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RADIOACTIVE WASTE MANAGEMENT

VOLUME THREE

R. E. Blanco
K. E. Cowser, L. C. Lasher, L. Gemmell, and S. G. Pearsall

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INSTITUTO DE ENERGIA ATÔMICA
Caixa Postal 11.049 (Pinheiros)
Cidade Universitária "Armando de Salles Oliveira"
SÃO PAULO — BRASIL

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TREATMENT OF WASTES FROM NUCLEAR RESEARCH INSTITUTIONS IN THE UNITED STATES *

R. E. Blanco

The individual unit operations of waste management have been described in previous lectures. The objective of this lecture is to describe the complete waste treatment systems at several U. S. nuclear installations. The waste management system at each nuclear installation is generally based on local requirements. Probably no two installations in the world are exactly alike. The student is referred to the Technical Report Series of the IAEA, particularly #27, "Technology of Radioactive Waste Management Avoiding Environmental Disposal", and the IAEA Symposium on Practices in the Treatment of Low and Intermediate-Level Radioactive Wastes, Vienna, Austria, December 6-10, 1965, for description of complete management systems at most nuclear installations in the world.

I have reproduced a paper by Dr. K. E. Cowser et al. which describes in detail the systems at Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL). This paper is used as the basis for this lecture. The systems at Argonne National Laboratory (ANL) consists of evaporation, ion exchange, flocculation, neutralization, cementing, and shipment to Kentucky or Oak Ridge for burial. The system at Los Alamos has been described in a previous lecture.

I have also included a description of a scavenging-precipitation ion exchange process which was developed at ORNL for treating low-level waste. It was successfully demonstrated in a 15,000 gal/day pilot plant at ORNL.

OPERATIONAL EXPERIENCE IN THE TREATMENT OF RADIOACTIVE WASTE AT OAK RIDGE NATIONAL LABORATORY AND BROOKHAVEN NATIONAL LABORATORY[1,2]

K. E. Cowser[3], L. C. Lasher[4], L. Gemmell[5], and
S. G. Pearsall[5]

ABSTRACT

Management of radioactive liquid and solid wastes (categorized as low-activity level and intermediate-activity level by this symposium) at Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL) are reviewed. Modifications evolved from operational experience and the findings of research studies are discussed, and their effect on effluent quality and cost of operation are described.

(*) Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation. Notes for lectures on Nuclear Waste Treatment and Fuel Reprocessing presented at the Institute de Energia Atomica, Nuclear Engineering Division, Cidade Universitaria, Sao Paulo, Brazil, October-November, 1969.

(1) For presentation at the Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Wastes, Vienna, Austria, December 6-10, 1965, sponsored by IAEA and the European Nuclear Energy Agency of the Organization for Economic Cooperation and Development.

(2) Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

(3) Health Physics Division, Oak Ridge National Laboratory.

(4) Operations Division, Oak Ridge National Laboratory.

(5) Health Physics Division, Brookhaven National Laboratory.

At ORNL operating experience with the 1330 liter/min process waste treatment plant has demonstrated that plant efficiency exceeds the design specifications of 50% removal of gross radionuclides. The treatment in the horizontal flow water-softening plant includes a $CaCO_3$ precipitation using lime and soda ash, for removal of ^{90}Sr and the rare earths, and the addition of illitic clay, for removal of ^{137}Cs. Under normal operating conditions, an average of 80 to 90% of these important radionuclides are removed at a cost of about $0.1 per 1000 liters. By optimizing the combination of lime, soda ash, and clay, it has been demonstrated that removal of strontium and cesium can be increased to 96 and 92%, respectively. Increased rates of flow through the plant and detergents and complexing agents in the waste increased the suspended solids in the plant effluent and reduced efficiency. To counter decreased efficiencies, 25% of the plant effluent is recycled for additional treatment, a polyelectrolyte was added to improve coagulation, and the treatment plant effluent was routed through a 5.7-million-liter settling basin. Efficient monitoring of the process waste system has resulted in reduced volumes of waste that require treatment (750 liters/min) and reduced amounts of ^{90}Sr, ^{137}Cs, and rare earths in the waste ($< 3 \times 10^{-5}$ μc/ml). Intermediate-level liquid waste (average of 1.6×10^7 liters/year, containing about 5 μc/ml) will be treated by means of a 2300 liter/hr evaporator, scheduled for operation in 1965. Permanent disposal of evaporator concentrate below the zone of ground-water circulation by the hydraulic fracturing method is under study.

At BNL, low-level liquid wastes and sanitary wastes are combined to receive biological treatment by an Imhoff tank and sand filter. As a consequence of diligent waste management, the quantity of gross activity in these wastes (average flow, 2200 liters/min) has been reduced to 1×10^{-7} μc/ml. The filter beds are now removing about 20% of the radionuclides. An evaporator is used to process intermediate-level wastes that average 1×10^6 liters/years and contain 0.004 to 0.2 μc/ml. A decontamination factor of 10^4 and a volume reduction of 140 are achieved. Previously, 200-liter drums were used to package contaminated solid waste and evaporator concentrate slurried with cement for off-site shipment. As a result of increased volumes of solid waste and excessive external radiation levels, reinforced concrete vaults are now used with walls 15 cm to 43 cm thick for gamma shielding. Waste slurry from the evaporator is now solidified in the vaults with cement and vermiculite to reduce handling and shipping costs and to preserve the integrity of the container. Cost and exposure to personnel during solid waste processing are further reduced by baling contaminated filters, decontaminating metal objects with shotblasting, and dismantling large grossly contaminated items with shaped charges of high explosives.

INTRODUCTION

The principal objective of radioactive waste management at Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL) is to dispose of wastes as safely and economically as possible. Some of the radionuclides generated at both installations are eventually released to the off-site environment but only under controls consistent with the recommendations of the National Committee on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICPR), and the Federal Radiation Council [1] [2] [3]. Waste segregation at its sources and waste treatment play integral parts in the effective control of radioactive wastes. Continuous vigilance is maintained to determine the fate of radionuclides that are released and to ensure that such operations are conducted safely.

The purpose of this paper is to review briefly the methods used to segregate and treat low-activity-level and intermediate-activity-level liquid and solid wastes, and to describe the effects of system modifications on effluent quality.

WASTE TREATMENT FACILITIES AT ORNL

Oak Ridge National Laboratory is a nuclear research and development center that houses a variety of laboratories, chemical processing pilot plants, radioisotope production plants, nuclear reactors, and supporting service facilities [4]. Liquid wastes evolve from all of these operations, but principally by reactor operations, radioisotope production, and pilot plant operations. The goal of the waste management program is to reduce the concentration of radionuclides in liquid effluents released to the Clinch River to less than one-tenth of the continuous occupational exposure limits recommended by ICPR [5]. Dilution in the river will further reduce the concentration of radionuclides by a factor of about 500.

LOW ACTIVITY LEVEL LIQUID WASTE SYSTEM

Process waste from equipment cooling systems, floor drains, storage canals, and laboratory sinks is of low-activity level and has essentially the same chemical composition as tap water (alkalinity, 94 mg/liter; total hardness, 127 mg/liter; and pH, 7.3). Acid-bearing waste will occasionally lower the pH to as low as 2.0. As indicated by the flow diagram (Figure 1), a network of underground terra-cotta pipes is used to collect the wastes and convey them by gravity flow through a central monitoring station and diversion valve to the 3.8-million-liter equalization basin or to the 5.7-million-liter settling basin. Emphasis has been placed on in-plant monitoring as a consequence of a larger than normal and intermittent release of activity to the system [6] [7]. Additional pretreatment holdup capacity has been furnished (11.4-million-liter emergency basin) in case the level of activity entering the equalization basin becomes too high for treatment. The Melton Valley processing facilities are currently under construction. Although provision is made for treatment of wastes, most of the cooling water is likely to be uncontaminated and after surveillance will be discharged to the creek without treatment.

Since 1960, each of the main waste lines of the process waste system has been continuously monitored for activity and rate of flow, and each has been continuously sampled in proportion to flow. The flow and activity measurements are telemetered to the Waste Monitoring Center. Figure 2 shows a cross section of a waste water monitor. Additional GM tuber, either similar to the one shown or an end-window type*, are included in the assembly when greater sensitivity is required. The effect of an efficient in-plant monitoring and control program is reflected by the reduction in concentration of radionuclides in process waste (Table I). Concentrations of the important radionuclides ^{90}Sr and ^{137}Cs have been reduced by factors of about 7 and 100, respectively. Total rare earths have also been reduced significantly. By restricting or eliminating the entrance of uncontaminated waste water to the process waste system, the volume of waste that requires treatment can be significantly reduced. Two 0.57-million-liter ponds were installed January 1965 to collect and monitor waste water from the 4500 area. Since only a small fraction of these wastes requires treatment, the total flow of process waste requiring treatment during 1965 has been reduced by about one-third.

PROCESS WASTE TREATMENT PLANT

A number of water treatment processes were investigated in the laboratory to determine their removal efficiencies for ^{90}Sr, ^{137}Cs, and the rare earths [8] [9]. Pilot plant tests were performed using upflow coagulation and sedimentation units and an ion exchange assembly [10]. From these laboratory and pilot plant studies and the antecipated future needs of the process waste system, waste treatment requirements and plant features established [11]. Design specifications required that removal efficiencies be in excess of 50% for the important radionuclides, ^{137}Cs and ^{90}Sr.

Since August 1957 process wastes have been treated by a horizontal flow water-softening plant. The process includes lime-soda softening with provision for alternate use of phosphate coagulation for ^{90}Sr removal and the addition of clay for increased ^{137}Cs removal. The plant, capable of treating 1.9 million liters/day, is so constructed that an additional treatment line of similar capacity could be added. Figure 3 is a drawing of the plant showing the treatment units and their arrangement. The principal features include two 1300 liter/min centrifugal pumps with rate-of-flow controllers, three

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(*) Lionel Anton, Model 210T, mica end window 1.4 mg/cm^2 thick.

200 lb/hr gravimetric chemical feeders, a flash mixing basin (1.3-min retention), a coagulation basin divided into three compartments by wooden baffles (31-min retention), and a sedimentation basin (129-min retention) with an automatic sludge collector. Sludge is stored for brief periods in a 61,000-liter underground concrete tank before it is discharged into a disposal pit excavated in a local shale formation. Flexibility in design allows recirculation of the plant effluent, sludge return from the collecting sump or underground tank, and addition of chemicals to each coagulation compartment and the flash mixer.

The primary treatment process is the precipitation of $CaCO_3$, using lime and soda ash, for removal of ^{90}Sr and the rare earths; illitic clay is added for removal of ^{137}Cs. In Table II the efficiencies of radionuclide removal are listed. Various combinations of lime and soda ash feed rates were originally tried on a plant scale. Since March 1958 the stoichiometric requirements for the precipitation of calcium and magnesium in the waste have been satisfied, and 200 mg/liter of excess soda ash have been added. During normal plant operation, the removal of strontium and rare earths averaged 84 and 86%, respectively. Addition of 200 mg/liter of Grundite, a commercial clay product containing about 70% illite, began October 1958 and resulted in the removal of 86% of the cesium. The clay feed was also instrumental in increasing the removal of ^{106}Ru (76%), of ^{60}Co (78%), and of gross beta activity (88%). Occasionally small quantities of gross alpha activity were detected in the waste; about 90% of these materials was removed. Detergents and chelating agents in the waste were responsible for the low efficiencies during January 1960 to June 1961; these factors and the increased rates of flow through the treatment plant also hindered plant performance during July 1961 to December 1964. During July 1961, the treatment plant effluent was rerouted to the inlet of the 5.7-million-liter settling basin; some improvement in the total removal of radionuclides resulted due to the additional settling of suspended solids. By removal of uncontaminated process water from the system, the rate of flow through the treatment plant was reduced to 900 liters/min (first 8 months of 1965). About 25% of the waste is now recycled for additional treatment, and strontium and cesium removals are averaging 86 and 81%, respectively.

In an effort to further improve the process, studies were undertaken to establish the optimum combination of lime, soda ash, and clay and to increase removal of suspended solids by the use of coagulant aids [12].

To determine the importance of each of the important chemical variables on treatment efficiency, a statistically designed experiment was performed in the laboratory with process waste. The principles of response surface methodology and, in particular, the method of the path of steepest ascent were used to discover those combinations of lime, soda ash, and clay feed which would remove the greatest amounts of strontium and cesium [13]. As shown in Table III, laboratory studies indicated an optimum of 96% removal of strontium with the listed additions of lime, soda ash, and clay; cesium removal with this combination was 92%. Removal of these two radionuclides during normal plant operation are listed for reference. The increased rates of chemical feed were applied in the plant for a period of 1 week, during which time 12.5 million liters of waste was treated. Strontium removal was increased from 84 to 94% and cesium removal from 86 to 91%. Although the chemical feeders handled the extra load satisfactorily, a build-up of sludge occurred in the flash mixer and settling basin. About twice the volume of sludge normally generated, 0.3% of the volume of waste treated, was accumulated during this period. Therefore, it was decided to optimize the chemical feed only when warranted by increased quantities of strontium and cesium in the waste.

Investigation of radionuclide distribution in the plant effluent showed that 40 to 50% of the ^{90}Sr, ^{137}Cs, and gross beta activity was associated with suspended solids. During normal operation, the effluent contained an average of 35 mg/liter of suspended material. Several coagulant aids were tested in the laboratory under conditions designed to simulate plant operation; [14] a combination of two of these polyelectrolytes improved the settling characteristics of the solids in the system under most conditions. A combination of 0.75 mg/liter of Hagan 50 and 5.0 mg/liter of Hagan 18 increased the removal of gross activity to 95%, strontium to 86%, and cesium to 93% in 40 million liters of waste treated (Table IV). Laboratory studies also showed that improvement of ^{137}Cs removal could be expected by using Grundite heated for 15 to 30 min at 600°C; [15] the heat treatment improved the sorption and settling properties of Grundite. To enhance the removal of suspended solids, 1.0 mg/liter of Hagan 50 is used currently.

The total cost of the treatment plant was $306,000. For chemicals, operator's time, and utilities, the cost of waste treatment is about $0.05 per 1000 liters. Maintenance cost, analytical costs, and a 10-year amortization of the initial investment in the plant would add another $0.05 to $0.08 per 1000 liters.

INTERMEDIATE ACTIVITY LEVEL LIQUID WASTE SYSTEM

Intermediate-level wastes from chemical processing and reactor operations are collected in stainless steel monitoring tanks (3800 to 15,000 liter in size), neutralized with NaOH, and pumped to three 0.64-million-liter concrete storage tanks (Figure 4). Liquid levels in the monitoring tanks are transmitted to the Waste Monitoring Center, and this information is used to control the volume of waste received from the many different sources. During the past 5 years the volume of waste has averaged 16 millions liters/year. Cesium-137 (4.5 μc/ml), ^{90}Sr (0.74 μc/ml), and ^{106}Ru (0.10 μc/ml) were the principal radionuclides in the waste. Sodium nitrate and sodium hydroxide comprised about 70% of the total solids. The soil disposal system shown in Figure 4 (seepage trenches) will shortly be abandoned and will be replaced by a waste evaporation system.

WASTE EVAPORATION SYSTEM

A 2300 liter/hr waste evaporator has been completed and is scheduled for operation late in 1965. The evaporator was designed to cencentrate (by a factor of 10) wastes that contain 0.3 M $NaNO_3$, 0.2 M NaOH, and a total activity of about 3 μc/ml. More recently, total solids in the waste have been reduced by a factor of about 10; consequently, a volume reduction greater than 10 may be expected. A decontamination factor of 10^6 to 10^7 is antecipated in the evaporation system. The evaporator will be capable of concentrating higher-level waste to an activity level of 740 curies/liter [16].

The principal units of the evaporation system (Figure 5) include: (1) a 1700-liter feed tank equipped with external cooling coils; (2) the 2300 liter/hr evaporator equipped with an internal de-entrainer and expected to give a decontamination factor of about 10^4; (3) a vapor filter containing a bed of Yorkmesh (stainless steel wire, 0.011 cm in diameter) and expected to give a decontamination factor of about 10^3; a condenser that requires 1400 liters/min of cooling water at an inlet temperature of 130°F; and (5) a condensate catch tank (570-liter capacity) equipped with an automatic diversion valve activated by a continuous recording radiation monitor. Two water-cooled stainless steel storage tanks (0.19-million-liter capacity) are furnished in case higher-level acidic wastes are produced. A caustic scrubber (4.2 cubic meter/min capacity) and a scrub liquor tank (630-liter capacity) can be used when acidic wastes are evaporated. The cost of the evaporation system was 1.5 million dollars.

Operation will be semi-continuous with feed material jetted to the evaporator during boildown. The contents of the evaporator will be cooled and then jetted to a concrete storage tank. Condensates will be discharged to the process waste system or returned for additional cleanup. Eventually, permanent disposal of evaporator concentrates well below the water table may be accomplished by use of the hydrofracturing plant. The feasibility of disposal in bedded shale by hydraulic fracturing has been demonstrated in a series of successful injections of actual waste. Plants are understudy to upgrade the design of the pilot plant.

WASTE TREATMENT FACILITIES AT BNL

Brookhaven National Laboratory has a variety of nuclear research and development facilities that may produce small quantities of radioactive waste. Most wastes are attributed to operation of reactors, decontamination facilities, accelerators, and the hot laboratory. Waste treatment requirements are closely associated with AEC limitations imposed on waste releases to the environment. The concentration of radioactivity in the liquid waste effluents leaving the BNL site must not exceed the limits specified in AEC Manual Chapter 0524, Standards for Radiation Protection. The total activity that can be discharged from the BNL site per year depends on the isotopic composition and volume of the effluent. If negligible alpha activity is present, no more than 20% of the activity is due to ^{90}Sr, and the remainder of the activity is not determined (the present situation); the limit is 4.4×10^{-10} curie/liter at the edge of the site. This amounts to a release of 0.5 curie/year for the present volume of 1.4×10^{9} liters/year. Only 0.1 curie/year of long-lived activity (half life greater than 5 years) can be buried locally as solid waste. These restrictions reflected the limited dilution available by surface waters and the extensive use of ground water as a source of water supply [17] [18].

LOW ACTIVITY LEVEL LIQUID WASTE SYSTEM

The laboratory wastes containing small quantities of radionuclides and the sanitary wastes are collected and flow togheter through common tile pipe to an Imhoff tank in the sewage treatment plant (Figure 6). Emphasis is placed on segregating waste of various levels of activity at or near the source. As a consequence each area in which liquid wastes are produced is provided with holdup tanks and provisions for monitoring. All wastes from drains at reactors, floor drains, and sinks are collected in monitoring tanks (size range, 5700 to 28,500 liters) in the Hot Laboratory area. Normally, these wastes are of low-activity level and are released to the sewerage system after analyses. Occasionally, such waste must be routed to three 0.38 million-liter mild steel tanks and then to the evaporator. Low-activity wastes originating in areas of higher-level operations (hot cells and dissolver units) flow through a diversion tank, are scanned by a GM tube assembly, and are automatically discharged to the appropriate system [19].

Low-level wastes entering the Imhoff tank are considerably more dilute than domestic sewage and have an average alkalinity of 100 mg/liter, and a total hardness of 44 mg/liter; the pH ranges from 6 to 10, suspended solids from 5 to 110 mg/liter, and BOD from 2.5 to 44 mg/liter. As shown in Table V, the volume of these wastes has increased by 60% in the past 5 years, while the concentration of radionuclides has decreased by an order of magnitude. The growth of the Laboratory and the extensive use of air conditioning account for the increase in volume to 2.9×10^{6} liters/day. A change in fuel enrichment in the Brookhaven Graphite Research Reactor (BGRR) (and the accompanying change to improved fuel element cladding) was about completed about 1962. A lower leakage rate of fission products from irradiated fuel elements stored in the canal and the effective use of the liquid waste monitoring system are primarily responsible for the decrease in activity concentration in low-level waste (1.0×10^{-10} curie/liter). Mixed fission products (^{137}Cs and ^{90}Sr) and ^{60}Co are the principal radionuclides in the waste [20] [21].

SEWAGE TREATMENT PLANT

The sewage treatment plant shown in Figure 7 was built by the military in 1942 and reactivated by BNL in 1947. It consists of an Imhoff tank for removal of suspended solids and a system of six intermittent sand filters, each about 1 acre in size [22]. The liquid from the Imhoff tank is automatically applied to the filter beds through open troughs by a 95,000-liter dosing tank. Each filter contains about 1 ft of sand and 5 ft of gravel. Most of the liquid passing through the filters is picked up by a tile underdrain system and is released to the river; however, during periods of low rainfall as much as 30% of the liquid is lost to the soil. Proportional samples are collected before and after filtration, and flow is measured by recording meters.

Laboratory and field studies indicated an initial retention of about 90% of the ^{131}I, ^{32}P, ^{90}Sr, and mixed fission products applied to sand filters. About 50% of the radionuclides in BGRR canal water was removed. A gradual reduction in efficiency occurred as more wastes were applied to the filters, and this reduction was believed to be due to washing down of activity that was loosely adsorbed in the bed. Plant experience has supported these findings. About 60% of the gross activity was removed by the filters during 1957-1958, and efficiency has gradually decreased to about 20% at the present time. Current plans call for complete renovation of the filters and replacing the sand and gravel.

Operating costs cannot be stated precisely at this time. The system has been particularly trouble free, requiring only occasional removal of weeds from the filter beds and sludge draw-off from the Imhoff tank.

INTERMEDIATE ACTIVITY LEVEL LIQUID WASTE SYSTEM

Liquid wastes that cannot be released to the sewerage system are routed to two hold-up tanks (6460-liter capacity each) in the Hot Laboratory area where caustic soda is added for neutralization. Hot wastes are also collected in 7.6-liter containers elsewhere in the Laboratory and are brought to this location and dumped into the proper tank. These intermediate-level wastes are eventually transferred to two 0.38-million-liter storage tanks; a separate 0.38-million-liter tank is reserved for storage of canal water. During the past 5 years the volume of intermediate-level waste has decreased from 3.0 million liters/year to 1.0 million liters/year. Gross activity now averages about 5×10^{-6} curies/liter, and ^{137}Cs, ^{90}Sr, and ^{60}Co are the principal radionuclides in the waste. The wastes contain an average of 3 mg/ml of total solids. The reduction in waste volume and in radionuclide concentration is attributed to the decreased rate of cladding failure of BGRR fuel elements.

WASTE EVAPORATOR SYSTEM

A 1100-liter/hr vapor compression evaporator, in operation since July 1953, is used to treat intermediate-level waste. The evaporator and entrainment separator were designed to concentrate (by a factor of 100) wastes that contain 0.2% dissolved solids and a gross activity as high as 5×10^{-4} curie/liter with a decontamination factor of 10^6 to 10^7. The principal units of the evaporation system (Figure 8) are of stainless steel construction and include: two 19,000-liter blending tanks, equipped with recirculating pumps; a duplex strainer; a double pipe heat exchanger; the 1100-liter/hr evaporator containing a calandria having 894 tubes, 7/8-in. diam x 16 gage; an entrainment separator containing a 100 micron spun copper filter packed to a bulk density of 15 lb per cu ft; and an axial flow steam compressor, which operates at a differential as high as 9 psig [23] [24].

Waste pumped from the storage tank is adjusted to a pH close to 7 in the blending tank. At this point antifoam agents (0 to 1.0 parts per thousand) and a water softener (72 mg/liter of Permutit for scale control) are added. Waste is then evaporated in 38,000-liter batches at the design rate. After passing through a duplex strainer, the waste is preheated to about 200°F in the heat exchanger before entering the evaporator. Research studies showed that the presently used spun-copper filters in the entrainment separator were mechanically more durable than Fiberglass filters at all evaporation rates. De-entrained steam passes through the vapor compressor, then to the evaporator calandria (providing most of the heat to boil the feed), and then through the heat exchanger before it is monitored and released to the sewerage system. Heat economy has been measured at 17.3 lb of distillate per 1000 Btu input. Routinely, a volume reduction of 85-100 and a decontamination factor of 10^4 - 10^6 are obtained. The original cost of the evaporation system (including design) was $267,000. Operating costs (excluding overhead charges), plus depreciation of capital equipment, average $0.03/liter.

SOLID WASTE TREATMENT

Solid wastes may be produced wherever experimenters are working with reactors, particle accelerators, and radioactive materials. A variety of objects have become contaminated with fission products, naturally occuring radionuclides, and induced activities. Liquid concentrates from the evaporator are also converted to solids.

By segregating solid wastes at the source, it is possible to predetermine which wastes require off-site disposal. Originally, solid wastes too radioactive for local disposal were packaged in 210-liter drums. Evaporator concentrate was mixed with cement in a batch-type mixer and poured into the drums to harden. The cost of packaging (labor and materials) amounted to $20/container, and the total cost of packaging and disposal was $37/container. Use of these small drums became impractical in 1959 when the quantity of waste and associated external radiation levels increased. Reinforced concrete vaults (normally 1.5 m x 1.5 m x 1.9 m) that may weigh up to 10 tons when filled are now used. Wall thickness is normally 15 cm, but may be increased to 43 cm for special gamma shielding. Waste Slurry from the evaporator is solidified in the 15-cm wall concrete vaults by the addition of exfoliated vermiculite (2.7 cu m/vault) and Portland cement (0.68 cu m/vault). Vermiculite is used for its absorptive capacity, and the cement provides the necessary strength; this mixture has less expansion than cement alone, thus eliminating hair-line cracks in the vaults. The batch mixer is no longer used, since adequate mixing of slurry with vermiculite and cement is obtained by introducing the slurry into the partially filled vault through a perforated pipe. The cost of packing each vault is $375, and the total cost of disposal is $875.

Solid wastes are concentrated by a number of techniques. This program has resulted in the reduction of both the quantity of waste that requires disposal and the total cost of processing and disposal (Table VI) [25] [26]. A power-operated baler is used to reduce the volume of easily compressible materials into rectangular blocks, 66 cm x 66 cm x 76 cm. Volume reductions range from about 5:1 to as great as 10:1. In 1960 a decontamination and reclamation facility was completed at a cost of $80,000. Operations that include decontamination by use of detergents, steam, grinding, acids, etc., and volume reduction by acetylene cutting, sawing, or disassembly are carried out in a 2400-sq-ft room, equipped with a bridge crane. In order to clean large bulky items with rough surfaces such as tanks, fan housings, fan impellers, etc., a shot blasting facility (using No. G50 steel shot) was installed as part of the decontamination facility in 1962. The cost of this addition was $70,000. To reduce personnel exposures involved in cutting and dismantling large grossly contaminated items, flexible linear-shaped charges (Petin, manufactured by E. I. du Pont de Nemours and Company, Incorporated) are being used to cut steel plate up to 0.65 cm thick. Essentially no spread of contamination has resulted from the use of shaped charges. The value of the decontamination and reclamation program is evidenced by the reduced volume of waste and lower processing and disposal costs incurred during 1964.

EFFECTIVENESS OF WASTE MANAGEMENT PROGRAMS

One measure of the effectiveness of waste management at ORNL and BNL is the concentration of activity released to the uncontrolled environment. At ORNL, the concentration of ^{90}Sr (the critical radionuclide) in the liquid effluents released to the Clinch River has been reduced to 4.5×10^{-10} curie/liter; that is, to about 1/9 of the continuous occupational limits recommended by ICRP. The concentration in the river above the nearest water intake has been further reduced by dilution to about 1×10^{-12} curies/liter. At BNL, the concentration of activity in liquid effluents released to the headwaters of the Peconic River has been reduced to an average of 1×10^{-10} curie/liter, which is about 1/4 of the acceptable limit.

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Table 1. Characteristics of Process Waste

Period	Waste Generated	Waste Treated		Radionuclides in Waste Treated (curies/liter)		
	(liters/day)	(liters/day)	(liters/min)	$^{89+90}Sr$	^{137}Cs	TRE
	x 10^6	x 10^6		x 10^{-7}	x 10^{-7}	x 10^{-7}
Sept. 1957 to Sept. 1959	2.9	1.5	1020	1.5	1.7	1.8
1960	1.9	1.7	1160	1.5	8.3	5.2
1961	1.4	1.3	900	0.51	0.11	0.53
1962	1.7	1.6	1150	0.34	0.23	0.15
1963	2.3	1.8	1250	0.27	0.098	----
1964	2.3	1.9	1320	0.37	0.15	----
Jan. 1965 to Aug. 1965	1.7	1.4	990	0.22	0.097	----

TABLE 2

AVERAGE PERCENTAGE REMOVAL OF RADIONUCLIDES BY TREATMENT SYSTEM

Period	$^{89+90}$Sr	TRE	^{137}Cs	^{106}Ru	^{60}Co	Gross Beta Activity
Sept. 1957 - Feb. 1958	77	79	--	--	--	--
Mar. 1958 - Aug. 1959	84	86	--	--	--	--
Sept. 1957 - Sept. 1958	--	--	21	49	35	66
Oct. 1958 - Aug. 1959	--	--	86	76	78	88
Jan. 1960 - June 1961	72	81	31	72	42	--
July 1961 - Dec. 1964	74	71[a]	75	75	73	--
Jan. 1965 - Aug. 1965	86	--	81	b	67	88

[a] Average through December 1962.

[b] Infrequently detected in untreated waste.

TABLE 3

PROCESS WASTE WATER TREATMENT

PERCENT REMOVAL OF RADIONUCLIDES

	Normal Plant[1] Operation	Laboratory[2] (Response Surface)	Adjusted Plant[3] Operation
^{89,90}Sr	84	96	94
^{137}Cs	86	92	91

[1] Lime: Stoichiometric; Soda Ash: 200 ppm Excess; Clay: 200 ppm.

[2] Lime: 2.5 x Stoichiometric; Soda Ash: 520 ppm Excess; Clay 360 ppm.

[3] Same as 2.

TABLE 4

PROCESS WASTE WATER TREATMENT

PERCENT REMOVAL OF RADIONUCLIDES

	Average Plant Operation[1] (6 Weeks Prior to Test)	Adjusted Plant[2] Operation
Gross Beta	96	95
^{89,90}Sr	65	33
^{137}Cs	81	92

[1] Lime: Stoichiometric; Soda Ash: 200 ppm Excess; Clay 200 ppm.

[2] Same as 1, with addition of Coagulant aids; Hagan 50 and Hagan 18.

Table 5. Characteristics of Liquid Wastes at BNL

Period	Low-Level Wastes		Intermediate-Level Waste	
	Volume Treated (liters/day)	Activity (curies/liter)	Volume Treated (liters/year)	Activity (curies/liter)
	$x\ 10^{6}$	$x\ 10^{-9}$	$x\ 10^{6}$	$x\ 10^{-4}$
1960	1.8	1.5	3.0	2.3
1961	2.1	0.53	2.5	0.92
1962	2.5	0.15	1.6	0.93
1963	3.0	0.12	1.1	0.038
1964	2.9	0.10	1.0	0.05

Table 6. Solid Waste Processed at BNL for Off-Site Shipment

Year	Number and Type of Container	Total Volume (cubic meters)	Total (curies)	Total Cost of Processing and Disposal ($1000)
1958	692 (210-liter drums)	140	1,100	26
1959	1,354 (210-liter drums) 20 (concrete vaults)	330	2,500	68
1960	92 (concrete vaults)	210	2,000	81
1961	60 (concrete vaults)	140	1,300	53
1962	55 (concrete vaults)	130	1,200	48
1963	45 (concrete vaults)	100	980	39
1964	38 (concrete vaults)	86	75	33

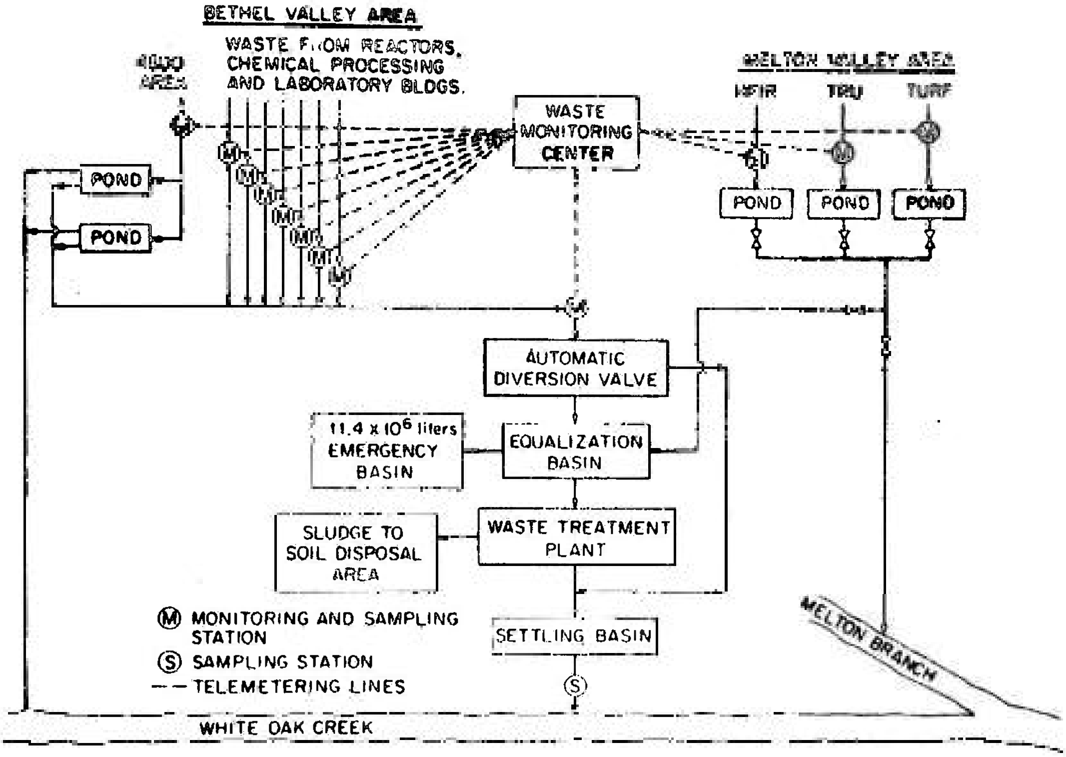


Fig. 1. Process Waste System.

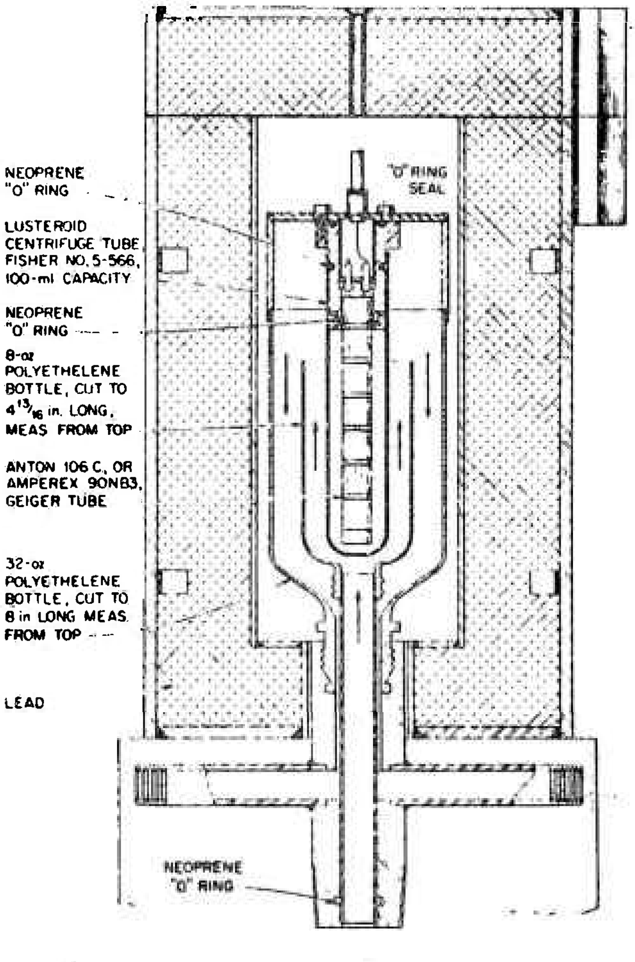
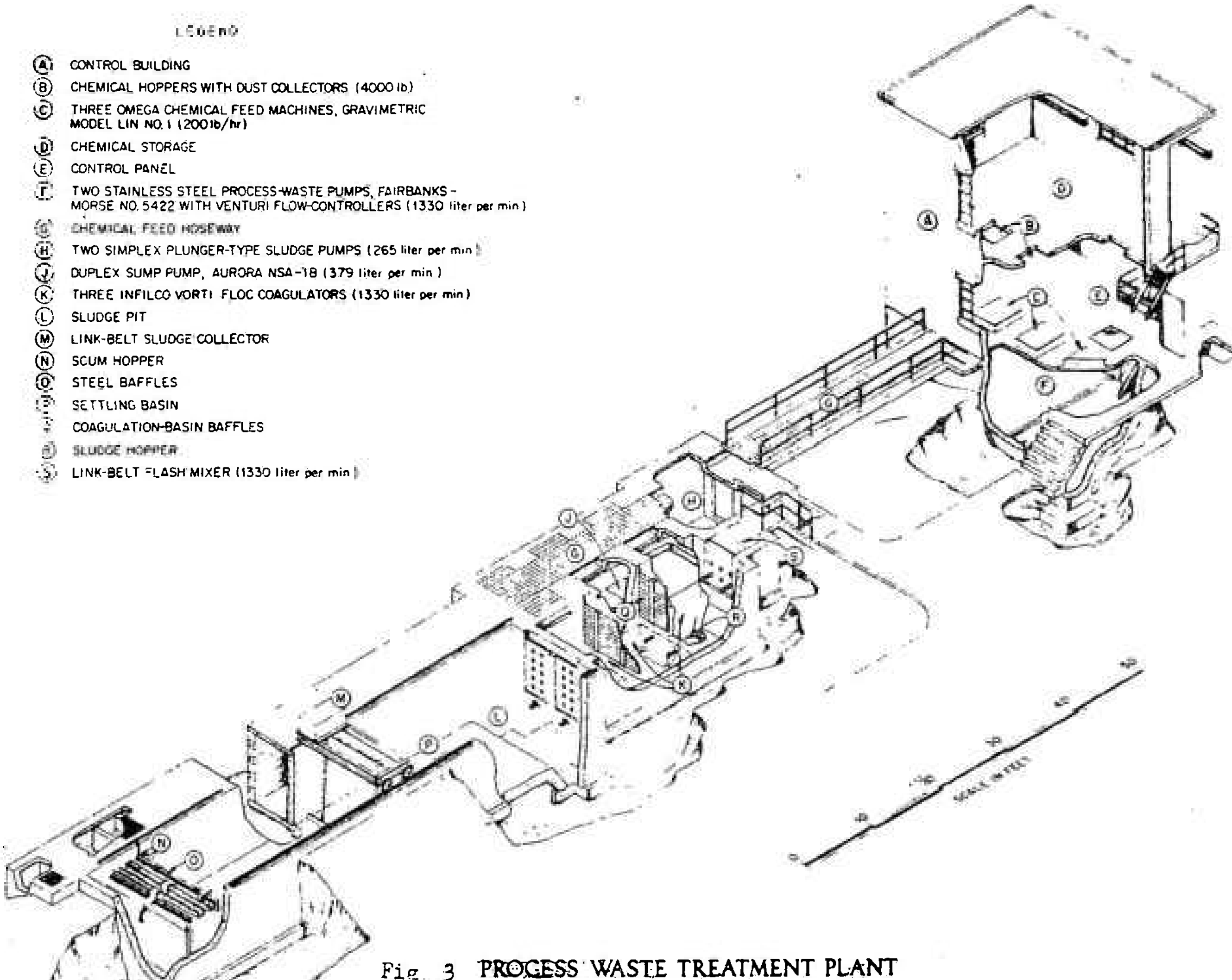


Fig. 2. Waste Water Monitor Detector for Beta and Gamma Radiation Assembly.



(A)	CONTROL BUILDING
(B)	CHEMICAL HOPPERS WITH DUST COLLECTORS (4000 lb)
(C)	THREE OMEGA CHEMICAL FEED MACHINES, GRAVIMETRIC MODEL LIN NO. 1 (200 lb/hr)
(D)	CHEMICAL STORAGE
(E)	CONTROL PANEL
(F)	TWO STAINLESS STEEL PROCESS-WASTE PUMPS, FAIRBANKS-MORSE NO. 5422 WITH VENTURI FLOW-CONTROLLERS (1330 liter per min)
(G)	CHEMICAL FEED HOSEWAY
(H)	TWO SIMPLEX PLUNGER-TYPE SLUDGE PUMPS (265 liter per min)
(J)	DUPLEX SUMP PUMP, AURORA NSA-18 (379 liter per min)
(K)	THREE INFILCO VORTI FLOC COAGULATORS (1330 liter per min)
(L)	SLUDGE PIT
(M)	LINK-BELT SLUDGE COLLECTOR
(N)	SCUM HOPPER
(O)	STEEL BAFFLES
(P)	SETTLING BASIN
(Q)	COAGULATION-BASIN BAFFLES
(R)	SLUDGE HOPPER
(S)	LINK-BELT FLASH MIXER (1330 liter per min)

Fig. 3 PROCESS WASTE TREATMENT PLANT

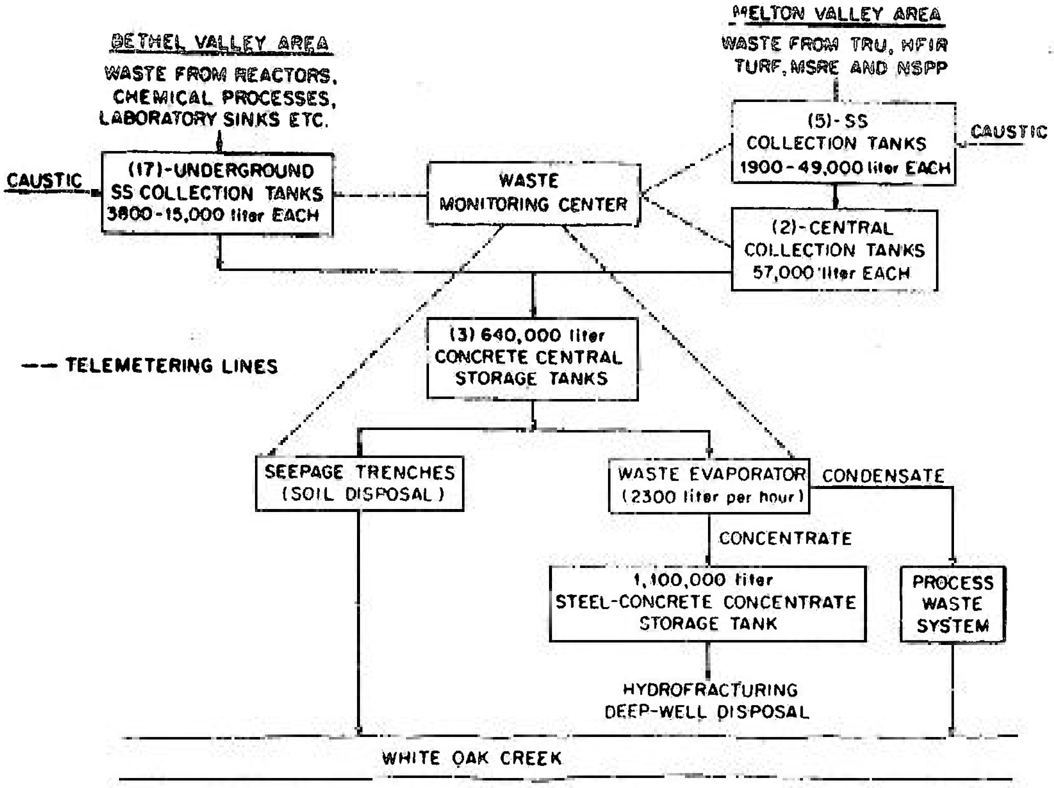


Fig. 4. Intermediate Level Waste System.

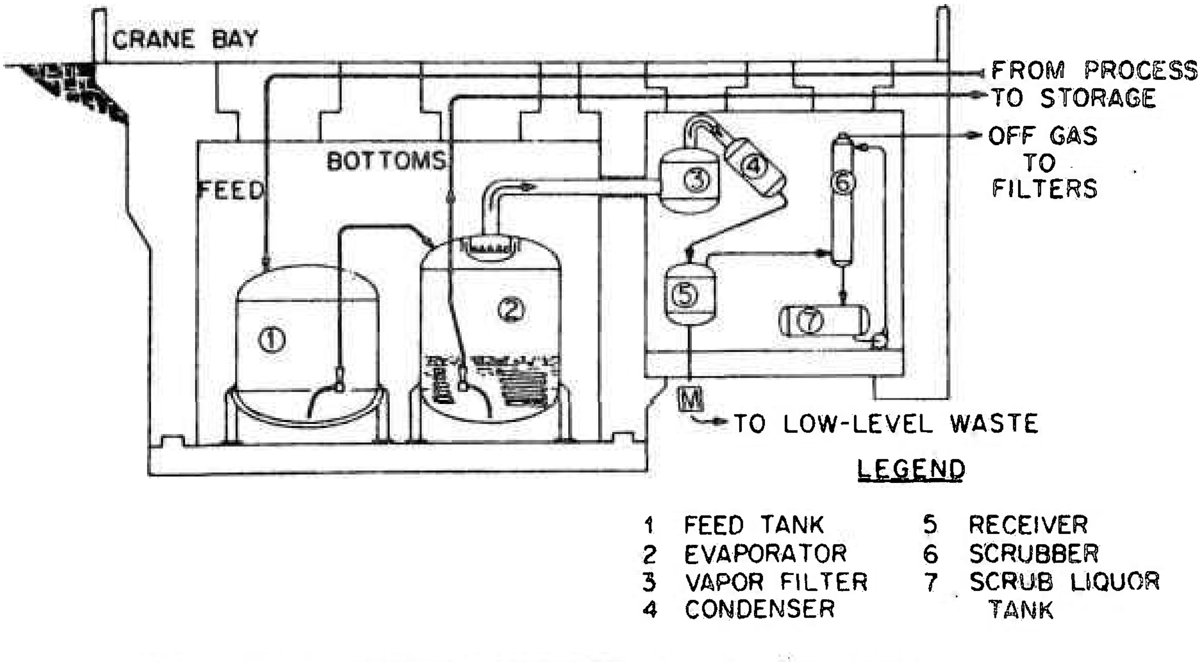


Fig. 5. ORNL INTERMEDIATE-LEVEL WASTE EVAPORATOR

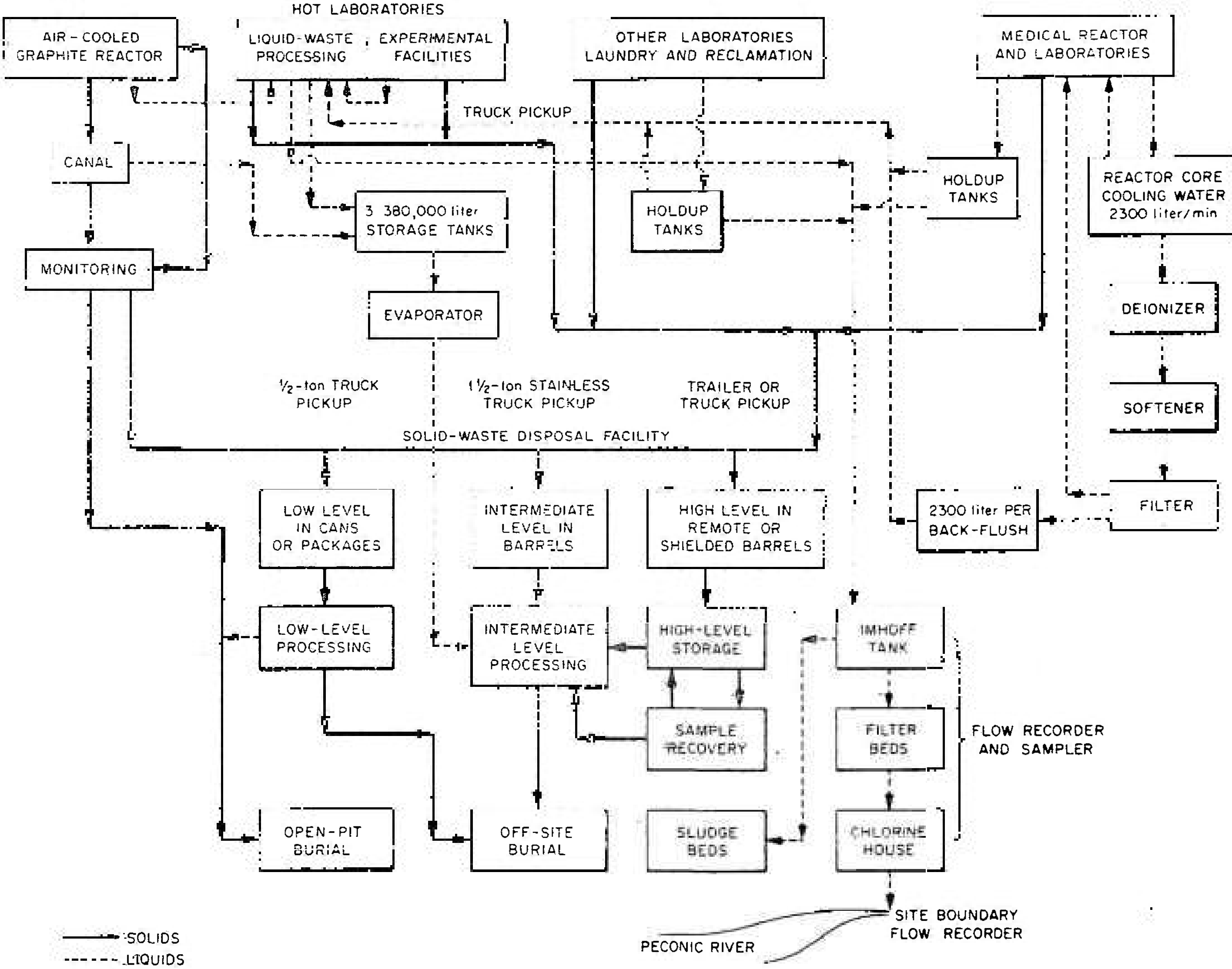


Fig. 6. Simplified BNL Radioactive Waste System

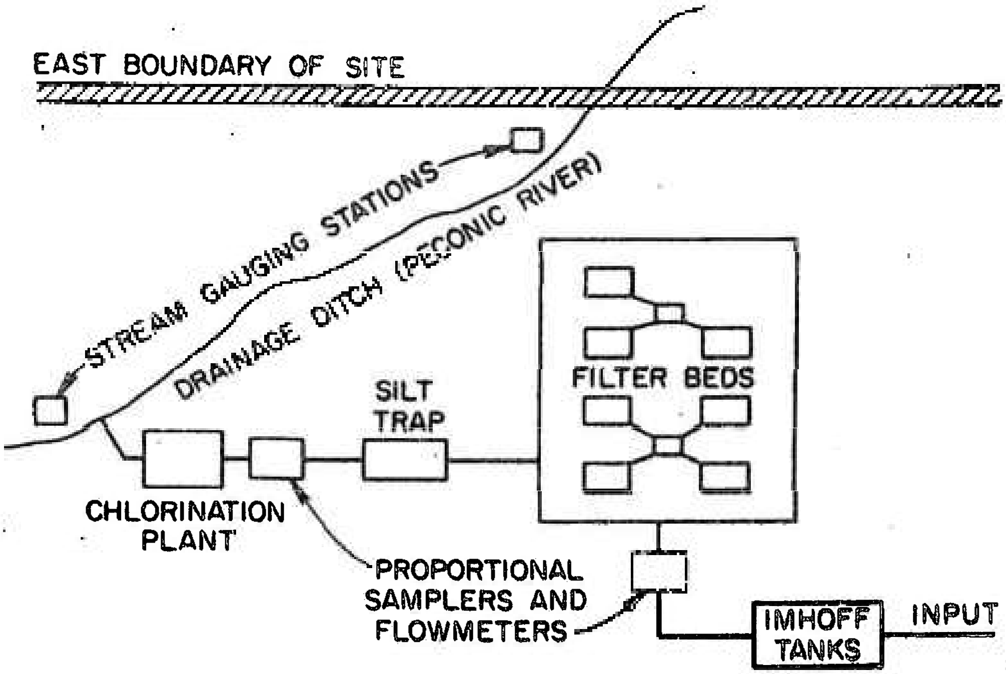


Fig. 7. BNL Sewage Processing and Monitoring System.

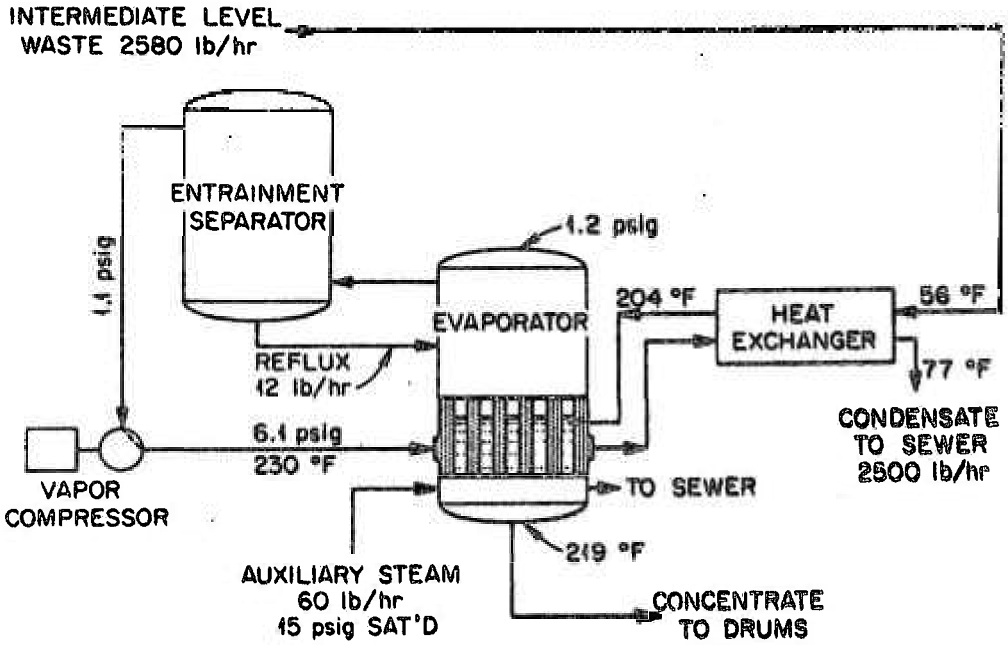


Fig. 8. BNL Vapor Compression Evaporator

ORNL Scavenging Precipitation Ion Exchange Process for Decontamination of Low-Level Liquid Wastes

SLIDE 1 Constituents of Feed to Pilot Plant (64-8476)

The remaining slides show the details of a new process developed at Oak Ridge National Laboratory which achieves high decontamination with low operating costs.

SLIDE 2 Scavenging-Precipitation Ion-Exchange Process for Decontamination of Low-Level Radioactive Waste (64-7501 A)

Describe process (1) alumina to remove phosphate, (2) scavenging-precipitation at pH 11.8 to remove colloids and 60-80% of radionuclides, (3) settle in clarifier, (4) pass through a phenolic-carboxylic ion-exchange resin to remove radionuclides, and (5) regenerate resin with HNO_3 and recycle to the head-end stream. All radionuclides are then removed in the solid sludge which is incorporated in asphalt. The purified water meets drinking water standards.

SLIDE 3 Results of Demonstration Runs Using Modified Flowsheet (64-8477)

SLIDE 4 Low-Level Waste Pilot Plant (Photo 68999R)

SLIDE 5 Low-Level Waste Treatment Pilot Plant – Ion Exchange Columns (Photo 56219)

SLIDE 6 Estimated Cost for the Recycle Scavenging-Precipitation Ion-Exchange Process (65-12228)

The unit cost is about $0.80/1000 gal in a large plant (700,000 gal/day) and $4.52/1000 gal in a small plant (15,000 gal/day).

REFERENCES AND EXCERPTS FROM:

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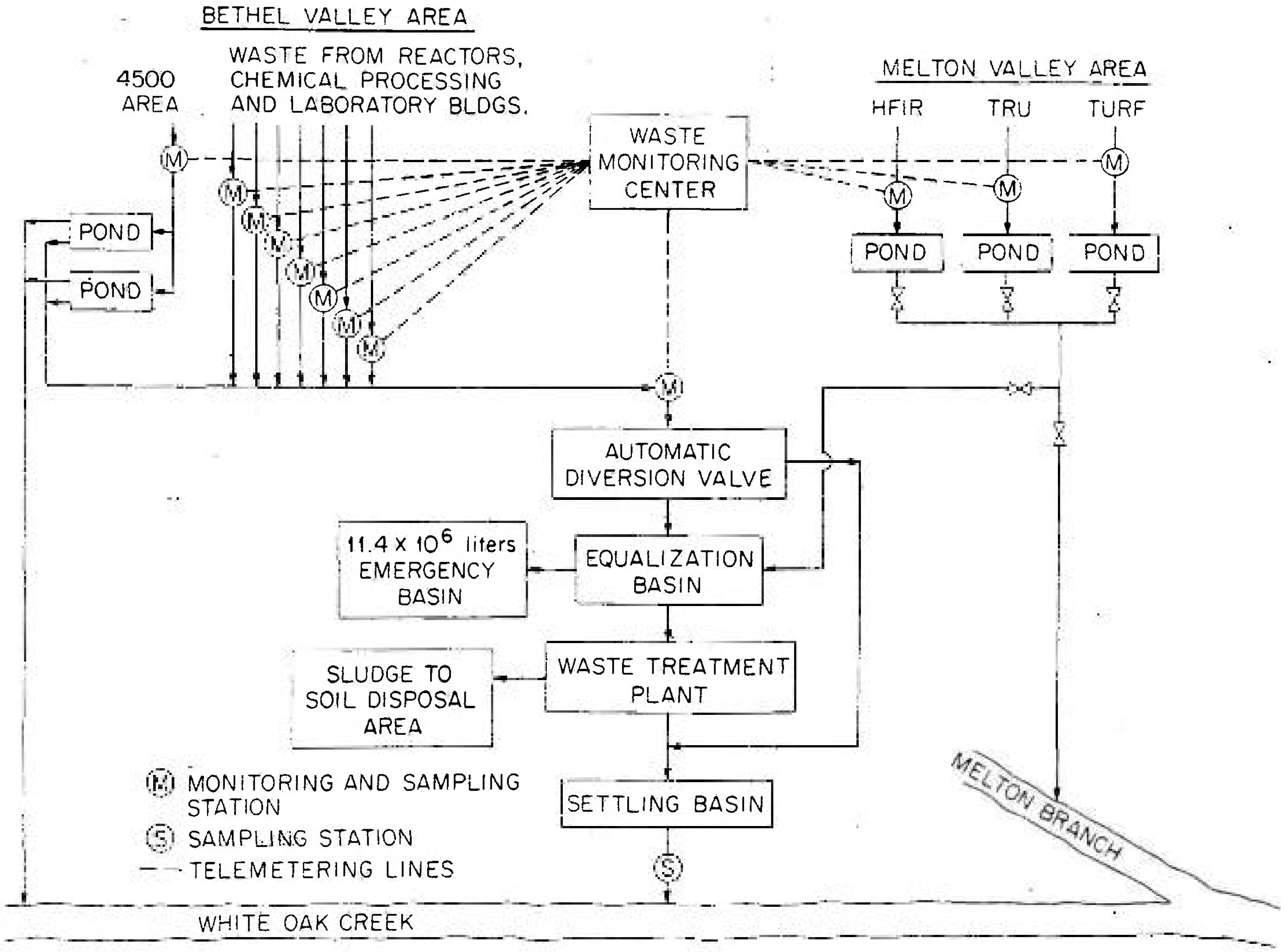
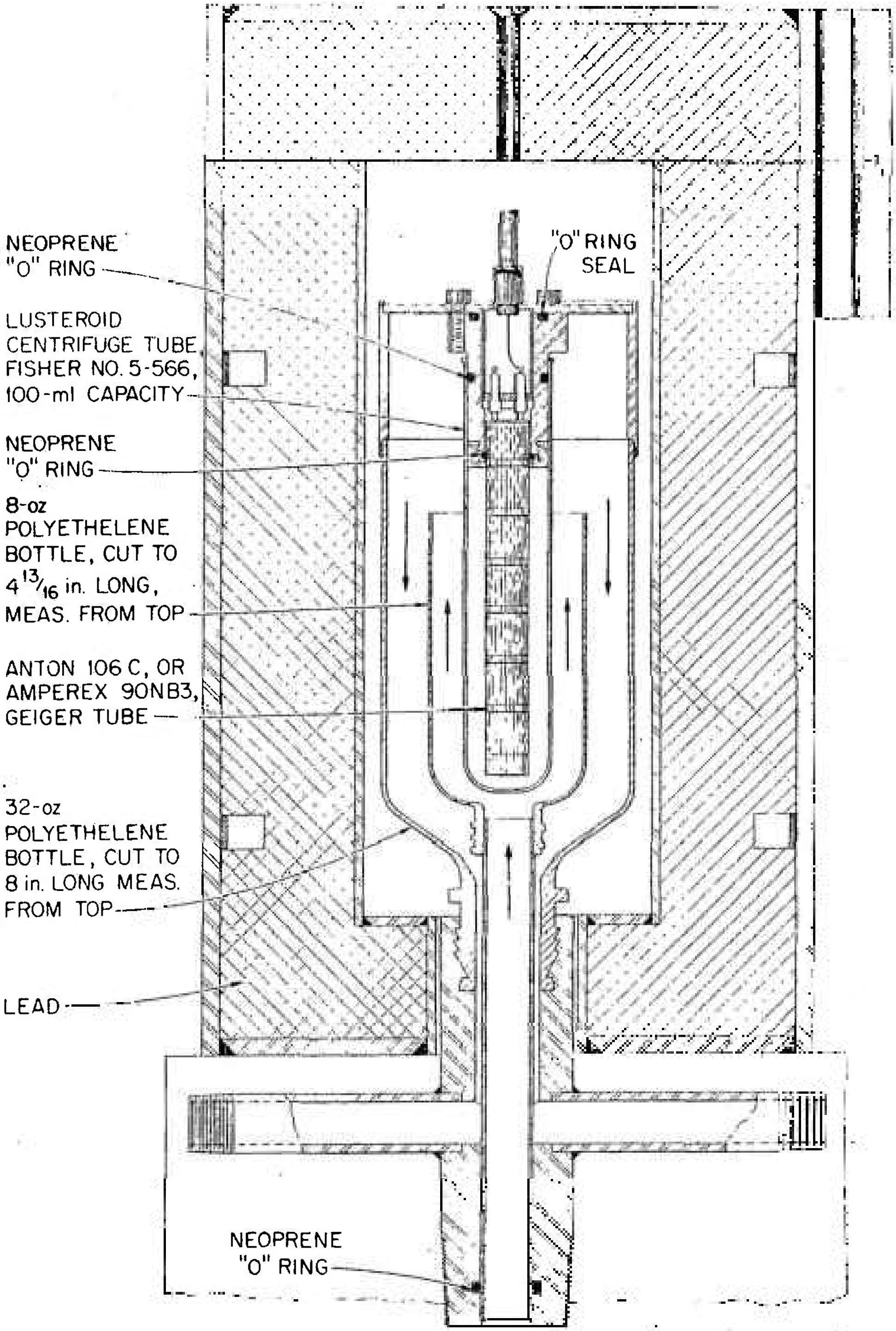
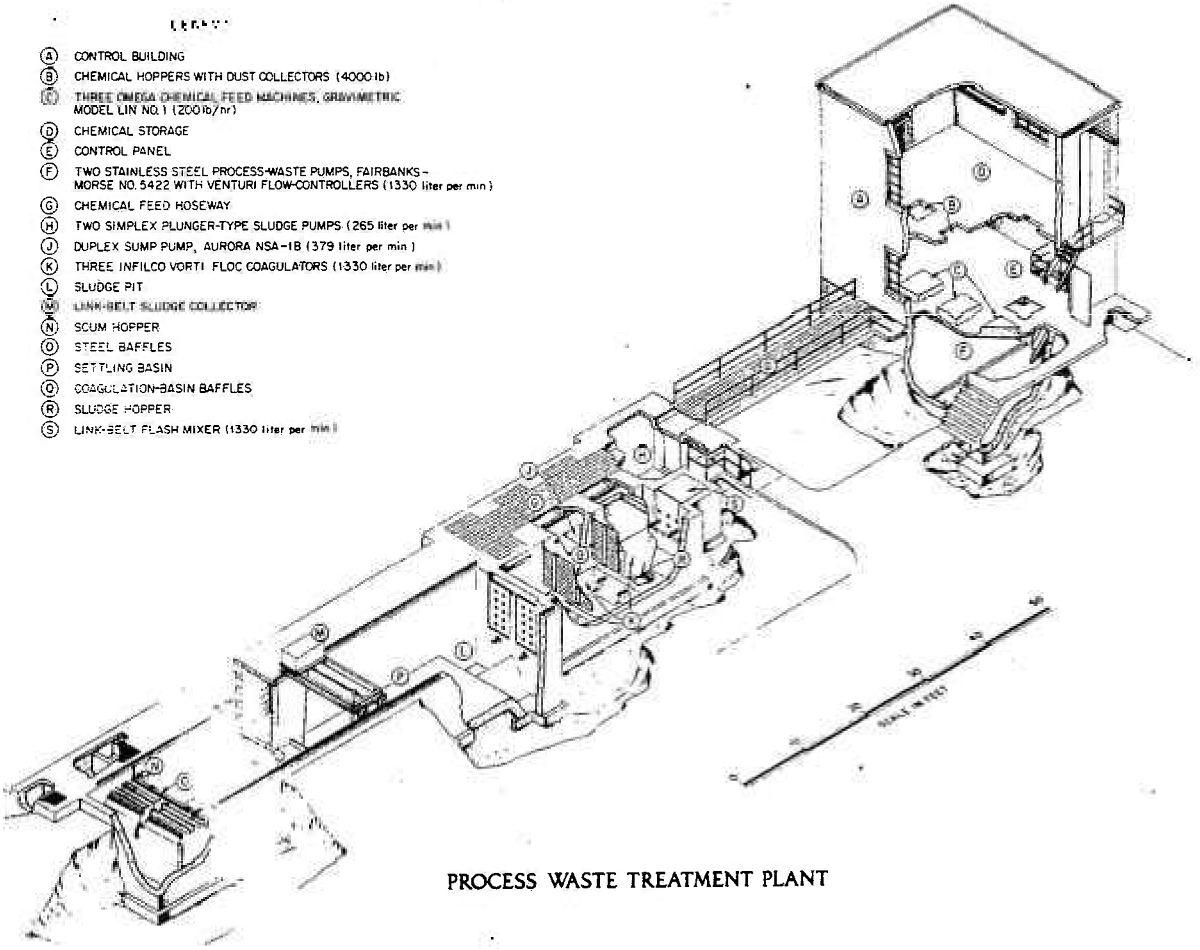


Process Waste System.


NEOPRENE "O" RING
LUSTEROID CENTRIFUGE TUBE, FISHER NO. 5-566, 100-ml CAPACITY
NEOPRENE "O" RING
8-oz POLYETHELENE BOTTLE, CUT TO 4 13/16 in. LONG, MEAS. FROM TOP
ANTON 106 C, OR AMPEREX 90NB3, GEIGER TUBE
32-oz POLYETHELENE BOTTLE, CUT TO 8 in. LONG MEAS. FROM TOP
LEAD
"O" RING SEAL
NEOPRENE "O" RING

CHARACTERISTICS OF PROCESS WASTE

Period	Waste Generated (liters/day)	Waste Treated		Radionuclides in Waste Treated (curies/liter)		
		(liters/day)	(liters/min)	$^{89+90}Sr$	^{137}Cs	TRE
	x 10^{6}	x 10^{6}		x 10^{-7}	x 10^{-7}	x 10^{-}
Sept. 1957 to Sept. 1959	2.9	1.5	1020	1.5	1.7	1.8
1960	1.9	1.7	1160	1.5	8.3	5.2
1961	1.4	1.3	900	0.51	0.11	0.53
1962	1.7	1.6	1150	0.34	0.23	0.15
1963	2.3	1.8	1250	0.27	0.098	--
1964	2.3	1.9	1320	0.37	0.15	--
Jan. 1965 to Aug. 1965	1.7	1.4	990	0.22	0.097	--



Ⓐ	CONTROL BUILDING
Ⓑ	CHEMICAL HOPPERS WITH DUST COLLECTORS (4000 lb)
Ⓒ	THREE OMEGA CHEMICAL FEED MACHINES, GRAVIMETRIC MODEL LIN NO. 1 (200 lb/hr)
Ⓓ	CHEMICAL STORAGE
Ⓔ	CONTROL PANEL
Ⓕ	TWO STAINLESS STEEL PROCESS-WASTE PUMPS, FAIRBANKS-MORSE NO. 5422 WITH VENTURI FLOW-CONTROLLERS (1330 liter per min)
Ⓖ	CHEMICAL FEED HOSEWAY
Ⓗ	TWO SIMPLEX PLUNGER-TYPE SLUDGE PUMPS (265 liter per min)
Ⓙ	DUPLEX SUMP PUMP, AURORA NSA-1B (379 liter per min)
Ⓚ	THREE INFILCO VORTI FLOC COAGULATORS (1330 liter per min)
Ⓛ	SLUDGE PIT
Ⓜ	LINK-BELT SLUDGE COLLECTOR
Ⓝ	SCUM HOPPER
Ⓞ	STEEL BAFFLES
Ⓟ	SETTLING BASIN
Ⓠ	COAGULATION-BASIN BAFFLES
Ⓡ	SLUDGE HOPPER
Ⓢ	LINK-BELT FLASH MIXER (1330 liter per min)

PROCESS WASTE TREATMENT PLANT

AVERAGE PERCENTAGE REMOVAL OF RADIONUCLIDES BY TREATMENT SYSTEM

Period	$^{89+90}$Sr	TRE	^{137}Cs	^{106}Ru	^{60}Co	Gross Beta Activity
Sept. 1957 - Feb. 1958	77	79	--	--	--	--
Mar. 1958 - Aug. 1959	84	86	--	--	--	--
Sept. 1957 - Sept. 1958	--	--	21	49	65	66
Oct. 1958 - Aug. 1959	--	--	86	76	78	88
Jan. 1960 - June 1961	72	81	81	72	42	--
July 1961 - Dec. 1964	74	71[a]	75	75	73	--
Jan. 1965 - Aug. 1965	86	--	81	b	67	88

[a] Average through December 1962.

[b] Infrequently detected in untreated waste.

PROCESS WASTE WATER TREATMENT

PERCENT REMOVAL OF RADIONUCLIDES

	Normal Plant[1] Operation	Laboratory[2] (Response Surface)	Adjusted Plant[3] Operation
$^{89,90}Sr$	84	96	94
^{137}Cs	86	92	91

[1] Lime: Stoichiometric; Soda Ash: 200 ppm Excess; Clay: 200 ppm.

[2] Lime: 2.5 x Stoichiometric; Soda Ash: 520 ppm Excess; Clay 360 ppm.

[3] Same as 2.

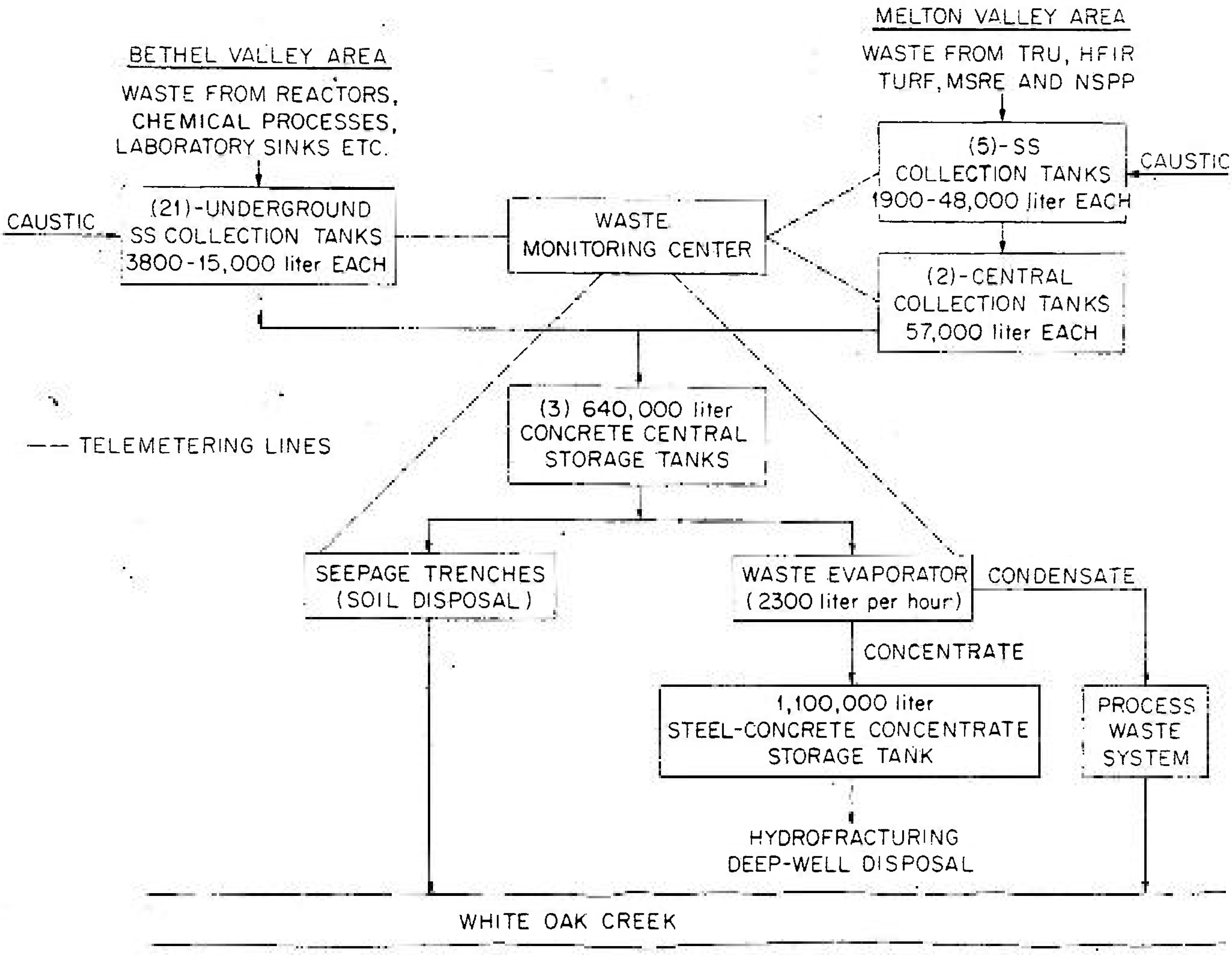
PROCESS WASTE WATER TREATMENT

PERCENT REMOVAL OF RADIONUCLIDES

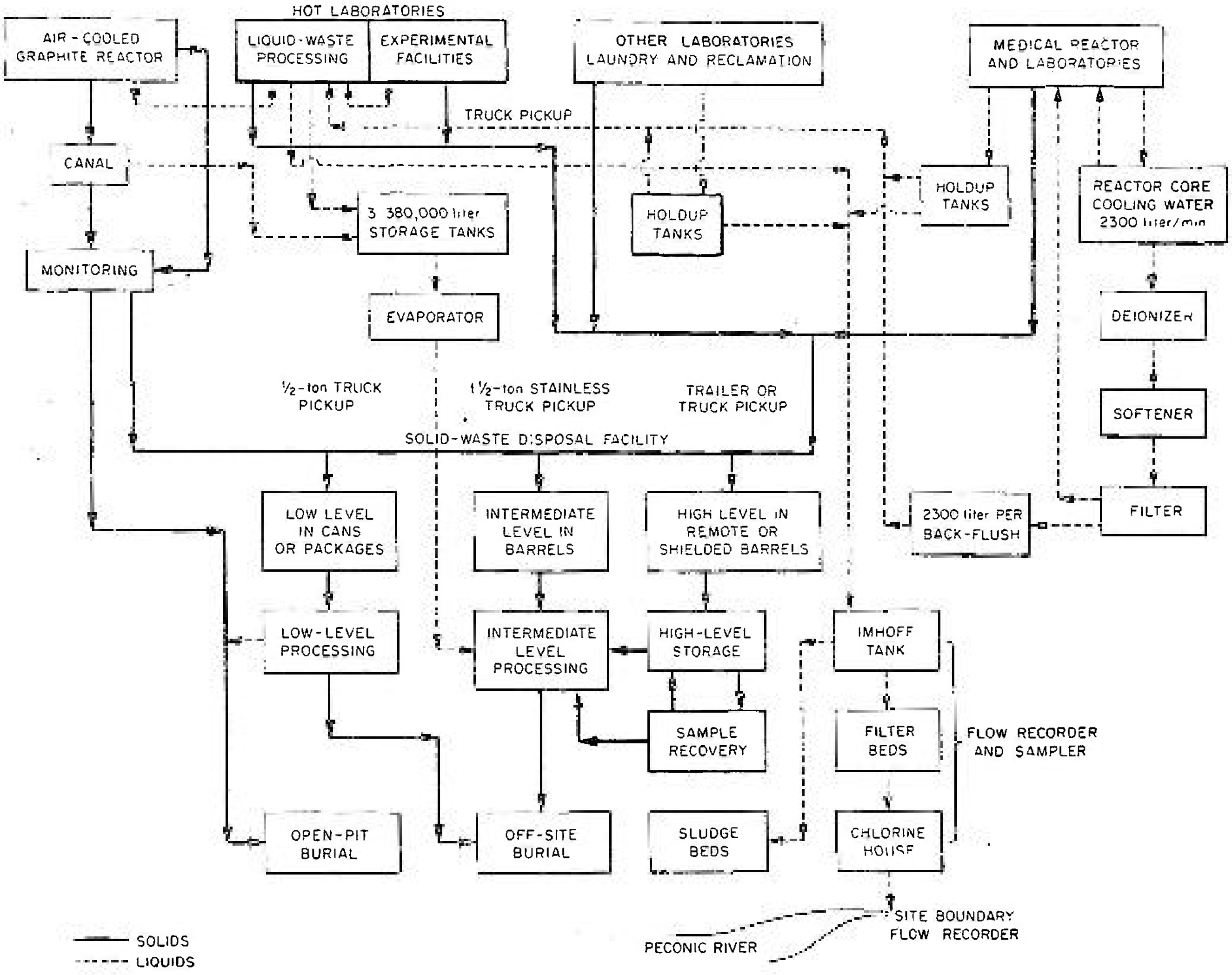
	Average Plant Operation[1] (6 Weeks Prior to Test)	Adjusted Plant[2] Operation
Gross Beta	86	95
$^{89,90}Sr$	65	86
^{137}Cs	81	93

[1] Lime: Stoichiometric; Soda Ash: 200 ppm Excess; Clay 200 ppm.

[2] Same as 1, with addition of Coagulant aids; Hagan 50 and Hagan 18.



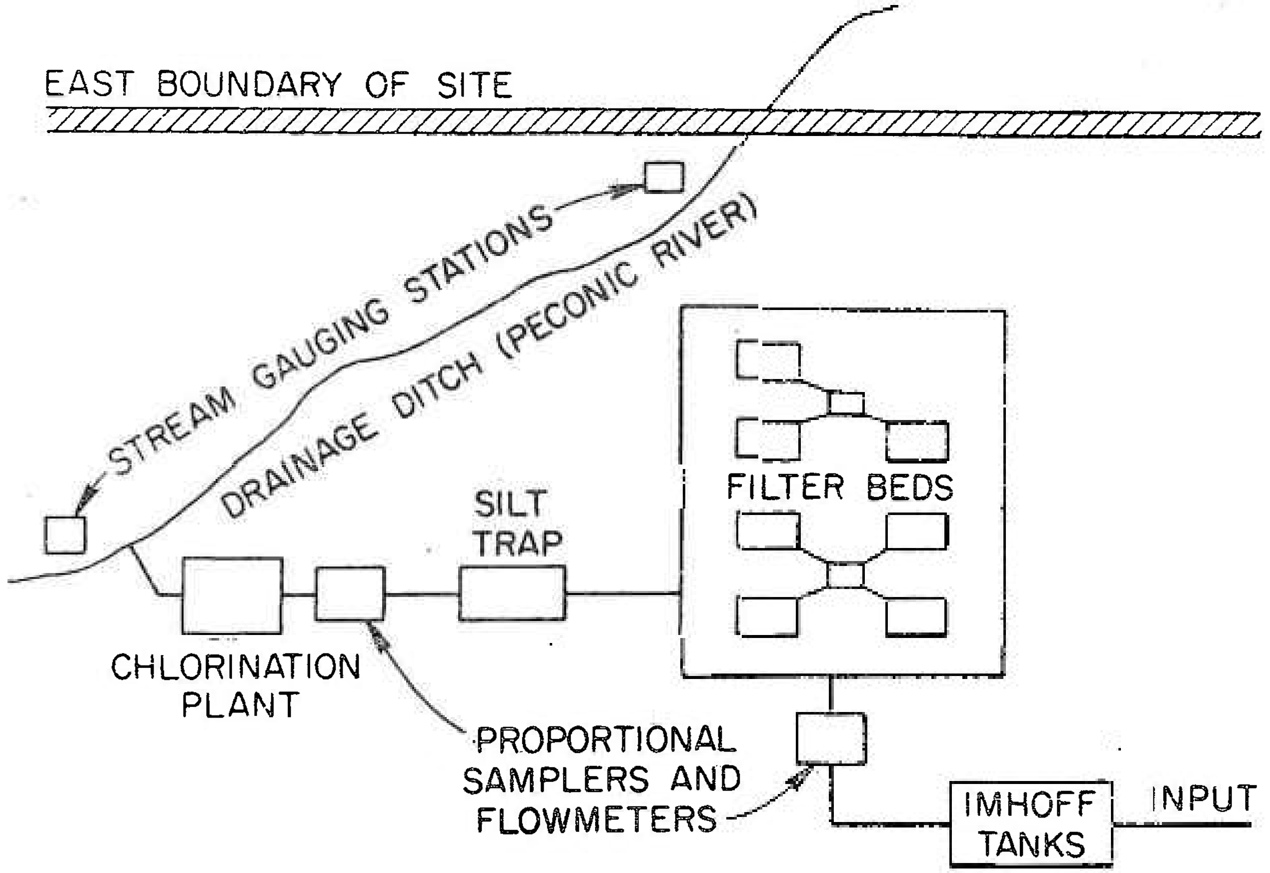
Intermediate Level Waste System.



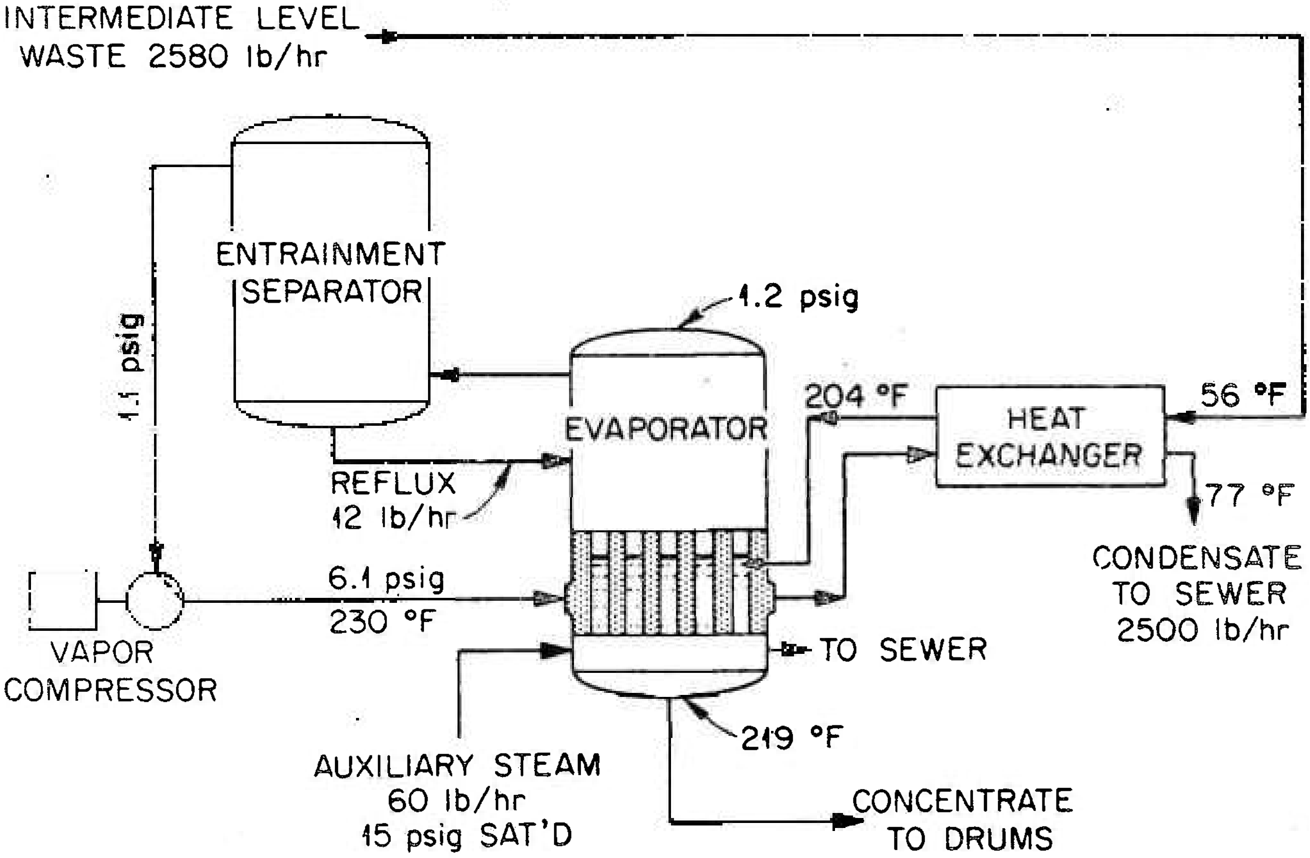
Simplified BNL Radioactive Waste System

CHARACTERISTICS OF LIQUID WASTES AT BNL

Period	Low-Level Wastes		Intermediate-Level Waste	
	Volume Treated (liters/day)	Activity (curies/liter)	Volume Treated (liters/year)	Activity (curies/liter
	x 10^6	x 10^{-9}	x 10^6	x 10^{-4}
1960	1.8	1.5	3.0	2.3
1961	2.1	0.53	2.5	0.92
1962	2.5	0.15	1.6	0.93
1963	3.0	0.12	1.1	0.038
1964	2.9	0.10	1.0	0.05



BNL Sewage Processing and Monitoring System.



BNL Vapor Compression Evaporator

SOLID WASTE PROCESSED AT BNL FOR OFF-SITE SHIPMENT

Year	Number and Type of Container	Total Volume (cubic meters)	Total (curies)	Total Cost of Processing and Disposal ($1000)
1958	692 (210-liter drums)	140	1,100	26
1959	1,354 (210-liter drums)			
	20 (concrete vaults)	330	2,500	68
1960	92 (concrete vaults)	210	2,000	81
1961	60 (concrete vaults)	140	1,300	53
1962	55 (concrete vaults)	130	1,200	48
1963	45 (concrete vaults)	100	980	39
1964	38 (concrete vaults)	86	75	33

CONSTITUENTS OF FEED TO PILOT PLANT

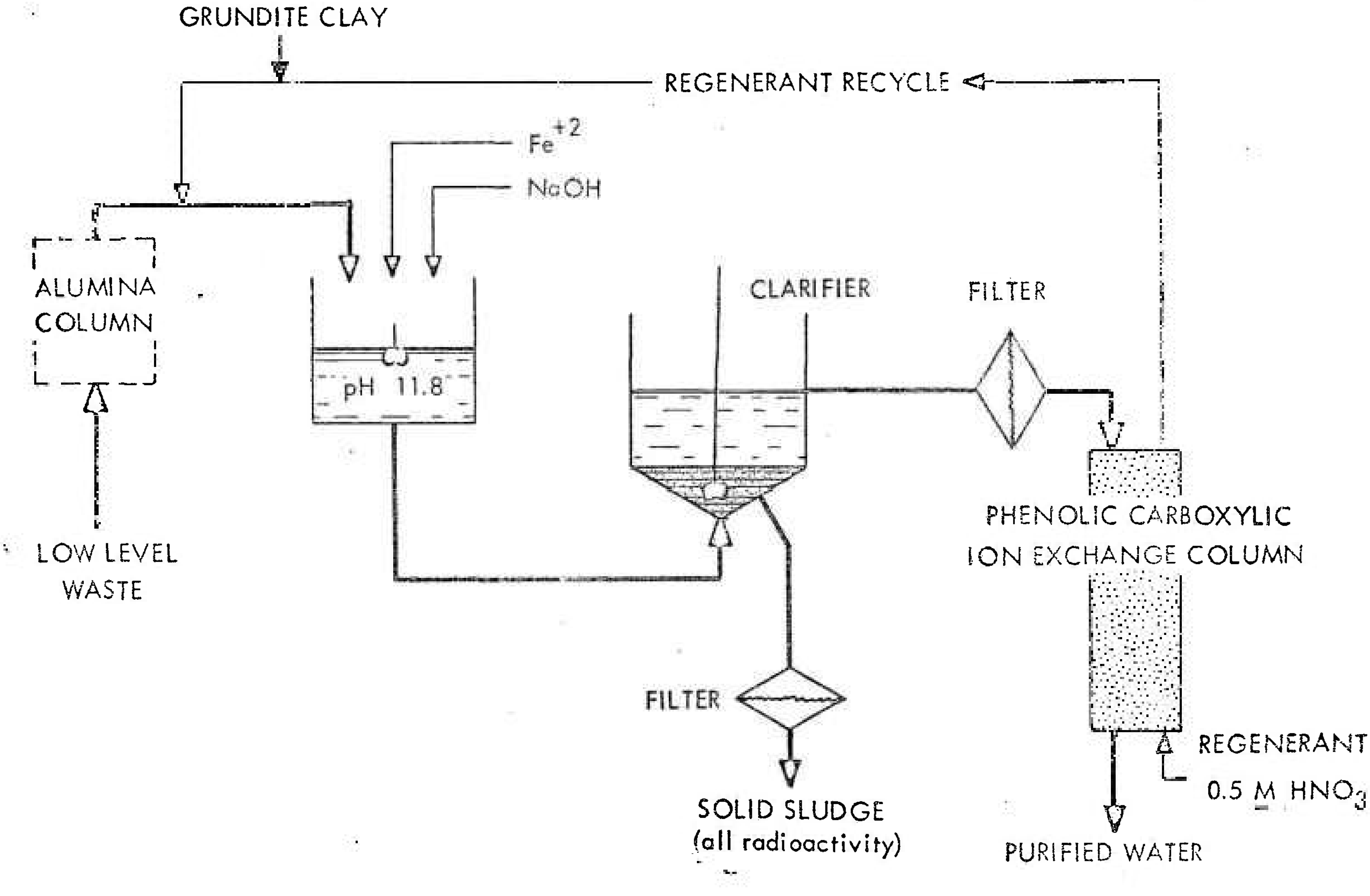
Nov. 1961 to July 1962

TOTAL HARDNESS (equiv. $CaCO_3$) 90-150 (115) ppm

pH 7-9 (7.7)

	RADIOACTIVITY dis min^{-1} ml^{-1}		MPC_w dis min^{-1} ml^{-1}
	RANGE	AVG.	
GROSS β	10-150[a]	36[a]	—
GROSS γ	8-730[a]	89[a]	—
^{90}Sr	25-200	86	2.2
^{137}Cs	10-275	54	440
^{60}Co	5-1600	137	1100
^{106}Ru	0-15	8	220
TRE	5-20[a]	9[a]	670

[a] COUNTS min^{-1} ml^{-1}



SCAVENGING-PRECIPITATION ION-EXCHANGE PROCESS FOR DECONTAMINATION OF LOW LEVEL RADIOACTIVE WASTE

RESULTS OF DEMONSTRATION RUNS USING MODIFIED FLOWSHEET

	DECONTAMINATION FACTOR	RADIOACTIVITY IN SLUDGE (% OF FEED)	RADIOACTIVITY IN EFFLUENT (% MPC_W)
GROSS β	17	61	—
GROSS γ	11	52	—
^{90}Sr	1400	73	1.4
^{137}Cs	1200	61	0.003
^{106}Ru	2.6 (14)[a]	—	0.07
^{60}Co	7 (11)[a]	—	0.05
TOTAL RARE EARTHS	580	79	0.003[b]

a. D. F. AT 300 BED VOLUMES USING ANION EXCHANGE.

b. MPC_W FOR ^{91}Y

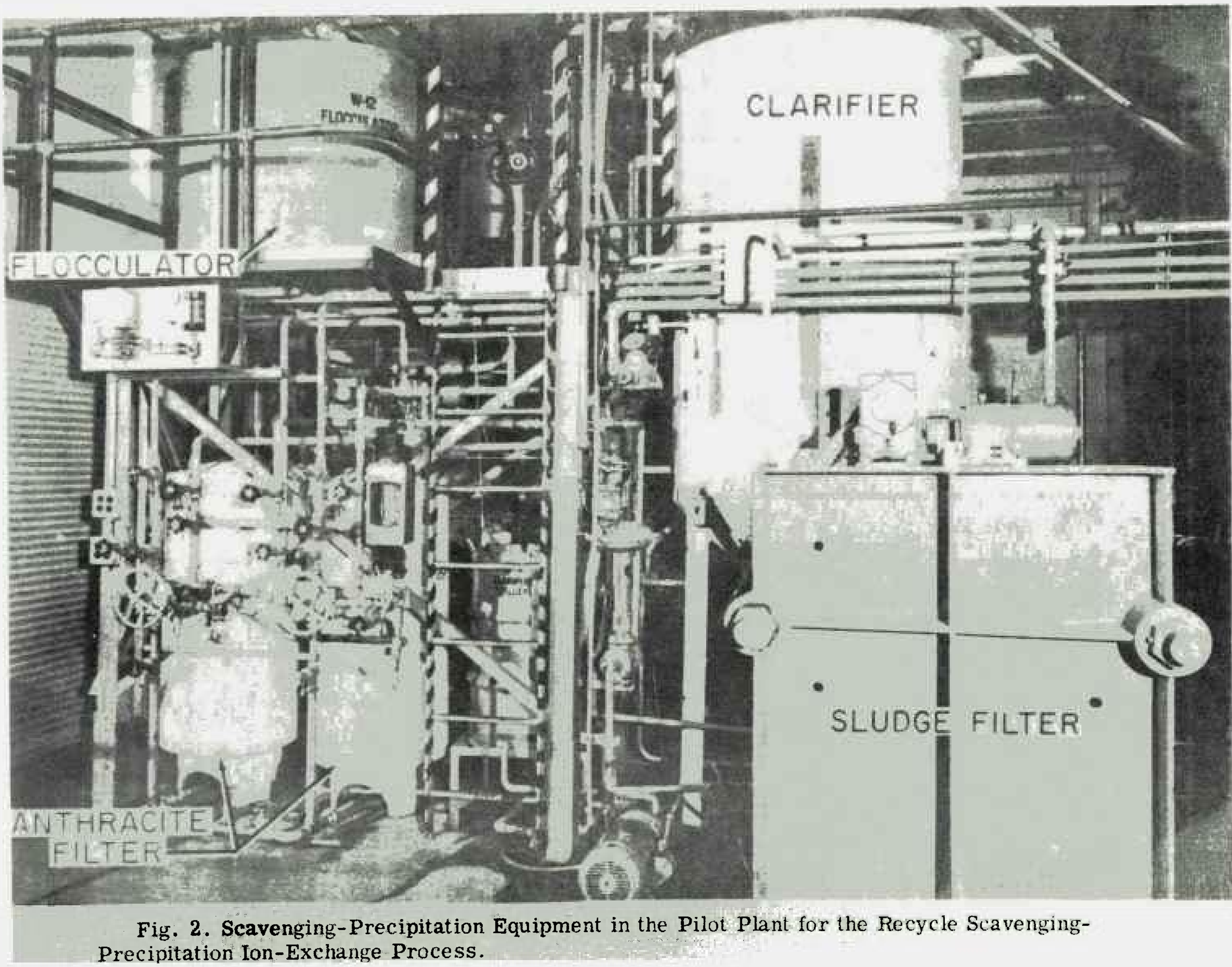
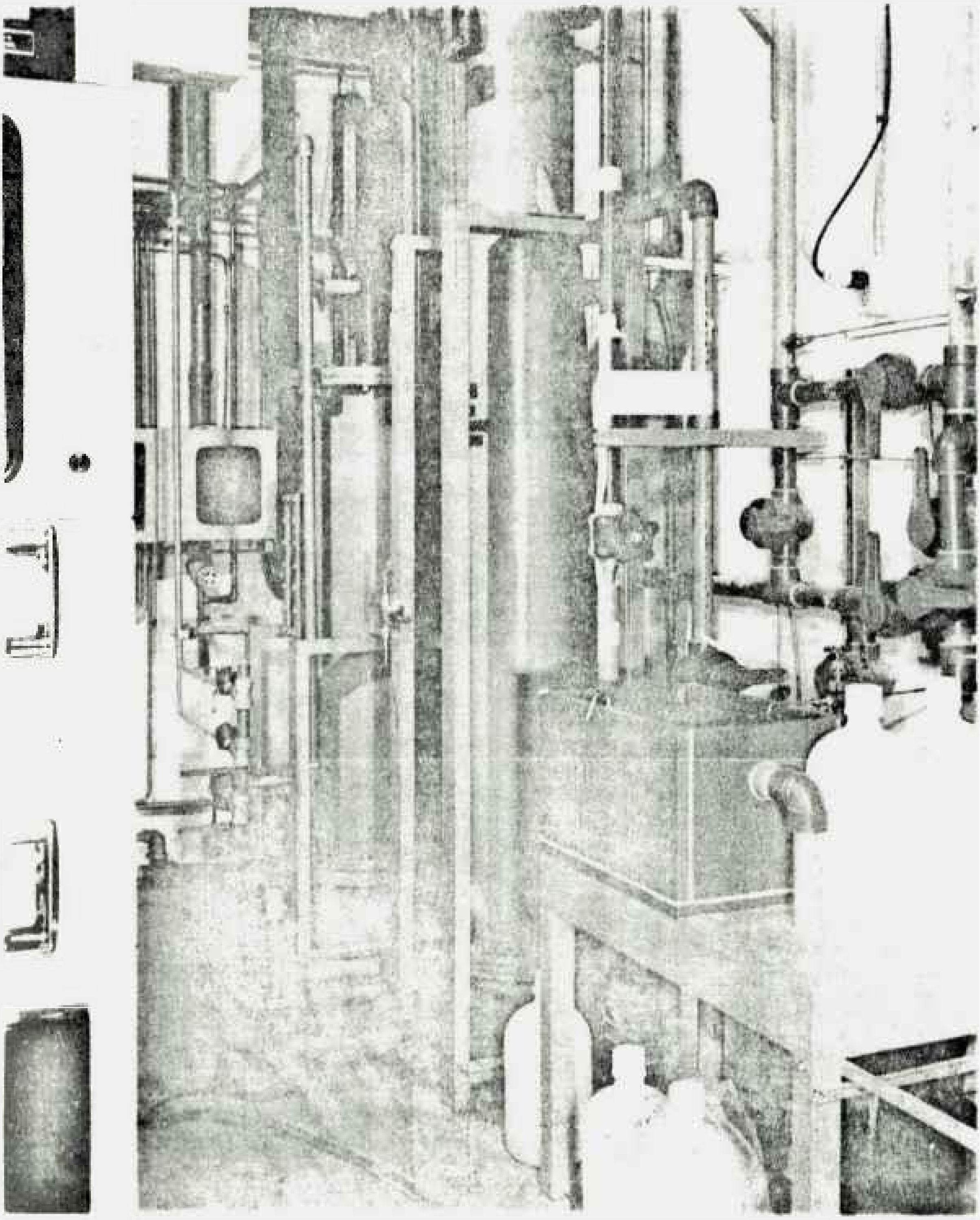


Fig. 2. Scavenging-Precipitation Equipment in the Pilot Plant for the Recycle Scavenging-Precipitation Ion-Exchange Process.

13-5

ORNL PHOTO 56219



ESTIMATED COST FOR THE RECYCLE SCAVENGING-PRECIPITATION ION-EXCHANGE PROCESS

(cents/1000 gal)

Capital Cost - $91,455

	750,000 gal/day	15,000 gal/day
Fixed Costs		
Utilities	1.0	1.0
Labor	17.28	205.0
Depreciation	12.88	123.0
Maintenance	8.57	83.5
	39.73	412.5
Other Costs*		
Chemicals		16.71
Resin Regeneration		4.48
Alumina Replacement (12¢/lb)		7.80
Alumina Burial (124¢/ft^3)		0.63
Sludge Burial (124¢/ft^3)		9.90
		39.52
Total		$4.52

*Same unit charge as for 750,000 gal/day plant.

TREATMENT OF WASTES FROM POWER REACTORS

R. E. Blanco

The treatment of wastes produced at nuclear power reactors is one of the first problems that a country must face when it decides to produce power from nuclear reactors. Significant amounts of gaseous and liquid wastes are produced and waste management requires specialized care. However, the numbers of curies produced are orders of magnitude lower than those produced at a fuel processing plant. Most nations can store the used fuel elements or have them reprocessed in other countries and need not face the problem of reprocessing fuel till much later.

Power reactors have been operated for a number of years in the U. S. and the wastes have been successfully treated to meet U. S. licensing standards. Waste treatment has mainly consisted of filtration and decay hold up of gases and demineralizature, filtration, and evaporation for liquid wastes. Liquid waste releases have ranged from 0.0002% to 10% of the allowable limits. Gaseous wastes varied from 0.0026% to 28% of the limits. The concentration of tritium released in the liquid wastes has been much less than 1% of the limit.

No future problems are antecipated which cannot be successfully handled.

SLIDE 1 Production of Wastes at a PWR (69-9224)

a) Most radioisotopes in waste effluents originate in the primary coolant.

b) Activation of corrosion products

c) Chemical additives such as boron, hydrazine, LiOH, or ammonia

d) Activation of components of air

e) Fission products from tramp ^{235}U

f) Fission products from leaky fuel

g) Diffusion of tritium through the fuel cladding

h) Ion Exchange demineralizer beds

SLIDE 2 Production of Wastes in a BWR (69-9225)

Generally the same wastes are produced here as in the PWR except that the gases are removed continuously with large volumes of air and are held only about 20 to 30 min before discharge.

SLIDE 3 Sources of Wastes at a Power Reactor (69-8595)

SLIDE 4 Gaseous Waste Products in Power Reactors (69-8592)

SLIDE 5 Sources of Tritium at Power Reactors (69-8594)

a) Tritium mixed with water cannot be economically separated from water.

b) Has not been a disposal problem since its MPC is relatively high and large amounts of water for dilution are available

c) Tritium release has been at much less than 1% of the permissible limits

d) Diffusion through Zr-2 is believed to be much less than for stainless steel and could prove to be a method of control if boron is not used as a sham control

SLIDE 6 Maximum Permissible Discharge Rate at Big Rock Point (69-8597)

These limits are typical for power reactors.

SLIDE 7 Radioactive Liquid Wastes Processed for Reuse at Dresden Nuclear Power Station (69-8596)

SLIDE 8 Spent Resins in Storage at Dresden Nuclear Power Station (60-8588)

SLIDE 9 Radioactive Solid Waste Shipments to Permanent Land Burial from Dresden Nuclear Power Stations (69-8587)

SLIDE 10 Significant Radionuclides Present in Liquid and Solid Wastes at Power Reactors (69-8598)

SLIDE 11 Waste Management at a Large BWR (67-17-R2)

a) Gaseous Wastes

In normal operation, the principal gaseous radioisotope is ^{13}N. In addition, there are usually some gaseous fission products from tramp uranium. Over 99% of these isotopes leave the system by way of the turbine condenser air ejectors; the remainder by the turbine gland seal exhaust system. Since the volume of the air ejector stream is too large for long-term storage, expanded off-gas lines are provided to allow 15- to 30-min holdup for decay before discharge. When operation continues with fuel cladding defects, isotopes of the fission-product gases krypton and xenon are stripped from the coolant by the steam and are also discharged by way of the air ejectors through absolute filters to the stack.

b) Liquid Wastes

Management of liquid wastes is based on segregation of the streams into four categories, depending on their concentrations of dissolved inert solids. Clean (i.e., low solid) wastes consisting mainly of drainage from the primary reactor coolant system contain approximately 1 ppm of total dissolved solids. Following cleanup by filtration and demineralization, these wastes can be returned to the reactor water system for reuse. The "dirty wastes" (primarily floor drains) cannot be demineralized economically because of their high concentration of dissolved solids. Thus these wastes, along with the laundry wastes, are normally filtered and released to surface waters after dilution with condenser cooling water to a concentration below that prescribed for the release of unidentified radioisotopes (10^{-7} μc/cc). The "chemical wastes" contain the greatest concentrations of dissolved inerts; and after neutralization, they to, are usually filtered, diluted, and discharged. However, an alternative procedure includes evaporation of these wastes to a viscous solution or slurry that can be stored temporarily in a special concentrated waste tank before encapsulation for off-site shipment and disposal.

c) Solid Wastes

Solid wastes are stored at the reactor site in underground tanks and concrete vaults or in sheltered areas above ground, to await shipment to regional burial grounds by licensed contractors. The greatest volume of waste shipped needs little or no shielding in order to meet federal regulations and is most often packaged in 55-gal steel drums. Baling machines are widely used to reduce the volume of compressible solids by factors of 4 to 6 prior to shipment, while incinerators have generally fallen into disfavor because of their unfavorable economics. The more-radioactive wastes such as spent resins and filter cartridges are frequently encased in concrete-lined drums, and "wet" wastes such as evaporator concentrates and sludges are usually immobilized in drums with drying agents or vermiculite.

SLIDE 12 Waste Management at a Large PWR (67-33-R3)

a) Gaseous Wastes

The reactor operates with a closed primary cooling circuit, usually maintained with a hydrogen overpressure to suppress the radiolytic decomposition of water. Gases are released from the primary system when the coolant is withdrawn or as a result of leakage from the circuit. The radioisotopes and hydrogen are separated from the water by passage through a gas stripper and are then combined in a header with gases (principally nitrogen cover gas) from tank, equipment, and sampler vents. The gases are compressed, passed through moisture separators, and then collected in decay tanks where they are retained until they can be released at a controlled rate through absolute filters to the atmosphere. From 50,000 to 300,000 cfm of ventilation and containment building air may be released with the reactor gases through the containment building vent.

b) Liquid Waste

Liquid wastes from various drains and from resin regeneration, containing relatively large amounts of dissolved salts, are collected in holdup tanks and then treated by evaporation. The evaporator overheads may be either passed through mixed-bed demineralizers and returned to the plant or discharged. The evaporator bottoms are mixed with cement and put into drums for off-site disposal. Coolant letdown from the primary circuit contains boric acid, which is usually recovered. This stream is passed first through a gas stripper and then to an evaporator. The boric acid in the evaporator concentrate is prepared for reuse by filtration and ion exchange, while the overheads may be discharged or passed through demineralizers and reused. This same line of equipment is used to process so-called "clean wastes" of very low dissolved-solids content.

c) Solid Waste

All solid wastes are prepared for off-site disposal. Evaporator concentrates and spent resins are mixed with cement and packaged in steel drums, and compressible solids are generally baled. Sheltered areas are provided for temporary storage.

SLIDE 13 Gaseous Waste Releases Through 1966 (67-9820)

Gaseous Wastes – Each station is licensed to release gaseous activation and fission products under the limitations of 10 CFR 20 (U. S. Federal Regulations). These regulations specify that the releases shall be such that the dosage above natural background at any off-site location will not exceed 0.5 rem/year, and the dosage at any on-site location will not exceed 5 rem/year. These permissible rates were calculated with that limitation.

Maximum annual average releases os activation and noble gases have ranged from 0.7 μc/sec at Yankee to 35,000 μc/sec at Big Rock Point. The releases varied from a maximum of 0.003% of the limit at Indian Point to as much as 28% of the limit at Humboldt Bay.

The maximum rates of release from the four boiling-water reactors occurred during periods when operation continued with substantial numbers of defective stainless-steel-clad fuel elements in the cores. As the defective fuel was replaced during scheduled refueling outages, the discharge rates were markedly reduced. In the case of Humboldt Bay, reactor power was reduced (from a full rating of 68 Mw net to 40 Mw net) to maintain the gaseous release rate below the annual average limit of 50,000 μc/sec; but the maximum off-site dosage measured above background, integrated over 12 consecutive months was only 50 millirems. Off-site air monitoring at other sites has yielded measurements at or very near background in all cases. Maximum releases at Indian Point and Yankee occurred as a result of leaks in the steam generator tubes.

SLIDE 14 Liquid Waste Releases Through 1966 (67-9821)

All the stations included in this study have been able to operate within the limits authorized for discharge of liquid wastes without undue difficulty, and releases to the environment have, in general, been made on the conservative basis of unidentified radioisotopes, with ^{226}Ra known to be absent, that is, after diluting first to a concentration less than 10^{-7} μc/cc.

The average activity released annually in liquid wastes varies from 0.01 curie at Yankee and Elk River to 11.1 curies at Indian Point, exclusive of tritium. Periodic isotopic analyses of individual batches of waste at the stations indicate that the appropriate limits for discharge range from 10^{-6} to 10^{-5} μc/cc. On the basis of an allowable continuous discharge of activity at an annual average concentration of 10^{-6} μc/cc, the releases have ranged from as little as 10^{-3}% of the limit at Yankee to as much as 10% of the limit at Big Rock Point.

The concentration of tritium in the wastes has not been routinely determined in the past, but recent analyses give evidence that 5 to 20 curies have been released annually by each of the BWR stations, and about 500 and 1300 curies have been discharged annually at Indian Point and Yankee, respectively. On the basis of an allowable continuous discharge of tritium at an annual average concentration of 3×10^{-3} μc/cc, this isotope has been released at much less than 1% of its limit in all cases.

SLIDE 15 Waste Management Costs a Power Reactors (69-7900)

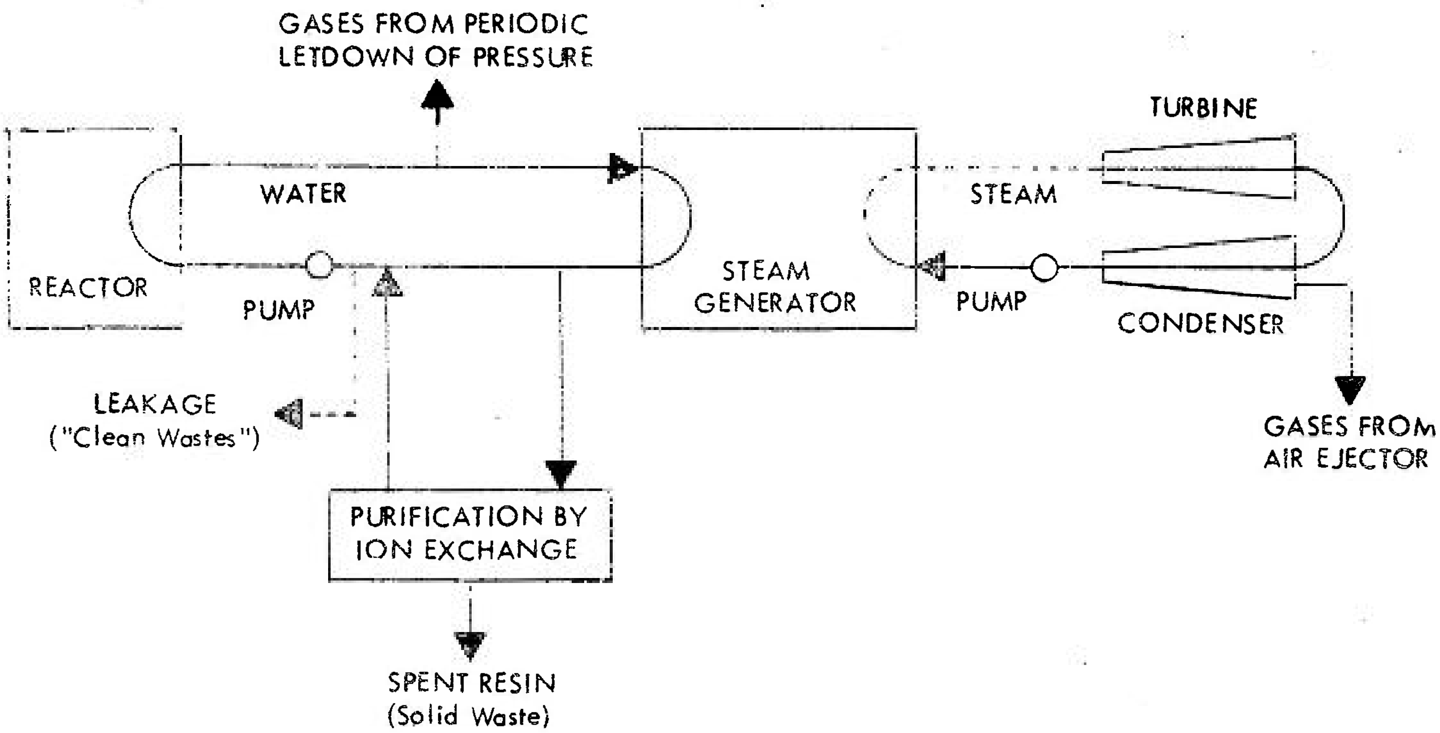
In conclusion, discharges of both gaseous and liquid wastes have been typically less than 10% of the established limits. Even during periods of operation with defective fuel or auxiliary equipment, the limits have not been exceeded, and these limits appear to be conservative judging by the results of off-site monitoring and occasional isotopic analyses of liquid wastes. The fact that no major changes in design or operating philosophy are anticipated in the new, larger plants now under construction is, itself, the best indication of satisfactory performance. Waste management practices at light-water power-reactor stations have been quite adequate in the past, and future problems are not anticipated that will not be resolved in the normal course of an expanding industrial technology.

REFERENCES AND EXCERPTS FROM:

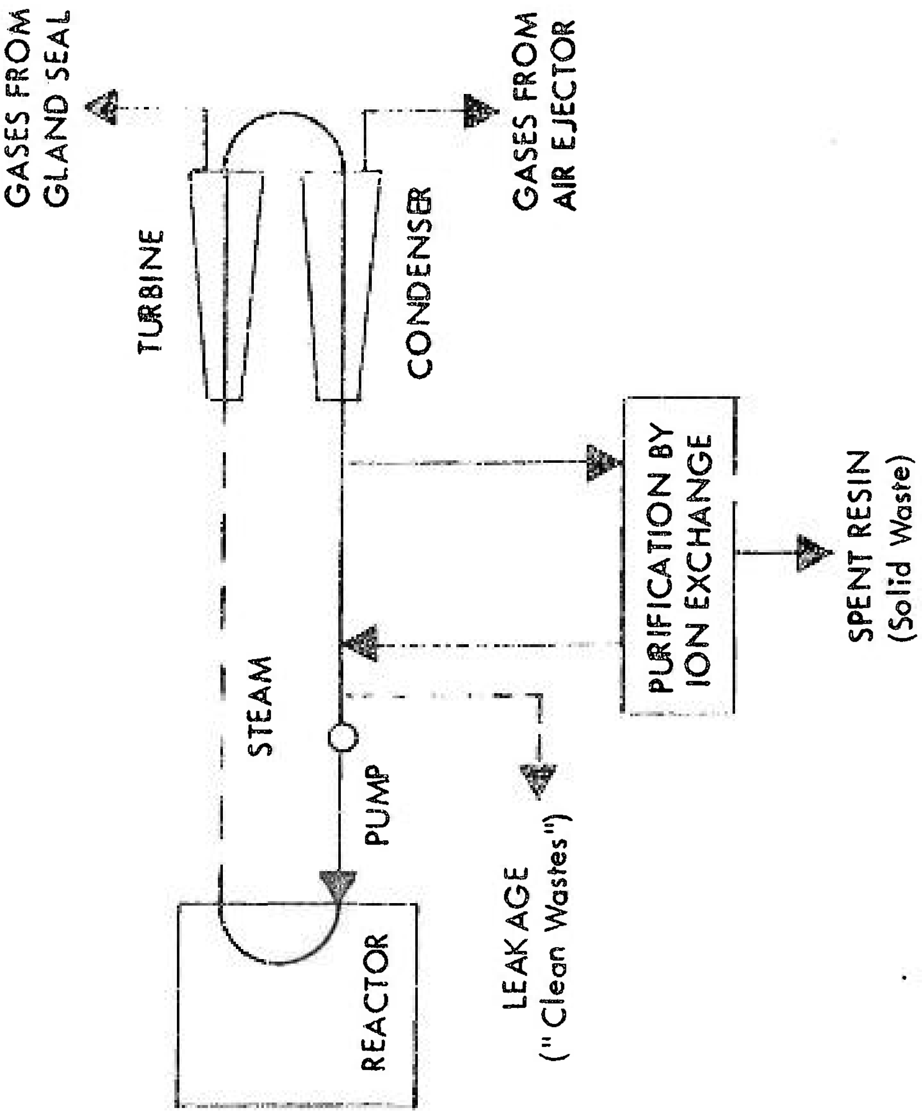
1. J. O. Blomeke, F. E. Harrington, *Management of Radioactive Wastes at Nuclear Power Stations*, ORNL-4070 (February 1969).

2. *Radioactive Waste Handling in the Nuclear Power Industry,* Edison Electric Institute, New York, 1960.

3. *Management of Radioactive Wastes at Nuclear Power Plants,* IAEA Safety Series No. 28 (1968).



PRODUCTION OF WASTES IN A PRESSURIZED WATER REACTOR



PRODUCTION OF WASTES IN A BOILING WATER REACTOR

SOURCES OF WASTES AT POWER REACTORS

LIQUID

1. Primary coolant
2. Shielding water for refueling operations
3. Decontamination
4. Laundry
5. Analytical and research laboratories
6. Fuel storage pool

SOLID

1. Tools
2. Laboratory equipment
3. Floor coverings
4. Paper
5. Rags
6. Plastic bags
7. Laundry
8. Used clothing

SIGNIFICANT GASEOUS WASTE PRODUCTS FROM POWER REACTORS AFTER 5 MIN DECAY

^{13}N (10 min), ^{16}O(p,α) ^{13}N

^{41}Ar (1.8 hr), ^{40}Ar(n,_) ^{41}Ar

Kr isotopes (fission product)

Xe isotopes (fission product)

At 3 day decay – Activity is decreased 10 to 100 times
– ~90% of activity is ^{133}Xe (5.27 day)

SOURCES OF TRITIUM AT A POWER REACTOR

1. From Tripartite Fission and Diffusion Through the Zr-2 or Stainless Steel Cladding

2. Leakage from Failed Fuel Elements

3. Neutron Activation of Hydrogen or Deuterium in Water

4. Neutron Reactions with Boron
   $^{10}B(n, 2\alpha)\ ^{3}H$ and $^{10}B(n, \alpha)^{7}\ Li(n, n\alpha)^{3}H$

MAXIMUM PERMISSIBLE DISCHARGE RATE AT BIG ROCK POINT

GASEOUS WASTES:	
Noble and activation-product gases:	
Annual average, curies/sec	1.0
Instantaneous, for periods up to 1 week, curies/sec	10
Particulates and halogens:	
Annual average, μc/sec	3.6
LIQUID WASTES:	
Maximum unidentified isotopes in discharge canal, μc/cc	10^{-7}
Maximum waste inventory in system, curies	5,000
SOLID WASTES:	
Maximum waste inventory on-site, curies	40,000

LIQUID WASTES PROCESSED FOR REUSE AT DRESDEN

	TO DEMINERALIZER			TO REUSE	
PERIOD	1,000 gal	mc	10^6 μμc/l	mc	10^4 μμc/l
1959	1,665	<1	<1	<1	<1
1960	7,728	152,840	5.2	390	1.4
1961	7,764	379,910	12.9	3,180	10.8
1962	12,640	430,980	9.0	1,620	3.4
1963	9,900	545,140	14.5	540	1.5
1964	10,020	174,460	4.6	190	5.1
1965	8,260	659,320	21.0	4,620	15.0
1966	11,590	794,113	18.1	7,046	16.2
Total	69,567	3,136,763		17,586	

[a] This table is taken from Diederich and Kiedaisch

SPENT RESINS IN STORAGE AT DRESDEN NUCLEAR POWER STATION[a]

PERIOD	RADWASTE (ft^3)	CLEANUP (ft^3)	CONDENSATE (ft^3)	TOTAL (ft^3)	ACCUMULATIVE TOTAL (ft^3)
1959	0	0	105	105	105
1960	400	160	120	680	785
1961	627.5	240	82.5	950	1735
1962	640	80	105	825	2560
1963	700	130	270	830	3390
1964	400	80		480	3870
1965	480		485	965	4835
1966	560	160	0	720	5555

[a] This table is taken from Diederich and Kiedaisch

RADIOACTIVE SOLID WASTE SHIPMENTS TO PERMANENT LAND BURIAL FROM DRESDEN NUCLEAR POWER STATION[a]

PERIOD	ESTIMATED ACTIVITY (curies)	TOTAL (ft^3)	ACCUMULATIVE TOTAL (ft^3)
1960	0.1	1,147	1,147
1961	62	5,112	6,259
1962	561	3,312	9,571
1963	1.1	5,970	15,541
1964	0.4	1,918	17,459
1965	41	3,524	20,983
1966	0.5	1,994	22,977

[a] This table is taken from Diederich and Kiedaisch

SIGNIFICANT RADIONUCLIDES PRESENT IN LIQUID AND SOLID WASTES AT POWER REACTORS

Activation Products

^{99}Mo (67 hr)

58 ~~^{88}~~Co (71 day)

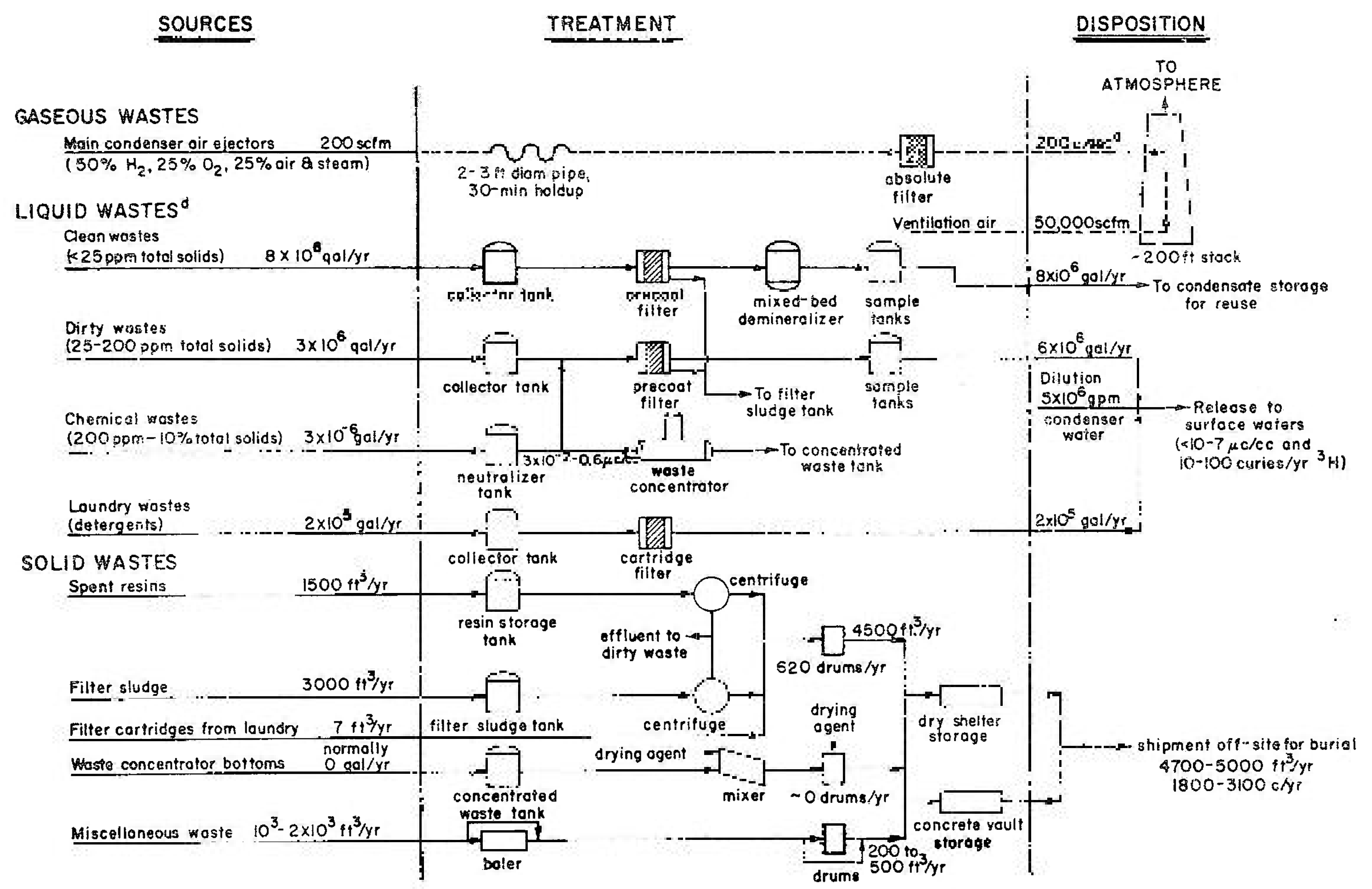
^{110m}Ag (60 day)

^{51}Cr (28 day)

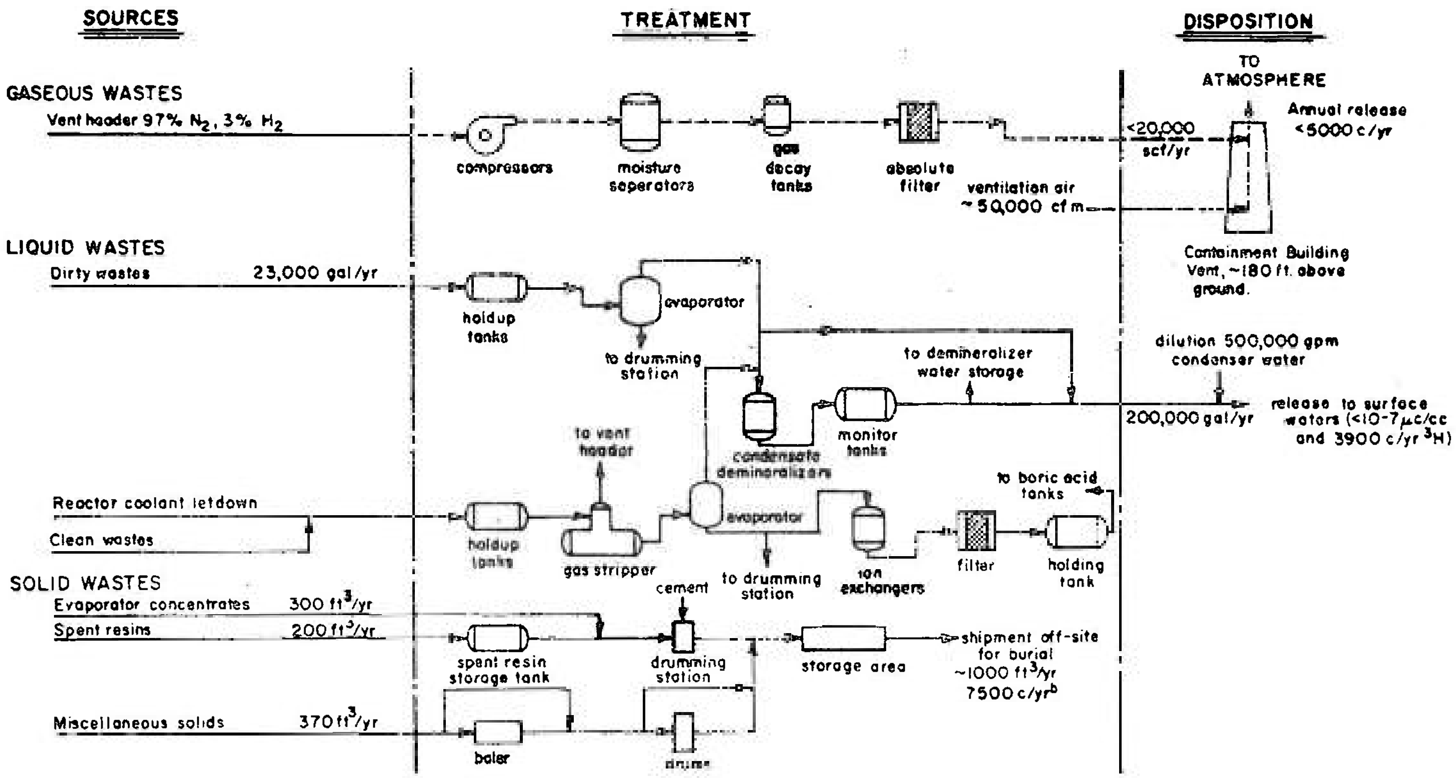
Fission Products

^{131}I (8 day)

^{133}I (21 hr)



WASTE MANAGEMENT FLOW DIAGRAM FOR A TYPICAL LARGE BWR



WASTE MANAGEMENT FLOW DIAGRAM FOR A LARGE PWR

GASEOUS WASTE RELEASES (THROUGH 1966)

	ANNUAL AVERAGE RELEASE, μc/SEC.		PERCENT OF PERMISSIBLE
	PERMISSIBLE RATE*	ACTUAL RELEASE RATE	
DRESDEN	700,000	<100-25,000	0.02-4
BIG ROCK POINT	1,000,000	< 20-35,000	<0.002-4
HUMBOLDT BAY	50,000	40-14,000	0.08-28
ELK RIVER	600	~0-110	~0-18
INDIAN POINT	50,000	0.07-2	10^{-4}-0.003
YANKEE	2,000	10^{-4}-0.7	~0-0.03

*DERIVED FROM LIMITS STIPULATED IN AEC LICENSES.

LIQUID WASTE RELEASES (THROUGH 1966)

	TOTAL ACTIVITY, CURIES/YR*	PERCENT OF LIMIT**	ESTIMATED RELEASE OF TRITIUM	
			CURIES/YR	% OF LIMIT
DRESDEN	4.1	0.3-4	5-10	$<10^{-3}$
BIG ROCK POINT	5.8	2-10	20	0.007
HUMBOLDT BAY	1.3	0.3-1	20	0.004
ELK RIVER	0.01	~0-0.05	10	0.006
INDIAN POINT	11.1	0.03-5	500	0.03
YANKEE	0.01	10^{-3}-10^{-2}	1300	0.16

*EXCLUSIVE OF TRITIUM.

**LIMIT BASED ON A CONTINUOUS DISCHARGE OF 10^{-6} μc/cc.

WASTE MANAGEMENT COSTS AT POWER REACTORS

(In $1,000)

		Capital Costs			Annual Operating Costs		
	Mw (e)	Total	Waste	Waste (%)	Total	Waste	Waste (%)
Dresden	200	51,000	2400	4.7			< 10%
Big Rock Point	50	26,700	450	1.7	590	28	4.7
Humboldt Bay	52	24,200	680	2.8	397	9.34	2.3
Yankee	185	43,300	1300	3.0		~60	
Indian Point	270	61,300	4060	6.6			

MANAGEMENT OF WASTES AT FUEL REPROCESSING PLANTS — PRESENT PRACTICE

R. E. Blanco

This lecture will describe the methods used at present for waste management at existing or proposed fuel reprocessing plants. Emphasis is placed on management practices at U. S. installations, i.e., the AEC — Savannah River and Hanford Plants, the private plant of Nuclear Fuel Services (NFS), and proposed plans for the General Electric, Atlantic Richfield, and Allied Chemical plants. Some additional discussion is included on practices at European plants.

This lecture contains excerpts from papers by Ferguson and Blanco on a survey of fuel reprocessing, P. E. Smith on waste management at Hanford, and a general survey paper by W. H. Hardwick on present waste practices (see references at end of lecture).

SLIDE 1 Approximate Waste Volumes and Processing Costs (66-640)

This slide does not present a precise calculation, it is used only to show an approximate distribution of costs. Note the volume increase from high to intermediate to low levels. With mechanical decladding, the intermediate wastes have low volume, but the volumes are high if chemical decladding is used. The relative importance of high-level wastes is apparent. For both economic and safety reasons, research and development in waste management has concentrated on high-level waste.

SLIDE 2 Wastes from Precdited Nuclear Power Economy (68-10566)

This slide shows an estimate prepared by J. O. Blomeke (ORNL) of liquid waste generation and the inventory of some significant fission products, assuming that the installed nuclear capacity in the U. S. arises from 12,000 Mw (electrical) in 1970, to 123,000 Mw (electrical) in 1980, to 675,000 in 2000. By 2000, we will be generating about 5 million gal/year of high-level liquid waste, and may have accumulated a total of something like 55 million gal containing 143,000 megacuries of fission product activity, including about 15,000 megacuries of ^{90}Sr. This is a smaller volume than the ~ 70 million gal now in storage at AEC sites, but the curies of biologically significant isotopes will greatly exceed those of the AEC before 1980. The quantities of krypton and tritium shown here are of interest because these nuclides would normally be released during fuel processing unless steps are taken for their separation and containment.

SLIDE 3 Intermediate- and Low-Level Solid Wastes from Spent Fuel Processing (68-10572)

This slide shows the projected volumes of low- and intermediate-level solid wastes which will be generated between 1970 and 2000. If buried according to present practice, they will consume about 940 acres of land by 2000. The large amounts of existing space in salt mines is a very attractive possibility for the disposal of this type of waste. More than 40 million ft^2 of area (900 acres) has already been mined in salt, and this could contain all the solid wastes of the type shown here that are projected for the Civilian Nuclear Power Program through 2010.

SLIDE 4 No Title (65-9305)

This slide shows compositions of typical high-level wastes. Additional data is included in the lecture on waste solidification.

LIQUID WASTE STORAGE

At the present time, a high-level waste from all reprocessing plants in the world are stored in underground tanks. The Savannah River, Hanford, and Nuclear Fuel Services plants concentrate the wastes by evaporation, neutralize with caustic and store in mild steel tanks. The Idaho and all other reprocessing plants in the world store the acidic, concentrated wastes in stainless steel tanks. (More recently, wastes which have cooled by decay have been solidified by calcination at Idaho and "in tank solidification" at Hanford. See lecture on solidification.) The chief advantage of neutralized storage is the low cost of mild steel. However, sludges form in the tank bottoms which cause bumping problems when steam is formed in the sludge cakes. The higher cost of stainless steel is compensated for by the greater volume reduction achieved($\approx$ lower storage volume) and the absence of solids or sludges.

SLIDE 5 Storage Tank for Non-Boiling Waste (50486-22)

This slide shows a tank used for non-boiling wastes at Hanford. High level liquid wastes from the chemcial processing plants at Hanford have been stored as alkaline slurries in underground tanks since startup. Currently, wastes in-process are being stored on an interim basis. The complex of waste storage tanks includes 149 tanks, ranging in capacity from 50,000 to 1,000,000 gallons. These tanks are constructed of reinforced concrete with a carbon steel liner on the bottom and side walls. The first tanks were built for non-boiling wastes. These tanks are vented to the atmosphere, some through air-cooled reflux condensers. Instrumentation is provided to measure the sludge and supernatant temperature, and the liquid and sludge levels in the tanks. A grid of dry wells in each tank farm is used to monitor the soil for radioactivity, thus serving as a secondary leak detection system.

SLIDE 6 Storage Tank for Boiling Waste (50486-12)

Later tanks at Hanford were built to contain wastes with greater heat generating characteristics. These tanks are of the same general construction as those described for non-boiling wastes. Additional features have been provided, however, to permit self-concentration of waste, the vapors being routed through headers to cyclones and water-cooled condensers which are vented to the atmosphere after passing through a packed deentrainment vessel, filters and exhaust blowers. Condensate is routed to an underground crib or is returned to the waste tank to prevent over-concentration. The tanks are provided with airlift circulators for the agitation of the contents to prevent localized temperature buildup in the sludge and resultant bumping caused by sudden steam release. Each tank is closely encompassed by a complex of vertical and horizontal dry wells and so that any leak may be detected by monitoring for radioactivity in the soil surrounding the tank. Some of the tanks have a drainage grid beneath the tank liner which connects to an associated leak detection well. A temperature element, liquid level instrumentation, and a radiation detector are located in each well. Any leakage which might be detected can be removed by pumping.

SLIDE 7 Schematic of SRP Waste Storage Tank (46189)

This slide shows a Savannah River tank which contains cooling coils for heat removal. The inner mild steel tank is separated from an outer mild steel pan which is contained in concrete. Any leakage which might occur would be dectected in the pan.

SLIDE 8 Waste Treatment at Nuclear Fuel Services

This slide shows the waste treatment flowsheet for the Nuclear Fuel Services Plant in the U. S., the only private fuel reprocessing plant in the world at this time.

The acid waste streams are evaporated: (1) to concentrate them for storage, (2) to decontaminate the excess liquid, and (3) to recover nitric acid for recycle. The high-level radioactive waste from the first extraction cycle (HAW) is evaporated, and the concentrate is sent to the high-level radioactive waste storage tanks. The condensate is combined with low-level waste from other processing cycles and evaporated a second time. The resulting concentrate is combined with the first-cycle concentrate. The overhead vapors pass into a fractionation column for recovering the nitric acid, and are them condensed and re-evaporated before the condensate is discharged to the lagoon and on-site streams. Thus the liquid from the high-level waste, in effect, is purified by four distillations prior to discharge.

The high-level radioactive waste concentrate is neutralized and stored in 750,000-gal waste tanks that are buried underground. These wastes will be expected to boil for several years. The overheads are condensed and either returned to the storage tank or re-evaporated and discharged via the interceptor-lagoon system.

Thus the intent is to retain essentially all salts and fission products (except tritium) from all levels of liquid wastes. This must be contrasted to the plant at Windscale in the United Kingdom and the La'Hague plant in France where the lower level waste streams are chemically treated to remove most of the fission products and the remainder and inert salts are discharged to the ocean. However, thousands of curies of mixed fission products are discharged. Extensive surveys have been conducted at each site, however, to establish the safety of the methods.

SLIDE 9 Schematic Waste Disposal Flowsheet for MFRP (68-10617 R1)

This slide shows the proposed waste treatment system for the General Electric MFRP plant.

In the Aquafluor process, more than 99.9% of the fission product activity is found in the aqueous waste stream from the solvent extraction system. This waste stream is concentrated in an evaporator that is equipped with an acid fractionating tower, condenser, and condensate distributor pot.

An electrically heated fluidized-bed calciner is provided to convert the concentrated stream of high-level radioactive waste to a dry solid. An aluminum salt is added to the concentrated waste stream, and the fission product oxides formed in the calciner are them intimately mixed and diluted with an inert alumina matrix. The off-gases from the operation of the calciner are passed through sintered metal filters and are then subjected to multiple condensation and scrubbing steps before they are released to the off-gas disposal system. The calcined fission products are removed from the bottom of the calciner and transferred into waste storage containers.

The high-level radioactive waste storage containers, which are fabricated of stainless steel, are suitable for long-term use in a water environment. They are designed to limit the maximum temperature of the contents of a safe value (~ 800°C, max.) in the waste storage basin. The high-level waste storage basin provides water cooling and shielding of waste containers.

Aqueous waste that is not routed to the high-level radioactive waste treatment system is converted to a salt cake for long-term on-site storage. Low-activity process wastes, flush and decontamination solutions, etc., are collected and transferred to a pot-type evaporator for concentration. The vaporized water is condensed and recycled to the process water makeup system. Concentrator bottoms are first neutralized with caustic, then fed to a wiped-film evaporator to produce a concentrated slurry that is fluid at operating temperature and can be pumped to a stainless steel-lined below-grade storage vault. Here, as the mixture cools to ambient temperature, it solidifies. Provision is made for return of any supernatant liquid for reconcentration.

In addition to the on-site retention of process wastes in an immobile form, there are two aspects of the facility design which warrant emphasis. The first is that no liquid process wastes will be discharged to the environment; instead, they will be eliminated by extensive use of recycle water and by stringent limitations on water intake into the process. Thus the potential hazard of contaminating potable water sources with radioactive material will be avoided. The freedom from liquid discharges will also decrease the dependence of the location on the site on hydrology and geology.

The Allied Chemical Corp. has announced that they will store concentrated acidic waste in stainless steel tanks. Essentially all fission products, as well as the small, but inevitable, losses of plutonium, will be combined in a single stream that will be stored as an acid solution in cooled high-integrity stainless-steel storage tanks. Thus, the Allied Chemical Corporation will retain the option of future recovery of by products, calcination, vitrification, or the use of any advances in handling radioactive wastes that might be developed during the next few years without sustaining any foresseable penalty.

SLIDE 10 Windscale Highly Active Storage Tank (69-10573)

The new tanks being installed at Windscale illustrate the complexity that safe containment demands. This tank as a capacity of 150 m^3 and is an upright cylinder 20 ft diameter and 20 ft high with a flat top and bottom. The base of the tank and the lower 3 ft section of the walls are fitted with a water jacket. The inerior of the tanks contain cooling coils with a heat removal capacity of 6×10^6 Btu/h. The tanks also contain jet sparge units and airlift circulators in order to keep any solids uniformly dispersed throughout the tank. The liquor temperature is normally maintained at 50°C and equipment is provided to enable further evaporation to be carried out within the tank. Filtered air will be heated to 60°C before passing into the tank, and on leaving the air will be passed through a disentrainment column and a dehumidification tower. After final decontamination in a scrubber the air will be discharged to atmosphere through a stack. At the present time it is hoped to store the wastes at a concentration equivalent to 40 l/ton of uranium processed and it is anticipated that this figure can be reduced to 20 l/ton in the future.

SLIDE 11 Typical Hanford Disposal Crib (50486-16)

Cribs are used extensively at Hanford for the disposal of intermediate level wastes and low level wastes are judged to have a high potential for becoming contaminate. A crib is a sub-surface liquid distribution system. Cribs are long, narrow trenches about 10 feet wide at the bottom, filled with several feed of graded gravel. A plastic membrane is placed over the gravel and the trench back-filled with earth. A single distributor pipe traverses the length of the crib within the gravel bed about 4 feet above the bottom of the excavation. For Hanford soils the design percolation rate is 10 gallons per square foot per day. Wells are drilled to various depths within the ground water adjacent (60-100 yards) to the crib in the direction of ground water movement to permit monitoring the crib effluent as it enters the ground water.

The long-lived isotopes of concern (^{90}Sr, ^{137}Cs and ^{239}Pu) sorb on the soil and are restricted from migrating to man's environment.

SLIDE 12 Schematic of Disposal Site for Intermediate Level Wastes (50486-21)

SLIDE 13 Approximate Sorption Patterns Typical Handford Disposal Crib (50486-24)

Operating standards are imposed on cribs to ensure that the cribbed waste volumes and compositions are compatible with the soil to the extent that a reasonable crib life can be expected. These limits are responsive to the inherent degree of control the plant equipment affords over the waste composition and are based on laboratory soil adsorption studies with synthetic and actual plant waste.

The slide shows the pattern of sorption of fission products on the soil. Essentially all of the plutonium is held within 10 ft of the point of release.

SLIDE 14 Purex Cooling Water Pond (50486-13)

Ponds are used at Hanford for the disposal of large volumes of low level waste. Low level wastes are derived from such sources as water from cooling coils, from condensers and evaporators. The ponds are formed by the discharge of waste to natural depressions. The area of the pond increases until the amount of water lost by evaporation and percolation through the soil equals the rate of waste addition. A Hanford pond will accommodate a flow of about 2 to 5 gallons per square foot per day.

Operational controls on ponds are minimal. The volume of waste entering the pond is measured and the radionuclide content of the waste monitored by routine sampling and analyses. These controls are supplemented by in-line radiation detection instruments to ensure routine control of radionuclide concentrations below 5 x 10^{-5} μCi/ml of alpha and beta radioactivity. Wells adjacent to the ponds are also monitored to ensure complete control of the amount of radioactivity entering the ground water from this source.

SLIDE 15 Solid Waste Burial Trenches (Industrial and Dry Waste) (50486-2)

Burial trenches for small solid waste boxes or drums are about 12 feet deep. Boxes or drums are dumped into the shallow trenches, and the trenches are back-filled with earth by bulldozers. The weight of the soil covering the boxes crushes them, compacting the waste.

Failed or obsolete process equipment is packaged in wooden, concrete, or steel boxes and buried in trenches about 16 feet deep. The boxes are transported to the burial trench by railroad using spacer cars to separate the train crew from the radiation field emitted by the burial container. At the burial site the box is pulled into the trench by a bulldozer and a long cable. The trench is then back-filled with earth by the bulldozer. During trench filling the bulldozer operator maintains an earth shield between himself and the box.

SLIDE 16 Tunnel for Storage of Radioactive Equipment (50486-3)

Some very large pieces of processing equipment at Hanford are too large for practical packaging and transport to burial trenches. These equipment pieces are placed on railroad cars and pushed into burial tunnels especially constructed for this purpose. These tunnels connected into the plant railroad system within the plant containment walls so that no external problems are encountered. The equimpent is flushed and decontaminated before it is placed in the tunnel to minimize contamination problems.

SLIDE 17 Schematic of Ventilation Typical Separations Plant (50486-1)

Gaseous wastes are dispersed from controlled ventilation systems to prevent release of radioactivity. Ventilation air for process buildings is supplied from the atmosphere by fans through filters. After passing through the building the air is transferred through a rough filter and two stages of high efficiency filters before venting to the atmosphere through a 200-foot stack. The air is continuously monitored and sampled periodically to determine the quantity of radioactivity released to the atmosphere.

Treatment of off-gases from process equipment varies significantly depending on the history and gas composition. The vessel ventilation off-gases in the Purex reactor fuels processing plant pass through a condenser, a heater, a silver reactor for iodine removal and through two filters prior to stack entry. Ventilation and process off-gases from all other processing operations having potential for the release of particulate radioactive material are exhausted to the atmosphere through high efficiency filters in series to prevent radioactivity release. In addition, separators, demisters, scrubbers and absorbers are used as required to protect the high efficiency filters and retain noxious chemicals.

SLIDE 18 Ventilation Filters (50486-6)

A wide variety of high efficiency filters is in use. Each of these has advantages and disadvantages. The paper filters are very efficient but lose strength when wet and develop a high pressure drop at relatively low dust loading. The glass fiber filters have the same operating and failure characteristics as the paper filters, but are significantly less failure-prone. The sand filter is the least failure-prone and also the least efficient.

Air that comes in close contact with process materials picks up process chemicals and radioactive contaminants. These gases are segregated into separate vent systems according to pressure requirement and compatibility of the process gases. The gases are processed for removal of noxious materials in a variety of ways. Once the noxious materials have been scrubbed to acceptable levels, the process vent gases are exhausted to the ventilation system for additional filtration and return to the atmosphere.

REFERENCES AND EXCERPTS FROM:

1. Don E. Ferguson and R. E. Blanco, "General Survey of Reprocessing of Thermal Reactor Fuels," presented at the 1968 American Nuclear Society International Meeting in Washington, D. C., November 10-15, 1968; to be published in the *Proceedings.*

2. P. W. Smith and G. L. Hanson, "Radioactive Waste Management Policies and Practices," ARH-SA-30, Atlantic Richfield Hanford Company, April 2, 1969.

3. W. H. Hardwick, IAEA Advanced Training Seminar in Waste Management, Oxford- September 1967, Lecture Notes, "Presented Practices and Future Thoughts on High Level Liquid Waste Management."

4. J. O. Blomeke, unpublished data, Oak Ridge National Laboratory.

5. John O. Blomeke and J. Tom Roberts, "Waste Management," *Annual Review of Nuclear Science,* Vol. 15, 1965, pp. 151-74.

APPROXIMATE WASTE VOLUMES AND PROCESSING COSTS

(20,000 Mwd/ton U)

	Volume	Cost	
	(gal/ton U)	(mills/kwh)	(%)
High Level	100-200	0.02	82
Intermediate Level	110[a] (1500-2000)[b]	0.0002 (0.022)	1
Low Level	10,000	0.004[c] (<0.0004)[d]	17

[a] Mechanical decladding.

[b] Aqueous decladding.

[c] Distillation and burial.

[d] Chemical treatment and burial.

WASTES FROM PREDICTED NUCLEAR POWER ECONOMY

	CALENDAR YEAR		
	1970	1980	2000
INSTALLED NUCLEAR CAPACITY, 10^3 Mw(e)	12	123	675
VOLUME HIGH-LEVEL LIQUID WASTE [a]			
ANNUAL GENERATION, 10^6 GAL/YR	0.1	0.9	4.6
ACCUMULATED VOLUME, 10^6 GAL	0.2	3.5	55
FISSION PRODUCTS			
ACCUMULATED BETA MEGACURIES	1000	15,500	143,000
ANNUAL MEGACURIES ^{90}SR	16	260	1,300
ACCUMULATED MEGACURIES ^{90}SR	16	790	14,700
ANNUAL MEGACURIES ^{85}KR	2.3	38	220
ANNUAL MEGACURIES ^{3}H	0.07	1.1	6

[a] BASED ON 100 GAL HIGH-LEVEL ACID WASTE PER 10^4 Mwd(th) IRRADIATION

INTERMEDIATE-AND LOW-LEVEL SOLID WASTES FROM SPENT FUEL PROCESSING

	CALENDAR YEAR			
	1970	1980	1990	2000
VOLUME OF CLADDING WASTE[a]				
ANNUAL, 10^3 FT^3	5	50	160	290
ACCUMULATED, 10^3 FT^3	12	220	1,270	3,460
TOTAL VOLUME OF SOLID WASTE[b]				
ANNUAL, 10^6 FT^3	0.07	0.6	2	4
ACCUMULATED, 10^6 FT^3	0.2	3	15	47
BURIAL GROUND AREA[c]				
ANNUAL, ACRES	1	12	40	80
ACCUMULATED, ACRES	4	60	300	940

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[a] BASED ON 14.7 FT^3 OF CLADDING HULLS PER TON OF FUEL PROCESSED.

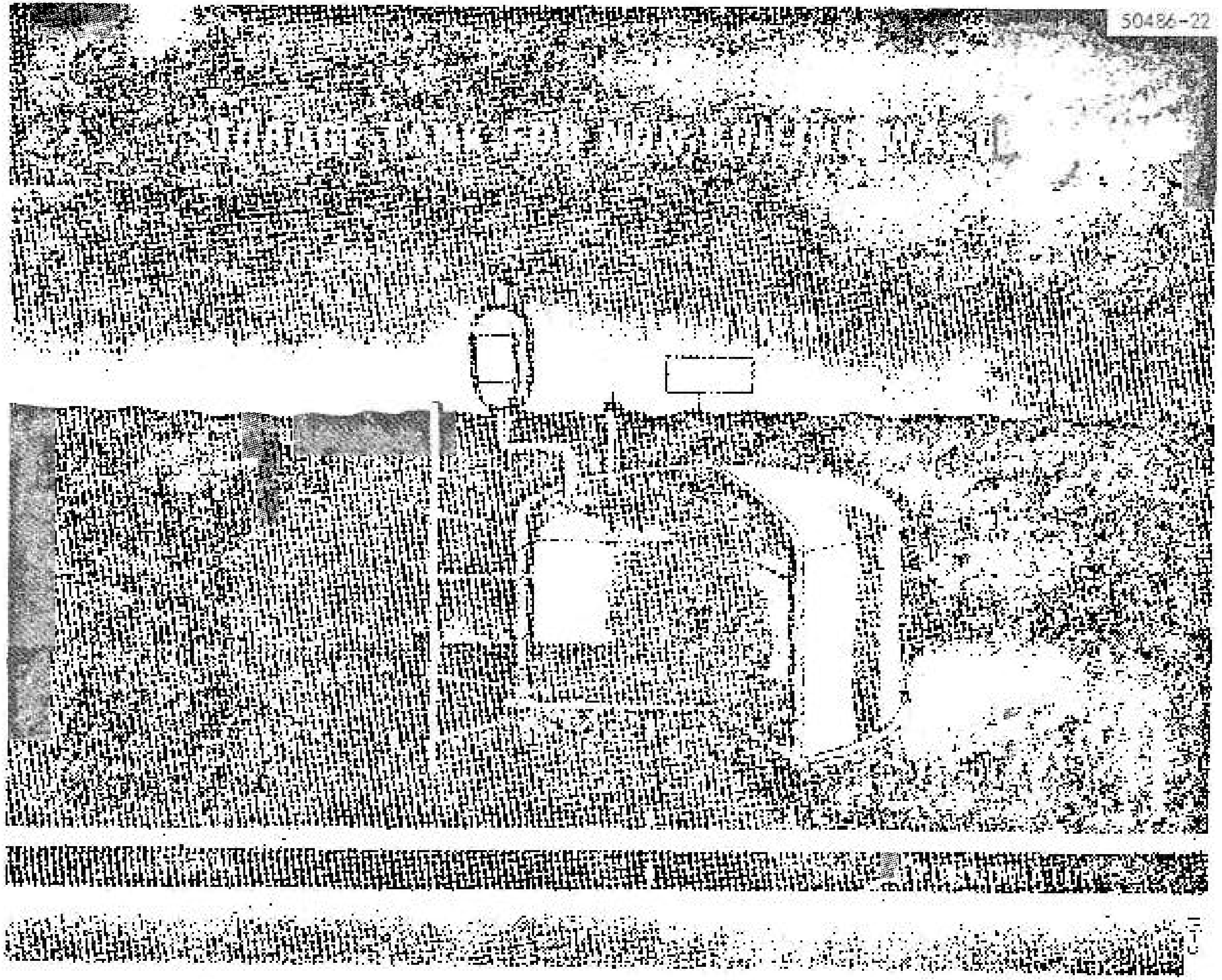
[b] BASED ON AN AVERAGE VOLUME OF 200 FT^3 OF SOLID WASTES PER TON OF FUEL PROCESSED.

[c] BASED ON BURIAL OF 50,000 FT^3 SOLID WASTE PER ACRE OF BURIAL GROUND.

COMPOSITION RANGES OF SIMULATED WASTES

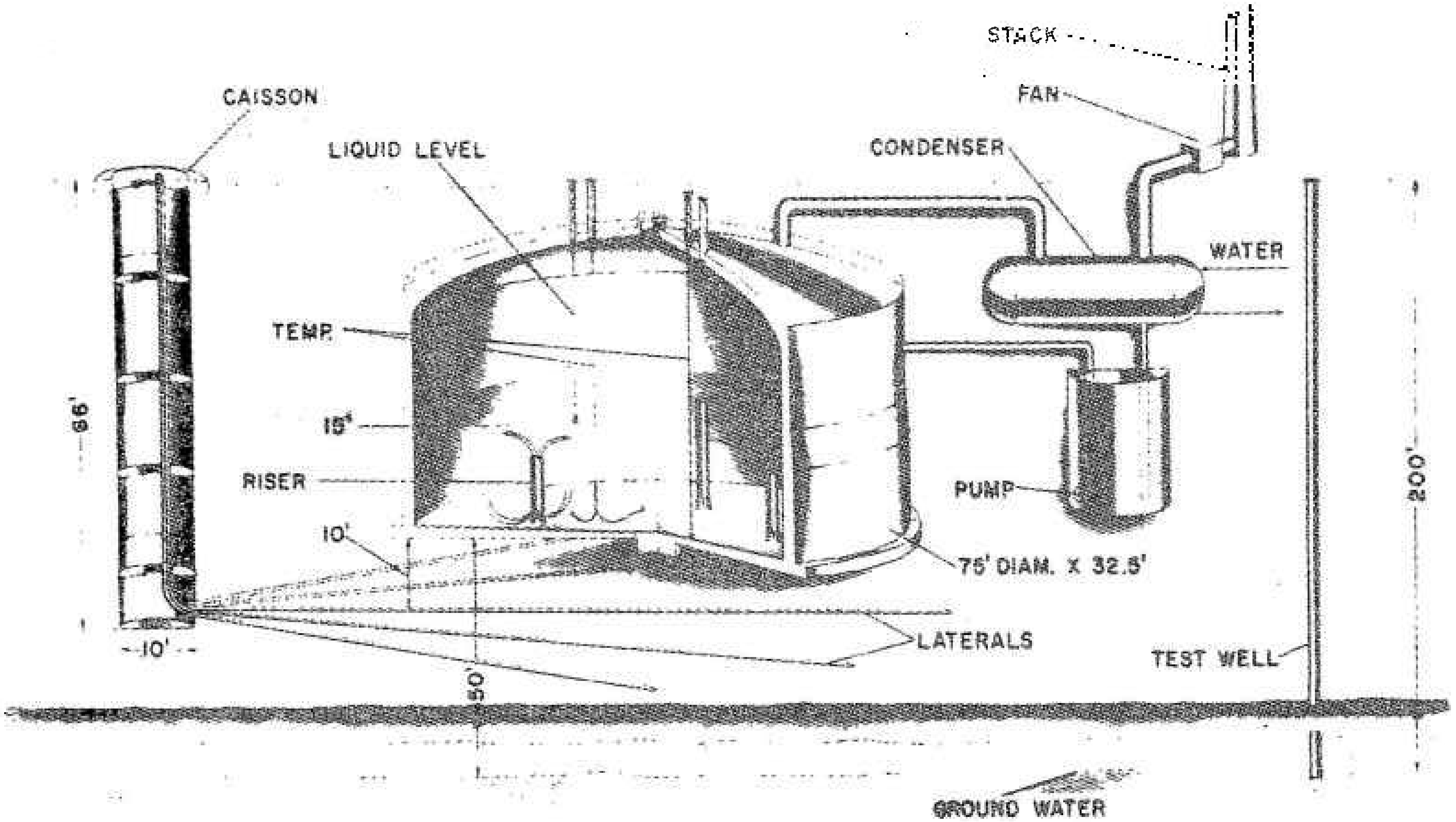
Cons't. (g·moles/liter)	High SO_4^2 (1 WW; FTW-65)	High Fe (Purex No. 1; Darex)	Aluminum (TBP-25)
H^+	0.5-5.6	2.0-5.0	1.26
Na^+	0.3-0.9	0-0.14	0.1
Al^{3+}	0.05-0.15	0-0.001	1.6
Fe^{3+}	0.1-0.5	0.9-1.3	0.002
Cr^{3+}	0.01-0.06	0.1-0.4	–
Ni^{2+}	0.01-0.03	0.05-0.2	–
SO_4^{2+}	0.15-1.0	–	0.03
NO_3^-	1.0-6.1	7.2-8.8	5.4
Sim. FPs. (max.)	0.144	0.397	0.002

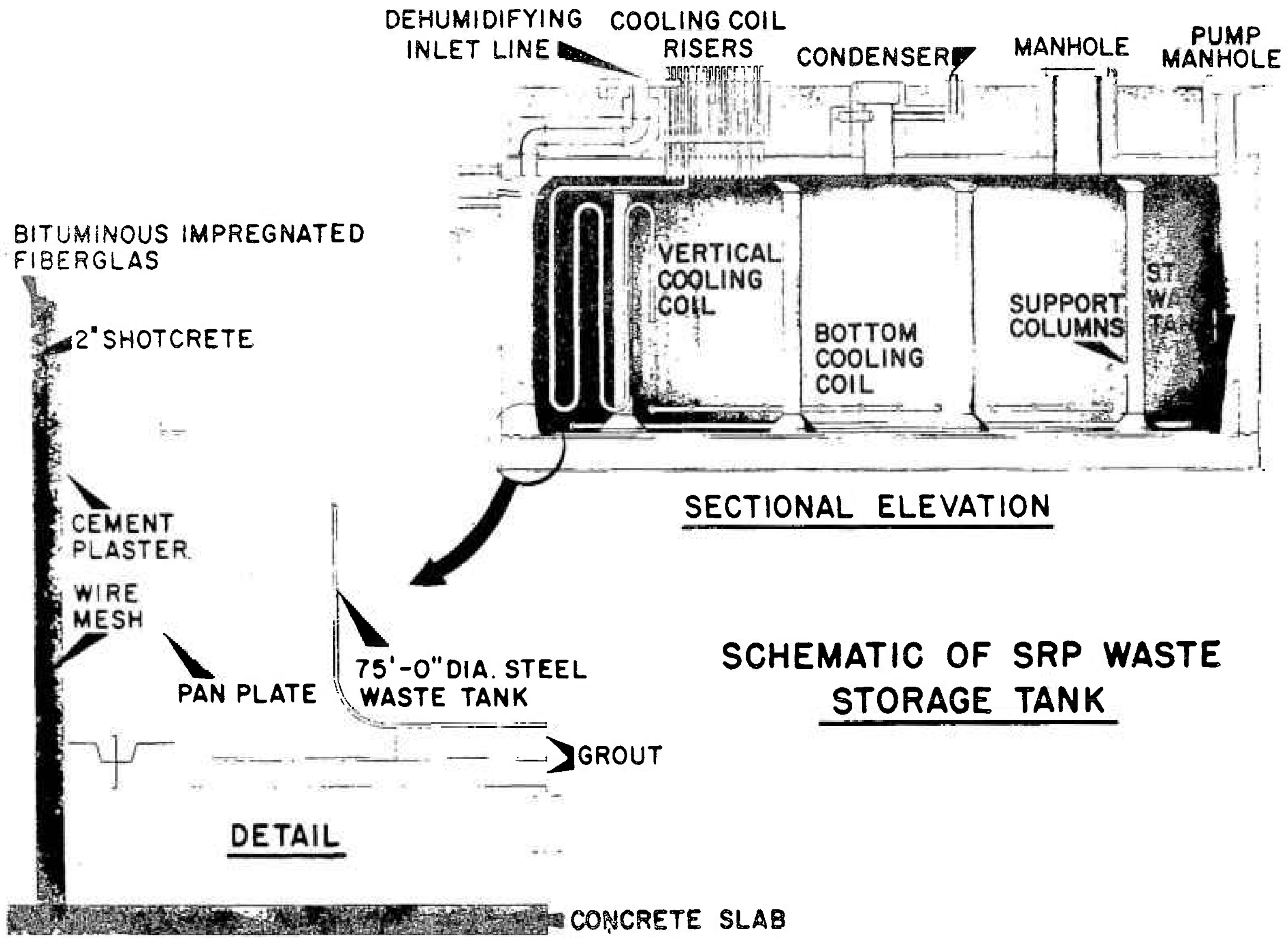
Plus varying traces (>0.01 M) of Hg^{2+}, PO_4^{3-}, SiO_3^{2-}, F^-, Cl^-


50486-22
STORAGE TANK FOR NONCOOLING WASTE

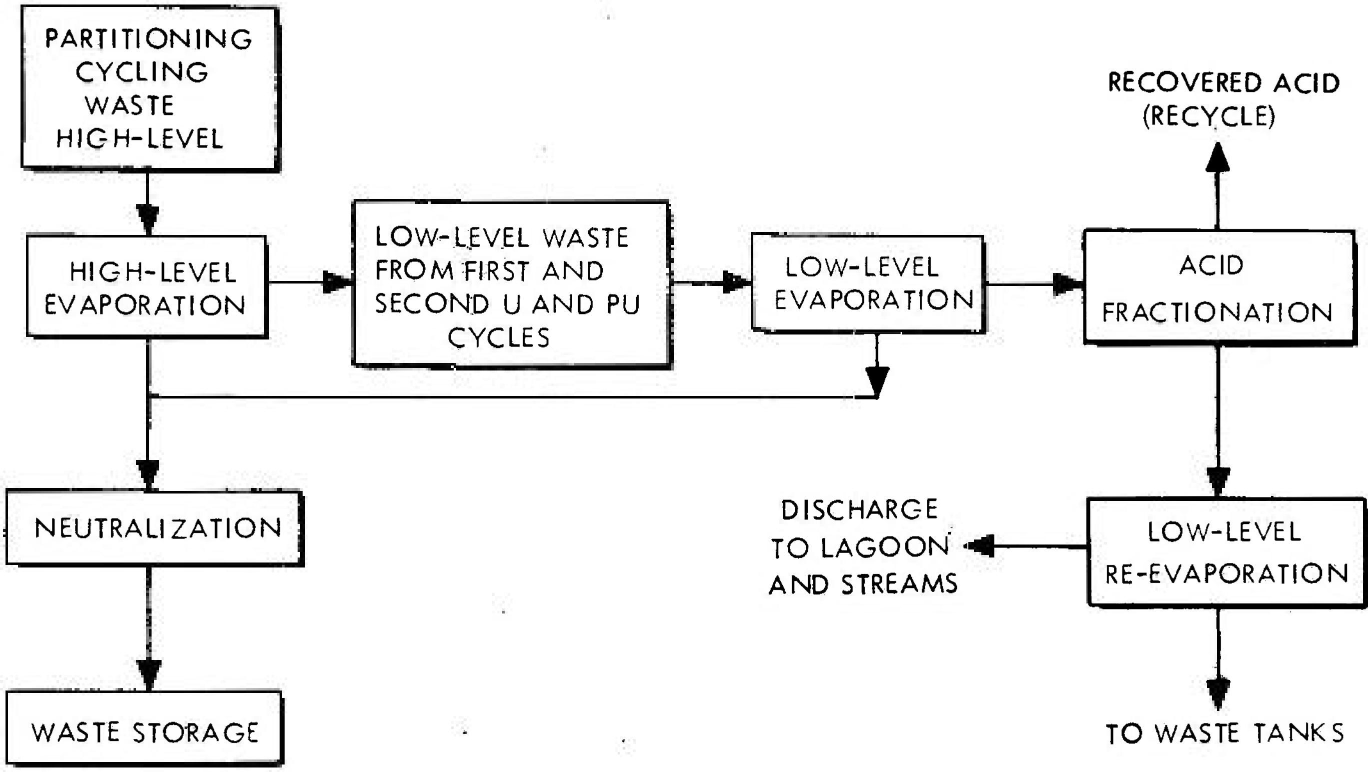
15-6

STORAGE TANK FOR BOILING WASTE

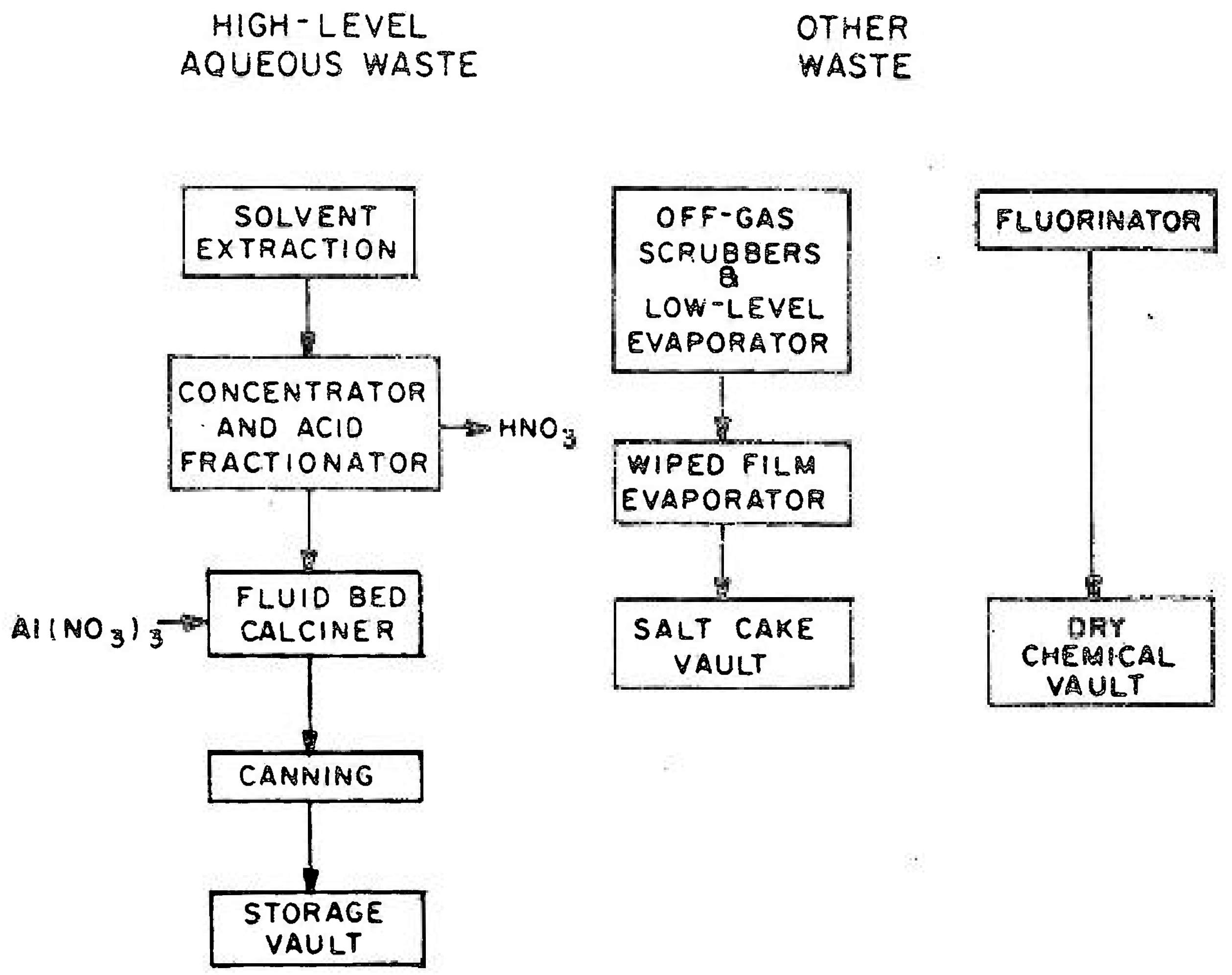




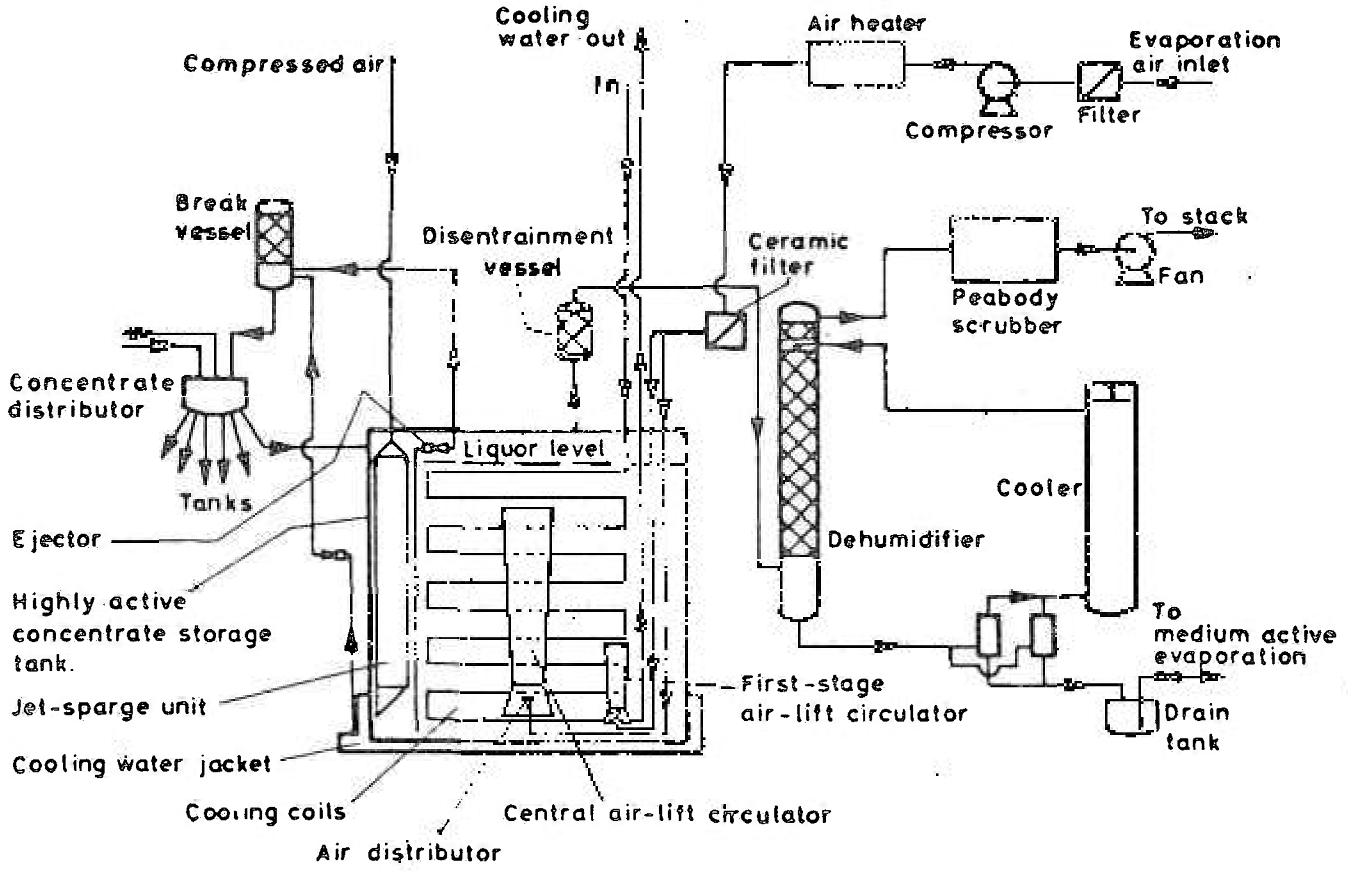
SCHEMATIC OF SRP WASTE STORAGE TANK



WASTE TREATMENT AT NUCLEAR FUEL SERVICES

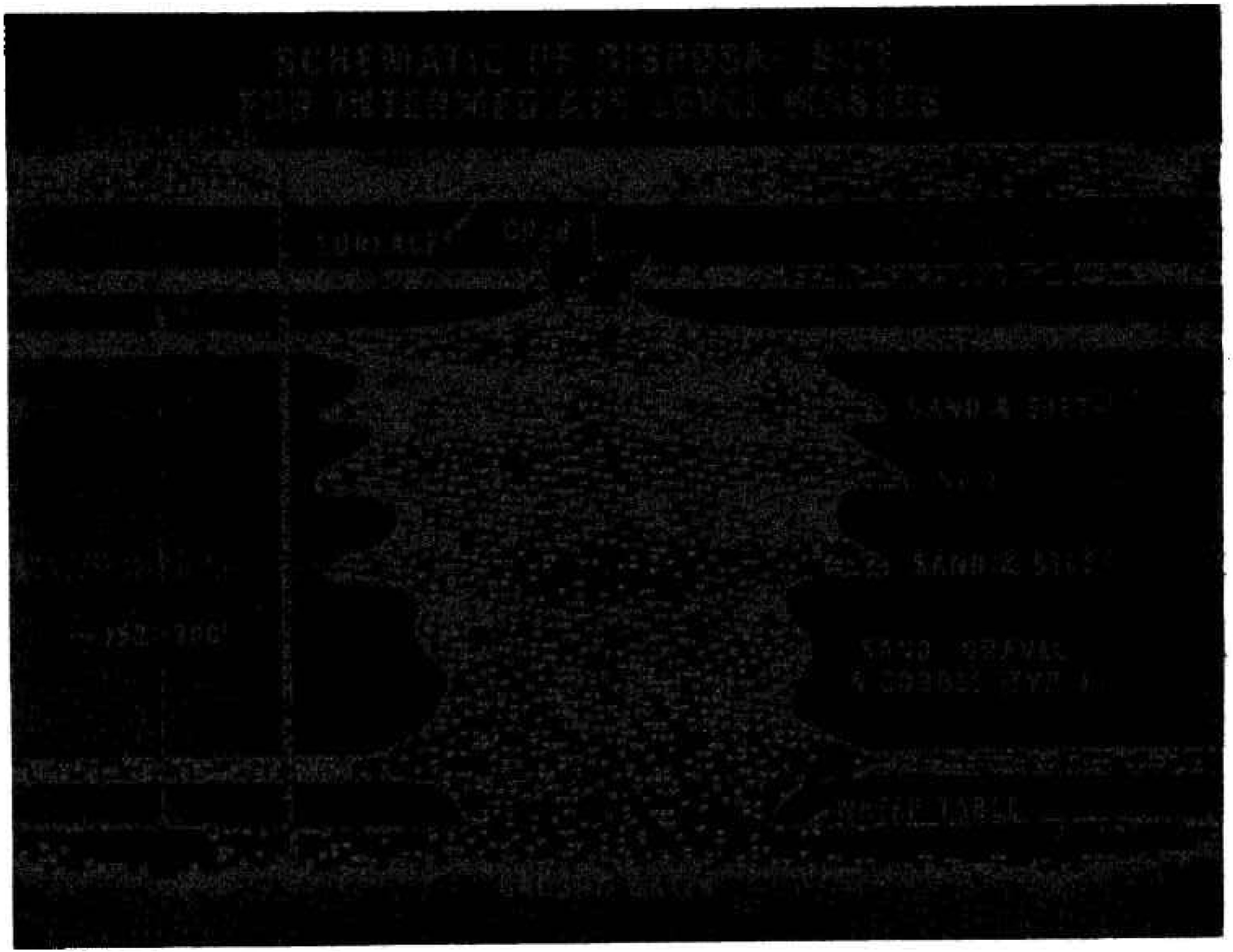


SCHEMATIC WASTE DISPOSAL FLOWSHEET FOR MFRP

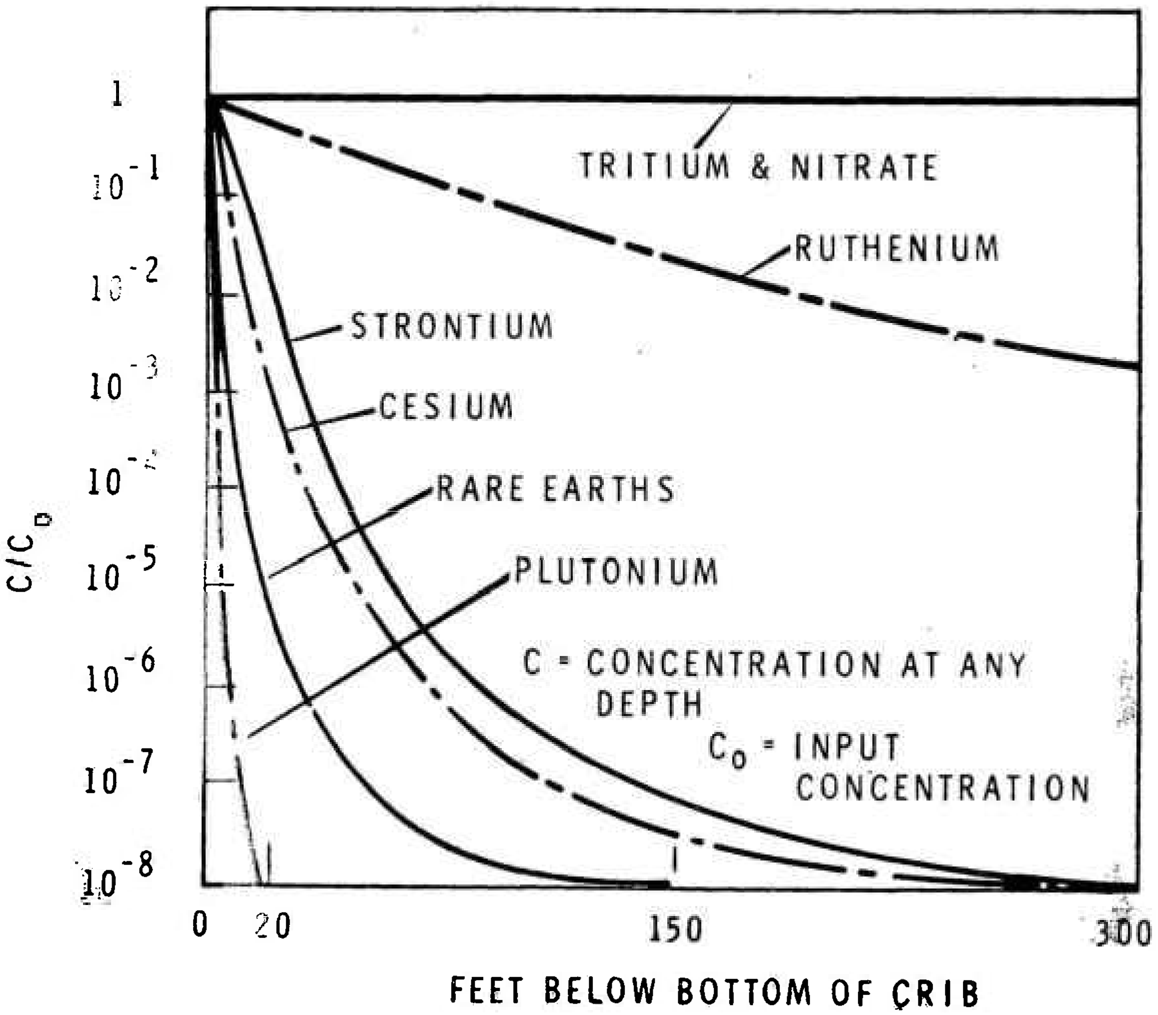


WINDSCALE HIGHLY ACTIVE STORAGE TANK.


50486-16
TYPICAL HANFORD DISPOSAL CRIB


SCHEMATIC OF DISPOSAL SITE
FOR INTERMEDIATE LEVEL WASTES
SURFACE
CLAY
SAND & SILT
CLAY
SAND & SILT
~150-300'
SAND, GRAVEL
& COBBLES
WATER TABLE

APPROXIMATE SORPTION PATTERNS TYPICAL HANFORD DISPOSAL CRIB



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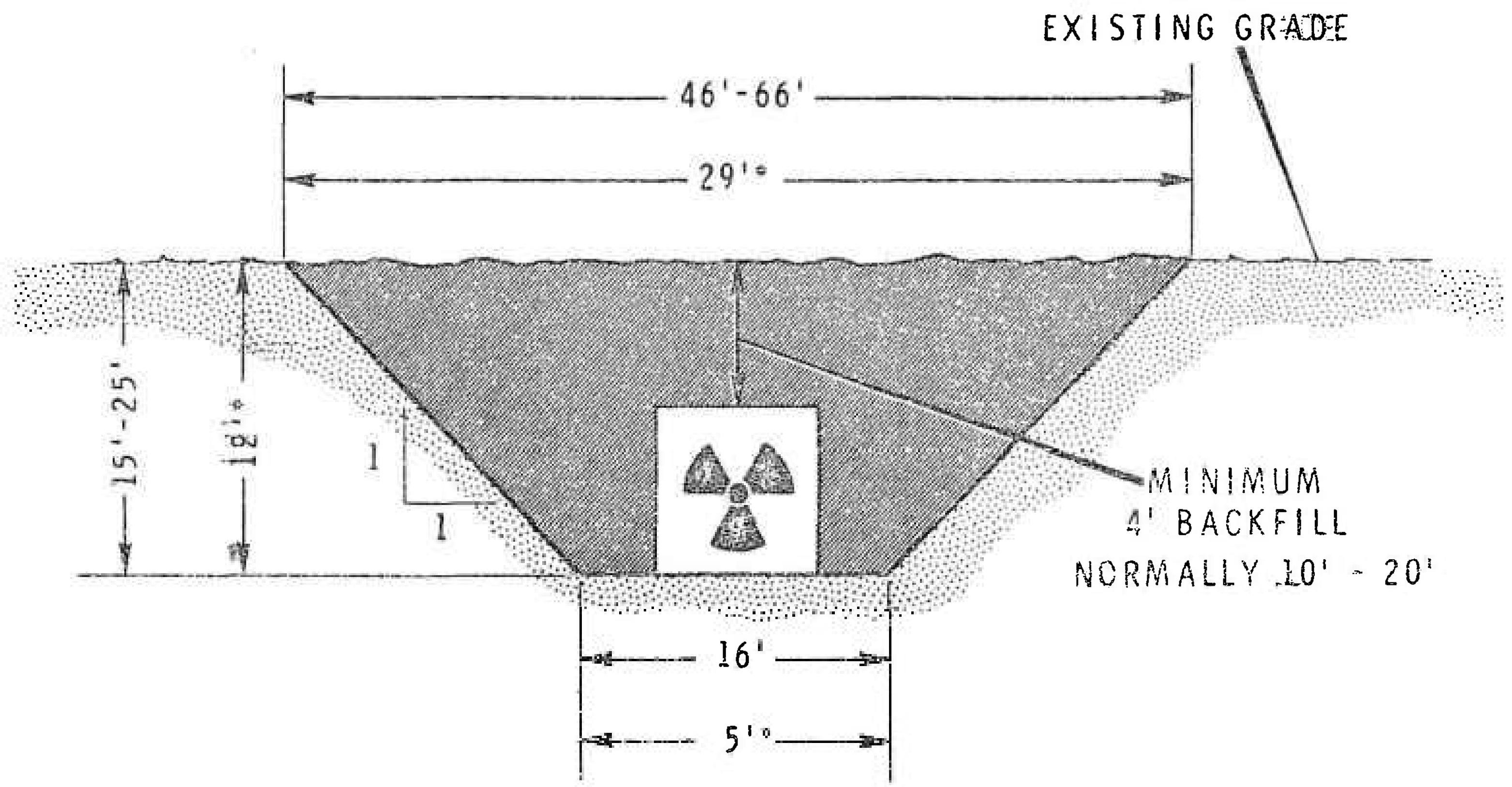
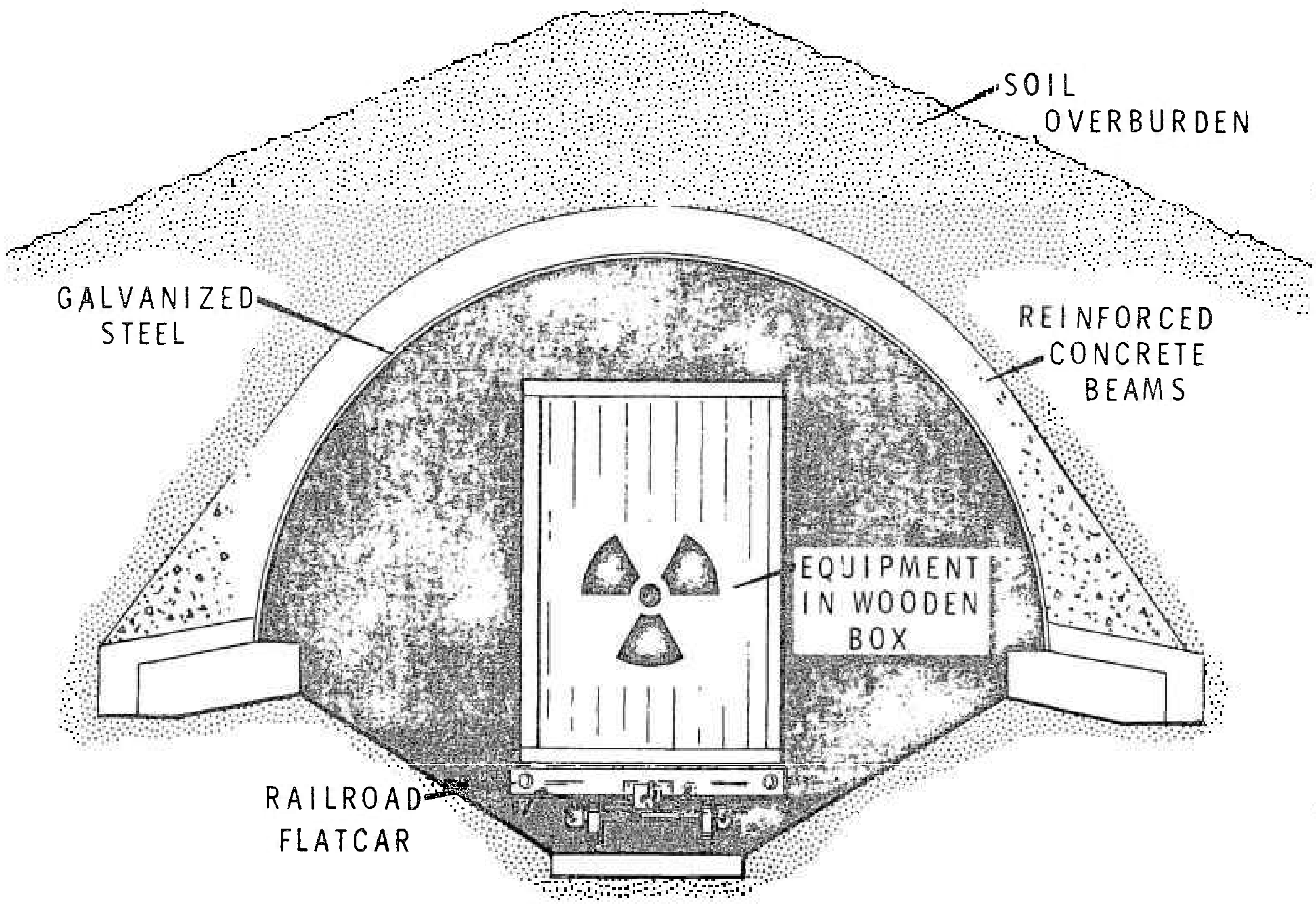
[illegible]

SOLID WASTE BURIAL TRENCHES

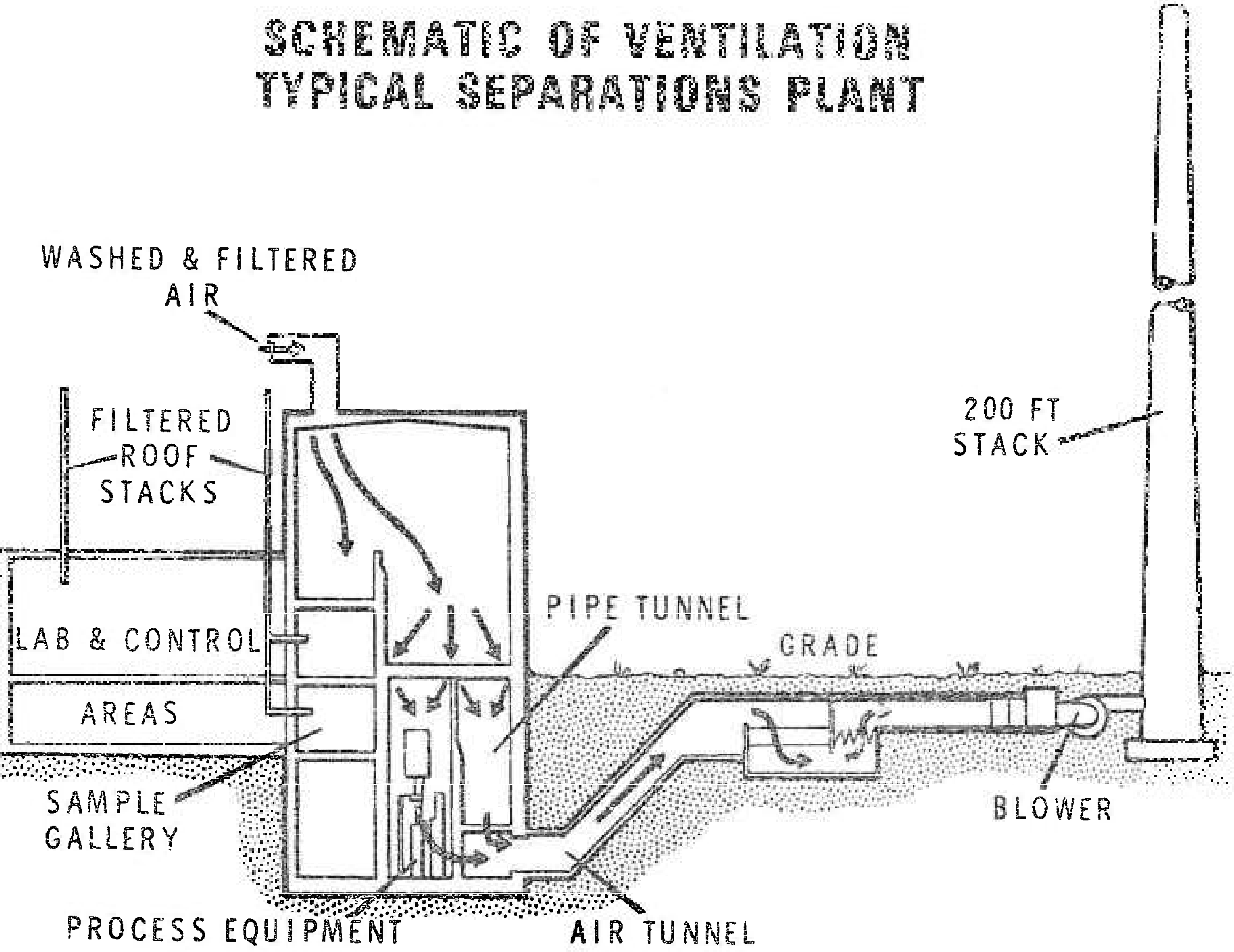
INDUSTRIAL AND DRY* WASTE



TUNNEL FOR STORAGE OF RADIOACTIVE EQUIPMENT



SCHEMATIC OF VENTILATION TYPICAL SEPARATIONS PLANT



VENTILATION FILTERS

TYPE	PLANT	RATED EFFICIENCY, %*
THREE STAGE PAPER	Z PLANT	99.99 +
TWO STAGE PAPER	B PLANT, AR VAULT	99.99
TWO STAGE GLASS WOOL	PUREX	99.9
SAND	UO_3, T PLANT	99.5

* FOR 0.3 MICRON PARTICLES

SOLIDIFICATION OF HIGH LEVEL WASTES – FUTURE PROSPECTS

R. E. Blanco

In the previous lectures we have seen that at present time all countries are storing their high-level wastes as liquids in tanks. However, it is generally recognized that it is desirable to convert these wastes to solids prior to sending them to the final waste storage or permanent disposal site. All countries which produce high-level wastes have carried out research and development on methods for converting these wastes to solids. Advantages for conversion of wastes to solids include (1) large volume reduction, (2) decrease in mobility, (3) production of insoluble solids, and (4) overall increase in safety during the initial interim storage period and in the final permanent disposal location.

The solidification processes being developed in all countries will be described, but particular emphasis will be given to the Waste Solidification Engineering Prototype (WSEP) pilot plant work being carried out at Battelle Northwest Laboratory (BNWL) at Richland, Washington, in the United States, since tests with wastes simulating that produced from reprocessing fuel exposed to 20,000 Mwd/ton are nearly complete and tests with wastes simulating advanced thermal reactor fuel exposed to 45,000 Mwd have begun. The WSEP program is a cooperative program with Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL) to demostrate and evaluate the processes developed at these installations, the pot calcination process at ORNL, the phosphate glass process at BNL, and the spray solidification process at BNWL.

This lecture contains excerpts from reports (principally BNWL-1083) prepared by A. G. Blasewitz, K. J. Schneider et al from BNWL and from the lecture presented by C. W. Christenson on Treatment of High-Level Wastes at the IAEA Conference in Rio de Janeiro. Many additional references are listed in the bibliographies of these reports (see references of end of lecture).

STATUS OF SOLIDIFICATION

The USAEC recently published a proposal in the United States federal register wherein private fuel reprocessors in the United States would be required to solidify their wastes within five years after they are produced and to ship them to a federal final disposal area within ten years. It is intended that the macro amounts of fission products would be removed continuously from the plant site and when operations at the plant are completed, that the fission product levels at the plant can be reduced to suitably low levels. This proposed policy is now being discussed with industry.

The United Kingdom (U. K) has no plans at present to initiate solidification of wastes although a process has been developed in the U. K France is making plans to solidify the wastes in the early 1970's as soon as the pilot plant program, now in progress at Marcoule, is completed. Belgium, West Germany, India, and the USSR are studying solidification but have not announced future plans.

PROCESSES DEVELOPED IN THE U.S.A.

WASTE TYPES

SLIDE 1 Hanford-Type Purex Wastes (Photo 48506)

This slide shows simulated wastes. (a) Relative volumes. (b) About factor 10 reduction from liquid to solids. (c) What is the safety and economic advantages of conversion to solids?

SLIDE 2 Conversion of Waste to Glass (Photo 53673-C)

(a) Typifies the conversion step. (b) Could also just evaporate to dryness and calcine to 900°C. This composition of product is not currently used. The slide is shown to illustrate conversion to glass.

SLIDE 3 Solidified High-Level Waste Samples from WSEP (Neg 0691648-1)

This slide illustrates the three types of product – ceramic, glass, and calcine. A powder composed of spherical particles of alumina (Al_2O_3) should also be shown to represent the product from fluid bed calcination.

SLIDE 4 Chemical Composition of High-Level Liquid Wastes for WSEP Demonstration (Table 3.1)

The TBP-25 waste is from research reactor fuels. (65-9305)

SLIDE 5 Examples of Glassy Solids Incorporating Oxides from Waste (65-11881)

This slide shows various types of phosphate glasses prepared at ORNL. Phosphates are favored in the U. S. These glasses are very insoluble and have relatively low melting points. Other countries such as France, the United Kingdom, and the USSR favor silicate glasses with higher melting points. Phosphate glasses are more corrosive to normal metals of construction but we use platinum melters on our pilot plant. This slide also shows an example of a low melting silicate glass we have studied. Lithium lowers the melting point of the glasses.

The solidified wastes from the spray and phosphate glass processes demonstrated in WSEP are monolithic melt-formed phosphate ceramics. Compared to silicate and borate melts, phosphate melts have the advantages of simple, all-liquid chemical adjustment of the waste solutions before solidification, easily handled hydrous precipitates in the chemically adjusted wastes, flexibility for a wide variety of waste compositions, and solid products with relatively low melting points. The disadvantages of phosphate melts are the high corrosion rates at temperatures greater than 875°C, and the approximately one order of magnitude higher leach rates. (Table 4.3)

ORNL POT CALCINATION PROCESS

SLIDE 6 Flowsheet for Converting High Activity Wastes to Solids by Pot Calcination (DWG 50885 R)

The pot calcination process, as developed at ORNL, consists of a calciner-evaporator loop and a fractionator. The waste is concentrated in an evaporator and the concentrate fed to a pot contained in a furnace. The pot overhead is condensed and recycled to the evaporator. All solids and essentially all of the radionuclides are retained in this loop and eventually end up in the pot. The evaporator and pot are regarded as a single unit. The overhead from the evaporator is redistilled in the fractionator to provide additional separation from radionuclides and to recover nitric acid for reuse.

SLIDE 7 ORNL Pots (Photo 54842)

This shows the pots developed in the ORNL experimental program. After filling, the filling tops would be removed and the pots sealed by welding a plate over the hole. They would then be shipped to the final disposal site. They are stainless steel 20 to 60 cm in diameter and 2.5 m high. One 30-cm-diam pot x 3 in. long holds 6.08 ft^3 – 15 pots/yr of glass would hold all high-level waste from a 1000 Mw(e) reactor at 10,000 or 20,000 Mwd/ton.

SLIDE 8 ORNL Hot Pot from Furnace (Photo 59579C)

Removing an experimental pot from furnace.

SLIDE 9 Pot Solidification System (Neg 0682348-2)

The pot solidification process is a batch system, and the principal processing vessel is the permanent storage pot. In pot calcination, liquid waste is added to a heated pot held at a nearly constant liquid level. After the waste is boiled for 5 to 10 hours, calcine scale begins to form on the pot wall. The calcine deposits grows inward and causes increased resistance to heat transfer, which in turn requires gradual reduction in feed rate until the pot is full of calcine except for a small core of thick sludge. At that time, feeding is stopped, and heating is continued until all the waste is converted into a calcine at about 900°C. The pot is then cooled and sealed for storage. The product for pot solidification is a mixture of oxides (and sulfate, if sulfate is present in the waste) of the mettalic constituents in the original liquid waste. The product is a porous, friable calcine with low thermal conductivity and relatively high solubility in aqueous solutions.

SLIDE 10 Spray Solidification System (Neg 0682348-3)

In spray solidification, liquid waste is fed into the top of a heated tower through a pneumatic atomizing nozzle. The spray is progressively dried and calcined to powder as it travels downward inside the reactor. The powder falls directly into the melter, while the process gas flows into the filter chamber, carrying along some of the finer waste powder as dust. The dust collects on porous metal filters and is periodically blown off the filters and into the melter by puffs of high pressure stream or air directed backward through the filters by small nozzles at the outlets of the filters. In the melter, the powder is melted at 700 to 1200°C. The molten waste then flows through an overflow weir tube or freeze valve into the receiver storage pot. The pot is then cooled and sealed for storage. The product from spray solidification is a monolithic solid formed after cooling the melt. The solid is a tough, microcrystalline, rock-like material with fairly good thermal conductivity and moderately low solubility in aqueous solutions.

SLIDE 11 Phosphate Glass Solidification System (Neg 0682348-1)

In continuous phosphate glass solidification, liquid waste is mixed with glass-forming chemicals (normally phosphoric acid) and concentrated by a factor of about five to a thick slurry in an evaporator. The slurry is fed to a continuous melter where final evaporation and glass formation occur at temperatures of 900 to 1200°C. The molten glass flows into the heated receiver-storage pot.

The pot is then cooled and sealed for storage. The product from the phosphate glass process is a monolithic, moderately brittle glass formed after cooling the melt. The glass has fairly good thermal conductivity and low solubility in aqueous solutions.

SLIDE 12 Status WSEP Radioactive Demonstrations of Solidification Methods (69-10577)

As of April 1, 1969, a total of 22 demonstrations have been completed in WSEP, 6 in each of the pot and phosphate glass solidification processes, and 10 with the spray solidification process. Approximately 30 megacuries of mixed radionuclides representing waste from about 232,000 Mwd of electrical power (assuming 33% thermal efficiency and 20,000 Mwd/tonne have been converted to solids. This amount of waste represents 88 days of all the nuclear power capacity in the United States as of March 1, 1969, about 5 days of all the nuclear power contracted to be on line by the end of 1973, and about 1.6 days of all the nuclear power anticipated in 1980. The total on-line processing time for solidification of these wastes was 56 days. The average on-line processing rates vary from 0.3 to 1.0 tonnes/day of fuel, depending upon the waste composition, the process, and the chemical flowsheet used.

The maximum amount of heat from radionuclide decay that can be incorporated into 8-in.-diam pots which are air-cooled by natural convection has been demonstrated for the spray and pot processes. The heat-rate values in the solidified wastes range from the highest of 205 W/liter for the spray solid to 85 W/liter for pot calcine. The out-of-reactor time represented by those maximum heat rates is less than half a year.

SLIDE 13 Characteristics of Solidified High-Level Waste (Part 1) (69-10578)

The next two slides summarize the characteristics of the waste products obtained in the WSEP program. The products from the three processes can be described as calcine cake for the pot calcination process, microcrystalline or rocklike for the spray solidification process, and glass for the phosphate glass process. The pot calcine product has a high leach rate when compared with the spray solidifier and phosphate glass products. The pot calcine material may be more stable at elevated temperatures. The spray and phosphate glass melts can be stored in mild steel containers; stainless steel containers must be used in the pot calcination process. Because of the differences in heat conductivity, the maximum fission product heat per unit volume which can be stored in pot calcine product is less than half that which can be contained in the melt products.

SLIDE 14 Characteristics of Solidified High-Level Waste (Part 2) (69-10579)

SLIDE 15 Waste Solidification Engineering Prototypes (67-9416)

The slide shows the flowsheet for the three processes in the WSEP. Each process can be operated separately. The volatile materials pass to the evaporator and then to the fractionator and off-gas cleanup units is sequence.

Cleanup of the gaseous effluents from solidification with a WSEP type of cleanup system has been demonstrated to readily meet requirements in 10CFR20 for gases. The aqueous effluents for the WSEP system contain up to one million times the allowable radionuclide content for drinking water, but these effluents should be readily returnable to similar but larger systems required in fuel reprocessing plants for treating other similar effluents. The waste solidification demonstration program has been progressing very satisfactorily.

Although evaluations are not complete, the major conclusion from the program is that solidification of high-level aqueous wastes by the pot, spray, and phosphate glass processes is feasible and essentially ready for application by commercial processors of spent nuclear fuels.

None of the three processes have been selected as the best process for all uses. Each process and its solidified waste has advantages and disadvantages which must be weighed carefully for each specific application. A list of advantages and disadvantages are shown in Table I.

The solubility of the waste products perhaps requires additional comment. The glass and microcrystalline products products are very insoluble in water, whereas the calcine is moderately soluble. The philosophy in support of calcination is that the product will be placed in a final disposal location, such as a salt mine, where water has not been present for thousands of years and water will not be present again in the foreseeable future. Thus, the solubility of the product, or container integrity, is of minor importance in that location. Proponents of glassy products state that some additional safety (in addition to the 1.25 cm thick stainless steel pot) is incurred during interim storage. All are agreed that the higher thermal conductivity of the glassy products is a substantial advantage.

The next series of slides show equipment in the WSEP installation at BNWL, Richland, Washington, USA.

SLIDE 16 Plan View of B Cell Equipment Arrangement (Neg 0653173)

This cell is about 6.7 m x 7.6 m x 9.3 m high and the concrete walls are about 1,6 m thick.

SLIDE 17 Looking Down on WSEP Equipment in Cell (Neg 0662668-7)

SLIDE 18 Looking Into WSEP Equipment Cell (Neg 0662668-21)

SLIDE 19 Platinum Melter (2-231-66)

The platinum melter is about .75 m hugh x .3 m in diameter. It is a nionel vessel with a 1.5-mm-thick platinum liner.

SLIDE 20 WSEP Containers for Solidified Waste (Neg 41607-2)

SLIDE 21 Environmental Test Pods Before Installation in SSETF (Neg 0673128-6)

The Solids Engineering Test Facility (SSETF) is available for evaluating the products. Samples can be taken of the material during the filling of the receiver pot for the two processes wherein the product is in a molten condition at the time it is placed in the receiver. These samples can be taken to a shielded chemical laboratory cell for measurements of leachability, crystallinity, and dispersibility. When filled with solidified waste, the pots are sealed, weighed, dimensionally checked, leak tested, ultrasonically scanned for pot wall tickness, and calorimetry checked. The pots are then placed in interim storage; temperature and pressure data are periodically monitored during this period.

Some of the receiver pots will be placed into controled environment storage. This involves placing the receiver pot in an environmental storage pod wherein different storage media can be used and different storage temperatures can be maintained. Data will be monitored continuously regarding the environmental storage conditions, temperature distribution throughout the solidified waste, and pressure within the receiver pot. Samples can also be taken from the vapor space above the solidified waste. The capability will exist to periodically remove the pot from the environmental pod, drill into the pot and remove a sample of the solidified waste.

Table 1. *Comparison of Pot, Spray, and Phosphate Glass Solidification Processes*

	Advantages			Disadvantages		
	Pot	Spray	Phosphate Glass	Pot	Spray	Phosphate Glass
1. **System**	Simple				Complex	Complex
2. **Type** —Amenability to Scale Up		Continuous Good	Continuous Good	Batch Limited		
3. Flexibility for Waste Variations	Good				Limited	Limited
4. R&D Required for New Wastes	Minimum				Moderate	Moderate
5. In-Process Inventory	Low	Lowest				Moderate
6. Specific Restrictions					Requires Platinum Melter Requires Good Waste Flow Control High Ruthenium Volatilization	Requires Platinum Melter Requires Good Concentrate Flow Control Does Not Retain Sulfate in Solid
7. Product Leachability		Low	Lowest if Glass	High		
8. Thermal Conductivity		Highest	Good	Poor		
9. Toughness		Tough		Friable		Brittle
10. Storage Container		Mild Steel	Mild Steel	Stainless Steel		

SLIDE 22 Solids Storage Engineering Test Facility (Neg 067 1892-3)

SLIDE 23 View into Cell Housing Waste Solidification Engineering Prototypes Through Air Lock (PNL 066 1893-4)

This illustrates the size of the equipment.

SLIDE 24 View of Pot After Filling with Radioactive Waste by Pot Calcination Porcess (PNL 43634-2)

SLIDE 25 Purex Solids in Pot After Pot Calcination (Photo 49212)

SLIDE 26 Fluid Bed Waste Calcining Facility (69-10575)

The first method of waste solidification adapted to a plant scale was that of continuous calcination in a fluidized bed at the National Reactor Testing Station, U. S. A. The process was originally conceived at Argonne National Laboratory and first applied at the Idaho Chemical Processing Plant (ICPP).

The heart of the Waste Calcining Facility (WCF) process is the 48-in.-diam fluidized bed shown schematically in this slide. Solutions are injected through pneumatic atomizing spray nozzles into a heated bed of granular solids fluidized with preheated air. The feed, which contains a full range of aged fission products, is divided equally among three nozzles located about six inches below the surface of the bed. Metallic oxides or fluorides coat the bed particles, and gaseous decomposition products are carried with the fluidizing and atomizing air to the off-gas scrubbing system.

Heat required for decomposition of the calciner feed is supplied through an indirect heat transfer system which uses the eutectic mixture of sodium and potassium (NaK) as the heat transfer fluid. The fluidized bed is maintained at the proper temperature, usually 400°C, by controlling the temperature of the NaK. NaK is heated in an oil-fired furnace and pumped through the calciner heat exchanger which contains 42 bayonet tubes protruding horizontally into the bed.

The solid product is transported pneumatically to solids storage bins and enters the transport air line from two sources: the first consists of small bed particles and fines elutriated from the bed and removed from the off-gas in the calciner cyclone, and the second consists of bed particles removed from near the bottom of the bed at a controlled rate for maintaining a constant bed level. At the storage area, the solids are removed in a second cyclone above the storage bins and fall by gravity into the bins. The transport air returns to the top of the calciner to be decontaminated with the calciner off-gas.

Separation of the WCF off-gas from radioactive particulate matter and volatile species is accomplished in a series of devices which include (in sequence) a spray quench tower, a venturi-scrubber and associated cyclone-type entrainment separators, heaters, silica-gel adsorbers, and high-efficiency filters. Approximately 80 percent of the water in the calciner feed is carried with the off-gas by adjustment of the gas humidity; the remaining 20 percent is used for scrubbing particulate matter from the off-gas as well as for control of the off-gas temperature and is then recycled batchwise to the feed tank for reprocessing. The heaters superheat the off-gas to prevent condensation in the adsorbers and filters. The silica gel adsorbs volatile ruthenium species, and the filters remove the remaining particulate matter before the gas is discharged through the stack, 76 meters in height.

The characteristics of the feed and product are given in Tables II and III, and operating data are shown in Table IV.

TABLE 2. TYPICAL FEED COMPOSITIONS

		Radioactive 48-inch Unit	
Component		Zirconium Fluoride	Aluminum Nitrate
H^+	(M)	1.76	1.10
Zr^{+4}	(M)	0.12	
F^-	(M)	1.08	0
Al^{+3}	(M)	0.47	1.70
NO_3^-	(M)	4.24	6.20
B^{+3}	(g/l)	1.00	0.1
Ca^{+2}	(M)	0.78	0
Undissolved Solids	(g/l)	31	0.1
Cs^{137}	mCi/l	500	1800
Ru^{106}	mCi/l	30	600

TABLE 3. PROPERTIES OF THE PRODUCT FROM THE CALCINER

	Radioactive	
Physical Properties:	Zirconium	Aluminum
Mass Median particle diameter, mm	0.75	0.6
Bulk density, g/cc	1.70	1.1
Chemical Properties:		
Aluminum, as Al_2O_3 (wt.%)	21.9	89.0
Zirconium, as ZrO_2	21.4	0
Nitrogen, as N_2O_5	1.0	4.0
Calcium, as CaF_2	54.2	0
Fluorine, as CaF_2	53.6	0
Radiochemical Properties:		
Heat generation, cal/hr-kg	55	222
Cs^{137} Ci/kg	2.8	17.6
Ru^{106} Ci/kg	0.03	0.1

TABLE 4. CALCINER DATA

<u>Feed Rates:</u>	<u>Zirconium Waste</u>	<u>Aluminum Wastc</u>
Gross to Calciner, l/hr	350	360
Recycled scrubbing solution, l/hr	90	94
Net to calciner, l/hr	260	264
Net from liquid storage tanks	200	250
<u>Product Rates: (kg/hr)</u>		
To solids storage	35	23
To scrubbing solution	11	6
<u>Calcination Conditions:</u>		
Superficial Fluidizing Velocity, m/sec	0.37	0.28
Volume ratio of nozzle air to feed rate	725	600
Bed temperature, °C	400°C	400°C
NaK inlet temperature, °C	735°C	725°C
<u>Decontamination Factors:</u>		
Particulate matter (based on Cs-137)	1×10^8 (min.)	1×10^8 (avg.
Volatile (based on Ru-106)	8×10^4 (min.)	1×10^3 (avg.

The overall decontamination factors for particulate matter from the raw feed to the stack were slightly better with zirconium than with aluminum waste. A minimum decontamination factor (df) of 1.1×10^8 was obtained for zirconium while the average value for aluminum waste was 1.0×10^8. Radiochemical analyses for cesium-137 were used to determine these dfs. The cesium release rate to the environment generally was less than 0.02 millicuries per day. For volatile ruthenium, the df with zirconium waste always exceeded 8×10^4 as compared to that with aluminum waste of 1.0×10^3. The release rate of ruthenium generally was less than one millicurie per day. The concentration of all radionuclides in the off-gas discharged from the ICPP stack was below the established guide limits for continuous exposure.

The WCF process has operated successfully and reduced large volumes of aluminum and zirconium-type wastes to solids. This is the only solidification process that has been used on a routine basis. Unfortunately, it has never been operated with high-level Purex wastes. Purex processing has been confined to a few early small-scale runs. Also the aluminum and zirconium wastes contain 100 to 1,000 times lower concentrations of fission products than power reactor Purex wastes. Therefore, little is known regarding the ability of the process to handle solids with a high heat production rate and to decontamine the off-gases. Therefore, comparisons with the order candidate processes is difficult. The high processing rate is a definite advantage and the low thermal conductivity of the product powder is a disadvantage.

SLIDE 27 Simplified Flowsheet for Incorporation of Fission Products Into Glass (69-10574)

The Fingal process was developed in the United Kingdom (U. K.) for the conversion of wastes from the Windscale separation plant into a leach resistance glass for long-term storage. Concentrated radioactive waste solution, together with glass-forming materials such as silica and borax, are fed into the first of three stainless steel vessels; this vessel is held at a temperature of 1050°C and evaporation, denitration, sintering and glass formation occur steadily during the loading cycle. The off-gases from the glass-making vessel are passed through second and third vessels which contain a small primary and a large secondary filter respectively. These filters are packed with ferric oxide adsorber for volatile ruthenium but most of the particulate material carried over is trapped by the primary filter. At the end of the process cycle the cylinder containing the glass is removed to storage and the cylinder in the second position containing the primary filter is moved into the furnace. A new cylinder with a new filter is put in the middle position. At the beginning of the next cycle of operations as the furnace temperature is raised, a fusible link allows the filter to fall to the bottom of the vessel and this filter together with its accumalated dust, is incorporated into the next block of glass. The off-gases from the three cylinders pass directly to a condenser and scurbbing system. The entire process equipment is maintained below atmospheric pressure by vacuun ejector.

SLIDE 28 Pot Vitrification Pilot Plant at Marcoule (69-10576)

The research carried out in France on the solidification of fission product solutions involve wastes of three distinct types:

a) Solutions from the processing of natural uranium rods from Marcoule. A silicate glass is proposed which is produced by a pot vitrification process.

b) Solutions from the processing of uranium-molybdenum alloy fuels from power reactors. A silico-phosphate glass is proposed and studies are being carried out on both the pot vitrification process and a continuous process.

c) Solutions from the dissolution of uranium aluminum fuels from experimental reactors. Phosphate glasses are proposed for these wastes after removal of the mercury.

Studies on the pot vitrification process are nearing completion at Fontenay-aux-Roses and a pilot unit is being built at Marcoule. This plant will handle the high-level wastes which are currently being stored in tanks. The layout of the equipment is illustrated in this slide. Radioactive waste solutions and a suspension containing the glass additives are fed directly into the top of a pot 2 m high and 25 cm diameter. The feed is first calcined at about 600°C until the pot is three quarters full. The temperature is then raised to 1200°C to melt the calcine to glass and the glass is transferred to a final storage container by heating the bottom outlet of the pot. In this way it is expected that the pot can be used at least 30 times before it has to be replaced. The off-gases pass through a ruthenium adsorber and are then condensed. The ruthenium adsorber consists of ferruginous granules and these are recycled to the melt pot. The condensate is evaporated and the concentrate recycled to the pot.

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4. E. J. Tuthill, G. G. Weth, L. C. Emma, Gerald Strickland, and L. P. Hatch, "Phosphate Glass Process for Disposal of High-Level Radioactive Wastes," I&EC Process Design and Development 6, 1967.

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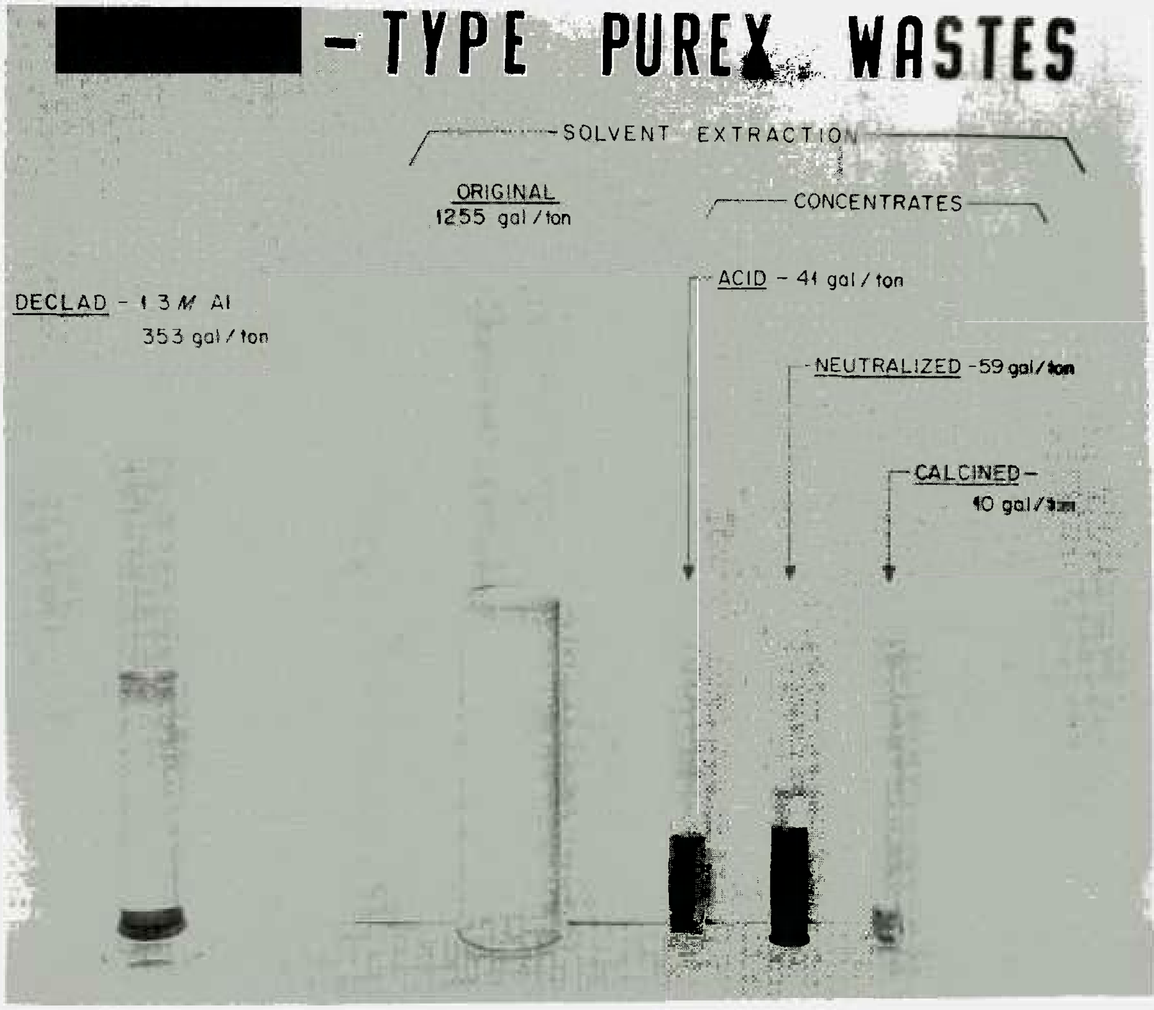
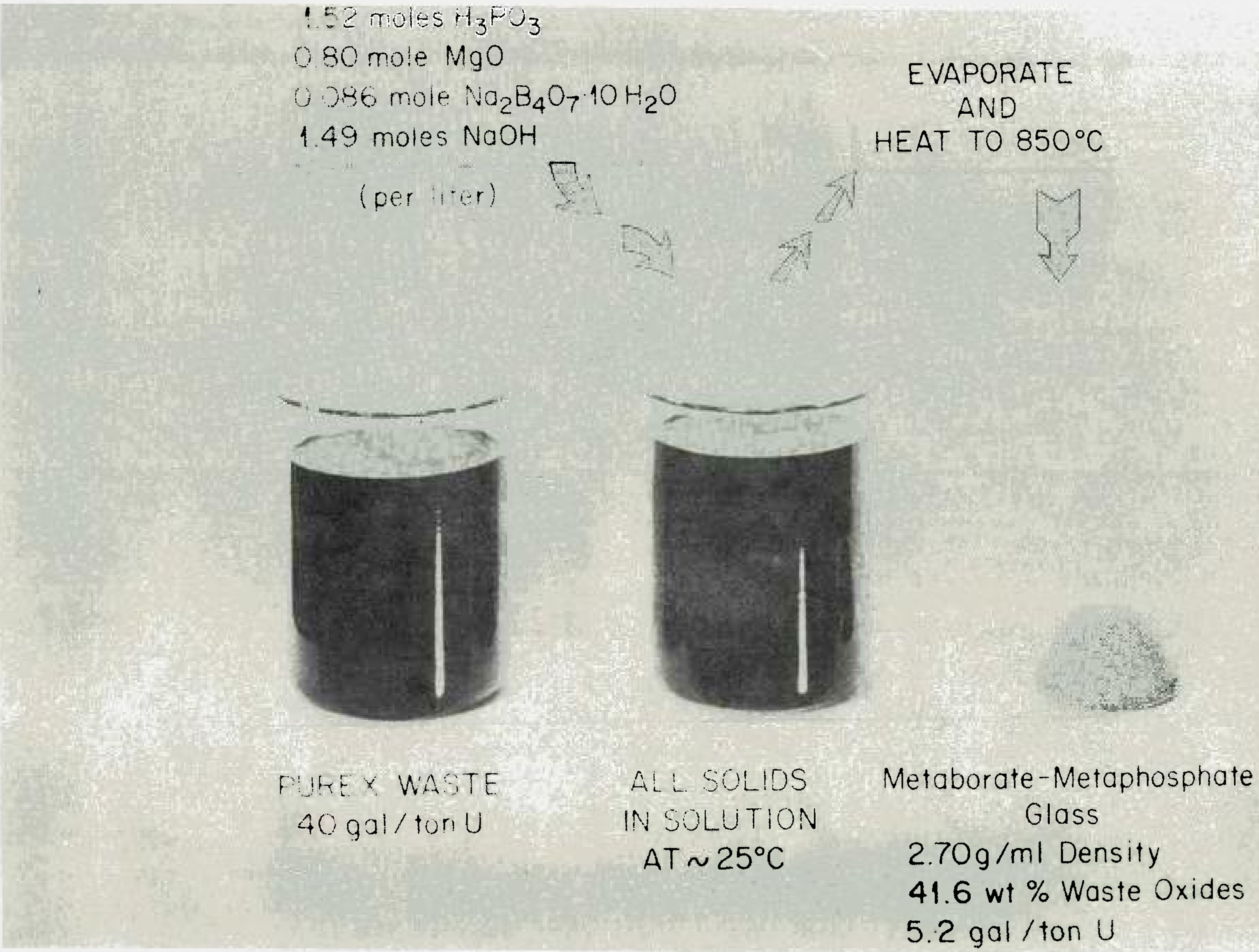
8. W. H. Hardwick, "Present Practices and Future Thoughts on High Level Liquid Waste Management," IAEA Advanced Training Seminar in Waste Management, Oxford, September 1967.



[redacted] – TYPE PUREX WASTES


1.52 moles H_3PO_3
0.80 mole MgO
0.086 mole $Na_2B_4O_7 \cdot 10\,H_2O$
1.49 moles NaOH
(per liter)
EVAPORATE AND HEAT TO 850°C
PUREX WASTE
40 gal/ton U
ALL SOLIDS IN SOLUTION AT ~ 25°C
Metaborate-Metaphosphate Glass
2.70 g/ml Density
41.6 wt % Waste Oxides
5.2 gal/ton U

Neg 0691648-1 (16-3)

Spray Solidification

Phosphate Glass

Pot Calcination

SOLIDIFIED HIGH-LEVEL WASTE SAMPLES FROM WSEP

COMPOSITION RANGES OF SIMULATED WASTES

Cons't. (g·moles/liter)	High SO_4^2 (1 WW; FTW-65)	High Fe (Purex No. 1; Darex)	Aluminum (TBP-25)
H^+	0.5-5.6	2.0-5.0	1.26
Na^+	0.3-0.9	0-0.14	0.1
Al^{3+}	0.05-0.15	0-0.001	1.6
Fe^{3+}	0.1-0.5	0.9-1.3	0.002
Cr^{3+}	0.01-0.06	0.1-0.4	–
Ni^{2+}	0.01-0.03	0.05-0.2	–
SO_4^{2+}	0.15-1.0	–	0.03
NO_3^-	1.0-6.1	7.2-8.8	5.4
Sim. FPs. (max.)	0.144	0.397	0.002

Plus varying traces (>0.01 $\underline{M}$) of Hg^{2+}, PO_4^{3-}, SiO_3^{2-}, F^-, Cl^-

TABLE 3.1. *Chemical Compositions of High-Level Liquid Wastes for WSEP Demonstration*

Constituent	Concentration, Molarity at 378 liters/tonne[(a)] PW-1	PW-2	PW-4m	LMFBR
General Chemical Composition of Inert Materials				
Na	low	high	low	low
Fe	high	med	low	low
SO_4	0	high	0	0
Actual Chemical Composition of Inert Materials				
H	3.70	3.90	0.50	0.50
Fe	0.93	0.44	0.050	0.16
Cr	0.012	0.024	0.012	0.046
Ni	0.005	0.010	0.008	0.023
Al	0.001	0.001	0.001	--
Na	0.14	0.93	0.10	0.10
U	0.010	0.010	0.010	0.010
Hg	<0.001	<0.001	<0.001	--
NO_3	7.5	5.4	2.4	4.7
SO_4	--	0.87	--	--
PO_4	0.003	0.006	0.003	0.060
SiO_3	0.010	0.010	0.010	--
F	<0.001	<0.001	<0.001	--
$\sum M^+_{chem}$ (b)	3.03	2.48	0.37	0.83
kg oxide/tonne	31.7	28.1(c)	4.6	9.7

Chemical Composition of Major Materials from Nuclear Fission

Constituent	Fuel Exposure in Thermal Reactors 20,000 MWd/tonne at 15 MW/tonne	45,000 MWd/tonne at 30 MW/tonne	Core Fuel Exposure in LMFBR 100,000 MWd/tonne at 200 MW/tonne
Mo	0.065	0.130	0.24
Tc	0.014	0.031	0.060
Sr	0.015	0.036	0.040
Ba	0.019	0.041	0.068
Cs	0.035	0.078	0.21
Rb	0.007	0.014	0.019
Y+RE(d)	0.12	0.27	0.53
Zr	0.065	0.14	0.22
Ru	0.032	0.082	0.22
Rh	0.007	0.013	0.060
Pd	0.017	0.043	0.16
Ag	0.0008	0.0016	0.019
Cd	0.0008	0.0025	0.010
Te	0.006	0.014	0.034
Sn	0.0007	0.0016	0.010
Sb	0.0002	0.0004	0.003
$\sum M^+_{fp}$ (b)	0.91	1.73	4.10
kg oxide/tonne	22.0	49.0	101.1

(a) *Tonne is a metric tonne, 1000 kg or 2205 lb.*

(b) *M^+ is metal equivalents, or normality of metal ions (does not include acid).*

(c) *Does not include the sulfate. If sulfate is not volatilized, approximately 27 kg/tonne of additional oxides are formed.*

(d) *RE is rare earth elements.*

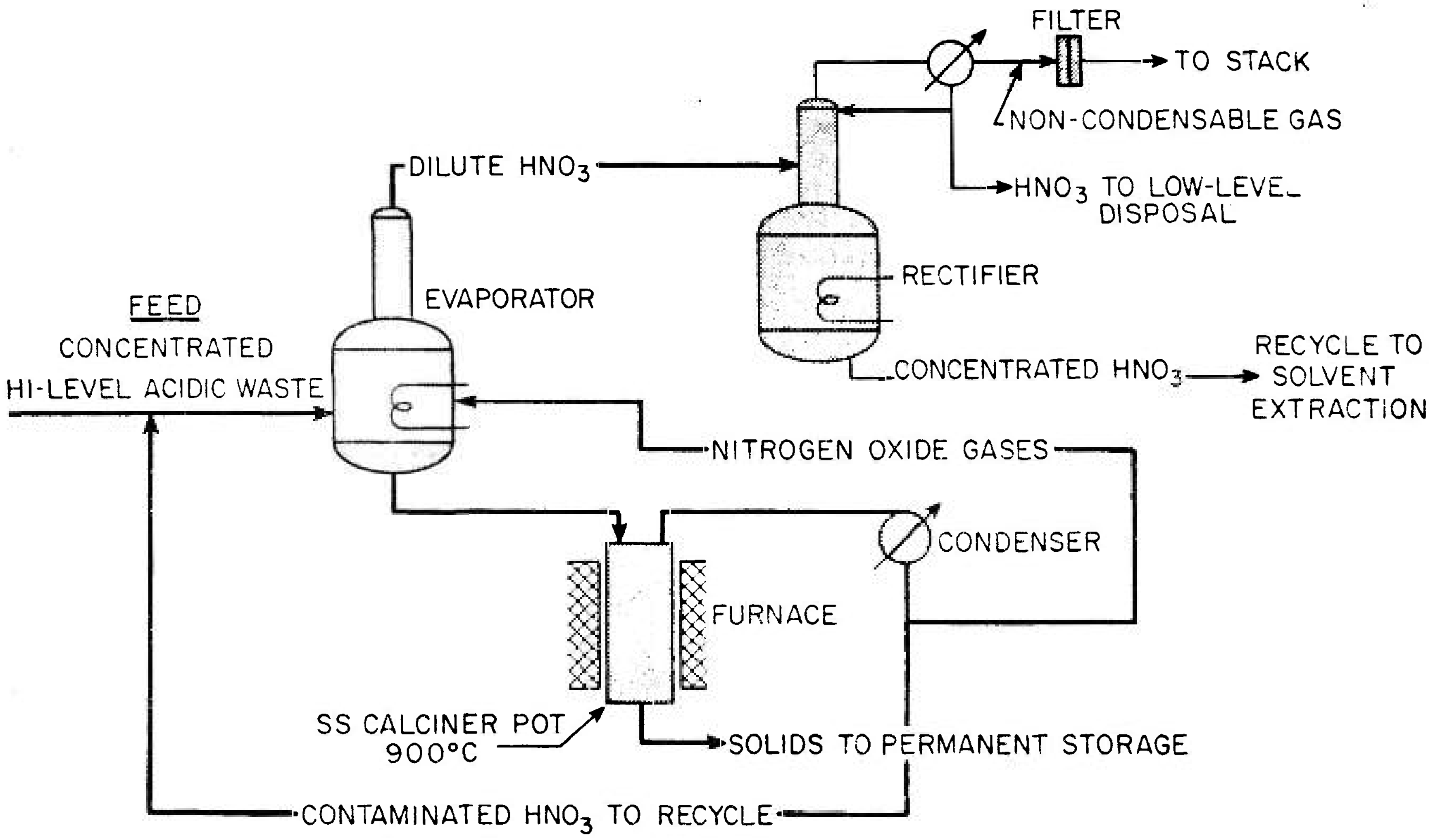
EXAMPLES OF GLASSY SOLIDS INCORPORATING OXIDES FROM WASTE

Additives (g-moles added to a liter of waste)	Waste Types			
	Purex-1*	Purex-1*	Purex-2*	TBP-25
Phosphate	2.0	–	2.5	2.0
Silicate	–	0.8	–	–
Borate	–	0.8	–	–
Sodium	1.1	1.0	1.38	2.0
Lithium	1.1	–	1.38	–
Calcium	0.1	–	0.2	–
Lead	–	0.9	–	0.25
Melt temp (°C)	900	850	800	900
Waste oxides (wt%)	40	30	43	26
$\left(\frac{\text{Volumes waste}}{\text{Volume glass}}\right)$	9	11	7	8

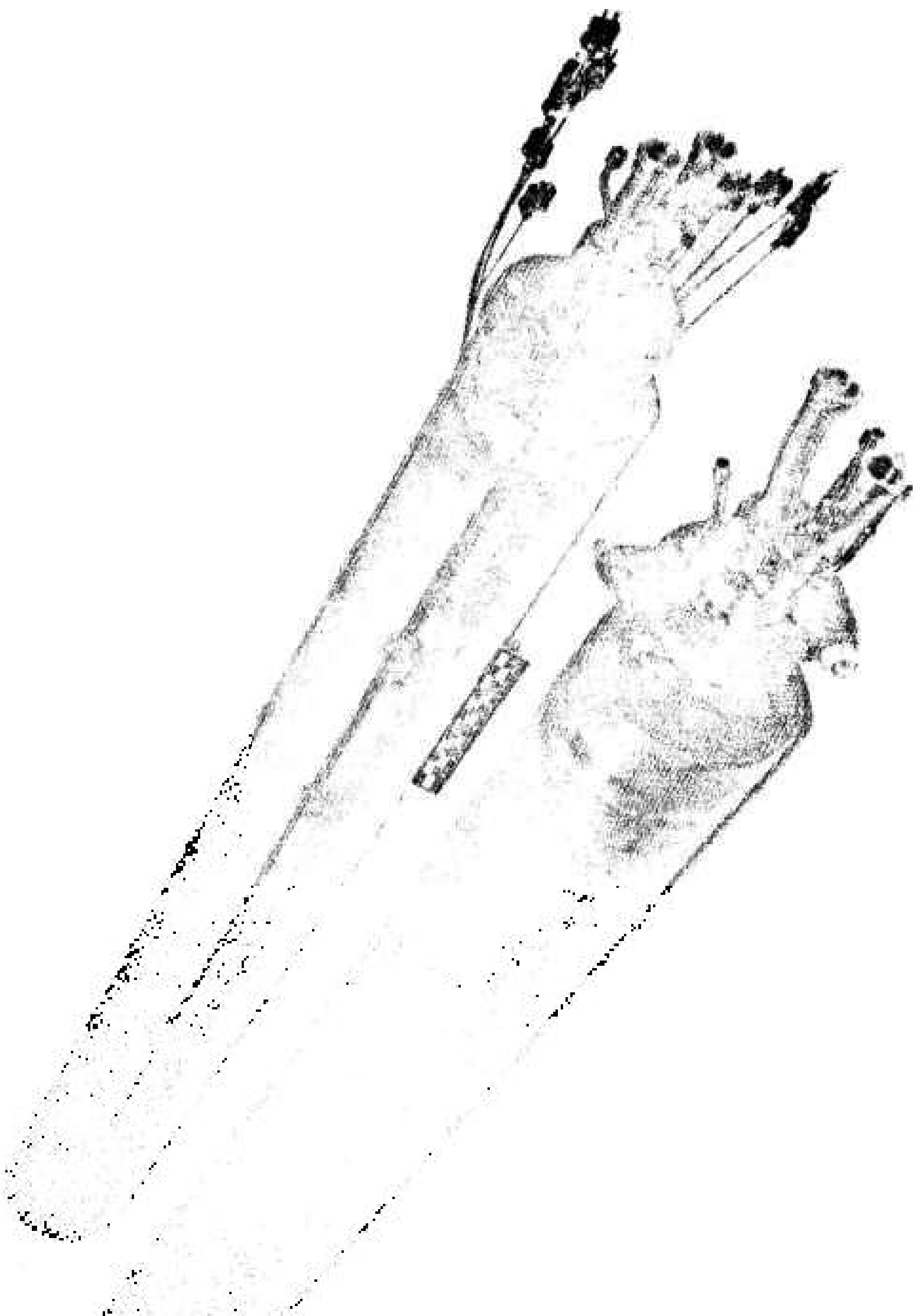
*Simulated fission products expected from fuel burned to 20,000 Mwd/tonne.

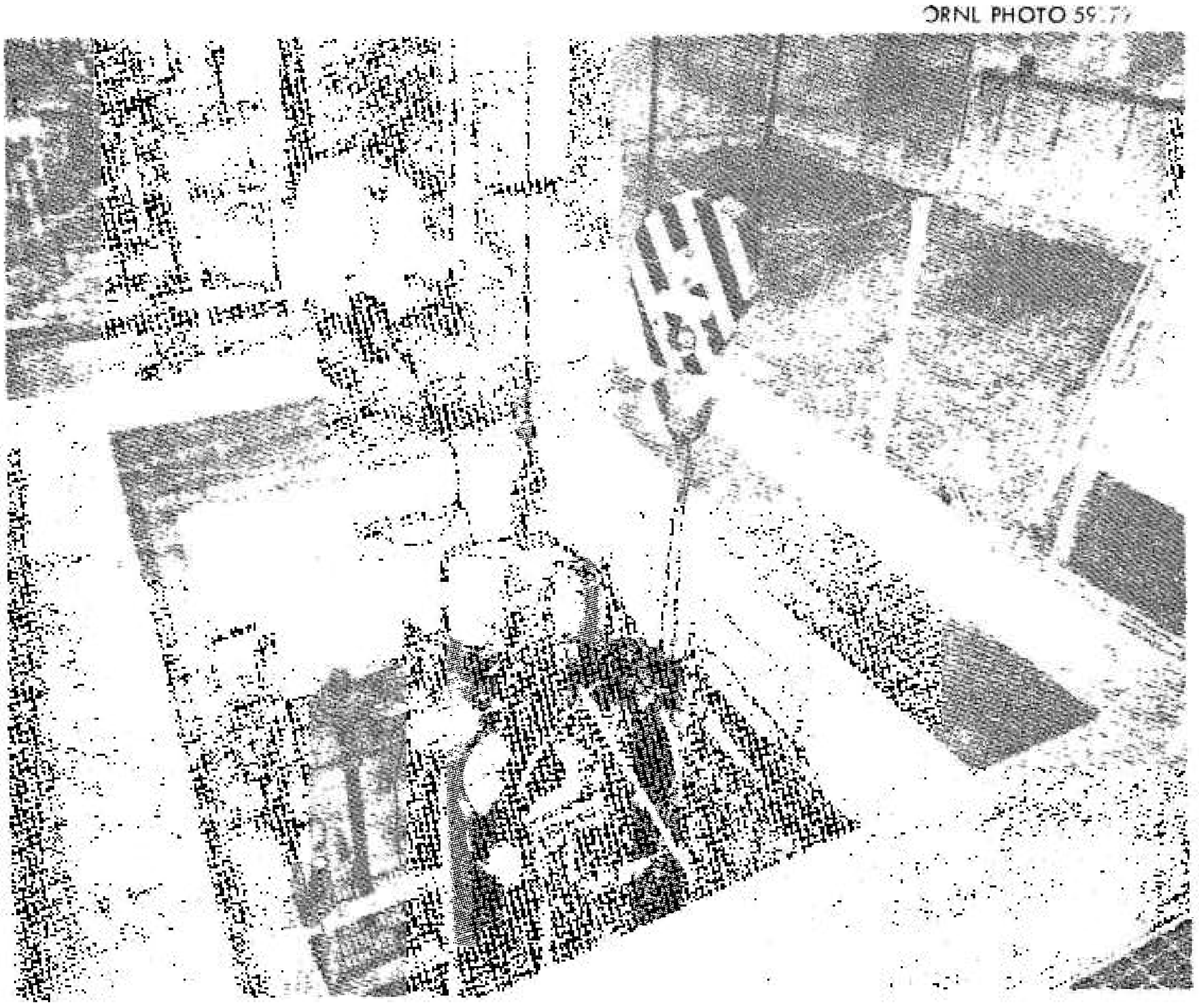
TABLE 4.3. Phosphate Melt Compositions Used in the WSEP Demonstrations

Additives, Moles per Liter of Waste at 100 gal/tonne	Spray Solidifier Process			Phosphate Glass Process		
	PW-1	PW-2	PW-4m	PW-1	[illegible]	[illegible]
Na	0.5		0.75	2.45		[illegible]
Li	0.5	1.17	0.75		[illegible]	
[illegible]	--		1.1	[illegible]	--	[illegible]
Ca	--	[illegible]			[illegible]	[illegible]
Al	--	0.25		[illegible]	--	--
H_3PO_4	1.74	1.61	2.80	6.23	5.36	[illegible]
$NaPO_3$	--	1.27 (in melter)	--	--	--	
	(M/P = 2.75)	(M/P = 2.7)	(M/P = 2.50)	(M/P = 1.0)	(M/P = 1.0)	(M/P = 1.0)
Product Composition, Oxide wt%						
Fission Products	18	10	25	8	14	16
Waste Inerts	30	28	4	13	21	5
Melt Forming Additives	52	62	71	79	65	79



FLOWSHEET FOR CONVERTING HIGH ACTIVITY WASTES TO SOLIDS BY POT CALCINATION

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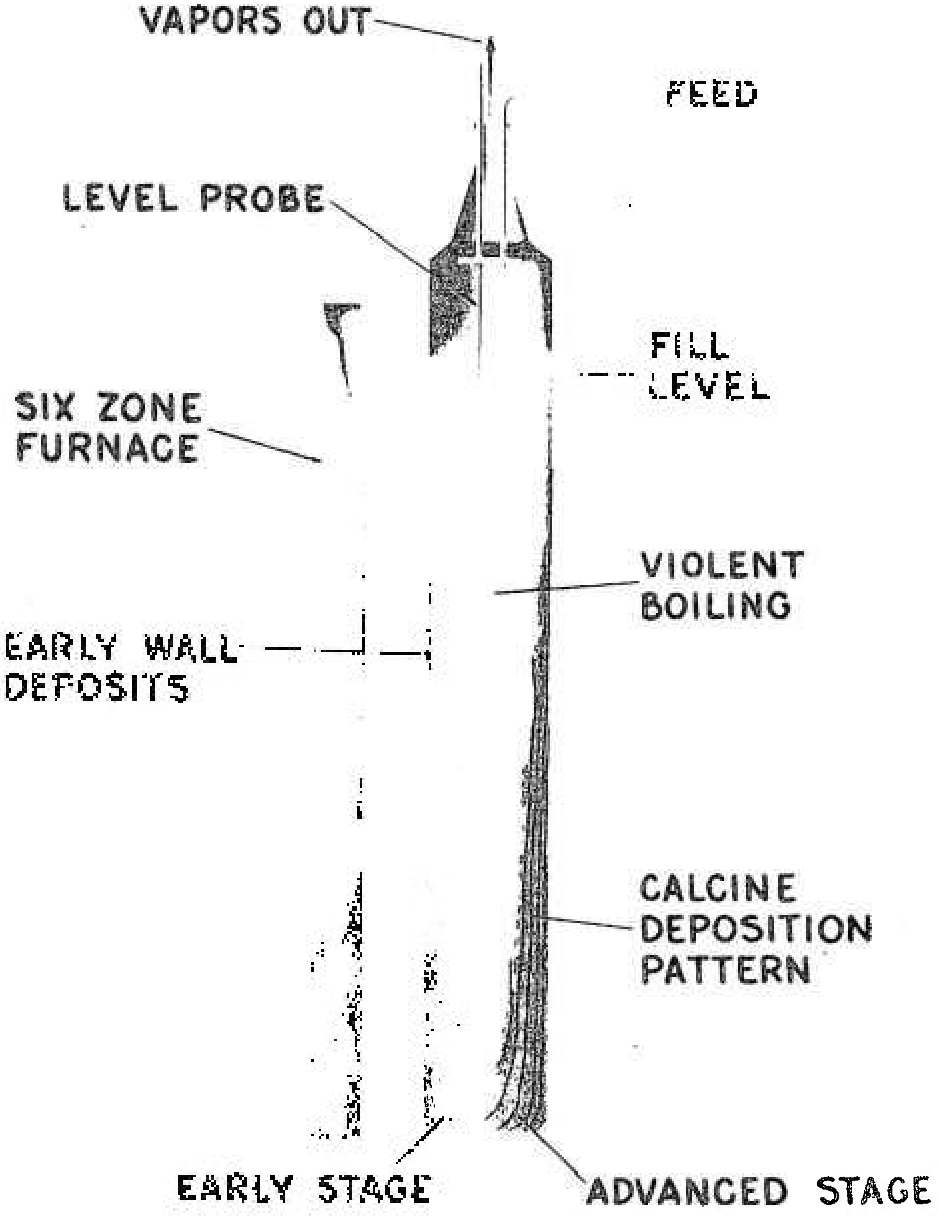
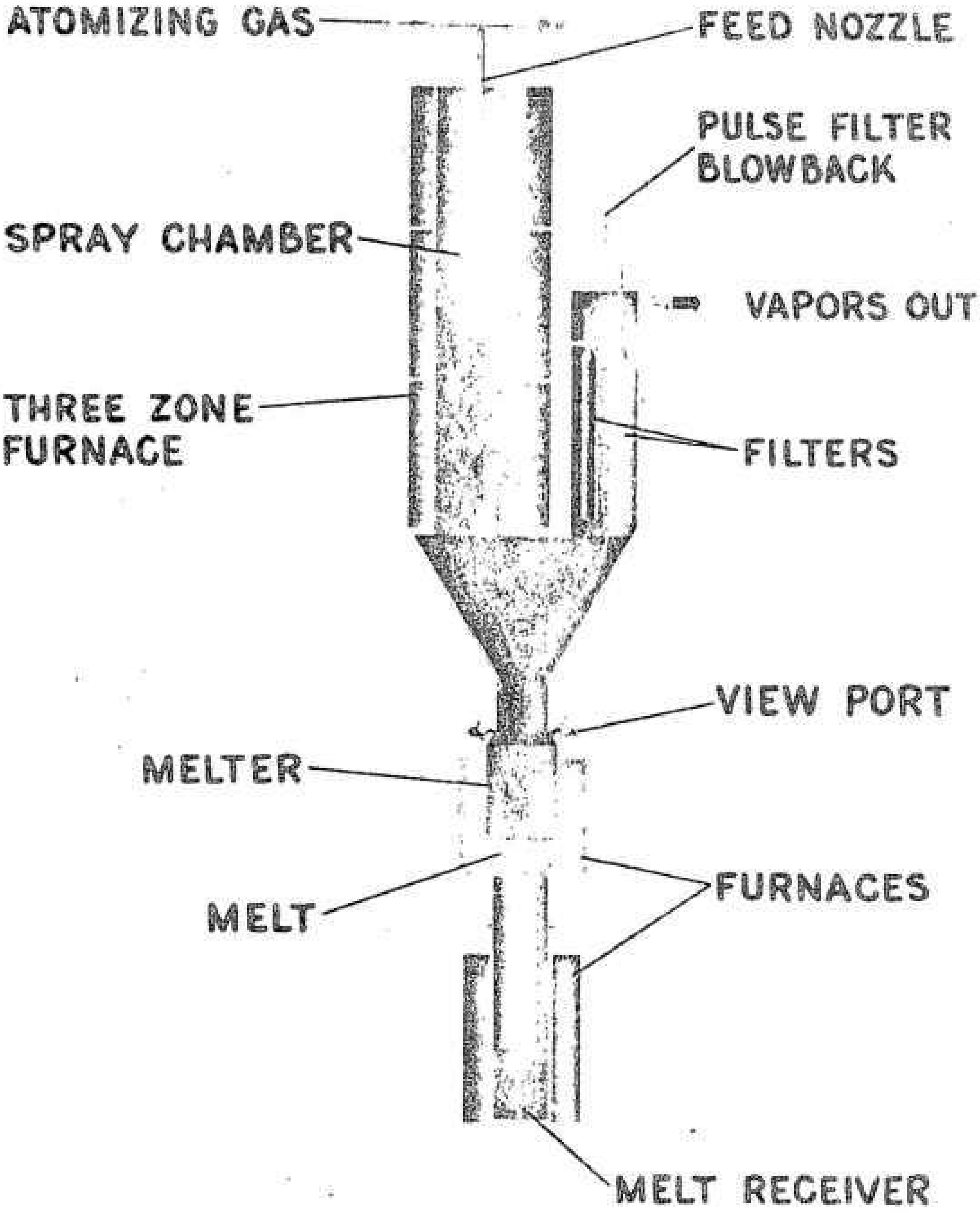
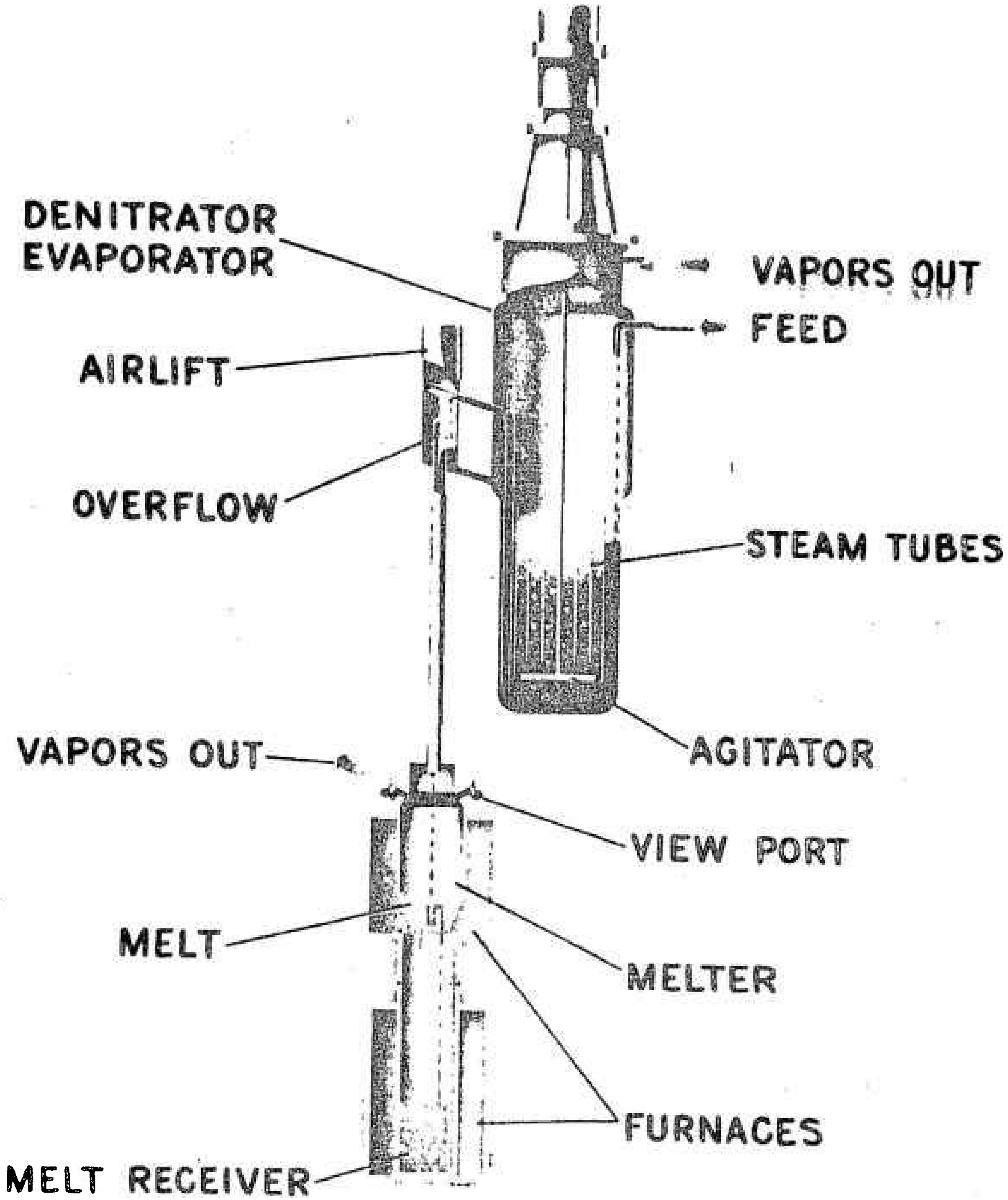


POT SOLIDIFICATION SYSTEM



SPRAY SOLIDIFICATION SYSTEM



PHOSPHATE GLASS SOLIDIFICATION SYSTEM

STATUS WSEP RADIOACTIVE DEMONSTRATIONS OF SOLIDIFICATION METHODS

	POT	SPRAY	PHOSPHATE GLASS	TOTAL
RUNS COMPLETED	6	10	6	22
MEGACURIES SOLIDIFIED	4.0	17.5	8.2	30
EQUIVALENT TONNES PROCESSED	11.3	14.6	9.0	35
MWd_e REPRESENTED BY WASTE	75,000	98,000	59,000	232,000
TONNE/DAY RATE	0.6 to 1.0	0.5 to 0.9	0.3 to 0.7	- -
MAXIMUM kW IN ONE POT	5.1	12.7	9.8	108[a]
MAXIMUM W/ℓ IN POT	85	205	160	- -
MAXIMUM (CENTER) TEMP. IN POT, °C	940	930	780	- -
LITERS SOLID/TONNE	40 to 50	30 to 65	50 to 100	- -
RUNS TO COMPLETE	6	3	5	14

[a] Total kilowatts encapsulated to date.

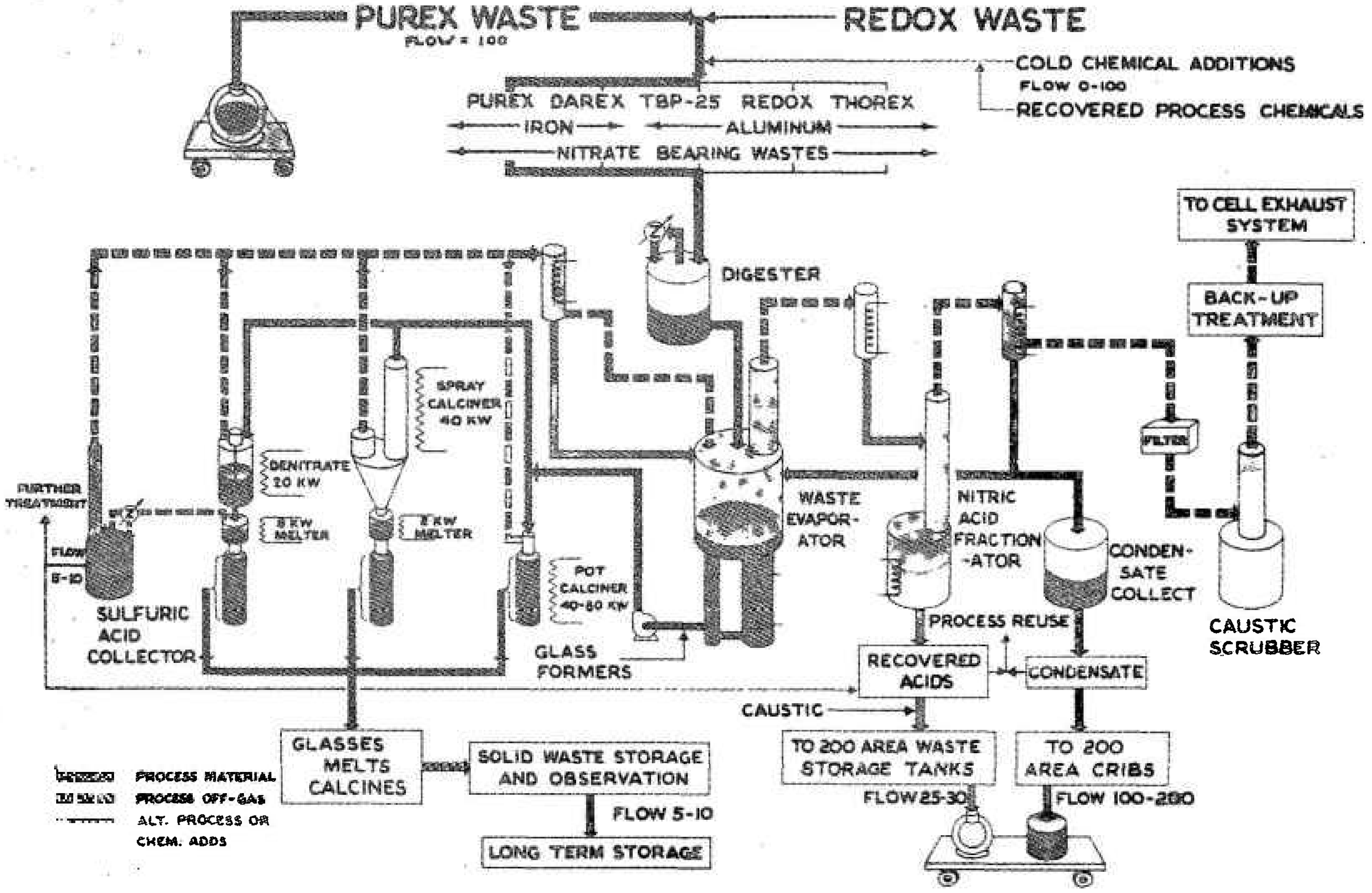
CHARACTERISTICS OF SOLIDIFIED HIGH LEVEL WASTE (PART 1)

	POT CALCINE	SPRAY MELT	PHOSPHATE GLASS
FORM	MONOLITHIC	MONOLITHIC	MONOLITHIC
DESCRIPTION	CALCINE CAKE	MICROCRYSTALLINE	GLASS
CHEMICAL COMPOSITION, mole %			
FISSION PRODUCT OXIDES	up to ~80	up to 20	up to 25
INERT METAL OXIDES	10 to 50	40 to 50	15 to 30
SULFUR OXIDES	0 to 40	0 to 40	0
PHOSPHOROUS OXIDES	~0	25 to 40	~60
BULK DENSITY, g/ml	1.1 to 1.5	2.7 to 3.2	2.7 to 3.0
THERMAL CONDUCTIVITY, W/m-°C	0.3 to 0.5	1.0 to 1.7	0.8 to 1.3
MAXIMUM HEAT, W/liter solid	100	205	190

CHARACTERISTICS OF SOLIDIFIED HIGH LEVEL WASTE (PART 2)

	POT CALCINE	SPRAY MELT	PHOSPHATE GLASS
LEACHABILITY IN COLD WATER, g/cm^2-day	10^{-1} to 10^{-2}	10^{-3} to 10^{-5}	10^{-4} to 10^{-6}
HARDNESS	SOFT	HARD	VERY HARD
FRIABILITY	CRUMBLY	TOUGH	BRITTLE
RESIDUAL NITRATE, wt % of product	≤ 0.05	≤ 0.005	≤ 0.005
VOLUME, liters/1000 MWd_{th}	1 to 2.5	1.2 to 3	1.5 to 5
MAXIMUM STABLE TEMPERATURE, °C	~900	PHASE SEPARATION AT ~900	DEVITRIFIES AT ~500
CONTAINER MATERIAL	STAINLESS STEEL	MILD OR STAINLESS STEEL	MILD OR STAINLESS STEEL

WASTE SOLIDIFICATION ENGINEERING PROTOTYPES



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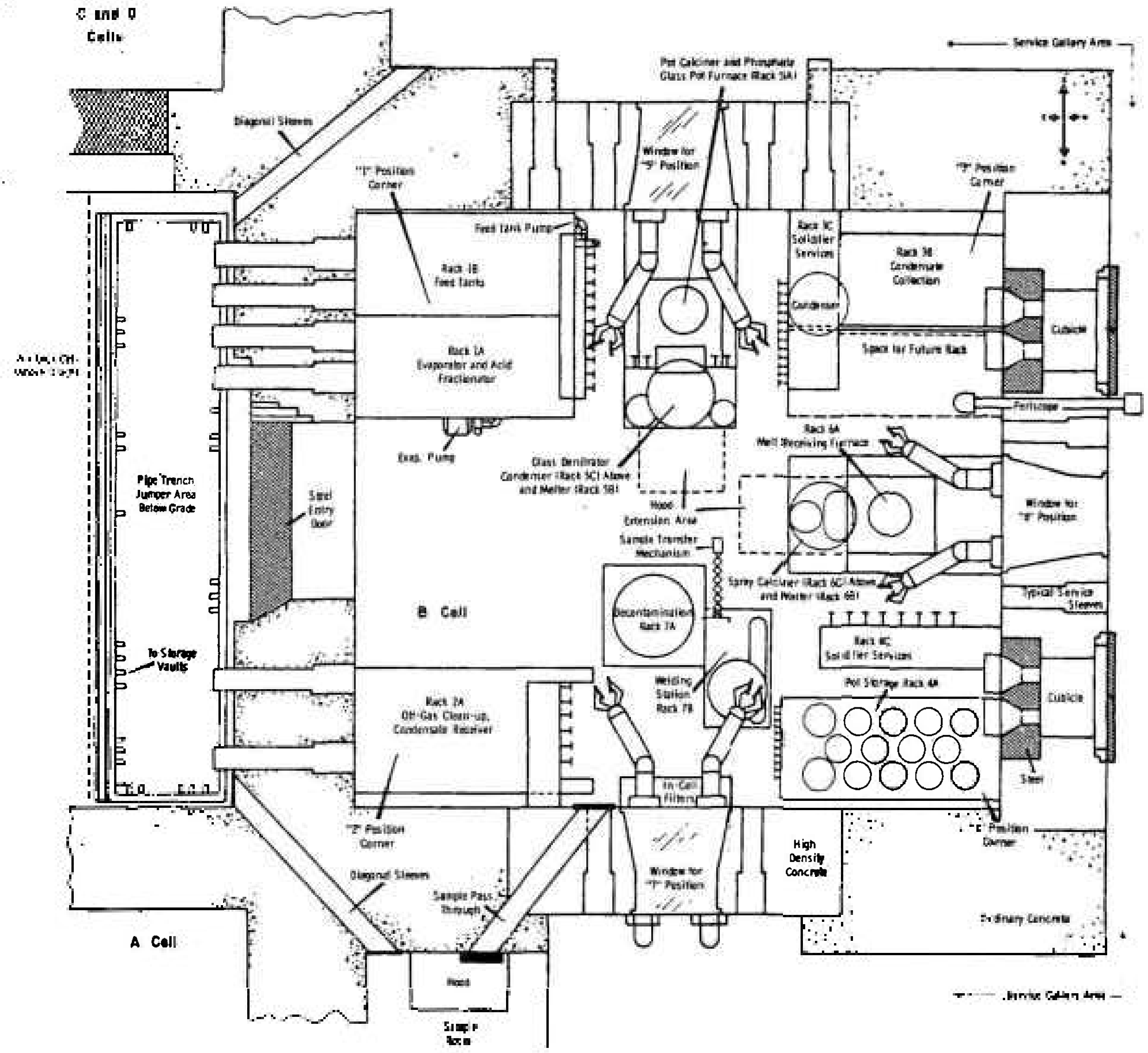
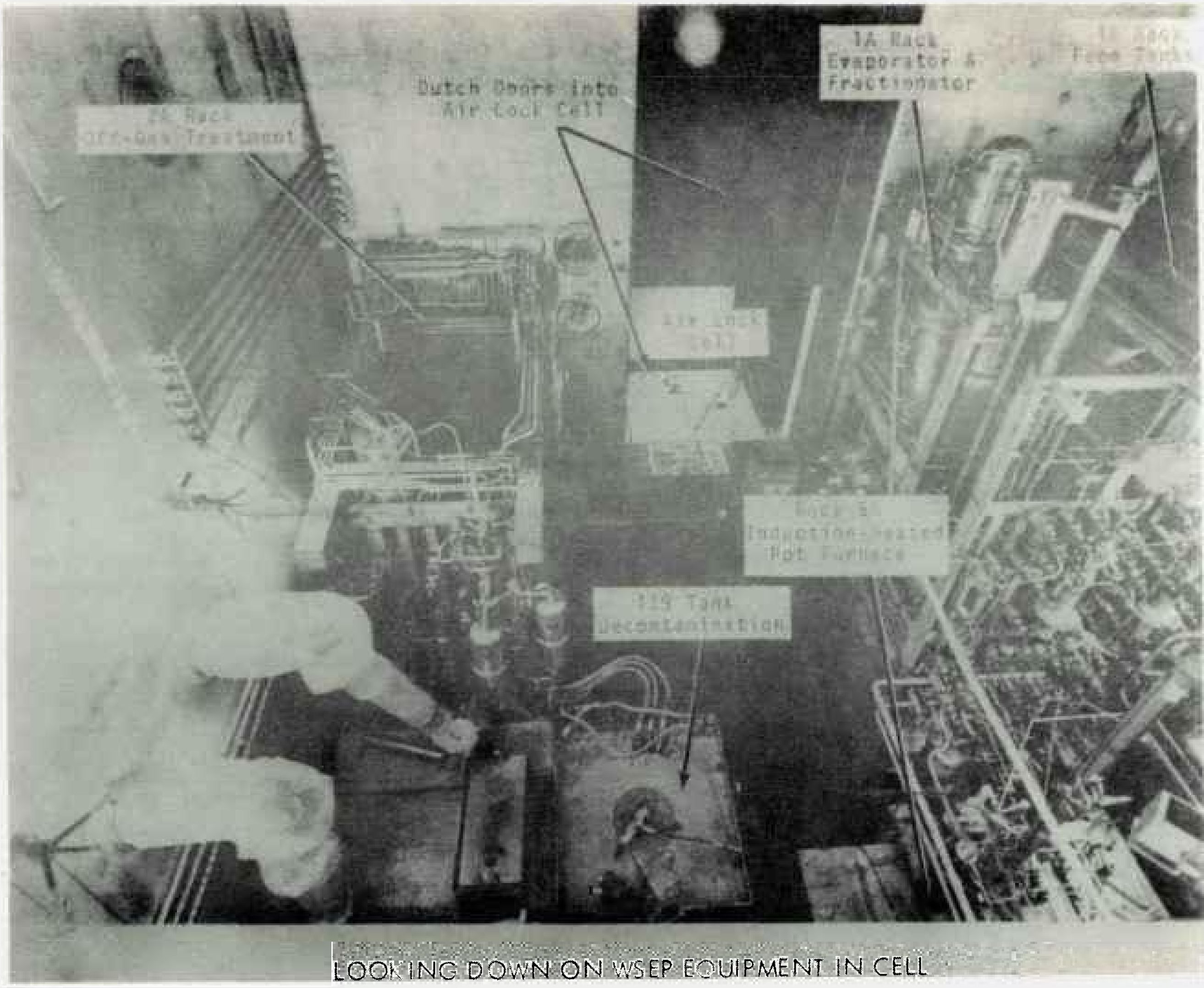
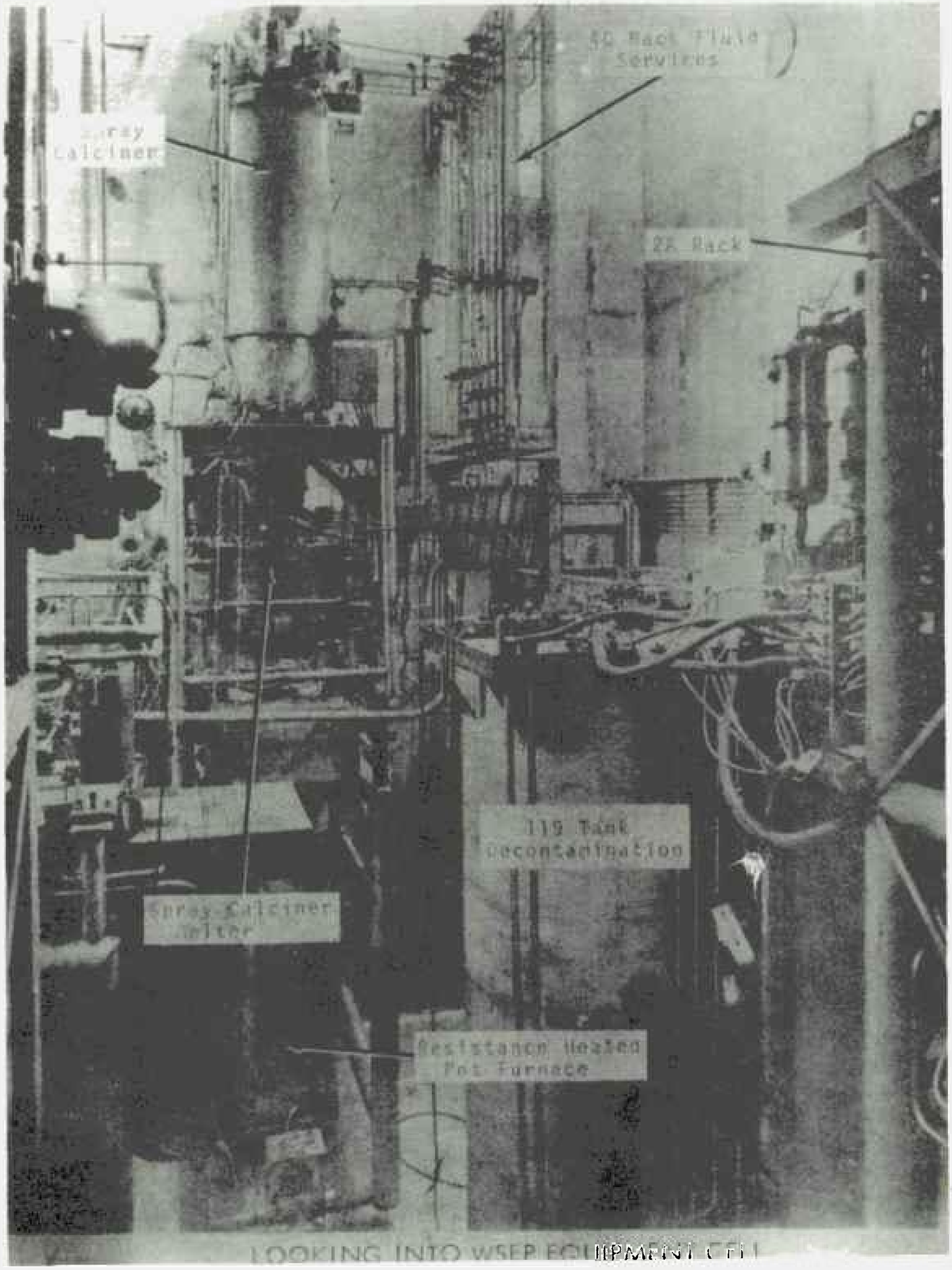


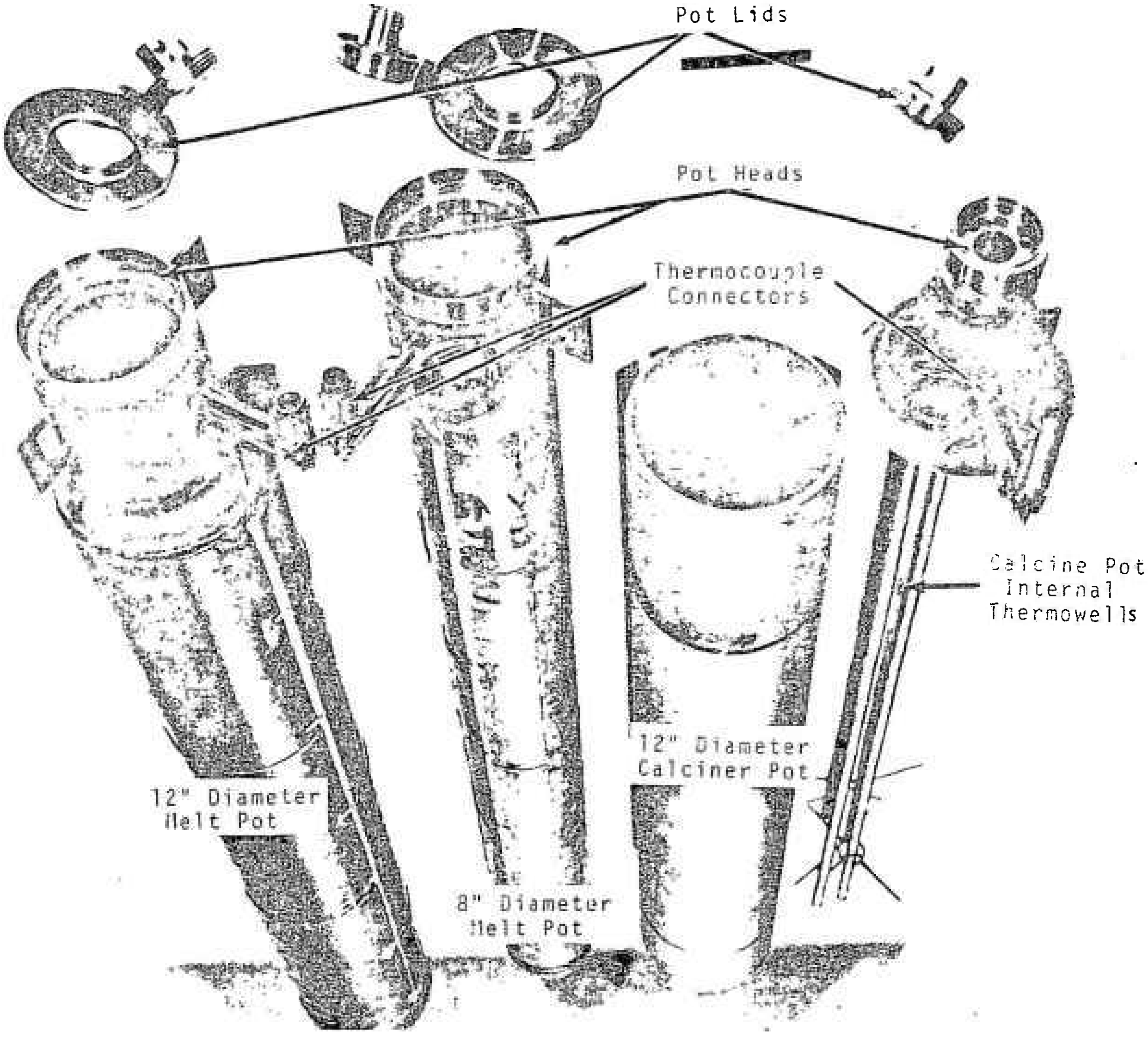
PLAN VIEW OF B CELL EQUIPMENT ARRANGEMENTS



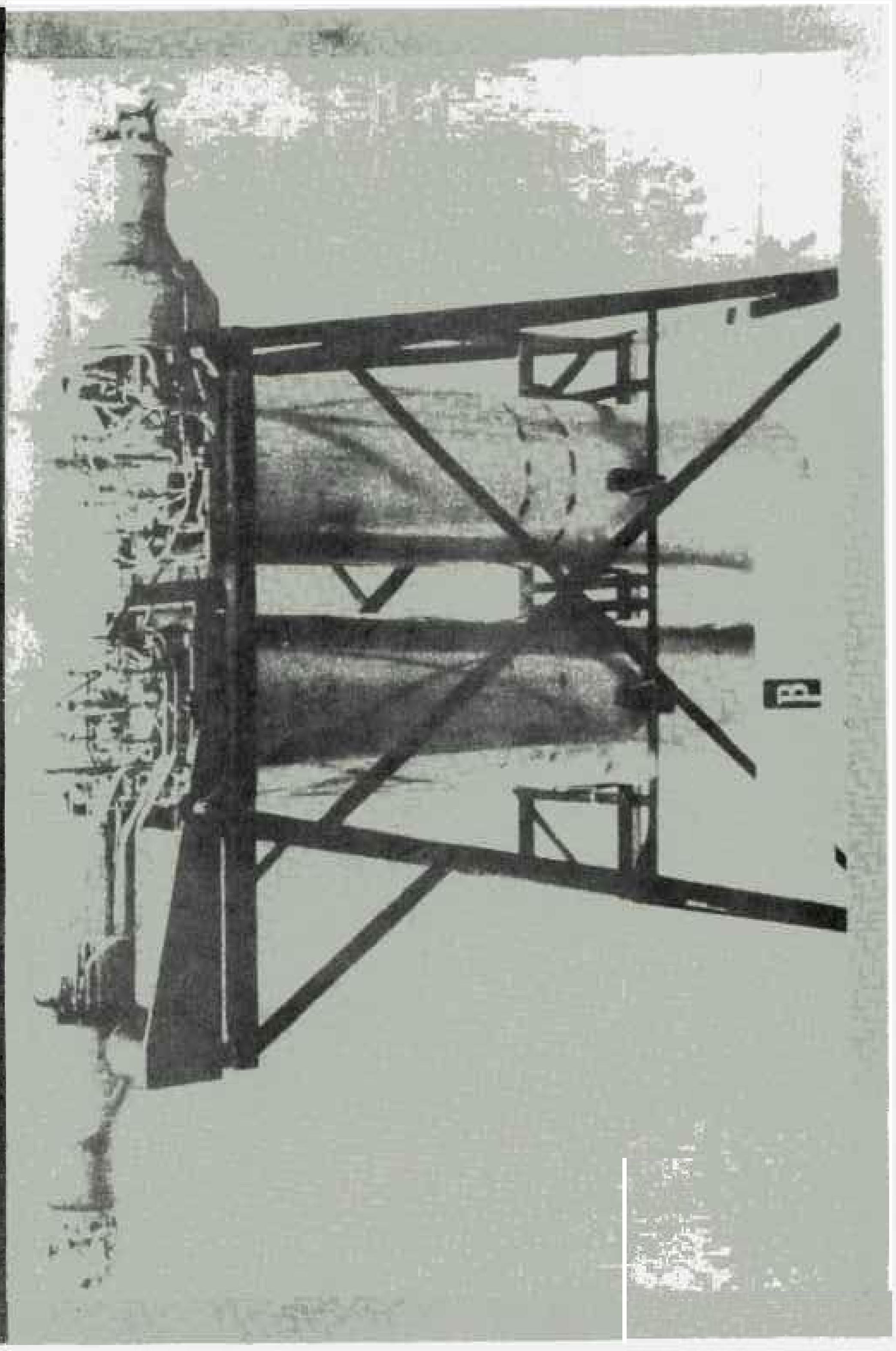
LOOKING DOWN ON WSEP EQUIPMENT IN CELL



LOOKING INTO WSEP EQUIPMENT CELL



WSEP CONTAINERS FOR SOLIDIFIED WASTE



ENVIRONMENTAL TEST PODS BEFORE INSTALLATION IN SSETF

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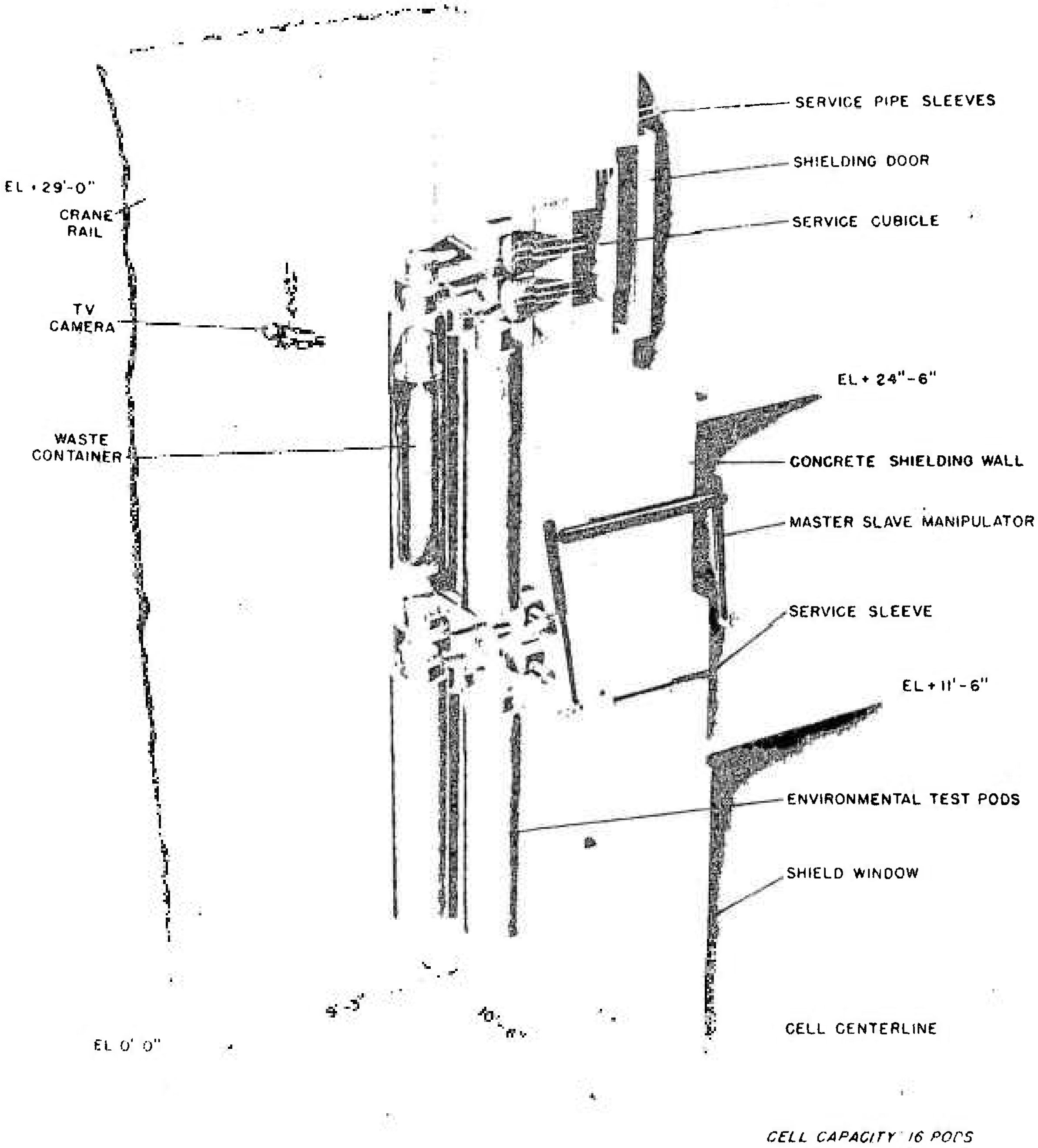
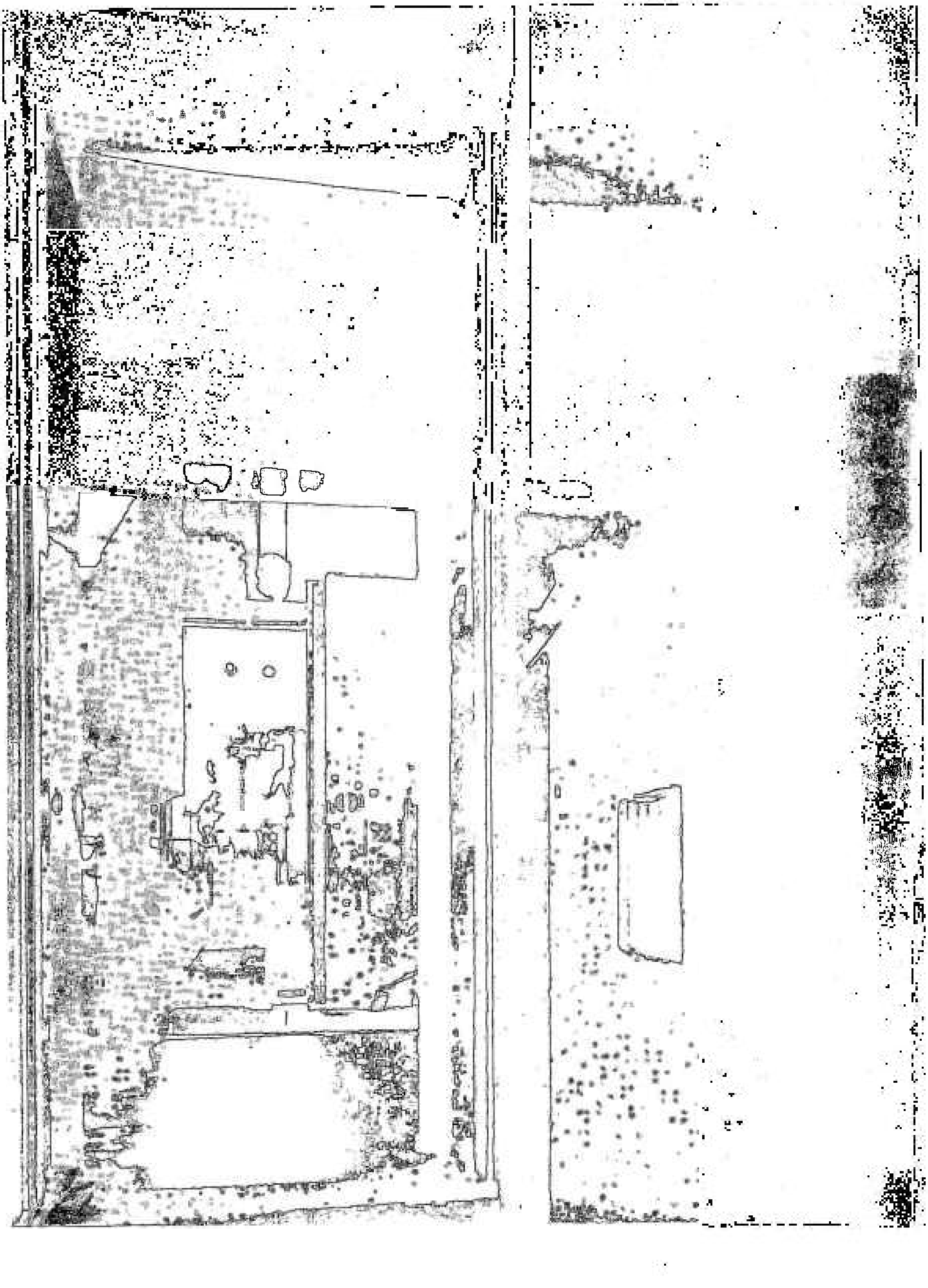
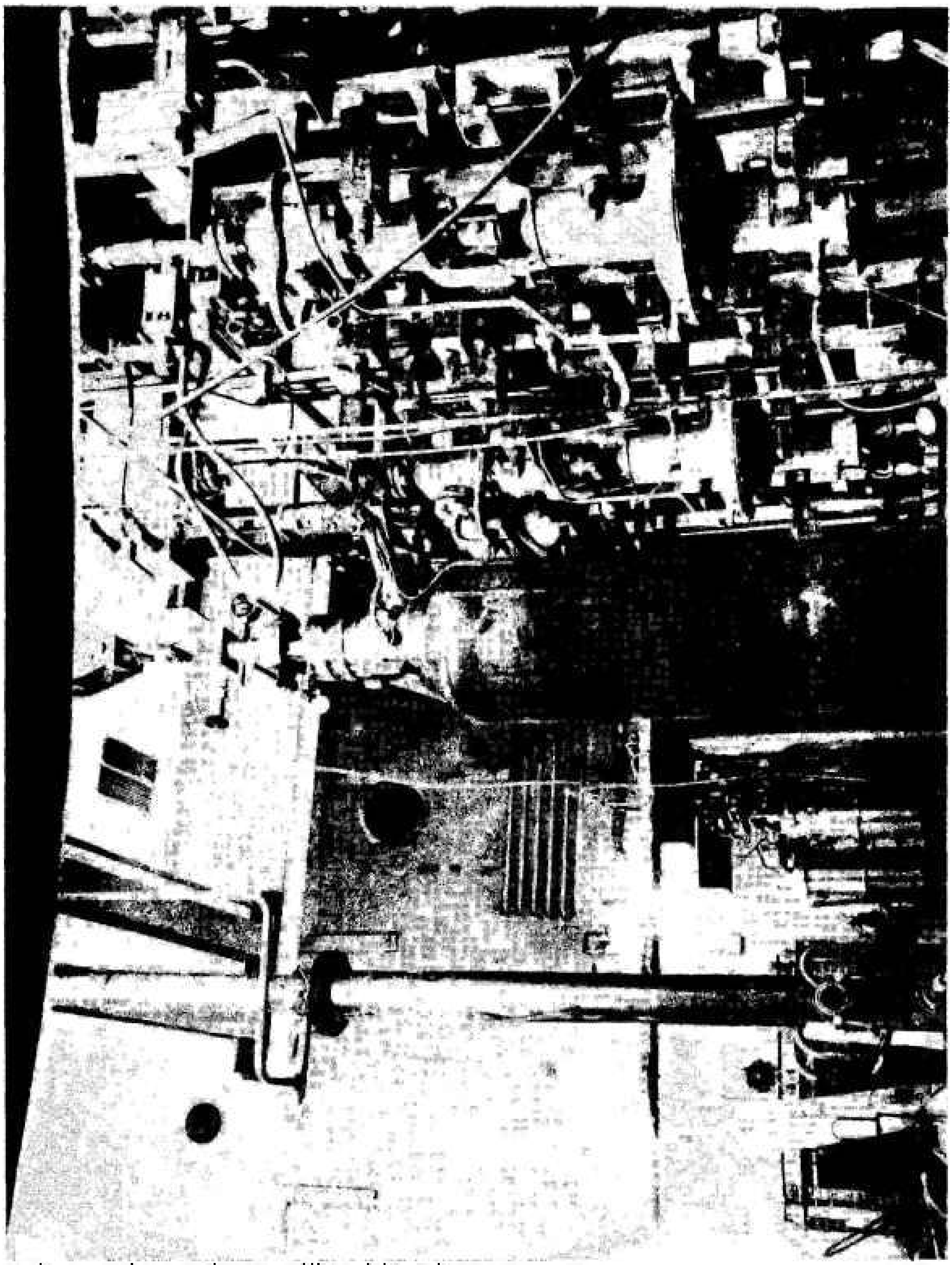
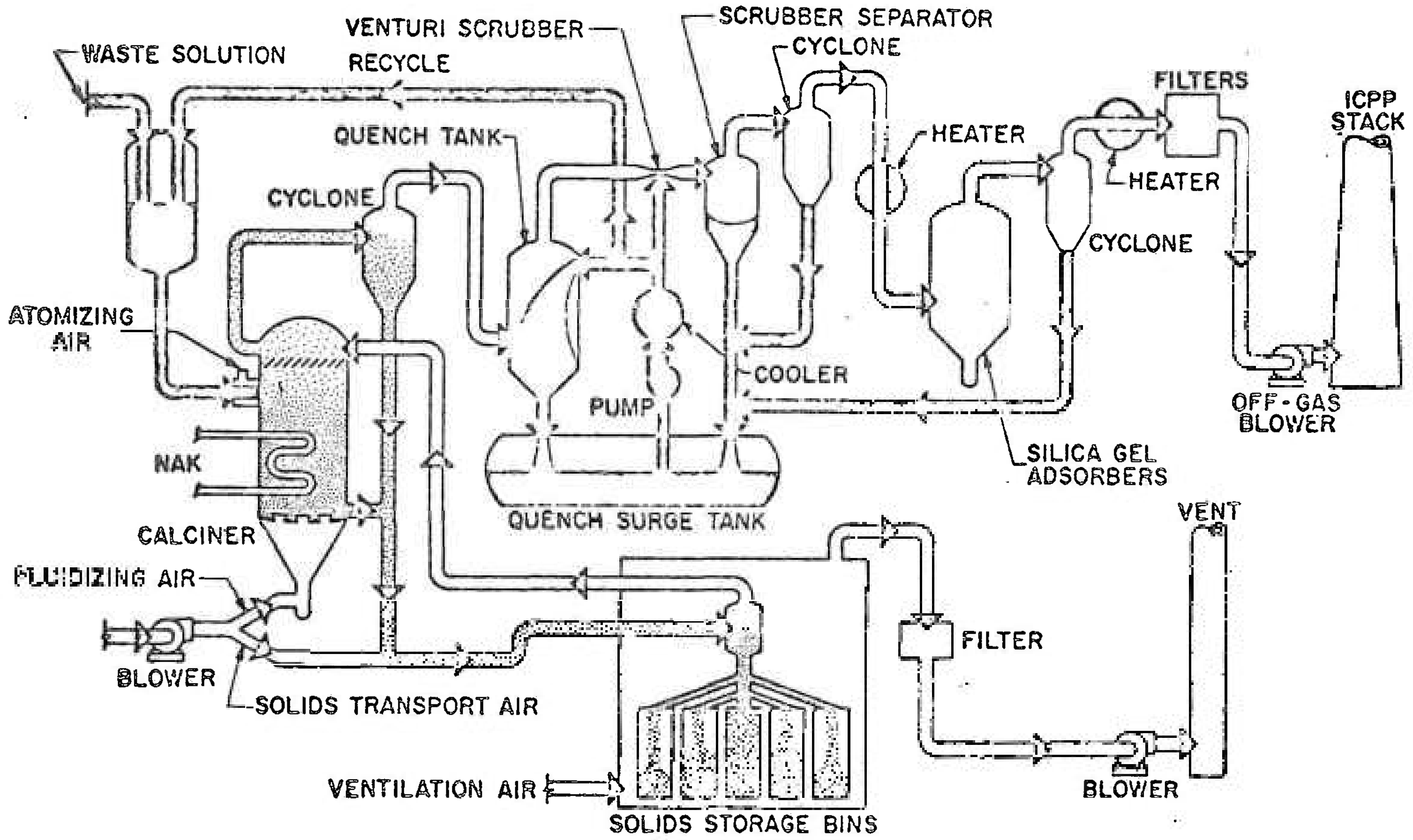


SOLIDS STORAGE ENGINEERING TEST FACILITY

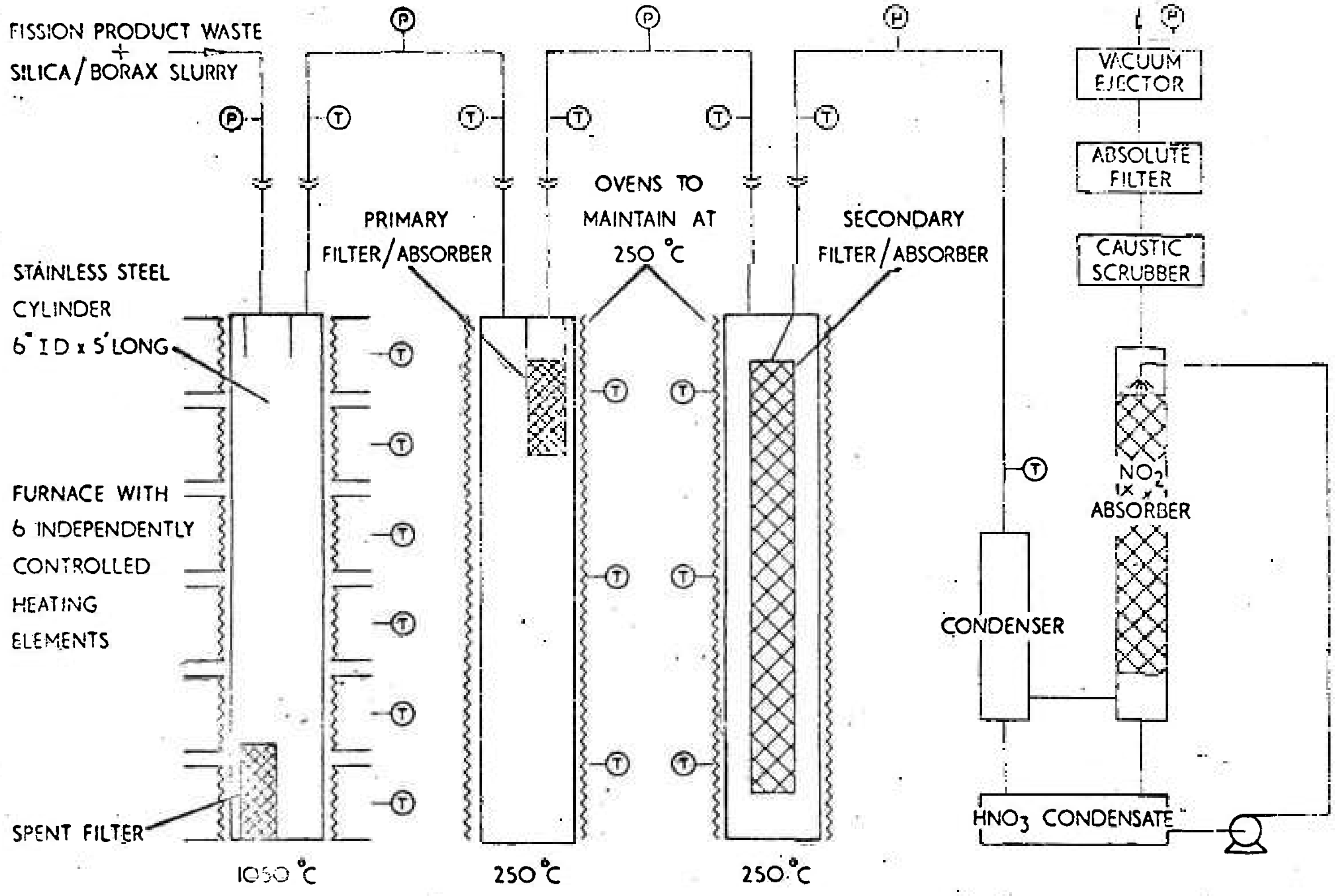




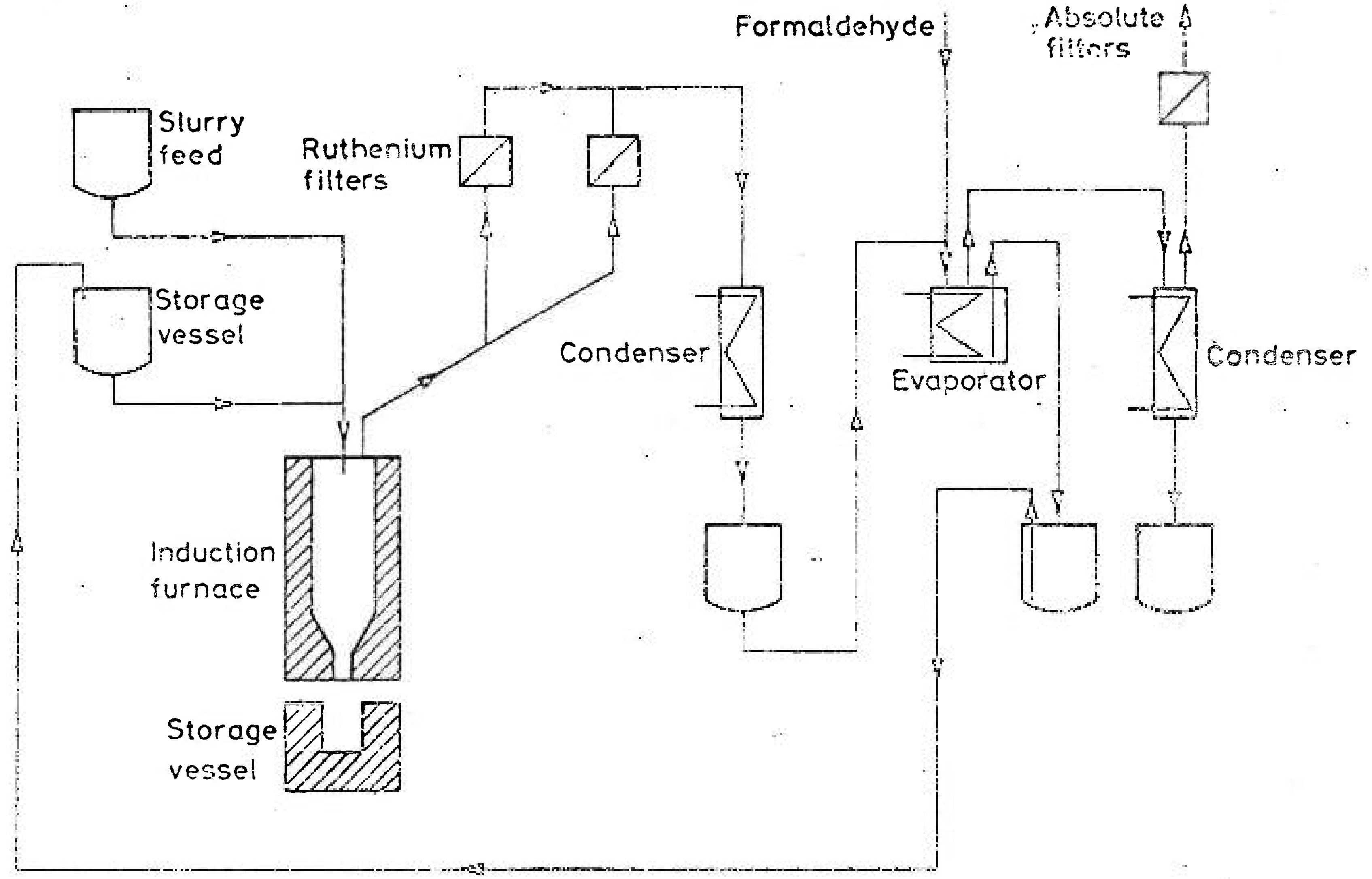
FLUIDIZED BED WASTE CALCINING FACILITY



SCHEMATIC FLOWSHEET OF THE FLUIDIZED BED CALCINATION PROCESS



SIMPLIFIED FLOWSHEET OF PROCESS FOR INCORPORATION OF FISSION PRODUCT WASTES INTO GLASS



POT VITRIFICATION PILOT PLANT AT MARCOULE.

DISPOSAL OF SOLIDIFIED HIGH LEVEL WASTES – FUTURE PROSPECTS

R. E. Blanco

In the previous lecture we discussed the solidification of high level wastes. We must now face the problem of final or ultimate disposal of the solid wastes encapsulated in stainless or mild steel pots (containers).

In September 1955, at the request of the Atomic Energy Commission, a committee of geologists and geophysicists was established by the National Academy of Sciences-National Research Council to consider the disposal of high-level radioactive wastes in geologic structures within the continental United States. The committee proposed storage in natural salt formations as the most promising method for the near future. As a result of the recommendations of the committee, study was begun at ORNL on the problems of disposal of such wastes in salt.

Project Salt Vault was a demonstration of disposal of high-level radioactive waste solids from power reactors, with irradiated Engineering Test Reactor (ETR) fuel assemblies in lieu of actual solidified wastes. The site of the demonstration was the Lyons, Kansas, mine of The Carey Salt Company. Preparations for the demonstration began in 1963, and the first radioactive material was placed in the mine in November 1965. Radioactive operations were terminated in June 1967, and the Lyons mine was placed on standby February 1, 1968.

Other ultimate disposal locations, such as concrete vaults or mined holes in granite have also been surveyed at ORNL, but costs are estimated to be higher than in salt. The Hanford Engineering plant is considering the removal of the principal heat producing isotopes ^{137}Cs and ^{90}Sr, and solidifying the remaining wastes in large underground tanks – "in tank solidification." The Savannah River plant is considering the solidification of wastes in underground bedrock vaults.

This lecture is based on data and excerpts from papers by R. L. Bradshaw, et al. on "Project Salt Vault," by J. F. Proctor and I. W. Marine on "Storage in Bedrock Vaults," and P. W. Smith and G. L. Hanson on "In Tank Solidification." (See references at end on lecture.)

PROJECT SALT VAULT

SLIDE 1 Salt Occurrences in the United States (68-10576R1)

Illustrates the wide availability of salt. Although kilometers of mined salt corriders and rooms exist, it is probable that a new mine or addition to an existing mine would be used.

SLIDE 2 Advantages of Salt Formations for Storage of Radioactive Wastes (61939)

SLIDE 3 Demonstration Objectives (64-3418R)

SLIDE 4 Demonstration of Radioactive Solids Disposal in Salt (63-239R)

This slide shows a schematic view of the demonstration site at Lyons, Kansas. The assemblies, after canning in Idaho, were shipped on a specially designed truck trailer in an ORNL carrier, modified for this purpose. At Lyons, the carrier was removed from the trailer and placed vertically over a steel-cased charging shaft which extends to the mine working area, approximately 1000 ft below. The fuel canisters will be lowered one at a time down the 19.1-in.-ID waste charging shaft into a shielded cask, mounted on an underground fuel assembly transporter. The transporter will then move to the experimental area and the fuel canister lowered into a suitably lined and shielded 12.in.-diameter, 13-ft-deep hole. Rock salt is approximately equivalent to concrete as an absorber of gamma radiation. Approximately 5 ft of solid salt or 7 ½ ft of crushed salt (assuming 1/3 voids) will give adequate biological shielding to allow unlimited access to a room in a salt mine filled with the most radioactive waste of the future. The containers would be located in backfilled holes in the floor with the tops of the containers at the proper depth, and that container spacing is based on fission product-decay heat dissipation calculations. In an actual disposal operation it is anticipated that rooms would be filled with waste, and then backfilled with crushed salt rapidly enough that roof bolts would probably not be required.

SLIDE 5 No Title (63-774A)

This slide shows the details of the experiments in Project Salt Vault. Fourteen irradiated fuel assemblies from the Engineering Test Reactor served as a source of radiation, since high-level solidified wastes do not exist at the present time. The assemblies will be placed in a circular array of holes in the floor of a newly mined area (14 ft above the existing mine floor,) with one can in the center and the other six cans located peripherally, spaced 5 ft on centers. During the course of the 2-year test, four sets of assemblies were used to achieve a peak dose in the salt of about 8 x 10^8 rad and the temperature of the salt adjacent to the center hole will be maintained at 200°C with supplementary electrical heaters. A second radioactive array, located in the existing mine floor, received the fuel canisters which were removed from the main array at the end of each 6-month period The purpose of this array is to study the problems which may be encountered if the waste containers are located in salt interbedded with water-bearing shale. A third array, consisting only of heaters, will be operated as a control to determine those effects due solely to heat.

In addition to the radioactive and control arrays, a rib-pillar located between the main radioactive array and the control array will be heated electrically around its base to produce significant information on salt flow characteristics and mine stability at elevated temperatures.

At the time of shutdown of the demonstration arrays, temperatures at the wall of the center hole of each of the arrays were at or slightly above 200°C and a roughly circular ellipsoid of salt of about 12 x 15 ft diameter was at a temperature between 100 and 200°C.

SLIDE 6 Mechanism of Stress Transference from Floor to Ceiling; Numerical Sequence (68-7583)

The heaters in the floor along the sides of the pillar elevated a volume of salt about 20 ft deep, 40 ft wide, and 60 ft long, in the floor beneath the pilar, to temperatures in excess of 60°C. The insertion of heat sources in the floor of a mine room produced a thermal stress whose effects are instantaneously transmitted around the opening (to the pillars and roof). These stresses produce increase plastic flow rates in the salt, and could possibly cause trouble if the roof of the room is too close to a shale layer (a plane of weakness). In the demonstration area such a shale layer existed, but is was found that conventional roof bolting techniques were adequated to handle the problem.

SLIDE 7 No Title (673512R)

Measurements showed that plastic flow of salt was not excessive.

SLIDE 8 Comparison of Theoretical and Experimental Migration Rates of Brine-Filled Cavities in Rock Salt.* Data Normalized to Salt Temperature Gradient of 1°C/cm (67-9435)

This slide shows the migration rate of small brine-filled cavities as a function of temperature. Small brine-filled cavities (generally roughly cubic in shape, with sizes ranging from a few millimeters down to microscopic) migrate toward a heat source. A typical bedded salt deposit might contain about ½% by volume water. Theoretical studies and laboratory tests of the migration rates as a function of temperature were in reasonable agreement. Based on the theoretical calculations one might expect a total inflow per waste disposal container hole of from 2 to 10 liters. Taking place over a period of 20 to 30 years after burial. (The void space in the crushed salt above the can is about 45 liters). The peak inflow rate would occur at around 1 year after burial and be somewhere in the range of 200 ml to 1 liter per year per hole. Similar migration rates were observed in the demonstration. This water inflow rate would be expected to taper off and approach zero after 20 to 30 years. (The total number of waste containers in a 50- x 300-ft room assumed in the above calculations would be 150.)

The field tests have indicated that once the migrating brine reaches the crushed salt backfilling the hole it moves upward and condenses in the colder regions above the waste containers. Thus, the water is expected to be retained in the 7 to 8 ft of crushed salt above the containers.

SLIDE 9 Comparison of Measured and Calculated Doses to Salt (67-9434)

During the 19-month operation of the radioactive phase of the demonstration, the average dose to the salt over the length of the fuel assembly container holes was about 8 x 10^8 rads, and the peak dose was about 10^9 rads. Dose dropped off very rapidly with distance out into the salt, with doses at 6 in. into the salt being only about 10^8 rads. Therefore, in each of the seven main array holes, a volume of about 14 ft^3 of salt was exposed to doses ranging from 10^8 to 10^9 rads. As antecipated from the laboratory studies, no significant effects due to the radiation were detected. Due to the relatively unusual requirements imposed on the dosimeter (dose rates to 10^5 rads/hr, integrated doses to 10^5 rads, and 200°C exposure temperature), radiophotoluminescent glass rods were used for most of the measurements.

SLIDE 10 No Title (64-3902)

This shows the design of a salt facility used in a cost study. It was assumed that a new salt mine would be used so that ventilation and access could be controlled.

SLIDE 11 No Title (Cost of Disposal in Salt) (67-9758R1)

Shows the estimated cost of disposal in salt. The maximum cost is for short-cooled wastes, reacidified Purex Waste in 6-in. diam pots (RP6) and reacidified Thorex waste in 6 in.-diam. pots (RT6). The shorter cooled wastes have a higher heat output and require lower diameter pots to hold the center temperatures to ~ 900°C maximum. Therefore, a larger number of pots (more extensive) are required for a given amount of waste. Also, more area of salt mine is required to dissipate the heat and hold the maximum temperature of the salt to 200°C. Additional details on costs were given in the lecture on costs of waste management.

SLIDE 12 **Waste Management Data for Conversion-To-Solids Concept (30 Years Interim Solids Storage) (68-10568)**

These are the estimated volumes of solid wastes and the salt mine area required in the U. S. by the year 2000. The basis for this estimate is contained in the lecture on present waste management practices at fuel processing plants, the slide on the expected accumulation of liquid high-level wastes. These estimates were prepared by J. O. Blomeke, ORNL.

This slide shows the estimated requirements over the remainder of the century for application of this solidification, storage, shipment, and disposal-in-salt concept. We assume immediate solidification in 6-in.-diam cylinders, followed by 30 years' on-site storage in canals, and then shipment and disposal in slat at age 30 years. The accumulated volume in storage rises from 2000 ft^3 in 1970 to 550,000 ft^3 in 2000, and will require, in the year 2000, 6100 linear feet of 48-ft-wide canals (or their equivalent). Following 30 years' storage, the waste is shipped 1000 miles to salt mines for disposal. If shipment takes place in casks containing thirty-six 6-in.-diam cylinders each, 20 shipments will be required in 2000 and no more than one leaded cask will be in transit at any time during that year. However, each year we will be committing an increasing number of pots for shipment 30 years later, until in 2030, about 900 shipments, corresponding to as many as five casks in transit at any time, will be required. The salt mine area required will increase from 2.6 acres/year in 2000 to 118 acres/year in 2030, which corresponds to a total utilization of 1415 acres (2-1/4 sq miles), a negligible fraction of U. S. salt reserves.

SLIDE 13 **Waste Management Data for Conversion to Solids Concept (2 Years Interim Solid Storage) (68-10567)**

This slide shows estimates for the accelerated case of only 2 years' on-site storage. The principal effect is to reduce markedly the waste inventories on-site at the expense of earlier shipment dates and nearly doubling the utilization of salt space in order to dissipate the heat from younger waste.

SLIDE 14 **Major Results (67-3689)**

The next series of slides are photos of Project Salt Vault.

SLIDE 15 **No Title (Photo 80765)**

The seven canisters were shipped in a 30-ton cask, equipped with forced circulation cooling. Cooling was accomplished by circulating the cooling water from a diesel engine equipped with an oversized radiator.

SLIDE 16 **No Title (Photo 80984)**

Arrival at mine shaft.

SLIDE 17 **No Title (Photo 80980)**

Tilting to mine shaft.

SLIDE 18 **No Title (Photo 67215)**

Sample unirradiated fuel sample in carrier.

SLIDE 19 No Title (Photo 80977)

SLIDE 20 No Title (Photo 81008)

SLIDE 21 No Title (Photo 80997)

Shows electrically heated pots in salt.

SLIDE 22 No Title (Photo 80969)

SLIDE 23 No Title (Photo 81000)

Many kilometers of mine corridors are in existance.

ROCK VAULT STORAGE

SLIDE 24 Storage Vault for Radioactive Wastes (69-10569)

This slide shows a diagram of a rock storage that is being studied at the Savannah Laboratory, at Aiken, South Carolina USA. A recent investigation established the technical feasibility and indicated the high degree of safety that could be afforded by the storage of high-level radioactive wastes in unlined vaults excavated in crystalline rock 1500 ft beneath the surface of the Savannah River Plant near Aiken, S.C.

The crystalline rock at the proposed site is covered by 1000 ft of unconsolidated sediments consisting predominantly of sand and clay. A virtually impermeable layer of clay separates the rock from the overlying sediments in which several prolific water-bearing zones occur. The separation of the waters above and below this clay layer is confirmed by their different chemical composition and by the presence of dissolved helium bearing gas only in the water in the rocks beneath the clay. Based on geologic and hydrologic information obtained in an intensive drilling and testing program upper limits on the rates of water movement through the crystalline rock are calculated to be 1.5 to 7 ft/year, depending upon the degree of fracturing of the rock. Comparable data on the unconsolidated sediments lead to a calculated maximum rate of water movement of 350 ft/year.

The most significant driving force for the migration of radionuclides from the storage site is derived from the natural water movement, coupled with effects due to dispersion and ion exchange. Characteristics of the waste, heat generation, and radiolysis have, by contrast, only small effects on migration. Three barriers prevent migration of the radionuclides: the very low permeability of the rock in which the storage vault is located, the virtually impermeable clay layer separating the rock and sediments, and the ion exchange properties of the sediments. Any one of these barriers is capable of confining the radionuclides well within the plant boundaries for a time much greater than the 600-year period required to render the wastes innocuous.

SLIDE 25 Waste Management Program (50486-5)

This slide shows the advanced waste management program at the Hanford plant. Self-boiling liquid wastes which have been generated and stored in underground tanks since about 1951 are now being processed for removal of ^{137}Cs and ^{90}Sr (by ion exchange and solvent extraction) prior to solidification by evaporation and cooling of the residual bulk (mostly sodium) salts in the tanks. Removal of the cesium and strontium is necessary prior to solidification in order to limit the temperatures which are attained in the salt cakes. The isolated cesium and strontium will be doubly encapsulated as dry cesium chloride and strontium fluoride compounds for interim storage in a water-cooled basin prior to movement to a long-term disposal site. Currently generated self-boiling wastes will be treated similarly to stored wastes except than an aging period of from 5 to 7 years, to permit short-lived fission products to decay, is required before the residual salt waste can be in-tank solidified. High-level non-boiling wastes will be processed directly to salt cakes. This type of permanent disposal by in-tank solidification is considered to be a suitable type of permanent disposal in the Hanford area, where desert (low rainfall) conditions prevail.

SLIDE 26 In-Tank Solidification (50486-19)

Non-boiling wastes are solidified in existing tanks at Hanford by evaporation with crystallization upon cooling. There are three types of evaporators in operation. One unit is a large electric immersion heater (4000 KW) installed in an airlift circulator in a storage tank. A second unit is a heated air unit in which about 3000 cfm of air at 1200 F is supplied to an airlift circulator in another waste storage tank. The third unit is a conventional steam heated tube bundle evaporator of about six million Btu/hr capacity. The evaporator units are operated on a recycle basis which permits solids to build up in air cooled bottoms receiver tanks and the supernatants to be returned to the evaporator.

This slide shows in-tank solidification with the electric immersion heater. The off-gas treatment facilities and operational mode are typical of the other units. The feed is pumped to the evaporator tank where it is concentrated to about 10 percent solids. The concentrate is pumped to a cascade of cooling tanks for further solids formation and deposition. The supernatant is recycled with fresh feed. The off-gas system includes a deentrainer, a condenser, a cyclone separator and high efficiency filters. The condensate is routed to a crib and the gases are vented to the atmosphere through a stack. The condensate's radioactivity is very near low level waste concentrations and the gases are well within release guide concentrations.

SLIDE 27 ITS-2 Immersion Heater Quadrant (50486-14)

SLIDE 28 Long-Term Storage of Radioactive Wastes as Salt Cakes in Tanks (50486-17)

The salt cakes might be prepared for long-term storage as shown in this slide. After it has been determined that the temperature profiles and rates of radiolytic gas formation are sufficiently low, sand or grout could be added to the void space in the tank to prevent total collapse should the tank dome fail. A layer of rock and gravel could then be placed over the tank farm area to give added protection from wind erosion. In this condition, the waste would be stored about 30 feet below ground surface and about 150 feet above the water table.

SLIDE 29 Long-Term Storage of Salt Slurry in Underground Caverns (50486-20)

Hanford is also studying the use of underground caverns in basalt rock. The salt resulting from in-tank solidification would be slurried to caverns some 3,000 to 4,000 ft below the water table.

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2. W. J. Boegly, Jr., R. L. Bradshaw, F. M. Empson, W. F. Schaffer, Jr. F. L. Parker, and J. O. Blomeke, Health Physics 12, 417-24 (1966).

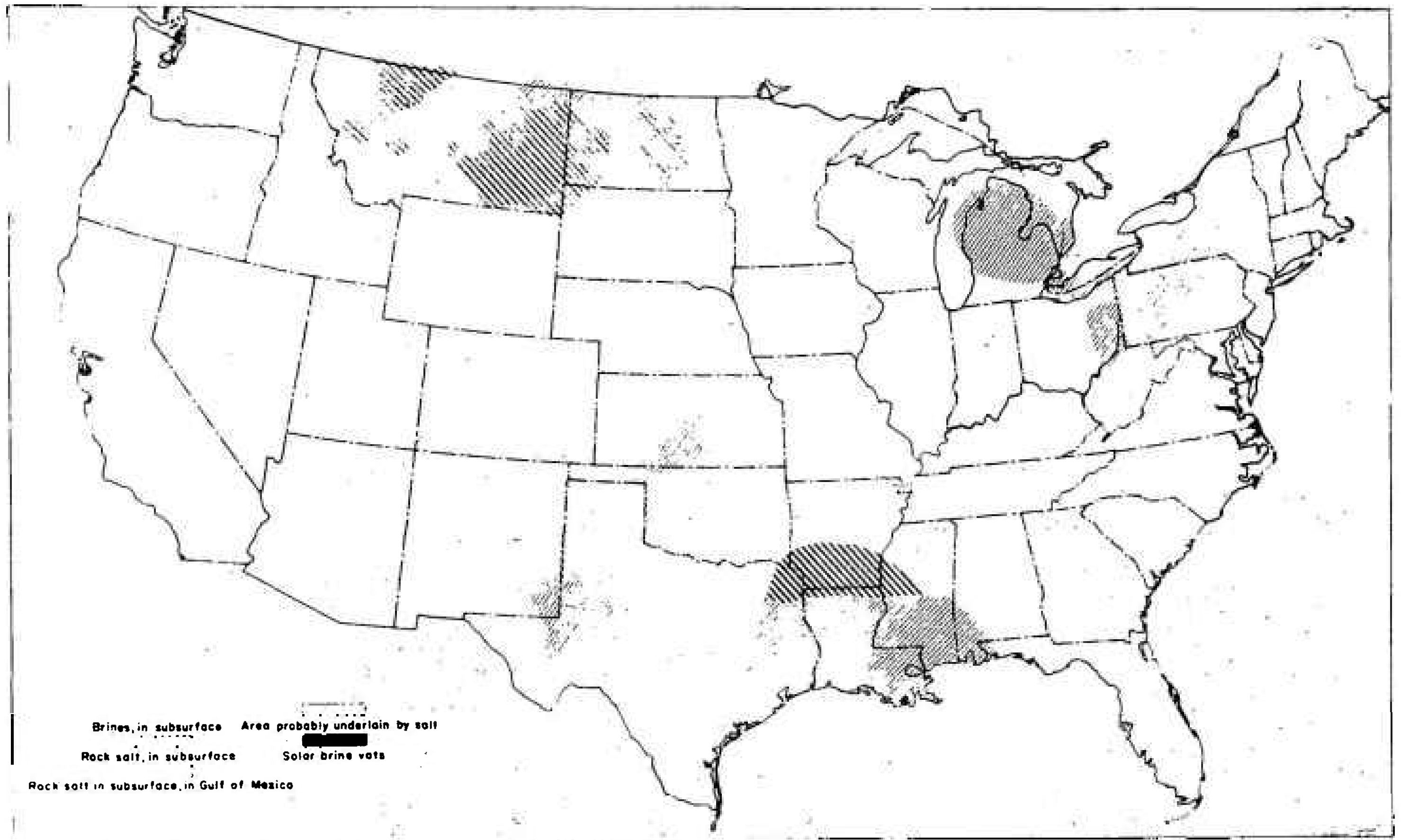
3. R. L. Bradshaw, J. J. Perona, J. O. Blomeke, and W. J. Boegly, Jr., *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes, Part VI: Disposal of Solid Wastes in Salt Formation,* ORNL-3358 (May 1968).

4. P. W. Smith and G. L. Hanson, *Radioactive Waste Management Policies and Practices,* ARH-SA-30 (April 1969).

5. J.F. Proctor, Nucl. Sci. and Eng. 22, 350-65 (1965).

6. R. E. Isaacson, "Future Options in Long-Term Waste Storage" in *Management of Radioactive Wastes at the Hanford Plant,* 29-42 (June 1969).

7. Unpublished data by J. O. Blomeke, ORNL.



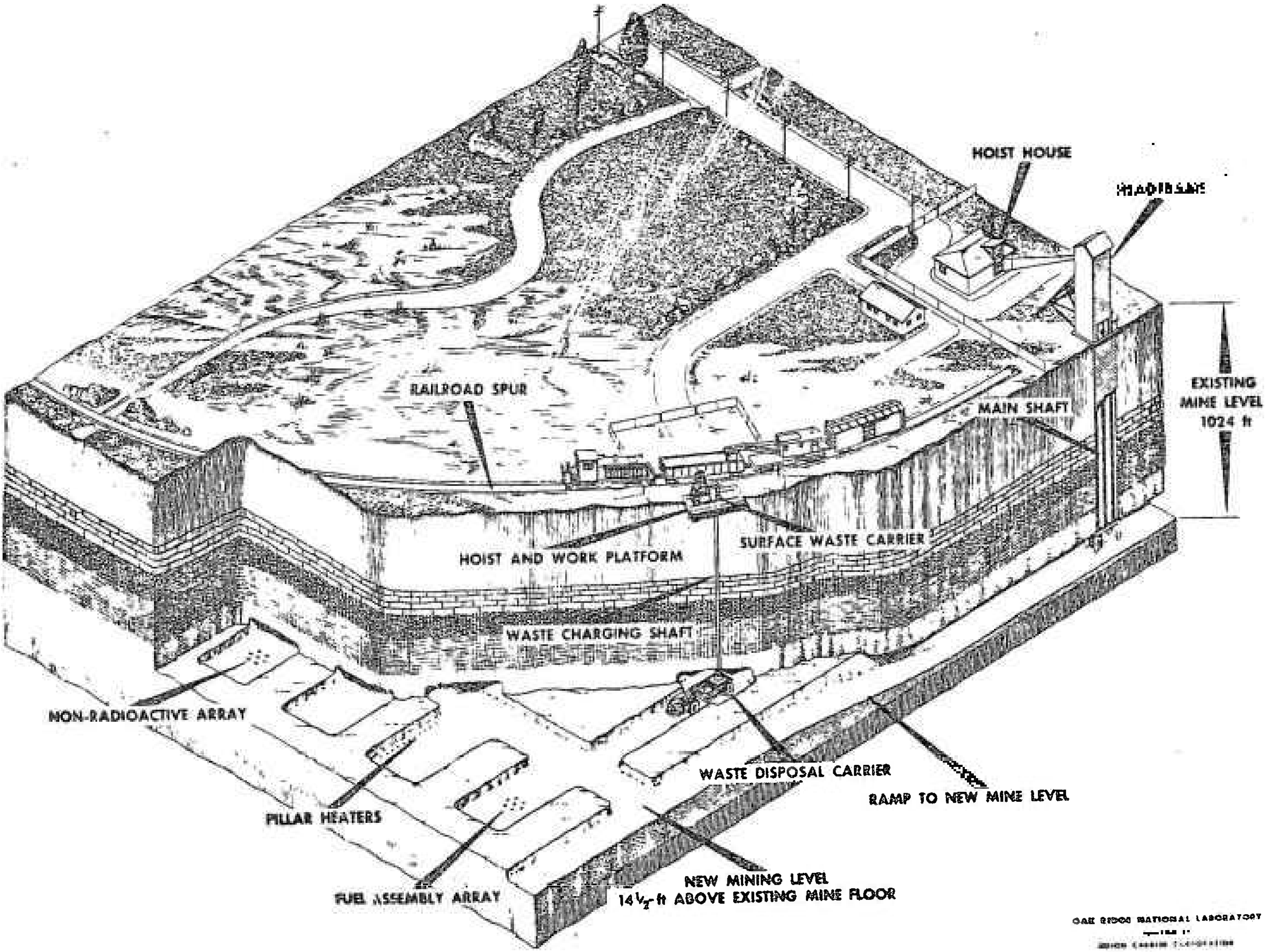
SALT OCCURRENCES IN THE UNITED STATES

ADVANTAGES OF SALT FORMATIONS FOR STORAGE OF RADIOACTIVE WASTES

1. WIDESPREAD AND ABUNDANT
2. HIGH STRUCTURAL STRENGTH
3. LOW COST OF DEVELOPING SPACE
4. GOOD THERMAL CONDUCTIVITY
5. LOCATED IN AREAS OF LOW SEISMICITY
6. ESSENTIALLY IMPERMEABLE

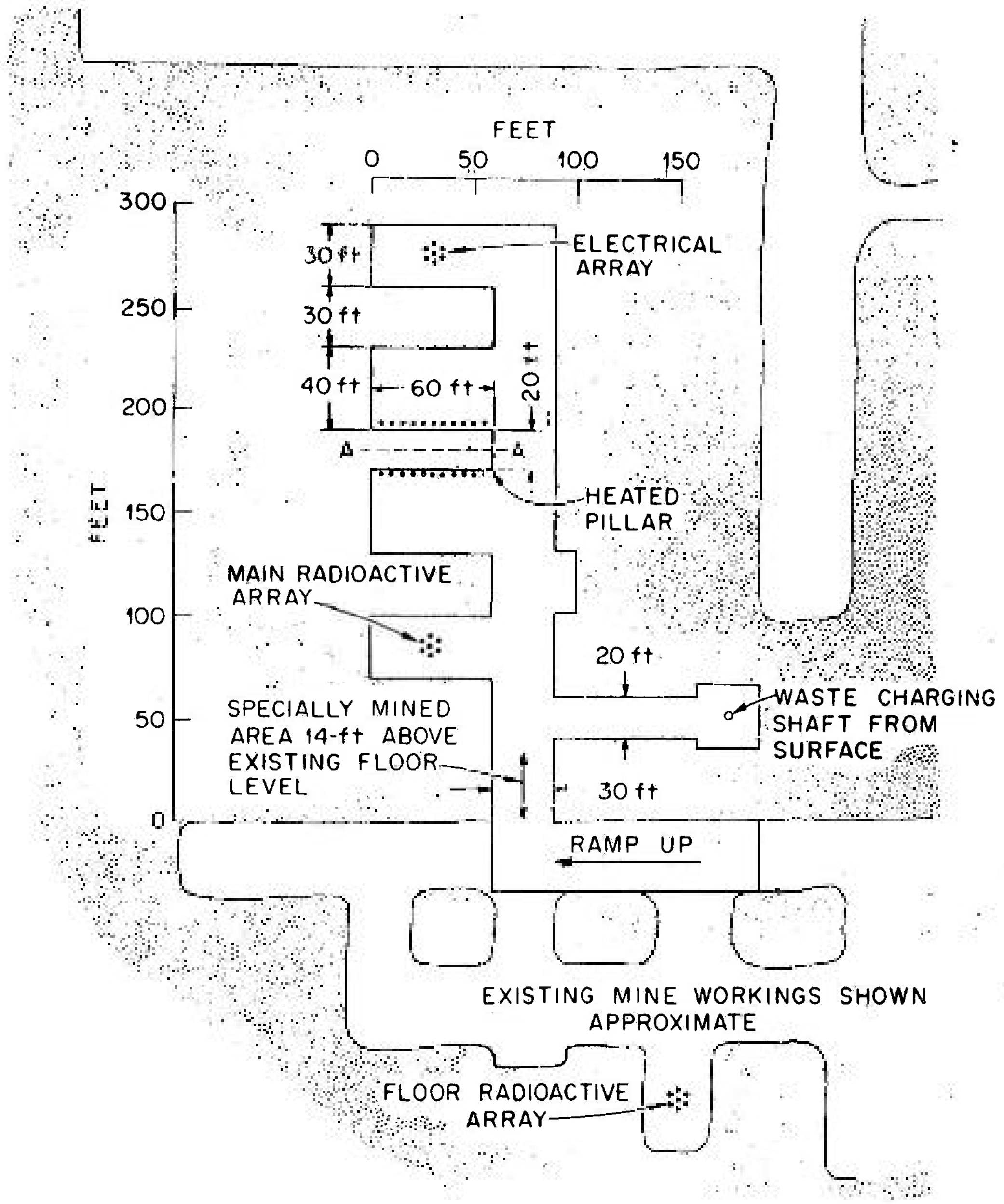
DEMONSTRATION OBJECTIVES

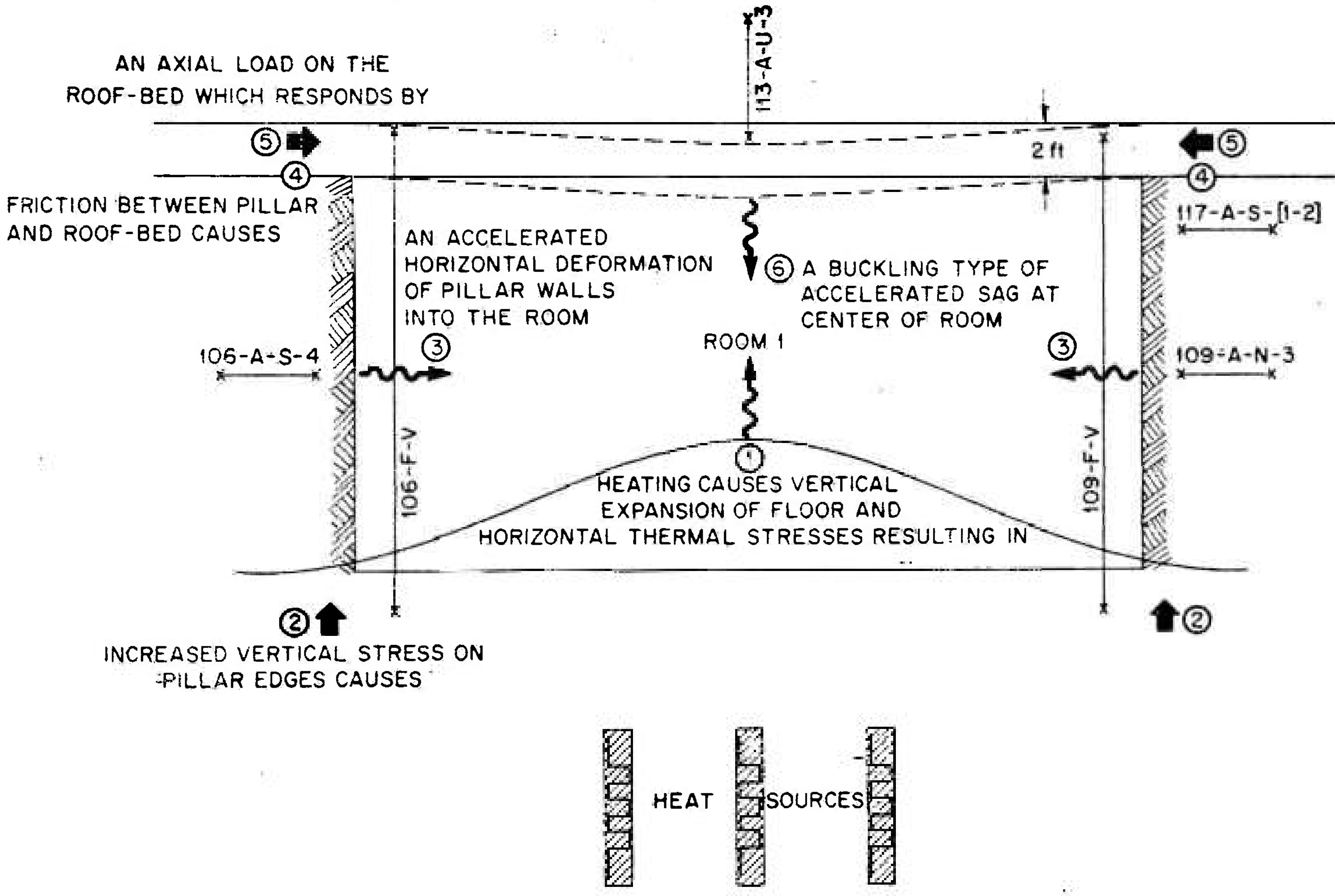
1. CONFIRMATION OF FEASIBILITY AND SAFET OF DISPOSAL IN SALT MINES.
2. THE DEMONSTRATION OF REQUIRED WASTE-HANDLING EQUIPMENT AND TECHNIQUES.
3. THE DETERMINATION OF THE STABILITY OF SALT UNDER THE INFLUENCE OF HEAT AND RADIATION.
4. THE COLLECTION OF INFORMATION ON CREEP AND PLASTIC FLOW OF SALT WHICH IS NEEDED FOR THE DESIGN OF AN ACTUAL DISPOSAL FACILITY.



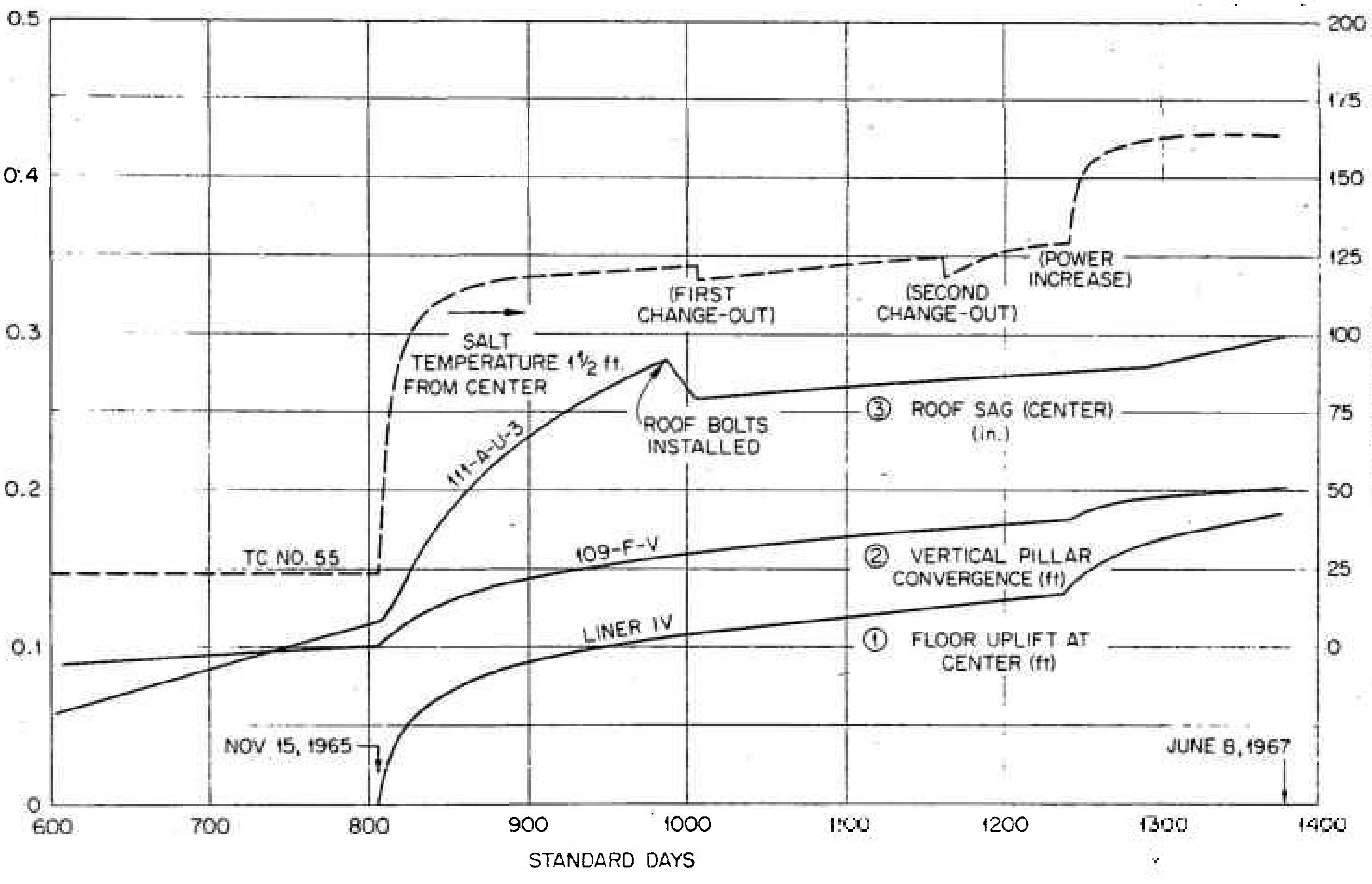
OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

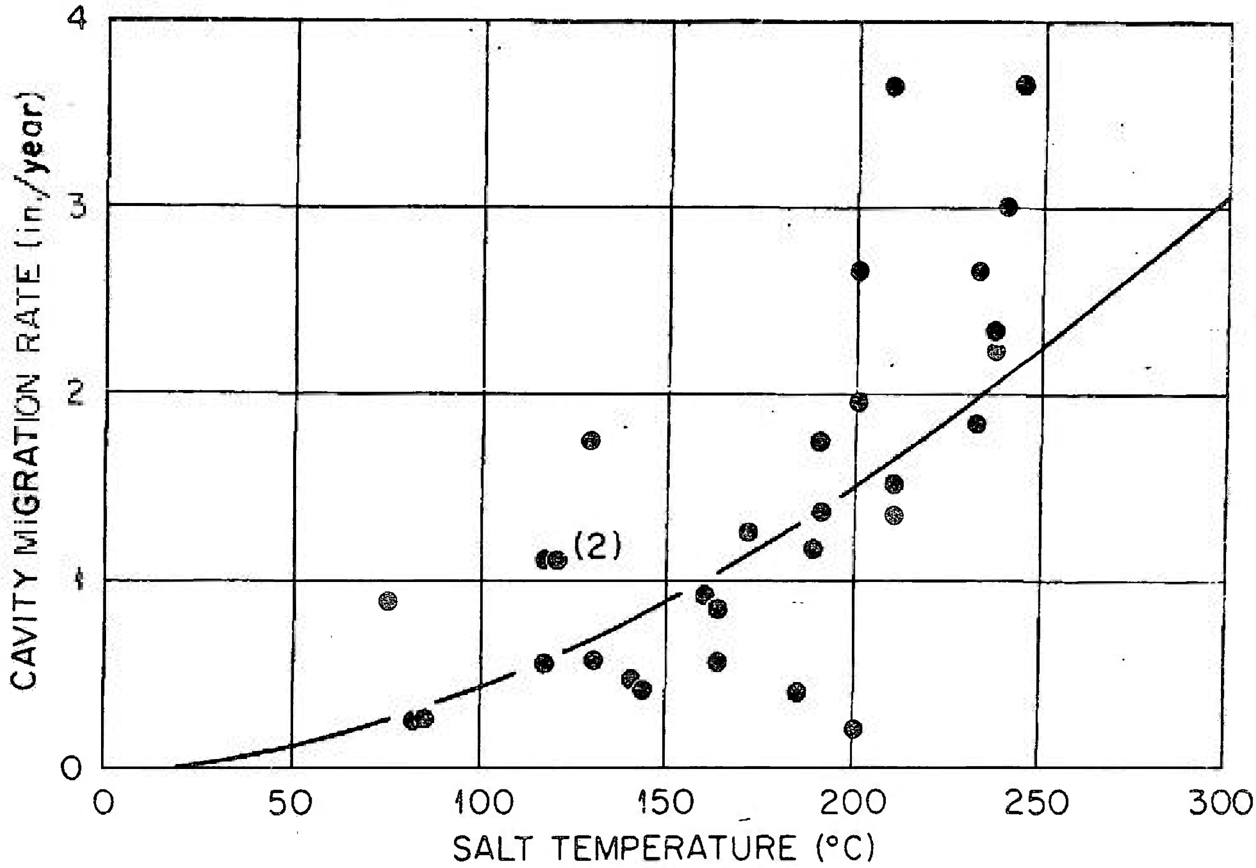
DEMONSTRATION OF RADIOACTIVE SOLIDS DISPOSAL IN SALT


FEET
0
50
100
150
300
250
200
150
100
50
0
FEET
30 ft
30 ft
40 ft
60 ft
20 ft
A
A
ELECTRICAL ARRAY
HEATED PILLAR
MAIN RADIOACTIVE ARRAY
20 ft
WASTE CHARGING SHAFT FROM SURFACE
SPECIALLY MINED AREA 14-ft ABOVE EXISTING FLOOR LEVEL
30 ft
RAMP UP
EXISTING MINE WORKINGS SHOWN APPROXIMATE
FLOOR RADIOACTIVE ARRAY



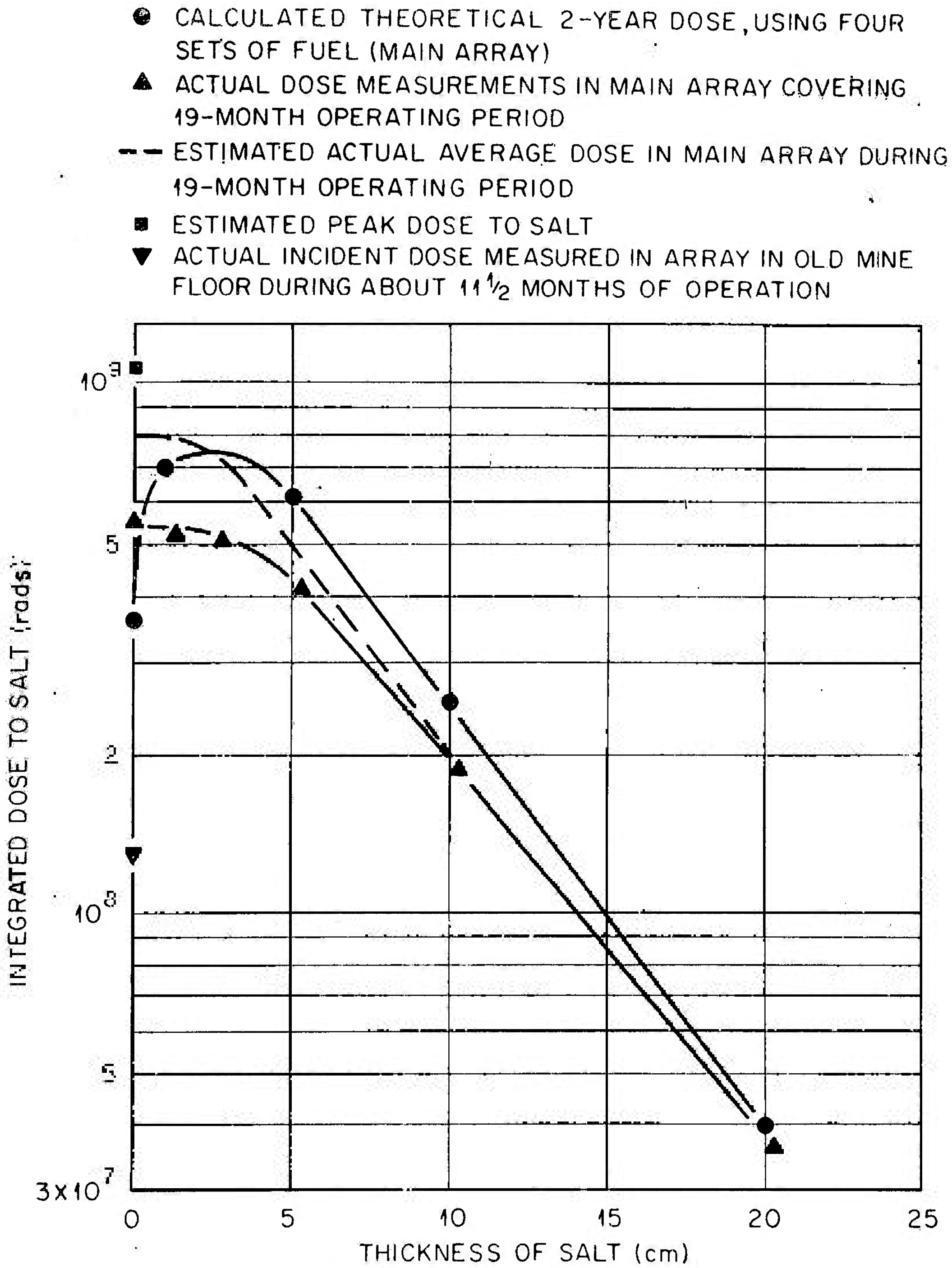
Mechanism of Stress Transference from Floor to Ceiling;
Numerical Sequence.


0.5
0.4
0.3
0.2
0.1
0
200
175
150
125
100
75
50
25
0
TEMPERATURE (°C)
600
700
800
900
1000
1100
1200
1300
1400
STANDARD DAYS
(POWER INCREASE)
(FIRST CHANGE-OUT)
(SECOND CHANGE-OUT)
SALT TEMPERATURE 1½ ft. FROM CENTER
11-A-U-3
ROOF BOLTS INSTALLED
③ ROOF SAG (CENTER) (in.)
TC NO. 55
109-F-V
② VERTICAL PILLAR CONVERGENCE (ft)
LINER IV
① FLOOR UPLIFT AT CENTER (ft)
NOV 15, 1965
JUNE 8, 1967

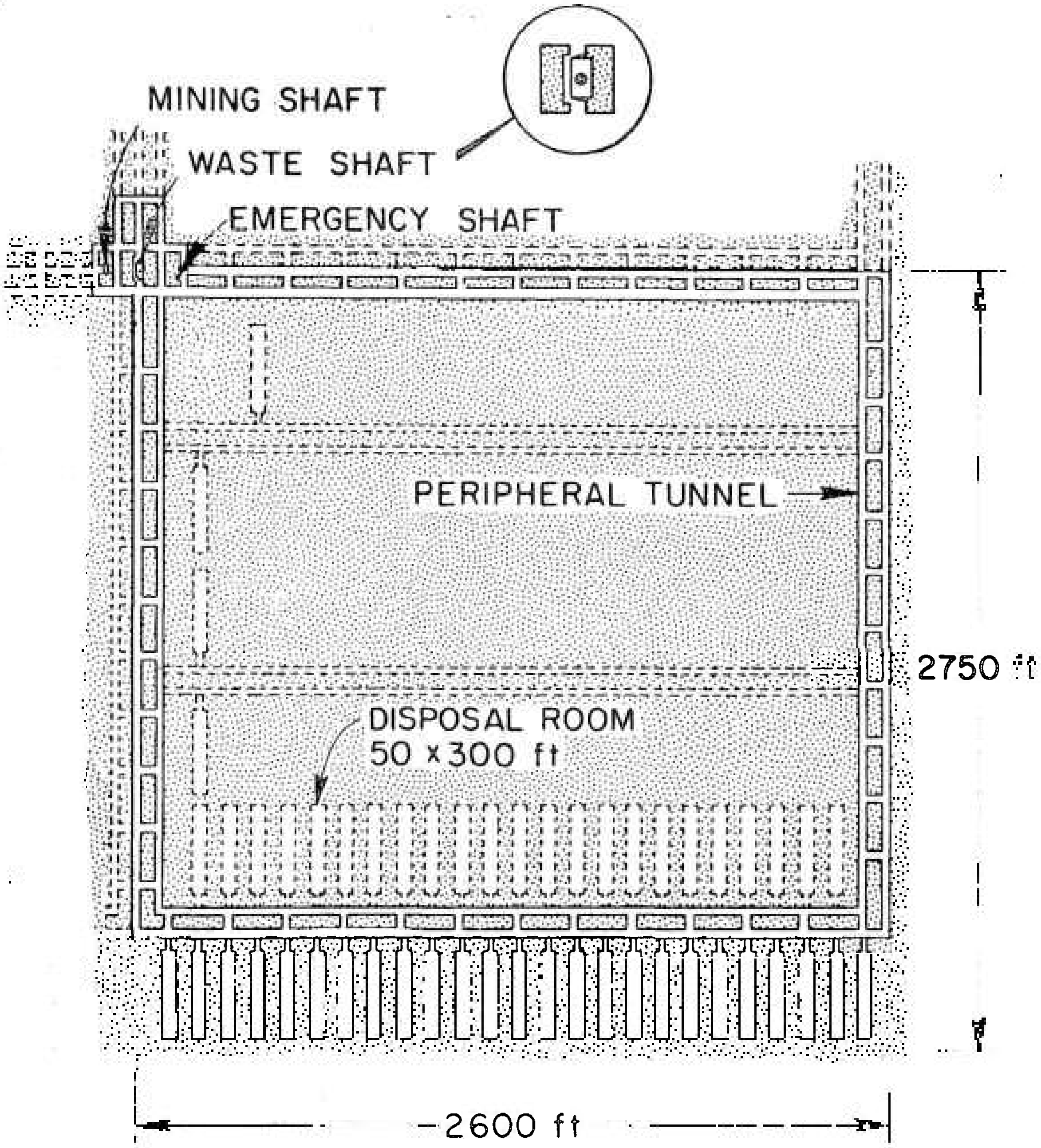


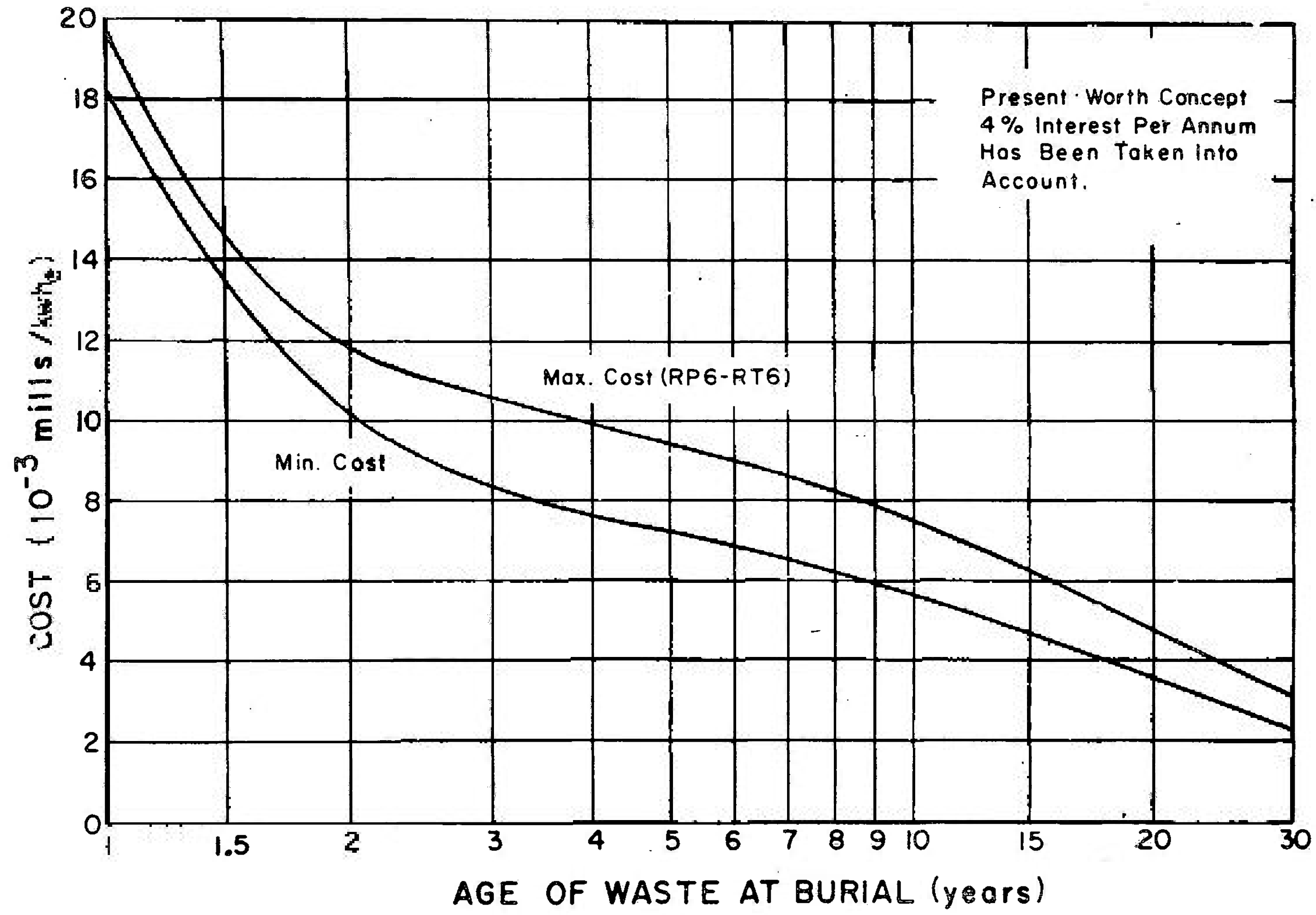
Comparison of Theoretical and Experimental Migration Rates of Brine-Filled Cavities in Rock Salt.*

* Data Normalized to Salt Temperature Gradient of 1°C/cm.



Comparison of Measured and Calculated Doses to Salt.


MINING SHAFT
WASTE SHAFT
EMERGENCY SHAFT
PERIPHERAL TUNNEL
DISPOSAL ROOM
50 x 300 ft
2750 ft
2600 ft



Present · Worth Concept
4 % Interest Per Annum
Has Been Taken Into
Account.
Max. Cost (RP6-RT6)
Min. Cost
COST (10^{-3} mills / kWh$_e$)
20
18
16
14
12
10
8
6
4
2
0
1
1.5
2
3
4
5
6
7
8
9
10
15
20
30
AGE OF WASTE AT BURIAL (years)

WASTE MANAGEMENT DATA FOR CONVERSION-TO-SOLIDS CONCEPT
(30 YEARS INTERIM SOLIDS STORAGE)

	CALENDAR YEAR			
	1970	1980	2000	(2030)[c]
SOLID WASTE PRODUCTION, FT^3/YR[a]	1,000	9,000	46,000	(--)
30-YEAR INTERIM SOLID STORAGE				
VOLUME IN STORAGE, FT^3	2,000	35,000	550,000	(--)
LENGTH 48-FT-WIDE CANALS, FT	22	390	6,080	(--)
1000-MILE SHIPMENT TO SALT MINES[b]				
NUMBER SHIPMENTS PER YEAR	0	0	20	(906)
NUMBER CASKS IN TRANSIT	0	0	1	(5)
DISPOSAL IN SALT MINES				
AREA REQUIRED, ACRES/YR	0	0	2.6	(118)
ACCUMULATED AREA USED, ACRES	0	0	5.1	(1415)

---

[a] ONE CU FT SOLID WASTE PER 10,000 Mwd(t) FUEL EXPOSURE.

[b] THIRTY-SIX, 6-IN.-DIAM CYLINDERS PER SHIPMENT CASK.

[c] COMMITMENTS MADE IN THE YEAR 2000.

WASTE MANAGEMENT DATA FOR CONVERSION TO SOLIDS CONCEPT
(2 YEARS INTERIM SOLID STORAGE)

	CALENDAR YEAR			
	1970	1980	1990	2000
SOLID WASTE PRODUCTION, 10^3 FT^3/YR[a]	1	9	25	46
2-YR INTERIM SOLID STORAGE				
VOLUME IN STORAGE, 10^3 FT^3	2	15	47	85
LENGTH 24-FT-WIDE CANALS, FT	45	330	1030	1890
1000-MILE SHIPMENT TO SALT MINES[b]				
NUMBER SHIPMENTS PER YEAR	12	105	415	790
NUMBER CASKS IN TRANSIT	1	1	3	5
DISPOSAL IN SALT MINES				
AREA REQUIRED, ACRES/YR	3.4	30	120	230
ACCUMULATED AREA USED, ACRES	3.4	150	890	2680

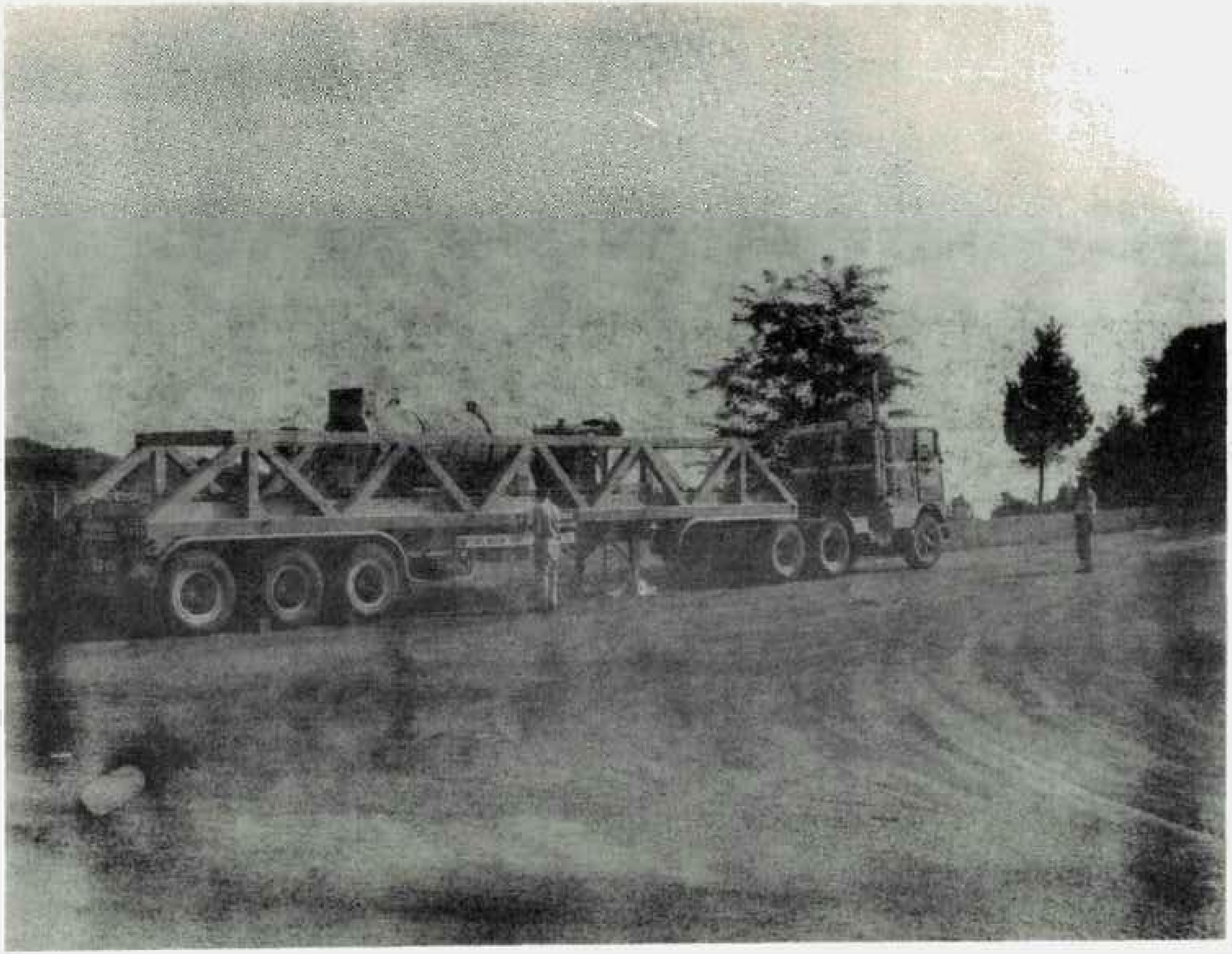
---

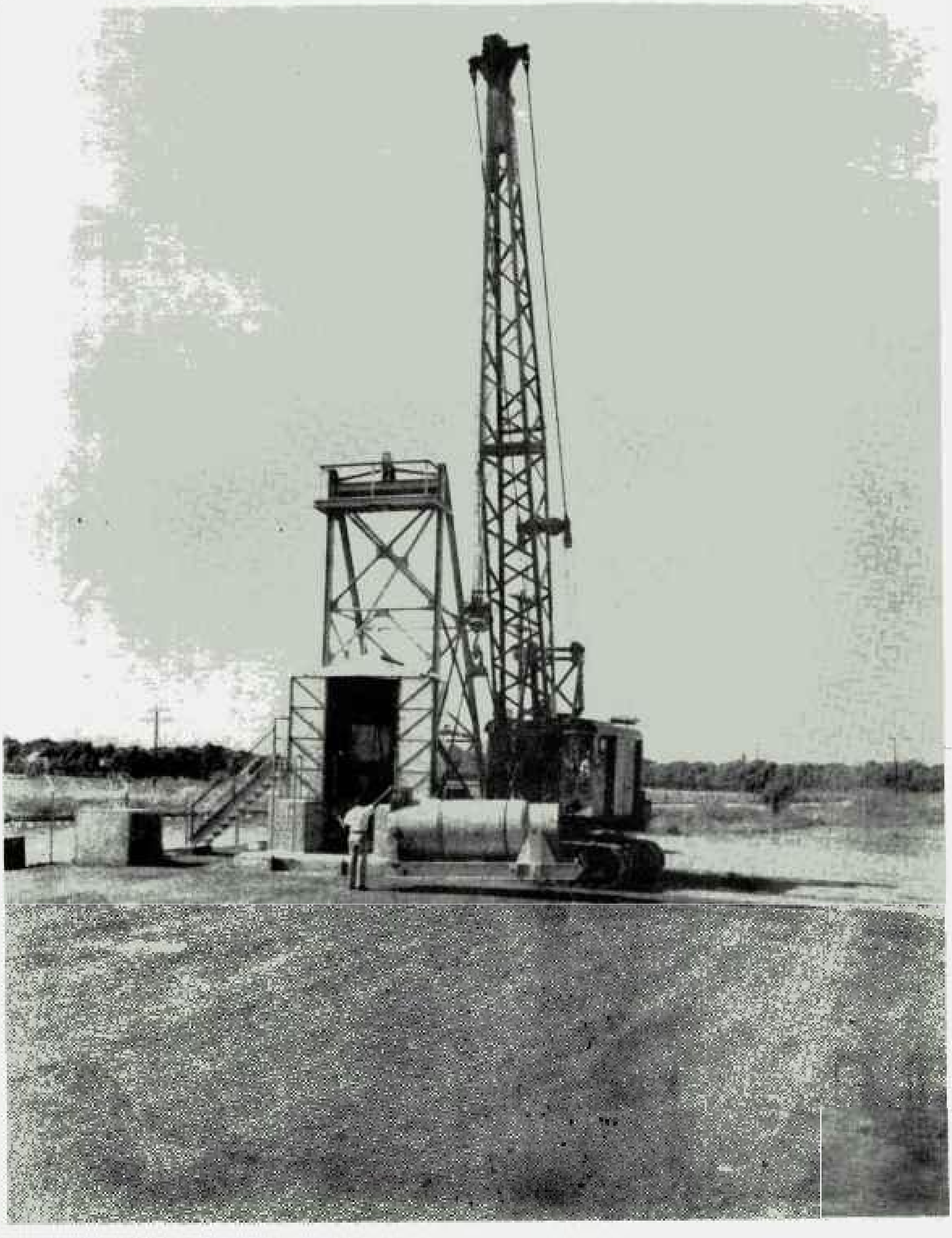
[a] ONE CU FT SOLID WASTE PER 10^4 Mwd(th) IRRADIATION.

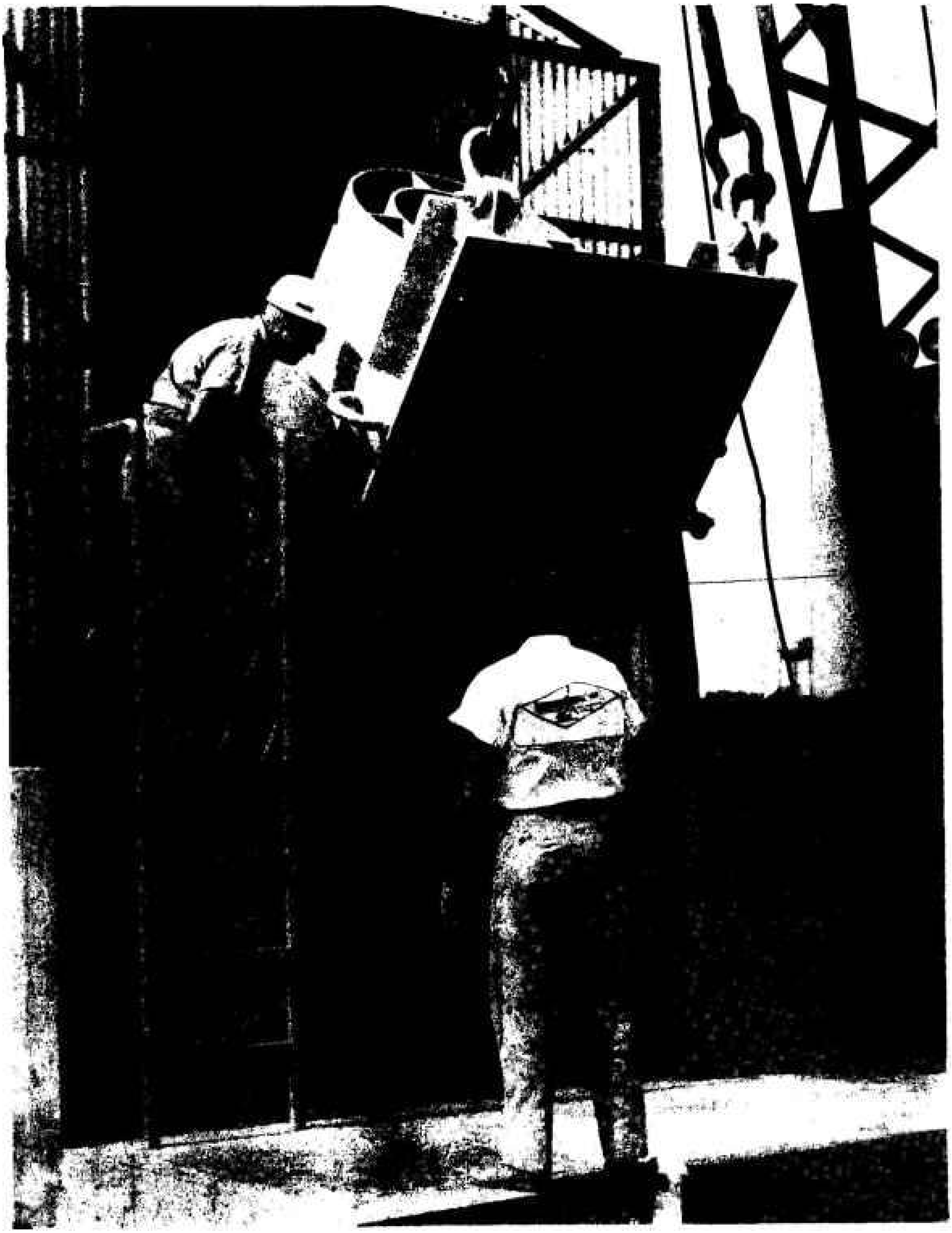
[b] THIRTY-SIX, 6-IN.-DIAM CYLINDERS PER SHIPMENT CASK.

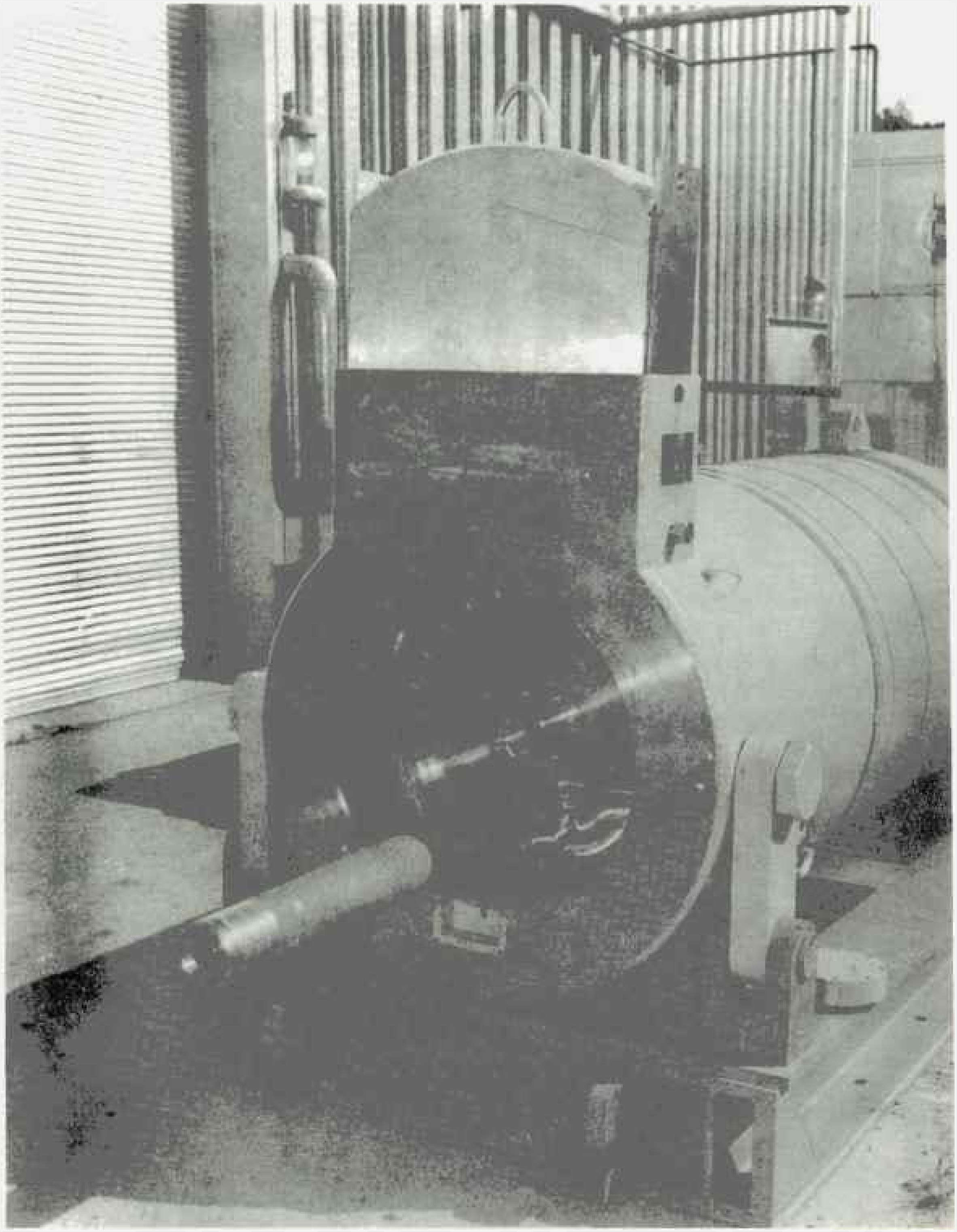
MAJOR RESULTS

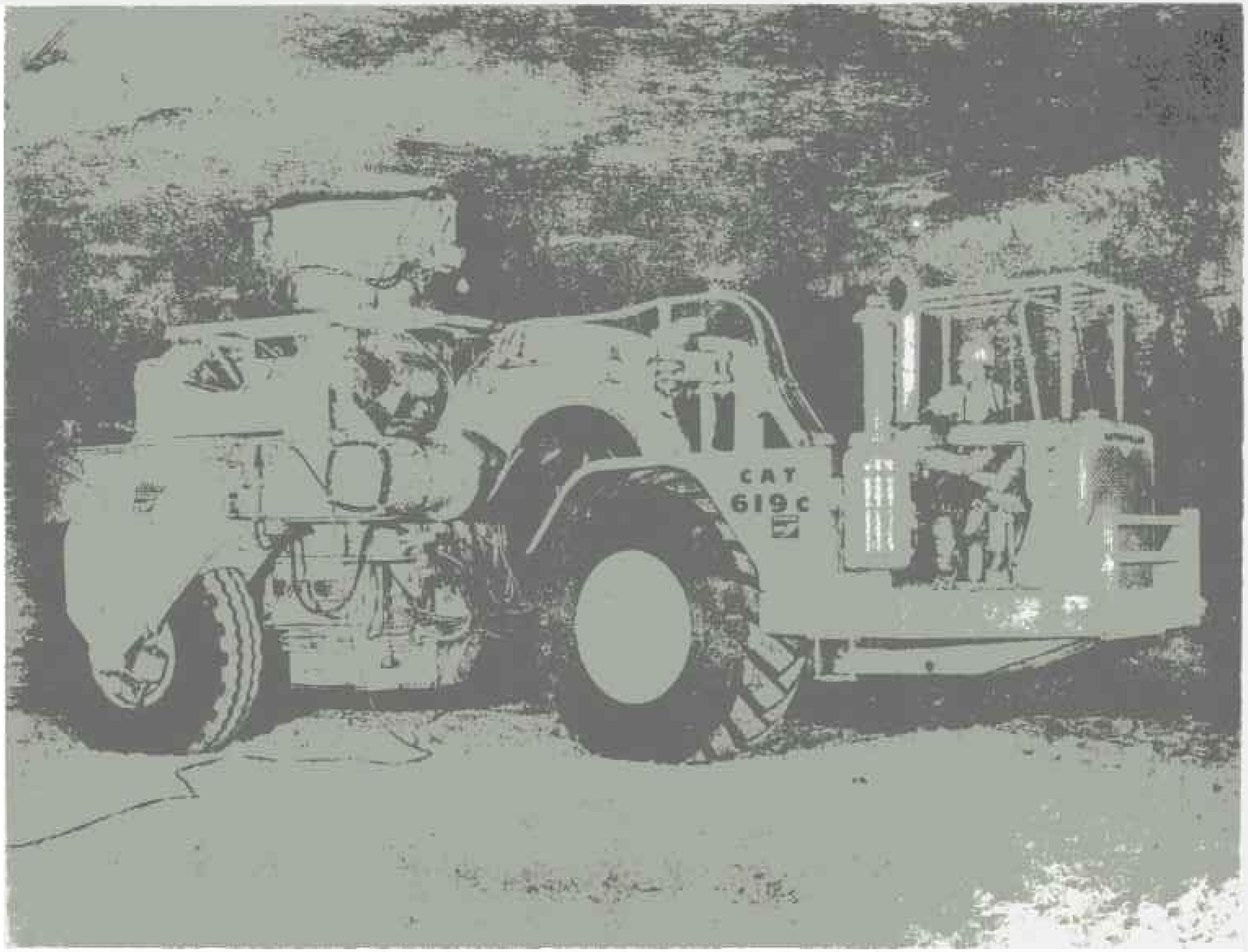
1. DEMONSTRATED EQUIPMENT AND TECHNIQUES.
2. ACHIEVED DOSAGES IN SALT OF 10^9 RAD.
3. RADIATION HAS NO SIGNIFICANT EFFECT ON SALT BEHAVIOR.
4. NO RELEASE OF CHLORINE DETECTED.
5. THERMAL STRESSES FROM CONS IN FLOOR TRANSMITTED RAPIDLY TO PILLARS AND ROOF.
6. SMALL BRINE-FILLED CAVITIES IN SALT MIGRATE TOWARD HEAT SOURCE.
7. SALT FLOW CHARACTERISTICS AS FUNCTION OF TEMPERATURE.

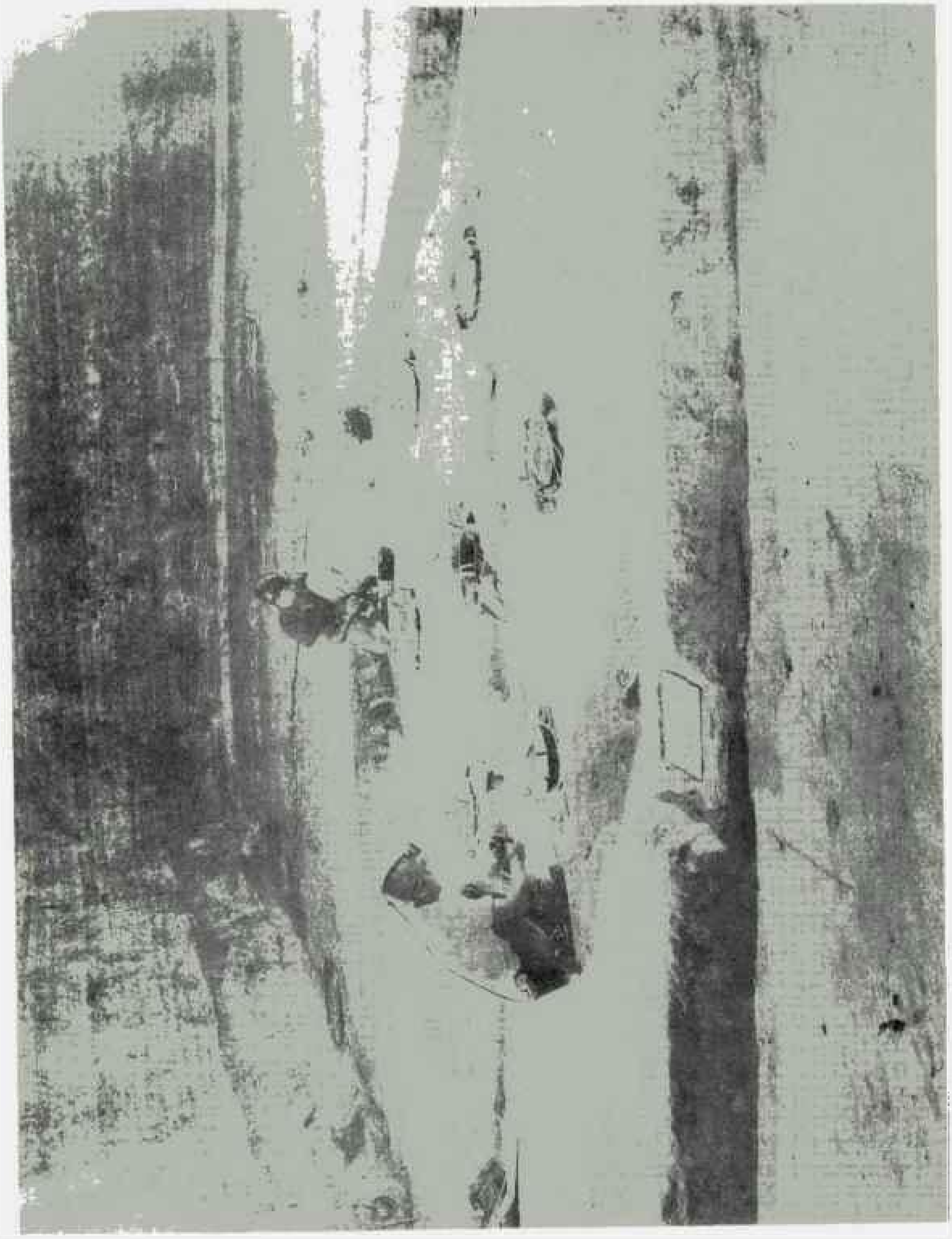


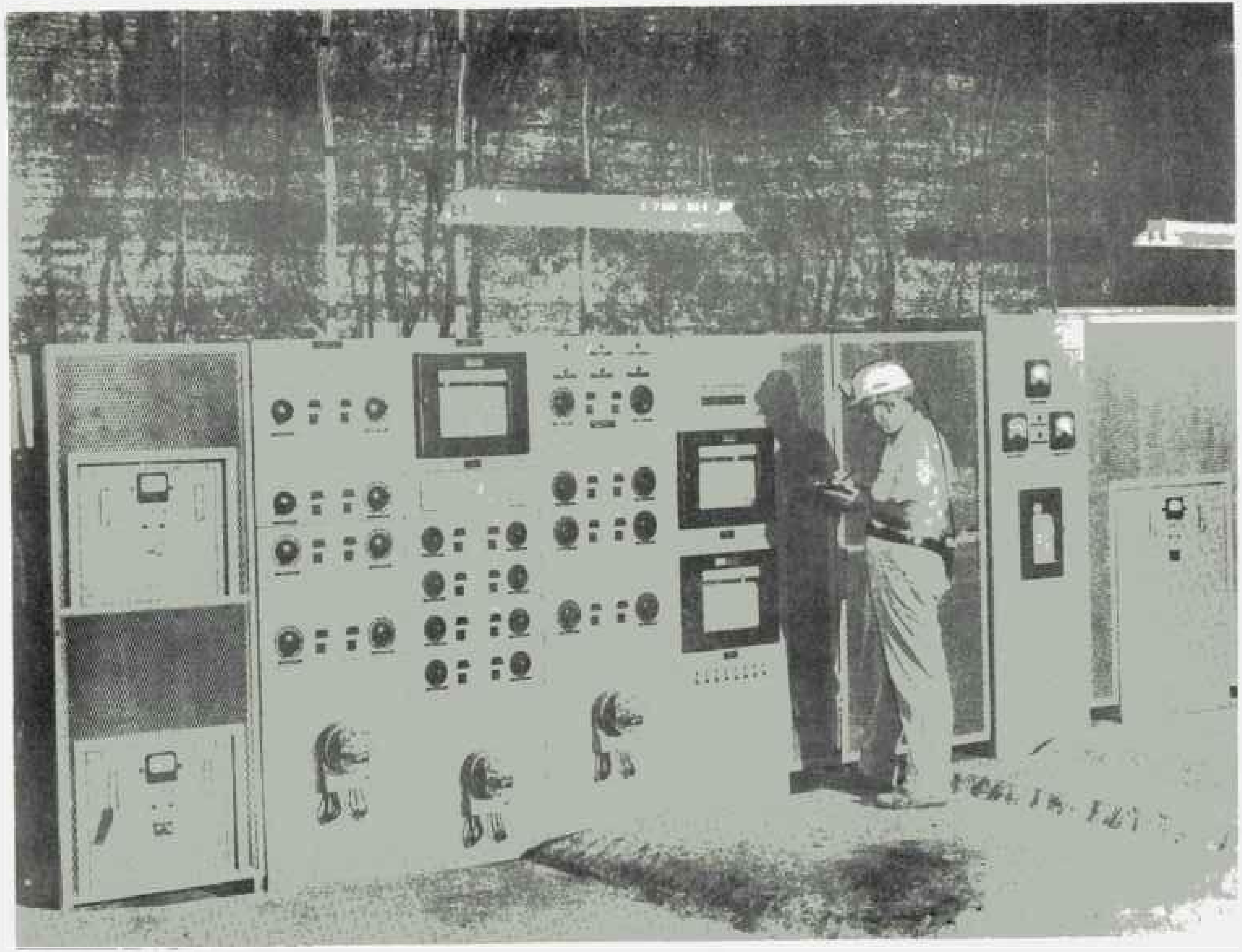


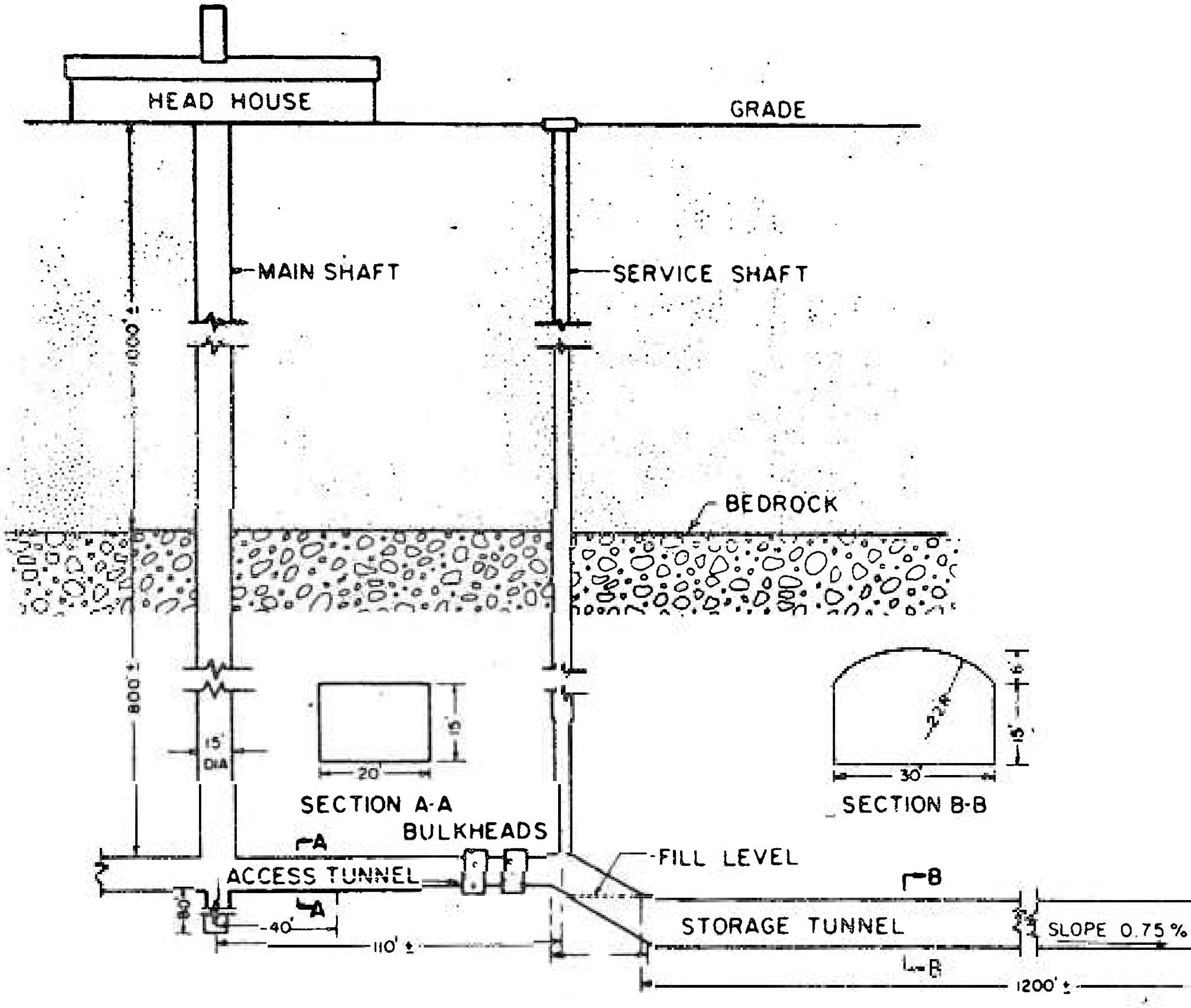





CAT
619C

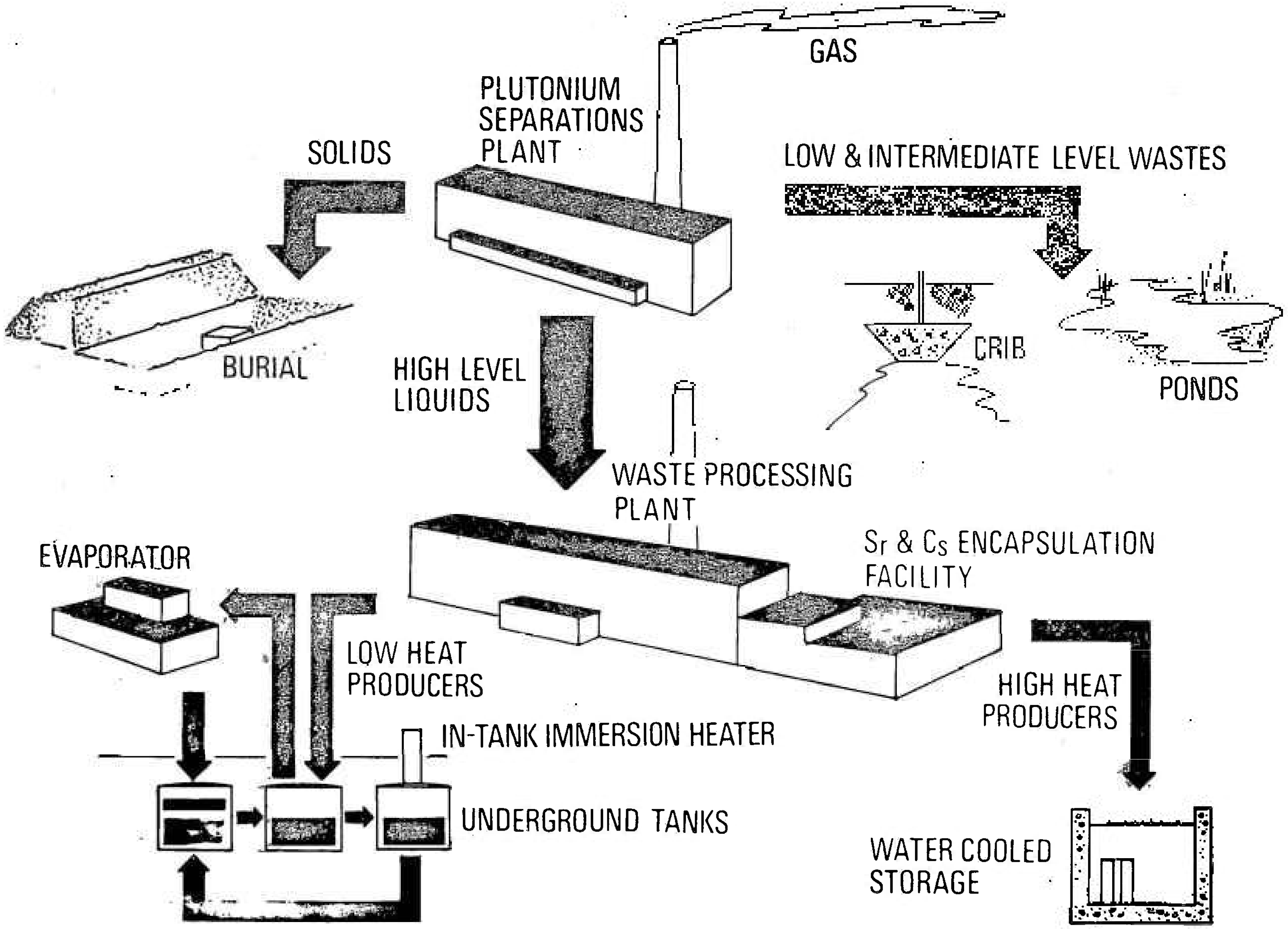




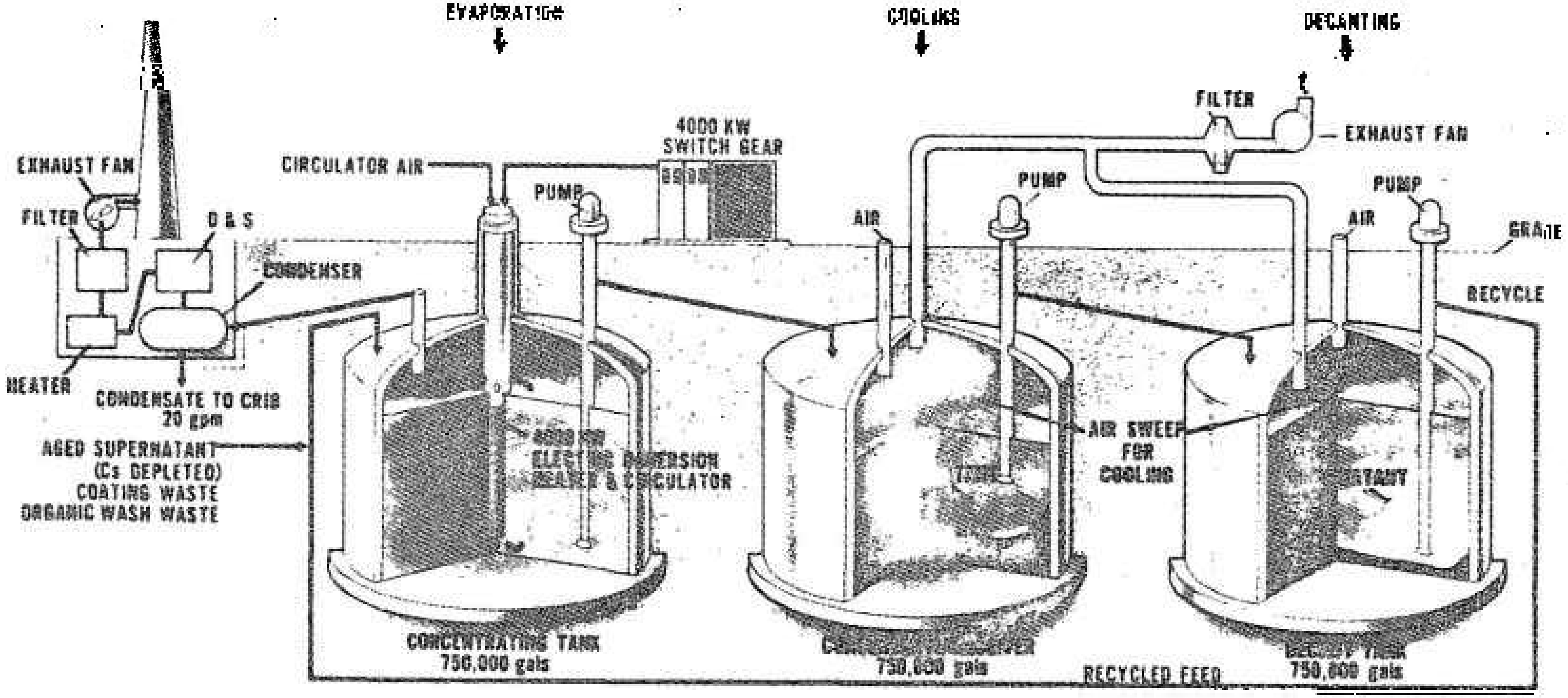


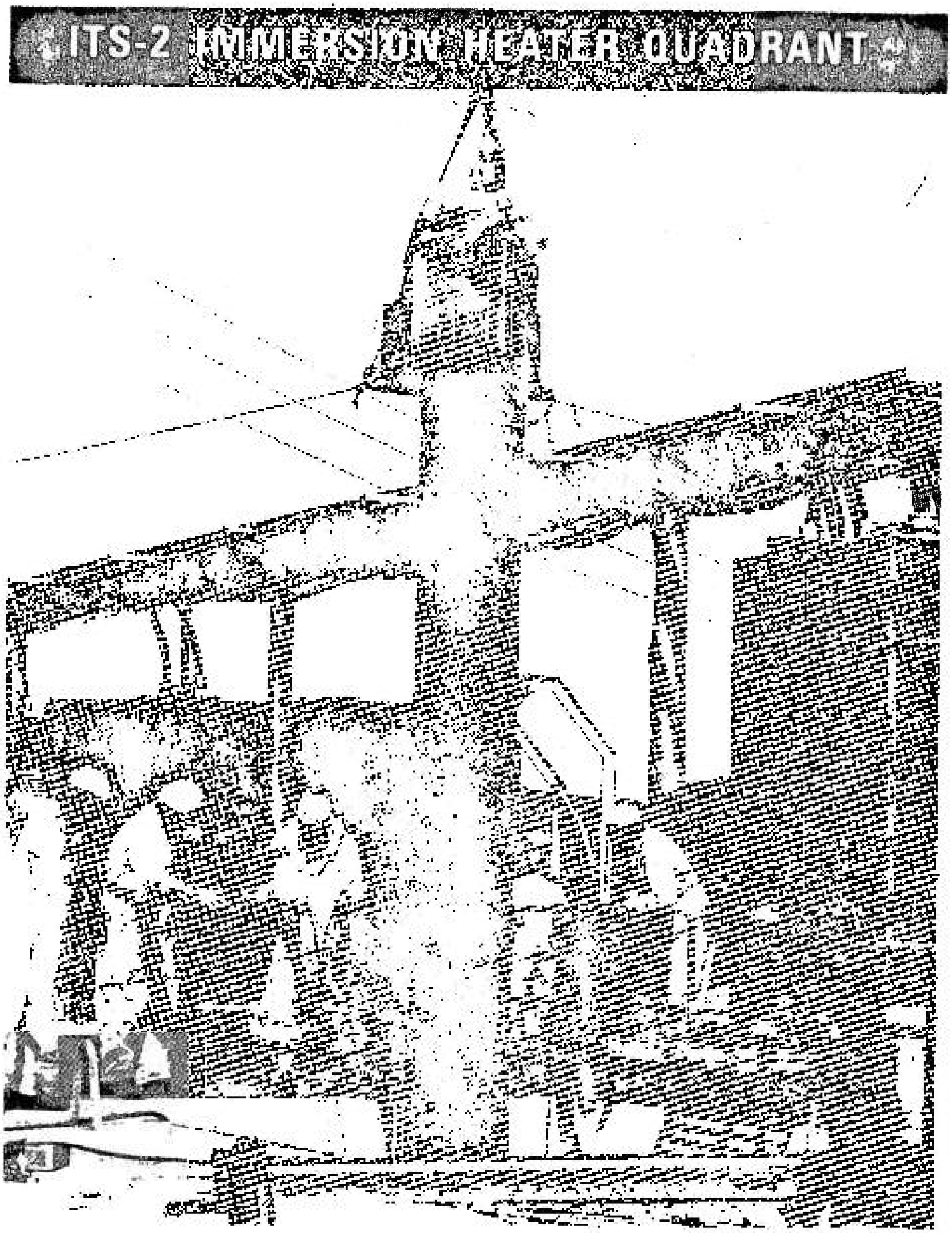
STORAGE VAULT FOR RADIOACTIVE WASTES

WASTE MANAGEMENT PROGRAM

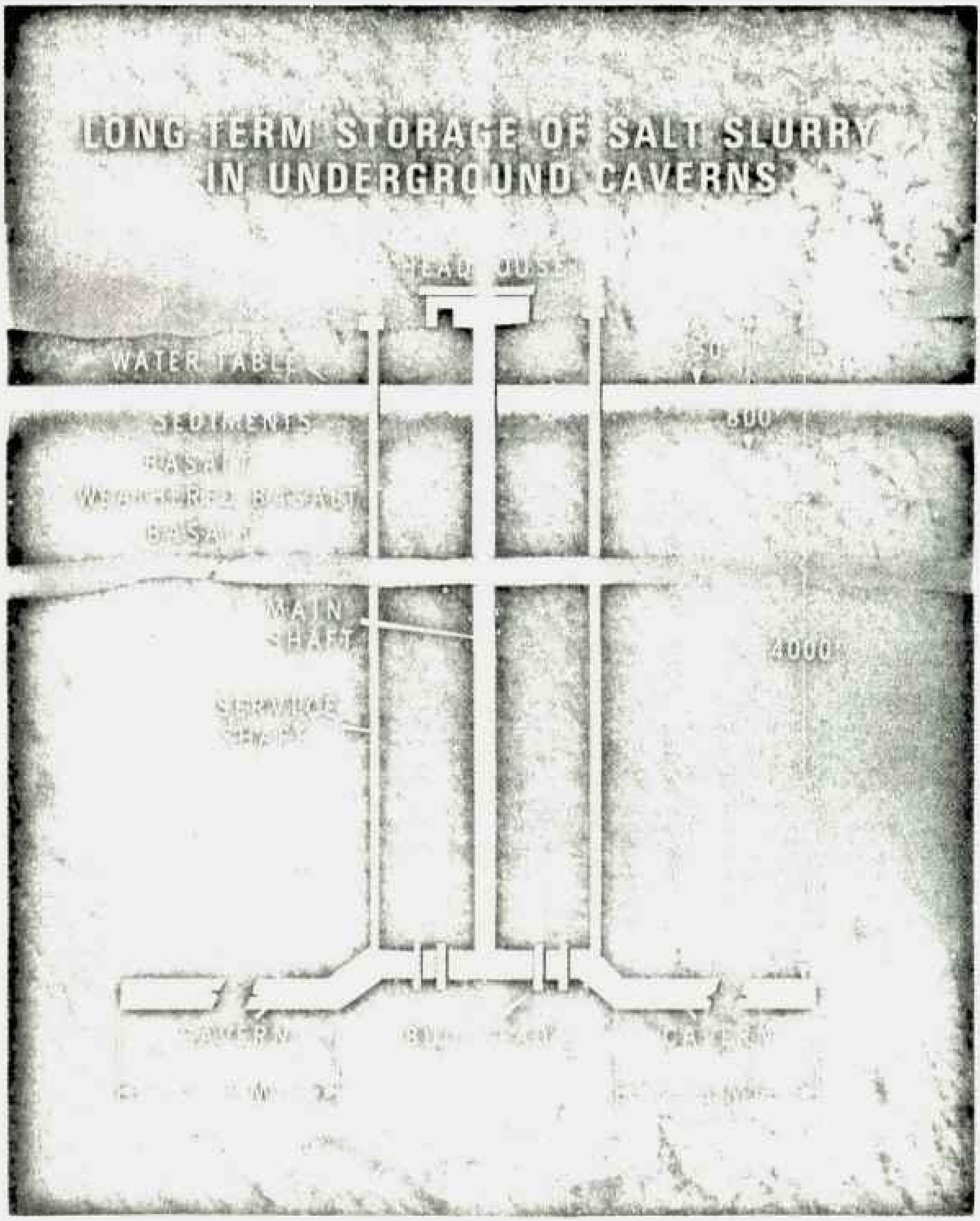


IN-TANK SOLIDIFICATION




ITS-2 IMMERSION HEATER QUADRANT

LONG TERM STORAGE OF RADIOACTIVE WASTES
SALT CAKES IN TANKS


LONG TERM STORAGE OF SALT SLURRY IN UNDERGROUND CAVERNS
HEAD HOUSE
WATER TABLE
0
SEDIMENTS
800
BASALT
BASALT
MAIN SHAFT
4000
CAVERN

COSTS OF WASTE MANAGEMENT

R. E. Blanco

LOW AND INTERMEDIATE LEVEL WASTES

Methods for determining the cost of waste management are different at each nuclear installation, consequently, the costs of waste management at one site cannot be compared with those at other locations. A panel of experts convened at the IAEA to prepare a standard method for acessing waste management costs for low and intermediate level wastes. The recommendations are listed in Technical Reports Series No. 83, "Economics in Managing Radioactive Wastes." Reports to the IAEA should use this form.

The following data must be collected to make a complete cost estimate.

SLIDE 1 Elements of Waste Management (69-8589) (69-8590) (69-8591)

The costs must be determined for each of these basic elements.

SLIDE 2 Summary Guide to Waste Management Costs (69-9170) (69-9171) (69-9172) (69-9173)

a) The costs for each of the items listed at the top must be tabulated for each of the elements shown on the left.

b) These costs were tabulated for the seven atomic installations that joined in the panel study and are shown as percentage of total direct costs.

c) Large variations in total costs and unit costs are apparent. For example, labor costs varied by a factor of 50. The factors influencing the costs are shown in the next slide.

SLIDE 3 Factors that Influence the Cost of Waste Management (69-8593)

HIGH LEVEL WASTE COSTS

The method for reporting high level waste cost has never been standardized. The next series of slides illustrate the method (and results) used by J. O. Blomeke and co-workers at ORNL to predict the costs of waste management for high level waste.

SLIDE 4 Management of High-Activity Wastes (72614-R5)

The first step was to prepare a conceptual design for each of the major operation interim liquid storage, pot calcination, interim solid storage, shipment, and final disposal in salt or alternatively for percentual storage in tanks. In the latter case, the tanks must be replaced every 25 to 75 yrs as they wear out. This slide shows the operational steps in high-level waste management from generation in a fuel processing plant to disposal in salt mines or vaults. The ORNL report numbers on this slide refer to published documents on economic studies that have been made.

SLIDE 5 Basis for Study (65-7825-R1)

Slide 5 gives the basis for the economic studies. Although the case of alkaline waste management was included, costs and operational efficiency favor management of these wastes in acid form. Plants were sized on the basis of fission-product throughput rather than fuel tonnages [100 gal acid waste per 10^4 Mwd (thermal)]; so fuel burnups had little effect on results.

FISSION-PRODUCT SEPARATION

It may be appropriate to mention now our conclusions regarding the effect of fission-product removal. Using what we considered to be optimistic expectations of waste compositions from future fission-product separations processes, we were unable to show any substantial economic advantages to waste management by the route of solidification, encapsulation, shipping, and disposal in salt mines. The cost of managing wastes that are 90 to 99% depleted in fission products is about 70% as much as the cost of managing wastes with no fission products removed. The difference, about $500 to $1000 per ton of fuel processed, is not adequate to pay for the separation, encapsulation, and eventual disposal of fission-product concentrates. We concluded, therefore, that fission-product removal must be justified and paid for by the market for fission-product radiation or heat sources, with only a marginal credit from reduced costs of waste management.

PERPETUAL TANK STORAGE

SLIDE 6 Tanks (64-9146R)

For perpetual tank storage, the economics are examined for three representative types of financing: government ownership, private ownership, and a combination of governement and private ownership. The case of governement ownership includes only depreciation and interest at 4% on the investment capitalt whereas, in the case of private ownership, costs reflect a 15% return on equity, as well as allowances for depreciation, insurance, taxes, and interest. In the third case, private ownership is assumed during the 20-year period of waste accumulation, after which the government will assume responsibility for perpetual care of the tank farm. In each case, it is assumed that a permanent tax-free fund is established during the filling period of such size that the annual tax-free interest will be sufficient to provide for periodic replacement of tanks and for the annual operating expense of the facility. The tank farms were designed for storing high-level wastes in both acid and alkaline forms, and in tanks ranging in capacity from 200,000 to 5,000,000 gallons.

This figure shows a conceptual layout of a "completed" tank farm, containing 20 yesrs' accumulation of waste. The farm is divided into two areas, one containing tanks of high-level waste and their associated cooling and ventilation facilities, and the other containing tanks of cladding wastes which are not considered in the cost analyses of this paper. The high-level wastes are stored in 10^6-gallon tnaks grouped around three sides of an operations building containing many of the major equipment items of the cooling and ventilation systems. A cooling tower and pumps, an emergency water storage tank, a water surge tank, and a stack and fans are also located in the high-level waste area. In addition to filled tanks, an empty tank is always maintained "on standby" to receive the contents of any tank which may have failed.

SLIDE 7 Schematic of SRP Waste Storage Tank (Photo 46189)

The tanks are similar is design to those in use at the Savannah River Plant. They are cylinders having a diameter-to-height ratio of 3, are fabricated of ½- to 1-in.-thich steel plate, and are housed in steel-lined vaults with walls 3 to 3-½-ft-thick, buried under 10 ft of earth. Heated air is circulated through the annular space between the tank and the vault for dehumidification, and the annulus is monitored for both liquid and airbone radioactivity. The tanks have steel-lined, internal columns for support, and are equipped with water-cooling coils which maintain the contents at or below 140°F during storage. The water in these coils is circulated to heat exchangers in the operations building where the heat is transferred to a secondary cooling circuit, and is subsequently dissipated in the cooling tower. A water-cooled condenser located in the operations building serves as the secondary means for heat removal, and ateam produced from self-boiling wastes during emergencies is vented to this condenser through a 4-ft-diam off-gas header in the top of the tank. During normal operation, air is swept through the vapor space in the tanks to remove radiolytic hydrogen, and it is vented through the off-gas header to fibrous glass filters and the stack. The tanks are provided with instrumentation to measure temperature, radiation, liquid levels, etc., and with access ports for introducting the necessary equipment for evacuation for the tanks when this becomes desirable.

SLIDE 8 Range of Incremental Costs During 20-Year Accumulation Period (66-13)

This slide shows the order-of-magnitude ranges of incremental costs incurred during the 20-year waste accumulation period.

SLIDE 9 Cost of Acid Waste Storage as a Function of Tank Capacity, Tank Life and Method of Financing for (a) Case 1, Government Ownership; Case 2, Private and Government Ownership and (b) Case 3, Private Ownership (65-6188-R1)

This slide shows costs, in terms of mills/kwhr (electrical), as a function of tank capacity for tank lifetimes of 25, 50 and 75 years. Minima occur at a tank capacity of about 10^6 gal in all cases and costs range from about 0.0165 to 0.0272 mill/kwhr.

CONVERSION TO SOLIDS AND DISPOSAL IN SALT

SLIDE 10 Costs of Interim Storage of Acid Wastes as a Function of Storage Time (65-7856)

This slide shows costs for interim liquid and solid waste storage as a function of storage time (government financing). Liquid storage costs were based on a model similar to that used for perpetual liquid storage and provide for reuse of tanks over short storage intervals. Solid storage is carried out in water-filled canals and the costs are only from 1/3 to 1/5 those of liquid storage. Government financing was used for this study on solidification and burial in salt.

SLIDE 11 Costs of Calcination and Shipment as a Function of Age of Waste (65-7854)

This slide presents pot calcination and waste shipment costs as a function of age of the waste. Costs were estimated for calcination in cylinders 6, 12, and 24 in. in diameter over the appropriate time intervals for each pot size as determined by the fission-product heat generation rate. These costs ranged from about 10.7 x 10^{-3} mills/kwhr for calcination in 6-in. pots at waste ages from 1/3 to 3 years, to about 5.7 x 10^{-3} mills/kwhr for calcination in 24-in. pots for wastes 6-½ to 30 years old. Costs for 1000-mile shipment of the pots in casks weighing 50 to 90 tons, and without forced convection cooling en route, were from 1 to 2 x 10^{-3} mills/kwhr at the minimum permissible ages for shipment. These costs decreased with increasing age of the waste because less cask shielding is required. The wavy, vertical line crossing the 6- and 12-in.-diam pot curve indicates the minimum age for shipment of 12-in.pots.

SLIDE 12 Cost of Disposal in Salt as a Function of Age of Waste at Burial (68-4168)

This slide presents the estimated costs for disposal of the solidified wastes in a salt mine. These costs are based on burial of the posts in vertical holes in the floor of a mine, 1000 ft below the surface. The pots are spaced so as to dissipate the decay heat without raising the temperature of the salt above 200°C at any point. Disposal costs lie in the range 6.5 to 10.2 x 10^{-3} mills/kwhr and are inversely related to pot diameter because heat is dissipated easier from the smaller vessels and this permits more efficient utilization of space in the mine. It is of interest to note that the minimum age for disposal of 24-in.-diam pots is 30 years.

SLIDE 13 Management Costs for Acid Purex and Thorex Wastes (68-10569)

This slide presents the present costs for two cases requiring no interim liquid storage. In both cases, the wastes are calcined immediately in 6-in.diam pots. In one instance, they are stored as solids for 2 years, then shipped at this earliest possible time 1000 miles and buried in a salt mine. The total cost is 22.9 x 10^{-3} mills/kehr. In the second case, the calcined wastes are stored on-site in canals for 30 years before shipment and disposal. This case results in a total cost of only 17.5 x 10^{-3} mills/kwhr. It is less than the first case because shipment and disposal are postponed an additional 27 years, and the savings reflect the 4% interest. Also, the costs of shipment and disposal of old wastes are less.

SLIDE 14 Management Costs for Acid Purex and Thorex Wastes (68-10570)

This slide shows that management can be carried out for essentially the same costs in 24-in.-diam pots, but on a different schedule. In the first case, the wastes are calcined in the large vessels after 5-2/3 years' liquid storage, which is the earliest possible time. Since wastes in 24-in.-diam pots cannot be buried in salt according to our criteria at less than 30 years of age, the pots are stored on-site for 24 years before shipment and disposal The total cost for this schedule is 21.2 x 10^{-3} mills/kwhr. In the second case, the wastes are stored as liquids in tanks for 30 years, then calcined and shipped immediately to a salt mine. This cost is 18.3 x 10^{-3} mills/kwhr.

These costs indicate that the least-expensive management consists of immediate solidification and then providing interim liquid or solid storage for 30 years before disposal in salt. This could be done for about the same total cost as that for perpetual liquid storage. If for any reason, such as safety, it were desired to carry out the schedule with the least practical delay, we would expect to pay about 30% more.

These costs represent only about 10% of the total estimated for fuel processing in 1970. However, it must be remembered that they are based on a fixed size, or scale of operations. We have not looked carefully at the effect of scaleup on waste costs, but early indications are that waste management costs do not scale as favorably with size as do fuel processing costs. So while they may represent only about 10% of 1970 fuel processing costs, they may represent 20 to 25% of a 0.05 mill processing cost predicted for the year 2000.

PRINCIPAL REFERENCES AND EXCERPTS FROM.

Low and Intermediate Waste

1. *Economics in Managing Radioactive Wastes*, Technical Reports Series No. 83, IAEA, April 1968.

High Level Waste

1. J. O. Blomeke, R. Salmon, J. T. Roberts, R. L. Bradshaw, and J. J. Perona, "Estimated Costs of High-Level Waste Management," in Proceedings of the Symposium on the Solidification and Long-Term Storage of Highly Radioactive Wastes, Feb. 14-18, 1966, Richland, Washington, pp. 830-43, ed. W. H. Regan USAEC, CONF-660208 (November 1966).

Additional References:

1. R. L. Bradshaw, J. J. Perona, J. T. Roberts, and J. O. Blomeke, *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes. I. Interim Liquid Storage*, USAEC Report ORNL-3128, Oak Ridge National Laboratory, August 1961.

2. J. J. Perona R. L. Bradshaw, J. T. Roberts, and J. O. Blomeke, *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes. II. Conversion to Solid by Pot Calcination*, USAEC Report ORNL-3192, Oak Ridge National Laboratory, September 1961.

3. J. O. Blomeke, J. J. Perona, H. O. Weeren, and R. L. Bradshaw, *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes. III. Interim Storage of Calcined Solid Wastes*, USAEC Report ORNL-3365, Oak Ridge National Laboratory, October 1963.

4. J. J. Perona, R. L. Bradshaw, J. O. Blomeke, and J. T. Roberts, *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes. IV. Shipment of Calcined Solids*, USAEC Report ORNL-3356, Oak Ridge National Laboratory, October 1962.

5. J. J. Perona, J. O. Blomeke, R. L. Bradshaw, and J. T. Roberts, *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes. V. Effects of Fission Product Removal on Costs of Waste Management*, USAEC Report ORNL-3357, Oak Ridge National Laboratory, June 1963.

6. R. L. Bradshaw, J. J. Perona, and J. O. Blomeke, *Evaluation of Ultimate Disposal Methods for Liquid and Solid Radioactive Wastes. VI. Disposal of Solid Wastes in Salt Media*, USAEC Report ORNL-3358, in preparation.

7. J. J. Perona, R. L. Bradshaw, and J. O. Blomeke, *Comparative Costs for Final Disposal of Radioactive Solids in Concrete Vaults, Granite, and Salt Formations*, USAEC Report ORNL-TM-664, Oak Ridge National Laboratory, October 1963.

8. J. O. Blomeke E. J. Frederick, R. Salmon, and E. D. Arnold, *The Costs of Permanent Disposal of Power-Reactor Fuel-Processing Wastes in Tanks*, USAEC Report ORNL-2873, Oak Ridge National Laboratory, September 1965.

Part 1

ELEMENTS OF COSTS OF WASTE MANAGEMENT

1. Collection – Containers, Hold Up Tanks But Not Pipes, Sinks
2. Transportation – Pumps, Pipes, Trucks to Treatment Center –
From Center to Disposal.
3. Monitoring
4. Treatment – Evaporation, Chemical, Sorting Solid Wastes.
5. Concentrate Conditioning – Asphalt, Cement, Packaging - - -
6. Concentrate Storage or Disposal – Transportation, Burial Grou
Costs, Containers, Escorts - - -.
7. Effluent Discharge – Pumping, Weirs, Stacks, Dams - - -.
8. Environmental Monitoring.

Part 2

ELEMENTS OF COSTS OF WASTE MANAGEMENT

9. Input – Cubic Meters/yr.
10. Labor Charge – Man Years.
11. Direct Labor – Costs Including Social Charges.
12. Utilities, Materials, Services – Steam, Electricity, Chemicals Laundry, Analytical, Engineering - - -.
13. Current Maintenance – But Not Replacement of Buildings or Equipment.

Part 3

ELEMENTS OF COSTS OF WASTE MANAGEMENT

14. Overhead –
    a. Operational – Direct Supervision, Office Supplies, Travel, Health Physics - - -.
    b. General – Establishment Supervision, Stores, Medical Training - - -.
15. Depreciation – 20 yrs at 5%/yr for Buildings; 7 1/2 yrs at 13.3% for Plant and Equipment.
16. Applied Research – Process Improvement.
17. Interest on Capital.

SUMMARY GUIDE TO WASTE-MANAGEMENT COSTS

TYPE OR ELEMENT OF MANAGEMENT	THROUGHPUT (m^3/yr)	TOTAL DIRECT COST	
LIQUID WASTES		(US \$/yr)	(US \$/m^3)
Collection by pipeline, monitoring, discharge without treatment	8×10^4 to 8.2×10^5	4.3×10^4 to 2.7×10^5	0.1 to 0.7 ave. 0.36
Collection by pipeline, monitoring, simple treatment discharge	6.3×10^4 to 5×10^5	1.5×10^5 to 4×10^5	0.50 to 2.65 ave.* 1.65
Collection, monitoring, evaporation, discharge	900 to 3.4×10^3	8×10^4 to 2.5×10^5	29 to 207 ave. 120
Chemical treatment (only)	6.3×10^4 to 5×10^5	7.4×10^4 to 2.3×10^5	0.2 to 1.50 ave. 1.02
Evaporation (only)	900 to 3.4×10^3	1.3×10^4 to 1.3×10^5	14 to 112 ave. 43
Collection by pipeline	3×10^4 to 8.4×10^5	1.5×10^4 to 1.8×10^5	0.2 to 1.6 ave. 0.7
Collection by tanker	950 to 6.4×10^3	1.2×10^4 to 9.4×10^4	7 to 28 ave. 15
Effluent discharge	2×10^4 to 5×10^5	1.4×10^3 to 1.7×10^4	0.03 to 0.13 ave. 0.06

SUMMARY GUIDE TO WASTE-MANAGEMENT COSTS

TYPE OR ELEMENT OF MANAGEMENT	PERCENTAGES OF TOTAL DIRECT COSTS				
LIQUID WASTES	LABOR	UTILITIES, SERVICES	MAINTENANCE	OVERHEADS	DEPRECIATIO
Collection by pipeline, monitoring, discharge without treatment	5 to 28	3 to 21	2 to 19	15 to 38	21 to 60
Collection by pipeline, monitoring, simple treatment discharge	7 to 25	3 to 25	3 to 17	9 to 58	19 to 45
Collection, monitoring, evaporation, discharge	10 to 40	5 to 24	1 to 13	10 to 31	19 to 44
Chemical treatment (only)	4 to 18	7 to 20	4 to 25	9 to 38	28 to 47
Evaporation (only)	14 to 18	3 to 20	1 to 9	7 to 35	18 to 50
Collection by pipeline	6 to 25	2 to 16	2 to 18	10 to 50	20 to 63
Collection by tanker	14 to 63	1 to 19	1 to 22	15 to 35	19 to 56
Effluent discharge	6 to 40	2 to 32	1 to 20	14 to 43	16 to 88

SUMMARY GUIDE TO WASTE-MANAGEMENT COSTS

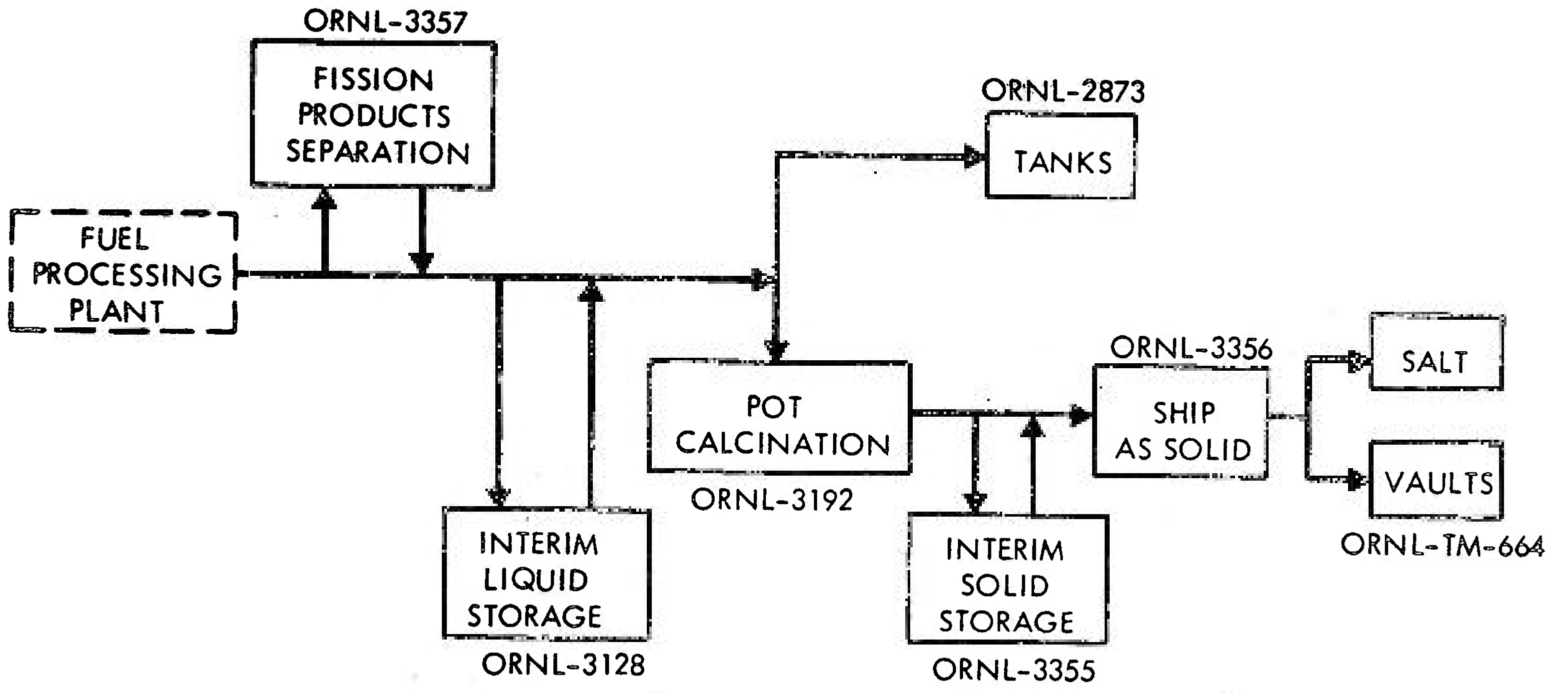
TYPE OR ELEMENT OF MANAGEMENT SOLID WASTES	THROUGHPUT (m^3/yr)	TOTAL DIRECT COST (US %/yr)	(US $\$/m^3$)
Total Management	250 to 4×10^3	1.5×10^4 to 1.4×10^5	35 to 84 ave. 60
Collecting, monitoring and transport	300 to 4×10^3	5×10^3 to 6.6×10^4	9 to 58 ave. 22
Compacting	270 to 3.5×10^3	7×10^3 to 1.6×10^5	26 to 49 ave. 40
Incineration	120 to 1.5×10^3	8.6×10^3 to 5.5×10^4	20 to 72 ave. 50

SUMMARY GUIDE TO WASTE-MANAGEMENT COSTS

TYPE OR ELEMENT OF MANAGEMENT SOLID WASTES	PERCENTAGES OF TOTAL DIRECT COSTS				
	LABOR	UTILITIES, SERVICES	MAINTENANCE	OVERHEADS	DEPRECIATION
Total Management	7 to 32 ave. 20	9 to 18 ave. 12	2 to 13 ave. 6	9 to 34 ave. 25	12 to 60 ave. 28
Collecting, monitoring, and transport	13 to 56 ave. 31	8 to 37 ave. 22	4 to 11 ave. 7	17 to 53 ave. 31	3 to 37 ave. 20
Compacting	3 to 37 ave. 16	9 to 59 ave. 34	1 to 31 ave. 11	5 to 13 ave. 11	14 to 66 ave. 28
Incineration	10 to 28 ave. 17	2 to 8 ave. 5	7 to 19 ave. 10	19 to 28 ave. 24	21 to 61 ave. 36

FACTORS THAT INFLUENCE THE COST OF WASTE MANAGEMENT

1. Labor Cost – Factor of 50 Difference Between Installations.
2. Radioactivity Level – $> 10^{-1}$ Ci/m^3 were 50 Times Higher Than $10^{-7} - 10^{-4}$ Ci/m^3.
3. Capacity.
4. Decontamination Factor.
5. Nearness to Disposal Site.
6. Chemical and Physical Composition.
7. Power Cost.
8. Overheads.
9. Variation in Capital Costs
10. Maintenance Costs.



MANAGEMENT OF HIGH-ACTIVITY WASTES

BASIS FOR STUDY:

22,400 MWe INSTALLED CAPACITY

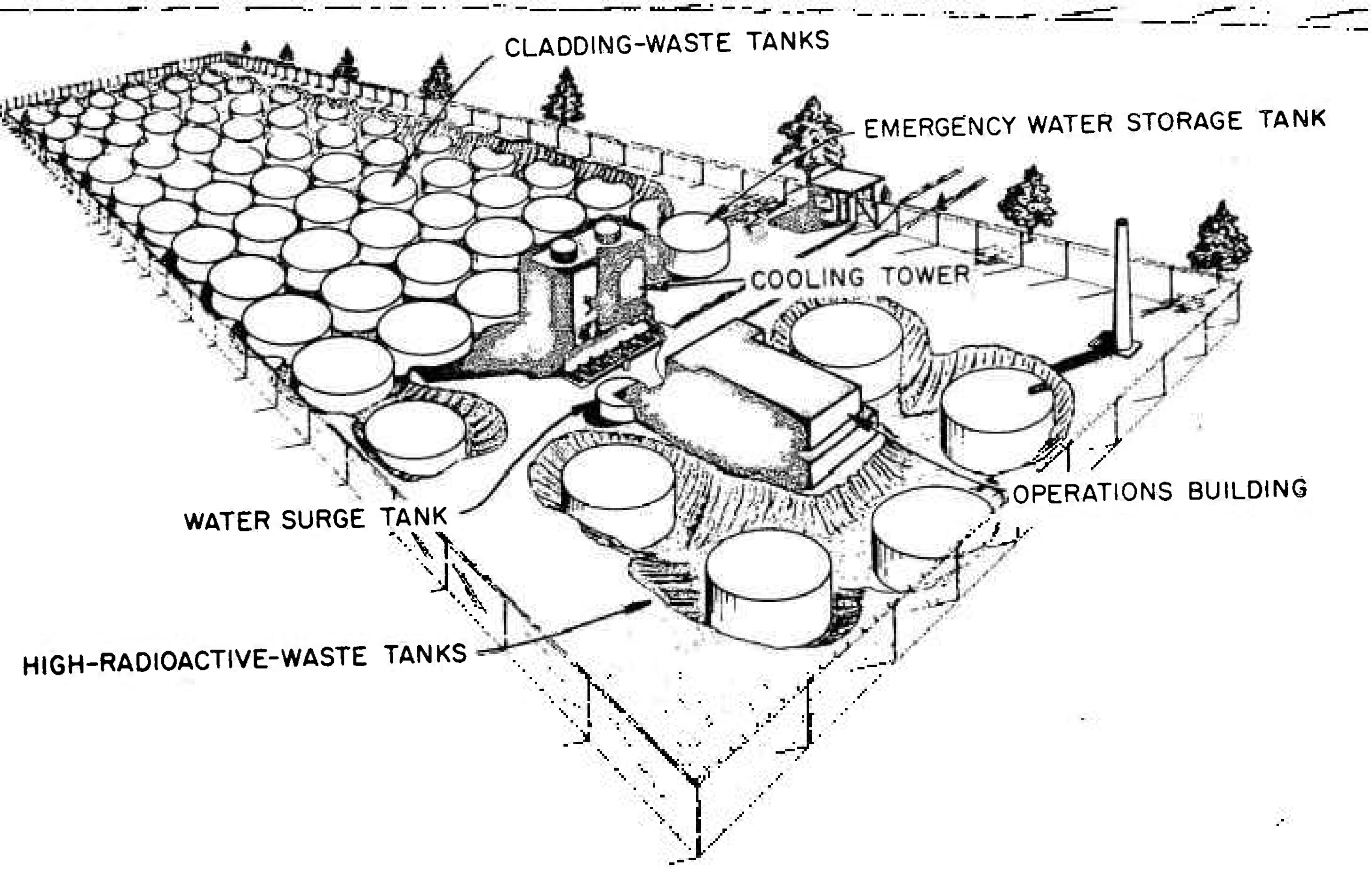
32% Thermal Efficiency
80% Reactor Load Factor

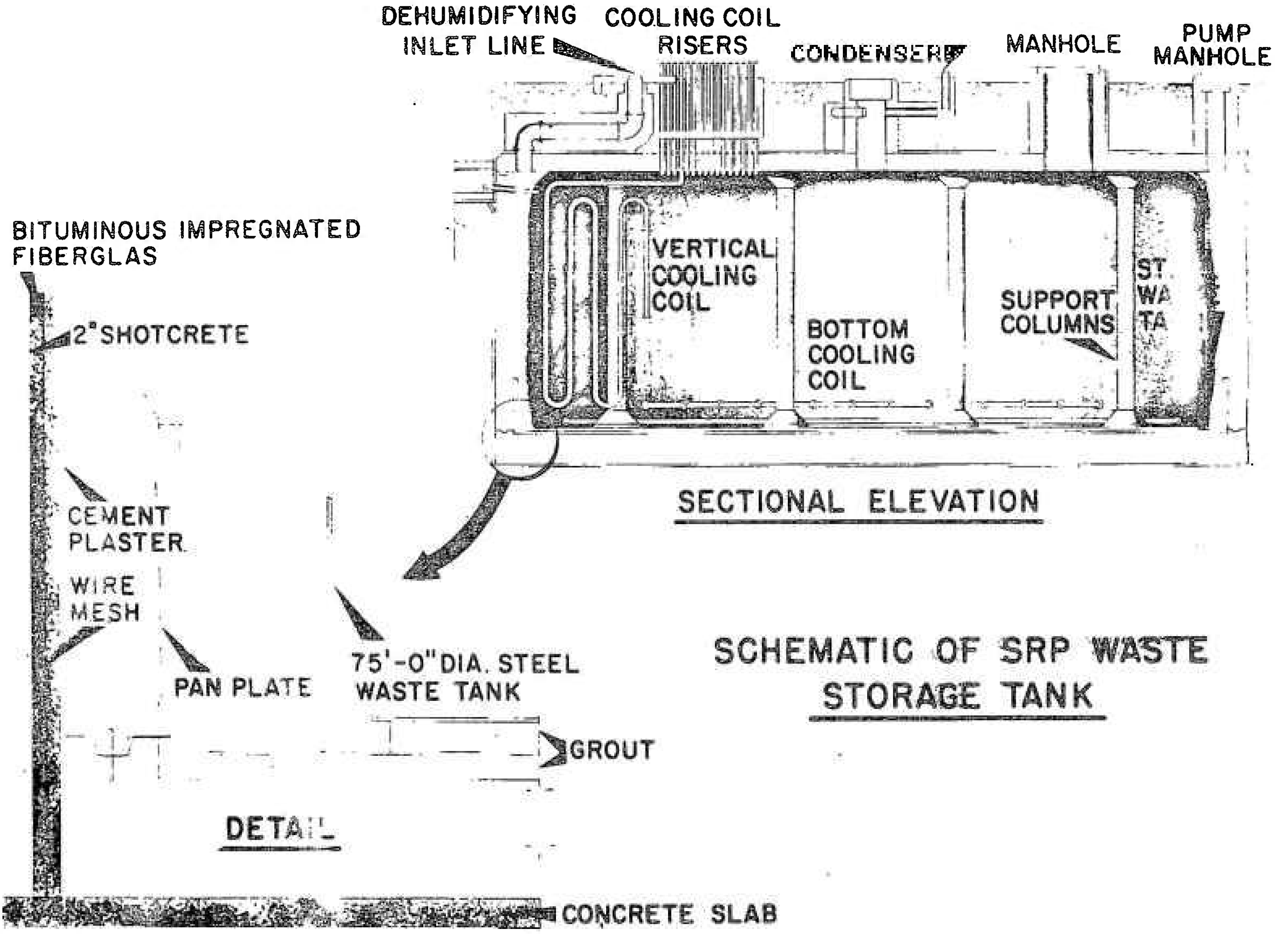
URANIUM CONVERTER FUEL

1500 MT/year
10,000 MWD/MT
100 gal/MT (acid)
600 gal/MT (alkaline)

THORIUM CONVERTER FUEL

270 MT/year
20,000 MWD/MT
200 gal/MT (acid)
1200 gal/MT (alkaline)

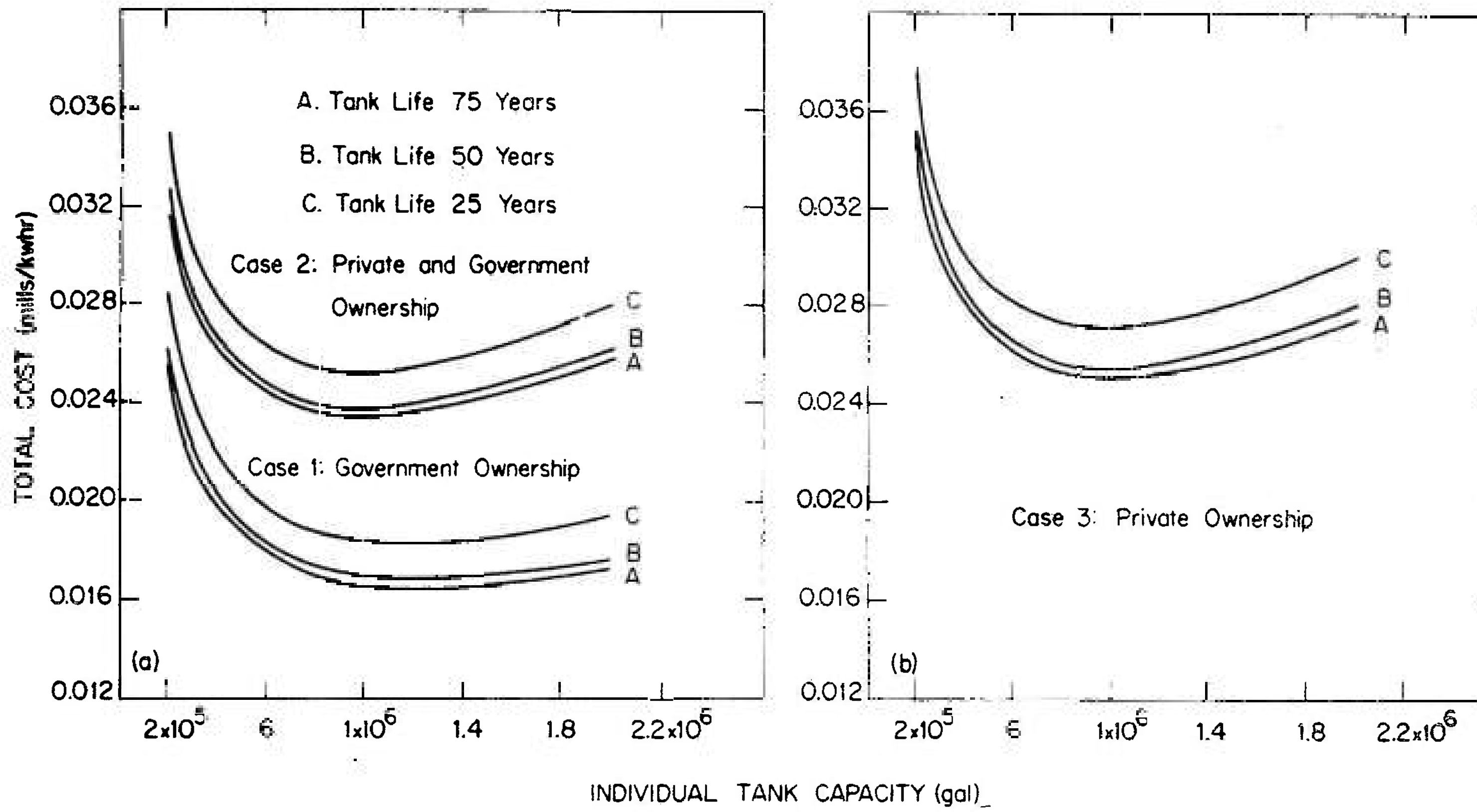

CLADDING-WASTE TANKS
EMERGENCY WATER STORAGE TANK
COOLING TOWER
OPERATIONS BUILDING
WATER SURGE TANK
HIGH-RADIOACTIVE-WASTE TANKS



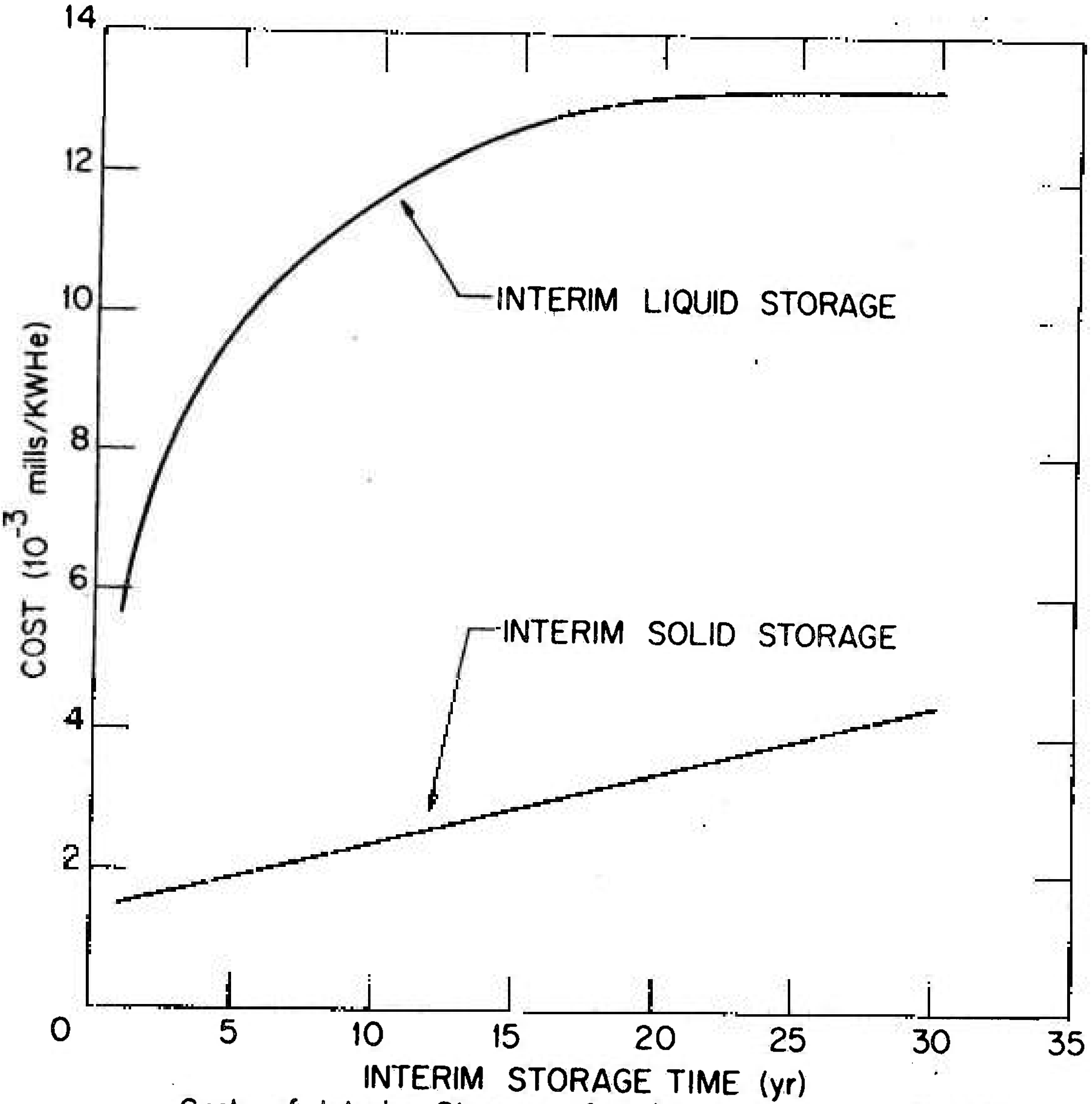
SCHEMATIC OF SRP WASTE STORAGE TANK

RANGE OF INCREMENTAL COSTS DURING 20-YEAR ACCUMULATION PERIOD

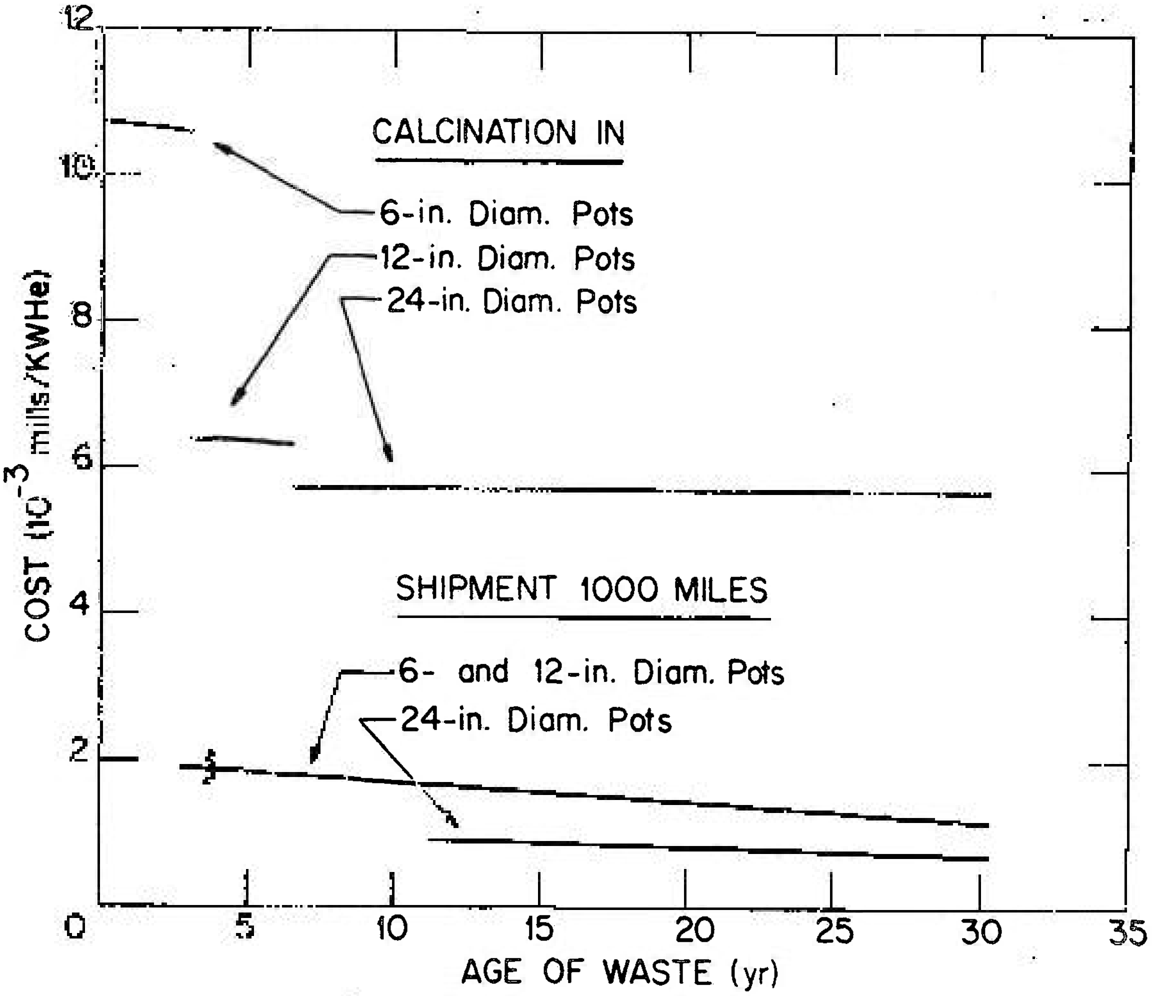
COST ITEMS	COST RANGE, $\$10^6$
CAPITAL	
INITIAL	8 - 16
TOTAL OVER 20 YEARS	22 - 61
ANNUAL OPERATING	
DURING FIRST YEAR	0.3 - 0.4
DURING TWENTIETH YEAR	0.5 - 2.0
PERMANENT FUND	
MAGNITUDE	22 - 72
ANNUAL DEPOSIT	0.7 - 2.6



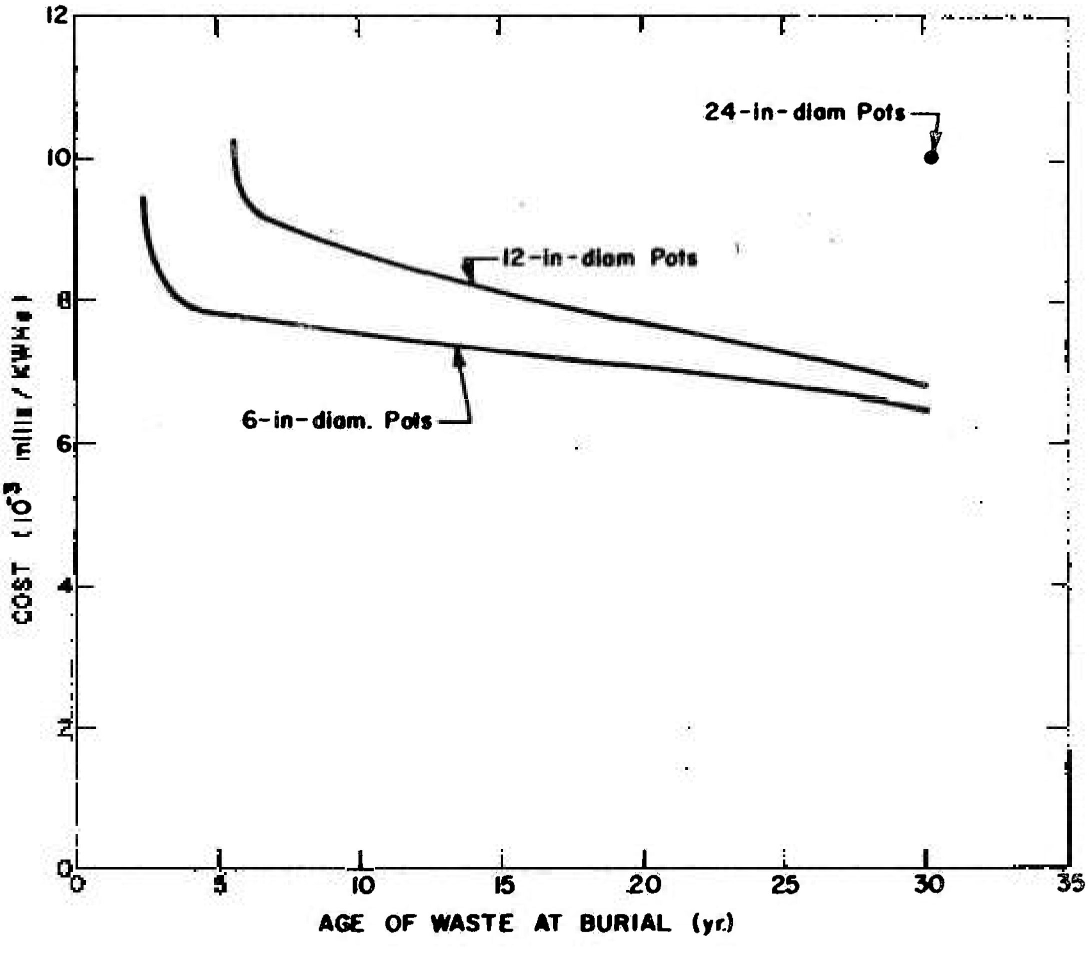
Cost of Acid Waste Storage as a Function of Tank Capacity, Tank Life and Method of Financing for (a) Case 1, Government Ownership; Case 2, Private and Government Ownership and (b) Case 3, Private Ownership.



Costs of Interim Storage of Acid Wastes as a Function of Storage Time.



Costs of Calcination and Shipment as a Function of Age of Waste.



Cost of Disposal in Salt as a Function of Age of Waste at Burial

MANAGEMENT COSTS FOR ACID PUREX AND THOREX WASTES[a]

MANAGEMENT OPERATION	INTERIM SOLID STORAGE TIME: 2 YEARS: 10^{-3} MILLS PER KWHR(E)	INTERIM SOLID STORAGE TIME: 2 YEARS: $/TON[b]	INTERIM SOLID STORAGE TIME: 30 YEARS: 10^{-3} MILLS PER KWHR(E)	INTERIM SOLID STORAGE TIME: 30 YEARS: $/TON[b]
INTERIM LIQUID STORAGE	0	0	0	0
POT CALCINATION IN 6-IN.-DIAM POTS AT 1/3 YR	10.7	1640	10.7	1640
INTERIM SOLID STORAGE	1.7	260	4.4	760
SHIPMENT (2000 MILES RT)	1.8	280	0.4	60
DISPOSAL IN SALT	8.7	1340	2.0	310
TOTAL	22.9	3520	17.5	2770

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[a] PRESENT-VALUED AT 4%.

[b] BASED ON A FUEL EXPOSURE OF 20,000 Mwd/MT.

MANAGEMENT COSTS FOR ACID PUREX AND THOREX WASTES[a]

MANAGEMENT OPERATION	INTERIM LIQUID STORAGE TIME: 5 2/3 YEARS, 10^{-3} MILLS PER KWHR(E)	5 2/3 YEARS, \$/TON[b]	30 YEARS, 10^{-3} MILLS PER KWHR(E)	30 YEARS, \$/TON[b]
INTERIM LIQUID STORAGE	10.3	1580	13.2	2030
POT CALCINATION IN 24-IN.-DIAM POTS	4.6	710	1.8	280
INTERIM SOLID STORAGE	3.0	460	0	0
SHIPMENT (2000 MILES RT) AT 30 YR	0.2	60	0.2	30
DISPOSAL IN SALT AT 30 YR	3.1	480	3.1	480
TOTAL	21.2	3290	18.3	2820

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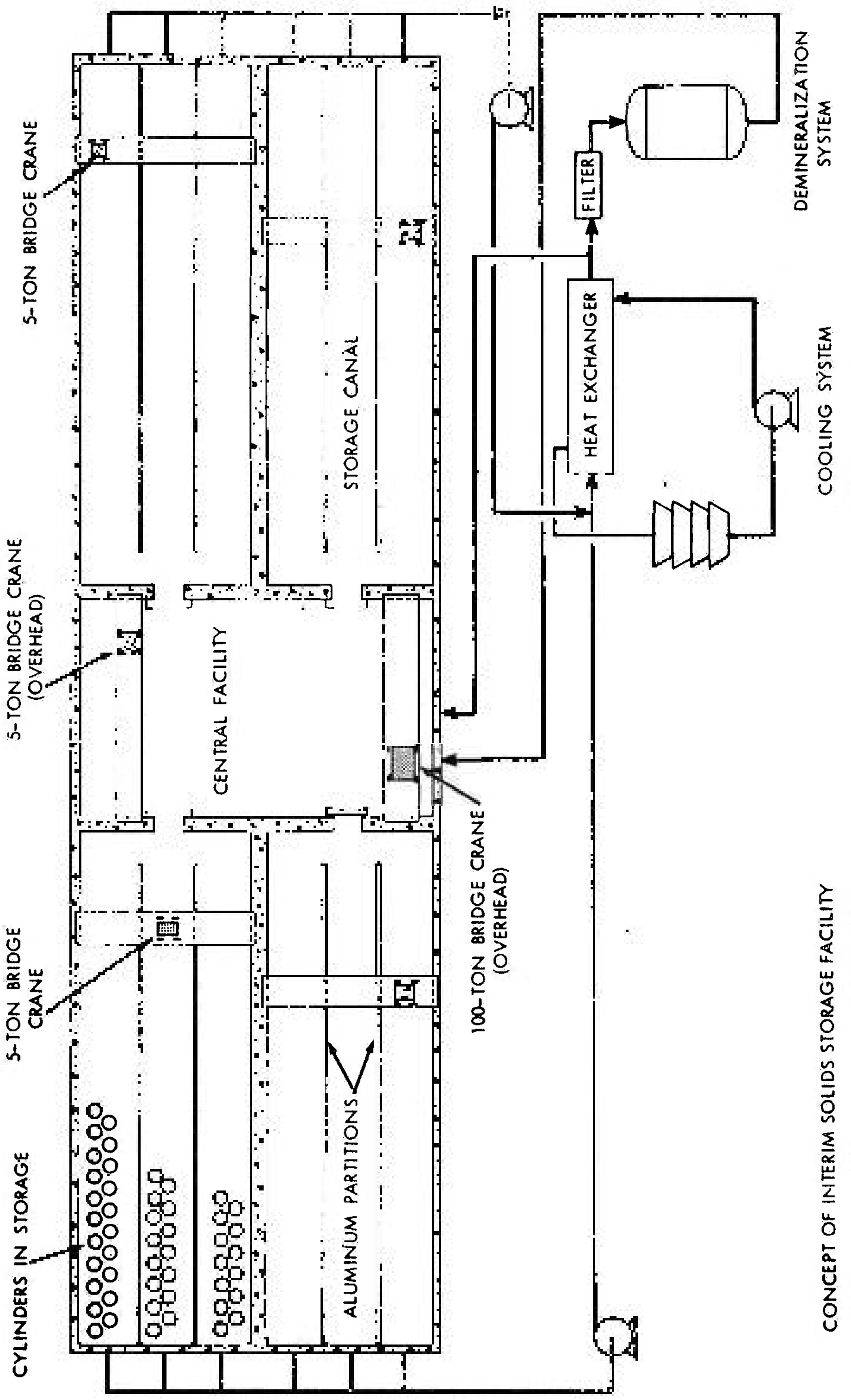
[a] PRESENT-VALUED AT 4%.

[b] BASED ON A FUEL EXPOSURE OF 20,000 Mwd/MT.



CONCEPT OF INTERIM SOLIDS STORAGE FACILITY