU materials from PORTS were limited to those fractions of receipts not processed (cylinder heels, rejected materials, unconverted oxides, etc.) and through March 1999 totaled 15.6 MTU. There was 8.3 MTU of RU materials remaining on site March 31, 1999 (not including wastes, sludges, etc.).

Enriched UF_6 from the PORTS site was, and continues to be, essentially free of TRU contamination. Low but detectable levels of ^{99}Tc were, and continue to be, present in the product. At times, high levels of this contaminant have required additional processing in order to produce product within specifications.

UF_6 tails from the gaseous diffusion cascade are and have been essentially free of TRU/FP.

Locations where worker exposure to TRU was most likely to occur:

1. Cascade cells near RU feed points;

2. Equipment removed from these cells during maintenance and change-out evolutions; and

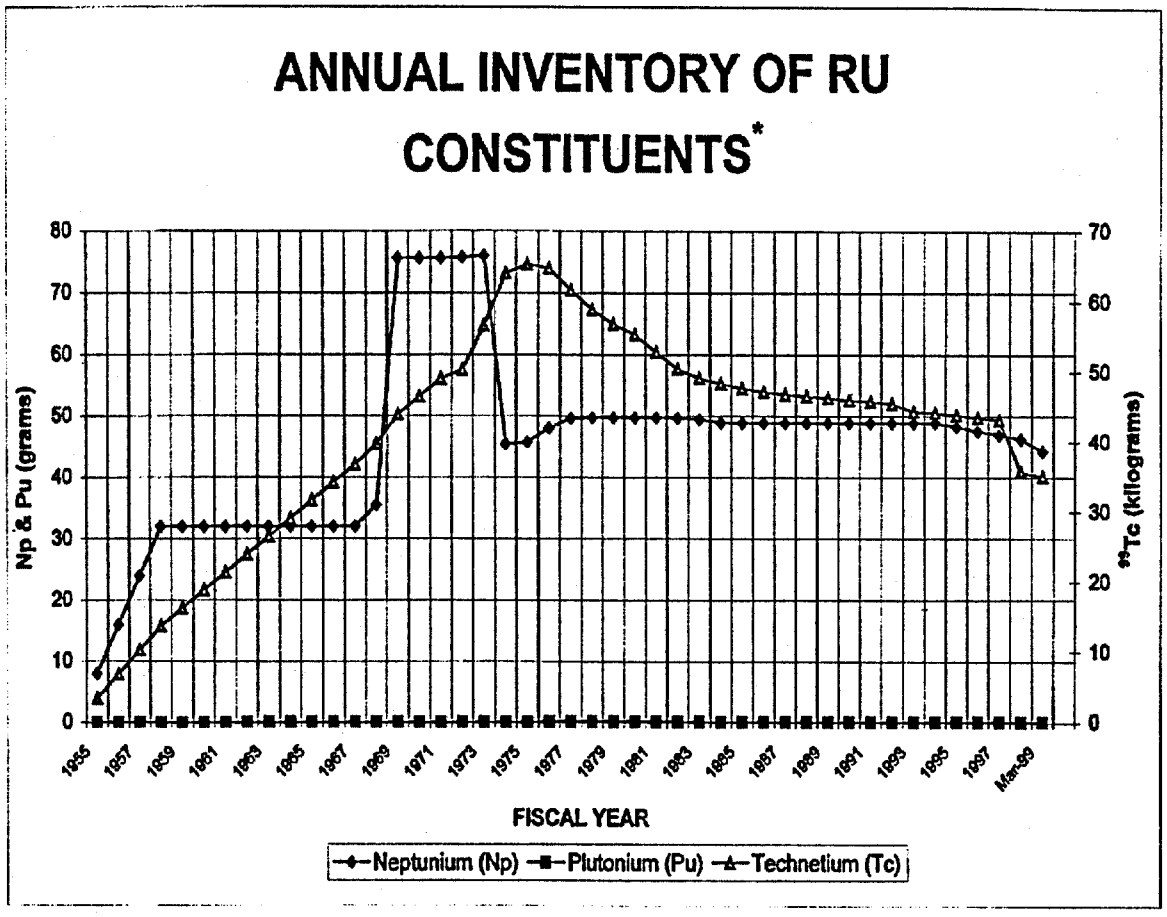
3. Oxide conversion operations – especially during evolutions involving the handling of filter ash while in operation on RU feed.

Locations where worker exposure to ^{99}Tc was most likely to occur are:

1. Top cascade cells (purge cascade);
2. Cascade vent alumina traps;
3. Magnesium fluoride (MgF_2) traps for ^{99}Tc reduction; and
4. Decontamination and Uranium Recovery Building (X-705) solution recovery raffinates and their treatment sludges.

Figure ES-1 depicts the PORTS annual inventory of TRU/FP constituents of RU.

Figure ES-1



* Does not include 4.6 MTU that was potentially utilized for development activities

Ongoing ES&H concerns from past operations dealing with TRU/FP are primarily those associated with trace quantities of TRU and ^{99}Tc. Technetium continues to be detected throughout much of the cascade (albeit at low levels) in plant process vents, effluents, and enriched product streams.

The majority of TRU constituents have been removed from the cascade with two equipment change-out programs. These constituents would also be in Low Enriched Uranium (LEU) oxides produced from the solutions used to decontaminate and clean this change-out equipment. These oxides were containerized or shipped and no longer present a significant concern at PORTS. A small fraction of these constituents remains in the process equipment that was not changed-out.

Activities during the 1990's associated with suspension of HEU production introduced 1.4 MTU of RU containing low levels of TRU into the X-326 cascade under the HEU refeed program. Constituents introduced during this program will remain until the process equipment is removed.

Since plant startup, many cases of worker exposure to and uptake of uranium are known and documented to have occurred. While no internal dose has been assigned to workers from TRU constituents of RU, it is likely that an uptake of these constituents has occurred at very low levels in the range of the limits of detection. Workers are known to have been exposed to ^{99}Tc.

1. PORTSMOUTH, OHIO RECYCLED URANIUM MASS BALANCE PROJECT

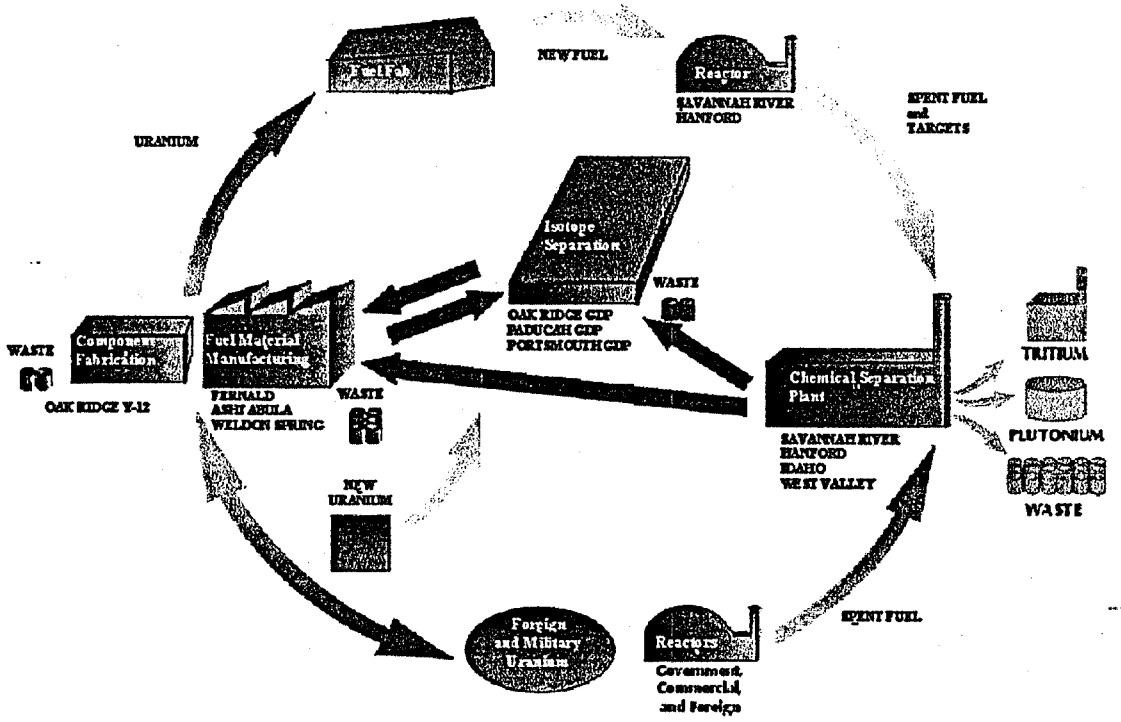
1.1 Project Overview

The gaseous diffusion plants (GDP) were an integral part of the flow path for uranium reprocessed from spent fuel from plutonium (Pu) and tritium production reactors. See Figure 1.1-1. Issues were raised surrounding this activity at the PGDP as to its potential for having affected the health of workers through exposure to the constituents of the RU. The Office of Environment, Safety and Health (EH) initiated five projects to investigate these legacy issues at the GDP's and linked plants. The second of these projects involves conducting a review of the characteristics and flow of uranium throughout the Department of Energy. This project is under the auspices of the Office of Nuclear Safety (EH-3) and is referred to as the mass balance project. It is the mass balance activity for PORTS that is addressed with this report.

The Bechtel Jacobs Company, LLC, under prime contract to DOE, was directed to prepare the PORTS site report for inclusion in the overall mass balance project report. A team consisting of seven long-term experienced current and former site and contractor employees with a cumulative service of 185 years was organized to review and research records of activities and RU operations. The team divided the site into four principal focus areas for directing its investigation. These areas represented the principle facilities and/or processes having the potential for concentrating the constituents of RU. Additionally, these facilities would present the greatest likelihood for worker interaction with uranium-bearing TRU/FP.

Figure 1.1-1

Principal Flow Streams of the Uranium Processing Cycle



These focus areas are: (1) uranium tetrafluoride (UF_4) to UF_6 feed manufacturing plant; (2) cascade and feed facilities; (3) uranium recovery; and (4) oxide conversion. Other facilities are addressed only to the extent that they were considered potential contributors to the mass balance issue. Activities at each of these facilities that involved either RU or the constituents of RU were reviewed from initial introduction of uranium during plant startup in FY 1955 through March 1999.

Four primary sources of RU were identified that bound the issue for Portsmouth. These sources were: (1) PGDP/ORGDP UF_6 manufactured from usually depleted RU-UO_3; (2) PGDP product; (3) RU-UF_6 supplied from foreign sources; and (4) RU non-UF_6 supplied from many sources in small quantities. By tracking each of these four sources with time throughout each of the facilities and summing by facility, an annual inventory of RU constituents was created.

Data sources were researched to determine quantity, source, and transaction date of all uranium, regardless of form received or shipped at PORTS. A database of this information was prepared as the initial phase of this effort. Attempts to corroborate shipments and receipts with other sites were made for the principal RU shippers and receivers.

Classification of flows as RU was accomplished based on the following criteria:

1. Analytical data supports the presence of TRU or FP

2. Materials were of the characteristic enrichment levels of RU

3. Suppliers identified the materials as RU

4. Materials were coded as RU in Source and Special (SS) Accountability Reports

For this report, RU was assumed to maintain its identity as RU until it was fed. From that point on, the constituents of the RU were tracked. This approach was adopted since small amounts of RU were intermixed with much larger amounts of non-RU yielding product and tails streams containing deminimus quantities of TRU constituents.

Processing of RU was found to have occurred in three of the four primary facilities studied. No record could be found of any RU based materials (UF_4) having ever been processed during the 46-1/2 months of X-344 feed manufacturing operation.

In the cascade, 1,094.6 MTU of RU was introduced as feed made at PGDP/ORGDP from depleted UO_3. Other foreign and domestic sources supplied 1.8 MTU of RU-UF_6. Some of this material was fed as late as FY 1998. The PORTS Oxide Conversion Facility manufactured 1.9 MTU of RU-UF_6 (manufactured by various feed sources) that was fed to the cascade. The cascade concentrated the TRU constituent, neptunium (Np) and small quantities of Pu, at or near the RU feed points.

An estimated total of 60 to 90 kg of the fission product ^{99}Tc was fed into the cascade, over the life of the plant, from low concentrations in large quantities of Paducah Product Feed (PPF). The cascade concentrates ^{99}Tc in the top purge area and in top vent stream traps. A trap that was installed to remove ^{99}Tc from the cascades' area of peak ^{99}Tc concentration, concentrated ^{99}Tc in the trap media.

In uranium recovery, 38.2 MTU was recovered in the form of triuranium octoxide U_3O_8 from all sources during the period covered by this report. This oxide contained TRU (primarily Np) that had been removed from the cascade equipment during the two major cascade equipment change-out programs. The ^{99}Tc concentrates in the sludges produced during the uranium recovery process.

In oxide conversion, 233 MTU of UF_6 was produced, over approximately 20 years of operation, from various on-site and off-site (including foreign) sources. Of this production, 5.6 MTU was identified as having been made from RU source materials. The process of oxide conversion is shown to concentrate TRU primarily in filter ash, and to a lesser degree, in tower ash and trap media.

Adding all of the quantities of TRU/FP constituents over time bounds this issue at PORTS. Less than 0.3 g of Pu were estimated as received at the plant. Approximately 0.003 g entered the process equipment. A total of approximately 140g of Np were received with about 46 g entering the process equipment. Some 60 to 90 kg of ^{99}Tc must be considered as having been processed with much of this having been removed through venting to the environment or through sludges originating from uranium recovery operations or with the product stream. The quantities of TRU/FP constituents that did not enter the process equipment remained in the cylinders.

Worker exposure analysis consisted of a review of available workplace monitoring data that included TRU results and reports which summarized workers' exposure monitoring results of the In-Vivo and urine bioassay programs.

The greatest potential for worker exposure was determined by reviewing the operations where concentrations of TRU/FP occurred.

In the cascade, worker exposures may have occurred as:

1. Maintenance evolutions, including change-out of equipment at the RU feed points; and

2. Maintenance evolutions, including changes in the top purge and ^{99}Tc collection traps.

For uranium recovery, worker exposures may have occurred in the handling of the sludges and perhaps UF_6 cylinder washing operations. For oxide conversion, worker exposures may have occurred with the handling of the filter and tower ashes.

Since plant start up, many cases of worker exposure to and uptake of uranium are known and documented to have occurred. While no internal dose has been assigned to workers from TRU constituents of RU, it is likely that an uptake of these constituents has occurred at very low levels in the range of the limits of detection. Workers are known to have been exposed to ^{99}Tc.

1.2 Purpose and Scope

The purpose of this project is to quantitatively estimate the historical mass flows and characteristics of RU within the PORTS site and between PORTS and other sites. The information generated from this project will enable the DOE to assess the potential for worker exposure and environmental contamination at PORTS resulting from the RU streams, specifically that caused by the TRU isotopes of Pu and Np and the fission product ^{99}Tc. These constituents were known to be present in trace amounts in uranium that had been recycled from DOE reactor programs and other sources. This project focuses on:

1. Identifying the mass flow of DOE RU from startup to March 31, 1999, from receipt to ultimate disposition. An intersite flow sheet was created showing how the PORTS site interfaced with other sites in the flow of RU. The flow includes all types of uranium including depleted, normal and enriched in several forms. The chemical forms specific to the site include uranium oxides (UO_3, U_3O_8) and uranium dioxide (UO_2), uranium fluorides (UF_4 and UF_6), and uranium wastes that contain amounts of uranium that would affect mass balance studies;

2. Identifying the major facilities where the various forms of RU were received, processed, or treated, thereby concentrating the various TRU and FP constituents. The processes and activities are sufficiently described, including feed and product specifications, and the uranium streams characterized as to their content of TRU and FP to permit addressing worker or public health and safety issues; and

3. Performing a site mass balance to the degree existing mass and analytical data permit.

Items specifically excluded from this study and the rationale for exclusion is as follows:

1. Radioactive sources and standards.

   These items are usually in sealed configurations or are in laboratory reagents. Their isotopic masses are accounted for under either the nuclear materials control and accountability system or the source control system. Their use is and has been controlled to assure worker safety and, as such, are not considered relevant to this study.

2. Very Highly Enriched (VHE) UF_6 shipments.

   In order to assure that this effort remains unclassified, materials in this flow were not considered. Due to the behavior of TRU/FP in the gaseous diffusion cascade, it is likely that only ^{99}Tc may have been a constituent in this stream. Due to the extremely high alpha levels of the VHE uranium, the fractional contribution of any ^{99}Tc to dose calculations would be minimal.

3. Uranium Management Center inventories.

   Materials were received after March 1999 and, therefore, are out of the project scope. These materials, however, are merely being stored with minimal worker interaction.

1.3 Project Implementation Strategy

The project goals are to:

1. Identify the mass flow of DOE RU from early production to March 31, 1999, including ultimate use or disposition;

2. Identify the characteristics and contaminants in the major uranium streams, specifically Pu, Np, and ^{99}Tc or other isotopic constituents of concern to worker or public health and safety (includes waste and scrap streams); and

3. Conduct PORTS site specific mass balances sufficiently thorough to identify significant implications for potential employee exposure to environmental contamination.

The strategy for accomplishing the PORTS mass balance project includes:

1. Utilizing existing DOE, Bechtel Jacobs Company, LLC, and United States Enrichment Corporation (USEC) protocols, procedures, and controls;

2. Obtaining and utilizing necessary "staff" specialists and support personnel through contractual means;

3. Establishing a structured approach to meeting the project goals, including the use of key assumptions;

4. Ensuring effective communication of progress, issues, and problem resolution through regular meetings with project personnel ; and

5. Coordinating with other sites and sharing of results.

This strategy is implemented through an organizational structure and a hierarchy of work elements described in Sections 1.3.1 and 1.3.2, respectively.

1.3.1 Project Organization

The mass balance project is planned and implemented through a matrix chain of responsibility and authority as shown in Figure 1.3.1-1.

The Office of Nuclear Safety within the DOE has overall responsibility for conducting this project as part of its plan to review the characteristics and flow of uranium throughout the DOE complex. The DOE headquarters team provides overall project direction and compiles the complex-wide report with assistance from the working group team leaders. Working group teams consisting primarily of DOE headquarters federal staff, are designated for each site to validate site data and results and assist in resolving any discrepancies between sites on shipper/receiver data, as well as provide assistance, as necessary, to complete the final site report. A data analysis subteam assists the DOE headquarters team in the analysis and consolidation of site report data for the final complex-wide report.

The PORTS site team is lead by a senior Bechtel Jacobs Company, LLC manager reporting to the PORTS Manager of Projects. The site team is composed of a subcontractor-based technical staff possessing over 185 years of combined experience in the maintenance, operations, engineering, analytical laboratory, and health and safety areas of the Portsmouth site. Support services, covering primarily nuclear materials control and accountability and records management, are provided, as needed, by the USEC. The site team is responsible for obtaining and summarizing site-specific RU data over the site's history, including mass flow and balance, TRU/FP constituent data, site inventory as of March 31, 1999, and for identifying major facilities/processes contributing to potential worker exposure.

1.3.2 Project Work Plan

Appendix B of the DOE Project Plan (Ref. 1) provides the foundation for the site's review of historical documents, extraction and evaluation of relevant data and preparation of this report. The framework to accomplish this project is shown in Figure 1.3.2-1. Data collection and evaluation were based upon the following key project assumptions:

1. Materials are classified as RU if:
   a. Analytical data showed a positive indication for TRU;
   b. Identified as such by the shipper;
   c. Supplied as UF_4 or UF_6 at the characteristic RU enrichment (~0.63-0.68% ^{235}U);
   d. Coded as RU in the source and special accountability records.
2. Materials that contained the fission product ^{99}Tc without significant accompanying quantities of TRU are not considered as RU (i.e., Paducah Product Feed);
3. Once RU materials enter into the cascade, or other process they are considered to have lost their RU identity;
4. Quantification of TRU constituents in UF_6 cylinders from PGDP and ORGDP and their subsequent feed rates are assumed to be in accordance with the Historical Impact of Reactor Tails on the Paducah Cascade (Ref 2, and 2a);
5. TRU/FP behavioral assumptions in the PORTS diffusion cascade are as follows:
   a. The minute amount (if any) of Pu that enters the cascade lodges in the immediate vicinity of the feed point;

PORTS MASS BALANCE PROJECT
ORGANIZATIONAL STRUCTURE

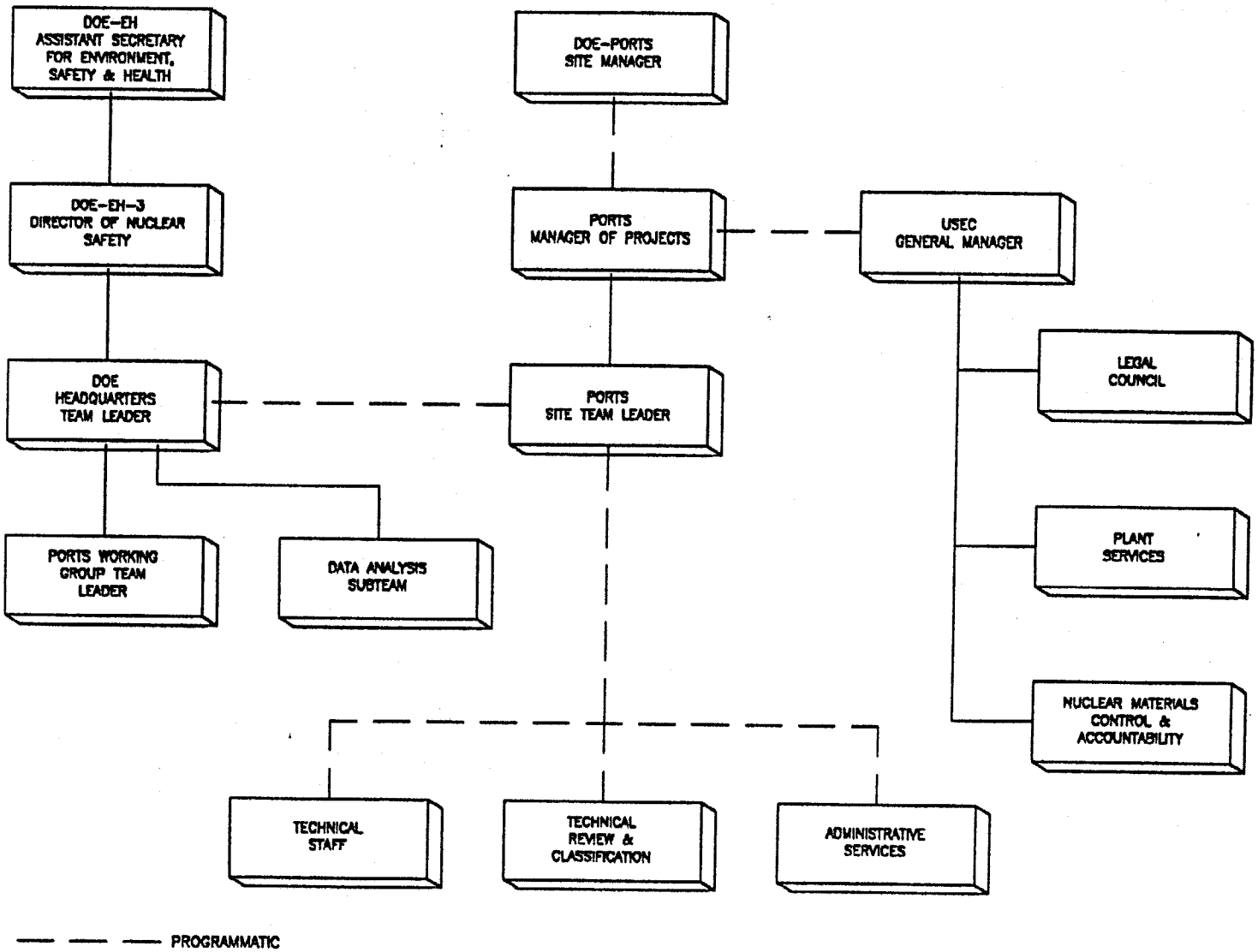


Figure 1.3.1-1

PORTS MASS BALANCE PROJECT
WORK BREAKDOWN STRUCTURE

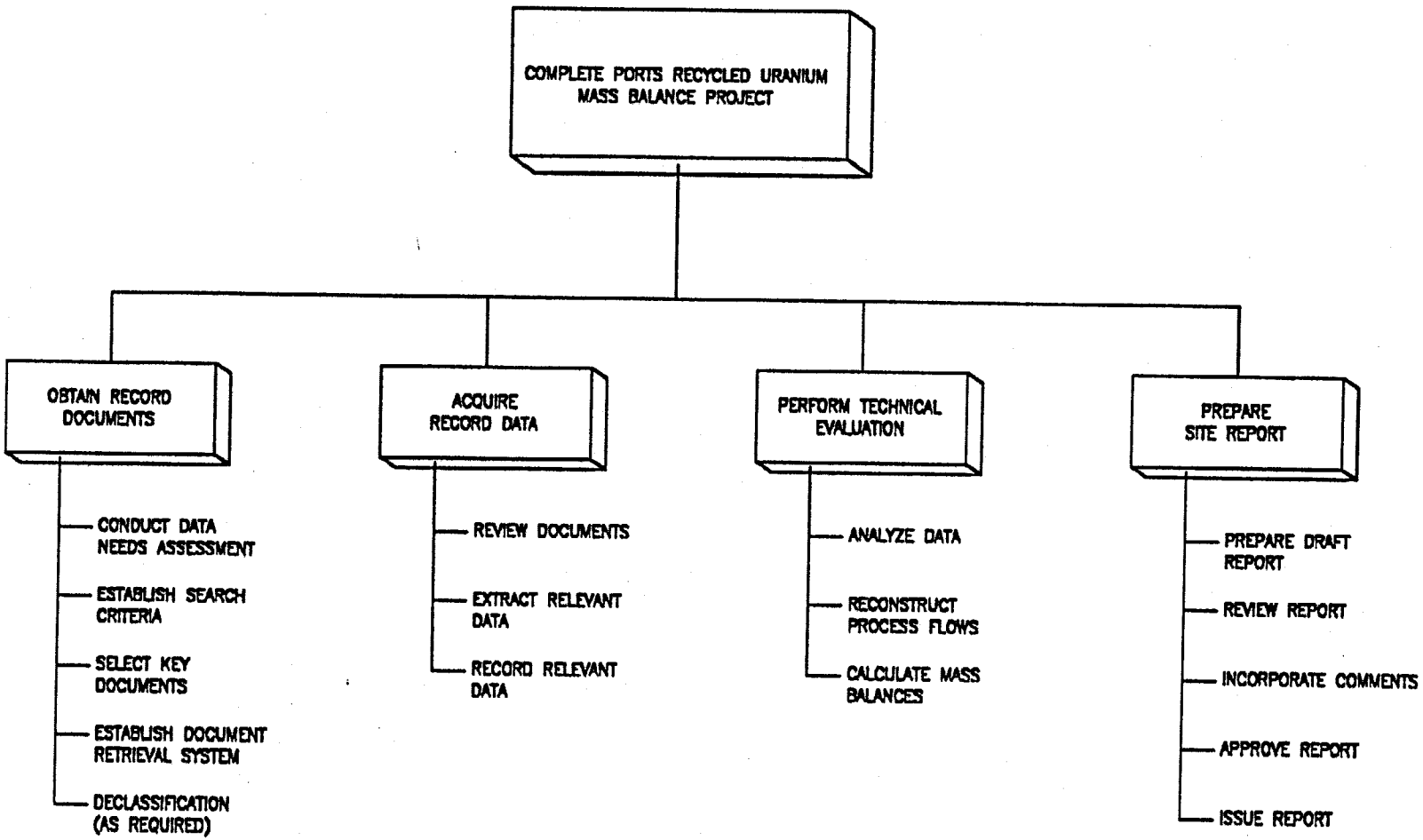


Figure 1.3.2-1

b. Np entering the cascade becomes lodged on barrier and unplated surfaces close to the feed point until its removal during an equipment change-out; and

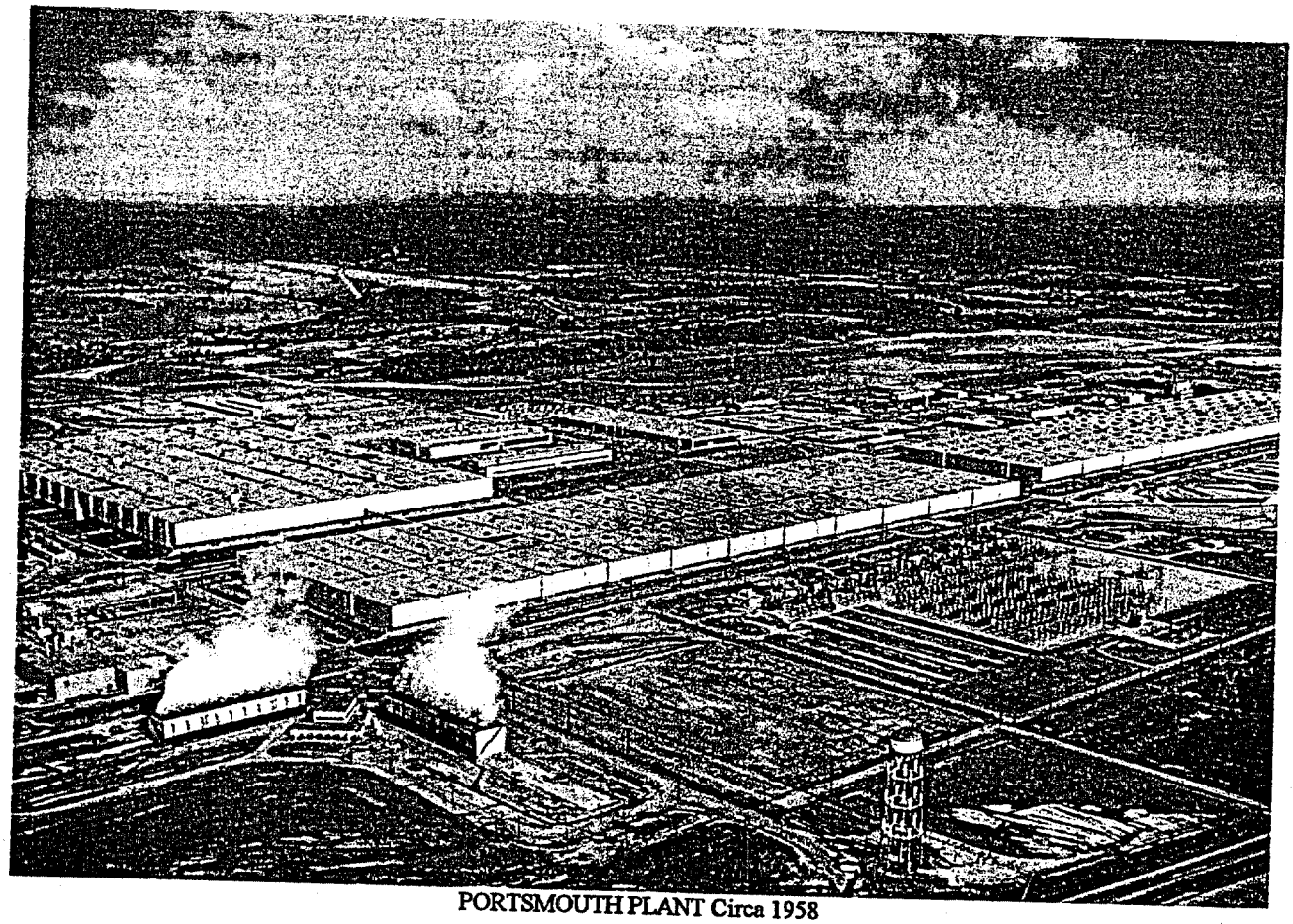
c. ^{99}Tc, due to its strong affinity for metallic surfaces, starts plating out at the feed point and then progresses up stream. The ^{99}Tc continues sorbing on metallic surfaces in successively higher cascade cells until the entire cascade above the feed point reaches equilibrium. The ^{99}Tc will then migrate to the top of the plant due to its lighter molecular weight.

6. A process is considered to have the ability to concentrate TRU/FP if it increases the mass of TRU/FP constituents relative to uranium or decreases the uranium mass relative to the amount of TRU/FP.

2. SITE HISTORICAL OVERVIEW

2.1 Site Description

The PORTS facility is one of two operating uranium enrichment production facilities in the United States. Both facilities are owned by the DOE. The other operating facility is located in Paducah, Kentucky. (A third uranium enrichment facility, in Oak Ridge, Tennessee, was placed in standby in 1985 and shut down in 1987). Each facility utilizes the gaseous diffusion process to enrich uranium from a natural state of less than 1% ^{235}U to increased concentrations varying from 2 to 5% ^{235}U for use as fuel for nuclear power generation. The Paducah facility presently enriches uranium to approximately 2.5% ^{235}U and then ships it to PORTS for further enrichment. PORTS had the capability of achieving a higher percentage of enrichment; however, highly enriched uranium operations were shut down beginning in 1991.



PORTSMOUTH PLANT Circa 1958

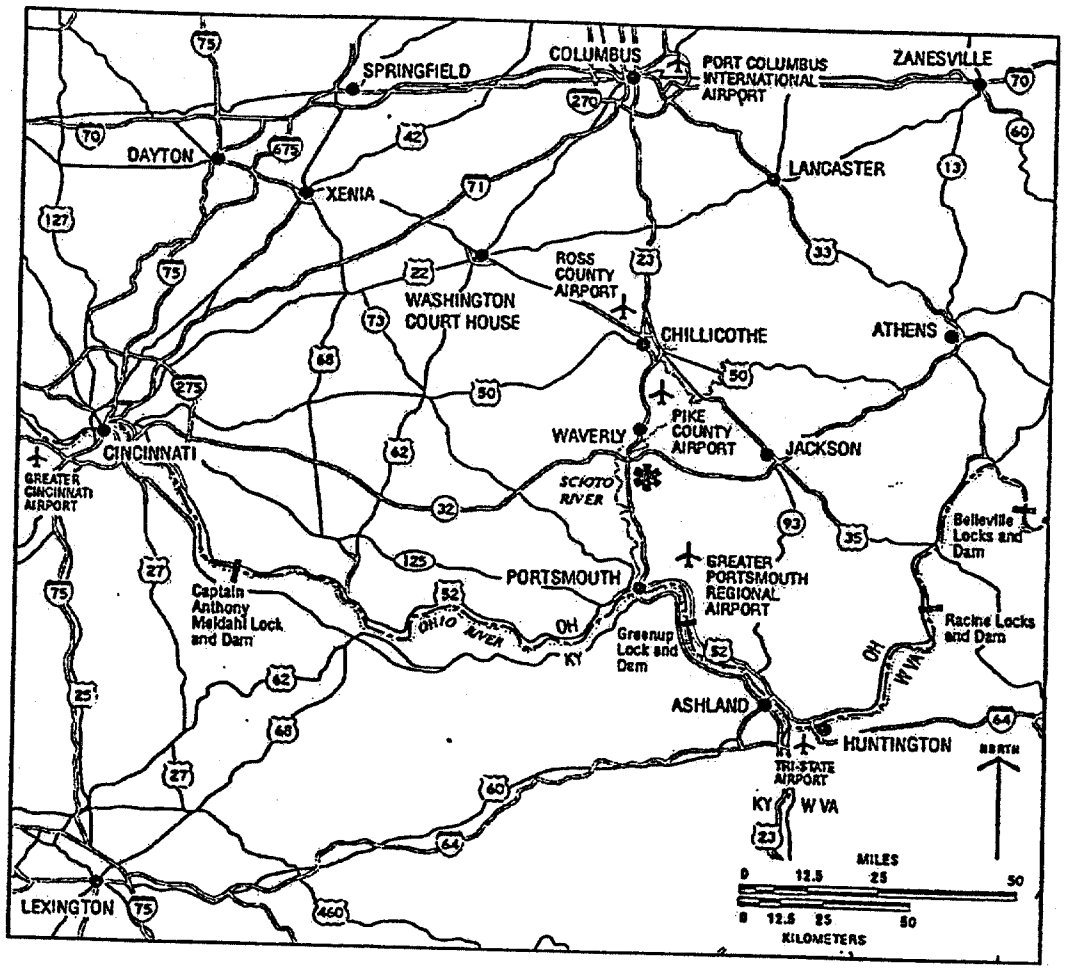
In June 1951, the Atomic Energy Commission (AEC) began design studies that would lead to the construction of a gaseous diffusion plant that could be added to the Oak Ridge—Paducah complex and provide ^{235}U production at rates substantially above those of the existing complex. The AEC began site selection for a new gaseous diffusion plant and selected PORTS in 1952.

PORTS consists of 109 buildings occupying 500 acres located on a 3708 acre DOE-owned reservation in Pike County in south central Ohio about two miles east of the Scioto River and 18 miles north of Portsmouth, Ohio (refer to regional location, Figure 2.1-1). Construction of the plant began in late 1952 and production of enriched uranium began in early FY 1955, one year before completion of construction. In the late 1970's, PORTS was chosen as the site for construction of a new uranium enrichment facility utilizing gas centrifuge technology. Construction of the Gas Centrifuge Enrichment Plant (GCEP) began in 1979 in an area southwest of and adjacent to the existing gaseous diffusion plant. Construction of this facility was halted in the summer of 1985.

With the Energy Policy Act of 1992, the responsibility for the operation of the gaseous diffusion plant at PORTS transferred to the newly created USEC effective July 1, 1993. With this transfer of responsibility, DOE leased to USEC property shown in Portsmouth Gaseous Diffusion Plant Building Lease Status, Figure 2.1-2. Although the USEC has managed the gaseous diffusion operations at PORTS since July 1, 1993, the DOE continues to have a significant presence, particularly in the area of environmental restoration and the responsibility for treating and disposing of wastes resulting from GDP operations prior to July 1, 1993.

Figure 2.1-1

Regional Location of PORTS



LEGEND

❋ PORTSMOUTH GASEOUS DIFFUSION PLANT

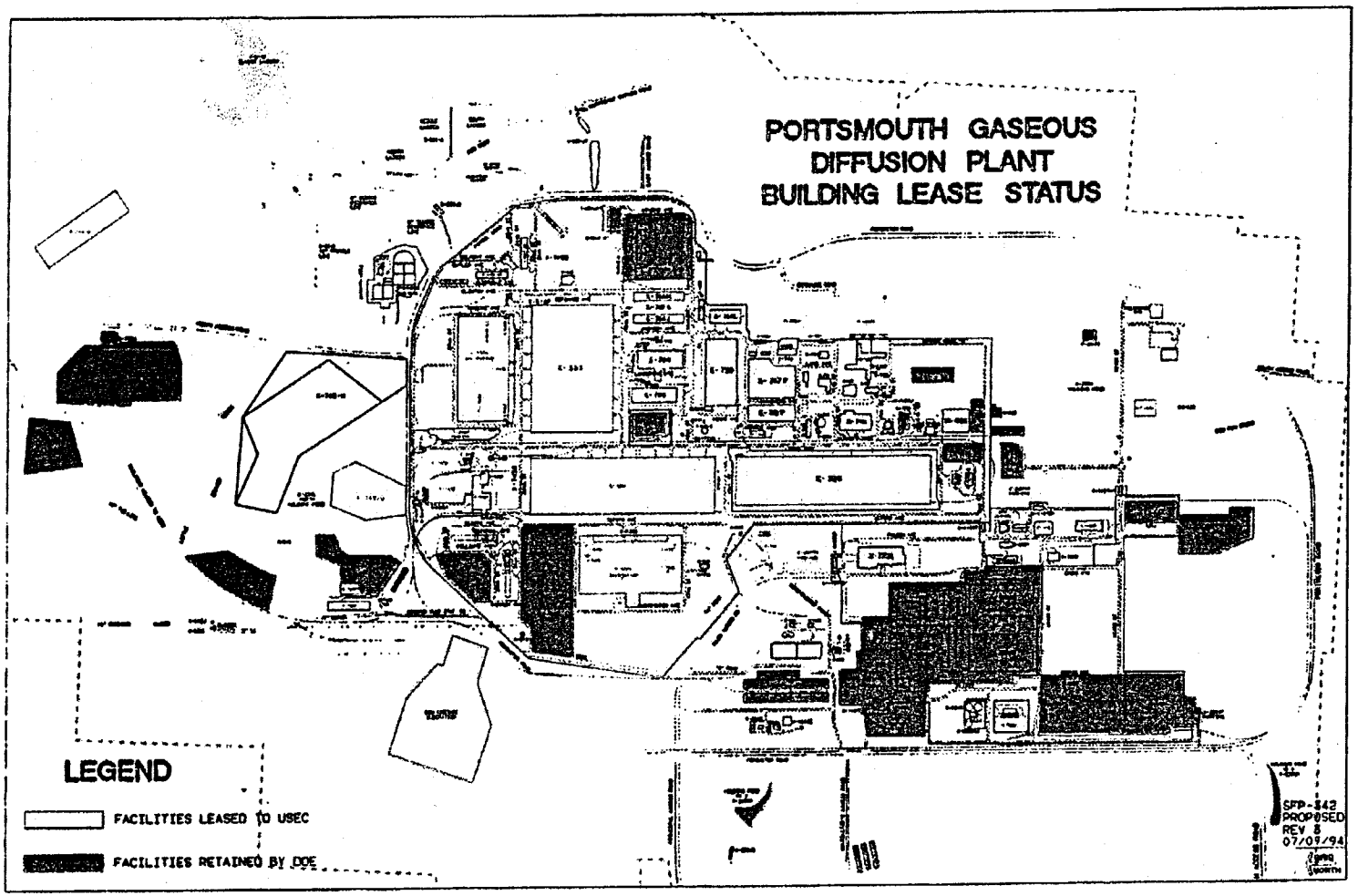

PORTSMOUTH GASEOUS
DIFFUSION PLANT
BUILDING LEASE STATUS
LEGEND
FACILITIES LEASED TO USEC
FACILITIES RETAINED BY DOE
SFP-342
PROPOSED
REV 8
07/09/94

Figure 2.1-2

The gaseous diffusion enrichment process takes place in three large process buildings; X-333, X-330, and X-326. Process buildings X-333 and X-330 were built in 1955 and used for the initial and intermediate phases, respectively, of uranium enrichment. The process building (X-326), which was built in 1956, was originally used for the high enrichment phase, but is no longer used for this purpose. The X-326 is currently used for product withdrawal and side feeding. In addition, from early 1997 to mid 1998 the X-326 product withdrawal equipment was used for HEU blending activities.

Various UF_6 feed, withdrawal, and sampling systems and UF_6 cylinder operations are located in the three process buildings, as well as the X-342A Feed Vaporization and Fluorine Generation Facility, X-343 Feed Vaporization and Sampling Facility, and the X-344A Toll Enrichment Services Facility.

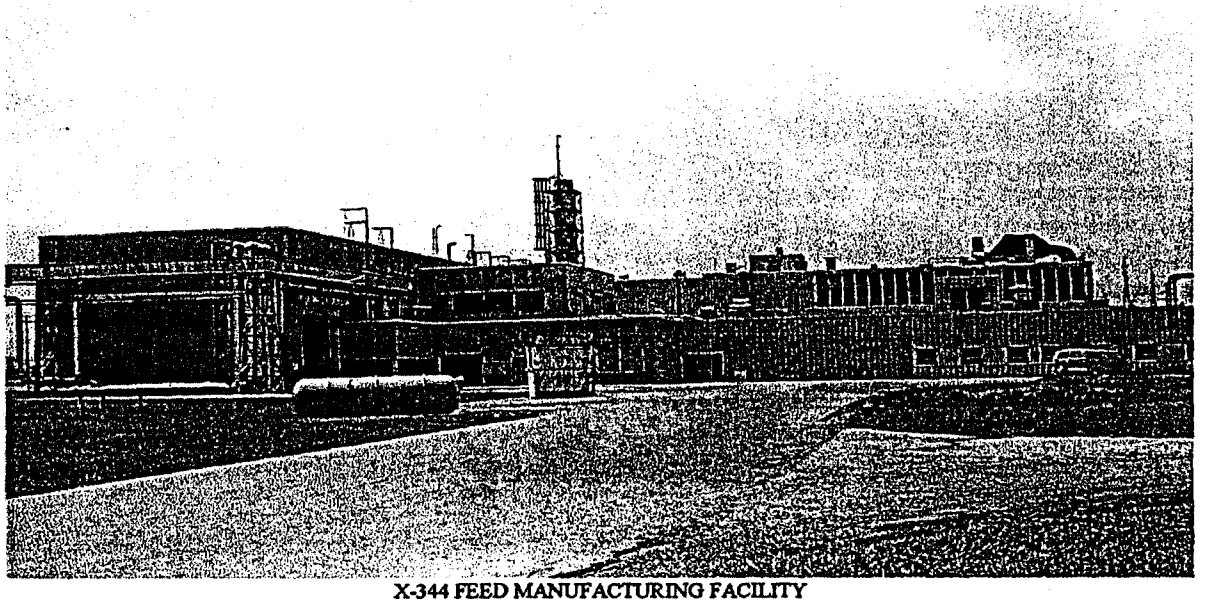
Three large facilities, the X-700 Converter Shop and Cleaning Building, X-705 Decontamination and Recovery Facility, and the X-720 Maintenance and Stores Building provide most of the equipment maintenance support for the diffusion cascade. Equipment removed from the cascade is disassembled and decontaminated in X-705, which also houses equipment/systems for the recovery of uranium from decontamination solutions.

2.2 Key Uranium Processing Facilities

Four major facilities were suspected of containing processes where RU constituents may be concentrated and, thereby, present the potential for worker exposure or environmental contamination. These facilities are listed below and their description and processes are described in subsequent sections:

1. X-344 Feed Manufacturing Plant;
2. Cascade (X-333, X-330, and X-326) and associated feed, withdrawal and sampling facilities;
3. X-705 Decontamination and Recovery Facility; and
4. X-705 Oxide Conversion Facility.

2.2.1 Feed Manufacturing Plant



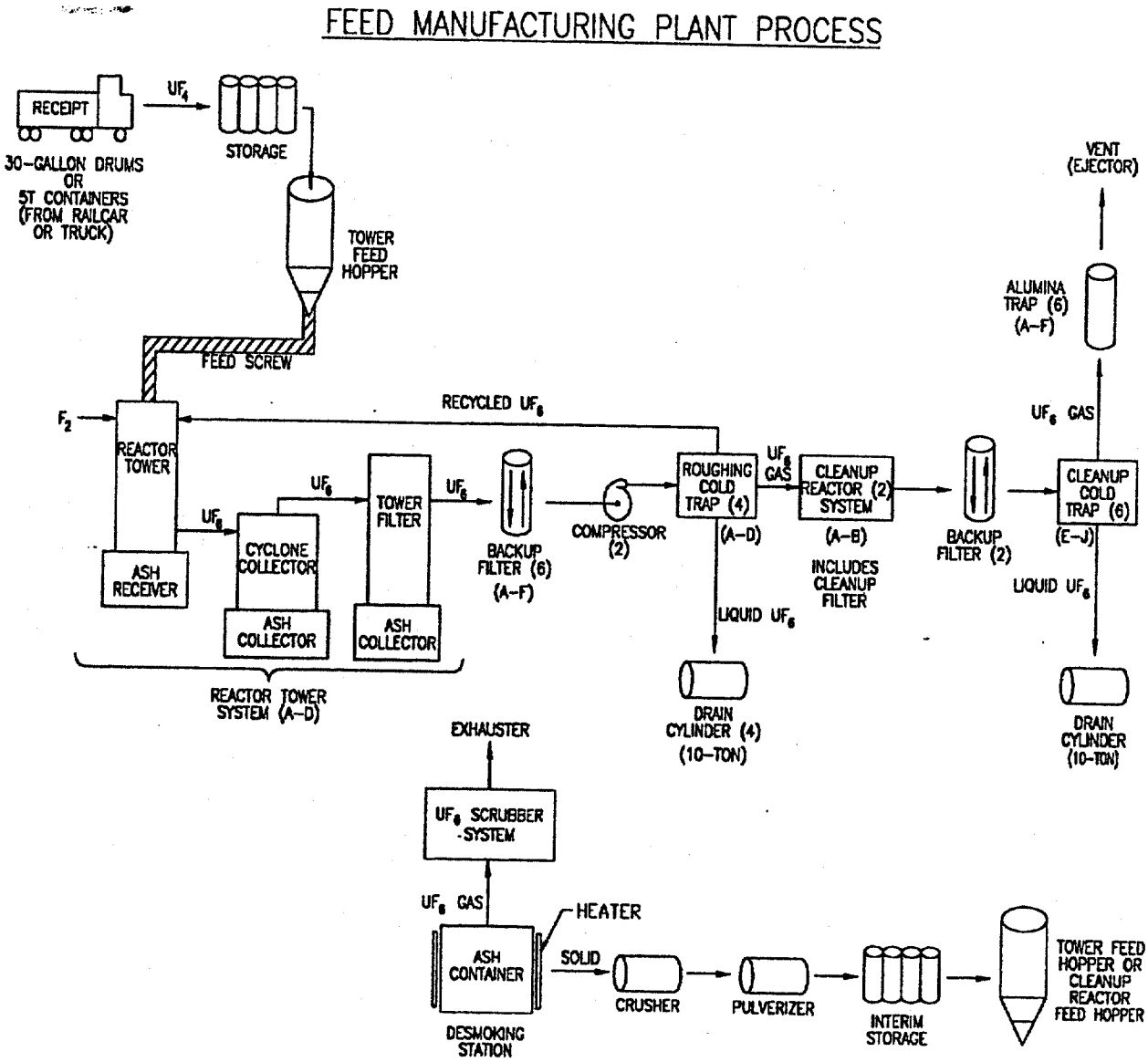
X-344 FEED MANUFACTURING FACILITY

2.2.1.1 Plant Description

The X-344 Feed Manufacturing Plant was located largely in what is now the X-342A Feed Vaporization and Fluorine Generation Building and X-344A Toll Enrichment Services Facilities, which are located north of the X-745B Toll Enrichment Process Gas Yard in the north-central region of the PORTS site. The feed plant was constructed and operated in the early days for the production of UF_6 from UF_4 green salt. The plant consisted of: 1) a building to house the process equipment, offices, and maintenance areas; 2) an auxiliary building for ash storage and acid neutralization; 3) an acid unloading and storage facility; and 4) modifications to the existing X-342A.

The process for producing UF_6 was by direct fluorination of UF_4 in a tower reactor as shown in Figure 2.2.1.1-1. The product UF_6 was piped to refrigerated cold traps where it was separated from other gases by condensation. Subsequently, the cold traps were isolated, heated, and the UF_6 drained as a liquid into 10-ton cylinders for eventual vaporization into enrichment cascades.

Figure 2.2.1.1-1



The large quantities of fluorine required for the process were produced by electrolysis of hydrogen fluoride (HF) in fluorine cells. The HF was stored in three 12,500 gallon storage tanks located within a concrete diked area protected by a Butler-type HF storage building (X-344C) located next to a spur track northeast of the X-344 building. Pumps were provided to transfer HF from the storage tanks to vaporizers located in the X-344 and then to the fluorine cells. A total of 40 cells were available for fluorine production. Equipment was provided to filter the fluorine and remove any residual electrolyte, and heat it prior to being metered into the tower reactor system(s).

Handling equipment was provided for receiving green salt powder in 30-gallon drums or five-ton containers in railroad cars or trucks. Facilities were provided for washing and drying of empty 30-gallon drums. A means was provided for adding one 30-gallon drum of ash to each five-ton container prior to its rotation and positioning over the tower feed hoppers.

The green salt fluorination system consisted of four tower reactors and the fluorine cleanup reactors. Each reactor system included a fluorine preheater, feed hopper, feed screw, reactor tower, cyclone barrier filter, and ash receiver similar in design to that used in the Oak Ridge feed plant. Following the reactor tower systems, the UF_6 gas was collected and piped to a backup filter station prior to being compressed and passed through one of four roughing cold traps. A portion of the gas from the cold traps was recycled back to the reactor towers and a portion was piped to a cleanup reactor system to remove any residual fluorine. The gas from the cleanup reactors was passed through one of the six cleanup cold traps. Any uncondensed UF_6 in the gas stream from the cleanup cold traps was removed in the alumina traps prior to venting to the environment.

Four drain positions were equipped for collecting the liquid UF_6 drained from the cold traps. Each position was designed to handle a 10-ton cylinder.

Ash grinding and storage facilities were provided for storing and processing unreacted green salt collected in the ash receivers under the towers, filters, and cyclones. These facilities were equipped for heating, crushing, pulverizing, and containerizing the ash for blending with fresh green salt for use in the tower reactors.

A system similar to that used in the Paducah feed plant was provided for neutralizing waste acid from the water scrubbing system before the water was passed to the sewer. This system provided for the mixing of lime with the acid solution.

Maintenance areas were provided for dismantling and repairing fluorine cells and other process equipment.

2.2.1.2 Material Flowsheet

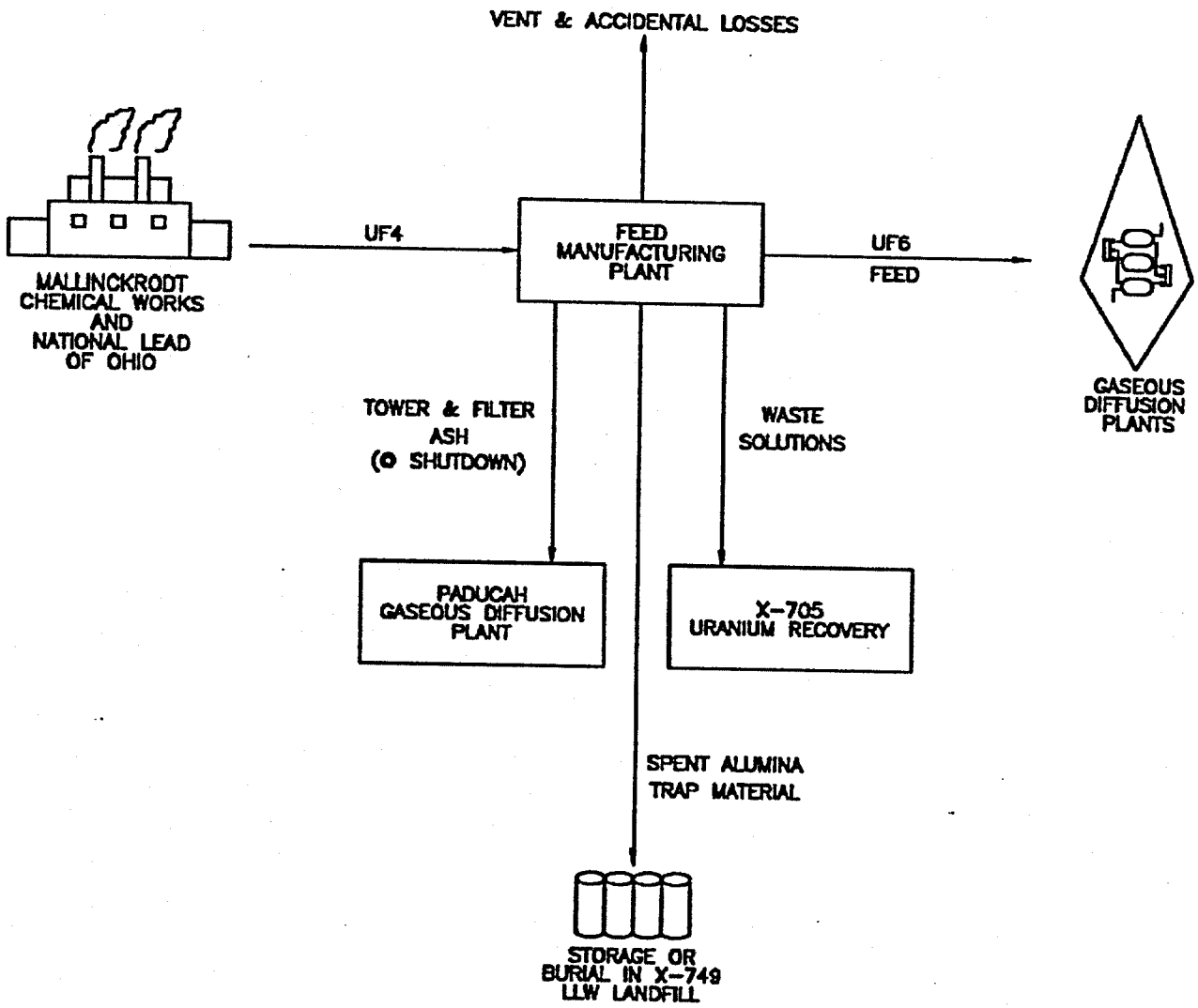
The materials flowsheet for feed manufacturing is shown in Figure 2.2.1.2-1.

A review of plant records (Ref. 3 and 4) indicates that all UF_4 fed into the feed manufacturing plant during its history was of normal enrichment from two sources; namely Mallinckrodt Chemical Works (MCW) and National Lead of Ohio (NLO). No record can be found of UF_4 produced from RU forms ever having been fed in the feed manufacturing plant. The UF_6 product was stored and either fed to the PORTS enrichment cascade or shipped to one of the other enrichment cascades. The reaction of UF_4 and fluorine in the tower reactors did not consume all of the UF_4 and some of it was caught in the ash receivers under the towers, filters and cyclones. The ash, which contained useable uranium, was recycled though the tower after further processing. The ash receivers were stored for approximately two months to allow the radiation level to drop to the point where the ash could be handled. Following the storage period, the ash was heated and the absorbed UF_6 driven off and recovered in a water scrubber system. The UF_6 free ash was then crushed, screened, pulverized and containerized for future blending with fresh green salt for use in the tower reactors. The only time ash was removed from the plant was after shutdown when it was pulverized, containerized and shipped to Paducah.

The waste acid from the water scrubbing system was neutralized with lime and permitted to pass into the sewer or containerized and transferred to the X-705 for processing.

Figure 2.2.1.2-1

FEED MANUFACTURING PLANT MATERIAL FLOWSHEET



The spent alumina from the chemical traps was containerized and either stored in the X-744G Bulk Storage Facility or buried in the X-749 Low Level Radioactive Waste Burial Ground.

2.2.1.3 Operating History

The Feed Manufacturing Plant was turned over to Goodyear Atomic Corporation (GAT) on April 25, 1958. Initial testing of the plant was performed using a special allotment of UF_4 shipped from Mallinckrodt Chemical Works for the production of UF_6 standards. The initial testing was completed on May 14, 1958 and the plant started production operations on May 15, 1958. The plant continued operations until February 1962 at which time it was shut down.

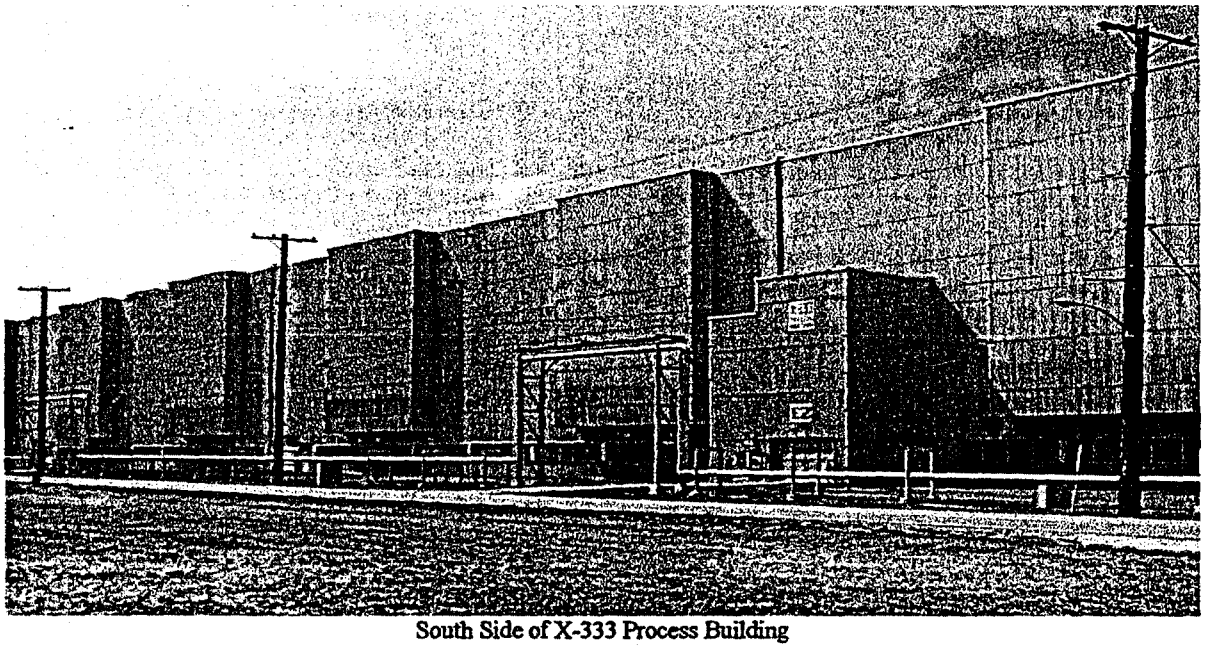
During its 46-1/2 month operational life 11,983 MTU of UF_4 was fed, at an average rate of 9.5 MTU per day, producing a total of 11,890 MTU of UF_6 (see Appendix I). This operation experienced numerous operating and maintenance problems resulting in significant radionuclide emissions to the atmosphere as well as contributing to its shutdown. The plant lost an average of 407 kgU per year to the atmosphere from 1959 until its shutdown in 1962, (Ref. 5).

At shutdown all material inventory was removed from the system, containerized and moved to storage for final disposition. About 23.3 MTU as ash was removed, pulverized containerized and shipped to Paducah for processing.

2.2.1.4 Current Status

After shutdown in 1962 and material removal, the process equipment was dismantled. In the early 1970's, new plans for the X-344 facility were prepared and the building was converted by 1975 to serve as the shipping/receiving point for low-assay UF_6 (less than 5%). Currently the facility is used for sampling 10-ton product cylinders and transferring product into smaller (2-1/2 ton) customer-owned cylinders for shipment. Use of all but four fluorine generation cells was discontinued with the remaining fluorine cells used for maintenance spares.

2.2.2 Cascade and Feed Facilities



South Side of X-333 Process Building

2.2.2.1 Description

The PORTS cascade is comprised of 4080 stages of process equipment and is capable of enriching uranium to 97% ^{235}U assay. Table 2.2.2.1-1 summarizes the stage arrangements per building and itemizes the number of stages of the various size equipment. The 33-size equipment located in the X-333 Process Building is the largest. Figure 2.2.2.1-1 shows a typical cell flow diagram for X-33-size equipment. The 25-size equipment located in the X-326 Process Building is the smallest. The cascade originally was fed from the X-342 building which housed 12 steam vaporizer bays used to heat 2-1/2 and 10-ton UF_6 cylinders. Two product withdrawal facilities located in the X-326 building, one being for Very Highly Enriched (VHE) product and one for Extended Range Product (ERP) for lower assays and one tails (depleted stream) withdrawal facility located in the X-330 building were part of the original plant design. A purge facility used to vent light gases is also located in the top (i.e., near VHE product areas) cascade of the X-326 building.

Table 2.2.2.1-1

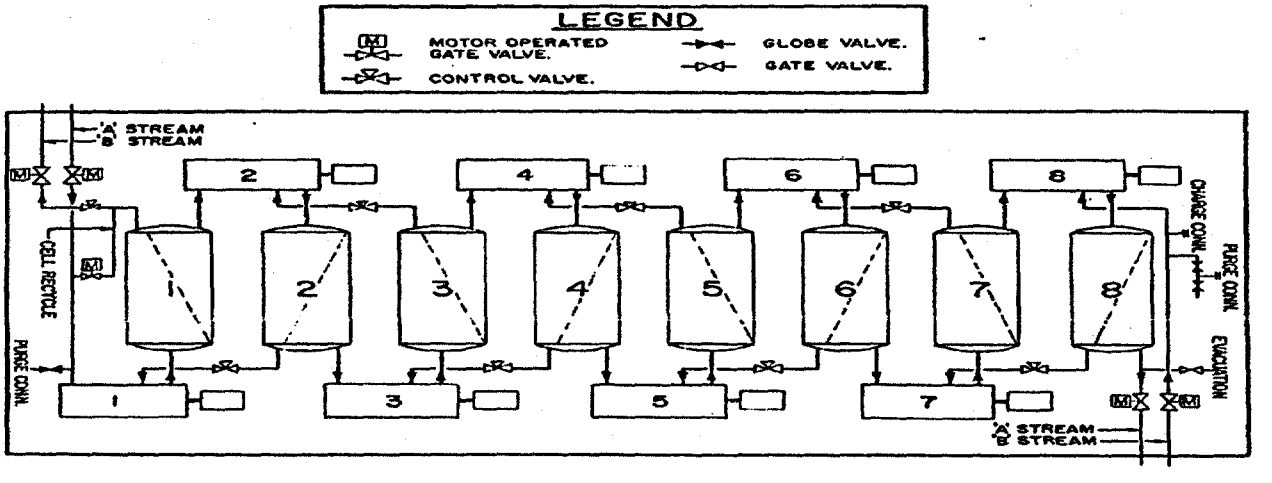
Cascade Configurations

Process Building	Equipment Size	Number of Stages
X-333	33 or (000)	640
X-330	31 or (00) 29 or (0)	500 600
X-326	27 25	720 1620
TOTAL		4080

Through the years, PORTS' mission evolved from high assay production for military uses to providing low enrichment services for fuel to be used in commercial nuclear power plants. Additional facilities have been built and existing facilities have been modified to reliably support the new mission. A new feed facility, the Feed

Figure 2.2.2.1-1

Cell Flow Diagram for X-33 Units



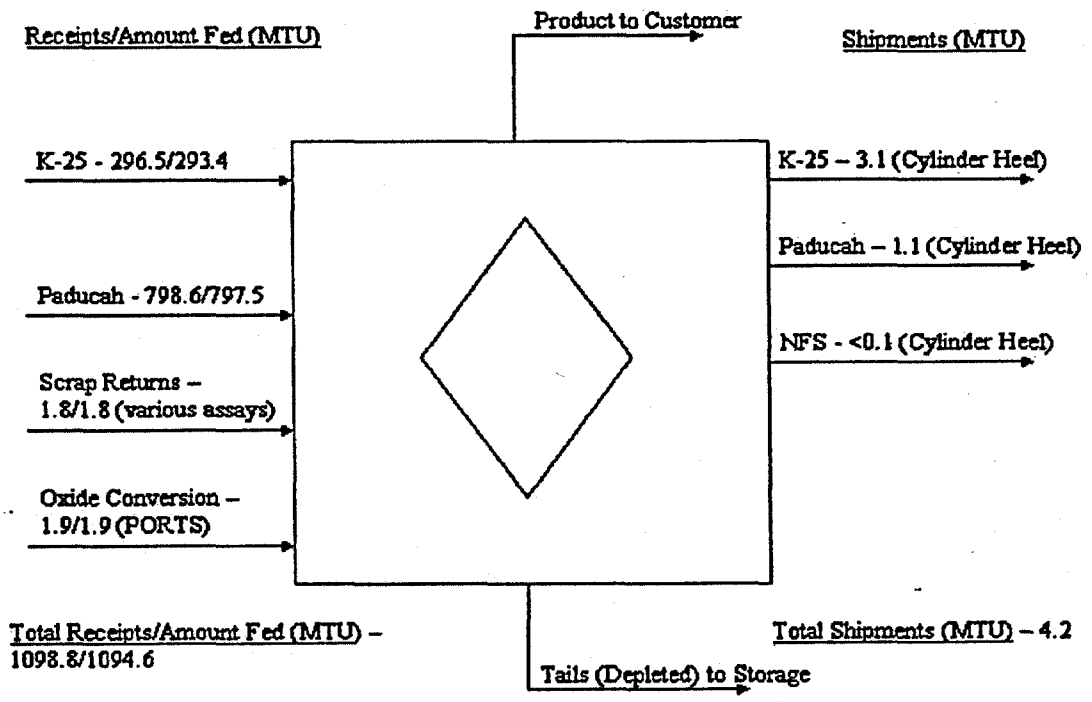
Vaporization and Sampling Building (X-343), was built and placed on line in 1983 and houses seven autoclaves. The 12 steam chests in X-342 were removed and two new autoclaves were installed to feed the cascade. An additional Low Assay Withdrawal (LAW) facility was installed in the X-333 building and modifications were made to enable product withdrawal at the tails withdrawal area, as well as tails withdrawal at LAW. The X-344 facility was modified into a toll enriching facility where product transfers from 10-ton processing cylinders to 2-1/2 ton customer cylinders with associated sampling is accomplished and cylinders are loaded onto trucks/railcars and shipped off-site.

2.2.2.2 Material Flowsheet

See Material Flowsheet – Cascade – RU-UF_6 , Figure 2.2.2.2-1

Figure 2.2.2.2-1

Material Flowsheet – Cascade RU-UF_6



2.2.2.3 Feed Specifications

1. Feed <5% - American Society for Testing Materials (ASTM) specification C-787-96 (or previous revisions) for natural uranium and UF_6 that has been received from irradiated uranium which has been reprocessed and converted to UF_6 (see Ref. 6 and 7).

   - Primary specification item – total alpha activity from Np and Pu to be limited to 1,500 disintegrations per minute/gram uranium (dpm/gU).

2. Feed >5% assay – scrap return program beginning in 1968 that had reactor return constituents was under a 1961 specification stating that total activity level to be less than 1,500 dpm/gU. Although these specifications were reissued in April 1971 by the AEC which raised the alpha activity level to 15,000 dpm/gU, PORTS continued to use 1,500 dpm/gU limit for scrap acceptance except for approved deviations on the NLO material received at 3,000 dpm/gU.

3. Shipments between the diffusion plants are exempt from ASTM specification criteria, although no known receipts or shipments have been identified that exceed the total of 1,500 dpm/gU limit.

2.2.2.4 Product Specifications

1. Product <5% - DOE memo dated January 22, 1993 and ASTM specification C-996-96 (or previous revision) (see Ref. 7 and 8).

2. Product ≥5% - military uses exempt from ASTM specifications. Naval and weapons program specifications were used.

2.2.2.5 Operating History

Startup of the PORTS cascade began early in FY 1955. Appendices II and III summarize the total uranium and RU received and shipped from PORTS (all forms) from startup to March 31, 1999. Appendix IV shows all the UF_6 fed to the cascade and its source, including reactor returns on an annual basis over the same 44-year timeframe. Reactor returns were fed to the cascade in timeframes shown in Table 2.2.2.5-1.

Table 2.2.2.5-1

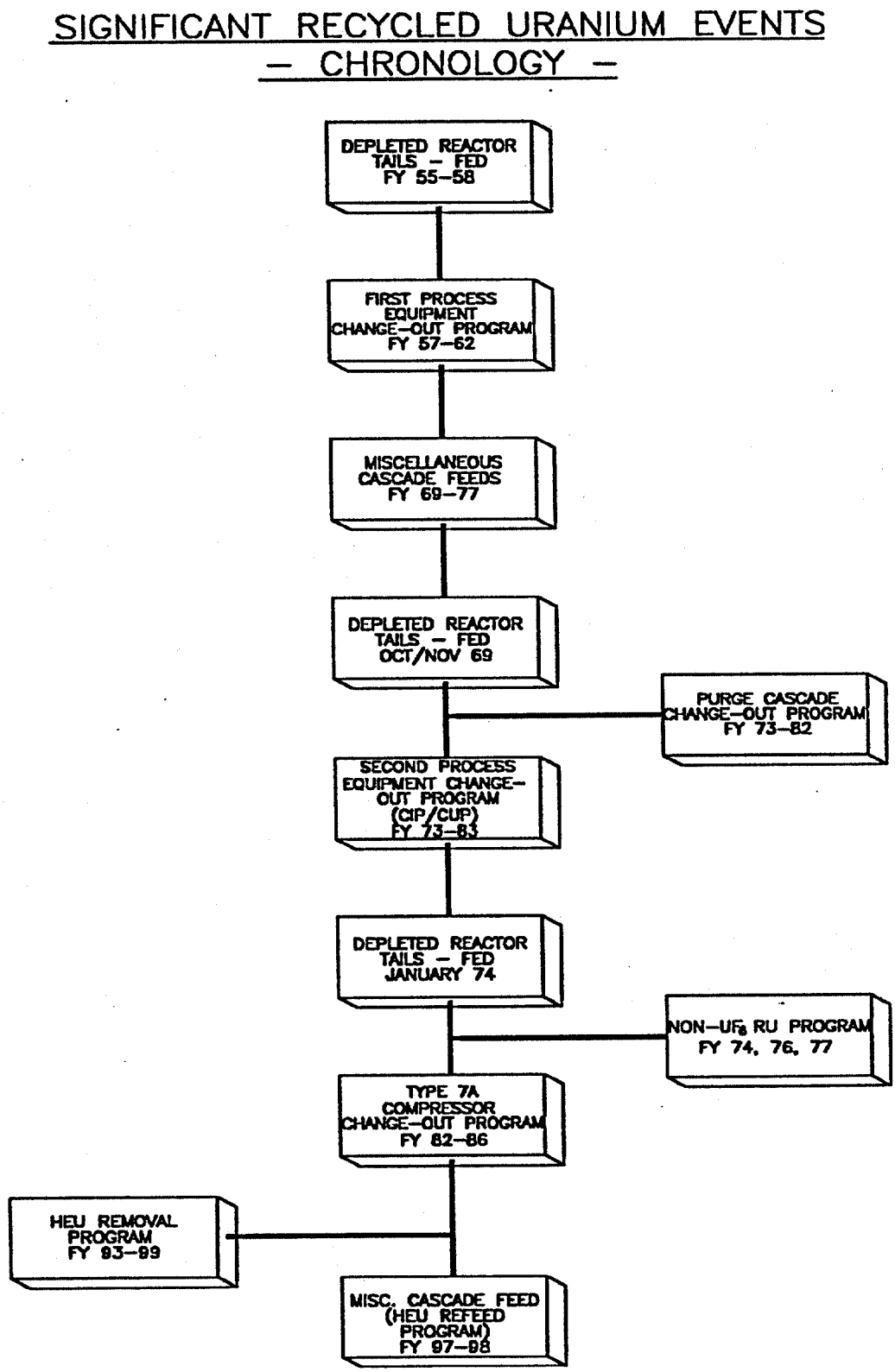
REACTOR RETURN CASCADE FEEDS (UF_6)

Fiscal Year	Amount Fed MTU	Assay % U^{235}	Source	Remarks
1955	105.8	0.64-0.68	Paducah	Fed May – Sept. 1955
1956	54.5 293.4	0.64-0.68 0.64-0.68	Paducah Oak Ridge	
1957	6.2	0.64-0.68	Paducah	
1958	64.2	0.64-0.68	Paducah	
1970	168.1	0.64-0.68	Paducah	Fed Oct. & Nov. 1969
1974	398.8	0.64-0.68	Paducah	Fed Jan. 1974
1968-1977*	0.15	78-80	Division of International Affairs	
1977-1998*	0.15	78-97	Babcock & Wilcox	
1969-1993*	0.07	78	USAEC Office of Safeguards & Materials Management	
1997-1998	1.10 0.33	56-82 .80	France NUMEC	
1974-1978*	1.86	2-50	PORTS Oxide Conversion	

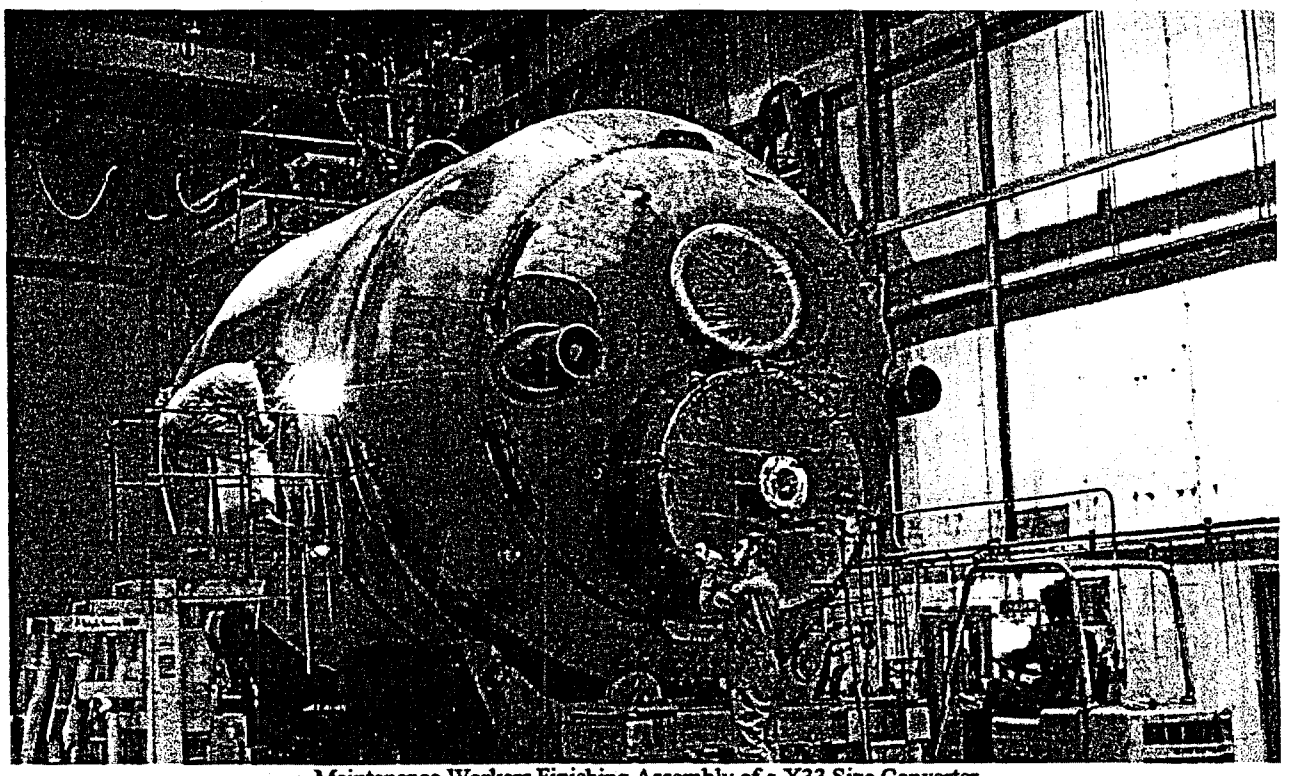
*Streams will require additional research to pinpoint the feed date if deemed necessary.

The operating history will be discussed with focus on significant RU events shown in Figure 2.2.2.5-1. These same events will be portrayed as campaigns in Section 5.

Figure 2.2.2.5-1



During plant startup, some RU that had been converted to UF_6 at Paducah and Oak Ridge was fed to the cascade. Five hundred twenty-seven (527) MTU RU of a total feed of 33,356 MTU (a total of about 1.6%) was fed during the FY 1955 - FY 1958 timeframe. Feed cylinder numbers, assay, date fed, and cascade feed points have been located, analyzed and tabulated (see Appendix V through VII). The Oak Ridge feed (296.5 MTU) came in 2-1/2 ton cylinders, whereas the Paducah material (230.6 MTU) came in both 2-1/2 ton and 10-ton cylinders. It appears that some of the 200 2-1/2 ton cylinders used were cycled between the plants with RU materials two to four times, thereby concentrating the RU in the cylinder heels.



Maintenance Workers Finishing Assembly of a X33 Size Converter

Following this period of RU feeds, the plants' first equipment change-out program commenced. This converter barrier replacement program ran from FY 1957-FY 1962 during which time 560 stages of X-33-size, 500 stages of X-31 size, and 280 stages of X-29 size converter barrier were replaced. The amount of TRU in the RU that could have been removed from the cascade and the process equipment during decontamination in this timeframe is based upon the concentration data provided in Refs. 2 and 2a, and for this period is calculated as 32 g Np and 2.1 mg Pu.

Reactor returns were again received from Paducah in FY 1968 – FY 1969 (568 MTU). One hundred sixty-eight (168) MTU was fed to the cascade in October and November 1969 (FY 1970), and 400 MTU (which is the largest amount fed at PORTS in any one year, as well as comprising 35% of all RU fed to the plant to date) was fed in January 1974. A manual UF_6 product cylinder history card system giving cylinder transactions since startup is available in the USEC Nuclear Materials Accounting Department for cylinders located at PORTS. Some other interplant cylinder history is also available. From these records, it appears that the 26 10-ton cylinders fed in FY 1970 were received, fed, and filled with PORTS tails, and returned. The 62 10-ton cylinders fed in FY 1974 were returned to Paducah empty where 28 were cleaned and returned to PORTS empty in the June 1975 timeframe.

Following this period of RU feeds, the plant's second major equipment change-out cascade improvement program (CIP) began in FY 1973 and was completed in FY 1983. Essentially all X-33 size (640 stages) and X-31 size (500 stages) of process equipment and piping were removed, decontaminated, and modified. The amount of

TRU that could have been removed from the cascade during this campaign is based on the concentration data provided in Refs. 2 and 2a, and for the period is calculated as 12.8 g Np and 1.3 mg Pu.

During the period of FY 1968 - FY 1978, small quantities of various assays RU were received as UF_6 or converted to UF_6 at PORTS' Oxide Conversion Facility. This was part of the government's scrap returns program. See scrap returns program (UF_6) and converted oxides for cascade feed, Table 2.2.2.5-2, for amount of RU-UF_6 received per shipper, amount of RU-UF_6 produced at PORTS oxide conversion, assay, amount fed through March 1999, and the amount in storage as of March 31, 1999. Detailed information is available for this UF_6, except for the specific dates cylinders were fed. A total of 1.4 MTU was fed to the cascade in the X-326 Building during the HEU refeed program in the FY 1997 - FY 1998 timeframe was from France and NUMEC; UF_6 assays ranged from 56-82% (see Appendices VIII through XII).

Table 2.2.2.5-2

SCRAP RETURN PROGRAM (UF_6) AND CONVERTED OXIDES FOR CASCADE FEED

Shipper		Assay %	Fiscal Year Received/ Converted	Amount of UF_6 Received/ Converted (MTU)	No. & Sizes of Cylinders	Timeframe Fed to Cascade (FY)	Amount Fed MTU)	In Storage as of March 31, 1999 (MTU)	Rejected & Returned (MTU)
Division of International Affairs		78-80	1968	0.15	10-5"	1968-77	0.15	0	--
Babcock & Wilcox		78-97	1977	0.15	11-5"	1977-78	0.15	0	--
USAEC Office of Safeguards and Material Management		1.35	1969	2.73	2 (2-1/2 ton)	1969-93	--	2.73	--
		78		0.07	3 (5")		0.07	0	--
France		56-82	1972-78	1.6	67-5"	1997-98	1.1	0	0.3
NUMEC		80	1972	0.33	20-5"	1997-98	0.33	0	--
Oxide Conversion at PORTS from:	Idaho Chemical Processing Plant (ICPP)	50	1974 (Jan. & Feb.)	1.4	Unknown	1974-77	1.4	0	--
	NLO	2.9	1976 (Jan. & May)	4.2	13 (5" & 12")	1976-77 (10 cyl.)	0.46	0.12 (3 cyl.)	--
					4 (2-1/2 ton)	0	--	3.62	--

Throughout the plant's history, many pieces of process equipment have been changed out under routine plant maintenance in addition to the two major upgrade programs discussed above (see Annual Equipment Replacement Summary Table 2.2.2.5-3). Two smaller equipment change-outs were also reviewed and are discussed below.

The purge cascade converters began to plug with ^{99}Tc compounds and other metallic impurities from feeds. A program to remove, dismantle, and clean approximately 25 converters occurred during FY 1973-FY 1978. An estimated 0.375 kg of ^{99}Tc was removed from the cascade with this program. An off-stream hot-gas treatment

process was developed that allowed some ^{99}Tc plugged converters to be unplugged in place. The same process was used to clean equipment of ^{99}Tc prior to maintenance/equipment removal activities.

Table 2.2.2.5-3

ANNUAL EQUIPMENT REPLACEMENT SUMMARY

Calendar Year	Compressor Removals						Converter Removals					Total
	X-33	X-31	X-29	X-27	X-25	Total	X-33	X-31	X-29	X-27	X-25	
1954	--	47	--	--	--	47	--	6	--	--	--	6
1955	2	9	7	14	8	40	1	1	4	2	12	20
1956	20	10	7	13	13	63	12	59	2	0	2	75
1957	30	18	2	17	7	74	169	304	0	0	1	474
1958	22	80	18	16	8	144	345	15	1	2	0	363
1959	28	106	2	8	8	152	208	0	2	0	0	210
1960	8	203	54	12	9	286	40	20	213	2	0	275
1961	6	80	257	17	10	370	63	40	40	0	0	143
1962	0	102	70	4	12	188	5	50	0	0	0	55
1963	12	50	38	5	6	111	17	20	20	0	0	57
1964	5	42	45	5	12	109	4	20	30	1	0	55
1965	8	23	95	10	24	160	5	20	22	0	0	47
1966	4	37	15	13	14	83	5	10	0	12	2	29
1967	19	4	2	10	15	50	26	0	0	0	0	26
1968	6	1	2	12	18	39	20	0	0	0	0	20
1969	22	1	26	3	20	72	20	1	0	0	0	21
1970	21	1	20	2	6	50	1	14	0	0	0	15
1971	15	1	12	2	5	35	12	1	0	0	1	14
1972	36	10	18	4	6	74	12	0	0	0	8	20
1973	24	0	10	5	10	49	30	0	0	0	6	36
1974	40	4	13	3	5	65	44	0	0	0	0	44
1975	127	42	0	6	10	185	119	0	0	0	1	120
1976	148	11	16	5	8	188	118	1	1	0	20	140
1977	198	66	5	2	12	283	150	40	0	0	0	190
1978	150	70	2	5	10	237	148	52	0	0	4	204
1979	54	40	8	6	18	353	54	30	0	0	0	84
1980	106	3	4	15	15	143	110	6	0	0	18	134
1981	12	197	12	7	21	249	6	154	84	0	0	244
1982	4	190	5	34	23	256	2	190	16	1	18	227
1983	25	40	6	42	25	138	4	8	0	0	0	12
1984	32	6	6	54	13	111	14	0	0	0	0	14
1985	44	10	2	38	20	114	14	2	0	0	0	16
1986	21	2	0	18	3	44	20	2	0	0	0	22
1987	2	2	2	11	23	40	18	0	0	0	1	19
1988	2	2	3	7	23	37	18	1	0	0	0	19

MgF_2 sidestream traps to remove ^{99}Tc were installed for use with cells X25-7-15, 17, and 19. These cells correspond to peak concentrations of ^{99}Tc in the cascade. This trapping system was successful in reducing the cascade ^{99}Tc inventory, but created a source of concentrated ^{99}Tc trap media. The quantity of ^{99}Tc in this media has not been quantified.

Another program consisted of the removal and subsequent decontamination and rebuilding of approximately 190 Fairchild type 7A compressors during FY 1982-FY 1986. These compressors were located in the X-326 building and are part of the X-27 size equipment. Due to a design problem, these compressors had wet air inleakage at the bolted compressor flange area. No TRU was considered to be removed with this activity.

In FY 1991, high assay production was terminated and 1,680 stages of equipment were shut down. Treatment for deposit removal, as needed, and mothballing of these shutdown cells occurred between FY 1993 and FY 1998. A total of 240 X-27 size isotopic stages and 180 X-25 size purge cascade stages remain in operation

In July 1993, USEC leased the enrichment facilities from DOE with Lockheed Martin Utility Services becoming the Maintenance and Operating (M&O) contractor.

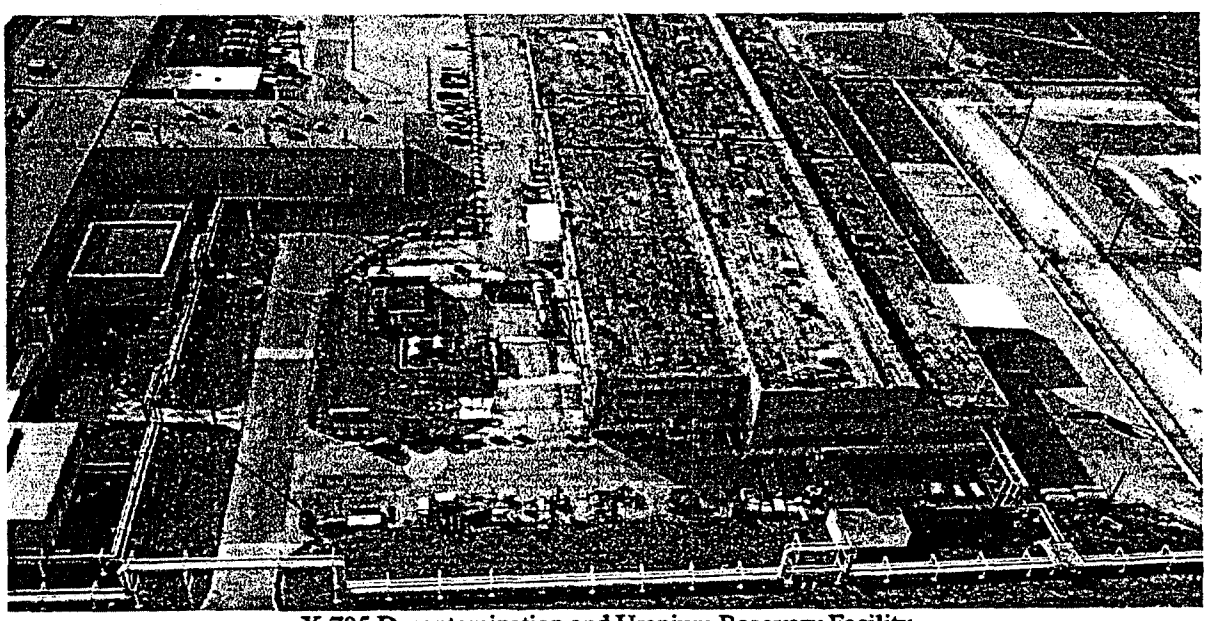
During FY 1997 - FY 1998, HEU UF_6 stored on site was fed and blended to LEU specifications in an agreement between DOE and USEC as part of a program to reduce PORTS inventory of HEU.

In March 1997, regulatory oversight of enrichment operations transferred from DOE to the Nuclear Regulatory Commission (NRC). In December 1998, X-326 side purge cell X25-7-2 caught fire due to internal compressor rubbing and resultant exothermic reaction. Three other side purge cells were damaged and are undergoing rework/rebuild. In May 1999, USEC took over direct operation of enrichment facilities.

2.2.2.6 Current Status

USEC is currently leasing and operating the cascade and its support facilities. The X-326 is essentially shut down except for 240 X-27 size isotopic stages and 180 X-25 size purge cascade stages. Varying amounts of X-27 size equipment and X-25 size equipment are running. Efforts are under way to rebuild the side purge cascade cells damaged from a 1998 fire. The shutdown equipment has been cleaned of large deposits and mothballed. All X-333 and X-330 building equipment is available for USEC's use. Approximately 88% of the X-33 size equipment, 60% of X-31 size equipment, and 8% of X-29 size equipment are running as of April 2000. The HEU-UF_6 in storage has been refed/down-blended and other uranium bearing materials of greater than 20% assay have been shipped off site. Six cylinders with RU-UF_6 (6.5 MTU) were in storage as of March 31, 1999 at PORTS.

2.2.3 Uranium Recovery



X-705 Decontamination and Uranium Recovery Facility

2.2.3.1 Plant Description

The uranium recovery operation is contained primarily in the X-705 with ancillary waste treatment operations located in the X-700 and at various times at the X-701B Holding Pond. A pictorial diagram of the integrated uranium recovery process is shown in Figure 2.2.3.1-1. The oxide conversion segment of this process will be discussed as a separate plant in Section 2.2.4 due to its unique mission.

The uranium recovery facility was designed to reclaim uranium from the following principle sources:

1. Decontamination solutions from equipment removed from the cascade and areas (small and large parts)
2. Field decontamination solutions
3. Trap media leaching
4. Oxide conversion, UF_4 conversion, and incinerator ashes and filtrates
5. UF_6 cylinder cleaning
6. Laboratory and miscellaneous sources.

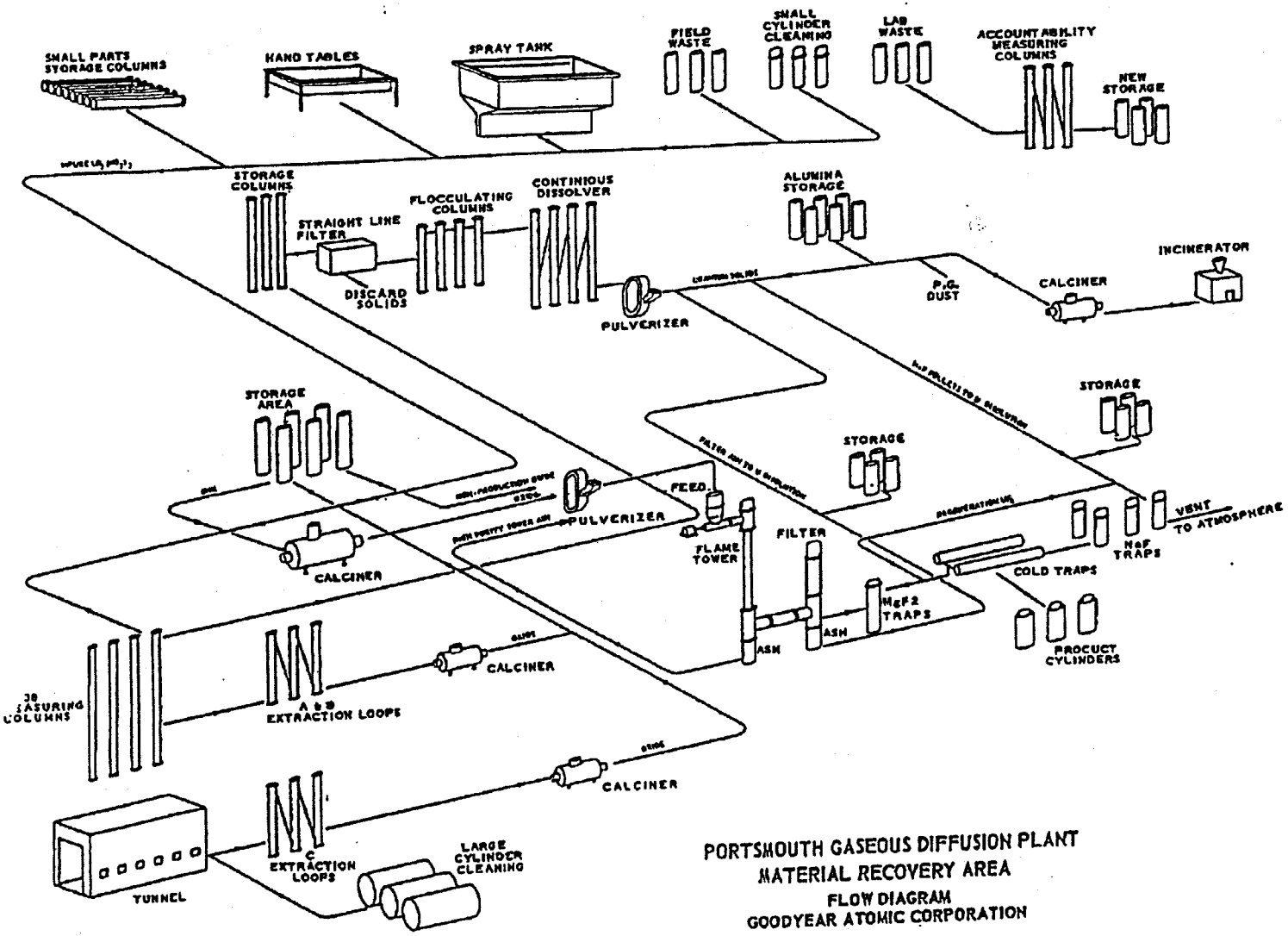
The processes used to create recoverable quantities of uranium from the above sources are varied in size, complexity, location, and throughput. They all, however, involve chemical or mechanical removal of uranium compounds from metallic surfaces and the subsequent dissolving of the removed uranium in aqueous or acidic solutions. These operations do not themselves selectively concentrate TRU or FP, but rather maintain the relative input concentrations of the constituents of concern. Concentration of the constituents in total is realized as solvents are reused/recycled until saturated.

The process used to recover uranium from solutions has been referred to as solution recovery, counter current extraction, or solvent extraction and appears to be a variant of the Plutonium and Uranium Extraction (PUREX) process developed for use at Hanford.

In this process (Figure 2.2.3.1-2), uranyl solutions are first evaporated to increase the specific gravity. The solution is then fed, along with nitric acid, to an extraction column where the uranyl ion is selectively captured by a mixture of tributyl phosphate (organic solvent) and Varsol (petroleum distillate). Next, this mixture has the uranyl ion as a nitrate stripped from it in a second (stripping) column with deionized water. The solvent solution is recycled to the extraction column for reuse. The aqueous uranyl nitrate solution (product) is fed to an evaporator where excess water is evaporated. The dewatered solution is then fed to a calciner (rotary kiln) where it is denitrated and further dried to produce U_3O_8, the final product, for storage, shipment, or fluorination to UF_6. Waste from this uranium recovery process is principally the depleted acid as raffinate from the extraction column.

TRUs in the uranyl solutions appear to follow the uranyl ions throughout all steps of the oxide formation process. The minor amounts that accompany the raffinate are discussed later. Experimental data collected and published by Walker (circa 1977) (Ref. 9) are as follows:

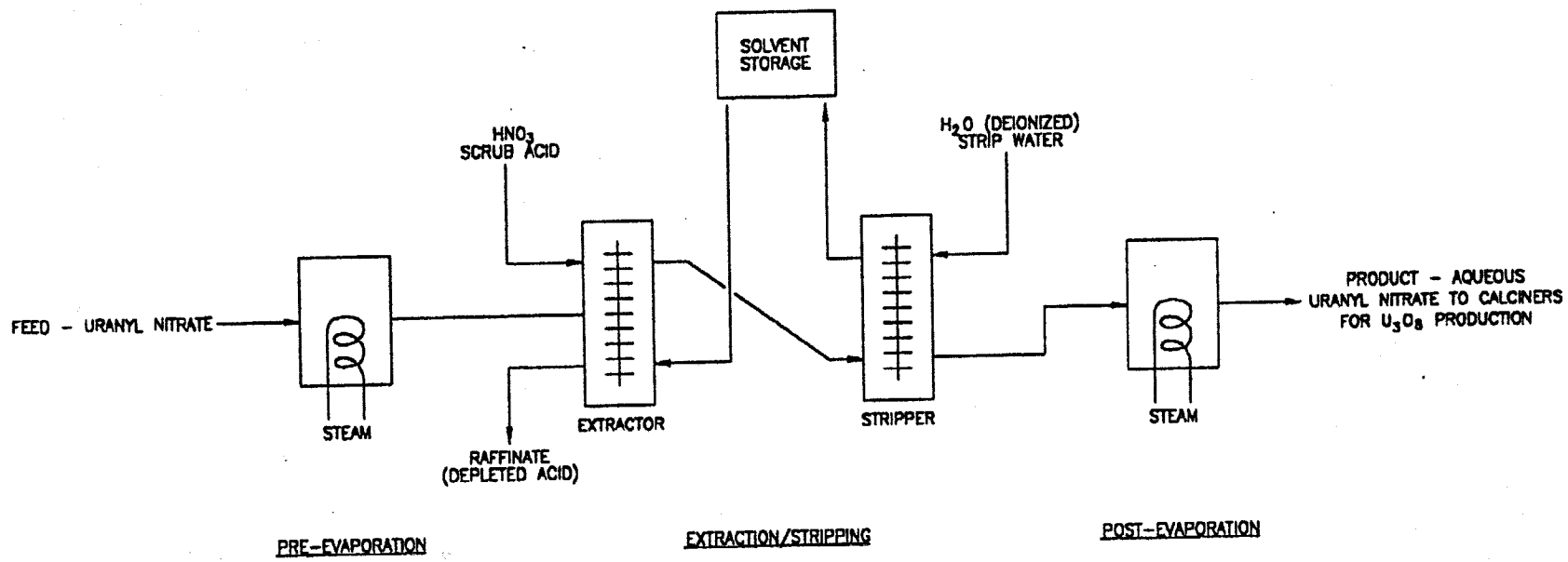
Process Step	Total TRU	^{237}Np	$^{239-240}Pu$	^{238}Pu
Loaded solvent (dpm/ml)	60,700	16,900	23,700	20,100
Stripped solvent (dpm/ml)	675	222	230	223
% stripped	98.89	98.69	99.03	89.98



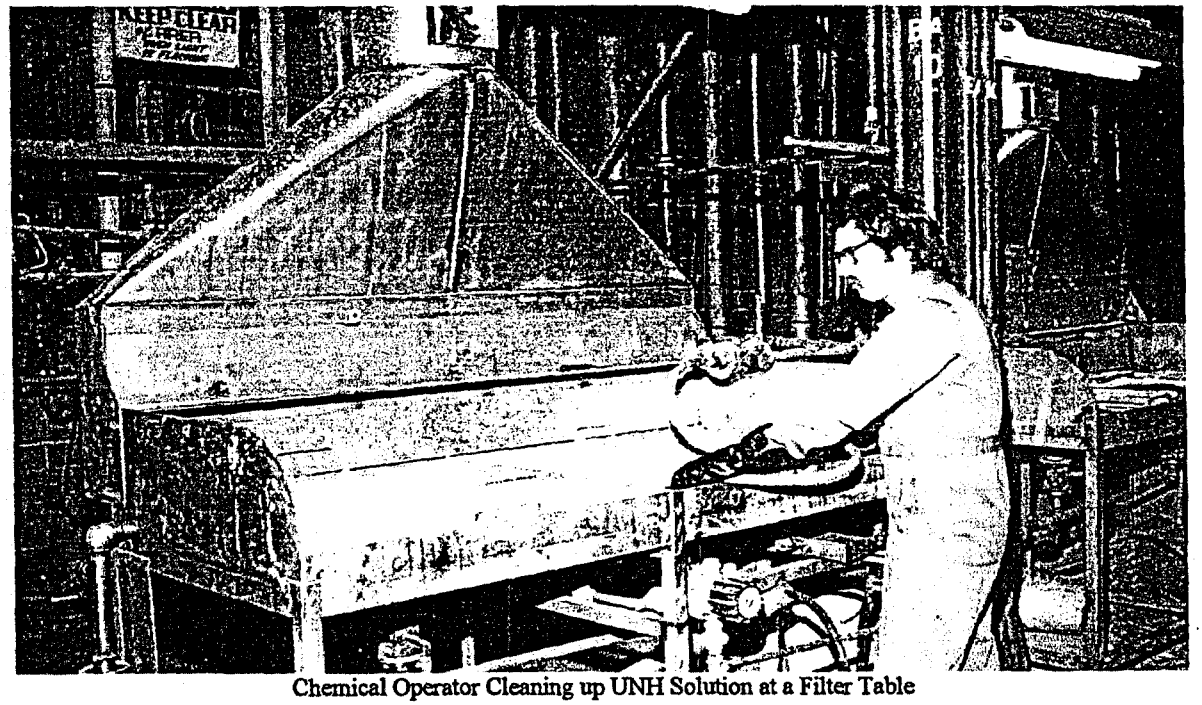
PORTSMOUTH GASEOUS DIFFUSION PLANT
MATERIAL RECOVERY AREA
FLOW DIAGRAM
GOODYEAR ATOMIC CORPORATION

Figure 2.2.3.1-1

FIGURE 2.2.3.1–2
SOLUTION RECOVERY PROCESS



SOLVENT: 20% TRIBUTYLPHOSPHATE $(C_4H_9)_3PO_4$
80% VARSOL (PETROLEUM DISTILLATE – KEROSENE)



Chemical Operator Cleaning up UNH Solution at a Filter Table

The ^{99}Tc appears to take the opposite course, being highly soluble in both nitric acid and water, and forming pertechnetic acid ($HTcO_4$) (Ref. 10). This acid, however, is not stripped from the solvent and remains in the waste raffinate. While quantitative data could not be found, the literature suggests that essentially all ^{99}Tc enters the raffinate stream and that has been the assumption for this study.

Concentration of ^{99}Tc, therefore, takes place in solution recovery in the form of the sludges and precipitates formed from raffinate treatment.

Treatment of the ^{99}Tc containing raffinate has evolved throughout the plant's history from: (1) rudimentary pH adjustment and discharge to the X-701B/east drainage ditch/Scioto River to (2) secondary pH adjustment and retention at the X-701B/east drainage ditch/Scioto River to (3) today's configuration of extensive treatment via (a) heavy metals precipitation; (b) technetium ion exchange; and (c) biodenitrification with discharge only to a permitted waste water treatment facility.

Concentration of ^{99}Tc has occurred in: (1) soils surrounding the settling pond and drainage ditch (pre FY 1972), (2) sludges/precipitates occurring from neutralization with lime (FY 1972-FY 1984), and (3) heavy metals neutralization sludge and spent ion exchange resins (since FY 1984)

2.2.3.2 Material Flowsheet

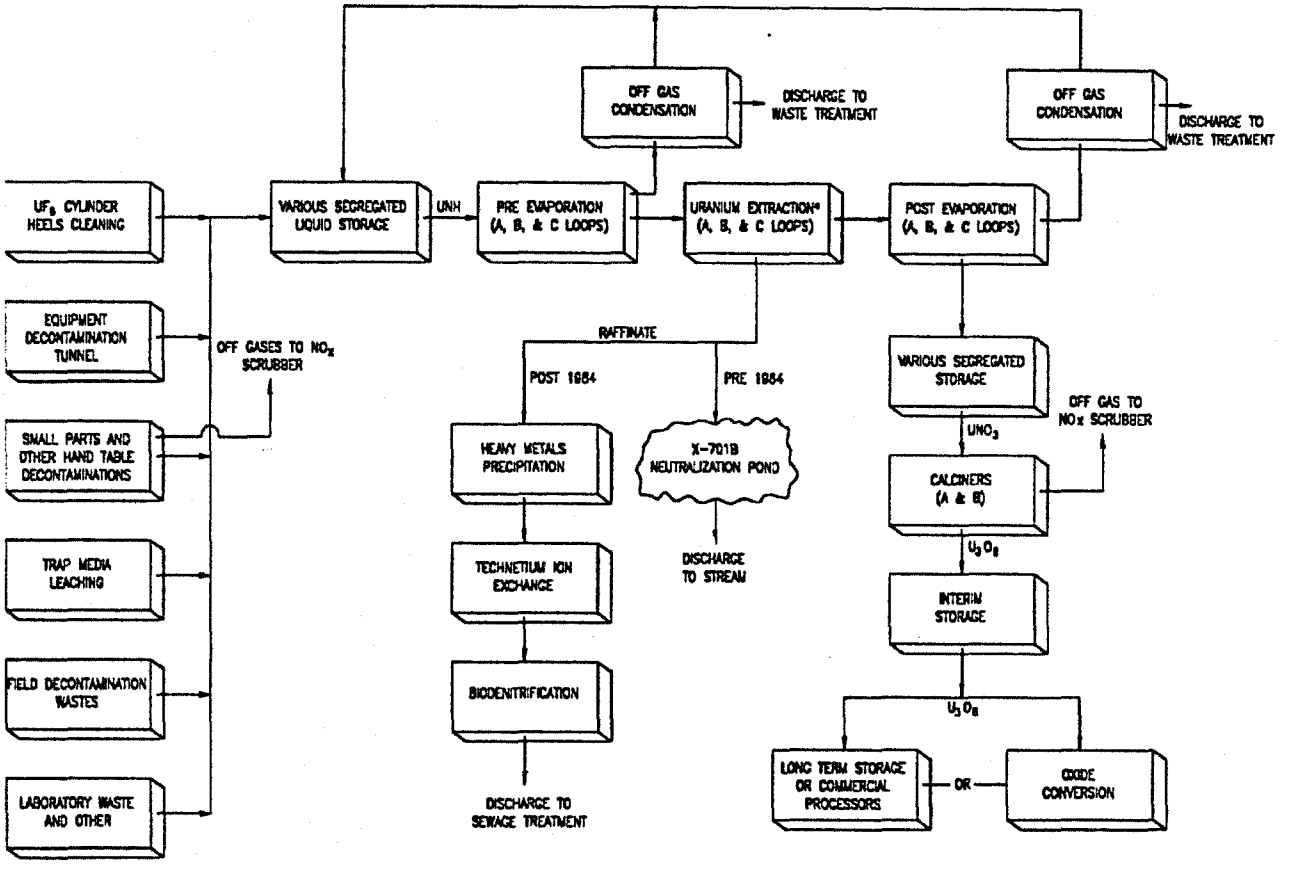
Uranium-bearing material forms and flows into and out of the uranium recovery process, (including oxide generation) are shown as Figure 2.2.3.2-1.

Measurement of these streams has been an uneven process throughout the years covered by this study. Table 2.2.3.2-1 presents the oxide output flows to the extent that the information could be located in plant records (Ref. 11).

Of principal concern with this uranium recovery facility is the ^{99}Tc content of the raffinate waste stream. Other than spot samples taken during process upset and discovery periods, representative sample data for operating periods can not be found.

Figure 2.2.3.2-1

Uranium Recovery



* TRIBUTYL PHOSPHATE AND PETROLEUM DISTILATE COUNTER CURRENT EXTRACTION PROCESS.

When ^{99}Tc was first detected in the X-701B (FY 1975), extensive sampling to quantify the ^{99}Tc constituent was undertaken. As part of the environmental sampling program, monitoring of this stream has continued unabated ever since. Amounts of ^{99}Tc in this flow are shown in Table 2.5-1 (Ref. 12).

It is postulated that the best quantification of ^{99}Tc in this waste consists of the characterization studies performed on the sludges dredged from the X-701B. This environmental remediation captured a high percentage of precipitated ^{99}Tc compounds that were deposited prior to ceasing use of the X-701B facility in FY 1983. Three hundred ninety-one grams (g) of ^{99}Tc were measured in approximately 2.5 million pounds of sludge. Considering that this sludge contained 1,652 kgU, ^{99}Tc was concentrated here to 237 ppm on a uranium basis. Adding this to the 1,024 g for years prior to FY 1983 (Table 2.5-1) gives an estimate of 1,415 grams as the total ^{99}Tc eminating from the raffinates of uranium recovery. It should be noted that in addition to ^{99}Tc, 0.03 and 3.3 grams of Pu and Np respectively, were measured in the X-701B sludge when it was characterized for LLW disposal. This serves to confirm the imperfect performance of the solution recovery process or the possibility that all TRU-containing materials were not processed through recovery. These sludges are currently being shipped to Envirocare of Utah, Inc.

Table 2.2.3.2-1

Uranium Recovery Production Data Summary

Fiscal Year	Oxide kgU Produced	Fiscal Year	Oxide kgU Produced
1956	726	1978	1105
1957	1313	1979	1758
1958	1233	1980	311
1959	1799	1981	1240
1960	2896 (Peak Production)	1982	891
1961	1559	1983	1127
1962	2032	1984	888
1963	526	1985	592
1964	493	1986	926
1965	640	1987	1263
1966	508	1988	1330
1967	581	1989	561
1968	537	1990	640
1969	713	1991	641
1970	875	1992	218
1971	1745	1993	167
1972	863	1994	299
1973	585	1995	125
1974	447	1996	328
1975	773	1997	456
1976	1020*	1998	333
1977	1069	1999	125**
TOTAL			**38,257**

*15 months
**through March 1999

2.2.3.3. Feed Specifications

Go-NoGo criteria for feed to uranium recovery consisted in the past of uranium content (ppm) and enrichment (% ^{235}U) nomograms derived from the value of the material and the cost to recover the uranium. A typical example of this is given as Figure 2.2.3.3-1 Discard Criteria for Solutions (Ref. 13). Materials not meeting the reclaim criteria were usually dumped to the X-701B through building drains. No record can be found of consideration of either TRU or FP concentrations as a criteria for feeding. It is suspected that most input streams were never measured for either TRU or FP. Current day operations also do not control inputs. Discharge through building drains was discontinued circa 1984.

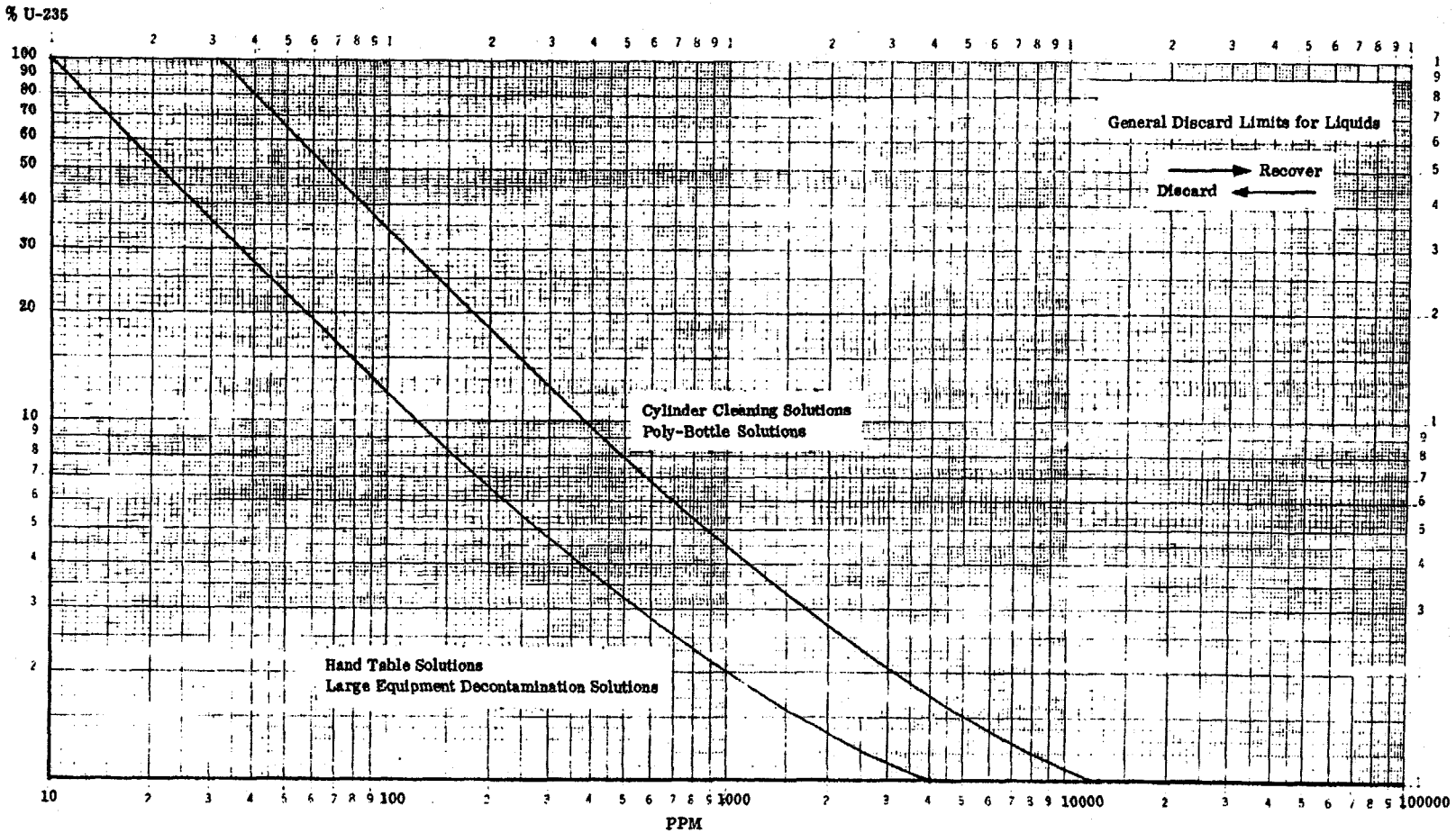
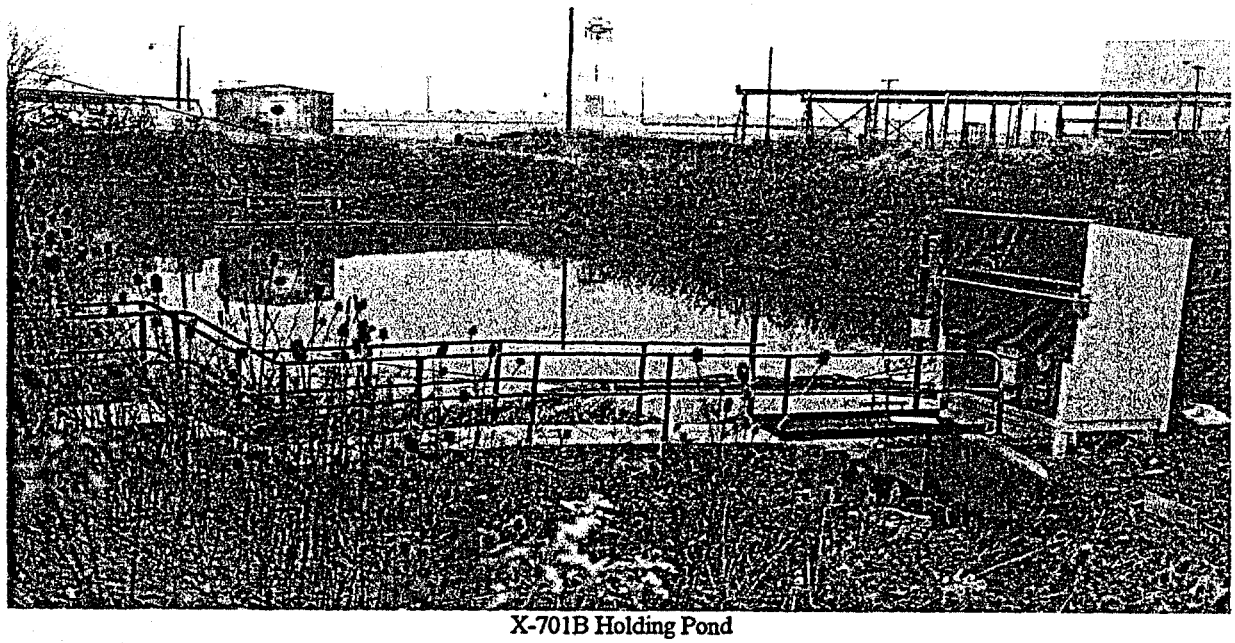


Figure 2.2.3.3-1

2.2.3.4. Product Specifications

Product from uranium recovery consists of fully dried and denitrated U_3O_8. No attempt was made to control purity, TRU/FP content, etc. Levels of denitrification and dehydration are controlled by feed rate to the rotary kiln to produce a product at a prescribed color and dryness based on experience.

2.2.3.5. Operating History



X-701B Holding Pond

The solution recovery system was turned over to GAT on August 3, 1955 and began .producing production quantities of U_3O_8 during late FY 1956. While many improvements for operational efficiency and safety (especially nuclear criticality safety) have been made over the life of the facility, the primary process of solvent extraction/stripping and calcination have remained unchanged. Decontamination chemicals (boric acid, sodium carbonate, citric acid, sodium bisulfate, etc.) have been introduced to enhance decontamination efficiency and/or adopt more environmentally friendly technologies during the precursor steps in the process. Major modifications or operational changes to the effluent (raffinate) treatment occurred as follows:

1. In 1972, the precipitation of uranium in the X-701B was enhanced by the addition of facilities to feed slaked lime and photoelectrolyte proportional to the influent rate. This raised the influent stream pH to about 8 in order to promote precipitation of uranium. (This would have also enhanced the precipitation of Np and Pu and some ^{99}Tc compounds).

2. Once or twice each year, the pond was dredged or sludge pumped to containment ponds elevated above and alongside the settling pond.

3. The east containment pond held the dredgings from 1973 through 1980. The west containment pond held the dredgings from 1980 to1984

4. During 1984, the discharge of raffinate to X-701B was permanently suspended with the startup of the heavy metals precipitation process (X-705), ^{99}Tc ion exchange process (X-705), and the biodenitrification process (X-700).

5. The X-701B and the two companion containment ponds were dredged with their contents and about one foot of bottom soil being containerized and characterized for disposal as LLW. Shipment of these soils to Envirocare of Utah, Inc. is in progress.

The peak year of operation occurred in FY 1960 when this facility processed nearly 8 MTU probably predominately from solutions generated from the first process change-out program. Average production rates were approximately 0.9 MTU per year over the life of the facility. Total production of U_3O_8 for the approximately 44 years of operation through March 1999 was about 38.2 MTU. Highlights of the operation include sustained high levels of throughput during the period FY 1974 through FY 1983, much of which was in support of the second diffusion process change-out known as the Cascade Improvement Program/Cascade Uprating Program (CIP/CUP).

2.2.3.6 Current Status

Uranium recovery remains in service to support ongoing efforts of USEC to maintain the gaseous diffusion plant operation. Extensive revisions to nuclear criticality safety driven procedures have been made and other changes have been initiated in support of Technical Safety Requirements of the NRC-USEC license.

2.2.4 Oxide Conversion

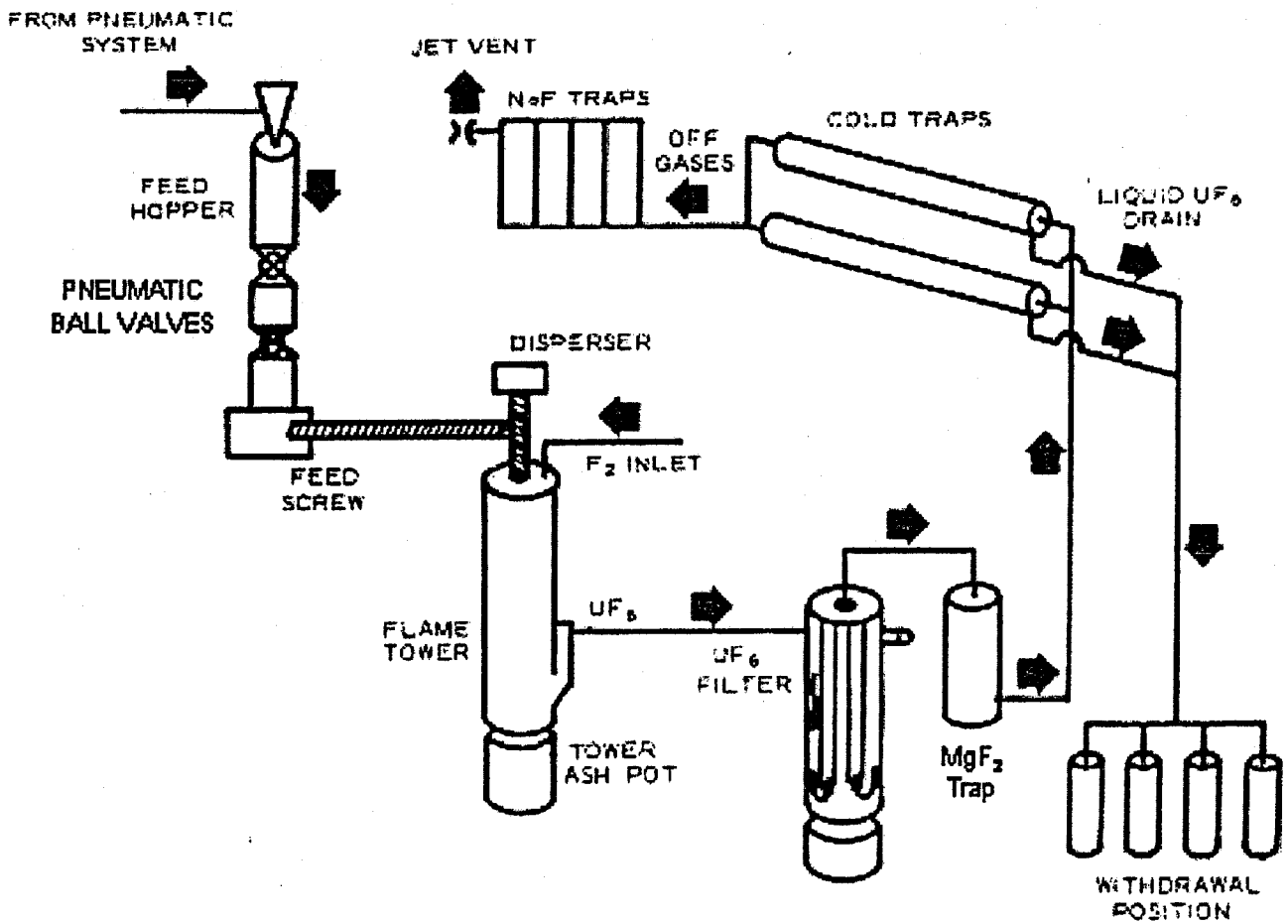
2.2.4.1 Plant Description

The Oxide Conversion Facility is located in "E", "F", and "H" areas of X-705. It was supplied as part of the original plant equipment compliment and was designed to be an integral part of the uranium recovery process (see Figure 2.2.3.1-1). Over time, this facility developed a somewhat unique mission and set of operating streams and as such is dealt with separately for this report.

To be used as feed to the gaseous diffusion process, uranium oxides originating from a variety of operations (both on-site and off) must be converted (fluorinated) to UF_6. This process of fluorination was originally performed at PORTS in a system of three parallel, horizontal, screw-fed, stirred-bed reactors. In these reactors, U_3O_8 was reacted with fluorine to form UF_6 which was cold trapped in chilled 5" cylinders. Little is known about this early system other than the fact that it was unreliable and had inadequate production capabilities. In 1959 the stirred-bed reactors were replaced with a site-developed 4" diameter open flame tower using direct fluorination. Stated capacity of the system was 7,200 kgU/yr output as UF_6.

In mid-1965, AEC approved a project to upgrade the Oxide Conversion Facility to achieve a production capability of 20,000 kgU/yr as UF_6 in support of its designation of PORTS as the processor of uranium scrap >10% enrichment for the DOE complex. The upgrade was accomplished in 1966 and 1967 and consisted of installing a significantly improved 5" flame tower, an improved feed system, an improved product filtering and withdrawal system, and an improved off-gas treatment and monitoring system. A pneumatic conveying system was installed to facilitate movement of the oxides and ashes. Where release of dusty powders might be expected, gloveboxes were installed. A pictorial diagram of the fluorination process is shown as Figure 2.2.4.1-1. A more detailed description of the process and its history are available in Ref. 14.

The process worked as follows: Oxides were fed into the top of the flame tower where they fell by gravity through a fluorine jet. Unburned (unreacted) oxide and some impurities would fall into the ash pot below the tower where, on a batch basis, the ash would normally be removed, ground and refed to the top of the tower. UF_6 generated in the tower would pass through a sintered metal filter to remove particulates and then through a MgF_2 trap for sorption of certain impurities.

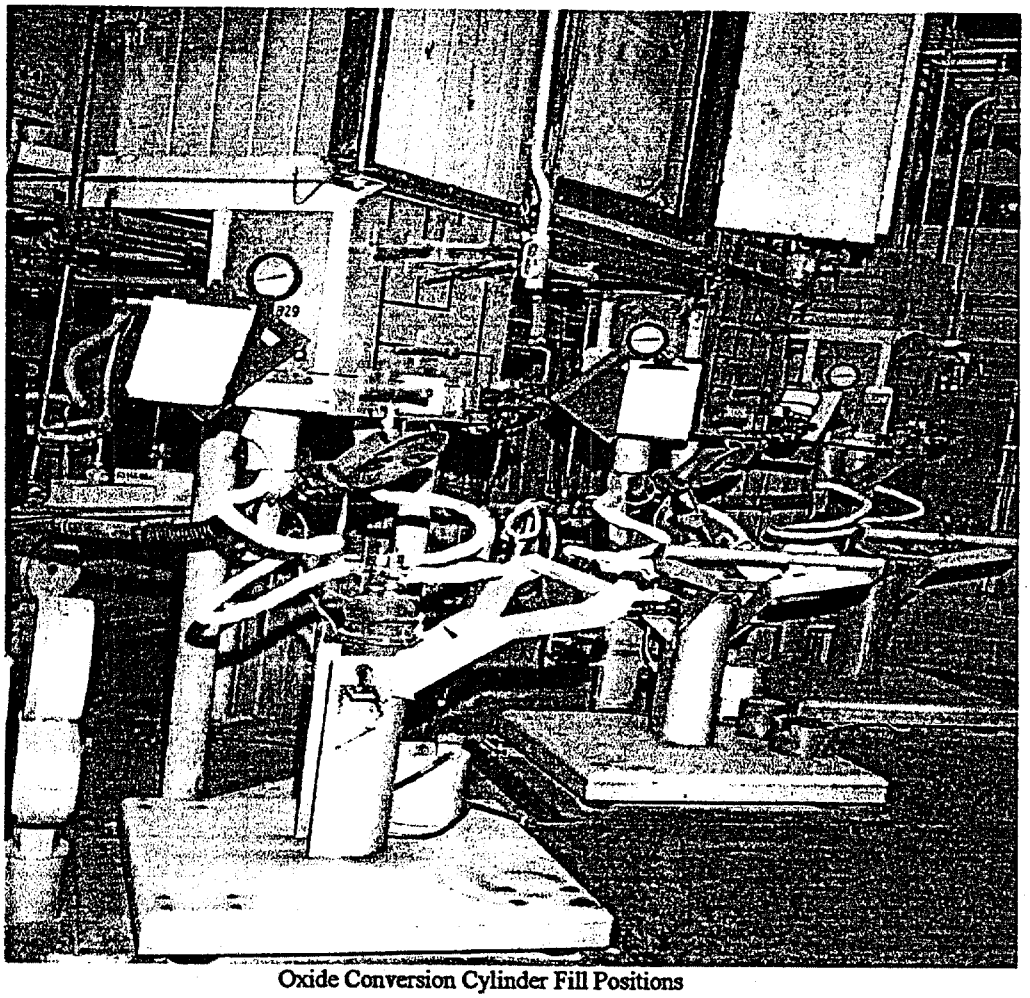


Oxide Conversion System Flow Diagram

Figure 2.2.4.1-1

Gaseous UF_6 was condensed in cold traps with liquid UF_6 drained by gravity to 5", 8", 12", or 2-1/2-ton cylinders. Off gases from the cold traps were passed through NaF traps to remove remaining minor amounts of UF_6 prior to ejection through a monitored vent.

In theory, TRU/FP's fed to this process take the following paths: Pu oxide formed gaseous PuF_6 in the flame tower should quickly become solid PuF_4 due to spontaneous dissociation and fall out in the tower ash or be filtered by the sintered metal filter (filter ash). Fluorinated Np will be sorbed on the MgF_2 trap. Technetium is generally not expected in the feed. If it was present, it would be easily fluorinated, sorbed on the MgF_2 or NaF traps or be vented. Based on these assumptions, Pu should concentrate in the tower ash and filter ash. Np should concentrate on MgF_2.



Oxide Conversion Cylinder Fill Positions

The ^{99}Tc will not be a factor except in the vent. Product cylinders should contain only trace amounts of TRU/FP's. In reality, Pu and Np both are found to be concentrated (based on uranium feed) from time to time in tower and filter ashes, MgF_2 trapping media, and in UF_6 product cylinders. Based on sample analysis data (Ref. 15), it appears that concentrations of constituents of concern in the waste and product streams are highly dependent on the degree of ash recycle, the in-service life of the trap media, as well as the constituent concentrations in the oxide feed. Perhaps operating parameters such as tower temperature also influence these concentrations. While a detailed parametric analysis of all variables could not be found, an analysis based on TRU sample data from a May 1976 run involving TRU material received from NLO gives insight into the distribution of constituents. Table 2.2.4.1-1 shows some of this data.

Table 2.2.4.1-1

Material Balance of Transuranics in Oxide Conversion During May 1976

Process Step	Uranium (kgU)	^{237}Np (g)	^{238}Pu (ug)	^{239}Pu (ug)	^{241}Pu (ug)
Input Oxide	3241	1.133	96.3	13510	231
Output					
Solution	6.7	0.003405	0.5	196	0.7
Filter Ash	0.9	0.1419	21.2	5152	8.8
Tower Ash	14.8	0.0112	2.2	344	3.1
Oxide	5.1	0.0034	0.9	84	1.3
Sodium Fluoride (NaF)	3.5	0.0118	0.1	34	-0-
Total Output	31.0	0.1717	24.9	5810	13.9
%Output/Input	0.96	15.2	25.8	43.0	6.0
% In product Cylinders (max)	99	84.8	74.2	57.0	94.0

From this limited data (UF_6 output and MgF_2 holdup were not analyzed), it can be seen that as much as 85% of the ^{237}Np and 57% of the ^{239}Pu may be present in the UF_6 product cylinders. Of that not in the UF_6 product, the filter ash is seen to contain the majority of the TRU constituents (12.5% of ^{237}Np and 38% of ^{239}Pu). The concentrating effect of these streams relative to input uranium concentration is illustrated in Table 2.2.4.1-2.

Table 2.2.4.1-2

TRU Concentration Factors from Table 2.2.4.1-1

Process Step	^{237}Np (g/gU)	Concentration Factor (^{237}Np)	^{239}Pu (g/gU)	Concentration Factor (^{239}Pu)
Input Oxide	3.5×10^{-7}	Base	4.2×10^{-9}	Base
Filter Ash	1.6×10^{-4}	451	5.7×10^{-6}	1375
Tower Ash	7.6×10^{-7}	2.2	2.3×10^{-8}	5.6
NaF	3.4×10^{-6}	9.7	9.7×10^{-9}	2.3

The filter ash for this case is seen to have concentrated the ^{237}Np by 451 times and the ^{239}Pu by 1,375 times relative to the levels in the uranium fed to the facility.

Analysis of filter ash remaining on site and analyzed as part of the HEU removal characterization studies showed average concentrations of: 1.9×10^{-4} and 3.3×10^{-6} g/gU, respectively, and peak concentrations of 3.9×10^{-4} and 5.8×10^{-6} g/gU, respectively, of ^{237}Np and ^{239}Pu which are reasonably close to those in the NLO material balance study.

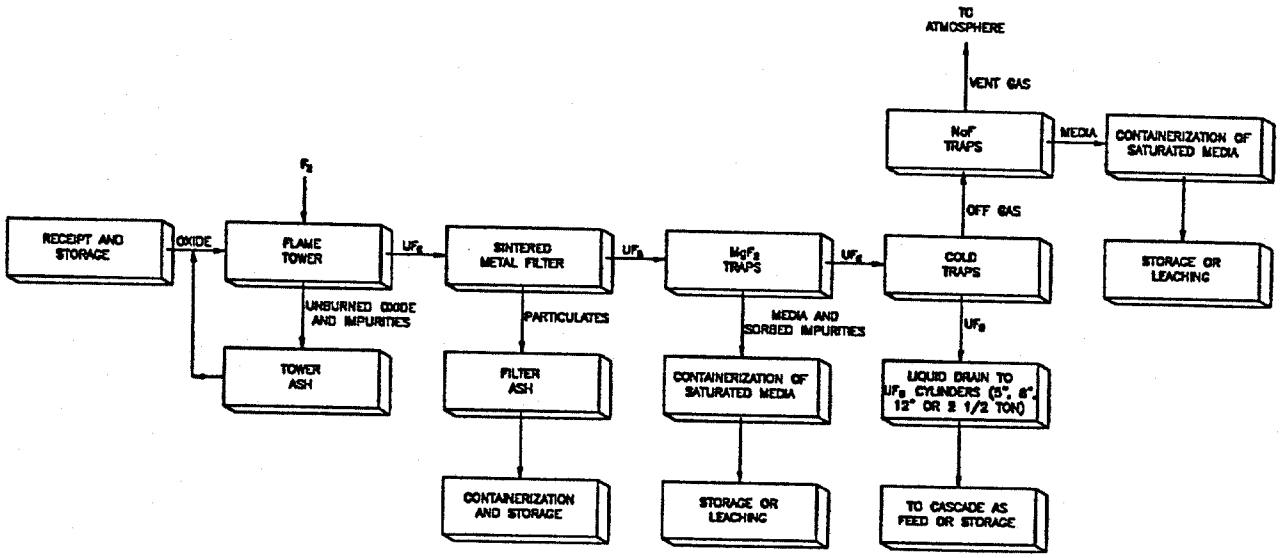
As a result of these data, the oxide conversion process is considered as removing at least 15% of ^{237}Np and 43% of ^{239}Pu.

2.2.4.2. Material Flowsheet

Uranium bearing material flows through the oxide conversion process are shown in Figure 2.2.4.2-1.

Figure 2.2.4.2-1

Oxide Conversion Flow Sheet



Accurate quantification records of these streams has been an uneven process throughout the life of the facility. Table 2.2.4.2-1 presents the UF_6 produced from oxide based on information available in plant records (Ref. 11).

Table 2.2.4.2-1

Oxide Conversion Production Data Summary

Fiscal Year	UF_6 Produced kgU	Fiscal Year	UF_6 Produced kgU
1958	2,795	1967	0
1959	1,425	1968	6,622
1960	887	1969	20,257
1961	1,170	1970	7,712
1962	866	1971	1,580
1963	497	1972	48,550
1964	978	1973	51,344
1965	4,085	1974	16,650
1966	4,995	1975	22,915
Total Old System	17,698	1976 (5 Quarters)	28,710
		1977	7,806
		1978	3,252
		Total New System	215,398
Total Life of Facility 233,096 kgU as UF_6			

Of principal concern with this facility is the TRU content of the feed stocks and their concentration in the product and waste streams.

Significant amounts of TRU materials were known to have been processed during the two periods shown in Table 2.2.4.2-2 (Ref. 16).

Table 2.2.4.2-2

Period of Operation	Quantity Processed kgU	Origin of Material	Probable Total Alpha from TRU dpm/gU
Jan.-Feb. 1974	1373	ICPP	<2000
Jan. & May 1976	4214	NLO	<3000

There is some indication that as much as 3.7 MTU of recycled depleted oxide (UO_2 and UO_3) from the Oak Ridge K-25 site may have been processed in the 1958 – 1961 time frame. Shipping records for this material specify that it was to be used for research and development purposes. It is believed that this material was used during the development of the fluorination tower (FluorOx) in the X-760 Chemical Engineering Building or during unsuccessful development efforts for oxide pelletization. Since this material was depleted and of extremely low value compared to HE/VHE materials it seems unlikely that the limited oxide conversion capacity in the X-705 would have been used to process this material. It would have amounted to approximately 75% of the total production of the Oxide Conversion Facility during the 1958 –1961 time period. The provenance of this material is unknown and remains in the disposition uncertain category.

There was a significant amount of TRU/FP (6847 kgU) as UNH calcined to U_3O_8 in the oxide conversion calciner during March – December 1977. This material was never converted to UF_6 at PORTS and was later shipped to NLO.

Side streams from the process were generally treated as follows:

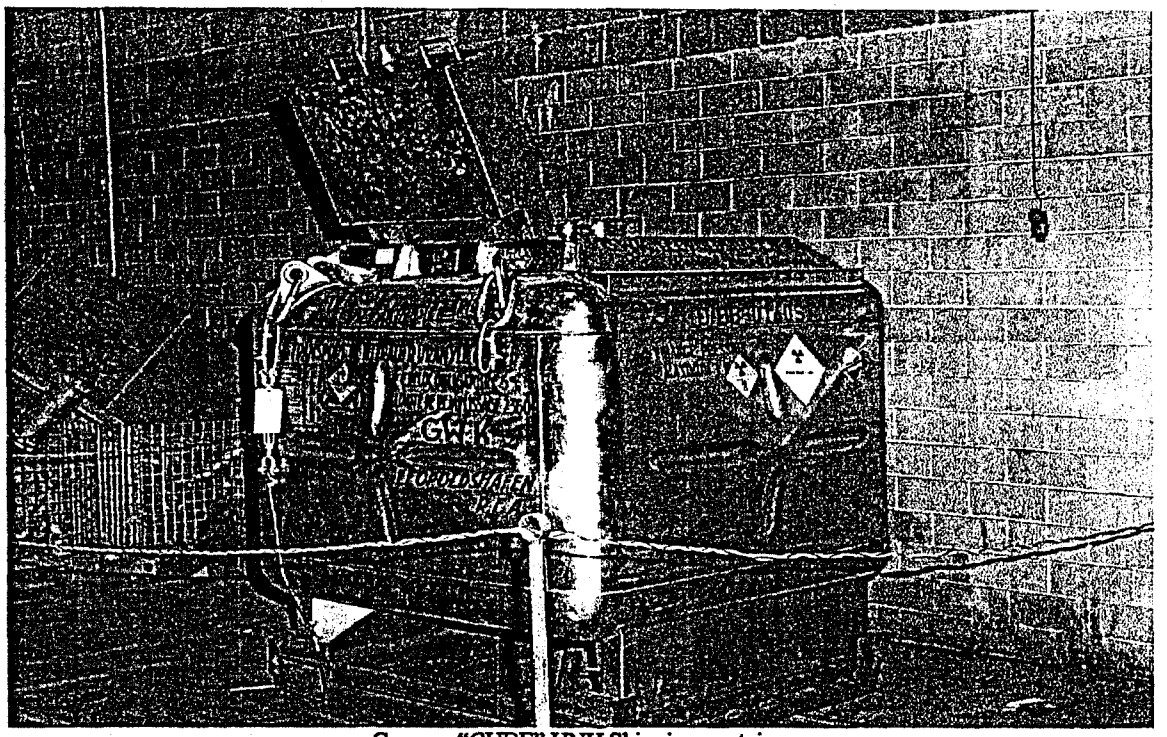
1. Tower Ash: Recycle to tower as feed as soon as possible after generation.
2. Filter Ash: Some unknown (possibly minor) amounts were digested at uranium recovery to create U_3O_8 for a second attempt at conversion to UF_6. The filter ash that remains on-site is stored in the X-326 "L" Cage. Partial contents of the amount remaining are shown in Table 2.2.4.2-3.

3. Magnesium Fluoride: There are indications that some of this material may have been leached at uranium recovery. There remains on site a quantity of MgF_2 that, in part, may have been generated from oxide conversion. Quantification of this stream is to be determined.

Table 2.2.4.2-3

Filter Ash (Partial Contents) Stored in X-326 "L" Cage

Constituent	Quantity (g)
Filter Ash	40,725
Uranium	3,007
^{235}U	1,233
^{237}Np	0.563
^{239}Pu	0.010



German "CUBE" UNH Shipping container

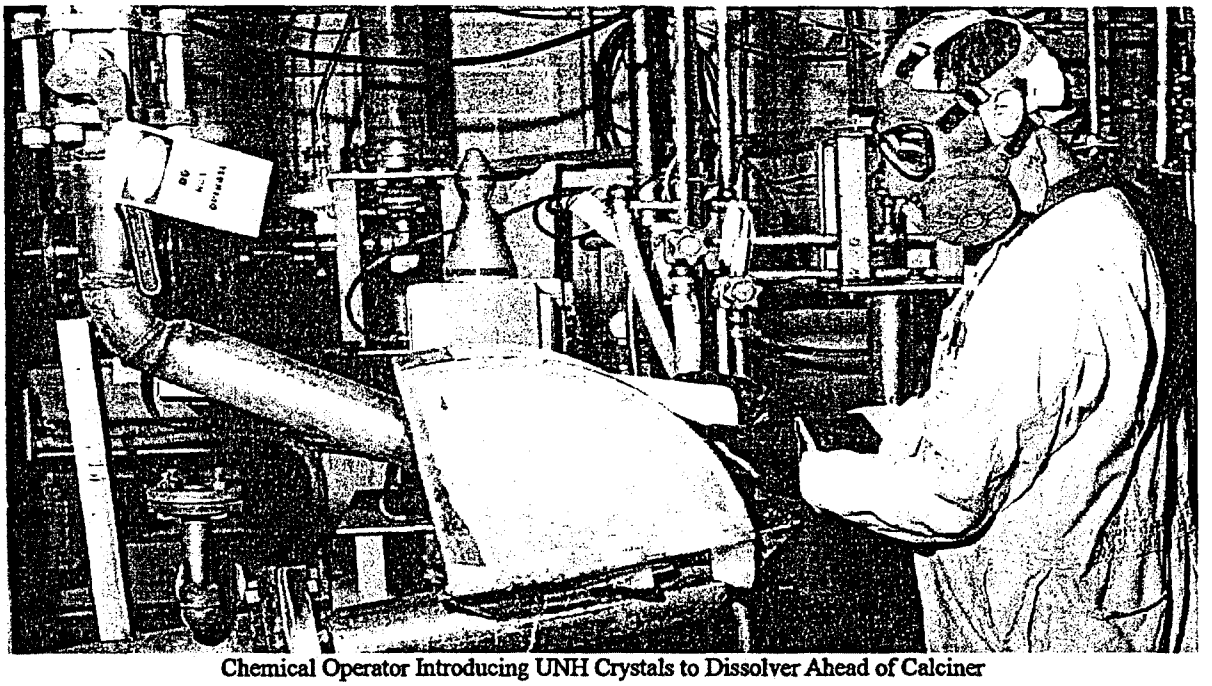
4. Sodium Fluoride: The facility was designed to regenerate the NaF traps through in-place heating and vaporization of sorbed uranium. Regenerated UF_6 would be introduced into the inlet of the cold traps, condensed, and drained into UF_6 cylinders. After several regeneration cycles, the NaF loses its absorption capability and must be replaced. This depleted/removed trap media may contain TRU. Also, whenever inventory of the system was required, usually monthly, the NaF was removed and the uranium leached, measured, and reintroduced into the uranium recovery process, if economical. The TRU on the media may accompany the media or be leached with the uranium. Depleted/leached media was containerized, the cans placed into site-prepared wooden boxes with void spaces filled with lime, and the boxes were buried at the X-749A landfill.

2.2.4.3. Feed Specifications

Feed to oxide conversion originated from both on-site and off-site sources. A specification for material originating from on-site could not be found. It is probable that if the oxide was of an enrichment level corresponding to a planned production campaign, it was considered an acceptable feed stock. The facility was operated to minimize assay mixing losses and, as such, quantities of materials at desired enrichments became the criteria for feeding. It is probable that materials originating on-site were seldom, if ever, fully analyzed except for uranium and ^{235}U content. Materials originating off-site were required to satisfy: "Feed Specification for U-235 Enriched Uranium Returned to AEC" (Ref. 17).

Since the UF_6 produced in oxide conversion was destined to be fed to the diffusion cascade, specific limits were placed on many parameters (usually metals) that would result in out-of-specification product from the diffusion cascade (nuclear poisons, etc.). Total TRU limits were expressed as total alpha activity from TRU and from the earliest days of the scrap returns program were set at 1,500 dpm/gU. Deviations from the acceptance specification were numerous throughout the life of the scrap program. These deviations were primarily due to excessive amounts of various non-TRU metals. Numerous letters accompany the transaction records prescribing minimal monetary penalties in an effort to indemnify AEC for cost of dealing with off-specification materials. No examples, however,

of deviations for TRU with accompanying penalties were found. There was substantial correspondence between AEC site managers or operating contractors discussing capabilities to accept various forms and levels of TRU. Materials from NLO were knowingly accepted in the FY 1975 - FY 1976 timeframe with gross alpha from TRU levels up to 3,000 dpm (Ref. 18).



Chemical Operator Introducing UNH Crystals to Dissolver Ahead of Calciner

2.2.4.4 Product Specification

No formal specification existed covering the production of UF_6 from oxide conversion. As stated earlier, impurities in product became an issue in-as-much as they would impact the cascade and its product. Since this UF_6 would ultimately be blended in the cascade with a large amount of other feed stocks, timing and scheduling its feeding to the cascade could frequently mitigate any adverse contaminates. Materials unable to be fed to the cascade were produced at oxide conversion and were still present at the suspension of HEU production in 1991.

2.2.4.5 Operating History

A significant event timeline and a detailed explanation of oxide conversion's history are available in Reference 14. In summary, the facility operated from 1957 to 1978. It produced about 233 MTU of UF_6 from materials originating from at least 47 feed sources. It is known that 5.6 MTU of feed contained TRU's. Operations were terminated when it became apparent that the existing facility could not meet current standards for containment as manifested in high levels of airborne contamination. Additional efforts to modernize, renovate, or replace the facility were terminated for the last time in July 1981 when cost estimates to provide this capability at PORTS indicated prohibitively high costs.

2.2.4.6 Current Status

Attention to this facility since shutdown has been limited to custodial activities that assure it remains safe and secure. The facility is locked to prevent spread of transferable contamination. Minor amounts of contamination remain internal to the system. There currently are no known funded plans for Decontamination and Decommissioning (D&D) of this facility.

2.2.5 Other Uranium Handling Facilities

Other uranium handling facilities are X-710 Technical Services Building (laboratory), X-760 Chemical Engineering Building, X-744G Bulk Storage Warehouse (uranium storage warehouse), X-345 Special Nuclear Materials (SNM) Storage Building, and X-745 Cylinder Storage Yards.

1. X-710 Technical Services Building

The X-710 laboratory was part of the original plant facilities. With respect to uranium, the primary analytical capabilities of the laboratories are:

a. Isotopic analysis (mass spectrometry) for feed and product;

b. Uranium purity (Davies Grey, wet chemistry);

c. Metallic impurities (ICP, spectral analysis);

d. Radiological analysis (radiochemistry, alpha, beta, gamma, counting); and

e. Sample preparation and waste handling processes for all of the above.

For this report, quantities processed were considered small with minimal likelihood of concentration.

Development of processes and procedures for plant use has been an integral part of this facility's mission for most of its life. There were a myriad of experiments that involved TRU and FP that potentially separated or concentrated these constituents. These development activities involved minor/trace amounts of these elements/isotopes. For this report, no attempt to quantify these is made. The likelihood of relevance to the site mass balance is low, but not zero.

2. X-760 Chemical Engineering Building

The X-760 Chemical Engineering Building had, as its mission, the pilot-scale development work on new chemical processes prior to or in aid of plant deployment. This facility had a small process laboratory, a small machine/fabrication shop, a worker change/shower area, and essentially all plantsite utilities. Early development projects, including decontamination process experiments, boiling freon heat exchanger experiments, UF_6 heating studies, uranium oxide pelletizing experiments, freon drying tests, and controlled UF_6 releases in a sealed environmental chamber, were conducted in this facility. Most relevant for this study is the prototype development work on what is referred to in reports as the fluorox process. This (what appears to be) fluorination tower may have been prototypical or a developmental aid either for the UF_6 feed manufacturing facility (X-344) or the oxide conversion fluorination system (X-705). Records show that 0.86 MTU of UF_4, 0.4 MTU of UO_2, and 3.3 MTU of UO_3 were received from K-25 in 1957 and classified as recycled uranium for research and development studies. The final disposition of this material could not be determined. It is likely that any materials converted to UF_6 were fed to the cascade, and that unconverted materials were converted to UF_6 during early oxide conversion X-705 operations (probably after 1962). The provenance of the TRU/FP constituents can only be speculated upon. Ash and trapping media disposition records can not be located.

3. X-744G Bulk Storage Warehouse

The X-744G was one of the earliest facilities constructed at PORTS. It was the Peter Kiewit (prime construction contractor) pipe fabrication facility. Here, process piping assemblies were fabricated for cascade and utilities. Interior surfaces of the process piping were also degreased in this facility using organic solvents. Later through most of the early years of plant operation, the building served as the non-UF_6 and small cylinder UF_6 storage area. Overflow materials from X-705 (solutions, UNH, oxides, etc.) were also stored here. Security systems provided protection for HEU materials.

With the advent of the uranium scrap returns program (circa 1966), X-744G was designated as the scrap storage warehouse and central receiving facility. Here oxides, UNH, and small cylinders of UF_6 were off-loaded from transport, with non-UF_6 containers opened and sampled (usually in a hood or glovebox), and placed on shelves or in holders for storage and future conversion at X-705. Materials not meeting acceptance criteria were packaged and shipped from this facility. After cessation of oxide conversion operations, the facility remained in service for storage of all oxides until X-345 was placed in service to store HEU materials. Throughout the life of this facility, no concentrating operations could be determined. There was, however, significant material throughput with opportunity for worker interaction.

This facility currently supports the DOE PORTS Uranium Management Center activities. Quantities of materials containing TRU are being stored here as part of this program. These materials were received after the March 1999 timeframe and, as such, are outside of scope for this report. This facility, over time, has also been the center for other activities on a shared basis such as aluminum smelting, waste sorting, etc. None of these activities are considered as relevant to this report.

4. X-345 Special Nuclear Materials Storage Building

X-345 was constructed, circa 1978, and was basically designed to fill the mission of X-744G for storage of HEU materials but in the more secure environment. An additional mission, added after initial construction, was the sampling of HEU UF_6 small diameter cylinders using autoclaves.

5. X-745 Cylinder Storage Yards

X-745 yards (a through h) are the eight UF_6 cylinder storage yards that exist or have existed at PORTS. These yards, at any one time, constitute the majority of the uranium materials at PORTS and contain UF_6 not in process or not shipped (i.e. feed, product and tails). The depleted UF_6 storage areas are of particular interest due to the buildup of naturally occurring uranium daughter products. Breaching of cylinders due to mechanical failures has occurred. There are no known cases of these breechings occurring in cylinders containing RU.

The repeated filling of UF_6 cylinders that contain RU heels without washing/removing the nonvolatile heel from the cylinder has the effect of concentrating TRU and FP in the heel. The degree to which the constituents concentrate depends primarily on the amounts added with each filling and the fraction removed through feeding. While the exact fraction removed is variable, it is assumed for this study to be as shown in Table 2.2.5-1.

It is known that large numbers of 2-1/2 and 10-ton cylinders were filled with RU off-site (PGDP, ORGDP) and fed at PORTS. These cylinders were either returned empty or filled with non-TRU (tails) and sent back. The potential exists for several such cycles to have been made on a very few cylinders either with RU before they were cleaned, or, they never have been cleaned and the TRU heel remains.

Table 2.2.5-1

Percentage of TRU/FP Removed from Cylinders During Feeding

Constituent	Removal Fraction With Each Feeding
^{237}Np	33% (Refs. 2 and 2a)
^{239}Pu	10% (Refs. 2 and 2a)
^{99}Tc	90% (Refs. 2a and 19)

PORTS has had a closed system for small diameter (i.e., 5", 8", and 12") cylinder cleaning since the X-705 facility startup in 1957. The capability to clean large (2-1/2, 10, 14-ton) cylinders was established through facility modifications made circa 1970. For this study, no TRU/FP is considered to have entered the PORTS site from cleaning of interplant RU cylinders.

Small diameter cylinders were used at PORTS to contain UF_6 (usually HEU) from X-705 oxide conversion. Specific cylinders were used for this flow and were selected for use based on uranium assay of last use so as to minimize mixing losses with the heel. Since on at least two known occasions (i.e., 1974 and 1976) oxide conversion produced UF_6 from oxides containing elevated TRU levels, the UF_6 produced is considered to contain TRU's. The small diameter cylinders used for this production may have been used several times without cleaning. It is probable that concentrating of ^{239}Pu and ^{237}Np took place in these cylinders. Records to corroborate this have not been found, other than records noting the presence of TRU contamination at the facility. There were records of small diameter cylinders being cleaned in about the correct timeframes, but records of specific cylinder numbers cleaned have not been found.

Several small diameter UF_6 cylinders containing HEU RU were received at PORTS from offsite sources {France, NUMEC, Department of International Affairs, Babcock & Wilcox (B&W), and United States Atomic Energy Commission Safeguards and Security (USAEC S&S)}. The majority of this material was only recently fed (circa 1997) to the cascade, and the cylinders with heels were either cleaned in X-705 or Nuclear Fuel Services (NFS). Solutions generated from the recent cleanings in X-705 appear not to have ever been analyzed for TRU. Due to X-705 operational problems, the majority of these solutions (blended down to <5% enrichment) appear to never have been processed and remain in storage. If warranted, they could be sampled and analyzed to ascertain TRU content. The disposition of the cylinders sent to NFS and their solutions are unknown.

2.2.6 Intrafacility Flow of TRU/FP Constituents

Opportunities for cross flow or cross contamination between various streams are worthy of mention. Examples are:

1. Dissolved oxide conversion filter ash reintroduced into uranium recovery;

2. Dissolved ash from personal protective equipment (PPE) and other incinerated burnables, and subsequently, introduced into uranium recovery;

3. X-705 solution complexing to aid recovery of uranium from NaF with the use of alumina, and processing through uranium recovery;

4. Field decontamination solutions originating from TRU contaminated equipment being processed through uranium recovery; and

5. Wastes generated from laboratory operations involving TRU being introduced into uranium recovery.

Any of these intermittent batch operations, as well as perhaps others, could have caused RU to appear at a time other than when RU operations were known to have occurred. The overall effect of these flows would primarily be a slight, but unquantified, increase of chances for personnel exposure.

2.3 Activity Summaries (concentrating processes and other site specific issues related to processing and plants)

The Table 2.3-1 summarizes concentrating processes and site specific issues for PORTS.

Table 2.3-1

Concentrating Processes at PORTS

Concentrating Process/Location	Dates of Operation	Comment
X-344 UF_4 – UF_6 Tower Ash	5/58 – 2/62	Operation only on non-RU (virgin) material (normal assay)
X-344 UF_4 – UF_6 Filter Ash	5/58 – 2/62	Operation only on non-RU (virgin) material (normal assay)
X-705 U_3O_8 – UF_6 Tower Ash	2/57 – 7/77	TRU's known to be processed Jan./Feb. 1974 – ICPP, and Jan. & May 1976 – NLO
X-705 U_3O_8 – UF_6 Filter Ash	2/57 – 7/77	TRU's known to be processed Jan./Feb. 1974 – ICPP, and Jan. & May 1976 – NLO
GDP Cascade Operations	Entire Period	RU constituents concentrated at feed points
GDP Cascade Operations	Pre 1975	Concentrated ^{99}Tc in purge cells
X-326 MgF_2 ^{99}Tc Traps	After 1975	Successfully removed ^{99}Tc in top of cascade
X-701B Holding Pond	Before 1984	Collected ^{99}Tc from X-705 solution recovery
X-705 Heavy Metals Precipitation	1984 and after	Collects current ^{99}Tc from X-705 solution recovery
Site Specific Issues		
HEU refeed of RU-UF_6	1/97 – 6/98	1.1 MTU French 0.3 MTU NUMEC
Side Purge Fire	12/98	Possible involvement with TRU materials

2.4 Activities where workers were likely to be in contact with recycled uranium through direct physical contact or airborne dust.

Table 2.4.1 summarizes activities where workers were likely to have been in contact with RU through direct physical contact or airborne dust at PORTS.

Table 2.4-1

Activities at PORTS Where Workers Were Most Likely to Contact RU

Activity	Comment	TRU/FP
Oxide Sampling (X-744G)	Done in hood	TRU
Oxide Analysis (X-710)	Generally done in hood	TRU
Removal and disassembly of cascade equipment near TRU feed points	In process building and X-705 high bay	TRU
Oxide Conversion Operation (X-705)	High airborne uranium levels	TRU
Oxide Conversion Maintenance (X-705)	Filter handling for ash clean-out, ash handling	TRU
MgF_2 and Alumina Trap Change-out and Media Handling (X-326)	Maybe HEU removal issue	FP
X-701B Sludge Handling	Pond sludge material to Envirocare	FP
X-705 Heavy Metals Precipitation Handling	Shipped to Envirocare	FP

2.5 Activities that caused reportable environmental releases of recycled uranium constituents

The only record of reported environmental releases of TRU elements occurred between October 1976 and March 1977. One sample above the minimum detection limit (MDL) was detected in the outfall from X-701B. A concentration of Np and Pu of 3.7×10^{-8} uCi/ml was measured which is just slightly above the MDL (at that time) of 2.3×10^{-8} uCi/ml. Activities ongoing on or before this time were recovery of uranium solutions and fluorination of oxides that later were discovered to have TRU constituents. Specific containers and sources of specific TRU constituents could not be accurately determined due to timing and cross flow anomalies. It is speculated that the source was the NLO oxides processed in Jan./May 1976.

Measurable and reported quantities of ^{99}Tc released to the environment have occurred since initial recognition of the concern as shown in Table 2.5-1.

As discussed earlier in this report, the activities resulting in surface water releases are those associated with recovery of uranium in X-705 and any episodes where solutions may have bypassed uranium recovery. Since uranium recovery processed materials generated from a myriad of internal plant sources (large and small parts cleaning, cylinder cleaning, field decontamination. etc.), the attribution to each source is somewhat speculative. The vast majority of these surface water releases are, however, felt to have originated from large parts (tunnel) operations where cascade equipment (especially converters) involved in CIP/CUP was processed.

Airborne releases of ^{99}Tc are felt to be predominantly attributable to cascade purging operations (top and side) when trap media became saturated or when trap media were ineffective due to extremely low concentrations of ^{99}Tc.

Environmental monitoring for TRU/FP appears to have been largely nonexistent prior to the mid 1970's. Beginning with the discovery of ^{99}Tc in the east drainage ditch (outfall 001) during the first quarter of FY 1975, sampling activities ramped up significantly. Since that discovery, and continuing until 1995, annual environmental reports have been issued that quantify ^{99}Tc releases and report any detection of TRU. Monitoring of air and water were and are conducted using both fixed sampling instruments and programmatic grab samples. Initially, all surface water sampling was manual. Current systems are capable of either time proportional or flow proportional sampling.

Current outfall sampling is done on a continuous basis, with seven day composites analyzed for gross alpha, beta, uranium, and ^{99}Tc.

Air monitoring has taken a similar evolutionary course. As part of the original complement of equipment, ionization chamber instruments (space recorders) were installed on cascade vents for detecting UF_6 releases. These later served to also detect the presence of ^{99}Tc in-as-much as the equipment would become dysfunctional in the presence of ^{99}Tc due to a rapid buildup in background radiation reading. A continuous sampling system installed during FY 1984 takes flow proportional samples and passes the gas through small calibrated alumina traps. Weekly (or more often if an anomaly occurs), the alumina is changed out and counted using radiochemistry techniques for uranium, ^{235}U, and ^{99}Tc. Data from this process takes as long as two weeks to receive, but is felt to be more accurate for quantifying releases.

Table 2.5-1

Annual Releases of ^{99}Tc to Surface Water and Air at PORTS

Fiscal Year	Annual Releases of ^{99}Tc					
	To Surface Water (Ci)	To Air (Ci)	Total (Ci)	To Surface Water (kg)	To Air (kg)	Total (kg)
1975	77.5		77.5	4.5		4.5
1976	19.1		19.1	1.1		1.1
1977	31.0	4.5	35.5	1.819	0.264	2.083
1978	17.7	0.823	18.523	1.039	0.048	1.087
1979	2.790	0.170	2.96	0.164	0.010	0.174
1980	7.740	0.210	7.95	0.454	0.012	0.467
1981	24.720	0.108	24.828	1.451	0.006	1.457
1982	11.840	11.100	22.94	0.695	0.651	1.346
1983	2.990	0.561	3.551	0.175	0.033	0.208
1984	9.340	0.127	9.467	0.548	0.007	0.556
1985	8.450	0.123	8.573	0.496	0.007	0.503
1986	2.480	0.122	2.602	0.146	0.007	0.153
1987	1.220	0.169	1.389	0.072	0.010	0.082
1988	0.870	0.162	1.032	0.051	0.010	0.061
1989	0.480	0.096	0.576	0.028	0.006	0.034
1990	1.350	0.049	1.399	0.079	0.003	0.082
1991	0.330	0.044	0.374	0.019	0.003	0.022
1992	0.210	1.230	1.44	0.012	0.072	0.085
1993	0.580	7.830	8.41	0.034	0.459	0.494
1994	0.167	0.122	0.289	0.010	0.007	0.017
1995	--	0.0147	0.0147		0.001	0.001
TOTAL	221	28	249	12.98	1.62	14.6

3. RECYCLED URANIUM MASS FLOW

3.1 Uranium Recycle Description

A diagram to illustrate the RU material mass flow at PORTS is shown in Figure 3.1-1. The total quantity received from each source and the total quantity shipped to each receiver is shown. The center of the diagram represents the several processes the RU material may have passed through after receipt. There is a large difference between the quantity received and the quantity shipped. This difference is due to the diluting nature of the processes at PORTS. When RU is fed to the cascade, the quantity fed is only a small fraction of the total amount of uranium present in the cascade. Once fed, the RU is mixed with the other material already present, and can no longer be tracked based upon the original uranium content. Each constituent is separated from the original uranium and follows a different path through the PORTS cascade and other facilities. Therefore, PORTS tracks RU only until it loses it's unique identity; from that point, PORTS tracks each constituent of TRU and ^{99}Tc individually to show the constituents' mass flow and to perform a mass balance.

Recycled uranium was first introduced at PORTS in FY 1955 as UF_6 feed manufactured at Paducah from UO_3 received from Hanford (HRT) and Savannah River (SRT) reactor tails. Also in FY 1955 PPF was provided for PORTS feed. The PPF was contaminated with ^{99}Tc at an estimated 1 ppm (Ref. 2). The UO_3 from HRT/SRT was contaminated with Np, Pu, and ^{99}Tc at an estimated 0.24 ppm, 4 ppb, and 7 ppm, respectively, prior to FY 1967 and 0.09 ppm, 2.2 ppb, and 7 ppm, respectively, thereafter (Ref. 2). After feed was manufactured from the HRT/SRT oxide it was contaminated with Np, Pu, and ^{99}Tc at an estimated 0.18 ppm, 0.04 ppb, and 6.65 ppm, respectively, prior to FY 1967 and 0.068 ppm, 0.021 ppb, and 6.65 ppm, respectively, thereafter (Refs. 2 and 2a).

To illustrate and track the movement of RU, TRU and ^{99}Tc through PORTS, four campaigns which cover all significant events at PORTS from startup in FY 1955 through March 31, 1999 were developed. Each campaign addresses a specific grouping of RU for a specific time period.

The Depleted Reactor Tails – Campaign #1 (Figure 3.1-2), addresses feed manufactured from HRT/SRT oxide and PPF from FY 1955 through FY 1967. The Depleted Reactor Tails – Campaign #2 (Figure 3.1-3) addresses feed manufactured from HRT/SRT oxide and PPF from FY 1968 through March 31, 1999. Note: The bars which extend beyond FY 1978 are assumed to remain constant through March 1999. The Non-UF_6 RU Program – Campaign #3 (Figure 3.1-4) deals with RU of all forms of uranium at PORTS other than UF_6. These campaigns do not include 4.6 MTU of non-UF_6 potentially utilized for development activities in FY 1957. The remaining RU is captured in the UF_6 feed as Miscellaneous Cascade Feed - Campaign #4 (Figure 3.1-5).

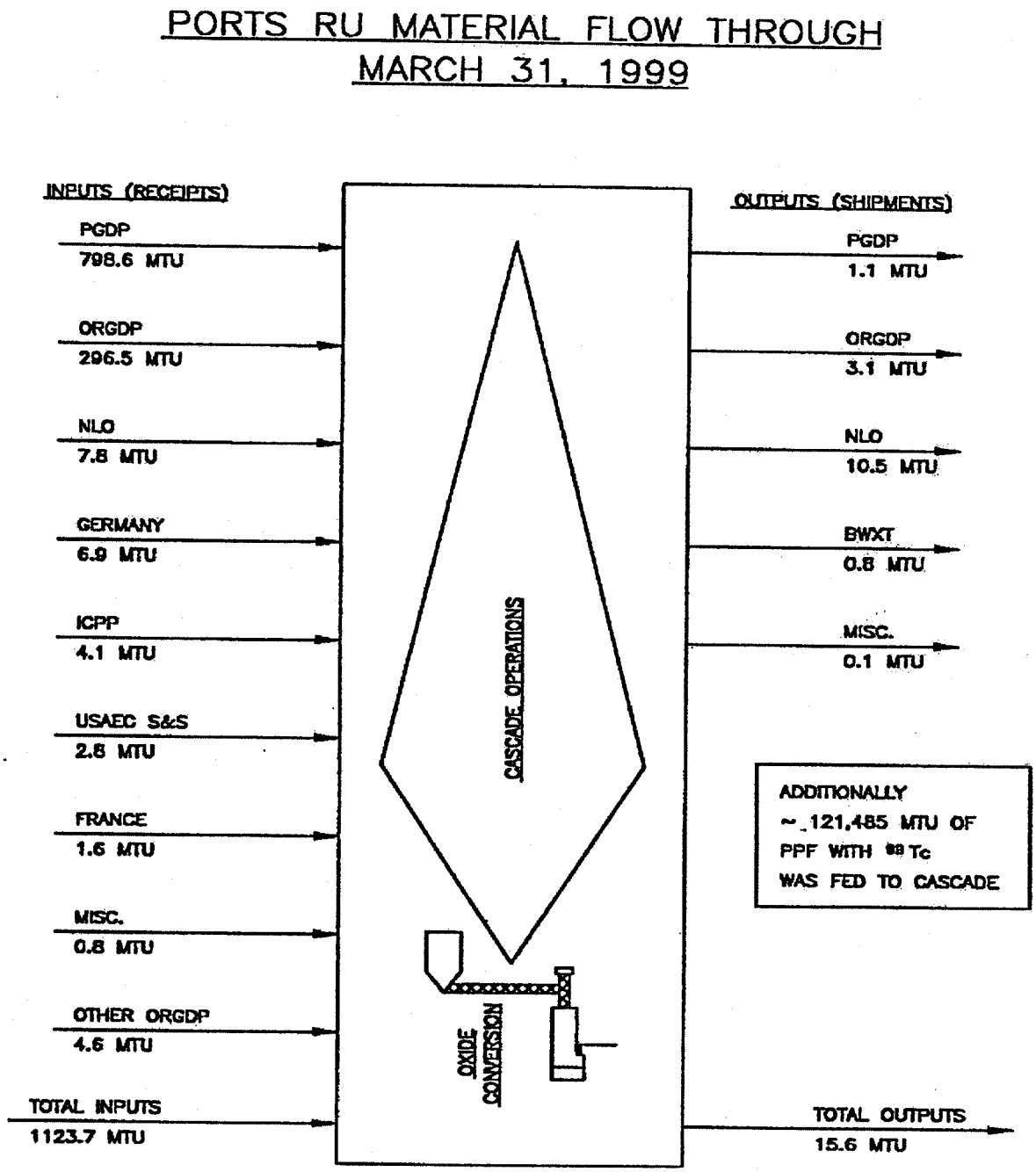
Each campaign shows what is known, estimated or projected regarding RU. Each figure identifies the source of the RU, year(s) received at PORTS, quantity of RU, which process(es) the RU, TRU and ^{99}Tc passed through, and when the material was shipped from PORTS. Significant events that occurred during the period are shown. This method allows for a tabulation of the TRU and ^{99}Tc by year to provide a year-end inventory, and establishes the RU constituent inventory as of March 31, 1999.

The RU, containing TRU and ^{99}Tc, was first introduced between FY 1955 and FY 1958 when approximately 527 MTU of feed manufactured from HRT/SRT oxide was received. This material is estimated to have contained a total of 95g Np, 0.021g Pu, and 3.7kg ^{99}Tc . Also, Paducah feed was utilized beginning in FY 1955 and continues to the present time. Between FY 1955 and FY 1971, ^{99}Tc was present at a concentration of approximately 1 ppm. During this time, 43.5 kg of ^{99}Tc is estimated to have been fed into the PORTS cascade. To establish the annual inventory of ^{99}Tc from Paducah feed, the total quantity received during this period was distributed evenly over the 17-year period. Various sections of this report discuss in detail specific plant facilities which processed/concentrated RU, TRU, and ^{99}Tc and will not be repeated here except as required to describe the flow of the RU and its constituents through each campaign.

3.1.1 Campaign #1

During Campaign #1, the TRU and ^{99}Tc contaminants that entered the cascade with feed manufactured from HRT/SRT oxide and early PPF were substantially removed during the first cascade change-out program. It is assumed that during this period, material was fed upon receipt and the empty cylinders with heels were returned to Paducah/Oak Ridge. These cylinders were not cleaned at PORTS; therefore, any TRU, RU, and ^{99}Tc contained in the heels went to Paducah or Oak Ridge.

Figures 3.1-1



DEPLETED REACTOR TAILS
CAMPAIGN #1

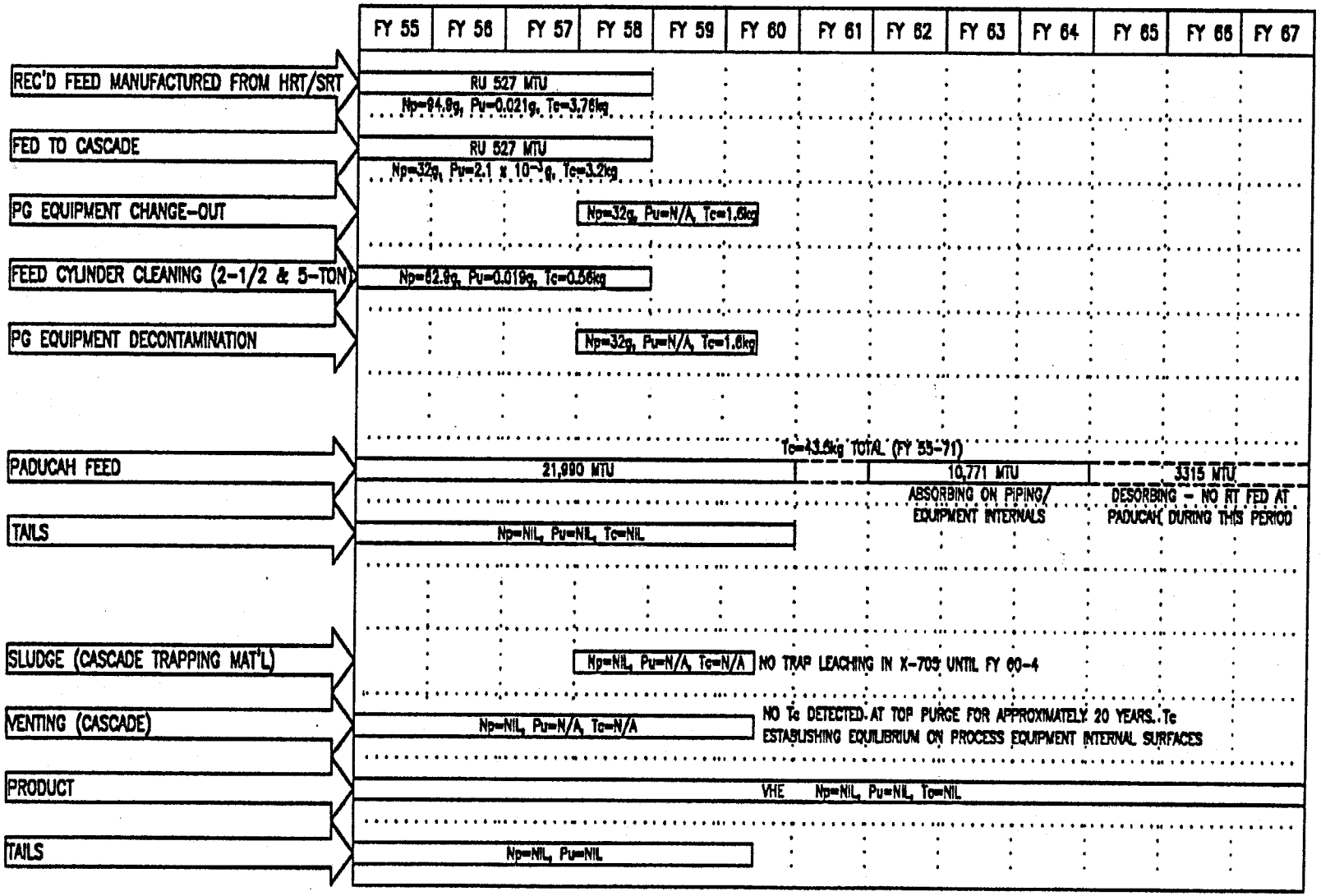


Figure 3.1-2

DEPLETED REACTOR TAILS
CAMPAIGN #2

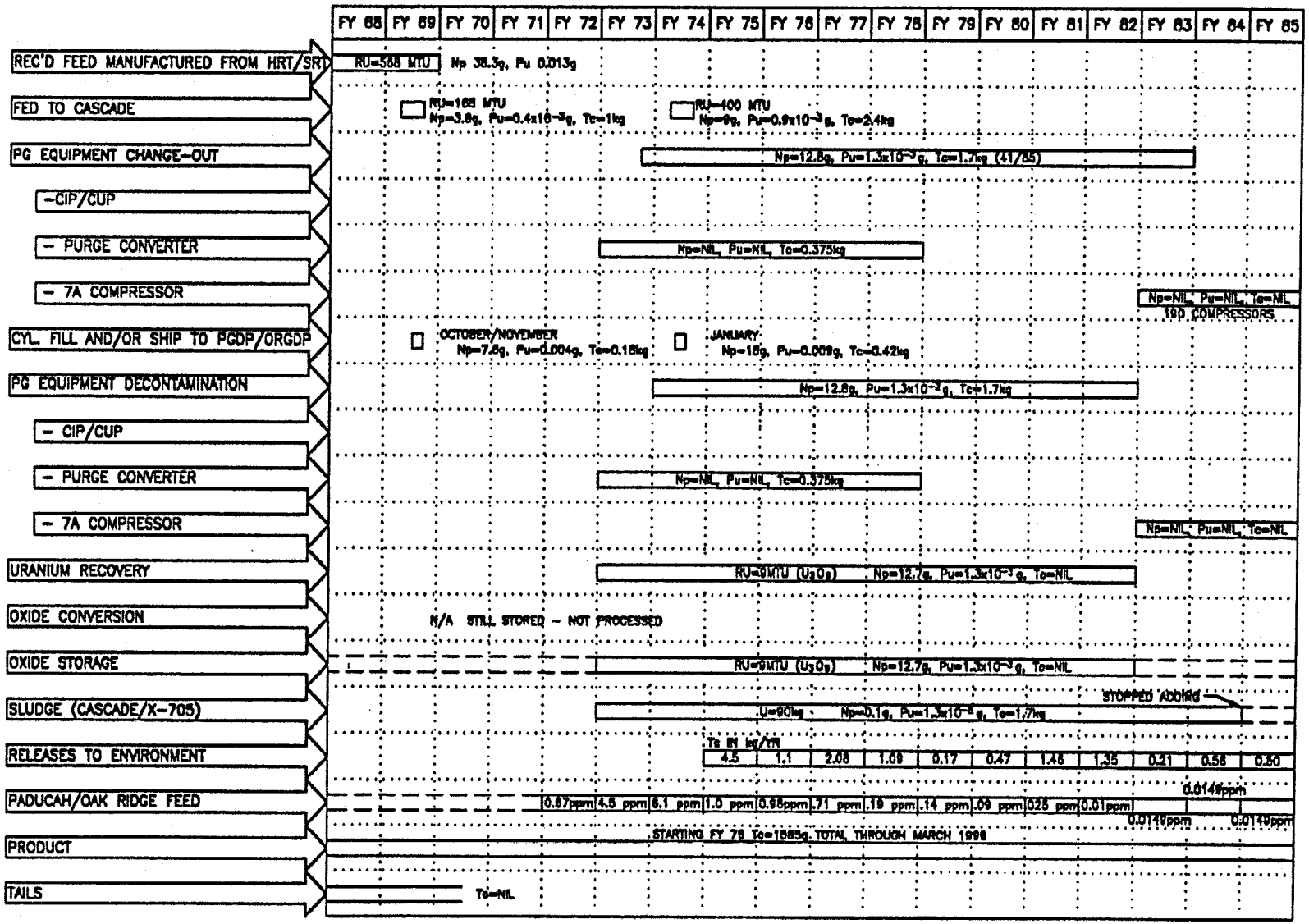


Figure 3.1-3

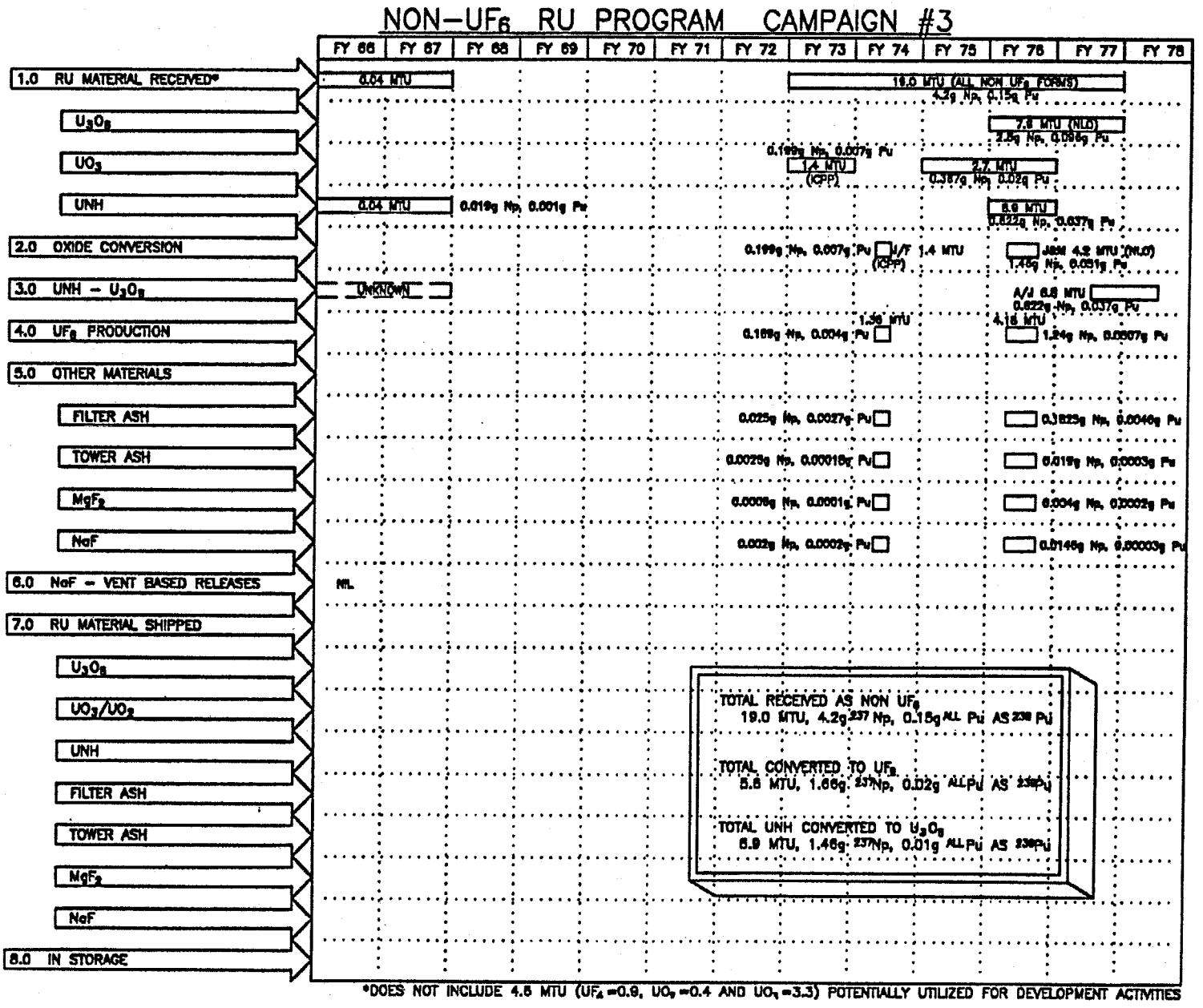

NON–UF6 RU PROGRAM CAMPAIGN #3
FY 66
FY 67
FY 68
FY 69
FY 70
FY 71
FY 72
FY 73
FY 74
FY 75
FY 76
FY 77
FY 78
1.0 RU MATERIAL RECEIVED*
U3O8
UO3
UNH
2.0 OXIDE CONVERSION
3.0 UNH – U3O8
4.0 UF6 PRODUCTION
5.0 OTHER MATERIALS
FILTER ASH
TOWER ASH
MgF2
NaF
6.0 NaF – VENT BASED RELEASES
7.0 RU MATERIAL SHIPPED
U3O8
UO3/UO2
UNH
FILTER ASH
TOWER ASH
MgF2
NaF
8.0 IN STORAGE
0.04 MTU
16.0 MTU (ALL NON UF6 FORMS)
4.2g Np, 0.15g Pu
7.8 MTU (NLO)
2.5g Np, 0.098g Pu
0.199g Np, 0.007g Pu
1.4 MTU
(CPP)
2.7 MTU
0.387g Np, 0.02g Pu
0.04 MTU
0.019g Np, 0.001g Pu
5.6 MTU
0.622g Np, 0.037g Pu
0.199g Np, 0.007g Pu
U/F 1.4 MTU
(CPP)
J&M 4.2 MTU (NLO)
1.46g Np, 0.051g Pu
UNKNOWN
A/J 6.8 MTU
0.622g Np, 0.037g Pu
1.36 MTU
4.16 MTU
0.169g Np, 0.004g Pu
1.24g Np, 0.0607g Pu
0.025g Np, 0.0027g Pu
0.3825g Np, 0.0046g Pu
0.0025g Np, 0.00018g Pu
0.019g Np, 0.0003g Pu
0.0008g Np, 0.0001g Pu
0.004g Np, 0.0002g Pu
0.002g Np, 0.0002g Pu
0.0146g Np, 0.00003g Pu
NIL
TOTAL RECEIVED AS NON UF6
19.0 MTU, 4.2g 237Np, 0.15g ALL Pu AS 239Pu
TOTAL CONVERTED TO UF6
5.6 MTU, 1.66g 237Np, 0.02g ALL Pu AS 239Pu
TOTAL UNH CONVERTED TO U3O8
6.9 MTU, 1.46g 237Np, 0.01g ALL Pu AS 239Pu
*DOES NOT INCLUDE 4.6 MTU (UF4 =0.9, UO2 =0.4 AND UO3 =3.3) POTENTIALLY UTILIZED FOR DEVELOPMENT ACTIVITIES

Figure 3.1-4

MISCELLANEOUS CASCADE FEED
CAMPAIGN #4

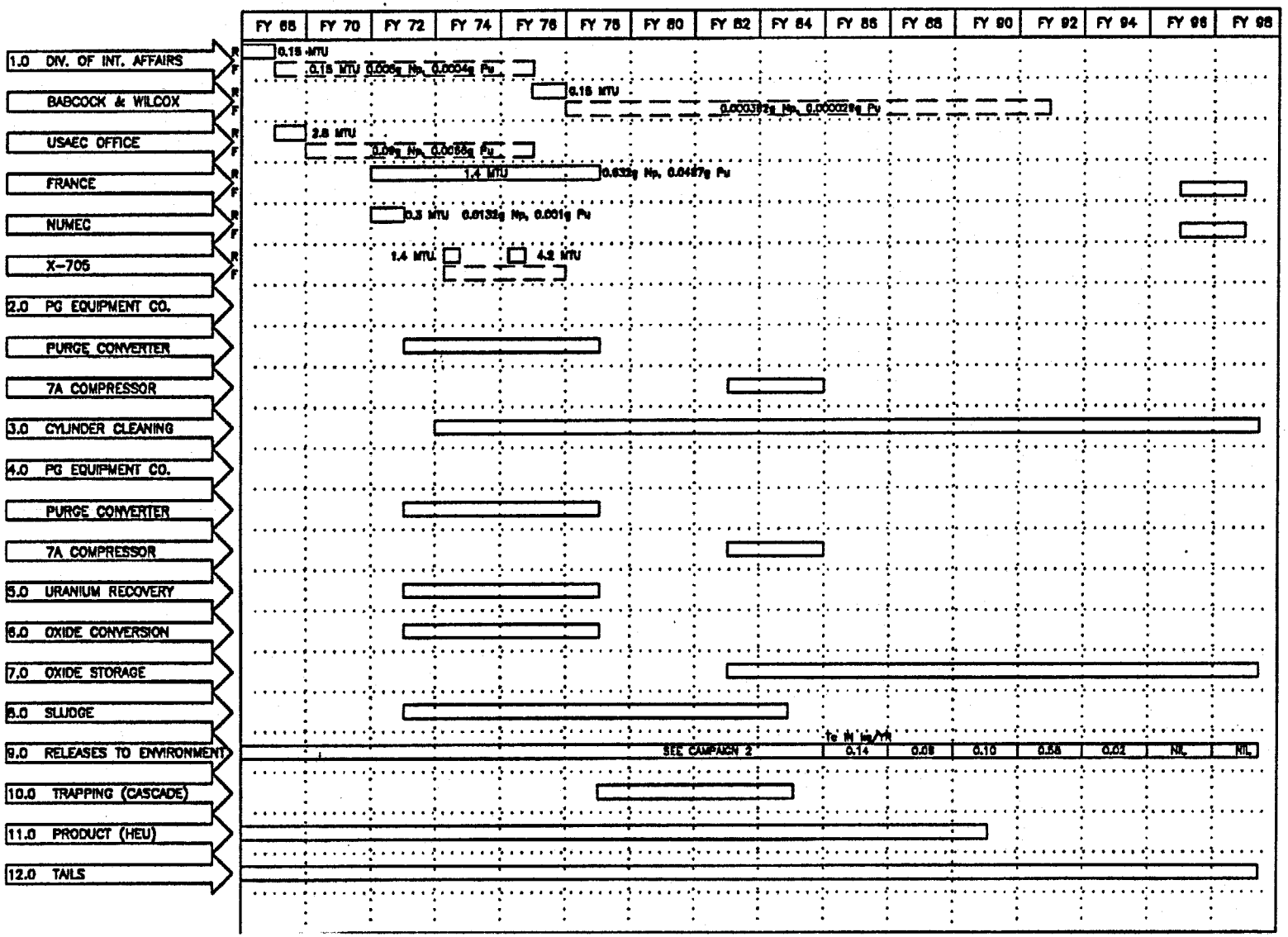


Figure 3.1-5

When RU/FP material was fed to the cascade, a portion of each constituent entered the cascade with the UF_6 while the balance remained in the cylinder. The split is assumed to be as shown in Table 2.2.5-1.

The barrier is assumed to contain essentially all of the TRU and was contaminated with Np, Pu, and ^{99}Tc at an estimated 0.24 ppm, 4 ppb, and 7 ppm, respectively, prior to FY 1967 and 0.09 ppm, 2.2 ppb, and 7 ppm, respectively, thereafter (Ref. 2) that was removed during the equipment change-out. This barrier was decontaminated in the X-705 large equipment tunnel, where essentially all of the TRU and ^{99}Tc was assumed to go into solution. This process removed essentially all Np and Pu from the PORTS cascade and approximately ½ of the ^{99}Tc introduced through FY 1959. The ^{99}Tc from the PPF is assumed to continue to absorb on cascade surfaces until it reaches equilibrium. No record of ^{99}Tc releases or its presence in the product were found to have occurred during this campaign.

3.1.2 Campaign #2

In Campaign #2, the barrier is again assumed to contain essentially all of the TRU and ^{99}Tc that was removed during the CIP/CUP and purge converter change-outs. The CIP/CUP change-out replaced equipment that contained the Np and Pu and a portion of the ^{99}Tc. The Np and Pu remain near the feed point, while ^{99}Tc may be found almost anywhere above the feed point. For this reason, only ^{99}Tc is shown as present in the purge converters changed-out. During this campaign, the barrier and other removed equipment were decontaminated. The TRU and ^{99}Tc are assumed to go into solution. The decontamination and processing of the decontamination solutions are assumed to have been performed in the same period as the equipment removal. All of the ^{99}Tc is assumed to go into the raffinate or traps. Ninety-nine percent of the Np and Pu are assumed to end up in the oxide produced.

Beginning with FY 1972, additional data on PPF became available and this was used to calculate the quantity of ^{99}Tc present. Starting with FY 1976, the product produced at PORTS is known to contain ^{99}Tc. An average of about 2% of the total amount of ^{99}Tc remaining in the cascade is estimated to be removed annually. It is estimated that a total of 1,585g of ^{99}Tc was removed in the product stream through March 1999. In FY 1975, quantities of ^{99}Tc that were detected in air/water releases to the environment were removed from PORTS inventory. The ^{99}Tc is shown as removed from PORTS inventory in the year the product is withdrawn from the cascade.

The HRT/SRT received in FY 1968 and 1969 was not all fed immediately. Therefore, the RU appears in the year-end inventory until fed. The TRU and ^{99}Tc fed to the cascade during this period is assumed to have been almost completely removed during the CIP/CUP program and/or purge converter change-out. No significant quantity of material is believed to have been removed during the 7A compressor change-out.

3.1.3 Campaigns #3 and #4

The UF_6 from the miscellaneous cascade feed and non-UF_6 scrap returns are summarized in the last two campaigns. The RU and contaminants are assumed to be fed or processed uniformly over the period from the earliest feed/processed date to the latest feed/processed date. Some of this material remains in storage.

3.2 Uranium Receipts

See Figure 3.1-1 and Table 3.2-1 for a summary of the RU received each FY and its source. A total of 1,123.7 MTU of RU (all forms) was received at PORTS. The table does not include Paducah or Oak Ridge product feed, which PORTS considers to be ^{99}Tc contaminated, but not RU. However, the mass flow includes the ^{99}Tc constituent of these PORTS feeds.

Table 3.2-1

PORTS Receipts Summary (RU Only)

		Net Weight (kgU)								
Shipping Facility	**Uranium Form**	**FY 1955**	**FY 1956**	**FY 1957**	**FY 1958**	**FY 1966**	**FY 1967**	**FY 1968**	**FY 1969**	**FY 1972**
Allied Chemical	UO_3									
Babcock & Wilcox	UF_6									
Division of International Affairs	UF_6							151		
	UNH					7	39			
Fernald	U_3O_8									
France	UF_6									65
Germany	UNH									
K-25	UF_4			865						
	UF_6		296, 504							
	UO_2			418						
	UO_3			3,319						
NUMEC	UF_6									330
Paducah	UF_6	105,873	54,649	6,156	64,311				567,620	
United Kingdom	UNH									
USAEC Office Safeguards & Materials. Management.	UF_6								2,833	
Y-12	U_3O_8									
Grand Total		**105,873**	**351,154**	**10,758**	**64,311**	**7**	**39**	**151**	**570,453**	**395**

Table 3.2-1 (Cont'd)

PORTS Receipts Summary (RU Only)

		Net Weight (kgU)							
Shipping Facility	**Uranium Form**	**FY 1973**	**FY 1974**	**FY 1975**	**FY 1976**	**FY 1976.5**	**FY 1977**	**FY 1978**	**Grand Total**
Allied Chemical	UO_3	1,376		1,403	1,295				4,074
Babcock & Wilcox	UF_6						153		153
Division of International Affairs	UF_6								151
	UNH								46
Fernald	U_3O_8				7,798				7,798
France	UF_6	202	324	128	273	112	152	235	1,586
Germany	UNH				6,860				6,860
K-25	UF_4								865
	UF_6								296,505
	UO_2								418
	UO_3								3,319
NUMEC	UF_6								330
Paducah	UF_6								798,609
United Kingdom	UNH			7					7
USAEC Office Safeguards & Materials. Management.	UF_6								2,833
Y-12	U_3O_8						104		104
Grand Total		**1,578**	**324**	**1,538**	**16,226**	**112**	**409**	**235**	**1,123,658**

3.3 Uranium Shipments

See Figure 3.1-1 and Table 3.3-1 for a summary by FY of the RU shipments from PORTS each FY and the receiving facility. PORTS shipped a total of 15.6 MTU of RU. The table does not include Paducah or Oak Ridge product feed, which PORTS considers to be ^{99}Tc contaminated, but not RU.

Table 3.3-1
PORTS Shipment Summary (RU Only)

Receiving Facility		Net Weight (kgU)						
	Uranium Form	FY 1955	FY 1956	FY 1972	FY 1974	FY 1982 - 1984	FY 1998	Grand Total
B&W	UO_3						800	800
France	UF_6			65				65
K-25	UF_6		3,102					3,102
NLO	U_3O_8					10,500		10,500
Paducah	UF_6	920	582		(368)			1,134
Grand Total		920	3,684	65	(368)	10,500	800	15,601

3.4 Recycled Uranium Waste

Central to the assumptions of this study is the concept of RU losing its identity through processing or treatment (i.e., the RU is blended with usually enormous amounts of non-RU resulting in product and tails streams containing deminimus quantities of RU). Wastes, therefore, are not classified as RU wastes, but rather wastes potentially contaminated with either TRU or FP. Such materials as alumina, NaF, and MgF_2 trapping media, contaminated pump oils, tower ash, and filter ash could constitute either waste or scrap depending upon the economics of processing and values of the recovered uranium. Holding pond and heavy metal sludges and ion exchange resins would constitute wastes from uranium recovery.

Quantification of the TRU/FP component of all of these streams could not be reliably accomplished within the time constraints of this report. Data on holding pond sludges have already been discussed. Data on filter ash have also been discussed. Some alumina and ion exchange resin data has been located, but not reviewed. NaF data remain to be discovered.

3.5 Recycled Uranium Scrap

For this study PORTS RU scrap is defined as RU scrap that was received from various sources either for conversion to UF_6 but was never converted to UF_6, or as RU-UF_6 feed that was never fed. Materials such as uranium heels in UF_6 cylinders that contained RU would meet this definition. There were 0.8 MTU of RU heels returned to PGDP and 0.8 MTU of RU heels returned to ORGDP. In addition, oxides (U_3O_8) produced from uranium recovery that contain TRU/FP could conceivably be considered RU scrap. In that regard, 0.85 MTU of highly enriched uranium oxides were shipped to BWXT during the HEU removal program. An unknown quantity of LEU oxides remain on site that potentially contain TRU/FP and may be considered scrap or waste depending upon the economics of processing and value of the recovered uranium.

3.6 Inventory as of March 31, 1999

A total of 8.3 MTU of RU (all forms) was in inventory at PORTS as of March 31, 1999. Table 3.6-1 shows the breakout by uranium form and includes the source of the material and the amount of uncertainty included in the inventory.

Table 3.6-1

PORTS March 31, 1999 Inventory of RU

Source Facility	Form	Amount Received (MTU)	Conv to UF_6 (MTU)	Conv to U_3O_8 (MTU)	UF_6 Fed to Cascade (MTU)	Amount Shipped (MTU)	Amount in Inventory 03/31/99 (MTU)	Inventory Uncertainty 03/31/99 (MTU)
Allied (ICPP)	UO_3	4.08	1.4		1.4	0.8	1.8	0.08
B&W	UF_6	0.15			0.15			0
Div. of I.A.	UF_6	0.15			0.15			0
	UNH	0.04						0.04
Fernald	U_3O_8	7.8	4.2		0.46	3.6	3.74	0
France	UF_6	1.6			1.1	0.01		0.4
Germany	UNH	6.9		6.9		6.9		0
NUMEC	UF_6	0.33			0.33			0
ORGDP	UF_4	0.86						0.86
	UF_6	296.5			293.4	3.1		0
	UO_2	0.4						0.4
	UO_3	3.3						3.3
PGDP	UF_6	798.6			797.5	1.1		0
USAEC	UF_6	2.8			0.07		2.73	0
Y-12	U_3O_8	0.1						0.1
TOTAL		1123.61	5.6	6.9	1094.56	15.6	8.27	5.18

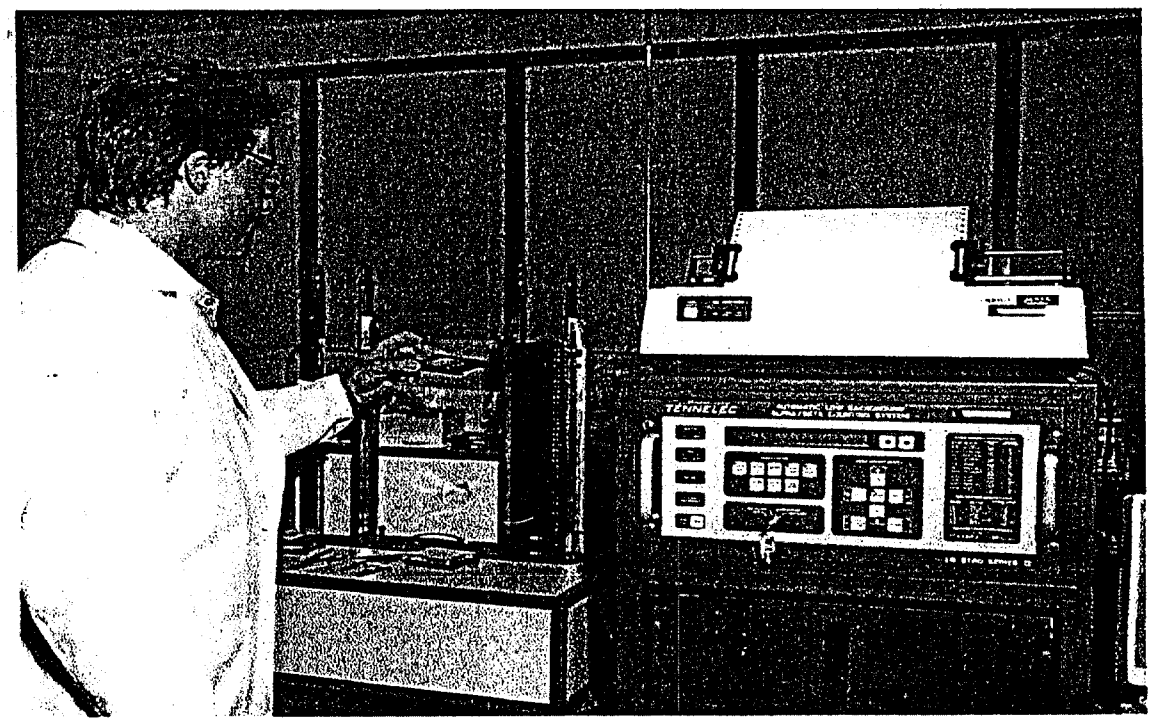
4. CONSTITUENTS IN RECYCLED URANIUM

4.1 Analytical Laboratories

There were several laboratories that analyzed samples for TRU or ^{99}Tc. These included the X-705 laboratory which, for production control purposes, analyzed recovery solutions for ^{99}Tc and oxides from the Oxide Conversion Facility, the Radiochemistry Department in the X-710, which performed more sophisticated TRU and ^{99}Tc analysis, and the Materials Sampling and Testing Department in the X-710, which did sample preparation work. Other areas within the laboratory undoubtedly handled samples that contained TRU's or ^{99}Tc as unknowns. ^{99}Tc was first detected in the laboratory as an unknown in 1974 and was confirmed as ^{99}Tc in 1976; the compound pertechnetyl fluoride ($^{99}TcO_3F$) was identified by infra-red analysis in 1977 (Ref. 21). While TRU analysis dated back at least as early as the early 1970's, no TRU analyses were regularly performed from 1984 to 1992. Generally, there is no TRU analysis being done on feeds or withdrawals from the cascade with the exception of Russian feed. A ^{99}Tc analysis is being performed on all feeds and withdrawals, and the results are easily retrievable, at least for the period of January 1, 1995 to date.

4.1.1 Analytical Procedures

A search of historical procedure manuals in the X-710 laboratory revealed one procedure "Analysis of Np and Pu Alpha Activity in Uranium Compounds" dating from 1976. This was a revision of an earlier procedure and is the basis of the procedure currently used. Two laboratory procedure manuals from the 1980's were found. They include the procedure previously mentioned, as well as procedures for ^{99}Tc in water and soils, Np and Pu in water and soils, ^{99}Tc in uranium and solvent raffinates, and ^{99}Tc in cell gases.



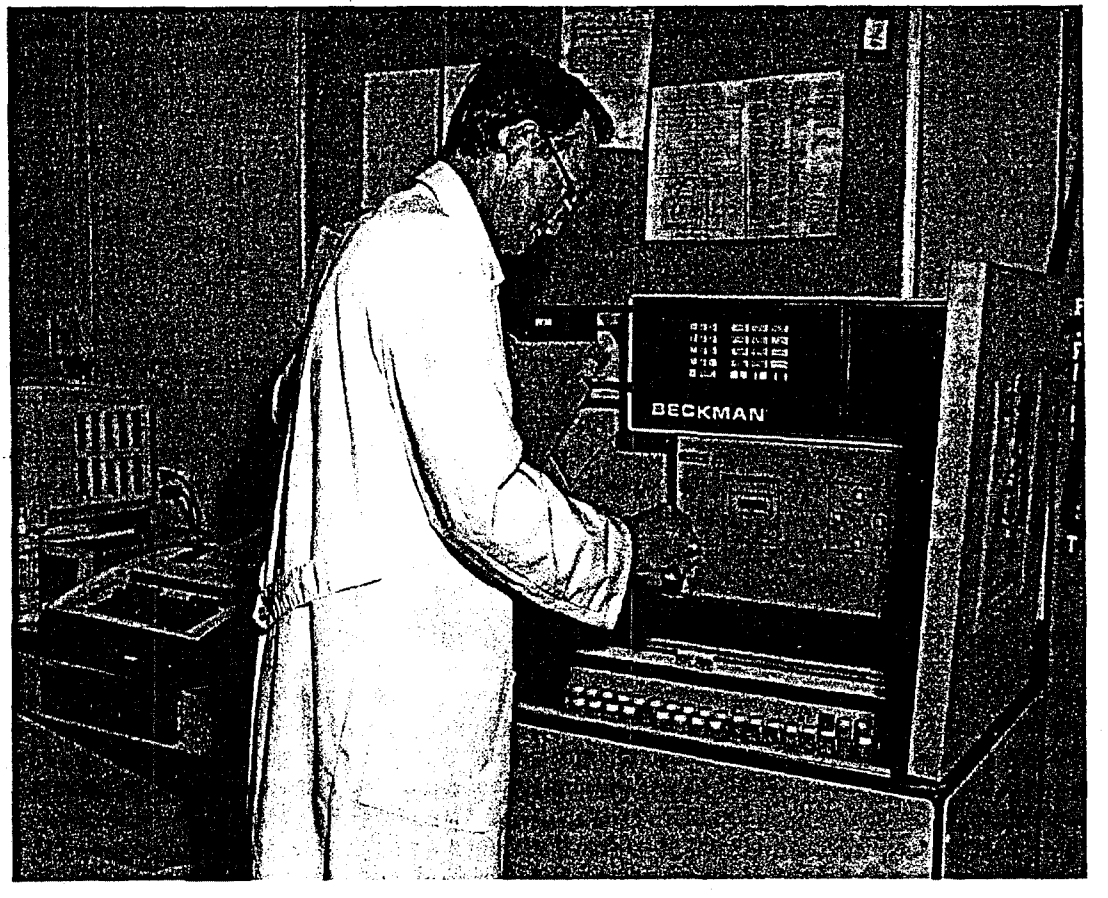
Tennelec Alpha/Beta Counting System in X-710 Labs

4.1.2 Analytical Methods and Errors

The 1976 procedure for Np and Pu in uranium compounds states the relative standard deviation (rsd) as "about 25%". The same procedure in the 1983 manual has the rsd as 10% indicating a refining of the method or better instrumentation. The 1983 manual lists the rsd of ^{99}Tc in raffinates and water as 5% with no estimate for ^{99}Tc in cell gases. A 1988 manual of environmental analytical procedures lists the rsd of ^{99}Tc in water, Np and Pu in water, Np and Pu in soils, and ^{99}Tc in soils as 10%. The current method for ^{99}Tc in UF_6 lists a rsd of 4.58%. The errors associated with these methods could be employed in determining a relative uncertainty value for calculations involving amounts or concentrations of the element of interest. The radiochemistry area employs the use of radioactive tracers in their current methods, with a known amount added to the sample matrix and the sample result adjusted for tracer recovery. This helps reduce errors associated with variables in the method.

4.1.3 Processing Issues

In the history of the laboratory, there were two major changes in the processing of samples for the analysis of TRU and ^{99}Tc. In the time period of 1979 to 1980, ^{99}Tc analysis was changed from a method of extraction, deposition on a planchette, and counting on a proportional counter to a method using liquid scintillation. The new method was quicker, cheaper, more accurate, and had a lower detection limit. The other major change involved the analysis of TRU. In the time period of 1989 to 1990 the advent of TRU-specific ion exchange resins gave an overall improvement in these analyses (Ref. 22). Other processing changes have involved the use of a glovebox for handling oxide samples and the increased use of hoods for samples known or suspected of containing TRU or ^{99}Tc.



Measuring ^{99}Tc Using a Liquid Scintillation Counter in the X-710

4.1.4 Quality Assurance

Quality assurance in the lab is provided by National Institute of Science and Technology (NIST) traceable standards, duplicate and spike analysis, and a program of blind and known controls. Control samples come from both an internal standards lab and external labs and inter-lab "round robin" testing. Control charts are maintained on a laboratory information system with oversight by plant statisticians. Currently, quality assurance data is available dating back approximately nine years; however, these quality control practices are laboratory and industry standards and have been historically practiced at the facility.

4.2 Analytical Results for Plutonium and Neptunium in Uranium Materials Shipped and Received

There is very little analytical data for TRU in product shipped or received. There was apparently a monthly sampling program in the mid-1970's that analyzed VHE, PPF and tails material for total transuranics and ^{99}Tc (Ref. 20). The few sample results available are "less than" values which indicate that TRU levels were below the detection limit of the method. Based on that data, there is no evidence of Np or Pu contamination in either the PPF or Side Withdrawal streams at that time; the likelihood of TRU contamination was greater in previous years when RU material containing higher levels of TRU was fed (Ref. 23). The ASTM specification C 996-90 for enriched reprocessed uranium states a limit of 200 dpm/gU for alpha activity attributed to Np and Pu. The few sample results available from the mid-70's are, again, "less than" values with detection limits of 5 or 2 dpm/gU. Regarding analytical results in materials received, there was not a rigorous sampling or analytical program for incoming recycled material. Some correspondence exists discussing acceptance of out-of-specification material; however, most of the analytical data that exists is for UF_6 which was manufactured from oxide known to contain TRU (Ref. 24). Much of this material was above ASTM specification limits for feed material. These limits are 1,500 dpm/gU for alpha activity attributable to Np and Pu in both the volatile and non-volatile components in a cylinder, or 200 dpm/gU for strictly volatile components. The range of data for TRU's in UF_6 range from the "less than" values of 2 or 5 dpm/gU to 23,800 dpm/gU for UF_6 produced at the Oxide Conversion Facility.

Sample data (Ref. 23a) covering the period February 1977 through May 1977 indicate that all VHE product, side withdrawals, and tails withdrawals had less than detectable quantities of TRU present. For this time frame, the laboratory instruments and procedures utilized claimed a 5 dpm/gU (2.25 pCi/gU) minimum detection level for total TRU. For the isotopes of concern, ^{239}Pu and ^{237}Np, this equates to concentration levels of 0.037 ppb and 3.19 ppb, respectively.

Deminimus levels of constituents are defined in the project plan as those resulting in no more than a 10% increase in health effect due to inhalation of the constituents present in the base uranium. The levels at which this occurs for ^{239}Pu and ^{237}Np is 2.17 ppb and 189 ppb, respectively, for the worst case of soluble uranium at tails enrichment levels. As enrichment levels increase the allowable constituent concentrations increase. Thus, ^{239}Pu and ^{237}Np constituents at 5 dpm/gU are below the 10% incremental health impact. Any combination of the two isotopes with a total activity below 5 dpm/gU must, also result in a determination of deminimus. The conclusion is that it is likely that all UF_6 product (VHE, and side withdrawals) and all tails withdrawn from the enrichment plant contained less than deminimus levels of TRU constituents ^{239}Pu and ^{237}Np.

4.3 Analytical Results for ^{99}Tc in Uranium Materials Shipped or Received

The ASTM specification for ^{99}Tc in enriched reprocessed UF_6 is 5ug/gU; however, the measurement of ^{99}Tc is not required unless the ^{236}U level is above 2,500 ug/gU or another control level agreed upon by the buyer and seller. The ^{99}Tc contamination product has, at times, been a problem with extra efforts sometimes needed to produce in-specification material. Within the past five years, estimates of total grams of ^{99}Tc shipped from PORTS have been in the range of less than 10 grams for a year with the estimates of total amount received from Paducah to be in the range of 20 to 200 grams a year (Ref. 25). Data from the time RU was being fed is sparse and there does not appear to have been a rigorous sampling program for incoming material for either TRU or ^{99}Tc. The data that is available is from sampling of tails, PPF, and VHE is in the range of less than 0.0002 ugTc/gU to 0.69 ugTc/gU (Ref. 20). The ^{99}Tc contamination in cascade equipment has been an acknowledged problem since the mid-1970's. Some

materials removed from the cascade at areas known to concentrate ^{99}Tc have been as high as 40% by weight ^{99}Tc (Ref. 26).

5. MASS BALANCE ACTIVITIES

5.1 Annual Mass Balance of Recycled Uranium

From startup in FY 1955 through March 31, 1999, approximately 371,000 MTU of uranium (all forms) received at PORTS, with 1,123.7 MTU determined to contain RU. Of this RU, 15.6 MTU are known to have been shipped from PORTS. The balance of the RU was either processed as discussed in other sections of this report or remains in storage.

The annual inventory of RU constituents is shown in Table 5.1-1 and Figure 5.1-1. The data sources are discussed later in Section 5.6

Table 5.1-1

Annual Inventory Of RU Constituents

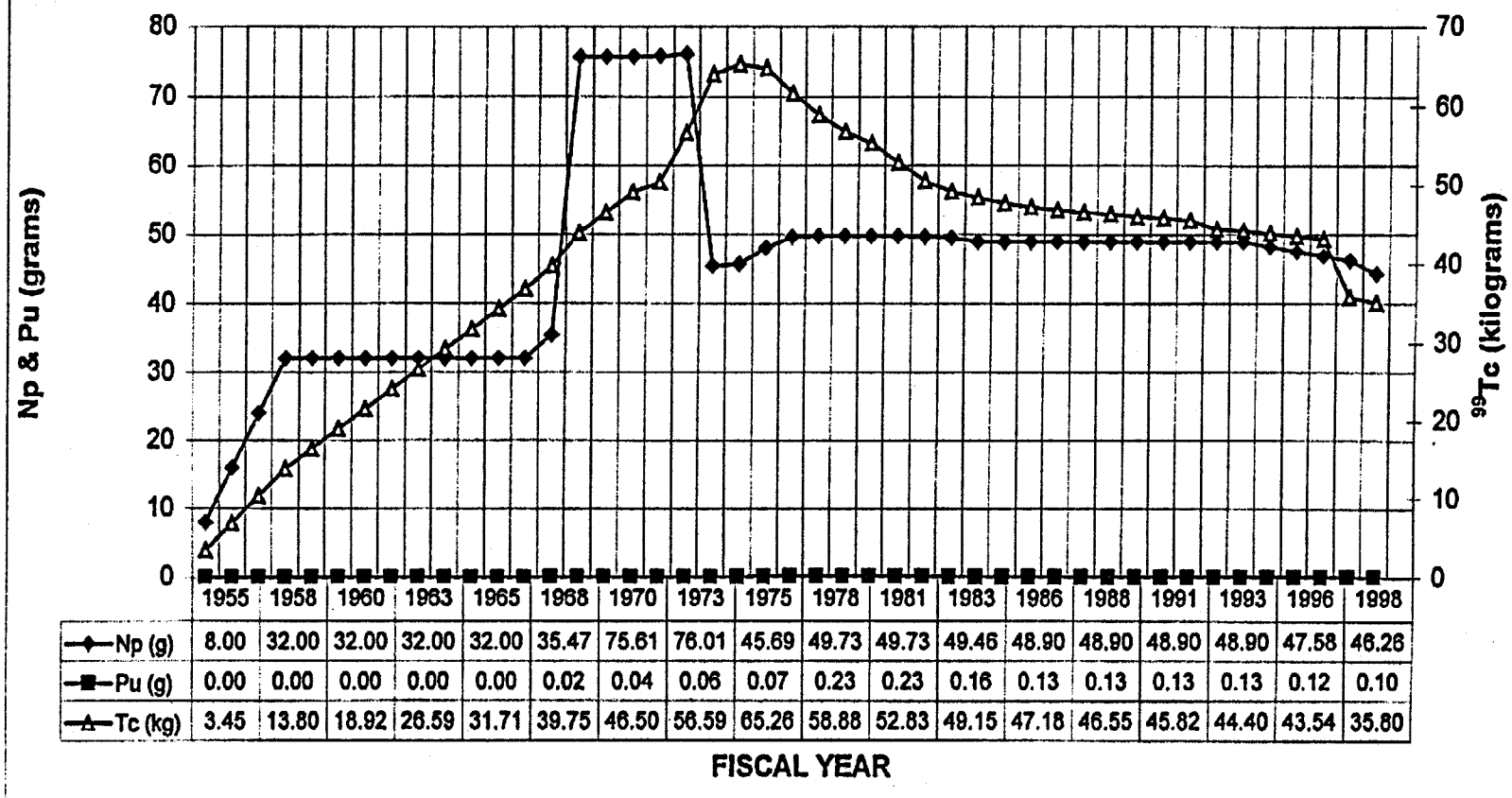
Fiscal Year	Np (g)	Pu (g)	^{99}Tc (kg)	Fiscal Year	Np (g)	Pu (g)	^{99}Tc (kg)
1955	8.00	0.0	3.45	1977	49.62	0.23	61.61
1956	16.00	0.0	6.90	1978	49.73	0.23	58.88
1957	24.00	0.0	10.35	1979	49.73	0.23	56.78
1958	32.00	0.0	13.80	1980	49.73	0.23	55.31
1959	32.00	0.0	16.36	1981	49.73	0.23	52.83
1960	32.00	0.0	18.92	1982	49.66	0.21	50.46
1961	32.00	0.0	21.48	1983	49.46	0.16	49.15
1962	32.00	0.0	24.03	1984	49.90	0.13	48.38
1963	32.00	0.0	26.59	1985	48.90	0.13	47.68
1964	32.00	0.0	29.15	1986	48.90	0.13	47.18
1965	32.00	0.0	31.71	1987	48.90	0.13	46.84
1966	32.01	0.0	34.27	1988	48.90	0.13	46.55
1967	32.02	0.0	36.83	1989	48.90	0.13	46.33
1968	35.47	0.02	39.75	1990	48.90	0.13	46.01
1969	75.61	0.04	43.94	1991	48.90	0.13	45.82
1970	75.61	0.04	46.50	1992	48.90	0.13	45.51
1971	75.61	0.04	49.06	1993	48.90	0.13	44.40
1972	75.72	0.05	50.32	1994	48.90	0.13	44.23
1973	76.01	0.06	56.59	1995	48.24	0.12	43.89
1974	45.42	0.06	63.99	1996	47.58	0.12	43.54
1975	45.69	0.07	65.26	1997	46.92	0.11	43.20
1976	48.03	0.17	64.72	1998	46.26	0.11	35.80
				Mid 1999	44.30	0.11	35.11

5.2 Annual Mass Balance of Plutonium in Recycled Uranium

As discussed in Section 3.1, RU when fed to the cascade loses its identity as RU. However, it is possible to identify the individual processes/facilities, which concentrate isotopes of Np, Pu, and ^{99}Tc and to estimate the mass flow/balance of the Np, Pu and ^{99} Tc.

This section addresses the annual mass balance of the Pu introduced into the PORTS cascade in RU.

ANNUAL INVENTORY OF RU CONSTITUENTS



	1955	1958	1960	1963	1965	1968	1970	1973	1975	1978	1981	1983	1986	1988	1991	1993	1996	1998
Np (g)	8.00	32.00	32.00	32.00	32.00	35.47	75.61	76.01	45.69	49.73	49.73	49.46	48.90	48.90	48.90	48.90	47.58	46.26
Pu (g)	0.00	0.00	0.00	0.00	0.00	0.02	0.04	0.06	0.07	0.23	0.23	0.16	0.13	0.13	0.13	0.13	0.12	0.10
Tc (kg)	3.45	13.80	18.92	26.59	31.71	39.75	46.50	56.59	65.26	58.88	52.83	49.15	47.18	46.55	45.82	44.40	43.54	35.80

Figure 5.1-1

PORTS used four major campaigns to model the flow of Pu from its arrival on plantsite through various facilities and processes to estimate the quantity of Pu present at a given location and to provide a basis for the annual mass balance (see Figures 3.1-2 through 3.1-5).

Plutonium was first introduced in FY 1955 with feed manufactured by Paducah from HRT/SRT. In the early years (FY 1955 – FY 1967) of the PORTS operation, Pu that remained in the feed cylinder heel after feeding was returned with the cylinder to either Paducah or Oak Ridge. The Pu contained in the cylinder heels is assumed not to be in inventory at year-end. Starting in FY 1968, RU was sometimes stored prior to feeding and some remains in storage as of March 31, 1999. In these instances Pu in the cylinders is included in the year-end inventory.

After cascade feeding, 90% of the Pu is assumed to remain in the cylinder heel. The Pu that does enter the cascade deposits on metallic surfaces within the immediate area of the feed point or in the feed lines. During change-out programs of FY 1958 - FY 1960 and FY 1974 - FY 1983 essentially 100% of the Pu introduced up to that point was considered removed. Due to the solubility of Pu in the decontamination solutions, the transfer of Pu from the equipment to the solution is assumed to be nearly 100%. Approximately 99% of the Pu remains with the U_3O_8 produced from uranium recovery. The balance of the Pu traveled with the raffinate to the X-701B. Trace quantities may remain in feed lines or cascade piping near the feed point.

During process equipment change-outs, the equipment typically was decontaminated soon after it was removed. For purposes of the annual mass balance the Pu is counted as either in the cascade, X-701B sludge, oxide storage, or feed cylinders. Material is always assumed to have been completely processed in the same year the processing started.

The amount of Pu in inventory annually is estimated to a peak at 0.23g in FY 1977 through FY 1981, with 0.11g Pu in inventory as of March 31, 1999. The estimate of annual Pu inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.3 Annual Mass Balance of Neptunium in Recycled Uranium

This section discusses the annual mass balance for the Np that was introduced with RU. This is not an annual mass balance of Np in RU (See rationale for Pu mass balance in Section 5.2) The model for estimating Np is similar to that of Pu with the only difference being that the percentage of Np fed to the cascade is assumed to be 33% with 67% remaining in the cylinder heel.

Like Pu, Np was first introduced in FY 1955 with feed manufactured at Paducah from HRT/SRT. Upon receipt, the material was fed and the cylinders returned to either Paducah or Oak Ridge. Beginning in FY 1968, the cylinders were sometimes held for a period of time before feeding; therefore, in these cases, the Np contained in the cylinders is included in the year-end inventory. During periods when a cylinder is fed and returned to Paducah, the Np in the cylinder is not included in the year-end inventory.

The Np that enters the cascade plates out with essentially all of the Np estimated to plate out on cascade components within a range of +6 cells to –4 cells of the feed point (Ref. 27). While Np may spread gradually over extended timeframes, the literature (Ref. 28) suggests sufficient immobility so that it can be assumed that during the cascade change-outs, FY 1958 - FY 1960 and FY 1974 - FY 1983, the equipment where the Np deposited was replaced and essentially 100% of the Np that was fed, up to this time, was removed. Trace quantities undoubtedly remained and perhaps are still present on surfaces not changed out.

The process equipment was decontaminated with essentially all of the Np going into solution through uranium recovery. The oxide produced was stored for oxide conversion at a later date. Approximately 1% of the Np that was processed through uranium recovery ended up in the raffinate at the X-701B with the remaining 99% in the oxide (U_3O_8).

The amount of Np in inventory annually is estimated to peak at 76.01g in FY 1973, with 44.3g Np in inventory as of March 31, 1999.. The estimated annual Np in inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.4 Annual Mass Balance of Technetium in Recycled Uranium

This section discusses the annual mass balance for ^{99}Tc that was introduced to the PORTS site with either RU or ^{99}Tc-contaminated PPF. This is not an annual mass balance of ^{99}Tc in RU (See rationale for Pu mass balance in section 5.2).

The ^{99}Tc mass balance is developed using the same campaigns discussed earlier for Pu and Np to model the constituent movement after arrival at PORTS.

^{99}Tc was first introduced at PORTS in FY 1955 with feed manufactured by Paducah from HRT/SRT oxide and Paducah or Oak Ridge product feed. Upon receipt, the material was fed and the cylinders returned to Paducah or Oak Ridge. Beginning in FY 1968, cylinders were sometimes held for a period of time before feeding. Any ^{99}Tc contained in the cylinders that were stored is included in the year-end inventory. During periods when a cylinder is fed and returned to Paducah, the ^{99}Tc in the cylinder is not included in the year-end inventory.

During cascade feeding, it is estimated that 90% of the ^{99}Tc enters the cascade with 10% remaining in the cylinder. The ^{99}Tc that enters the cascade initially absorbs on the metal surfaces as it moves up the cascade. While ^{99}Tc is highly mobile and moves quickly to the top of the cascade once equilibrium has been established, it was not unequivocally identified until 1974. This 19-year lag is assumed to be at least in part due to the time it took the ^{99}Tc to reach equilibrium (Ref. 19). Once at equilibrium, additional ^{99}Tc in the feed rapidly traveled from the feed point to the top of the cascade. The migration of ^{99}Tc in the cascade was slowed by the equipment change-out in FY 1958 - FY 1960 when much of the equipment contaminated with ^{99}Tc was removed and decontaminated

The process equipment was decontaminated with essentially all of the ^{99}Tc going into solution through uranium recovery. The oxide produced was stored for oxide conversion at a later date. All of the ^{99}Tc processed through uranium recovery is assumed to end up at the X-701B.

The model (Campaigns 2 & 3) includes ^{99}Tc releases to the environment as identified in Table 2.5-1.

The amount of ^{99}Tc in inventory annually is estimated to peak at 65.26 kg in FY 1975. The ^{99}Tc in inventory as of March 31, 1999 is estimated to be 35.11 kg. The estimated annual ^{99}Tc in inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.5 Potential for Worker Exposure from Recycled Uranium

Worker monitoring began in 1954 with the Film Badge and Bioassay Programs. Workers with the potential for external radiation exposure were provided film badges for monitoring. However, not all workers were provided film badges, and not all badges issued to workers were read. This changed in the mid 70's when the film badges were replaced with TLD badges. All workers, regardless of exposure potential since that time, have been provided TLD badges. Some badges are not read unless there is cause to believe a significant dose may have been recorded. Records of badge readings obtained since 1954 are retained by USEC.

The bioassay program began with urine sampling for uranium or gross alpha. Uranium sampling was used to monitor intake of workers with the potential for exposure to low assay soluble uranium. Workers with the potential for exposure to high assay uranium were monitored by gross alpha. In the mid-1990s, both methods were replaced with Inductively Coupled Plasma/Mass Spectroscopy (ICP/MS) methods. Results of urine bioassay monitoring since 1954 are retained by USEC.

Since 1988, internal doses have been reported for workers with positive bioassay results (>20 dpm/l) that resulted in calculated doses that exceeded 10 mrem. Up to several hundred employees per year were assigned internal doses based on the alpha bioassay results due to the 20 dpm/l detection limit. When the ICP/MS method replaced the uranium and alpha methods, the workers assigned internal doses dropped to a few each year since the method detection limit is about 10 nanograms (ng)/l for each isotope of uranium. The lower detection limit resulted in fewer workers being assigned doses from false positives. The uranium method (fluorimetry) used previously had a detection limit of about 5 ug/l.

5.5.1 Derived Air Concentration (DAC) and Maximum Permissible Concentrations (MPC)

Over the years of plant operation, the radiation standards have changed. The most recent standards are 10 CFR 20 (USEC) and 10CFR 835 which replaced DOE order 5480.11. Both of these used the DAC based on International Commission on Radiation Protection (ICRP) 26/30 recommendations. The DAC is defined as the concentration that if breathed by a worker for a work-year, would result in a limiting dose. The limiting dose is the more limiting of either 5 rem committed effective dose equivalent or 50 rem committed organ dose equivalent. DACs are listed by each radioisotope and by solubility. The solubility classes, from most to least soluble, are D (for days), W (for weeks), and Y (for years).

AEC/ERDA/DOE 0524 provided Radiation Protection Guidelines (RPGs) based on ICRP 2/10. The MPC if breathed by a worker for a work-year would result in 15 rem annual organ dose. Reporting of internal dose was required when cumulative intakes exceeded 50% of the RPG for the critical organ.

AEC/ERDA/DOE 0524 listed the MPCs for soluble uranium as $6x10^{-11}$ uCi/ml and $1x10^{-10}$ uCi/ml for insoluble uranium. The MPC's for TRU ranged from $1x10^{-10}$ uCi/ml for insoluble ^{237}Np, to $2x10^{-12}$ for soluble ^{238}Pu, ^{239}Pu, and ^{240}Pu. The current DACs from 10CFR 835 are listed in Table 5.5.1-1.

Table 5.5.1-1

Current Derived Air Concentration from 10CFR835

	Half Life (Years)	Specific Activity uCi/g	DACs		
			Class D uCi/ml	Class W uCi/ml	Class Y uCi/ml
^{99}Tc	213,000	17,000	$2x10^{-6}$	$3x10^{-7}$	
^{228}Th	1.91	$8.2x10^{8}$		$4x10^{-12}$	$7x10^{-12}$
^{230}Th	77,000	20,200		$3x10^{-12}$	$7x10^{-12}$
^{234}U	244,500	6,253	$5x10^{-10}$	$3x10^{-10}$	$2x10^{-11}$
^{235}U	7,038,000	2.163	$6x10^{-10}$	$3x10^{-10}$	$2x10^{-11}$
^{236}U	23,415,000	64.74	$6x10^{-10}$	$3x10^{-10}$	$2x10^{-11}$
^{238}U	$4.468x10^{9}$	0.3364		$3x10^{-10}$	$2x10^{-11}$
^{237}Np	2,140,000	705.3		$2x10^{-12}$	
^{238}Pu	87.74	$1.7x10^{7}$		$3x10^{-12}$	$7x10^{-12}$
^{239}Pu	24,065	62,200		$2x10^{-12}$	$6x10^{-12}$
^{241}Am	432.2	$3.4x10^{6}$		$2x10^{-12}$	

The most dramatic change between the old standards and the new standards are for the insoluble TRU DACs. For instance, the MPC for insoluble ^{237}Np was 50 times higher than the current DAC. Since the insoluble TRU MPCs were similar or higher than the insoluble uranium oxide MPC, exposure to insoluble TRU under the old limits would be considered to be adequately controlled if the exposure to insoluble uranium was controlled. The plant allowable limits (PAL) were about half the DOE/Energy Research and Development Administration limits.

To significantly decrease the effective DAC 10% for a mixture of Class D uranium and TRU compared to the uranium alone would require that only 0.04% of the total activity be TRU. The presence of the TRU increases the dose compared to the dose that would have been received by inhalation of the uranium alone. For Class W, there would only have to be 0.07% TRU present for the same 10% decrease in the DAC. For class Y uranium, there would have to be 1.2% present to result in 10% increase in dose.

For ^{99}Tc, the DAC is much higher than the DAC for other beta emitters present at the site. The ^{238}U decay products ^{234}Th and ^{234}Pa are present, especially at the feed facilities. The DAC for ^{234}Th Class Y (the most restrictive) is $6x10^{-8}$ uCi/ml compared to the Class W ^{99}Tc DAC of 3 $x10^{-7}$ uCi/ml. Since the uranium alpha DACs are at least 500 times lower, unless ^{99}Tc is present at 500 times the uranium activity, the uranium is the more limiting in terms of the hazard to workers. ^{230}Th is also present at PORTS in significant quantities. ^{230}Th arises from the decay of ^{234}U. About 9 uCi of ^{230}Th per year are produced per curie of ^{234}U. The presence of ^{230}Th is not related to RU since it would be present regardless. It is important to note that ^{230}Th is generally present in larger quantities compared to TRU. The ratio is generally 3 parts ^{230}Th to 1 part TRU. The DAC for ^{230}Th is comparable to TRU. Since 1993, internal dose assessments include a contribution from ^{230}Th and ^{234}U. The significance of TRU to the potential dose to workers in this report will not include the contribution from ^{230}Th.

5.5.2 Bioassay Monitoring Results

From 1965, until the early 1990s, a portable In-Vivo counter was employed to monitor lungs of workers for insoluble uranium. The counter was usually brought to PORTS twice per year. The capability to detect ^{99}Tc, and ^{237}Np were added in 1977. The detection limits for this counter were stated as: 100 ug ^{235}U (240 ug RPG limit); 10 mg total uranium (27 mg RPG limit); 1 uCi ^{99}Tc (9 uCi RPG limit); and 0.2 uCi ^{237}Np (17 nCi RPG limit). The most reliable results were for ^{235}U since the gamma emissions were monitored directly. ^{238}U is determined from the ^{234}Th decay product of ^{238}U, and ^{237}Np is based on the ^{233}Pa emissions. Naturally occurring radon daughters provide a false positive for ^{237}Np. ^{99}Tc is not a gamma emitter, thus a background adjustment is performed over a low energy range to obtain the ^{99}Tc count.

No records have been found that indicate that doses from uranium, ^{99}Tc or TRU have been assigned based solely on In-Vivo results. Usually, another In-Vivo count and a urine bioassay are requested if a result that exceeds the detection limit was obtained. To determine dose, the amount inhaled and how fast the material is removed from each organ of interest must be known. The RPG's are based on having the RPG present in the organ for a year. The results of In-Vivo monitoring were summarized in a 1986 report (Ref. 29.) The following table lists the number of counts taken and the number that exceeded the ^{235}U detection limit from 1965 to 1985: (Table 5.5.2-1)

Table 5.5.2-1

In-Vivo Summary (1965 – 1985)

Year	Total Counts Taken	No. of Counts >100ug ^{235}U	Year	Total Counts Taken	No. of Counts >100ug ^{235}U
1965	27	18	1976	411	58
1966	30	14	1977	971	96
1967	236	28	1978	542	29
1968	364	39	1979	497	15
1969	393	73	1980	924	4
1970	147	32	1981	868	2
1971	179	56	1982	910	1
1972	157	36	1983	632	3
1973	392	26	1984	613	0
1974	521	65	1985	798	4
1975	684	92			

Another summary produced at the same time (Ref. 30) listed the urine sampling results for uranium and alpha for the same period. The total collected each year and the number that exceeded the detectable level for alpha results (approximately 50 dpm/l) are summarized below: (Table 5.5.2-2)

Table 5.5.2-2

Urine Sampling Summary 1965 - 1985

Year	Total Samples	Positive Samples	% Positive	Year	Total Samples	Positive Samples	% Positive
1965	2,155	1,234	57	1976	5,066	3,689	73
1966	2,667	1,037	39	1977	5,601	4,414	79
1967	2,493	805	32	1978	5,933	2,412	41
1968	2,985	1,035	35	1979	3,694	1,963	53
1969	3,505	1,758	50	1980	7,794	5,525	71
1970	3,442	1,579	46	1981	8,203	5,449	66
1971	3,953	2,121	54	1982	6,571	3,487	53
1972	3,288	1,513	47	1983	4,412	2,455	56
1973	3,656	1,817	50	1984	3,302	1,681	51
1974	3,056	1,489	49	1985	5,108	2,472	48
1975	5,750	3,954	69				

The bioassay results indicate that a large proportion of monitored workers were exposed and had intakes of uranium. The dose to each worker cannot be determined without detailed analysis. Because the monthly samples were not consistently submitted by workers, the actual number of workers monitored each year or the number of workers with positive results can not be reliably extracted from the results above.

5.5.3 Facilities with the Potential for Worker Exposure to RU Constituents

5.5.3.1 X-705 Oxide Conversion Facility

A limited amount of information is available that describes the recycled constituents of the oxide processed in this facility. An unpublished draft report, circa 1977, (Ref. 31) which covered the conversion of TRU contaminated oxides from 1967 to 1975 included calculations of airborne TRU concentration if TRU contaminated oxide were to be processed. This report did indicate that the airborne uranium concentration exceeded the Plant Allowable Level (PAL) in 1368 occurrences in the Tower Room, in 826 occurrences in Oxide Unloading, and an additional 577 occurrences in the Cold Trap Room. The PAL, according to the May 1979 GAT-226 "Guide to Safety", was 3 dpm/ft^3 (4.8×10^{-11} uCi/ml). The actual data cited in this report could not be located. The use of respiratory protection was required by procedures in place at the time, and their use was encouraged by supervision.

Smear samples obtained in "E", "F" & "H" areas and analyzed for thorium, uranium and TRU during TRU characterization in the 1990's are shown in Table 5.5.3.1-1.

How much TRU was present in each year of operation is not known however, these sample results do verify that TRU contamination was present when the facility was shutdown in 1978. The samples in H-Area indicate that the TRU percentage is 0.12%. In soluble uranium (Class D or W), found in H-Area and the cold trap room, the levels of RU constituents has a significant effect on the DAC. For TRU in insoluble (Class Y) oxides, (E and F-Areas) the effect is lessened due to the decrease in the insoluble uranium DAC. Since the TRU percentage is less than 1.2%, the effect of TRU in the oxide at these levels is insignificant.

Table 5.5.3.1-1

TRU Characterization of Smear Samples from "E", "F", and "H" Areas

Location	Sample # HPX-	Am/Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
F-Area	93-934	26	32	7,630	0.76	8.18	12.4
H-Area	94-070, 077, 100, 101	3.2	7.5	8,699	0.12	18.86	30.3
E-Area	93-924, 925, 931, 96-039	231	61.0	81,623	0.36	11.66	20.0

Air sampling in the Oxide Conversion Facility measured only the total alpha concentration from uranium, thorium and TRU. Appendix XIV, taken from site internal correspondence (Ref. 32) summarizes the air sample concentration at the continuous air samplers located in the Oxide Conversion Facility from sampler start up through 1978. These levels warrant the use of respirators and, when worn properly, provide adequate protection to the worker.

The calculations for obtaining the DAC are shown in Attachment XIV. The calculated Class D DACs can be compared to the current Class D DAC of $1x10^{-10}$ uCi/ml which assumes that up 2% ^{230}Th is present. The samples from H-Area were used to calculate the DAC for the cold trap room in E-Area, since the TRU fraction is lower and the UF_6 handled is soluble (Class D). This is in line with removal of the TRU in the tower ash and the MgF_2 trap. The airborne radioactivity in the other E-Area samples may contain oxides or ash from the tower which are considered insoluble (Class Y).

5.5.3.2 X-705 Decontamination Area

The X-705 Decontamination Area contains a multitude of activities with the potential for worker exposure to the constituents of RU. There are two principal routes of entry for these constituents - PG cylinders to be cleaned and process equipment to be disassembled and decontaminated. The cylinder cleaning area is in the northwest corner of the X-705. Both large cylinders (2-1/2 ton and up) and small cylinders (5" to 13") are cleaned in separate areas. The cylinders are currently cleaned with a boric acid solution, rinsed and dried. The solutions from the cylinder cleaning are added to the solution recovery system.

Process equipment may be disassembled in one of several areas depending on size and the fixtures necessary to handle it. Gross internal contamination is removed after disassembly. The components are then either placed on carts and passed through the Large Parts Decontamination Tunnel or decontaminated by hand in the Small Parts Area. Solutions from these areas are also added to the solution recovery system.

Air contamination surveys taken in 1993-1994 and 1995-1996 in the X-705 Decontamination Area indicate that significant TRU was present. A summary of these results is shown in Appendix XV.

In the airborne samples taken in 1993-1994, both Pu and ^{237}Np are significant. In the samples taken in 1995 - 1996 only ^{237}Np is significant for the RU constituents. Since the analytical techniques were being refined during 1993 and 1994, it is possible that the difference is due to changes in the laboratory method, or to actual changes in the constituents present.

5.5.3.2-1 Cylinder Cleaning

The cylinder cleaning operations in the X-705 Decontamination Area potentially involve concentrated RU constituents. The results shown in Table 5.5.3.2-1 are from samples taken during decontamination of large cylinders area in 1993. The results in Table 5.5.3.2-2, which were obtained in 1999, were taken from small cylinder wash

solutions in the X-705 West Annex. In both cylinder-cleaning areas, the TRU percentage is significantly greater than 0.04%.

Table 5.5.3.2-1

Large Cylinder Area – Sample Results 1993

Sample # HPX-	Am/Pu pCi	^{237}Np pCi	Total pCi	% TRU	% Th + TRU	uCi/gU	% ^{235}U
93-291, 292, 293, 294, 802, 822, 825, 897, 916	130	60	33,305	0.57	7.94	3.44	4.3

Table 5.5.3.2-2

X-705 West Annex – Sample Results from Small Cylinder Wash

Sample #*	Note	Am/Pu pCi	Np pCi	^{238}Pu pCi	^{239}Pu pCi	Total pCi	% TRU	uCi/gU	% ^{235}U
340-110, 112, 114, 116, 118, 120	Wash	471	11,514	1,287	2,838	1,692,643	0.92	19.82	42.7
340-111, 113, 115, 117, 119	Rinse	0.11	4.8	0.39	1.7	1,305	0.53	17.09	37.1

*Laboratory Information Management System Identification Number (LIMS ID#)

5.5.3.3 X-744G Bulk Storage Building

Batching operations of trap materials were carried out in the X-744G in the mid 1990s. Respirators were worn by workers during these operations. In 1995, air samples were taken and analyzed to characterize the TRU constituents.. The results of all 28 samples taken in 1995 were combined to produce the data in Table 5.5.3.3-1. In this instance the TRU is significant since the TRU percentage is 0.15% of the total activity.

Table 5.5.3.3-1

X-744G Bulk Storage – Air Sample Results for Trap Batching (1995)

^{238}Pu pCi	^{239}Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
1.6	3.9	6.0	7,771	0.15	10.03	16.2

5.5.3.4 X-343 Feed Vaporization and Sampling Facility

This facility feeds UF_6 into the cascade. Airborne radioactivity measurements analyzed for TRU are summarized in Table 5.5.3.4-1. The results were obtained from 38 samples taken between 1994 and 1997. The TRU levels are insignificant at 0.01% of the total activity.

Table 5.5.3.4-1

X-343 Feed Vaporization and Sampling Facility – Airborne Radioactivity Summary (1994 – 1997)

^{239}Pu pCi	^{238}Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
0.0	0	1.2	19,986	0.01	2.10	4.0

5.5.3.5 X-344 Toll Enrichment Facility

The sample results shown in Table 5.5.3.5-1 were obtained from eleven air samples taken in 1994 - 1997 in the X-344. The TRU levels are insignificant at less than 0.01% of the total activity.

Table 5.5.3.5-1

X-344 Toll Enrichment Facility – Airborne Radioactivity Summary (1994 – 1997)

^{239}Pu pCi	^{238}Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
0.0	0.21	0.7	25,143	0.00	0.95	1.3

5.6 Environmental

Surveys to determine the extent of contamination in the USEC leased facilities covered more than 540 outside acres. Most outside areas covered with grass, gravel or pavement were surveyed with an array of radiation detectors towed behind a tractor at slow speeds. Other areas were surveyed with hand-held survey instruments.

USEC Health Physics Policy X38300-00-001 lists all areas of contamination within the USEC leased areas. One area is known to contain ^{99}Tc and TRU (X-701B), but most are posted only for protection of personnel from removable contamination. There are about 527,000 sq. ft. of contaminated areas, of which almost half (244,000 sq. ft.) are on the roofs of the X-705 and X-710. The X-701B is about 260,000 sq. ft. Contamination control zones are 4,116,800 sq. ft. of which 4,109,000 sq. ft. are associated with the three process buildings. There are 1,532,900 sq. ft. of fixed contamination areas with 530,000 sq. ft. in the X-530 switchyard. Soil contamination areas amount to 199,100 sq. ft. of which 130,000 sq. ft. are near the X-7721, 12,000 sq. ft. near the X-745F, and 14,000 sq. ft. near the X-705. Underground radioactive material areas amount to 15,800 sq. ft. and mostly associated with the X-705 (15,500 sq. ft.).

6. RESULTS AND CONCLUSIONS

6.1 Explanation of Flow Paths

The RU entered PORTS from two sources. The first source was the stream derived from spent reactor fuel that had been converted to UF_6 at the PGDP and ORGDP feed manufacturing facilities. A total of 1,094.6 MTU was fed to the cascade (~0.64 assay). The second source was materials received in various chemical forms through the scrap returns program (see Table 5.1-2).

The constituent ^{99}Tc reached PORTS through the RU and PPF. Since startup, some 58 kg of ^{99}Tc are estimated to have entered the PORTS cascade from PPF and 6 kg from RU for a total of 64 kg fed to the cascade. Some previous estimates have placed this level as high as 90 kg.

6.2 Identification and Evaluation of Processes or Facilities That Involved Worker Exposure to Recycled Uranium Constituents

There are no known documented cases of worker exposure to TRU constituents of RU at PORTS from any process. There have been cases of worker contamination due to ^{99}Tc. Exposures to ^{99}Tc through ingestion are TBD. There was the potential for worker exposure in the following facilities/work areas:

1. Oxide Conversion Facility;
2. cascade during removal of cascade equipment;
3. equipment decontamination in X-705 equipment; and
4. cylinders cleaning.

In-vivo results from workers are available since 1965. The HP staff was concerned about exposures to insoluble Np compounds during CIP/CUP. Urine bioassay results are available from 1955 and indicate that uranium and ^{99}Tc exposures have occurred. It is not yet known whether these exposures indicated by urine bioassay also include a contribution from TRU.

Dose assessments for workers with positive bioassay have been required and performed since 1988. No worker has been assigned an internal dose from ^{99}Tc or TRU since that time. Air and smear samples taken since 1993 to characterize the radioactive constituents in the facilities listed above also indicate that TRU is present, but may not be significant.

6.3 Identification and Evaluation of Processes or Facilities that Involved Potential Environmental Contamination

The only record of environmental releases of TRU constituents occurred between October 1976 and March 1977 when one sample above the minimum detection limit (MDL) was detected in the outfall from the X-701B. There are extensive records and cases of environmental contamination due to ^{99}Tc. The section above lists the facilities / work areas into which RU was introduced. Technetium has been found in site disposal areas where contaminated equipment was stored, oil and cleaning solvents were disposed of, and in air (stack and perimeter) and water effluent monitors. Perimeter air sampling generally indicates only naturally occurring materials, but ^{99}Tc was found after the 1998 fire in X-326. Groundwater sampling has indicated that technetium is present in plumes originating from the X-701B Holding Pond. Technetium has higher mobility than uranium and the other RU constituents.

The immediate area around the X-705 has contamination on pavement and in soil. Because the source was the X-705, ^{99}Tc and TRU are expected to be present. The X-705A Incinerator Facility was operated until 1986 to dispose of burnable waste. Contamination measurements made of the interior surfaces during demolition indicated that ^{99}Tc and TRU were present. Contamination surveys in the vicinity of these facilities have located small areas of surface soil contamination. These areas have been posted to protect workers from exposure, but have not been extensively characterized to determine if they contain ^{99}Tc or other constituents of RU.

6.4 Discussion of Data Sources and Confidence Levels

Information utilized in this report was gathered from various sources. Factors influencing the quality of the information vary from the level of documentation in which the information was found to the credibility of the individual supplying the data. Certain types of data clearly have, as their bases, physical and chemical measurements supported by reliable documentation including chain of custody records, weight tickets, and lab instrument printouts. At the other extreme, anecdotal testimony of "how things were done" may be highly dependent on fading memories or hearsay information. In some cases, conclusions could be arrived at only through deductive reasoning and, in a few cases, speculation. Table 6.4-1 lists the sources and/or types of sources utilized, with the team's assessment of the reliability of the information based on their collective experience.

Where deductive reasoning and/or educated speculation were critical in coming to closure with an issue in this report, such steps in logic are cited. In these cases, the 185 years of collective experience of the site team has been relied upon.

6.5 Conclusions

In reviewing the operating history of PORTS (including facilities and specific time periods), where there are significant implications for potential worker exposure or environmental contamination certain conclusions can be made with reasonable confidence. These are as follows:

1. The largest quantity of recycled uranium received and fed to the PORTS cascade was manufactured at PGDP and ORGDP from recycled UO_3 from Hanford and Savannah River. Of the approximately 320,817 MTU fed to the PORTS cascade through FY 1997, 1,095 MTU was RU processed at the PGDP or ORGDP feed plants.

2. The largest contribution of ^{99}Tc, which amounted to about 60 to 90 kg, was in the 121,485 MTU of feed produced by PGDP.

3. Facilities and associated processes where TRU constituents had the most potential for worker exposure were in the cascade near the RU feed points, particularly during equipment removal, and the Oxide Conversion Facility when changing the ash filters.

4. Possible exposure to ^{99}Tc could have occurred during maintenance of the top purge cells or change-out of trapping media near these locations. Potential exposure to ^{99}Tc from the handling of treatment sludges from the uranium recovery facility is considered unlikely due to the dilution of this stream with enormous quantities of non-radiological materials.

5. The X-344 Feed Manufacturing Facility was free of TRU/FP constituents during its operating history.

The site team lists opportunities for improvement to the PORTS RU mass balance effort in Table 6.5-1. The recommended actions are listed in order of priority for clarifying worker exposure or environmental contamination.

Table 6.4-1

Data Sources and Assessment of Data Quality – PORTS Site

Item	Data Source	Quality of Information			Comments
		High	Med.	Low	
1	SS Accountability Records	X			
2	Plant Monthly Inventory Reports	X			
3	NMMSS Data Base Queries		X		Dependent on setting of proper filters
4	NMC&A-DEC10 Journals (VHE-refeed)	X			
5	Supplemental Analysis Reports (Scrap) GAT-XXX-XX-XX		X		
6	Nuclear Materials Transaction Reports Typically DOE/NRC-741	X			
7	HEU Refeed Program Cylinder Cleaning and Shipment Schedules	X			Historical
8	Operational Log Books	X			
9	Oxide Conversion Feed Sheets		X		Were followed on a best effort basis only
10	Plant and Department Activity Reports		X		Evolving format and content
11	Plant Interdepartmental Correspondence		X		Citations and references at times are lacking
12	GAT.POEF OR and Paducah Formal Plant & Technical Reports	X			
13	Interviews with Plant Current and Former Employees			X	Unsubstantiated and opinionated
14	Consonant Interviews with Multiple Current and Former Employees		X		Subject to "group think"
15	Dissonant interviews with Multiple Current and Former Employees			X	
16	Personnel Notes, Memos to File		X		
17	Correspondence Between AEC/ERDA/DOE Site Operations Office and Operating Contractors	X			
18	Drawings/Photographs of Systems and Facilities	X			
19	Operating and Maintenance Procedures		X		Implemented on a graded approach

Table 6.5-1

Opportunities for Improvement - Prioritized

Priority	Improvement	Remarks
1	Evaluate potential for worker exposure and/or environmental contamination from 1998 X-326 side purge fire	Clean-up/repair work currently progressing; high potential for ^{99}Tc and Np contamination; documentation may be readily available.
2	Evaluate X-744G facilities/activities for potential worker exposure	Large quantities of oxide, solutions, nitrate, and trapping materials containing possible TRU/FP were sampled or batched over the operating history.
3	Evaluate X-710 facilities/activities for potential worker exposure	Handled small quantities of TRU/FP under experimental conditions
4	Quantify flow paths for recycled waste streams (i.e., ion exchange resins, heavy metals sludge, etc.) from uranium recovery facility	Streams are probably low in TRU content, but high in ^{99}Tc
5	Evaluate flow path of 17 MTU enriched (15.5 MTU with assay of 0.7103 and 1.5 MTU with assay of 0.85303) UF_6 received from ORGDP in FY 1961	Categorized as non-TRU, but lack certainty
6	Quantify flow paths for various trapping materials (i.e., alumina, NaF, and MgF_2)	High potential for worker exposure, but effort may be time consuming
7	Evaluate X-760 facilities/processes/activities (i.e., Fluorox process, etc.) for potential worker exposure (particularly in late 1950's and early 1960's)	Quantities of depleted RU (i.e., about 0.4 MTU UO_2, 3.3 MTU UO_3, and 0.9 MTU UF_4) received from ORGDP in FY 1957 may have been used for early plant reactor studies in X-760, but records could not be retrieved in time for this report
8	Quantify flow path for ^{99}Tc at X-231A and B Oil Biodegradation Plots	Environmental data may be available in RCRA closure documentation
9	Determine final disposition for unaccounted for 0.04 MTU of UNH received from Division of International Affairs in FY 1966 – 1967	Low quantity of RU present and locating records may be laborious
10	Quantify the TRU/FP constituents in LEU oxides in storage	Low potential for worker exposure in present storage configuration. High cost associated with sampling and analysis. Results would be of benefit to final disposition

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Reference 13 – GAT-T-2448, Discard Criteria Report, Griggs, December 8, 1975.

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APPENDIX I

X-344 FEED MANUFACTURING PLANT

DATE	BEGINNING INVENTORY (kgU)			RECEIPTS (kgU)			AMOUNT FED (kgU)			ENDING INVENTORY (kgU)					TOTAL UF_6 PRODUCTION (kgU)	UF6 DISPOSITION (kgU)			ENDING INVENTORY (kgU)	ASH PRODUCTION (kgU)
	MCW	NLO	TOTAL	MCW	NLO	TOTAL	MCW	NLO	TOTAL	MCW	NLO	TOTAL	N	*		PORTS CASCADE	PGDP CASCADE	ORGDP CASCADE		
Mar-58				17,238																
Apr-58	17,238		17,238																	
May-58	17,238		17,238		101,414	101,414	17,238	60,581	77,819	17,238		17,238								
Jun-58		40,545	40,545	55,302	304,371	359,673	11,059	333,537	344,596		40,545	40,545			61,903	17,357	13,085			11,283
Jul-58	44,243	10,129	54,372	42,630	237,266	279,896	86,863	205,825	292,688	44,243	10,129	54,372	N		314,822	133,675	170,741		31,562	
Aug-58		40,615	40,615	97,928	237,923	335,851	94,329	226,420	320,749		40,615	40,615	N		285,547	159,853	110,347		42,739	
Sep-58	3,267	51,112	54,379	212,902	271,572	484,474	162,549	253,815	416,364	3,267	51,112	54,379	N		307,049	168,831	139,451		58,042	
Oct-58	53,106	64,266	117,372	246,096	237,492	483,588	250,219	253,543	503,762	53,106	64,266	117,372	N		407,107	171,908	258,011		57,206	
Nov-58	47,996	47,485	95,481	235,988	206,931	442,919	249,191	219,399	468,590	47,996	47,485	95,481	N		503,474	174,384	325,045		35,255	
Dec-58	34,063	33,892	67,955	228,636	111,854	340,490	231,795	132,391	364,186	34,063	33,892	67,955	N		464,701	153,891	322,986		40,613	106,583
Jan-59	30,158	13,111	43,269	190,097	94,585	284,682	179,530	83,922	263,452	30,158	13,111	43,269	N		364,609	155,812	178,377		29,354	72,848
Feb-59	40,327	23,587	63,914	164,348	107,568	271,916	184,233	130,885	315,118	40,327	23,587	63,914	N		260,566	161,389	98,323		60,805	
Mar-59	20,072		20,072	259,242	30,276	289,518	231,581	30,185	261,766	20,072		20,072	N	*	310,202	147,714	160,278		62,149	30,311
Apr-59	47,143		47,143	300,202	29,341	329,543	329,239	29,268	358,507	47,143		47,143	N	*	263,376	163,004	111,414		64,651	69,881
May-59	17,317		17,317	303,290	82,438	385,728	250,355	82,001	332,356	17,317		17,317	N	*	356,366	158,340	191,157		54,467	81,486
Jun-59	70,011		70,011	374,757	9,711	384,468	370,990	9,688	380,678	70,011		70,011	N	*	331,035	170,632	180,550		62,570	83,415
Jul-59	73,126		73,126	224,490	29,816	254,306	222,190	29,771	251,961	73,126		73,126	N	*	378,037	173,788	198,792		43,498	122,989
Aug-59	74,837		74,837	242,403	19,667	262,070	269,010	19,531	288,541	74,837		74,837	N	*	260,776	150,324	109,697		50,109	89,519
Sep-59	47,570		47,570	232,370	9,787	242,157	234,620	9,744	244,364	47,570		47,570	N	*	289,904	204,395	102,545		51,083	74,419
Oct-59	44,657		44,657	119,007	87,916	206,923	155,324	84,376	239,700	44,657		44,657	N	*	243,834	178,430	48,759		34,341	82,249
Nov-59	7,973	3,371	11,344	178,850	68,654	247,504	176,121	61,931	238,052	7,973	3,371	11,344	N	*	253,063	175,757	79,504		51,404	39,881
Dec-59	7,298	13,057	20,355	194,438		194,438	197,837	13,048	210,885	7,298	13,057	20,355	N	*	221,865	178,924	65,613		49,490	115,812
Jan-60	3,438		3,438	178,002	29,407	207,409	174,771	29,331	204,102	3,438		3,438	N	*	222,253	188,424	11,999		27,062	3,174
Feb-60	6,273		6,273	236,979		236,979	242,654		242,654	6,273		6,273	N		202,526	188,146			49,085	27,976
Mar-60				269,184	29,307	298,491	221,812	3,288	225,100	46,885	26,018	72,903	N		245,389	244,749			63,473	20,035
															221,462	277,687			64,102	12,927
																			13,368	39,958

Notes:

N – Normal material

* - Totals include MCW material in 30 gallon drums (includes substandard material)

APPENDIX I (CONT'D)

X-344 FEED MANUFACTURING PLANT

DATE	BEGINNING INVENTORY (kgU)			RECEIPTS (kgU)			AMOUNT FED (kgU)			ENDING INVENTORY (kgU)			N	*	TOTAL UF_6 PRODUCTION (kgU)	UF6 DISPOSITION (kgU)			ENDING INVENTORY (kgU)	CASH PRODUCTION (kgU)
	MCW	NLO	TOTAL	MCW	NLO	TOTAL	MCW	NLO	TOTAL	MCW	NLO	TOTAL				PORTS CASCADE	PGDP CASCADE	ORGDP CASCADE		
Apr-60	46,885	26,018	72,903	248,222	19,928	268,150	256,591	45,853	302,444	37,818		37,818	N		303,296	237,665			79,194	8,799
May-60	37,818		37,818	203,299		203,299	224,415		224,415	15,916		15,916	N		220,316	255,127			44,171	18,914
Jun-60	15,916		15,916	232,761	29,112	261,873	217,066	29,035	246,101	30,721		30,721	N		235,395	233,741			45,820	12,227
Jul-60	30,721		30,721	232,885		232,885	181,246		181,246	81,606		81,606	N		185,457	182,359			48,977	17,662
Aug-60	81,606		81,606	249,906	69,056	318,962	300,183	45,768	345,951	33,412	19,799	53,211	N	*	347,325	195,880			200,923	36,971
Sep-60	33,412	19,799	53,211	199,915	29,568	229,483	162,777	49,192	211,969	69,768		69,768	N	*	217,692	214,756			203,780	45,300
Oct-60	69,768		69,768	139,007	9,820	148,827	175,818	6,440	182,258	32,208	3,367	35,575	N	*	181,505	227,775			157,306	35,386
Nov-60	32,208	3,367	35,575	178,385	128,126	306,511	184,287	124,609	308,896	25,597	6,422	32,019	N	*	307,491	245,057			219,924	33,062
Dec-60	25,597	6,422	32,019	187,175	98,453	285,628	166,414	101,559	267,973	45,888	3,066	48,954	N	*	267,988	257,375			230,610	13,128
Jan-61	45,888	3,066	48,954	232,655	39,208	271,863	257,258	38,824	296,082	20,515	3,360	23,875	N	*	289,664	268,270		46,509	205,482	17,428
Feb-61	20,515	3,360	23,875	153,231	137,475	290,706	149,299	137,131	286,430	23,981	3,374	27,355	N	*	286,574	236,769	38,781	70,843	146,562	56,638
Mar-61	23,981	3,374	27,355	88,175	265,010	353,185	94,427	251,449	345,876	17,412	16,271	33,683	N	*	341,144	248,956		144,034	99,992	21,095
Apr-61	17,412	16,271	33,683	36,995	255,464	292,459	36,827	231,451	268,278	17,412	39,584	56,996	N	*	272,500	242,035		69,707	60,917	61,653
May-61	17,412	39,584	56,996	68,448	236,133	304,581	67,973	245,810	313,783	17,689	29,194	46,883	N	*	308,999	261,401			108,403	33,541
Jun-61	17,689	29,194	46,883	68,105	265,619	333,724	75,416	244,891	320,307	10,207	49,217	59,424	N	*	321,639	252,990			177,578	31,148
Jul-61	10,207	49,217	59,424	55,653	88,417	144,070	35,781	107,529	143,310	30,070	29,818	59,888	N	*	139,692	252,384			63,667	32,422
Aug-61	30,070	29,818	59,888	22,290	118,508	140,798	38,331	118,428	156,759	13,603	29,509	43,112	N	*	158,167	137,370			84,932	19,574
Sep-61	13,603	29,509	43,112	39,872	106,588	146,460	36,472	116,689	153,161	16,932	19,122	36,054	N	*	150,416	130,853			104,441	17,349
Oct-61	16,932	19,122	36,054	79,517		79,517	70,440	19,019	89,459	25,805		25,805	N	*	89,131	72,253			121,617	20,857
Nov-61	25,805		25,805	79,673	27,712	107,385	81,593		81,593	23,675	27,712	51,387	N	*	82,683	119,768			84,286	36,285
Dec-61	23,675	27,712	51,387		187,487	187,487	20,378	58,443	78,821	3,240	33,947	37,187	N	*	77,911	93,119			68,943	108,294
Jan-62	3,240	33,947	37,187		193,836	193,836	3,240	190,217	193,457		36,959	36,959	N		194,730	155,959			108,027	
Feb-62		36,959	36,959		101,761	101,761		138,109	138,109				N		140,704	184,836			64,147	
TOTALS				7,400,548	4,744,577	12,127,887	7,379,742	4,602,926	11,982,668						11,890,285	8,604,112	2,915,455	331,093		1,762,479

Notes:
N – Normal material
* - Totals include MCW material in 30 gallon drums (includes substandard material)

APPENDIX II

PORTS SUMMARY HISTORICAL DATA RECYCLED URANIUM WORKSHEET (RECEIPTS)

Shipper	Uran. Form	Startup thru FY 1970			FY 1971 thru March 1999			Total Quantity		
		Total Quantity of Uranium (g)	Total Quantity of Recycled Uranium (g)	Percent Recycled	Total Quantity of Uranium (g)	Total Quantity of Recycled Uranium (g)	Percent Recycled	Total Quantity of Uranium (g)	Total Quantity of Recycled Uranium (g)	Percent Recycled
Allied Chemical Corp	UO_3	–	0		4,073,685	4,073,685	100	4,073,685	4,073,685	100
Babcock & Wilcox	UF_6	–	0		8,926,541	153,112	2	8,926,541	153,112	2
Division of International Affairs	UF_6	237,204	150,593	63	–	0		237,204	150,593	63
	UNH	65,202	45,855	70	–	0		65,202	45,855	70
Fernald	U_3O_8	21,069	0		44,831,588	7,798,373	17	44,852,657	7,798,373	17
France	UF_6	101,347,684	0		6,849,106,756	1,585,738	<1	6,950,454,440	1,585,738	<1
Germany	UNH	–	0		6,853,588	6,853,588	100	6,853,588	6,853,588	100
K-25	UF_4	864,952	864,952	100	–	0		864,952	864,952	100
	UF_6	714,852,220	296,504,829	41	1,929,840,198	0		2,644,692,418	296,504,829	11
	UO_2	802,892	418,481	52	–	–		802,892	418,481	52
	UO_3	3,319,042	3,319,042	100	–	–		3,319,042	3,319,042	100
NUMEC	UF_6	2,772,420	0		8,679,175	330,429	4	11,451,595	330,429	3
Paducah	UF_6	68,830,800,434	798,609,341	1	45,705,582,073	0		114,536,382,507	798,609,341	1
United Kingdom	UNH	–	0		6,957	6,916	99	6,957	6,916	99
USAEC Office Safeguards & Materials Management	UF_6	5,386,873	2,833,255	53	–	0		5,386,873	2,833,255	53
Y-12	U_3O_8	1,217,878	0		94,305,888	103,719	<1	95,523,766	103,719	<1
Total Receipts at PORTS		69,661,687,870	1,102,746,348	2	54,652,216,055	20,915,560	<1	124,313,903,925	1,123,651,908	1

APPENDIX III

PORTS SUMMARY HISTORICAL DATA
RECYCLED URANIUM WORKSHEET
(SHIPMENTS)

Facility	U Form	Startup thru FY 1970			FY 1971 thru March 1999			Total Quantity		
		Total Quantity of Uranium (g)	Total Quantity of Recycled Uranium (g)	Percent Recycled	Total Quantity of Uranium (g)	Total Quantity of Recycled Uranium (g)	Percent Recycled	Total Quantity of Uranium (g)	Total Quantity of Recycled Uranium (g)	Percent Recycled
France	UF_6	904,052	0		126,997,607	64,553	<1	127,901,659	64,553	<1
K-25	UF_6	456,452,730	3,102,118	1	191,526,286	0		647,979,016	3,102,118	<1
Paducah	UF_6	53,549,902,918	1,502,414	<1	48,837,760,004	(368,000)	<1	102,387,662,922	1,134,414	<1
Total Shipments from PORTS		54,007,259,700	4,604,532	<1	49,156,283,897	(303,447)	<1	103,163,543,597	4,301,085	<1

APPENDIX IV

PORTS CASCADE FEED

Fiscal Year	Paducah Product MTU	Reactor Tails MTU	Feed Source		PORTS Recycled Tails MTU	Commercial Sources MTU	Misc. MTU	Total MTU
			X-705 Oxide Conversion MTU	PORTS Feed Plant MTU				
1955	5,268.5	93.4	–	–	8,752.3	–	2.3	14,112.4
1956	2,852.9	363.1	–	–	5,507.2	1,090.7	0.8	9,814.7
1957	4,459.1	6.2	0.4	–	48.4	–	2.4	4,516.5
1958	3,392.2	64.2	1.9	151.0	19.8	1,283.1	0.8	4,913.0
1959	3,071.6	–	1.5	1,959.6	–	0.3	1.9	5,034.9
1960	2,945.2	–	0.8	2,513.5	–	–	2.5	5,462.4
1961	2,898.6	–	1.2	2,833.6	–	68.2	3.1	5,804.7
1962	2,841.9	–	0.8	1,210.0	–	1,575.1	1.4	5,629.2
1963	2,872.2	–	0.5	–	0.1	2,105.4	1.0	4.979.2
1964	2,157.9	–	0.9	0.2	52.4	2,623.2	1.3	4,836.1
1965	1,274.7	–	3.6	–	1129.7	–	14.5	7,981.5
1966	1,284.3	–	4.9	–	–	–	202.1	5,801.1
1967	755.8	–	0.3	–	–	–	28.6	4865.1
1968	690.5	–	4.8	–	–	0.5	29.6	4059.0
1969	2,236.2	–	15.8	–	6.3	0.4	19.7	4187.4
1970	1,950.9	168.2	9.6	–	–	0.3	42.0	4019.7
1971	2,553.9	–	3.5	–	–	3,374.7	28.2	6005.5
1972 (FX)	1,645.3	–	18.7	–	6.0	2,715.8	21.5	4,423.0
1972 (BXA)	439.0	–	8.8	–	7.5	965.6	78.8	1,634.5
1973 (BXA)	1,549.9	–	27.4	–	36.6	3,894.6	31.6	5,794.3
1974 (BXA)	1,118.4	400.0	15.1	–	207.6	4,007.9	86.5	5,907.4
1975 (BXA)	1,408.2	–	14.8	–	180.4	5,458.1	31.3	7,092.8
1976 (BXA)	1,166.5	–	22.1	–	4,649.3	4,649.0	2.6	10,489.5
1976-5 (BXA)	312.6	–	3.1	–	1,272.7	1,405.6	–	2,994.0
1977	1,008.8	–	7.7	–	5,055.9	5,981.2	2.7	12,056.3
1978	1,358.2	–	1.0	–	1,765.2	6,253.8	93.7	9,471.9
1979	996.8	–	–	–	–	6,587.1	152.5	7,736.4
1980	3,132.3	–	–	–	20.2	3,952.4	36.6	7,141.5
1981	2,749.0	–	–	–	16.6	3,825.7	279.9	6,871.2
1982	2,682.4	–	–	–	0.4	4,095.5	887.4	7,665.7
1983	4,179.8	–	–	–	–	4,063.6	958.6	9,208.2
1984	3,304.2	–	–	–	–	3,247.3	112.5	7,048.8
1985	3,107.8	–	–	–	–	4,012.0	98.2	7,218.0
1986	2,687.5	–	–	–	0.1	2,914.8	871.9	6,474.3
1987	3,339.2	–	–	–	1.1	3,864.3	609.4	7,814.0

APPENDIX IV

PORTS CASCADE FEED

Fiscal Year	Paducah Product MTU	Reactor Tails MTU	Feed Source		PORTS Recycled Tails MTU	Commercial Sources MTU	Misc. MTU	Total MTU
			X-705 Oxide Conversion MTU	PORTS Feed Plant MTU				
1988	3,901.2	–	–	–	1.7	2,681.3	507.6	7,091.8
1989	4,212.8				–	4,809.0	1,053.4	10,077.1
1990	3,795.5	–	–	–	2.5	8,349.7	197.9	12,345.6
1991	4,752.4	–	–	–	2.3	6,408.4	198.5	11,361.6
1992	4,305.3	–	–	–	3.7	6,448.6	125.8	10,883.4
1993	4,643.1	–	0.1	–	2.3	1,432.4	224.0	6,301.9
1994	3,884.9	–	5.7	–	1.7	5,907.8	119.3	9,919.4
1995	4,193.7	–	2.7	–	1.0	4,814.2	29.9	9,041.5
1996	3,924.9	–	–	–	7.3	3,266.7	27.4	7,226.3
1997	4,179.3	–	–	–	1.0	3,326.7	26.4	7,504.4
1998*								
1999* (3/31)								
Total	121,485.4							320,817.2*

*Data was unavailable for FY 1998 – mid-FY 1999.

APPENDIX V

REACTOR RETURN FEED CYLINDERS

FED TO PORTS CASCADE BETWEEN FY 1955-FY 1958

Item	Cylinder Number	% Assay	Feed Date	Item	Cylinder Number	% Assay	Feed Date
1	36842	0.6451	05/14/55	47	D-30764	0.6520	
2	85907	0.6606	05/18/55	48	D-22	0.6623	06/26/55
3	A-2877	0.6472	05/19/55	49	A-3581	0.6390	06/28/55
4	27185	0.6702	05/20/55	50	D-27575	0.6684	06/24/55
5	32731	0.6125	05/21/55	51	24841	0.6419	07/01/55
6	31041	0.6688	05/22/55	52	37293	0.6556	06/25/55
7	33680	0.6686	05/23/55	53	27474	0.6392	06/28/55
8	W-558	0.6680	05/24/55	54	76882	0.6405	07/01/55
9	A-3500	0.6690	05/25/55	55	28382	0.6753	06/27/55
10	97323	0.6708	05/26/55	56	32475	0.6448	06/27/55
11	36006	0.6688	05/27/55	57	29173	0.6406	06/29/55
12	W-589	0.6675	05/28/55	58	39303	0.6408	07/01/55
13	50754	0.6740	05/29/55	59	A-1398	0.6430	06/29/55
14	30802	0.6664	05/30/55	60	A-4130	0.6394	06/29/55
15	31684	0.6752	05/31/55	61	45464	0.6390	06/29/55
16	76516	0.6746	06/01/55	62	31152	0.6422	06/27/55
17	31347	0.6750	06/02/55	63	31512	0.6318	07/05/55
18	31458	0.6744	06/03/55	64	30042	0.6392	07/04/55
19	26241	0.6198	06/04/55	65	24956	0.6420	07/02/55
20	W-561	0.6188	06/05/55	66	37159	0.6422	07/03/55
21	43243	0.6766	06/06/55	67	26981	0.6406	07/04/55
22	28890	0.6726	06/07/55	68	39782	0.6411	07/03/55
23	A-2841	0.6724	06/08/55	69	85152	0.6422	07/04/55
24	27127	0.6621	06/09/55	70	2665	0.6416	07/06/55
25	31704	0.6716	06/10/55	71	33112	0.6389	07/01/55
26	21211	0.6705	06/11/55	72	D-18	0.6399	07/04/55
27	24004	0.6715	06/12/55	73	D-26071	0.6359	07/05/55
28	28063	0.6702	06/13/55	74	D-31903	0.6426	07/07/55
29	39398	0.6575	06/14/55	75	D-27293	0.6399	07/06/55
30	77252	0.6616	06/15/55	76	D-23970	0.6420	07/07/55
31	77020	0.6779	06/16/55	77	D-36164	0.6374	07/08/55
32	24106	0.6582	06/17/55	78	D-27122	0.6353	07/08/55
33	29199	0.6731	06/18/55	79	D-36326	0.6392	07/09/55
34	26429	0.6676	06/19/55	80	W-561	0.6393	07/09/55
35	21032	0.6737	06/20/55	81	D-86071	0.6366	07/10/55
36	D-32779	0.6490	06/20/55	82	D-31271	0.6379	07/10/55
37	D-40505	0.6432	06/21/55	83	23636	O. R.	07/11/55
38	D-37685	0.6610	06/18/55	84	32923	O. R.	07/11/55
39	D-48558	0.6548	06/27/55	85	80113	0.6375	07/12/55
40	D-52945	0.6469	06/19/55	86	A-3435	0.6363	07/12/55
41	D-78441	0.6503	06/22/55	87	85907	0.6377	07/13/55
42	D-31702	0.6468	06/21/55	88	29836	0.6370	07/13/55
43	D-80113	0.6537	06/19/55	89	39398	0.6368	07/14/55
44	D-89454	0.6411	06/23/55	90	36313	0.6804	07/14/55
45	D-31852	0.6636	06/17/55	91	A-3841	0.6385	07/15/55
46	D-29955	0.6400	06/23/55	92	37458	0.6372	07/15/55

APPENDIX V

REACTOR RETURN FEED CYLINDERS

FED TO PORTS CASCADE BETWEEN FY 1955-FY 1958

Item	Cylinder Number	% Assay	Feed Date	Item	Cylinder Number	% Assay	Feed Date
93	25350	0.6809	07/16/55	144	36164	0.6500	08/05/55
94	38905	0.6810	07/16/55	145	27123	0.6780	08/02/55
95	32601	0.6815	07/17/55	146	26620	0.6770	08/07/55
96	34756	0.6814	07/17/55	147	26716	0.6770	08/10/55
97	39533	0.6828	07/18/55	148	25350	0.6780	08/09/55
98	40787	0.6830	07/18/55	149	35144	0.6710	08/09/55
99	30392	0.6833	07/19/55	150	27474	0.6800	08/08/55
100	D-21241	0.6835	07/19/55	151	51313	0.6740	08/09/55
101	3500	0.6834	07/20/55	152	36337	0.6790	08/11/55
102	33680	0.6842	07/20/55	153	27599	0.6760	08/10/55
103	A-812	0.6816	07/21/55	154	32777	0.6720	08/11/55
104	27354	0.6824	07/21/55	155	A-3435	0.6500	08/07/55
105	31182	0.6527	07/22/55	156	80137	0.6610	08/06/55
106	26790	0.6806	07/23/55	157	23508	0.6720	08/13/55
107	24850	0.6815	07/23/55	158	A-3581	0.6880	08/11/55
108	A-3163	0.6822	07/24/55	159	23329	0.6880	08/14/55
109	78511	0.6824	07/25/55	160	86260	0.6880	08/14/55
110	39884	0.6820	07/26/55	161	25223	0.6721	08/15/55
111	23329	0.6446	08/01/55	162	37916	0.6743	08/15/55
112	23508	0.6447	07/24/55	163	28382	0.6743	08/09/55
113	37531	0.6410	07/24/55	164	D-33680	0.6796	08/15/55
114	36439	0.6410	07/27/55	165	D-78500	0.6806	08/16/55
115	D-80024	0.6448	07/29/55	166	A-2554	0.6781	08/16/55
116	W-340	0.6482	07/30/55	167	D-29961	0.6798	08/16/55
117	D-28432	0.6477	07/28/55	168	29836	0.6747	08/12/55
118	A-4367	0.6473	08/05/55	169	27757	0.6823	08/17/55
119	D-30853	0.6478	08/02/55	170	31041	0.6837	08/19/55
120	D-29961	0.6474	07/31/55	171	37508	0.6830	08/16/55
121	A-2955	0.6394	07/31/55	172	75688	0.6835	08/17/55
122	D-75445	0.6455	07/27/55	173	27505	0.6735	08/11/55
123	D-80335	0.6390	07/26/55	174	27724	0.6780	08/14/55
124	D-31012	0.6384	07/27/55	175	39533	0.6500	08/21/55
125	A-5118	0.6386	07/28/55	176	D-37927	0.6817	08/17/55
126	D-29468	0.6441	07/28/55	177	A-4367	0.6804	08/21/55
127	D-36173	0.6406	07/29/55	178	W-379	0.6835	08/18/55
128	25685	0.6391	08/03/55	179	32571	0.6802	08/19/55
129	56500	0.659	07/29/55	180	A-3841	0.6805	08/26/55
130	78500	0.6600	07/29/55	181	A-30042	0.6847	08/18/55
131	25875	0.6598	08/01/55	182	D-59394	0.6838	08/22/55
132	39356	0.6509	08/04/55	183	W-565	0.6833	08/20/55
133	38325	0.6730	08/02/55	184	38325	0.6844	08/19/55
134	34562	0.6750	08/05/55	185	36313	0.6835	08/18/55
135	31597	0.6720	08/05/55	186	28432	0.6837	08/27/55
136	40722	0.6700	08/06/55	187	A-3435	0.6852	08/20/55
137	77386	0.6700	08/14/55	188	26350	0.6665	08/25/55
138	W-565	0.6590	08/14/55	189	80113	0.6511	08/20/55
139	37309	0.6790	08/04/55	190	85907	0.6788	08/22/55
140	30042	0.6789	08/17/55	191	24241	0.6836	09/01/55
141	75688	0.6740	08/08/55	192	31271	0.6824	08/29/55
142	31494	0.6710	08/05/55	193	25685	0.6820	08/23/55
143	31271	0.6510	08/03/55	194	27474	0.6854	08/25/55

APPENDIX V

REACTOR RETURN FEED CYLINDERS

FED TO PORTS CASCADE BETWEEN FY 1955-FY 1958

Item	Cylinder Number	% Assay	Feed Date	Item	Cylinder Number	% Assay	Feed Date
195	35144	0.6820	08/21/55	246	31182	0.6743	09/08/55
196	CWS-562	0.6834	08/24/55	247	85907	0.6739	09/08/55
197	36437	0.6850	08/24/55	248	A-3435	0.6748	09/09/55
198	27123	0.6839	09/01/55	249	32779	0.6747	09/09/55
199	D-18	0.6842	08/26/55	250	37631	0.6764	09/10/55
200	36326	0.6845	08/22/55	251	23182	0.6808	09/10/55
201	48558	0.6851	08/22/55	252	37508	0.6787	09/11/55
202	75445	0.6844	08/23/55	253	48558	0.6735	09/11/55
203	40787	0.6848	08/26/55	254	26500	0.6719	09/12/55
204	A-3500	0.6827	08/23/55	255	34756	0.6743	09/12/55
205	27575	0.6835	08/11/55	256	W-56	0.6718	09/13/55
206	28382	0.6829	08/26/55	257	27122	0.6731	09/13/55
207	25350	0.6833	08/27/55	258	A-812	0.6737	09/12/55
208	29836	0.6834	08/28/55	259	77820	0.6722	09/14/55
209	39398	0.6829	08/30/55	260	A-4027	0.6725	09/13/55
210	31494	0.6839	09/03/55	261	A-495	0.6726	09/15/55
211	32777	0.6837	08/28/55	262	A-950	0.6737	09/12/55
212	32601	0.6831	08/29/55	263	39884	0.6740	09/11/55
213	31597	0.6865	09/03/55	264	W-558	0.6718	09/14/55
214	37309	0.6846	08/30/55	265	32479	0.6702	09/15/55
215	A-2955	0.6797	08/31/55	266	33202	0.6716	09/14/55
216	23329	0.6814	08/29/55	267	27973	0.6761	09/15/55
217	78500	0.6799	08/28/55	268	45525	0.6711	09/16/55
218	75688	0.6789	09/02/55	269	7615	0.6723	09/16/55
219	27724	0.6793	08/30/55	270	A-933	0.6708	09/17/55
220	80139	0.6796	08/31/55	271	25875	0.6671	09/18/55
221	27354	0.6816	09/01/55	272	24183	0.6666	09/18/55
222	78511	0.6804	09/03/55	273	D-34756	0.6683	09/19/55
223	W-558	0.6811	09/02/55	274	36861	0.6672	09/20/55
224	26716	0.6792	08/29/55	275	25066	0.6614	09/20/55
225	44783	0.6789	08/31/55	276	D-23182	0.6699	09/21/55
226	38726	0.6779	09/03/55	277	D-80142	0.6682	09/22/55
227	31779	0.6781	09/04/55	278	D-27757	0.6657	09/22/55
228	28056	0.6789	09/04/55	279	D-32601	0.6674	09/23/55
229	30768	0.6659	09/05/55	280	D-39884	0.6699	09/21/55
230	27757	0.6793	09/05/55	281	D-35146	0.6680	09/24/55
231	D-39394	0.6793	09/05/55	282	D-31182	0.6684	09/25/55
232	42656	0.6783	09/08/55	283	D-42656	0.6655	09/26/55
233	W-565	0.6743	09/12/55	284	A-812	0.6658	09/26/55
234	D-26790	0.6725	09/11/55	285	D-30581	0.6660	09/27/55
235	D-25875	0.6740	09/11/55	286	D-31597	0.6641	09/28/55
236	26790	0.6740	09/11/55	287	D-27575	0.6651	09/28/55
237	25875	0.6740	09/11/55	288	D-30993	0.6666	09/29/55
238	80142	0.6780	09/11/55	289	D-26350	0.6652	09/30/55
239	31708	0.6785	09/10/55	290	D-80113	0.6655	10/01/55
240	26730	0.6772	09/13/55	291	D-27123	0.6660	10/01/55
241	A-293	0.6770	09/10/55	292	D-28056	0.6655	10/02/55
242	39533	0.6750	09/09/55	293	D-33202	0.6655	10/02/55
243	A-4367	0.6756	09/08/55	294	45525	0.6644	10/03/55
244	D-80113	0.6754	09/04/55	295	37309	0.6690	10/03/55
245	D-36818	0.6744	09/06/55	296	4A002271	0.6671	11/28/56

APPENDIX V

REACTOR RETURN FEED CYLINDERS

FED TO PORTS CASCADE BETWEEN FY 1955-FY 1958

Item	Cylinder Number	% Assay	Feed Date	Item	Cylinder Number	% Assay	Feed Date
297	1714	0.6500	08/01/57	303	1668	0.6816	05/21/58
298	647	0.6600	04/19/58	304	82	0.6615	05/22/58
299	1830	0.6520	04/21/58	305	2365	0.6644	05/23/58
300	2016	0.6586	05/17/58	306	2371	0.6706	05/25/58
301	1943	0.6536	05/18/58	307	1632	0.6650	05/27/58
302	2372	0.6446	05/19/58				

Note:
1) Cylinder Nos. 1-295 inclusive are 2-1/2-ton cylinders
2) Cylinder Nos. 296-307inclusive are 10-ton cylinders
3) Cylinders Nos. 1-295 feed point cells were:
   a) 29-3-1,5
   b) 29-2-1,2,3,5,8,9,10
   c) 31-5-1,2,3,4,5,6,7,8,9,10
   d) 31-4-1,3,4,5,6,7,8,9,10
   e) 31-3-2,4,5,6,7,8,9,10
4) Cylinders Nos. 296-307 feed point cells were:
   a) 33-1-1,3,5,9
   b) 33-3-3

APPENDIX VI

REACTOR RETURN FEED CYLINDERS FED TO PORTS CASCADE (OCTOBER/NOVEMBER 1969)

Item	Date Fed	Cylinder No.	% Assay	Feed Point
1	10/26/69	2442	0.6448	33-1-6
2	10/28/69	1233	0.6443	33-1-4
3	10/29/69	1109	0.6448	33-1-2
4	10/31/69	1187	0.6444	33-1-2
5	11/1/69	761	0.6438	33-1-2
6	11/2/69	486	0.6440	33-1-2
7	11/4/69	1848	0.6425	33-2-7
8	11/5/69	1121	0.6446	33-1-2
9	11/6/69	671	0.6434	33-2-7
10	11/8/69	1570	0.6444	33-2-7
11	11/9/69	457	0.6440	33-1-2
12	11/10/69	1066	0.6438	33-1-2
13	11/11/69	273	0.6439	33-1-2
14	11/13/69	839	0.6429	33-1-2
15	11/14/69	100	0.6444	33-1-2
16	11/15/69	3304	0.6440	33-1-2
17	11/17/69	3174	0.6450	33-1-2
18	11/18/69	1053	0.6458	33-1-2
19	11/19/69	2412	0.6432	33-1-2
20	11/21/69	669	0.6454	33-3-1
21	11/22/69	1013	0.6452	33-3-1
22	11/23/69	1220	0.6456	33-3-1
23	11/25/69	1515	0.6456	33-3-1
24	11/26/69	3321	0.6442	33-3-3
25	11/27/69	267	0.6442	33-3-3
26	11/29/69	961	0.6425	33-3-3

APPENDIX VII

REACTOR RETURN FEED CYLINDERS FED TO PORTS CASCADE (JANUARY 1974)

Page 1 of 2

Item	Date Fed	Cylinder No.	% Assay	Feed Point
1	1/1/74	1624	0.6424	33-4-3
2	1/1/74	53	0.6440	33-4-3
3	1/2/74	105	0.6440	33-4-3
4	1/3/74	176	0.6435	33-4-3
5	1/3/74	116	0.6435	33-4-5
6	1/3/74	1006	0.6442	33-4-5
7	1/4/74	147	0.6434	33-4-5
8	1/4/74	900	0.6424	33-4-5
9	1/5/74	91	0.6441	33-4-5
10	1/5/74	577	0.6434	33-4-7
11	1/6/74	729	0.6424	33-4-7
12	1/6/74	181	0.6440	33-4-7
13	1/7/74	1063	0.6442	33-4-7
14	1/7/74	1008	0.6441	33-4-7
15	1/8/74	1735	0.6438	33-4-7
16	1/8/74	2341	0.6439	33-4-7
17	1/9/74	2204	0.6422	33-4-7
18	1/9/74	635	0.6436	33-4-7
19	1/10/74	2175	0.6429	33-4-7
20	1/10/74	1113	0.6442	33-4-7
21	1/10/74	834	0.6440	33-4-7
22	1/11/74	1316	0.6339	33-4-7
23	1/11/74	2198	0.6339	33-4-7
24	1/12/74	1342	0.6438	33-4-7
25	1/12/74	1075	0.6429	33-4-7
26	1/12/74	1641	0.6421	33-4-7
27	1/13/74	2017	0.6475	33-4-7
28	1/13/74	1206	0.6422	33-4-7
29	1/14/74	1737	0.6426	33-4-7
30	1/14/74	1143	0.6439	33-4-7
31	1/15/74	1531	0.6429	33-4-7
32	1/15/74	3106	0.6447	33-4-7
33	1/15/74	477	0.6434	33-4-7
34	1/16/74	387	0.6429	33-4-7
35	1/16/74	1042	0.6427	33-4-7
36	1/17/74	2281	0.6422	33-4-7
37	1/17/74	465	0.6422	33-4-7
38	1/17/74	119	0.6425	33-4-7
39	1/18/74	1545	0.6421	33-4-7
40	1/19/74	3147	0.6430	33-4-7
41	1/19/74	1077	0.6437	33-4-7
42	1/19/74	3241	0.6426	33-4-7
43	1/20/74	943	0.6416	33-4-7

APPENDIX VII

REACTOR RETURN FEED CYLINDERS FED TO PORTS CASCADE (JANUARY 1974)

Item	Date Fed	Cylinder No.	% Assay	Feed Point
44	1/20/74	2164	0.6442	33-4-7
45	1/21/74	3124	0.6428	33-4-7
46	1/21/74	259	0.6450	33-4-7
47	1/22/74	3312	0.6450	33-4-7
48	1/22/74	148	0.6426	33-4-7
49	1/22/74	1110	0.6448	33-4-7
50	1/23/74	2070	0.6445	33-4-7
51	1/23/74	2025	0.6430	33-4-7
52	1/24/74	3359	0.6432	33-4-7
53	1/24/74	1900	0.6432	33-4-7
54	1/25/74	3314	0.6450	33-4-7
55	1/25/74	211	0.6439	33-4-7
56	1/25/75	2176	0.6431	33-4-7
57	1/26/74	1693	0.6426	33-4-7
58	1/26/74	3329	0.6447	33-4-7
59	1/27/74	71	0.6452	33-4-7
60	1/27/74	1000	0.6425	33-4-7
61	1/27/74	513	0.6425	33-4-7
62	1/28/74	3353	0.6439	33-4-7

APPENDIX VIII

REACTOR RETURN FEED CYLINDERS RECEIVED FROM DIVISION OF INTERNATIONAL AFFAIRS AND FED TO PORTS CASCADE

Item	Cylinder No.	Date Received	Timeframe Fed*	Assay %	Net Weight kgU	Heel kgU
1	050189	12/27/67	1/68-1/93	78.23	15.680	0.3
2	050417	12/27/67	1/68-10/77	80.10	15.591	0.3
3	050502	12/23/67	1/68-10/77	82.55	16.323	0.3
4	050600	12/27/67	1/68-10/77	78.20	15/299	0.3
5	050626	12/27/67	1/68-10/77	78.90	15.421	0.3
6	053874	12/27/67	1/68-10/77	80.47	16.175	0.3
7	050142	1/26/68	1/68-10/77	83.91	14.523	0.3
8	050287	1/26/68	1/68-10/77	82.99	9.457	0.3
9	050240	1/26/68	1/68-10/77	83.24	15.844	0.3
10	051879	1/26/68	1/68-10/77	83.99	16.151	0.3
TOTAL					150.454	3.0

*Actual feed dates not available in time for this report.

APPENDIX IX

REACTOR RETURN FEED CYLINDERS RECEIVED FROM BABCOCK AND WILCOX AND FED TO PORTS CASCADE

ITEM	CYLINDER NUMBER	DATE RECEIVED	TIMEFRAME FED*	ASSAY %	Net Weight kgU	SHIPMENT NO. TO NFS	HEEL kgU
1	054066	12/17/76	10/77 – 1/93	96.700	12.0		
2	050020	12/17/76	10/77 -	91.471	13.6		
3	050360	12/17/76	10/77 – 1/93	97.487	15.3	22	0.310
4	050225	12/17/76	10/77 – 1/93	91.470	14.2		
5	054075	12/17/76	1/93 -	97.344	14.3	45	0.341
6	054060	12/17/76	1/93 -	83.76	14.6	45	0.210
7	051933	12/17/76	10/77 – 1/93	92.58	16.8		
8	050126	12/17/76	1/77 – 10/77	76.05	14.9		
9	050012	12/17/76	1/77 – 10/77	76.05	15.0		
10	050513	12/17/76	10/77 -	94.853	16.2		
11	050562	12/17/76	10/77 -	93.420	7.0		
		TOTAL			153.9		

*Actual feed dates not available in time for this report.

APPENDIX X

REACTOR RETURN FEED CYLINDERS RECEIVED FROM USAEC OFFICE SAFEGUARDS AND MATERIALS MGMT. AND FED TO PORTS CASCADE

ITEM	CYLINDER NUMBER	DATE RECEIVED	TIMEFRAME FED	ASSAY %	kgU	SHIPMENT NO.	HEEL kgU
1	D29462	12/2/68	*	1.332	1405.3		
2	D39829	12/2/68	*	1.374	1383.2		
3	050378	11/5/68	11/68 – 10/77**	78.03	16.4	24	0.262
4	050666	11/5/68	10/77 – 1/93**	79.04	16.4	32	0.317
5	050609	11/5/68	11/68 – 10/77**	71.69	12.9		
TOTAL					2834.2		

*Not fed to PORTS cascade as of March 31, 1999, remains in storage.
**Actual feed dates not available in time for this report.

APPENDIX XI

REACTOR RETURN FEED CYLINDERS RECEIVED FROM NUMEC AND FRANCE AND FED TO PORTS CASCADE

NUMEC							
ITEM	CYLINDER NUMBER	DATE RECEIVED	TIMEFRAME FED	ASSAY %	NET WEIGHT kg	SHIPMENT NO.	HEEL kgU
1	050178	8/3/71	6/97	79.85	14.587	20	0.595
2	051861	8/3/71	3/98	79.83	14.587	35	0.272
3	050013	8/6/71	6/98	79.86	13.239	54	0.467
4	050191	8/6/71	3/97	79.86	13.239	22	0.415
5	050059	8/6/71	3/97	79.76	13.239	19	0.350
6	050469	8/6/71	1/98	79.81	13.239	28	0.260
7	050433	8/12/71	6/97	80.08	16.172	27	0.476
8	051919	8/12/71	6/98	79.80	16.172	Stored in X-744G	
9	050207	8/17/71	5/97	79.34	12.487	21	0.738
10	050290	8/17/71	12/97	80.06	12.487	25	0.294
11	051467	8/20/71	2/98	80.03	10.552	33	0.341
12	052362	8/20/71	3/98	79.94	10.552	41	0.334
13	050384	8/23/71	6/97	80.02	15.067	27	0.577
14	051967	8/23/71	2/98	79.82	15.067	38	0.623
15	050112	8/24/71	1/97	80.13	13.628	1	0.575
16	050111	8/27/71	3/97	81.87	14.229	19	0.412
17	050131	8/27/71	2/97	80.14	14.229	24	0.610
18	051938	8/27/71	1/98	79.61	14.229	38	0.623
19	054053	8/30/71	1/98	80.58	13.110	44	0.496
20	052289	8/30/71	2/98	79.89	13.110	40	0.292
SUBTOTAL (NUMEC)					273.218		8.75

TransNuclear – France							
ITEM	CYLINDER NUMBER	DATE RECEIVED	TIMEFRAME FED	ASSAY %	NET WEIGHT kg/U	SHIPMENT NO.	HEEL kgU
1	050556	8/22/73		79.92	15.594	37	0.323
2	053934	8/22/73	5/98	79.92	15.594	43	0.260
3	050174	9/27/73	2/97	79.82	14.772	20	0.505
4	050619	9/27/73	3/98	79.79	14.772	31	0.196
5	050430	4/5/74	9/97	61.77	15.011	23	0.161
6	050681	4/5/74	11/97	64.89	15.011	35	0.153
7	051859	4/5/74	1/98	66.76	15.011	35	0.262
8	051942	4/5/74	9/97	57.96	15.011	38	0
9	050227	4/5/74	1/98	58.32	15.011	22	0.077
10	050014	4/5/74	6/97	57.61	15.011	22	0.117
11	050202	6/13/74	5/97	73.48	15.246	20	0.167
12	054032	6/13/74	9/97	73.48	15.246	44	0.161
13	054059	6/13/74	3/98	73.48	15.246	45	0.152
14	054090	6/13/74	11/97	82.59	15.246	46	0.238
15	050153	6/13/74	5/97	82.59	15.246	20	0.292
16	052365	7/30/74	5/98	79.97	16.407	37	0.358
17	050281	7/30/74	9/97	78.94	15.969	24	0.248
18	050104	7/30/74	--	78.94	15.969	1	0.201
19	051601	7/30/74	7/97	79.69	15.927	34	0.199
20	051520	7/30/74	1/98	79.43	15.927	34	0.197
21	051956	7/30/74	5/98	79.33	15.927	37	0.455
22	053928	7/30/74	3/98	80.05	15.927	43	0.219
23	051890	7/30/74	8/97	79.54	15.927	36	0.220
24	050296	10/22/75	6/97	77.46	8.867	25	0.962
25	050630	9/10/75	12/97	77.29	15.127	32	0.166
26	051473	9/10/75	7/97	78.26	15.127	33	0.502
27	051485	9/10/75	11/97	77.99	15.127	33	0.255
28	051528	9/10/75	1/98	78.17	15.127	34	0.090
29	052355	9/10/75	--	79.69	15.127	41	0.199
30	054088	9/10/75	9/97	78.36	15.127	45	0.216
31	054099	9/10/75	7/97	78.33	15.127	46	0.400
32	050143	9/10/75	1/97	76.78	15.127	20	0.315
33	054051	9/10/75	3/98	77.66	15.127	36	0.230
34	052300	5/5/76	3/98	79.68	5.219	40	0.207
35	053917	5/5/76	5/98	80.24	5.219	43	0.143
36	050293	5/5/76	3/98	81.59	5.219	23	0.352
37	050035	5/5/76	3/97	81.59	5.219	19	0.304
38	050320	9/10/75	6/97	78.48	15.127	25	0.571
39	053870	5/5/76	9/97	81.41	11.353	41	0.148
40	050343	7/21/76	6/97	80.04	15.965	23	0.227
41	050289	7/21/76	12/97	79.26	15.965	24	0.220
42	052335	7/21/76	9/97	79.26	15.965	40	0.190
43	051521	7/21/76	7/97	79.26	15.965	33	0.298
44	050203	7/21/76	5/97	80.04	15.965	21	1.010

TransNuclear – France (cont'd)							
ITEM	CYLINDER NUMBER	DATE RECEIVED	TIMEFRAME FED	ASSAY %	NET WEIGHT kg/U	SHIPMENT NO.	HEEL kgU
45	050250	7/21/76	11/97	80.04	15.965	22	0.381
46	050086	7/21/76	2/97	80.04	15.965	1	0.311
47	050713	2/4/77	1/98	79.69	15.965	36	0.110
48	050268	2/4/77	–	77.93	13.772	21	0.433
49	050367	2/4/77	8/97	76.33	13.772	22	0.462
50	050390	2/4/77	1/98	79.64	13.772	26	0.280
51	050163	2/4/77	1/97	79.68	13.772	20	0.283
52	050189	2/4/77	5/97	79.64	13.772	20	0.282
53	050018	2/4/77	–	77.03	13.772	1	0.900
54	050253	8/11/77	1/98	78.22	8.708	22	0.144
55	050264	6/2/77	6/97	82.42	16.335	21	1.006
56	050574	1/16/78	–	79.24	19.268		
57	053945	1/16/78	8/97	81.01	19.268	43	0.468
58	050338	1/16/78	–	–	19.268	22	0.221
59	050378	11/16/78	8/97	72.81	19.268	24	0.262
60	050397	1/16/78	9/97	56.15	19.268	22	0.125
61	050700	1/16/78	9/97	56.54	19.268	36	0.138
62	051522	1/16/78	7/97	72.95	19.268	34	0.282
63	050032	1/16/78	3/97	58.75	19.268	19	0.249
64	050010	1/16/78	3/98	87.60	19.268	24	0.961
65	050595	7/20/78	2/98	73.48	14.017	31	0.113
66	050491	7/20/78	3/98	72.85	14.017	30	0.127
67	050696	7/20/78	3/98	73.21	14.017	33	0.243
SUBTOTAL (FRANCE)					996.039		19.947
GRAND TOTALS					1,269.257		28.697

NOTES:

1. Material received FY 1972 through FY 1978 (8/3/71 – 7/20/98)
2. Material fed to PORTS cascade FY 1973 through FY 1998 (1/97 – 6/98)
3. Eighty-seven 5A cylinders fed, 1.4MTU
4. Cylinders shipped to NFS and cleaned (0.029 MTU heels)
5. Feedpoint cells for the 87 5" cylinders were:
   a. 25-7-3
   b. 25-7-7

APPENDIX XII

REACTOR RETURN UF_6 FEED
RECEIVED FROM NATIONAL LEAD OF OHIO (FERNALD)
THROUGH PORTS OXIDE CONVERSION FACILITY

Item	Cylinder No.	Date Produced	Timeframe Fed*	Assay %	Net Weight kgU	Heel kgU	In Storage as of March 31, 1999
1	STO261	1/76	1/76-10/77	2.922	1.096		
2	STO368	1/76	1/76-10/77	2.922	1.738		
3	STO183	1/76	1/76-10/77	2.922	1.176		
4	STO160	1/76	1/76-10/77	2.922	7.770		
5	STO403	5/76		3.562	1.716	--	Yes
6	120076	1/76		2.923	7.079	--	Yes
7	120157	1/76	1/76-10/77	2.922	135.837		
8	120154	1/76	1/76-10/77	2.922	134.322		
9	120007	1/76	1/76-10/77	2.922	137.561		
10	120180	1/76	1/76-10/77	2.922	133.375		
11	120187	5/76	--	3.017	114.555	--	Yes
12	120175	5/76	5/76-10/77	2.920	6,523		
13	120117	5/76	5/76-10/77	2.920	12,001		
14	300195	5/76		2.920	1,371.542	--	Yes
15	300076	5/76		2.920	517.279	--	Yes
16	300173	5/76		2.920	1,215.469	--	Yes
17	300208	1/76		2.920	1,491.435	--	Yes
TOTAL					5,290.474		

*Four 2-1/2 ton cylinders, two 12", and one 5" have not been fed (in storage) as of March 31, 1999;
**Actual feed dates and cylinder cleaning information to be determined.

APPENDIX XIV

OXIDE CONVERSION FACILITY AIR SAMPLER RESULTS – TOTAL ALPHA CONCENTRATION

TOTAL ALPHA CONCENTRATION (uCi/ml) BY SAMPLE LOCATION

Year/Loc. Code *	GI	GJ	GK	GL	GM	GN	GO	GP
1963	1.73E-10							
1964	1.53E-10							
1965	5.13E-10	5.79E-10						
1966	3.91E-10	1.13E-11						
1967	2.43E-11	1.46E-12	2.73E-12	1.46E-12	5.33E-12	8.87E-12	1.53E-12	5.39E-12
1968	9.58E-11	1.22E-11	7.01E-11	8.12E-11	1.21E-10	7.18E-11	8.63E-11	1.36E-10
1969	9.92E-11	2.95E-11	6.63E-11	2.17E-10	2.76E-10	2.42E-10	5.33E-11	2.60E-10
1970	8.36E-12	1.22E-11	4.40E-12	1.76E-11	1.69E-11	2.63E-11	4.69E-12	1.86E-11
1971	2.25E-11	3.49E-11	3.21E-11	6.77E-11	8.12E-11	3.15E-11	3.28E-11	1.04E-10
1972	3.45E-11	1.01E-11	3.53E-11	5.02E-11	6.94E-11	3.88E-11	3.15E-11	8.66E-11
1973	4.17E-11	1.72E-10	8.30E-11	1.30E-10	1.55E-10	1.82E-10	6.13E-11	3.87E-11
1974	2.22E-11	1.74E-11	1.78E-11	8.01E-11	1.18E-10	1.10E-10	1.74E-11	1.01E-10
1975	2.20E-10	7.92E-11	2.40E-10	1.09E-10	1.57E-10	9.60E-11	3.15E-11	1.68E-10
1976	4.34E-11	6.23E-11	294E-11	9.33E-11	9.27E-11	3.82E-11	1.52E-11	1.43E-10
1977	3.47E-11	1.43E-11	2.04E-11	5.74E-11	2.59E-11	2.72E-11	2.19E-11	1.14E-10
1978	1.16E-11	1.64E-12	5.26E-12	8.30E-12	5.17E-12	6.15E-12	2.86E-12	9.97E-12

*NOTE: The following provides the Oxide Conversion Air Sampler Locations, Descriptions, and Calculated DAC for this appendix

Loc. Code	Description of Sampler Location	Uranium Solubility Class	Calculated DAC** (uCi/ml)
GI	E-Area, Cold trap Room Scales Area	Class D	2.77×10^{-10}
GJ	H-Area, Glove Box	Class D	2.77×10^{-10}
GK	E-Area, Cold Trap Room NaF Trap Area	Class D	2.77×10^{-10}
GL	E-Area, Tower Room Ash Reactor, Glove Box	Class Y	1.9×10^{-11}
GM	E-Area, Tower Room Glove Box, West	Class Y	1.9×10^{-11}
GN	E-Area, Unloading Glove Box	Class Y	1.9×10^{-11}
GO	E-Area, Calciner	Class Y	1.9×10^{-11}
GP	E-Area, Tower Room Glove Box, South	Class Y	1.9×10^{-11}

**NOTE: The effective DAC for these areas as shown below:

$$f_{Th228}/DAC_{Th228} + f_{Th230}/DAC_{Th230} + f_{U234}/DAC_{U234} + f_{U235}/DAC_{U235} + f_{U236}/DAC_{U236} + f_{U238}/DAC_{U238} + f_{Np237}/DAC_{Np237} + f_{AmPu}/DAC_{AmPu} = 1/DAC_{Eff}$$

f is the ratio of the isotopic activity divided by the total activity

Using the Fractions of the total activity for the nuclides as listed above in H area, the effective DAC is for Class D U and Class W Th/TRU:

$.00016/4x10^{-12} + .00293/3x10^{-12} + 0.945/5x10^{-10} + .0346/6x10^{-10} + .0039/6x10^{-10} + .012/6x10^{-10} + 0.00086/2x10^{-12} + .00037/2x10^{-12} = 1/DAC = 3.6x10^{9}$; $DAC = 2.8x10^{-10}$

Without the TRU but including the Th, the DAC would be:

$f_{Th228}/DAC_{Th228} + f_{Th230}/DAC_{Th230} + f_{U234}/DAC_{U234} + f_{U235}/DAC_{U235} + f_{U238}/DAC_{U238} = 1/DAC_{Eff}$

$.00016/4x10^{-12} + .00295/3x10^{-12} + 0.950/5x10^{-10} + .0348/6x10^{-10} + .0124/6x10^{-10} = 1/DAC = 3.0x10^{9}$; $DAC = 3.33x10^{-10}$

(3.33-2.77)/3.33 = 0.168 or 17% more dose from inhalation of the recycled constituents present in the material. In this case the dose from TRU is significant using the 10% rule.

For the Case of Class Y uranium, Th, Pu and Class W Np in E Area:

$.00035/7x10^{-12} + .00793/7x10^{-12} + .922/2x10^{-11} + .037/2x10^{-11} + .006/2x10^{-11} + .023/2x10^{-11} + .00075/2x10^{-12} + .00283/6x10^{-12} =$
$1/DAC = 5.1x10^{9}$; $DAC = 1.94x10^{-11}$;

The class Y case without TRU does not need to be calculated since the effect is so small

(2.00-1.94)/2.00 = 0.03 or 3%

In this case of insoluble TRU being present, the dose increase from the TRU constituents is insignificant. The then current PAL of $4.8x10^{-11}$ is still two and a half times the current effective DAC.

APPENDIX XV

X-705 DECONTAMINATION AREA – AIR SURVEY RESULTS

1993-1994

Sample #'s (HPX-)	Location*	Am/Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
93-339, 94-06, 176, 135	GB	2.1	0.2	948	0.24	3.21	4.0
93-306, 307, 445, 94-39, 136	GC	1.4	1.2	2,065	0.13	6.97	8.8
93-308, 472, 94-01, 02, 23, 29, 33, 40, 51, 57, 81, 137, 206	GD	6.8	9.1	4,538	0.35	1.28	1.8
93-340, 473, 94-09, 41, 138, 207	GF	3.4	2.8	2,311	0.27	3.20	4.6
93-309, 446, 94-34, 42, 10, 67, 69, 82, 92, 96, 139 ,208	GG	7.9	7.1	8,660	0.17	3.71	5.6
93-341, 452, 474, 94-43, 11, 140, 209	GH	8.6	3.3	1,945	0.61	3.28	4.4
93-448, 94-12, 36, 46, 143, 300	GQ	2.8	0.2	2,299	0.13	2.34	3.2
93-343, 476, 94-47, 13, 85, 95, 99, 144	GR	3.2	1.3	2,657	0.17	2.61	5.0
94-31, 48, 58, 107, 115, 153, 204, 301, 218	GS	2.7	1.4	8,809	0.05	6.88	9.9
93-449, 94-14, 145	GT	2.0	0.6	1,902	0.14	3.35	4.7
93-311,450, 94-49, 146, 302	GU	2.9	1.2	1,819	0.23	2.99	3.9
93-344, 477, 94-32, 50, 16, 147, 303	GV	2.8	2.0	2,219	0.22	2.64	3.6

APPENDIX XV

X-705 DECONTAMINATION AREA – AIR SURVEY RESULTS

1995-1996

Sample #'s (HPX-)	Location*	^{239}Pu pCi	^{238}Pu pCi	^{237}Np pCi	Total pCi	% TRU	uCi/gU	% ^{235}U
95-10, 87, 113, 203, 282	GB	0	0	2.8	4,838	0.06	19.58	21.0
95-46, 135, 137, 223, 297, 96-330	GC	0.0	0.0	0.5	1,625	0.03	4.40	6.4
95-47, 136, 169, 294, 273, 298, 96-331	GD	0.5	0.0	2.1	4,120	0.06	2.59	3.8
95-11, 88, 170, 205, 274, 299, 96-332	GF	0.0	0.0	0.9	2,312	0.04	4.85	7.0
95-12, 89, 138, 171, 206, 275, 300, 96-333	GG	0.1	0.0	0.7	5,655	0.01	3.82	5.7
95-13, 90, 207, 301, 96-334	GH	0.3	0.0	0.4	1,437	0.05	5.44	7.5
95-15, 94, 140, 208, 277, 303	GQ	0.0	0.0	0.4	1,803	0.02	4.17	5.6
95-16, 141, 172, 278, 283	GR	0.1	0.0	1.0	3,634	0.03	4.26	7.8
95-92, 106, 112, 123, 159, 168, 180, 212, 214, 215, 216, 217, 222, 304, 96-336	GS	1.8	2.7	21.7	63,287	0.04	25.81	31.6
95-17, 93, 209, 305, 96-337	GT	0.0	0.0	0.2	998	0.02	7.06	10.2
95-18, 142, 210, 284, 306, 96-338	GU	0.0	0.0	0.6	1,915	0.03	5.55	8.2
95-19, 95, 143, 211, 285, 96-339	GV	0.0	0.0	0.7	2,399	0.03	3.86	5.5

*Note: The locations of the X-705 Continuous Air Sampler locations listed above (and described below) are all located within the High Bay of the X-705 except for the sampler GS, which is located within the South Annex. Since the mission of the South Annex is to permit the disassembly of process equipment, it was included, as well.

GB	Small Parts Decontamination @ Column J-17
GC	Calciner Area, Recovery between Cols. AA-13 and AA-14
GD	Compressor Maintenance @ Col. G-6
GF	Decontamination Tunnel @ Col. D-13
GG	Grinding Booth @ Col. D-5
GH	Detubing Booth @ Col. G-20
GQ	Compressor Maintenance
GR	Truck Alley
GS	South Annex
GT	Col. G-25
GU	Col. E-18
GV	Col. G-12

Appendix E
PORTSMOUTH DOCUMENT RELEASE FORM
Page 1 of 1

Document Description (To Be Completed By Requester)

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