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This document consists of
58 pages.

WASTE MANAGEMENT PROGRAM
CHEMICAL PROCESSING DEPARTMENT

Compiled by:

R. E. Tomlinson

And Staff of
Advance Technical Planning
Research and Engineering Operation
Chemical Processing Department

February 25, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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-3-

HW-76352

TABLE OF CONTENTSPage

1. INTRODUCTION	4
2. PREMISES AND OBJECTIVES	4
3. SUMMARY	5
4. HISTORY AND CURRENT STATUS	9
4.1 Dispersal of Gaseous Effluents	9
4.2 Burial of Solid Wastes	10
4.3 Ground Disposal of Low-Level Liquids	11
4.4 Tank Storage of High-Level Liquids	15
4.5 Fission Product Recovery	19
5. PLANNED ACTIVITIES	21
5.1 Gaseous Wastes	22
5.2 Solid Wastes	22
5.3 Low-Level Liquid Wastes	22
5.4 In-Tank Solidification of Non-Heating Wastes	23
5.5 Tank Storage of Self-Heating Wastes	26
5.6 Waste Fractionization	28
5.7 Miscellaneous Requirements for Waste Handling	33
5.8 Fission Product Recovery	33
6. ESTIMATED COSTS	35
7. ALTERNATIVES	37
7.1 Gaseous Wastes	37
7.2 Solid Wastes	37
7.3 Low-Level Liquid Wastes	37
7.4 High-Level Liquid Wastes	37
7.5 Fractionization Processes	42
7.6 Packaging of Separated Isotopes	43
8. REFERENCES	44
9. TABLES AND FIGURES	46

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HW-76352

-2-

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HW-76352

-4-

WASTE MANAGEMENT PROGRAM
CHEMICAL PROCESSING DEPARTMENT

1. INTRODUCTION

During the 18-year history of Hanford operations, a high degree of safety has been provided at reasonable cost by the storage of high-activity liquid wastes in underground tanks, by the percolation of low-activity liquid wastes (principally water) through the soil to the ground water, and by the adsorption or filtration of radioactive materials from gaseous effluents prior to their discharge to the atmosphere. Studies of the Hanford area indicate that current practices could be continued almost indefinitely without jeopardizing environmental safety, provided the stored liquid wastes are periodically transferred to new tanks before the existing ones fail. A risk still exists, however, in that fission products are stored in a mobile form, and continual surveillance is required to assure that the stored materials are in fact safely contained. This risk is especially pertinent when the adverse effects of potential events (e.g., earthquakes, military action, etc.) are considered. In view of these considerations, the in-tank storage of radioactive liquids has been considered to be a temporary practice, and improvement efforts have been focused on the immobilization of hazardous materials. As a guide for improvement efforts, waste management plans were documented formally in 1958⁽¹⁾, and revised in 1960⁽²⁾, 1961⁽³⁾, and 1962⁽⁴⁾.

In 1960, a Fission Product Recovery Program⁽⁵⁾ was prepared outlining a plan to meet the Atomic Energy Commission's needs for isolated fission products. Emphasis was placed on developing processes for isolating selected fission products and on providing interim production capability for recovering the fission products of immediate interest. In 1961, development efforts were channeled almost exclusively toward meeting the immediate needs of the Commission for separated strontium-90 and cesium-137. During this time, it became increasingly evident that substantial economies could be realized, both in immediate development and subsequent operating efforts, if the separate fission product recovery and waste management programs were merged. The integration of these programs was therefore effected, as described in the 1962 Waste Management Program.⁽⁴⁾ This document describes the current status and goals of the integrated program.

2. PREMISES AND OBJECTIVES

The Waste Management Program is concerned with the safe and economic processing and disposal of radiochemical effluents from the chemical processing operations at Hanford. The objectives of this program are to provide the technology and facilities required to achieve the following:

- 1) to control and minimize the chemical and radiochemical hazards, both short-term and long-term, associated with the waste effluents from the separations processes;

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HW-76352

-5-

- 2) to recover and store (or load for shipment) fission products as requested by the Commission;
- 3) to minimize the time and cost required to acquire the necessary technology and to implement the program; and
- 4) to maintain the programming flexibility to meet the changing needs of the Commission.

After the required technology is acquired, the costs of waste management are considered to include two elements:

- 1) the allocated capital and operating costs of processing radiochemical wastes to forms suitable for long-term storage, and of placing the processed wastes in such storage; and
- 2) the future costs for surveillance, replacement storage facilities, and waste transfer or other processing operations.

These costs must be appraised in terms of the following potential liabilities:

- 1) claims, arising from the inadvertent release of radioactive materials through some unforeseen mechanism; and
- 2) other intangible factors, such as loss of public confidence and support of atomic energy programs, potentially arising from the inadvertent release of radioactive materials.

3. SUMMARY

Essentially all of the fission products produced at Hanford are currently being stored in about sixty-four million gallons of alkaline slurries in underground tanks ranging from eight to nineteen years old. Research and development programs are being actively pursued to increase the certainty of confinement of these radioactive materials. The plans presented here are believed to be optimum for Hanford, making productive use of the environmental advantages and facilities available.

3.1 Gaseous Wastes

The current practice of discharging gaseous wastes to the atmosphere, after removing radioiodine by adsorption and contaminated particles by filtration, will be continued. Contamination of the Hanford environment is within the "completely acceptable" Range I as defined by the Federal Radiation Council; ecological studies indicate that no living organism in the environment has been sufficiently contaminated by exposure to separations effluents to be of concern.

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3.2 Solid Wastes

The current practice of burying contaminated wastes will be continued. Failed Purex equipment that is too heavily contaminated for disposal in this way will continue to be stored on old railroad cars in a tunnel adjacent to the Purex Building.

3.3 Low- and Intermediate-Level Liquid Wastes

The current practice of percolating process condensates and cooling water through the ground to the ground water will be continued. The small quantities of contained hazardous isotopes are adsorbed by the soil and are thus stored in the ground. This practice has been successful in confining the radioisotopes within the reservation at a low cost. Current emphasis will be continued to minimize, within practical economic limits, the quantity of short- and long-lived isotopes discharged to the ground.

3.4 Non-Heating Liquid Wastes

A program is being initiated to concentrate the non-heating wastes (coating wastes, BiPO_4 wastes, and some aged Redox supernates) to salt cakes in the existing tanks. Laboratory studies and prototype operation indicate that the passage of heated air through the solution can remove enough water that, with controlled carbonation to convert free caustic to sodium carbonate, a non-hygroscopic salt cake can be formed. After the salt cake has been formed, only minimal surveillance will be required, and no reliance need be placed on the continuing integrity of the tank. This program, if successful, will obviate the need for constructing additional tanks for non-boiling wastes at least until 1980. If this program were not implemented, new tanks for storing non-heating wastes would be required by November, 1966. The first concentration unit is scheduled to be operational in the spring of 1964.

3.5 Self-Heating Liquid Wastes

Additional storage space is needed by August, 1964 to receive self-heating Purex wastes and thereby permit continued operation of the Purex Plant; funds for construction have been requested. A remotely-operated diverter station is to be installed to obtain improved contamination control while changing the routing of Purex wastes outside the process building.

3.6 Interim Fission Product Recovery

Facilities now in place will permit the Chemical Processing Department to meet the projected demand for fission products for the next few years. A load-out facility is now operational in C-Farm; this unit permits cesium to be sent to Oak Ridge in the shielded transfer tanks on request. In the first phase of the programmed reactivation of an old canyon

building (B-Plant), new operating facilities have been provided and are to be activated in March, 1963. These facilities will permit the accumulation of strontium and rare earth concentrates for aging. The Strontium Semiworks is being used to purify strontium until more permanent facilities are available. This facility is contact-maintained, however, and does not have the operating reliability of other production facilities in the Hanford complex. The head-end of Purex must continue to operate as an integral part of the interim fission product recovery complex; these facilities are also being used to demonstrate precipitation processes of potential use in future operations.

3.7 Immobilization of Self-Heating Wastes

Assuming the concurrence of the Commission, an integrated waste processing complex is to be installed in B-Plant. Phase II is to provide, by late FY 1965, a packaging cell to demonstrate processes for packaging separated fission products, either for long-term storage or for shipment. The unit is to become a part of the subsequent operating complex. Phase III is to provide, by late FY 1966, the facilities for removing strontium, cesium and some rare earths from Purex acid wastes, and the cesium and strontium from the alkaline wastes accumulated in the Purex and Redox tank farms. The separated cesium and strontium are to be packaged in small, high-integrity containers for long-term storage. The rare earths and residual salt wastes, after about five years storage as liquid for decay of the short-lived fission products, are to be combined with coating wastes for concentration to salt cakes in existing tanks. Additional packaging facilities for the separated fission products will be provided, if needed, and a vault is to be provided to contain the packaged fission products.

A facility is to be provided by late FY 1966 to remove sludge from the Purex tank farms for the removal of cesium and strontium. Processes are being developed to recover plutonium and neptunium from these sludges for recycle to the Purex Plant; implementation of this step will depend on the cost of the operation in relation to the value of the products recoverable. Similar facilities are to be provided for the treatment of Redox sludges by FY 1969. According to current plans, the operation of the integrated facility would result in all stored wastes being processed to an equilibrium processing condition by about FY 1973.

3.8 Future Fission Product Processing

It is proposed to recover technetium-99 from the aged liquid wastes while they are being processed for cesium removal. This recovery operation is believed to be justified by the unique characteristics of technetium and by the nominal cost of the recovery operation. Failure to recover the technetium at this time would place a national resource in a relatively inaccessible position from which it would be recoverable only at much greater cost. Technetium does not occur in nature, and the radiochemical wastes from the processing of irradiated fuels comprise its only source

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HW-76352

-8-

of supply. Technetium is unusually effective as a corrosion inhibitor, and it is a super-conductor at a higher temperature than is any other metal. It is estimated that the expenditure of about \$800,000 (including \$250,000 capital) will permit the recovery of about one ton of this rare material. The cesium, strontium, and rare earths isolated by the waste management operations would be available as crude concentrates for any desired recovery operations. Purification of cesium, strontium, or cerium might be effected within the B-Plant complex, either by using the Phase III equipment on a campaign basis or by using additional equipment provided for that purpose. Promethium purification would require the addition of an ion-exchange facility.

In October, 1962 the Division of Isotopes Development authorized⁽⁶⁾ an engineering study of the feasibility of constructing and operating an isotopes fabrication and packaging facility adjacent to the B-Plant facilities. The isotope plant would be designed to process the crude concentrates produced by the waste management activities. The operation of such a plant would obviate the need in B-Plant for any activities specifically in support of fission product recovery, except for the recovery of technetium as discussed above.

3.9 Estimated Costs

The capital cost of implementing the proposed Waste Management Program is estimated at \$20,750,000, to be authorized during the next five fiscal years. If current operating practices were continued and none of the existing storage tanks fail during this time, the requirements for capital funds to store the wastes would be about equal to that indicated above. The additional operating cost of the proposed program (excluding general plant overheads) is estimated at \$2,180,000 per year, starting in FY 1966 and continuing about six years. The justification for these operating costs lies in improved safety and the recovery of valuable isotopes.

3.10 Alternatives

The current treatment of gaseous, solid, and low- and intermediate-level liquid wastes is effective and relatively inexpensive. No alternative course of action has offered sufficient incentive to warrant intensive study.

The non-heating salt wastes might continue to be stored in underground tanks in their present form. However, not only will the proposed solidification of these wastes immobilize the radioactive materials, but the solidification program will cost less during the next five years than would the continuation of current practice.

Alternatives of the proposed fractionization and solidification of self-heating wastes include 1) continued storage as at present in underground tanks, 2) solidification in tanks after sufficient aging to reduce the rate of heat generation to an acceptable level, and 3) calcination to a thermally-stable form with long-term storage in high-integrity containers.

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HW-76352

-9-

The continued storage of these wastes in present form would retain the hundreds of megacuries of cesium in a mobile form and would therefore fail to meet the stated objective of minimizing radiochemical hazards. While these wastes could eventually be solidified in underground tanks as non-heating wastes, the Redox and Purex process wastes would have to be stored as liquid for fifty and one-hundred years, respectively, before in-tank solidification could safely be effected. The calcination operation would be at least as expensive as the fractionization operation, and would be much less compatible with the recovery of transuranic and fission product elements.

The proposed fractionization approach therefore offers 1) improved safety through the immobilization of the hundreds of megacuries of cesium and strontium, 2) decreased future costs for continued confinement and surveillance, 3) recovery of the technetium, plutonium, and neptunium, and 4) improved availability of selected fission products for beneficial use. In the opinion of the Chemical Processing Department, the proposed fractionization operation is fully justified by the waste management advantage alone, and becomes an economic benefit if the separated fission products prove to have a real value to other programs.

4. HISTORY AND CURRENT STATUS

During the eighteen years of Hanford operation, waste management practices have included the dispersal of gaseous wastes in the atmosphere, the burial of solid wastes, the discharge of low- and intermediate-level liquid wastes to the ground, and the accumulation and confinement of high-level liquid wastes in underground storage tanks.

Starting in 1957, an increasing interest in recovering selected fission products for beneficial use became evident, and limited production capability for cesium and strontium was established in 1961.

4.1 Dispersal of Gaseous Effluents

Contaminated gaseous effluents from the radiochemical plants are discharged to the atmosphere through 200-foot stacks after being scrubbed and filtered to reduce contamination to acceptable levels. Since krypton and xenon are easily reduced to acceptable concentrations by atmospheric dilution, they do not constitute a significant radiation hazard. The use of chemical adsorbers to limit the release of iodine-131 to the atmosphere was initiated in 1946, and the use of filters to contain contaminated particles was started in 1947. At the present time, gases vented from process vessels must pass through at least two filters before being discharged to the atmosphere. Continuing ecological studies are conducted by personnel of the Hanford Laboratories, both to determine the paths taken by contaminants released to the atmosphere and to establish safe limits for the rate of such release.

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4.1.1 Operating Plan

From 1955 to 1961, the release of iodine-131 from the separation plants to the atmosphere was limited to ten curies per week. This limit was based on ecological studies⁽⁷⁾ of the area which showed that, under these conditions, vegetation and animal life in the environs would remain at acceptably low levels of contamination. In September of 1961, the Federal Radiation Council issued a guide⁽⁸⁾ for the release of selected radioisotopes, including iodine-131. Hanford's limit of ten curies per week was not compatible with the "completely acceptable" Range I, but did fall well within Range II which called for "continuing study and surveillance". The "completely acceptable" emission rate of Range I would allow an annual average iodine-131 emission of only about two curies per week. Hanford personnel considered it prudent to operate the Hanford facilities within the limits of Range I, and the necessary changes were made to establish the indicated controls.

The principal sources of air-borne radioactivity in the Redox and Purex Plants are the gases evolved during fuel dissolution and during the subsequent concentration and transfer of process solutions. Iodine-131 is the radioisotope of primary concern. To minimize the emission of iodine-131, fuel elements are held in storage for 120 days or more before processing. This storage permits the 8-day iodine-131 to decay to low levels of activity. Silver reactors and acid absorption towers are used to trap the iodine liberated during the dissolving step, and mercuric nitrate is sometimes added to process solutions to minimize the volatility of radioiodine in subsequent processing operations. Fibrous glass filters, having efficiencies of 99.9 percent for material one micron in diameter, are used to filter particulate matter from the process vent systems before discharging the off-gas to the ventilation system. Ventilation air is exhausted through high efficiency filters, the Redox unit being constructed of graded layers of sand, while the Purex and Strontium Semiworks units are constructed of glass fibers.

The contaminants in gaseous effluents from the plutonium fabrication plant are essentially all particulate. The air leaving the process areas is filtered at least two, and usually three, times before it is combined with the ventilation air from the rest of the building. The ventilation air passes through a large bank of fire retardant glass and asbestos filters (the CWS absolute type) before it is discharged to the atmosphere.

4.1.2 Operating Experience

As indicated in Table I, the rate of dispersing radioisotopes in the atmosphere has been steadily decreasing. Environmental studies⁽⁹⁾ indicate that no living organisms are receiving clinically observable damage from these emissions.

4.2 Burial of Solid Wastes

Contaminated solid wastes are usually placed in special sealed boxes and are buried in designated areas. Two of the four active burial grounds receive

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HW-76352

-11-

failed process equipment and other solid wastes emitting high levels of gamma radiation, and two are restricted to the burial of wastes having low levels of gamma contamination.

The burial operations are conducted in about seventy acres of ground set aside for this purpose. Burial trenches in the industrial grounds are about 22 feet deep, and trenches in the low radiation or "dry waste" grounds are usually 14 feet deep. Boxes of low-level waste are dumped into the trench, while the wood or concrete boxes containing bulky or grossly contaminated materials are dragged from railroad cars into the trench by bulldozers using long cables. In either case, the boxes must be strong enough to contain the materials until the wastes are in place and the trenches are backfilled by the bulldozers. The weight of soil covering the waste crushes the wooden or cardboard boxes and compacts the waste, while the concrete boxes are strong enough to support the soil loads without crushing.

Most burial operations are uneventful, with precautions being taken to prevent personnel exposure, contamination of heavy equipment and environs, and premature crushing of the burial boxes. In several instances during past years, the burial boxes have been broken prematurely, and costly decontamination efforts were required. This risk is now minimized by using double encapsulation in wooden boxes or by using concrete boxes to transport heavily contaminated materials to the burial site.

The size and radiation level of some equipment at the Purex Plant makes it impractical to transport the failed equipment to the burial trenches, so a burial tunnel was built at Purex. Failed equipment is placed on an old railroad flat car, and the car is pushed into the tunnel for dead storage. Enough equipment has been stored in this manner to fill most of the initially provided tunnel space, and a second storage tunnel is now being constructed.

4.3 Ground Disposal of Low Level Liquids

The low- and intermediate-level liquid wastes are discharged directly to the ground under carefully controlled conditions. Low-level wastes, primarily cooling water, have such a low potential for contaminating the environment that they can be discharged directly to natural surface depressions or swamps. The intermediate-level wastes, which consist primarily of process condensates and low-level process solutions, are discharged to sub-surface cribs that have an appropriate percolation area. This "cribbing" technique prevents surface contamination and still allows ground disposal of the wastes.

The ground disposal technique relies on the ion-exchange properties of the sand, clay, and gravel soils above the water table. As the liquids percolate downward to the water table, the radioisotopes of concern are removed and retained as a physically- and/or chemically-fixed inventory in the soil.

4.3.1 Operating Plan

The discharge of low- and intermediate-level wastes to the ground is regulated such that radiological hazards are not created at any point contacted by

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living organisms in the environment. Improvement efforts are directed toward minimizing the quantities of radionuclides routed to the ground, within economically justifiable limits.

The use of cribs is controlled by limiting the permissible concentration of radioisotopes in the ground water beneath the crib. The appearance of cesium or strontium in the ground water beneath a crib in a concentration approaching one-tenth the maximum permissible concentration (MPC) for drinking water (occupational exposure) is cause for discontinuing the use of that crib. This conservative limit makes allowance for drainage from the soil after the disposal site is abandoned, and for the fact that the maximum concentration in the ground water may reach a somewhat higher level than that detected in the ground water sample. Somewhat higher concentrations of ruthenium are permitted, since significant decay takes place during the several years transit time to the river. Concentrations of tritium slightly greater than its MPC are permitted in the ground water, since dilution with the ground water and the Columbia River is certain to reduce tritium concentration well below its MPC before it is contacted by living organisms.

A limit of 5×10^{-5} microcuries of beta emitters per cubic centimeter has been established for the discharge of low-level wastes containing mixed fission products to surface swamps. This low concentration limit was adopted to minimize the potential for significantly contaminating wild fowl with radioactive fission products, and to prevent surface winds from spreading small quantities of radioisotopes over the surrounding terrain.

4.3.2 Operating Experience

From start-up of the plant through 1962, approximately eighty-billion gallons of water have been pumped to CPD. Essentially all of this water has been discharged to the ground. Approximately fifty-five billion gallons of the water sent to the ground have been as low-level waste, and about five billion gallons have been as intermediate-level waste; the remaining twenty billion gallons of water either was evaporated, is stored in the high-level waste, or was discharged to ground as a normal industrial waste.

Trends in ground disposal practices are reflected by Tables II and III which show the volume and activity of low- and intermediate-level wastes that have been discharged to the ground through 1962. Since most of the radioisotopes entering the ground are associated with the intermediate-level wastes, the origin, volume, and activity of intermediate-level waste streams generated in 1962 are presented in Table IV.

Low-level wastes (Table II) averaged about 1.7 billion gallons per year prior to the start-up of the Redox and TBP plants in 1953, and reached a peak of about 5.3 billion gallons in 1958. Subsequently, process improvements have decreased the rate of discharge of low-level waste to ground to about 4.7 billion gallons per year. Total activity in this low-level waste has only been about 0.3 percent of the total activity routed to the ground, with the major portion of the total being the result of accidental leaks of process materials into cooling water systems.

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-13-

The peak discharges of radionuclides in the intermediate-level wastes (see Table III) resulted from a waste scavenging program that was initiated to free storage space urgently needed to meet production requirements. In this program, which ran from 1954 through 1957, aged liquid wastes from the bismuth phosphate process were scavenged to reduce the concentration of long-lived radionuclides in solution and were then discharged to the ground. Late in 1958, the practice of re-distilling the Redox head-end condensate was initiated and resulted in a sharp decrease in the quantity of beta emitters (principally ruthenium) that were discharged to the ground. In 1959, a surface condenser was substituted for the original contact condenser in the Purex "A" waste farm; this change permitted most of the process condensate to be returned to the waste tanks, thereby reducing the quantity of radioactivity (principally cesium) discharged to the ground. Most of the strontium discharged to the ground in 1961 was a direct result of the strontium-90 recovery efforts in the 325 Building "Hot Cells" and the Strontium Semiworks. During the past year, a plant test resulted in a relatively large release of ruthenium to the process condensate. This test was conducted to determine if the condensate from the Purex waste concentration step could be stripped of nitric acid and discarded without a second distillation. An improved demister had been designed to minimize contamination carried by entrainment, and the elimination of the second distillation offered substantial economies in reduced steam consumption and equipment replacement. While the test was successful in controlling entrainment to acceptable levels, enough ruthenium was volatilized to cause a substantial increase in activity in the recovered acid and in the condensate discarded. Tests are continuing to find a way to control the ruthenium volatilization to acceptable levels.

Most of the plutonium sent to ground was carried in the Recuplex extraction wastes. Since the Recuplex facility was shut down in April, 1962, the losses of plutonium to the soil have been substantially decreased. Project CGC-912, which will provide facilities for substantially reducing the plutonium content of all 234-5 Building wastes, is expected to be in operation during 1963.

Although waste streams continue to carry fission products into the ground, the inventory of fission products stored in the ground is steadily diminishing; this condition is realized because the disappearance by radioactive decay is greater than the rate of fission product input. The inventory of total beta activity is now less than 10 percent of the gross input, while the strontium-90 and cesium-137 inventories are about 86 percent of their gross input. At the present time, the strontium-90 and cesium-137 inventories are decreasing at a rate of about 500 and 1000 curies per year, respectively. Plutonium, because of its long half-life, has been accumulating in the ground with essentially no loss by decay. This plutonium accumulation is concentrated primarily in the top few feet of soil on the crib bottom.

The best indication of the effectiveness of ground disposal controls is the status of ground water contamination. While the detectable contamination is composed principally of 1.0-year ruthenium-106, longer-lived isotopes such as 5.3-year cobalt-60, 12.2-year tritium, 28-year strontium-90, and 30-year

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HW-76352

-14-

cesium-137 have been detected in the ground water. A summary of the long-lived isotopes detected in the ground water is presented in Table V. Ruthenium-106 has migrated about seven miles from the 200 East Area and can be detected about five miles from the Columbia River. This is an expected condition, since ruthenium is retained poorly by the soil, and the spread has occurred in aquifer areas of known high permeability. There is little concern about this ground water contamination, however, since ruthenium-106 has a short half-life, and its concentration at the migration front is less than one percent of its MPC. Current data show that the concentration in wells behind the migration front is decreasing, indicating that the rate of accumulation is less than the rate of radioactive decay. This trend may be reversed if the ruthenium content of the Purex condensates is not quickly reduced and controlled. Low concentrations of cobalt-60 could be carried into this same zone of high permeability since cobalt-60 has been found in the northern section of the 200 East Area. Samples from wells in this area have all shown the cobalt-60 concentration to be less than its MPC_w; since the principal source of cobalt-60 was the scavenged bismuth phosphate wastes sent to ground about six years ago, it is expected that the cobalt-60 concentration will continue to diminish steadily. The presence of fission-produced tritium in the ground water was recently confirmed, and a study was made of its migration pattern in the ground water⁽¹⁰⁾. While some tritium has probably reached the Columbia River, concentrations at that point are so low it cannot be detected by routine analytical techniques. Cesium-137, which was originally detected at two sampling sites in the 200 West Area, has not been detected recently in the ground water.

The natural hydraulic gradients underlying the processing area have been altered by the artificial water mounds created by the large quantities of water discharged to the ground. Two major ground water mounds now exist, one under 200 East Area and one under 200 West Area, both coinciding with the cooling water disposal sites. At these points, the ground water elevation has been increased about 25 and 75 feet above normal, respectively. This mounding decreases the depth of the soil above ground water, slightly reduces the quantity of soil available for the adsorption of radionuclides, and increases the hydraulic gradient for ground water flow from the disposal sites to the Columbia River; up to this time, no deleterious effects have been observed.

4.3.3 Monitoring Activities

As an integral part of the over-all ground disposal program, 600 monitoring and test wells have been drilled in the Hanford area to obtain the most effective data possible on the hydrology and geology of the area, and to provide ground water and soil samples for projecting the environmental effects of ground disposal. Well drilling is expected to continue on the reservation at a rate of about twenty percent of the historical rate, or about three wells per year. The cost of field sampling and analytical services is projected to continue at about \$40,000 per year, while the cost of well drilling is expected to decrease from about \$70,000 to about \$15,000 per year.

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4.4 Tank Storage of High Level Liquids

To the present time, in-tank storage has been considered the most practical method of high-level waste containment, although its temporary nature has been recognized from the start. The storage requirements for the large volumes of waste produced over the years have become an important economic factor in plant operation.

In general, the volumes of process wastes have been determined by the quantity and solubility of the various chemicals used. The highly efficient solvent extraction processes, which use smaller quantities of residual chemicals, have produced smaller volumes of process wastes than did the former bismuth phosphate process; the resulting higher concentration of radionuclides has resulted in these wastes generating heat at a faster rate than could be dissipated through the surrounding soil. Heat is removed from these tanks by permitting the waste to boil and condensing the vapors. This procedure has rather naturally divided the high-level waste into two categories, viz., 1) the self-boiling wastes, or those that generate more than about 0.1 watt of fission product heat per gallon of waste, and 2) the non-boiling wastes. All of these high-level wastes are currently stored as alkaline slurries in reinforced concrete tanks constructed with carbon steel liners.

4.4.1 Non-Boiling Waste Storage

The non-boiling wastes currently in storage consist of coating wastes, aged solvent extraction supernates from the Redox process, and various types of bismuth phosphate process wastes. Only coating wastes, wastes from the Strontium Semiworks operation, and some equipment decontamination wastes are now being added to this inventory routinely. Redox process wastes are initially self-boiling, but, after decaying a few years in storage, the supernatant solutions gradually cool below the boiling point and can then be classes as non-boiling wastes. The supernatant liquids in these aged Redox wastes are periodically transferred into storage tanks designed for non-boiling wastes.

4.4.1.1 Tank Farm Status (end of CY 1962)

A total of about fifty-two million gallons of sludge and non-boiling solution has been accumulated in the storage tanks. One-hundred eight underground tanks, each having a capacity of 500,000 or 750,000 gallons, are used for storing this non-boiling waste. At present, there is reserve space of approximately 7,400,000 gallons in East Area (excluding 1,600,000 gallons of space reserved for the fission product recovery wastes) and about 4,500,000 gallons in West Area (excluding 3,700,000 gallons of space reserved for aged Redox supernates). Assuming that forecasted production rates are as currently forecasted and that none of the storage tanks fail, non-boiling wastes can be accommodated in existing storage systems until November, 1966. The total residual space may be considered on an integrated basis, since there are cross-country transfer lines between the areas; without such transfer, the residual space would be adequate in East Area until January, 1966, and in West Area until August, 1968.

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HW-76352

-16-

4.4.1.2 Life Expectancy

The potential of tank failure must be considered when planning the continued use of these underground tanks since they are now between 11 and 19 years old. During the past six years, four of these tanks have started to leak; leakage was detected by observing a declining liquid-level in the tank and verified by detecting radioactivity in the adjacent soil. In each case, the tank contents were immediately transferred to sound tanks, and no contamination of the ground water, the ground surface, or the atmosphere occurred. Two of the tanks were about eight years old, one about twelve and one sixteen years old at the time of failure. Each tank had been subjected to repeated cycles of emptying and filling. During most of their history, they had been used to store wastes from the bismuth phosphate process which, according to laboratory studies, have a higher corrosion potential than have the Redox and Purex wastes now being generated. Additional failures in these tanks can be expected in the future. Each tank failure will reduce the remaining storage capacity by an effective period of two to three months, depending on the volume of the failed tank.

4.4.2 Self-Boiling Waste Storage

Self-boiling wastes are principally composed of Purex and Redox solvent extraction raffinates. These wastes contain over 99 percent of the non-gaseous fission products processed and are stored in the 241-SX (Redox) and 241-A (Purex) tank farms. These tanks are specially designed to contain the self-boiling waste. Headers connect the tanks with water-cooled condenser systems. The condensates are either returned to the storage tank to maintain proper dilution of the waste or they are sent to cribs if the tank contents are to be concentrated.

Prime factors in the definition of safe storage conditions for self-boiling wastes include the chemical composition of the waste solutions, the quantity and nature of any solids present, the specific rate of heat generation, and the vessel's structural strength and tolerance for thermal gradients. These variables cannot be adequately evaluated in a separate experimental facility because there is a recognized influence of storage time and radiation exposure on the characteristics of the waste and the container. Experimental studies of tank and waste characteristics are therefore combined with past experience in waste tank operation in the establishment of safe operating procedures for each storage area.

4.4.2.1 Tank Farm Status (end of CY 1962)

The 241-SX (Redox) tank farm consists of fifteen 1-million-gallon tanks that were built in 1954. Of these fifteen tanks, one is used as a de-entrainment pot to minimize the contamination of the condensate routed to ground, and one failed when its liner buckled (see 4.4.2.2. below). This tank farm has received a total of about 18,600,000 gallons of neutralized Redox solvent extraction raffinates and head-end treatment wastes. About 2,225,000 gallons of this waste has now aged sufficiently to permit its transfer to

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HW-76352

-17-

non-boiling storage; of this total, 984,000 gallons has actually been transferred. Self-concentration has reduced the remaining wastes to a current inventory of about 8,080,000 gallons. Under established 241-SX specifications, this tank farm can handle the Redox wastes from the forecast production through September, 1965, assuming no further tank failures or removal of non-boiling wastes. It is planned to continue to remove cooled wastes and to re-use the tanks. Assuming that tank failures do not occur, the use of the 241-SX tank farm can thus be extended to 1970 and beyond without significant expenditures.

The 241-A (Purex) tank farm consists of six 1-million-gallon tanks that were constructed in 1955. This tank farm has received a total of 28,450,000 gallons of neutralized Purex solvent extraction raffinates, aqueous washes from solvent regeneration, and miscellaneous cell and equipment flushes from the Purex plant. Self-concentration has reduced this volume to 4,131,000 gallons. Based on established operating limits and forecast production rates, additional storage capacity for Purex self-boiling wastes will be required no later than August, 1964.

The supernatant solution in tank 103-A was transferred to non-boiling storage in the 241-C Farm during January of 1963. This waste was produced during the initial operation of the Purex plant and is the only solution in the A-Farm sufficiently dilute in fission products to permit its transfer to non-boiling storage. While this move has created some storage reserve, there is presently no plan to use the space for the storage of fresh neutralized waste because of the likelihood of temperature excursions in the sludge layer if such a plan were followed. Unless steps can be taken to avoid this hazard, such as by sludge removal, the 103-A tank is considered to be available for the storage of self-boiling wastes on an emergency basis only. This space will be held available to receive the supernates from one of the other A-Farm tanks if a leak should develop. The supernates transferred to C-Farm will be held as a source of cesium to meet the market demand in the near future.

Following the removal of the supernate from the 103-A tank, approximately 300,000 gallons of water was added to the sludge; no appreciable dissolution occurred, even though the water was recirculated at boiling temperatures for several weeks. It is theorized that solubilization is inhibited by the presence of a hard water-resistant layer on the sludge surface.

4.4.2.2. Life Expectancy

During the ten years of operational experience in storing self-boiling wastes, one tank has failed from abnormal cause. In 1958, the bottom liner of tank 113-SX buckled shortly after self-heating Redox waste was first added to the tank. It is theorized that the heat from the waste vaporized water from the concrete grout between the steel liner and the concrete base of the tank, generating sufficient pressure to lift the steel liner and the small amount of waste in the vessel at the time. The wastes were transferred to another vessel, and operating procedures were revised to prevent the liner from buckling in other tanks. While initial evidence indicated that no leakage

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HW-76352

-18-

occurred, recent tests have shown that the vessel would leak if filled with liquid. Recent measurements have also revealed two small areas of slight radioactivity in the soil near 108-SX tank. The apparent contamination is not spreading or increasing in intensity. Similar areas of low-level activity have been found under the Purex A-Farm tanks.

Little basis exists to predict the service life of the tanks. While the chemical characteristics of the Redox and Purex wastes are such that pitting and general corrosion attack on the mild steel liners should be almost negligible, other factors suggest that a relatively short service life can be expected. Recent operational experience at Savannah River⁽¹¹⁾ indicates that stress corrosion can be expected in the alkaline nitrate systems; this phenomenon has been duplicated in experimental studies both at Savannah River and at Hanford. On the other hand, no leakage at Hanford can be attributed to this mechanism after ten plus years of storing alkaline nitrate wastes, and the seepage observed at Savannah River has been largely self-sealing. Test borings have shown that the concrete domes of the Purex tanks at Hanford are deteriorating from continued exposure to radiation and condensing steam; this deterioration is slow, however, and the concrete domes are predicted to retain their integrity for many years. In a few instances of abnormal operation, the temperature of the Purex sludge (in 241-A) has risen as high as 400 F for a few days or weeks. While short-term exposures of the tank to these conditions are not predicted to cause rapid deterioration, the actual effects of these isolated temperature excursions cannot be determined. While these observed conditions are not cause for immediate alarm, they do cast doubt on any estimate of extended tank life and underline the need for an improved design for tanks to be provided in the future. In the opinion of Hanford engineers, the life of these tanks is probably no more than twenty years, and may be as little as ten to fifteen years.

4.4.2.3 Operational Safety

Several improvements in operational safety have been realized in the storage of self-boiling wastes. In 1955, air-lift circulators were installed in nine of the fifteen 241-SX tanks to augment natural convection in maintaining the stored liquid at a uniform temperature. Prior to the use of circulators, heat accumulated in the liquid at the lower levels by heating the liquid to a higher temperature than its atmospheric boiling point. This stored heat was released periodically by circulation of the superheated liquid, causing a rapid release of steam and a slight pressurization of the tank. Circulators are now installed on all tanks containing boiling wastes, and have been effective in eliminating the pressure surges. These circulators, together with concentration control of the waste solutions, have also been very effective in controlling the temperature of the sludge that has accumulated on the bottom of the tanks. Both the 241-SX and 241-A tank farms have pressure relief devices that are designed to limit the amount of pressurization that might occur in these tanks.

Emergency water systems have been installed in each area to provide water to the condensers in case there should be a failure of the normal water

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supply system. The 241-SX tank farm has two water tanks having a total capacity of 300,000 gallons. This water supply should last two or three days and can be replenished by tank truck. The 241-A tank farm, which has a higher boil-off rate than does the 241-SX tank farm, is protected against both water and electrical failures by a diesel-driven generator, a deep-well pumping system, and a cooling-water tower. A 750,000-gallon reserve tank allows for temporary interruptions of the system. Leakage from the boiling tanks could be detected by monitoring grids of vertical and lateral channels around and under each storage tank in the two farms. These systems, which were installed during the past year, are monitored regularly to verify the integrity of the storage vessels.

4.5 Fission Product Recovery

In 1955, research was initiated on a limited scale to explore methods of recovering specific fission products from radiochemical wastes. Early results were promising, and the R&D program was expanded to include the development of packaging techniques. In 1959, the Atomic Energy Commission requested Hanford to develop the technology and to provide facilities for the isolation and accumulation of strontium, cesium, and rare earths⁽¹²⁾. This general request was supplemented in 1960 by specific orders for kilocurie quantities of cesium-137⁽¹³⁾ and for megacurie quantities of strontium-90⁽¹⁴⁾. In response to these requests, existing facilities were modified as required and were either activated or diverted to this use. A considerable portion of the separations research and development was diverted to this program during FY 1960 and 1961; in October, 1961, the responsibility for filling specific orders for fission products was assigned to the Purex Operation, and the emphasis of development programs was returned to the processing of waste for long-term storage. The history and status of the various facilities used in developing capability for fission product recovery are discussed below.

4.5.1 Purex Head-End

The Purex feed clarification system (centrifugation) was freed from normal production use late in FY 1960, and plant-scale tests were initiated to demonstrate fission product concentration by precipitation. Early in FY 1961, a load-out station was installed adjacent to the plant to permit fission product concentrates to be loaded out from Purex. Demonstration efforts were originally focused on the recovery of rare earth crudes, but these efforts were re-directed early in FY 1961 when the Commission officially requested Hanford to provide about 800,000 curies of purified strontium-90 for the SNAP-7 Program. Since that time, the primary function of the Purex head-end has been to isolate strontium as a crude concentrate for purification. The facilities are also used to clarify aged waste supernatant solutions preparatory to loading cesium onto Decalso in the shielded transfer tanks for shipment to Oak Ridge. Plant tests are also being made to demonstrate laboratory-developed precipitation flowsheets for the isolation of other products, such as cesium, cerium, or rare-earth concentrates appropriate for the isolation of promethium.

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HW-76352

-20-

4.5.2 325 Hot Cells

The High-Level Radiochemistry Facility (Hot Cells) was originally programmed to demonstrate the recovery and purification of selected fission products during FY 1961; during FY 1962, waste processing methods were to be demonstrated. When the SNAP-7 program required an immediate supply of purified strontium by the third quarter of FY 1961, the function of the Hot Cells was diverted to fill this need. Ion-exchange equipment already in place was used to purify strontium from crudes supplied by Purex; about 60,000 curies of purified strontium-90 recovered were shipped to Oak Ridge. When larger-scale equipment became available late in FY 1961, the Hot Cells were returned to the original program. Initial demonstration of fission product recovery flowsheets was completed in FY 1962. In addition to the strontium demonstration, kilocurie quantities of cerium-144, and promethium-147 were isolated, together with about ten grams of technetium-99. The "A" cell is now used for waste calcination studies, while process development for the removal of selected isotopes is continuing in the "B" cell. Experiments are also in progress to demonstrate the long-term stability of packaged fission products under conditions simulating those being proposed for long-term storage.

4.5.3 Strontium Semiworks - CR Vault

The Strontium Semiworks - CR Vault complex was activated late in FY 1961 to purify strontium-90 needed for the SNAP-7 program. The CR Vault was originally used in the recovery of uranium from underground storage (wastes from the BiPO_4 process) and had been inactive since 1952. The CR Vault is now used to accumulate the crude strontium concentrate from the Purex head-end, and to permit the decay of strontium-89 before purification in the Strontium Semiworks. The Strontium Semiworks was originally used to pilot the Redox and Purex processes at full levels of radioactivity; the facility had been inactive since 1957. The Strontium Semiworks has been revised to permit the purification of strontium by solvent extraction, using di(2-ethylhexyl) phosphoric acid (D2EHPA) and tri-butyl phosphate (TBP) in a hydrocarbon diluent. Over 3.3 megacuries of strontium-90 have been purified, of which 1.16 megacuries have been shipped to Oak Ridge and Quehanna, and the balance is being held in inventory. An additional 1.0 megacurie of crude strontium is aging in the 244-CR Vault.

With the completion of demonstration runs early in FY 1962, the responsibility for operating the Strontium Semiworks was transferred from Hanford Laboratories (R&D personnel) to the Chemical Processing Department (operating personnel). Additional storage facilities for purified strontium have been installed (Project CGC-948), and the operating complex has been prepared for semi-routine operation as required to meet anticipated Commission needs for strontium in the near future. This facility is contact maintained, however, and does not have the operating reliability of the other production facilities in the Hanford complex.

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HW-76352

-21-

4.5.4 Shipping Casks

During FY 1960, work was initiated to develop casks that would permit the safe shipment of fission products from Hanford facilities to off-site areas. The first cask design (HAPO-I) was developed for rare earths or cerium-144 products. These products were to be shipped as dried filter cakes, readily dissolvable by the receiver. Before the HAPO-I cask was tested, it was modified (to HAPO-I-A) for the shipment of the strontium-90 purified in the Hot Cells. This modification deleted the filter unit and substituted a container of commercially available inorganic ion-exchange material (Decalso) designed for the retention of up to 80,000 curies of strontium-90 during shipment. A second cask (HAPO-II) was designed specifically for 170,000-curie shipments of strontium from the Hot Semiworks. This cask was designed to permit the loading of strontium carbonate by filtering the precipitate from a slurry routed through the filter casks. The HAPO-I cask has been further modified (HAPO-I-B). It is similar to the HAPO-II cask except it has about twice the shipping capacity and has adequate shielding for cerium shipments as well. All of these casks are shipped in buffer units specially designed to protect the cask from impact and fire.

Shielded transfer tanks (STT) of ORNL design have been used to provide crude cesium-137 to ORNL since FY 1961. Each tank uses Decalso to adsorb and contain about 50,000 curies of cesium, and two such tanks are contained in a railroad car modified to provide some impact protection to the tanks.

While the shipping methods in use have met the production requirements of the Commission without incident, it is believed that safer methods of transporting these materials should be developed if inter-site transfers are to continue indefinitely or to increase in volume. An improved shipping cask, the HAPO-IV, is being designed for the transport of cesium adsorbed on a zeolite.

4.5.5 Cesium Load-Out Facility

A load-out facility has been constructed (Project CGC-950) at the C-Tank Farm to permit cesium to be loaded onto Decalso in the shielded shipping tanks without having to use the Purex head-end. This facility will relieve the interference with the primary assignment of the Purex head-end - that of strontium recovery and in-plant testing of flowsheets under development. The start-up of this facility is in progress.

5. PLANNED ACTIVITIES

The short-term plans for waste management are directed toward the continuation and improvement of current waste disposal methods to assure that plant operations continue without delay. Long-range planning is directed toward waste storage and disposal methods that will achieve maximum long-range containment and radiological safety at minimum cost. The major goals of the long-range program are:

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- 1) to control the radioactivity released to the atmosphere to a rate consistent with federal regulations and at which no clinically observable effects are observed in life in the environs,
- 2) to minimize the amount of hazardous radioisotopes stored in the ground,
- 3) to immobilize non-heating wastes by in-tank solidification,
- 4) to extract and package long-lived heat-producing radioisotopes for safe long-term storage,
- 5) to recover fission products for beneficial use to the extent desired by the AEC, and
- 6) to minimize the time and cost required to achieve acceptable performance in the goals noted above.

An intensive research and development program, item 02-53-04-04⁽¹⁵⁾, is in progress to define the technologies required to reach these goals. A tentative schedule of the design and construction activities required to implement the program is presented in Table VI.

5.1 Gaseous Wastes

Development efforts are continuing to define the technology required to reduce the quantities of radionuclides dispersed in the atmosphere. Areas of specific interest include the absorption of iodine-131 and the filtration of contaminated particles. The confinement capability of all separations is now receiving an intensive review, and projects will be recommended to improve areas found to be inadequate.

5.2 Solid Wastes

The current practice of burying solid wastes in trenches and tunnels will be continued. The current techniques are believed to be adequate to meet the needs of the foreseeable future.

5.3 Low Level Liquid Waste

The history and current status of low- and intermediate-level liquid waste disposal practices at Hanford attest to the safety and practicality of current ground disposal methods. In the future, it is planned to continue current methods, and to improve these methods as justified by new data, revised criteria, or economics.

5.3.1 Replacement Disposal Sites

An annual expenditure of about \$250,000 is scheduled for routine crib replacements. Significant reductions in the radioisotope content of cribbed

streams might reduce the costs of crib replacement, but budgeted funds cannot be reduced until these advances in technology have been firmly established.

5.3.2 New Treatments

5.3.2.1 Organic Wastes

The organic wastes from Purex and Strontium Semiworks operations are least amenable to ground disposal because they are poorly retained by the soil and have poor ion-exchange characteristics. Improved process performance has significantly reduced the volumes of these materials however, and no need for an improved disposal method is now foreseen.

5.3.2.2 Plutonium-Bearing Wastes

Completion of a new waste treatment facility (Project CGC-912) in 1963 will permit a significant reduction in the amount of plutonium being discharged to the ground by the 234-5 Building. Future technology may provide a justification for further treatment of these wastes to remove plutonium and other transuranium alpha emitters, but no major project expenditures are currently envisioned.

5.3.2.3 Fission Product-Bearing Wastes

Strontium-90 and cesium-137 are presently of concern in the ground disposal of waste streams (see Tables III and IV), and attempts are now being made to minimize the amount of these isotopes discharged to the cribs. High-integrity titanium tube-bundles have been installed in the Purex waste concentrators; by reducing the failure rate of the tube-bundles, the amount of fission products accidentally escaping from the process through the steam condensate will be minimized. Strontium-90 losses from the Strontium Semiworks operation are carefully controlled, with most waste being concentrated and sent to underground storage tanks. Only the organic wastes and condensates go to cribs. A limit of 1×10^{-3} curies/liter has been established as the maximum amount of strontium-90 permitted to be discharged in the condensate. Contaminated solvent, which is occasionally discarded, is reduced to a total of 5 to 10 curies per batch before it is discarded.

Further reductions in the radioactivity of cribbed streams may require special treatment such as ion-exchange or evaporation methods, but the use of these techniques cannot be justified at present. Adverse environmental conditions could increase the need for these techniques, however, so a continuing effort will be made to obtain both a full characterization of the individual waste streams and the technological and engineering information necessary for applying special treatment methods. On the basis of present data and technology, it is predicted that capital expenditures for the special treatment of cribbed wastes will not be needed for several years.

5.4 In-Tank Solidification of Non-Heating Wastes

In-tank solidification of essentially non-heating wastes is proposed as a low-cost form of long-term containment, while simultaneously increasing the

effective capacity of the old tanks. Solidification can be safely effected only with "non-heating" wastes. If the concentrated salt waste solutions are to cool sufficiently (about 140 F) to solidify, the rate of heat generation in the tank cannot exceed the capability of the system to dissipate the heat at that temperature. This limiting rate of heat generation is currently estimated at about 0.1 watt per cubic foot of concentrate; studies are continuing to verify this value. If the solidified wastes did not cool, the salt cakes would slowly evaporate the contained water and, as the thermal conductivity decreased, might self-heat to calcination temperatures.

5.4.1 Operating Plan

In this concept, water is to be evaporated from the stored waste so that it will solidify in the underground tank upon cooling. Evaporation is to be effected by the passage of heated air through the waste. See Figure 1. Air, pumped through a 1500 kw electric heater at about five thousand cubic feet per minute, is to be heated to about 800 C. The passage of heated air was chosen as the concentration method because most of the required equipment can be located above ground; the major equipment would not be grossly contaminated while in use and could be readily moved from tank to tank. Electricity was chosen as the source of power because hydroelectric power is relatively inexpensive in the Hanford area (\$17.50 per kilowatt-year), and because the waste evaporation unit can then be designed to be fail-safe so that constant operation attendance would not be required. The heated air is to be introduced well below the surface of the liquid within a shroud tube. Introducing the heated air in this way accomplishes three things. First, the heated air is brought into intimate contact with the liquid, thereby assuring the efficient use of the heat. Second, the air rising in the shroud tube creates liquid circulation rates up to thousands of gallons per minute, thereby promoting uniform concentration of the tanks' contents. Third, some heat is transferred from the incoming air to the rising liquid through the finned air-introduction tube, thereby reducing the air temperature at the time of initial air - liquid contact. Experimental work has shown that, in units of this type, entrainment in the effluent gases is minimized by keeping the temperature of the air relatively low (say, 400 - 500 C) at the time of initial air - liquid contact. The effluent gases are to be passed through de-entraining devices, a condenser and a fibrous glass filter. The filtered air is to be released to the atmosphere; if contamination of the environs becomes a problem, the air could be recycled. The condensate is to be cribbed as a high-level waste.

Laboratory studies and small-scale pilot experiments (up to 5000-gallon size) indicate that water can be removed from the non-heating wastes in this way to such an extent that, upon cooling, the residual salts will crystallize to a massive salt cake. During concentration, the temperature of the liquid waste will rise from about 70 C to the range of 95 to 100 C. Some carbon dioxide may be added to neutralize excess sodium hydroxide, thereby reducing the affinity of the salts for water. After the external source of heat is removed, the partially crystallized mass will cool at a rate determined by the rate of heat loss to the environment, and by the rate of heat generation

from the decay of contained fission products. It is predicted that, under operating conditions, a thick layer of salt cake will form on all sides within a few months, but that the center may remain liquid for a few years. If needed, techniques are available to solidify any residual liquid by the addition of calcium sulfate or sodium bicarbonate to the liquid in the tank.

After the wastes are solidified in the tanks, only minimal surveillance of the area will be required to assure the continued confinement of the radioisotopes; no reliance need be placed on the continuing integrity of the concrete tank. Rainfall in this area averages about seven inches per year; the water is re-evaporated to the atmosphere, with essentially none of it percolating downward to the water table. This condition has existed for at least ten thousand years, and is not likely to change in the thousand years required for the long-lived fission products to decay to innocuous levels. Plutonium would, of course, be present in significant quantities for many tens of thousands of years beyond that time. Because of the ion-exchange properties of the soil, however, the ground would serve as a very efficient back-up confinement mechanism for plutonium; in the many cribs receiving plutonium-bearing wastes, all tests have shown that plutonium migrates no more than a few feet. While it is conceivable that an earthquake could fragment the tank and salt cake, there need be no concern provided the salt cake were not exposed to the wind. In this very unlikely event, the exposure of living organisms to the salt cake would be minimal in the few days or weeks required to recover the cake. Similarly, an earthquake might cause landslides which could dam the Columbia River and force it to find a new course at a higher elevation. Geologists agree that this mechanism might elevate the river level a few tens of feet, but estimate the probability of its being elevated a hundred feet as minimal. The salt cakes would be located at least 150 feet above the water table. It is concluded that no natural phenomenon characteristic of the area would transport the radioactive material to a place where it could be contacted by living organisms.

5.4.2 Facilities Plan

Although existing storage space for non-boiling wastes may be adequate until FY 1967, the uncertainty of tank life requires the early achievement of the full-scale potential of the in-tank solidification program. The scope design of a five-million Btu/hr system has been completed, and the detailed design is in progress. A project proposal is being submitted for its installation. A similar concentration unit is to be installed in the 200 West Area, with beneficial use being expected by August, 1965. The capital cost of each unit is estimated at about \$500,000. Annual operating costs (excluding general plant overheads) are estimated at \$240,000 for the two units, including annual costs of \$100,000 for the transfer of the concentration units from tank to tank. Assuming that an over-all volume reduction of three and that an average boil-off rate of 7,200,000 gallons per year from the two units is realized, the solidification of non-heating wastes will be current with waste production about the end of FY 1973.

5.5 Tank Storage of Self-Heating Wastes

Solution storage of self-heating wastes will continue until the waste fractionation program can be activated, and as necessary to implement the program thereafter. In the immediate future, current practices must be extended and improved to assure production continuity, and to increase the efficiency and safety of existing systems. The primary concern is for the Purex self-boiling wastes, since they contain over half of the total fission product inventory and will contain over 75 percent of those processed in the future.

5.5.1 Purex Waste Processing

On the basis of past operating practices, sufficient storage capacity is available to store the self-boiling wastes from the forecast Purex production until April, 1964. Since the completion of the proposed new tank farm may be delayed beyond that date, Purex operating practices have been revised to increase the equivalent storage capacity of the two available tanks by about eight percent. The changes include the denitration of Purex wastes by reaction with formaldehyde or sugar, and an improved solvent washing procedure that features a reduced salt content in the aqueous waste to be stored. Additional self-boiling wastes cannot be added to the four filled tanks, since operating experience has shown that adding fresh Purex self-boiling wastes to a tank once-filled with self-boiling wastes results in undesirable temperature excursions in the sludge layer. However, the space remaining in the "once-filled" tanks will be effectively used to store the aqueous wastes from the solvent washing steps. Assuming no tank failures, the revised operating procedures will permit the storage of Purex wastes in A-Farm until August, 1964.

5.5.2 Additional Storage Tanks for Self-Boiling Wastes

The expenditure of \$2,700,000 has been budgeted to construct a 241-AX farm, to be comprised of four tanks similar to those currently in use. Additional tanks may be required at a future date if the re-use of the 241-A tanks is shown to be impractical. On the basis of this possibility, an additional \$2,300,000 is included in the FY 1966 budget forecast for constructing four additional tanks.

5.5.3 Purex Essential Waste Routing System

Operating experience has shown that the continued use of the existing diversion box system for changing the routing of Purex process wastes has become increasingly hazardous and costly. To avoid jeopardizing production capability, an improved method of re-routing Purex wastes is required. To fill this need, it is planned to provide an improved waste routing system for the Purex process wastes, and to limit the function of the existing 241-A-151 and 241-A-152 diversion boxes to routing non-boiling wastes. The new waste routing system will include four new waste transfer lines from the Purex 202-A Building to a central diverter station, from which waste can be diverted via additional new lines to the 241-A and 241-AX tank farms and to the 241-CR Vault system. In addition, provisions will be included for a future connection with lines to the B-Plant complex. The central diverter station (241-AX-151) will

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HW-76352

-27-

feature a diversion system which is a modification of the British gravity-flow system. By using remotely-positioned inlet and discharge heads within the diverter station, routing changes can be made without opening the station. This system is estimated to cost about \$750,000 and will be installed as soon as project approval is obtained. A diverter station (241-AX-152) is also included in the design of the new 241-AX tank farm.

It is not currently planned to replace other diversion boxes with diverters of this type, although a need may develop at the Redox Plant as the Redox waste system becomes older and as newer type wastes (NPR, PRTR) are processed.

5.5.4 OWW Segregation and Concentration

Purex organic wash wastes (OWW) are currently stored with the self-boiling process wastes to be concentrated by the heat of radioactive decay. In the past, the OWW has contributed about 20 to 25 percent of the sodium salts and approximately 30 percent of the sludge in the combined waste. While recent process changes have reduced the amount of sodium salts in the OWW about 30 percent, the storage requirements are still appreciable.

In 1960, the segregation of OWW from the neutralized Purex waste was proposed on the basis of cost savings to be realized by reducing the quantity of salt to be processed during subsequent waste calcination operations. The current waste management plan does not include waste calcination, and thus eliminates this major element of justification. The primary economic incentive for OWW segregation is now based on possible savings to be realized by increasing the effective capacity of the boiling-waste tank farms. As explained above, early operating practices resulted in tanks which are full with respect to the storage of self-heating wastes, but not with respect to OWW storage. Under projected operating conditions, the capacity of the 241-A tanks for OWW should not be exceeded before the 241-AX farm is available. The continued inclusion of OWW with the self-boiling wastes would consume some of the effective capacity of the 241-AX farm, and it may become desirable to concentrate the OWW for storage in other tanks.

Alternative plans for concentrating the OWW are therefore being evaluated. The engineering feasibility of using the existing 242-T evaporator facility for this purpose has been studied. The successful demonstration of the in-tank waste solidification prototype, however, may offer the possibility of a more economical OWW concentration process. A third possible OWW concentration method involves the use of a large Purex concentrator (the E-F11 concentrator) which was recently removed from service as a result of Purex process changes. Possible advantages of this latter method are reduced capital cost requirements and its ready availability for use on an emergency basis. The concentration of OWW in conventional evaporation equipment may be complicated by the presence of MnO_2 solids in the waste; these solids may foul the heat exchange surfaces and promote frothing. Assuming the separate concentration of OWW becomes necessary, the use of the Purex concentrator is probably the most economical method, and \$250,000 has been budgeted for this purpose.

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5.5.5 Redox Waste Storage

The current practice of transferring aged Redox supernates to tanks equipped to handle only non-boiling wastes will be continued. On this basis, existing storage capacity for Redox self-boiling wastes should be adequate through the expected life of the tanks.

Planned modifications of Redox practices are predicted to minimize the requirements for Redox waste storage. Prototype facilities have been installed to return dissolved sludge, rich in sodium nitrate, from the 241-SX tank farm to the Redox Plant for use in coating-removal operations. If this operation is successful, the quantity of sodium nitrate stored in the SX tanks will be reduced without a corresponding increase in the volume of non-boiling wastes requiring storage. Additional storage space for non-boiling wastes may be made available in the 241-S tank farm by transferring non-boiling supernates to the 241-SX tanks for additional concentration.

5.6 Waste Fractionization

It is proposed to fractionize the high-level wastes so that the long-lived heat producers, cesium and strontium, can be immobilized and stored separately in small-volume, high-integrity containers. The residual wastes are to be immobilized as salt cakes with the other non-heating wastes. According to the plan presented here, all stored wastes would be immobilized and the plant would be in an equilibrium operating condition by the end of FY 1973.

5.6.1 Processing Plan

The processing plan for fractionizing high-level waste is outlined in Figure 2. The plan provides for 1) the removal of strontium-90 and cesium-137 from future high-level acid wastes and from the alkaline high-level wastes in storage; 2) the packaging and storage of the strontium and cesium as compact solids; and 3) the temporary storage of the self-heating salt effluents as alkaline slurries followed by their subsequent reduction to a salt cake for permanent storage in existing underground storage tanks.

The fraction of cesium and strontium that can be permitted to remain with the salt wastes to be solidified is determined only by the heat generation associated with radioactive decay. Since many of the "non-heating" wastes to be solidified do in fact contain some heat-producing isotopes, the processed wastes must be stripped of heat-producers to such an extent that the total remaining heat in the combined wastes can be safely dissipated from the underground tanks available.

5.6.1.1 Purex Acid Waste Processing

- (1) Acid wastes from Purex operations are to be clarified by centrifugation at the Purex plant. Cesium, strontium, and rare earths would be leached from the solids and returned to the liquid waste stream. The solids would be sent to underground storage in the 241-AX farm for decay of the scavenged short-lived fission products.

- (2) The clarified waste is to be transferred to B-Plant for fractionization by solvent-extraction. Using a CSREX flowsheet similar to that presented in Figure 3, the cesium, strontium, and rare earths are to be extracted into an organic phase, leaving an aqueous waste stream containing short-lived fission products and miscellaneous salts. The extracted fission products are to be sequentially re-extracted into an aqueous phase, yielding a cesium-strontium fraction and a rare-earth fraction. The cesium-strontium removal is to be about 99 percent efficient, while the cerium-rare earth removal efficiency may be as low as 50 percent.
- (3) The rare-earth fraction may be stored in B-Plant for approximately a three-year decay period, after which it is to be combined with the aqueous salt wastes from step (2). The need for this interim storage has not been firmly established. The cerium in this fraction produces heat at significant rates up to seven years, while the other short-lived fission products in the salt wastes have decayed to innocuous levels in about five years. The advantage accrued from storing cerium separately will depend on the availability of tank space for self-heating wastes and the amount of heat that can be safely dissipated from the solidified "non-heating" wastes.
- (4) The Cs-Sr fraction is to be concentrated and stored as a solution for about one year to dissipate the heat from Sr-89 decay before packaging. The cesium and strontium are then to be packaged as described in section 5.6.1.5.
- (5) The aqueous waste streams from steps (1), (2), and (3) are to be aged, initially under self-boiling conditions, for about five to six years in underground storage tanks. In combination with other wastes (e.g., OWW), the cooled wastes are to be blended with coating wastes and converted to salt cakes by in-tank concentration, as described in section 5.4.

5.6.1.2 Aged Alkaline Supernate Processing

- (1) The aged alkaline supernatant solutions from the Purex (241-A, 241-AX) and Redox (241-S, 241-SX) tank farms are to be transferred to B-Plant.
- (2) The cesium is to be removed (with about 99 percent efficiency) by an ion-exchange process similar to that presented in Figure 4, using a synthetic alumino-silicate type of ion-exchange medium.
- (3) It is recommended that the waste stream from the cesium removal operation be processed for technetium recovery, using an anion-exchange process. The technetium solution would be concentrated and stored as a solution. A sequential exchange cycle using a cation resin may be required if a purified technetium product is desired.

While this step has no bearing on waste disposal parameters, Tc recovery is believed to be justified by the nominal cost of the recovery operation

and by the value of this element that does not exist in nature. See section 5.8.1.

- (4) The separated cesium is to be packaged as described in section 5.6.1.5.
- (5) The salt waste stream from the supernate processing operation is to be converted to a salt cake, in combination with other non-heating wastes, by in-tank concentration as described in section 5.4.

5.6.1.3 High-Level Sludge Processing

- (1) The long-lived isotopes are to be removed from the sludges in the A, AX and SX tanks. While current plans are based on using sluice-mining techniques to remove the sludges to stainless steel vessels for dissolution in acid, laboratory studies are in progress to find a way to leach the desired isotopes from the sludges without dissolving the hundreds of tons of iron oxides present. The removal of heat-producers need not be complete; any residual heat-producers would limit the amount of salt wastes that might be later solidified on top of the residual sludge layer.
- (2) If technically and economically feasible, plutonium and/or neptunium is to be recovered from the sludge; the recovered product would be transferred to the Purex Plant for purification.
- (3) The sludge fraction is to be processed in the B-Plant solvent extraction equipment for the removal (about 99 percent) of cesium and strontium.
- (4) The cesium and strontium are to be packaged as described in section 5.6.1.5.
- (5) The salt waste stream from the sludge processing operation is to be concentrated to a salt cake, in combination with other non-heating wastes, by in-tank concentration as described in section 5.4.

5.6.1.4 Redox Acid Wastes

The treatment of Redox acid wastes is not specifically included in this plan. Promising process adaptations are being developed to eliminate aluminum from the waste and thereby permit its being processed in the CSREX flowsheet. The goal is for the Redox acid wastes to be made compatible with the B-Plant operations by 1968.

5.6.1.5 Packaging and Storage of Cesium and Strontium

The separated cesium and strontium are to be adsorbed on a synthetic aluminosilicate ion-exchanger in a stainless steel canister. After thorough drying by passing heated air (about 800 °F) through the bed, the canister is to be closed and welded shut. The canister may take the form of a cylinder perhaps

ten to twelve inches in diameter, or an annular tank perhaps six-inches-thick and up to two feet in diameter; it is anticipated that up to one-half megacurie of cesium or strontium will be contained in each such canister. Experiments are in progress to demonstrate the thermal, chemical, and mechanical stability of this system. The sealed canister is to be stored under close surveillance until its equilibrium temperature and pressure can be determined; the unit is then to be transferred to a water basin or air-cooled vault for long-term storage. Later, the units may be transferred to a location requiring no artificial cooling, such as a shaft in the ground or salt mine.

While the canister is the primary containment mechanism for the cesium-strontium, its failure would not be catastrophic. The chemical affinity of the cesium and strontium for the alumino-silicate is so great that the leaching action of water would be very slow. In a storage basin, corrective action could be taken in the form of identification and repackaging of the leaker, and of demineralization of the basin water. Long-term storage of such a container would of course be best in a medium that excludes environmental waters, such as a salt dome or a natural desert.

5.6.2 Facilities Plan

New facilities are needed to remove the stored wastes from the underground tanks and to connect these facilities with existing piping systems for the transport of the radioactive solutions to B-Plant where the major processing steps are to be performed. Existing storage basins may be modified for the storage of the packaged long-lived heat-producers.

5.6.2.1 B-Plant Modifications

The B-Plant is a canyon-type building originally built in 1944 and operated to recover plutonium by the bismuth phosphate precipitation process. The building has been inactive since 1952. The B-Plant facilities are being modified and activated in three phases as the need develops. The functions of the cells to be activated under each phase are tabulated in Table VII.

In Phase I, the building was reactivated, the ventilation system modified to meet minimum standards in the portion activated, and the process equipment in ten cells was activated to permit the segregation and storage of crude concentrates of strontium and rare earths. This phase of the construction effort, which was justified by the need to accumulate a stockpile of separated fission products, is now complete. The operation of these facilities is scheduled to start in March, 1963.

In Phase II, the ventilation system is to be augmented, a more efficient ventilation filter is to be installed, and five additional cells are to be modified for the installation of a demonstration packaging unit for the separated cesium-strontium fraction. This unit includes several operations that have not yet been conducted under the production conditions, and the allocation of a substantial period for debugging prior to production assignment is considered to be prudent. This phase is ready for detail design, and is expected to be ready for debugging operations by about March, 1965.

Limited quantities of separated fission products may be packaged for shipment in this unit after the packaging operations are satisfactorily demonstrated.

In Phase III, the balance of the operating equipment is to be installed, and the equipment activated under Phase I and II is to be integrated into a single operational complex. The scoping effort for Phase III is scheduled to start in August 1963, and the construction effort is scheduled to be complete about May, 1966.

The operation of the B-Plant facility is estimated to cost \$3,200,000 per year, exclusive of general plant overheads. Of this amount, about \$100,000 is estimated to arise from the technetium recovery operations.

5.6.2.2 Waste Removal and Transport

As currently conceived, a sludge accumulation and dissolution facility is to be constructed adjacent to the 241-A and 241-SX tank farms. These units are to be similar to the process vaults used in the Uranium Recovery Program. The facility for removing A and AX farm sludges is scheduled to be operational by July, 1966, while the SX unit is to be operational by early 1969. If a satisfactory technology is developed for leaching the long-lived isotopes from the sludge in the tank, the capital expenditures for sludge treatment may be substantially reduced.

The operation of these units is currently estimated at about \$340,000 per year, exclusive of general plant overheads. The crew is programmed to remove the sludges from the Purex and Redox farms in sequence.

5.6.2.3 Storage of Packaged Isotopes

As currently conceived, one of the three 200 N cooling basins is to be modified for the storage of the packaged isotopes. These basins were originally constructed in 1944 for the interim storage of irradiated fuels prior to processing for plutonium recovery. The basins have been inactive since 1952.

A water-cooling system is to be provided to remove the heat (up to six million Btu per hour) from the basin water, and a demineralizing unit is to be installed to remove trace metallic ions and radioactive materials from the captive water coolant. Depending on the amount of cesium-strontium that can be safely packaged in a single canister, and on the amount diverted for beneficial use, one storage basin will be capable of storing between half and all of the isotopes to be stored. Provisions are being made to activate one of the basins in 1966. While a second unit may be needed in 1970, the probability of need is considered to be so low that the second unit is not included in the projected costs.

The cost of transporting the canisters to the storage basin and of maintaining the facility is estimated at \$100,000 per year, excluding general plant overheads.

5.7 Miscellaneous Requirements for Waste Handling

A fund of \$150,000 per year is budgeted to cope with unique problems that may develop in the management of radioactive wastes. As an example, funds may be necessary to improve the current capability of transferring boiling waste from one tank to another when a tank fails. This need could become pressing if transfers of boiling wastes between the East and West areas should become necessary.

5.8 Fission-Product Recovery

As presently conceived, the waste management processing complex provides no facilities specifically designed for fission-product recovery except those for the recovery of technetium-99 from tank-farm supernates. Facilities could be provided for the purification of technetium, cesium, strontium, cerium, or promethium if desired.

5.8.1 Technetium Recovery

The recovery of technetium is recommended as a part of the waste management processing effort because of the nominal cost of recovering this unique material. Failure to recover this material at this time would place a national resource in a relatively inaccessible condition from which it would be recoverable only at much greater cost.

Technetium has unusual properties. It is an efficient corrosion inhibitor when used in trace amounts, and it becomes superconductive at a higher temperature than any other known metal. This metal is not found in nature, and radiochemical wastes constitute its only source of supply.

The cost of recovering about one ton of technetium as a crude concentrate is estimated at \$250,000 for equipment and building modifications, and an operating expense of about \$100,000 per year for five to six years. These estimates predict a unit cost of about one dollar per gram for isolating this rare element.

5.8.2 Recovery of Cesium and Strontium

During the next three years, the needs of the Commission for recovering, stockpiling and purifying strontium will be met by operating an available complex which includes the Purex head-end, the CR Vault, the Strontium Semiworks, and the B-Plant Phase I equipment. This complex was activated to fill an interim need, and the operating units lack the reliability characteristic of the normal Hanford operational units. While this complex will be operated on a best-effort basis, a less-than-normal operating efficiency can be expected. With these reservations, it is anticipated that ten megacuries of crude strontium will be stockpiled in B-Plant and that strontium can be purified in the Strontium Semiworks at a rate up to four megacuries per year. Assuming an adequate supply of slurry filter casks, the strontium can be made available for shipment as dry strontium carbonate.

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HW-76352

-34-

Cesium can be adsorbed on an alumino-silicate in the shielded transfer tanks (STT) or other transfer casks for transport off-site. A new load-out station is now operational at the C Farm, and the supernate from the Purex waste in tank 103-A has been transferred to the C Farm as a source of cesium supply.

Assuming an adequate supply of shipping casks, cesium could be made available at a rate of five megacuries per year.

With the start-up of the Phase III B-Plant facilities in FY 1967, up to thirty megacuries each of cesium and strontium are to be segregated annually as a crude concentrate. At that time, it is intended that the Strontium Semiworks be shut down, and that the inventories of crude strontium and rare earths stored in the B-Plant Phase I equipment be merged with the comparable Phase III crude fractions. The separation of these isotopes could be effected in B-Plant at some additional cost, the magnitude of which would be dependent on the quantity and quality of isotope desired. For example, the solvent-extraction equipment might be operated on a campaign basis to separate and purify strontium. The separated and purified products might be loaded on zeolite beds for shipment. The operation would delay waste processing operations somewhat, and the allocated costs would be accrued to match the account of the products. Other processing schemes might be developed to match the needs defined at some future time. If a processing plant were located adjacent to B-Plant, it would be feasible to transfer the crude cesium-strontium concentrate to that plant by pipeline.

5.8.3 Recovery of Cerium and Promethium

During the next few years, the Purex head-end and B-Plant Phase I equipment will be operated to stockpile a rare-earth concentrate, possibly containing as much as ten megacuries of promethium. The primary purpose of the stockpiling operation is to create an available source of promethium-147 in which promethium-148 has decayed sufficiently to minimize gamma radiation. Plans to isolate the promethium will be made when a specific demand for the purified product is identified.

If cerium is needed to fill a specific demand, limited amounts of this isotope can be isolated in the Purex head-end equipment and loaded for shipment in a slurry-filter cask. This operation would interfere somewhat with the strontium stockpiling operation, and will be performed only when a need is identified.

When B-Plant Phase III is activated, a cerium-rare earth fraction will be segregated as a part of waste management operations. Up to 380 megacuries of cerium-144 and 55 megacuries of promethium will be available annually as a crude concentrate. Tentative plans are to store this concentrate for a few years before returning to the salt waste for storage and subsequent solidification; the need for this interim storage is not firmly established, and the temporary storage may not be required. This cerium-rare earth fraction could be made available for isotope recovery, either by loading on a zeolite for shipment or by transferring as a solution to an adjacent plant. Alternatively, the promethium might be isolated in new facilities (B-Plant -

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Phase IV) for about \$1,800,000 capital funds; this expenditure is not currently being programmed.

5.8.4 Isotope Processing Plant

An engineering study is currently in progress⁽⁶⁾, under the sponsorship of the Division of Isotopes Development, to determine the feasibility of constructing and operating an isotope processing and fabrication facility adjacent to the B-Plant. This study, which was authorized in October, 1962 is being conducted by Hanford Laboratories personnel with the assistance of engineers from the Chemical Processing Department. This study is based on the premise that the proposed waste fractionization plan is implemented, and that crude concentrates produced by the waste processing operations can be made available to the isotope processing plant. No changes were required in the proposed scope of the Waste Management Program to be compatible with these premises.

6. ESTIMATED COSTS

The implementation of the Waste Management Program will require authorization, during the next five fiscal years, to spend an estimated \$20,750,000 in capital funds in addition to those currently authorized. This request for capital funds is about equal to that which would be required to continue current practice, assuming that no failures of underground tanks were experienced. The pertinent construction projects are tabulated below, together with their estimated cost and possible source of funds. A tentative schedule of construction project activities is presented as Table VI. Estimates of the additional personnel and costs involved in operating the new facilities are also indicated below.

<u>Project</u>	<u>Possible Source of Funds</u>	<u>Estimated Capital Cost</u>	<u>Estimated Increase in Operational</u>	
			<u>Personnel</u>	<u>Annual Cost</u> ^(a)
<u>Construction Authorized</u>				
1. Waste Treatment, Z-Plant (CGC-912)	61-a-1	\$ 725,000		\$
2. Leak Detection - Self- Boiling Tanks (CAC-928)	61-a-1	510,000		
3. Equipment Disposal - Purex (CGC-964)	63-a-3	1,000,000		
<u>Planned for Future</u>				
1. New Waste Storage Tanks - Purex - 241-AX (CAC-945)	63-a-3	2,700,000 ^(b)		

0315587030

HW-76352

-36-

<u>Project</u>	<u>Possible Source of Funds</u>	<u>Estimated Capital Cost</u>	<u>Estimated Increase in Operational</u> ^(a)	
			<u>Personnel</u>	<u>Annual Cost</u>
<u>Planned for Future, cont.</u>				
2. Purex Essential Waste Routing System (CAC-970)	63-a-3	\$ 750,000		\$
3. In-Tank Waste Solidification -CGC-965- 200-E	Equip	520,000	+ 8	+ 120,000
- 200-W	Equip.	530,000	+ 8	+ 120,000
4. Packaging Demonstration Phase II, B-Plant (CGC-981)	62-a-2	1,500,000	+ 11	+ 300,000
5. Waste Fractionization Phase III, B-Plant	64-a-2	3,700,000	+200	+ 2,900,000
6. Sludge Removal & Waste Transfer - Purex -	PAC			
- Redox -	FY 65	2,000,000	+ 16	+ 340,000
	FY 68	2,000,000		
7. Packaged Waste Storage 200 N Storage Basin	PAC	500,000	+ 4 ^(c)	+ 100,000 ^(c)
	FY 66			
8. Additional Waste Storage Purex 241-AX, Phase II	PAC	2,300,000 ^(b)		
	FY 66			
9. Purex OWW Concentration - 202-A	Equip.	250,000		
10. Cribs and Test Wells	Annual	250,000/yr.		
	Equip.			
11. Miscellaneous Requirements - Equipment	Annual	150,000/yr.		
	Equip.			
Shut Down Strontium Semiworks and B-Plant-Phase I		-	-100	1,700,000
Total through FY 1968 -		\$20,750,000		
Net Change -			+167	\$ 2,180,000

(a) This estimate includes direct and indirect costs, but excludes general plant overheads.

(b) This cost estimate is based on the original scope design, which is currently being revised.

(c) These estimates include the transportation of canisters to the storage basin.

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7. ALTERNATIVES

The waste management plan as presented is considered to be optimum for Hanford, making productive use of the environmental advantages and facilities available. Studies of alternative courses of action will continue, however, to provide a basis for periodic reevaluation and modification of the program to achieve optimum results.

7.1 Gaseous Wastes

If in the future a reduction in the rate of iodine emission is desired, alternative courses of action may be taken. A factor of 30 reduction in the iodine emissions from the Purex Plant might be realized by installing a higher efficiency silver reactor system in the process vent system; based on an engineering study, a preliminary cost estimate of \$250,000 was obtained for implementing this step. As an alternative, a comparable reduction of iodine emission rate might be achieved without the expenditure of capital funds by increasing the cooling time of the irradiated fuel by 40 days prior to processing. This increased cooling time would be equivalent to increasing the working inventory of uranium and plutonium at Hanford by about six to seven percent.

7.2 Solid Wastes

As an alternative to burial, combustible wastes might be burned and the ash stored in a vault. In view of the successful burial practice currently in use, the construction and operation of such a unit is considered to be an unwarranted expense.

7.3 Low-Level Liquid Wastes

The use of cooling towers (at an estimated cost of \$1,800,000) has been considered to reduce the volume of cooling water released to the swamps. Such a reduction would reduce the level of the ground water under the area, thereby increasing the depth of soil available for fission product retention and storage; the decreased water flow would also increase the travel time to the river, allowing more time for decay of any fission products in the ground water. As currently evaluated, neither incentive is sufficiently large to justify the cost of towers, and installation is not planned.

The use of ion-exchange beds to remove and confine radioisotopes from the intermediate-level wastes has been considered as an alternative to the continued use of the ground to perform this function. The advantages of this step are judged to be insufficient to justify the cost of such installations.

7.4 High-Level Liquid Wastes

The safe confinement or disposal of high-level radiochemical wastes has received increasingly intensive study from technical and political leaders

throughout the world, and many alternative methods of waste "disposal" have been proposed. As part of this larger effort, the Chemical Processing Department has focused its effort on identifying the optimum method of coping with the large quantities of these wastes in existence and being generated at Hanford.

7.4.1 Criteria

In choosing a processing plan for managing high-level radiochemical wastes at Hanford, the following criteria were established, in decreasing order of importance:

- (1) Both existing and future wastes shall be immobilized in a chemical and physical form from which they cannot be readily released to the environment by natural phenomena.
- (2) The treatment methods shall require a minimum of near-term capital and operating expenditures, and shall require a minimum of time and cost to acquire the required technologies.
- (3) The conditions of long-term storage shall require a minimum of handling or surveillance.
- (4) The treatment methods shall be compatible with the recovery of fission products for beneficial use.

7.4.2 Approaches

Four courses of action are available for the treatment of high-level radiochemical wastes. These alternative courses differ primarily in the timing and degree of immobilization accomplished.

- (1) The wastes might be stored indefinitely as an alkaline slurry in underground tanks. Any heat from radioactive decay would be dissipated to the environment; the more active wastes would self-heat to boiling temperatures for periods up to fifty years unless some mechanical system of heat removal were provided. Under these conditions, radioactive cesium is retained in solution, and is mobile in the event that the storage container leaks. At Hanford the soil surrounding the storage tank would act as a secondary containment medium, and thereby provide some protection against sudden tank failure; continual surveillance would be required, however, and the contents of any leaking vessel would have to be promptly transferred to another vessel. The other radioisotopes of concern are associated with insoluble materials and are not so amenable to rapid movement through the soil; the solids in the more active wastes would, however, have to be kept covered with only moderately concentrated salt solution to permit the safe removal of heat. Failure to maintain conditions favorable for heat dissipation

could permit temperatures to be reached at which the nitrate salts would be decomposed and the concrete vessel would be weakened by dehydration.

- (2) The radiochemical wastes might be solidified as salt cakes in the underground tanks after storing the slurry from zero to one-hundred years to permit the heat-producing fission products to decay. In this case, nearly all of the cesium and strontium would be stored under self-heating conditions and would be subject to the same hazards and need for operational control as outlined in (1), above, during the one-hundred years of solution storage.
- (3) The wastes might be processed to remove the long-lived heat-producing isotopes for immobilization and separate high-integrity storage. The residual wastes might be solidified as salt cakes in underground tanks after five to seven years storage to permit the decay of the short-lived heat producers. This is the recommended approach.
- (4) The radiochemical wastes might be calcined or otherwise processed to convert the mixture to a stable chemical and mechanical form; the process waste might be packaged and stored. Storage as a liquid for three to five years would permit a significant reduction in the cost of the containers.

7.4.3 Evaluation

As reported last year⁽⁴⁾, the calcination of salts was studied⁽¹⁶⁾ intensively at Hanford. This approach was rejected because 1) this process showed no economic incentive over the recommended approach, 2) the adaptation of the calcination process to production-scale facilities in the existing B-Plant would require a relatively large development effort to resolve very difficult engineering problems, and 3) the process is not compatible with the recovery of radioisotopes for beneficial use. Development work in support of waste calcination was therefore no longer supported by the Chemical Processing Department. During the past two years, the calcination of wastes in sulfur has been demonstrated to be technically feasible.⁽¹⁷⁾ While this development might reduce the engineering complexity of a potential B-Plant installation, it does not provide significant financial incentive or compatibility with fission product recovery. The many processes designed to incorporate the radioisotopes in glass-like compounds are subject to the same disadvantages when evaluated for the processing of existing wastes in the underground tanks. Calcination and glass-making processes are therefore judged to be less desirable than the fractionization process proposed.

The first three alternatives were studied in sufficient detail to develop comparative cost estimates. Since the service life of the underground tanks is not known but has an overshadowing effect on the

0315587030

HW-76352

-40-

relative economics of the three alternatives, each alternative was estimated on the basis of a twenty- and a forty-year tank life. It is felt that the actual life of the tanks storing non-boiling wastes is probably bracketed by these values; the life of the tanks storing Purex and Redox self-boiling wastes is probably much shorter, however, and may well be less than twenty years. For this study, it was assumed that processing activities would continue at Hanford as currently scheduled through FY 1973, when all processing activity would cease. At this time, all current and future Redox and Purex wastes would have been processed to an equilibrium operating condition as defined by the approach taken. For each alternative, the capital and operating costs (excluding general plant overheads) were then estimated in 1964 dollars, using operating plans which were designed to make optimum use of available technology. The only expenses considered were those directly applicable to the treatment of high-level wastes; no fission product recovery activities were included in any estimate. For comparative purposes, the projected expenditures were reduced to the common basis of present worth, using an interest rate of four percent. These cost studies are summarized in Table VIII.

Approach I is intended to reflect a logical extension of current practice. New tanks would be required for the storage of non-heating wastes, starting in FY 1967. Redox process wastes would be stored under self-concentrating conditions in the SX Farm; boiling would stop after about seven years of storage, and the supernatant liquid would be transferred to older tanks. Purex process wastes would be stored under self-concentrating conditions. In addition to the four-tank AX Farm currently requested, continued tank storage would require a four-tank addition in FY 1966 and another four-tank farm in FY 1969. After about fifty years of storage, Purex wastes would cease boiling and they could be stored under non-boiling conditions thereafter. Replacement tanks would be provided for all wastes every twenty or forty years, depending on the tank life assumed.

Approach II is intended to reflect the impact of solidifying the wastes in underground tanks at such time that the rate of heat generation in the wastes would permit. Facilities for concentrating the wastes in the underground tanks would be provided in fiscal years 1964 through 1966. The solidification of non-heating wastes would start late in FY 1964 and continue about nine years, when the non-heating wastes would be solidified. The Redox process wastes would be stored as a slurry about fifty years, and the Purex process wastes about 110 years, when they would have decayed sufficiently to permit solidification as a salt cake a few feet thick. Tank requirements for self-concentrating wastes would be essentially the same as for Approach I, but no new tanks would be required for non-heating wastes; the Redox and Purex wastes could be stored in simple tanks (non-boiling) during the last forty to sixty years before solidification.

Approach III is intended to reflect the impact of removing the long-lived heat-producing isotopes for separate storage, and the solidification of

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HW-76352

-41-

residual salts in underground tanks after about five years (out of reactor) decay of the short-lived isotopes. As in Approach II, the solidification of non-heating wastes would start late in FY 1964 and would continue about ten years. Facilities would be provided in B-Plant to remove and package cesium and strontium from Purex acid and stored wastes, and cesium from stored Redox wastes, starting in FY 1966. The processing of Redox sludges and acid wastes would start about FY 1969. Plutonium and neptunium are assumed to be recovered from the stored Purex and Redox wastes as they are processed. The stored self-heating wastes would have been processed for cesium-strontium removal by about FY 1973. Since Purex wastes must age at least five to seven years (after reactor discharge) before solidification of the salts is permissible, the in-tank solidification program would be complete about 1980.

As indicated in Table VIII, a significant financial incentive exists for proceeding with the in-tank solidification program. The capital and operating costs through FY 1967 of implementing the in-tank solidification program are less than the cost of the new tanks required by that time to maintain continuity of production by present practices, even if no tanks fail in the meantime. The financial incentive for in-tank solidification becomes progressively larger each year thereafter, and reaches tens of millions of dollars when the original storage tanks start failing. For example, present worth calculations indicate an incentive for in-tank solidification of \$31,000,000 if a forty-year tank life is assumed, or \$60,000,000 if analyzed on the basis of a twenty-year tank life. Since the solidification of non-heating wastes provides significantly improved long-term safety (through immobilization of cesium) at reduced near-term costs, the justification for in-tank solidification is clear, and the project should proceed with all practicable speed.

While not so self-evident as with in-tank solidification, a financial incentive probably exists for the fractionization of high-level wastes. Assuming in-tank solidification proceeds as planned, the capital cost of implementing the fractionization program is estimated to be about seven-million dollars more than that of continued tank storage provided none of the tanks containing self-boiling waste fail. At such time, however, that the Redox and Purex (SX and A) tank farms must be replaced, the capital costs of continued tank storage of self-heating wastes will exceed that of fractionization by about six-million dollars. Similarly, present worth calculations indicate that fractionization is about \$10,000,000 more costly if the life of tanks containing self-boiling wastes is assumed to be forty years, but that fractionization has a financial incentive of nearly \$10,000,000 if these tanks need replacement after twenty years. While the life of the A and SX tanks cannot be predicted with certainty, the consensus of Hanford engineers is that the life expectancy of these tanks is no more than twenty years and may be appreciably less. The fractionization approach is therefore at least competitive with the decay-and-solidify approach, within the limits of accuracy of the analysis. The advantages of the

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fractionization approach include 1) improved safety, realized by immobilizing the hundreds of megacuries of cesium and strontium, 2) decreased future costs of continued confinement and surveillance, 3) the recovery of previously lost plutonium and neptunium, and 4) improved availability of selected fission products for beneficial use. In the judgment of the Chemical Processing Department, the fractionization approach is justified by the waste management advantages alone, and becomes an economic benefit if the separated fission products prove to have a real value to other programs. This course is therefore recommended for implementation.

7.5 Fractionization Processes

Many processes have been developed in the last few years to remove selected fission products from radiochemical wastes. The choice of processes to be used at Hanford is not yet firm, but the processing pattern is most likely to be as described under section 5.6.

7.5.1 Strontium Removal

As an alternative to the CSREX flowsheet, strontium might be recovered separately from the same streams by solvent-extraction using D2EHPA-solvent alone. The strontium-only flowsheet has been demonstrated, both in the laboratory and on a production scale in purifying strontium for fission product sales. Strontium might also be removed by co-precipitation with lead sulfate, but the efficiency of strontium removal is too low to meet the needs of the Waste Management Program. In neither case would simultaneous cesium removal be possible.

7.5.2 Cesium Separation

As an alternate to the CSREX process proposed, cesium might be precipitated from acid solution as the phosphotungstate or phosphomolybdate; these materials are soluble in sodium hydroxide and would thus be compatible with further processing. The chemical and operating costs of these precipitation processes are so much higher than with the CSREX process, however, that their use cannot be justified even though useable precipitation equipment exists in B-Plant. Solvent extraction with dipicrylamine in nitrobenzene may be used to recover cesium over a range of pH from about 3 to 13. While this process appears to be quite feasible, it cannot compete cost-wise with the CSREX process which extracts cesium and strontium simultaneously. Cesium may also be removed from acid solutions by adsorption on alumino-silicates; subsequent elution can be effected with ammonium carbonate solution. This process requires simple equipment and operating procedures, and is quite efficient. The cost of the equipment could not be justified, however, since cesium removal may be effected simultaneously with strontium in the CSREX equipment.

The most attractive alternative process for the extraction of cesium from the alkaline supernates involves the use of a solvent extraction flowsheet very

0315587030

HW-76352

-43-

similar to that presented in Figure 3, except that only BAMBP (4-sec butyl (α methyl benzyl) phenol) is used as the complexing agent in the solvent. This approach would permit all of the radioactive wastes to be processed for cesium-strontium removal in the same equipment. In this case, optimum process performance would require that the various solutions be processed separately on a campaign basis, and some additional storage capacity would be required in which to accumulate the feed solutions. Both the ion-exchange and the solvent-extraction processes are technically and operationally feasible, and the choice of process to be used will be based on the relative economics of the two approaches. The capital and operating costs of the two alternatives are currently being estimated in more detail.

Cesium removal from alkaline solutions has also been demonstrated by a variety of other processes. Ion-exchange processes have been highly successful in the laboratory, using either alumino-silicates (commercial zeolite molecular sieves, naturally-occurring clinoptilolite, or Decalso) or a phenolic resin, Duolite C-3. The efficacy of commercial zeolites and clinoptilolite has been demonstrated with full-level plant solutions. Decalso has also been used in production-scale operations for cesium shipment. A commercially available synthetic zeolite was chosen for this application because its cost and performance characteristics are competitive and because a mechanism for quality control is already established. Elution of the cesium from the zeolites has been accomplished with several reagents, the most promising being ammonium carbonate. Cesium may also be precipitated from alkaline solution as a double salt with zinc- or nickel-ferrocyanide. Metathesis with silver ion yields a relatively pure cesium solution which is readily separable from the residual metallic cyanides. The operating cost of this precipitation process is too high for economical use, even though the necessary equipment exists in B-Plant.

7.6 Packaging of Separated Isotopes

The desired characteristics of highly concentrated radioactive material for long-term storage have been identified as: 1) long-term stability at high temperatures and levels of irradiation, 2) low corrosivity, 3) low solubility and dispersability in the environment, 4) high thermal conductivity, 5) small bulk, 6) accountability, and 7) retrievability.

The most promising approach to date is the fixation of the long-lived heat emitters on alumino-silicate beads, with the dried beads being contained in a high-integrity canister. Methods other than adsorption on zeolites have been studied for long-term immobilization, including the preparation of various cesium and strontium glasses and the fixation of strontium oxides by fluxing with sulfur or lithium fluoride. These processes would require higher-cost installations than would the adsorption processes, and do not appear to offer sufficiently compensating advantages.

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HW-76352

-44-

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HW-76352

-45-

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031712281030

-46-

TABLE I

AIR-BORNE EMISSIONS FROM
THE SEPARATIONS PLANTS

<u>Isotope</u>	<u>Principal Source</u>	<u>1960</u> <u>in curies per day, average</u>	<u>1961</u>	<u>1962</u>
I-131	Redox, Purex	0.96	0.70	0.28
Zr-Nb-95	Redox, Purex	0.01	0.005	0.0024
Ru-106	Redox, Purex	0.006	0.005	0.0036
Ce-141	Redox, Purex	0.001	0.006	0.0002
Sr-90	Strontium Semiworks	*	0.0005	0.0015
<u>in microcuries per day, average</u>				
Plutonium	234-5 Building	31.2	19.4	10.5

*The Strontium Semiworks was activated during 1961.

TABLE II

ESTIMATED VOLUMES AND RADIOACTIVITY OF
LOW-LEVEL WASTES DISCHARGED TO GROUND SURFACE

<u>Year</u>	<u>Billions of</u> <u>Gallons Discharged</u>	<u>Total Curies of</u> <u>Fission Products</u>	<u>Grams of</u> <u>Plutonium</u>
Cumulative to 12/31/56	24.2	1770	63
1957	5.1	100	157
1958	5.3	600(a)	185
1959	4.5	39	41
1960	4.5	8	8
1961	4.7	1850(b)	80
1962	5.2	1774(b)	47

(a) The major contribution to this total was a process leak into the Redox cooling water system

(b) Most of these discharges came from the Purex Plant, where more frequent equipment failures permitted process solution to enter the cooling water system.

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HW-76352

-47-

TABLE III

ESTIMATED VOLUMES AND RADIOACTIVITY OF
INTERMEDIATE-LEVEL WASTES DISCHARGED TO GROUND

<u>Year</u>	<u>Millions of Gallons</u>	<u>Total Activity - Curies</u>			<u>Grams of Pu</u>
		<u>Gross Beta</u>	<u>Cs-137</u>	<u>Sr-90</u>	
Cumulative to 12/31/52	155	41,000	7,600	240	
1953	85	48,400	1,000	560	
1954	770	372,200	24,000	9,200	
1955	760	826,600	4,600	11,800	
1956	900	769,100	7,500	6,100	
1957	560	321,400	6,800	4,00	30,380 ^(a)
1958	470	127,000	1,500	1,000	7,250
1959	400	8,700	1,200	100	5,170
1960	320	7,800	700	50	4,110
1961	245	6,366	370	230	9,920
1962	315	61,850 ^(b)	300	190	5,434
<hr/>					
Estimated current Inventory ^(c)		249,000	48,000	29,000	63,200

(a) Through 1957.

(b) The increased beta activity is due entirely to ruthenium volatilized in the Purex Plant during a plant test.
See Section 4.3.2.

(c) Corrected for radioactive decay.

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031155071030

HW-76352

-48-

TABLE IV

ESTIMATED DISCHARGES TO SUB-SURFACE CRIBS DURING 1962

<u>Crib</u>	<u>Source</u>	Millions of <u>Gallons</u>	Curies of Fission Product Activity			<u>Pu - Grams</u>
			<u>Gross Beta</u>	<u>Cs-137</u>	<u>Sr-90</u>	
A-3	Silica Gel Waste	0.04	0.13	< 1	< 1	-
A-10	Purex Process Con- densate	103	55,900 ^(a)	4.2	86	71
A-21	Purex I-131 Scrub Waste	5	270	39	< 1	5
A-24	Tank Farm Con- densate	5	118	118	< 1	-
A-30	Purex Steam Con- densate	129	543	46	1.5	40
S-6	Redox Steam and Cooling Water	51	218	7.7	12.6	1
S-7	Redox Process Con- densate	9	1,420	16	56	13
SL-1 & 2	300 Area Lab. Waste	4	1,090	50	19	5
SL-1 & 2	222-S Waste	0.03	< 1	< 1	< 1	0.2
276-S	276-S Waste	0.005	2.3	< 1	< 1	-
SX-1	Tank Farm Con- densate	1	13	13	0.4	-
TY	Equip. Decontami- nation	0.7	2,190	-	-	10.5
U-12	UO ₃ Concentrator Condensate	4	5	-	-	-
Z-9	Recuplex	0.05	-	-	-	2,444
Z-12	Lab. & Process Wastes	6	-	-	-	2,845
C-6 & C-4	Strontium Semiworks	0.03	111	-	9.8	< 1
Total		317	61,880	294	185	5,434

(a) See section 4.3.2.

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TABLE V

HISTORY AND STATUS OF LONG-LIVED ISOTOPES IN THE GROUND WATER

Isotope	Source	Year Detected	General Site Location	Originally Detected		Status - 1/63	
				Distance from Disposal Site in feet	Conc. as Fraction MPC _w	Dist. ft.	MPC _w
Cs-137	Scavenged Waste	1956	N of B-Plant	1,000	0.005	No longer detected after 1956	
Co-60	Scavenged Waste	1956	N of East Area	1,000	0.02	3,000	0.02
Cs-137	BiPO ₄ Low-Level Waste	1956	W of T-Plant	100	0.07	No longer detected after 1956	
Sr-90	Redox Condensate	1957	W of Redox	15	1.00	15	0.3
Sr-90	B C Cribs	1962	SE of East Area	15	0.10	2,500	0.03
Tritium	200-E Process Waste	1962	SE of 200 East	39,000	< MPC _w	39,000	< MPC _w
Tritium	200-W Process Waste	1962	SE of 200 West	16,000	< MPC _w	16,000	< MPC _w

TABLE VI

TENTATIVE DESIGN AND CONSTRUCTION SCHEDULES
FOR MAJOR WASTE MANAGEMENT ACTIVITIES

<u>Function</u>	<u>Starting Period</u>		<u>Beneficial Use</u>
	<u>Scope Design</u>	<u>Detailed Design Construction</u>	
1. Storage of Purex Wastes-241-AX Phase I		4Q FY 63	1Q FY 65
-241-AX Phase II	2Q FY 64	2Q FY 66	1Q FY 68
2. Purex Essential Waste Routings		3Q FY 63	1Q FY 65
3. In-Tank Solidification - 200 East		4Q FY 63	4Q FY 64
- 200 West	4Q FY 64	1Q FY 65	1Q FY 66
4. B-Plant Modifications - Phase II		4Q FY 63	3Q FY 65
- Phase III	1Q FY 64	3Q FY 64	4Q FY 66
5. Sludge Removal and Transport - A, AX	3Q FY 63	4Q FY 64	4Q FY 66
- SX	2Q FY 67	3Q FY 67	3Q FY 69
6. Long-Term Storage - 200 North	1Q FY 64	4Q FY 65	1Q FY 67

-51-

TABLE VII

PRELIMINARY 221-B CELL ALLOCATIONS

<u>Cell No.</u>	<u>Function: (Initial; Final)</u>	<u>Phase(s)</u>
1, 2	Equip. Storage; Maintenance, Container Cleaning and Inspection	I; II and III
3	Railroad Tunnel	I, II and III
4	Temporary Container Storage	II and III
5	Sr Concentration; Sr-Cs Concentration	I and II; III
6	Sr Storage; Sr-Cs Storage	I and II; III
7	Sr Storage and Vessel Vent System; Sr-Cs Storage	I and II; III
8	Sr and Ce-Rare Earth Storage	I, II and III
9	Feed Receiver and Waste Collection	I, II and III
10	Cell Drainage	I, II and III
11	Sr Precipitation; Acid Waste Receiver	I and II; III
12	Sr Centrifugation; Feed Clarification	I and II; III
13	Cs, Sr Precipitation; Sludge Solution Receiver	II; III
14	Cs, Sr Centrifugation; Feed Clarification	II; III
15	Waste Packaging	II and III
16	Waste Receiver and Off-Gas Treatment	II and III
17	Cask Load-In; Waste Packaging	II; III
18	Spare	II and III
19	Cs Concentrator and Cs Receiver	III
20	Vessel and Condenser Vent System	III
21	Ce-Rare Earth Storage	III
22	Ce-Rare Earth Storage	III
23	Recovered Cs Eluent Receiver	III
24	Cs Eluent Feed Tank	III
25	Cs Ion-Exchange	III
26	Tc Ion-Exchange	III
27	Tc Concentration and Storage	III
28	Spare	III
29, 30	Waste Collection and Neutralization	III
31	Spare	III
32	Organic Treatment and IAX Tank	III
33	ICW and IBP Tanks	III
34	IB and IC Columns and ICP Tank	III
35	IAF and IAP Tanks	III
36	IA Column and IAW Tank	III
37, 38, 39 and 40	Spares Available for Phase IV Operations	-

Notes for TABLE VIII

- (1) The indicated inventory of recoverable isotopes is an approximation based on the irradiated fuel processed through Redox and Purex since 1958. The annual receipts are based on projected Purex production only.
- (2) The recovery capability is based on the assumption that the indicated inventory plus receipts are recovered in a six-year operating period, starting in FY 1967.
- (3) The tanks used to store non-boiling wastes probably have a service life measured in decades, and an evaluation of alternative treatments of non-boiling wastes using tank life expectancies of twenty and forty years appears reasonable. The Hanford tanks containing boiling wastes have been chemically and mechanically stressed, however, (see section 4.4.2.2) and the life expectancy of these tanks is uncertain. The consensus of Hanford engineers is that the A Farm will probably need replacement within five to ten years, and that a fifteen- to twenty-five year life is as much as should be assumed for the tanks of this type in this type of service.
- (4) The indicated costs include the capital costs of plant additions and the operating costs (including direct and indirect costs, but excluding general plant overheads) attributable to the handling of radioactive waste in the indicated way. These estimates do not include the capital cost of projects already authorized or of the Purex waste routing system, the operating costs for cesium-strontium recovery prior to FY 1967, or the capital and operating costs for technetium recovery. No escalation was included in any cost estimate. The times indicate the years in which the expenditures are assumed to be authorized.
- (5) The cost of constructing new storage tanks for non-boiling wastes was assumed to be 50¢ per gallon of storage space provided. It was assumed that the sludge in non-heating waste would not be moved, so replacement storage space was assumed to be provided for the supernatant solutions only. Since the in-tank solidification program would be operating by the time the T, U, B, and C tanks would require replacement under the twenty-year-life assumption, it was arbitrarily assumed that only half of the tanks would have to be replaced.
- (6) The cost of constructing new storage tanks for self-boiling wastes was assumed to be 80¢ per gallon of storage space provided. Replacement storage space was assumed to be required for the sludge as well as for the supernatant solution. Since the Redox wastes cease to self-boil after a few years of storage, one-half of the SX Farm capacity was assumed to be replaced with tanks for self-boiling wastes and half for non-boiling wastes.

Notes for TABLE VIII, continued

- (7) In Approaches II and III, the non-heating wastes are programmed to be solidified by about FY 1971 and 1973, respectively. In Approach III, the Purex and Redox acid wastes that have been processed for cesium-strontium removal are still self-heating from the decay of short-lived fission products; these wastes are programmed to be solidified in FY 1980.
- (8) These operating costs reflect the added costs of the sludge removal and treatment systems, the cesium-strontium removal systems, and the cesium-strontium packaging and storage systems. Since most of the functions of current Phase I-B-Plant and Strontium Semiworks operations are included in this activity, these costs have been deducted from the total cost of the fractionization effort. The costs of technetium recovery are also excluded from this number since this activity has no bearing on waste management.
- (9) The Redox process wastes were originally self-boiling and are assumed to be all non-boiling by 1980. These wastes are assumed to have cooled sufficiently without fractionization so that they can be solidified in tanks by 2020. The Purex process wastes are assumed to be self-boiling until 2030, after which they can be stored in tanks for non-boiling wastes; these wastes are assumed to have cooled sufficiently to permit in-tank solidification by 2080. The charges for in-tank solidification include \$500,000 for a new operating facility, plus one year's operation for Purex waste and two years' operation for the Redox wastes.
- (10) The 200 N storage basin was constructed in 1944 and is assumed to have a forty-year life. A replacement basin is assumed to cost one-million dollars. It is possible that a replacement basin would not be built, but that the packaged isotopes would be transferred to a natural environment requiring no replacement, such as a shaft in a salt mine.
- (11) The cost of maintenance, surveillance, and waste transfers are assumed to increase after 1980 in Approach I because of the assumption that there would be no other similar type activity in the vicinity. In Approach II, most of the wastes are assumed to have been solidified, so no penalty was assessed during the time of continued storage. Only nominal surveillance is assumed after all wastes are solidified in Approaches II and III.
- (12) All capital and operating costs listed above were reduced to the common basis of present worth in 1964 dollars, assuming an interest rate of four percent.

[illegible][illegible]

[illegible]

Estimated Cost of the Project 1970
(\$11,000 of 1970 dollars for year)

(7) 0000000000

(c)
The following information is being furnished to you under the provisions of the Freedom of Information Act, 5 U.S.C. 552, and the Privacy Act, 5 U.S.C. 552a.

 

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IN TANK SOLIDIFICATION

GRADE

COOLING PERIOD

FIGURE 1

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FIGURE 2

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Note:
D2EHPA is di(2-ethyl hexyl) phosphoric acid
BAIBF is 4-sec butyl (α methyl benzyl) phenol

THE CSREX FLOWSHEET
CESIUM-STRONTIUM-RARE EARTH EXTRACTION FROM PUREX ACID WASTES

FIGURE 3

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CESIUM EXTRACTION FROM ALKALINE WASTES
FIGURE 4

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END

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