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WASTE PRODUCTION AND MANAGEMENT AT EBR-II

by

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ANL-West Division and EBR-II Project

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ABSTRACT

This report surveys waste production rates and disposal practices at EBR-II, a small-scale liquid-metal fast breeder reactor, for the past decade. Normal airborne wastes have been reduced by switching to low-sulfur fuel oil in auxiliary boilers and by converting to use of reactor steam for part of the plant space heating. Atmospheric releases of gaseous radioactive wastes initially increased during the implementation of programs for testing reactor fuels to and beyond cladding breach; but the effluent rates have now been reduced to far below permissible limits by use of a cryogenic distillation column.

EBR-II's small production of liquid radioactive waste is handled by evaporation, followed by disposal of the evaporator sludge as solid waste. Liquid-waste releases meet all applicable state and federal standards. Solid wastes are sent to two storage/disposal areas--one of which is reserved primarily for storage of materials containing sodium, pending development of a facility for disposal of the sodium.

In overall waste emission, EBR-II compares very favorably with commercial light-water reactors.

I. INTRODUCTION

This report is a comprehensive survey of waste production rates and disposal practices at Experimental Breeder Reactor II (EBR-II) in the past decade. The report covers both normal industrial and radioactive waste streams.

After EBR-II went into operation in 1964, its mission, and that of the adjoining Hot Fuel Examination Facility (HFEF), gradually changed to meet the changing developmental needs of the liquid-metal fast breeder reactor (LMFBR) program. These changes affected the types and volumes of waste produced at EBR-II, especially radioactive gaseous waste. Therefore this

report must sketch the EBR-II facility and programs in historical perspective to provide a clear understanding of how the waste streams have evolved.

The objective of this report is to provide a baseline of comparable data on wastes from a small, but fully operational, LMFBR power plant.[1] It is pertinent to note that EBR-II is more than just a neutron producer; approximately 20 MW of electricity is routinely generated and is distributed on the Idaho National Engineering Laboratory grid for routine consumption. With regard to possible commercial LMFBR plants, it is not clear whether such plants will have integrated, individual facilities for fuel examination or processing, as EBR-II has had. The historical role of the HFEF in relation to EBR-II is discussed, and some judgements are made as to whether certain waste streams generated by HFEF or jointly by EBR-II and HFEF more properly accrue to reactor or HFEF operations. However, the waste streams that are quantified and reported in this document are those judged to be primarily attributable to reactor operations. Thus, the data base is not intermixed with data more properly associated with a fuel-processing facility.

This report also attempts to show that an LMFBR does not produce an inordinate amount of radioactive waste by making a comparison with commercial light-water nuclear power plants. Published data from operating pressurized-water reactors and boiling-water reactors for the year 1976 are compared with EBR-II data.

II. DESCRIPTION AND HISTORY OF THE PLANT

A. Description and History of EBR-II

1. History of EBR-II

EBR-II is an experimental LMFBR located at the Argonne-West site of the Idaho National Engineering Laboratory (INEL). It consists of an unmoderated sodium-cooled reactor with a thermal power output of 62.5 MW, an intermediate closed loop of secondary sodium coolant, and a steam plant that produces 19.5 MW of electrical power through a conventional turbine generator.* EBR-II was designed by and is operated by Argonne National Laboratory under contracts with DOE and the predecessor agencies.

Closely associated with EBR-II is the HFEF, a two-building complex, the south part of which was originally the EBR-II Fuel Cycle Facility. The HFEF now serves to assemble "hot" experiments for irradiation in EBR-II, to examine experimental fuels and materials after irradiation, and to prepare used EBR-II driver fuel for shipment. Figure 1 is an aerial view of EBR-II, HFEF, and other related facilities.

*About 4.5 MW serves to power the Argonne-West site, and the remaining plant output of 15 MW of electricity is routinely fed to the INEL distribution system.

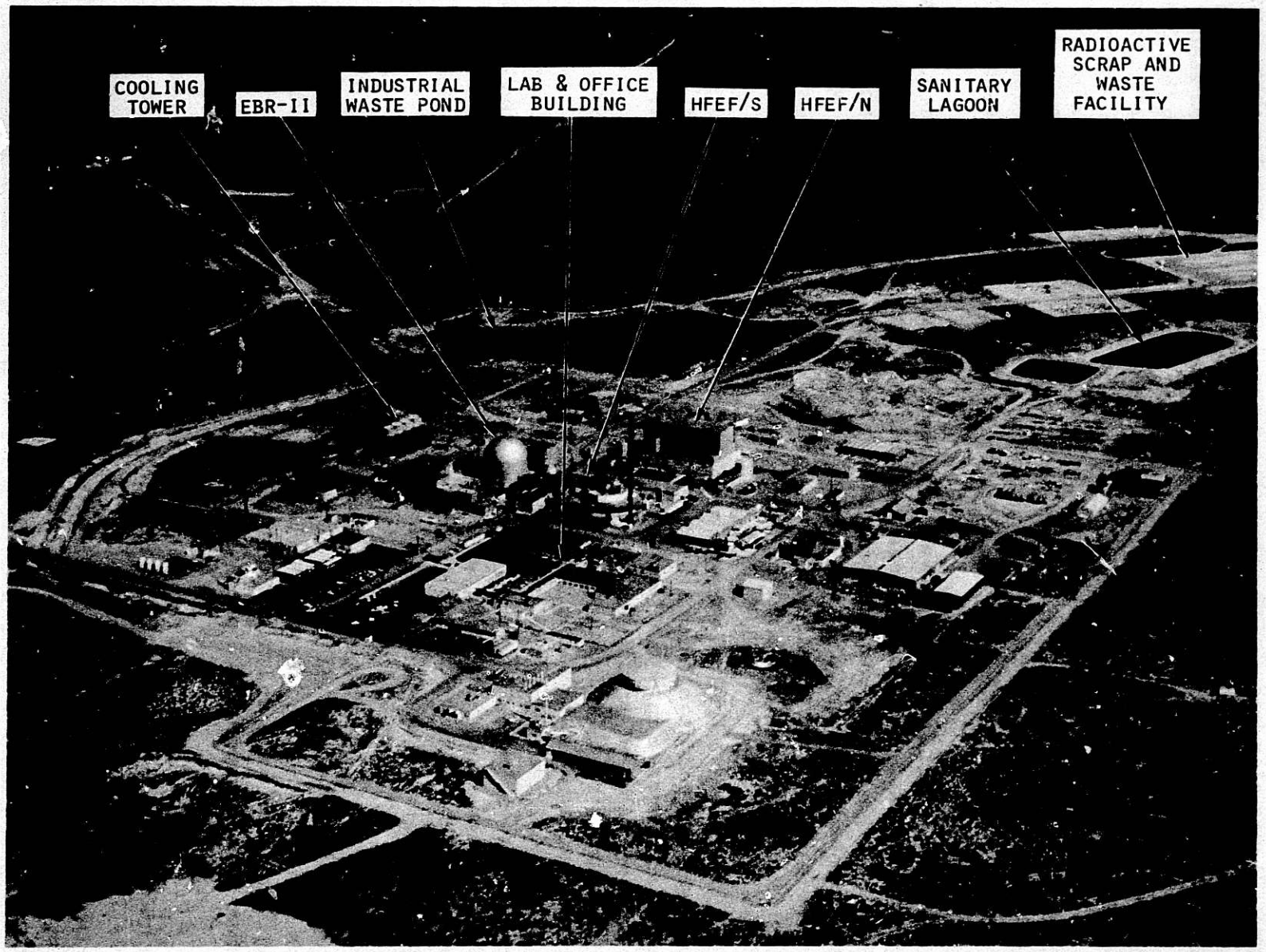

Fig. 1. Aerial View of Argonne–West Site

EBR-II was originally designed as an engineering facility to demonstrate the feasibility of fast reactors for central-station power-plant applications. It was also intended to prove that a breeding ratio greater than unity could be obtained in a power-producing reactor. The overall objective of the EBR-II demonstration was to prove the feasibility of a completely integrated plant in which fuel could be irradiated in the reactor, reprocessed in the Fuel Cycle Facility (FCF), and returned to the reactor without being removed from the site. The thermal performance of the reactor and the size of the system components were intended to be amenable to direct extrapolation to central-station application. The plant was designed to permit a maximum of experimental flexibility by separation of the plant systems, yet permit extrapolation to a commercial plant which might not require this same degree of separation.

EBR-II and the FCF were successfully operated as planned. However, experience during early operation indicated that EBR-II would also be useful as a high-temperature fast-neutron irradiation facility. To support development of commercial-scale plants, the purpose of the reactor was redirected in 1965 to provide irradiation services for the development of fuels and structural materials for the LMFBR program.

Because EBR-II was not designed as an irradiation facility, the transition required changes in several original concepts of operation. These included an increase in overall core size to compensate for reactivity losses to the system caused by additional irradiation experiments, changes in fuel enrichment and blanket composition for both economic and nuclear reasons, a shift of operating philosophy from that primarily of an engineering test facility to that of a high-priority neutron producer, and the design of a variety of irradiation test vehicles for use by experimenters. The redirection of the plant to the new high-priority role as an irradiation facility resulted in an improvement from early plant capacity factors of 25-40% to plant capacity factors in the last five years of 70-75%.

Table I is chronology of major events in EBR-II history. The evolution of EBR-II and HFEF can be separated into broad segments, within which radioactive-gaseous-waste release rates can be conveniently categorized. However solid and liquid wastes were not significantly affected by changes in the experimental programs.

For EBR-II, the first period was from the approach to power in 1964 until mid-1967, when the first breached experimental element was encountered. During this period, the plant was being fueled with recycled subassemblies from HFEF (then FCF), and the core and primary tank were free of fission products from breached fuel. Fission products did exist because of "tramp" uranium in the primary system, but these were at very low levels. The driver fuel was removed after about 1.2 at. % burnup. The reactor power during this period was restricted to 45 MW. At that time, the run-to-cladding-breach (RTCB) testing concept had not been implemented in EBR-II.

TABLE I. Chronology of Major EBR-II and HFEF Events

Year	Month	Description
1955	July	Original authorization of funds
1957	October	Start of construction
1961	September	Dry criticality achieved
1963	February	Filling of primary sodium system
1963	August	Argon cell of FCF filled with argon
1963	August	Filling of secondary sodium system
1963	November	Wet criticality achieved with a loading of 70 subassemblies
1964	July	Start of approach to power
1964	September	Start of irradiated fuel reprocessing in FCF
1965	March	Reactor power of 45 MW achieved
1965	May	First experimental subassembly placed in reactor
1965	May	Reactor operated with first recycled fuel from FCF
1967	January	Core enlarged to 91 subassemblies to accommodate experiments
1967	May	First breached experimental element encountered
1969	January	FCF remote fuel reprocessing shut down; hot cells dedicated to fuel examinations
1969	September	Reactor power raised to 62.5 MW
1970	November	One million MWh thermal accumulated
1971	March	First breached experiment identified by xenon tag
1974	October	Two million MWh thermal accumulated
1974	-	EBR-II converted to Mark-II driver core
1975	March	Start of hot (irradiated) fuel examinations in HFEF/North cells
1977	January	First experiment in run-beyond-cladding-breach program conducted
1977	June	Cover-gas cleanup system operational
1977	June	Three million MWh thermal accumulated
1978	April	Primary-coolant cesium trap installed
1978	October	One million MWh electrical generated
1978	December	3 645 000 MWh of operation accumulated

By the end of 1967, the second breached experimental element had been encountered, and the RTCB testing concept was being formulated. Designated experimental elements were ultimately allowed to run to breach (in 1971), but the release of fission gases within the plant, for experimental and safety reasons, was restricted to low levels; that is, when breaches were encountered, reactor shutdown was required. The xenon-tag method for locating leakers had not yet been conceived; thus the identification required multiple batch removals of subassemblies and reactor restarts to identify the source.

In the early 1970's, the xenon-tag method was applied to the leaker-location problem, and the first identification by this method occurred in 1971. As the method proved itself, it allowed the run-to-cladding-breach (RTCB) program to be extended and broadened. The RTCB program still continues, and it now involves tens of breached elements per year.

Understandably, the activity levels in the primary system, and in the gases vented from the primary system, increased during the second period (1968-1977). Also, during this period, the driver core was converted from the Mark-IA design, restricted to a maximum burnup of 3 at. %, to the Mark-II design, which is now qualified to 8 at. %. The fuel-qualification program also contributed fission-product activity to the primary system. The reactor was brought up to full thermal design power of 62.5 MW in 1969.

Although the RTCB program is still active, the implementation of the run-beyond-cladding-breach (RBCB) program, in early 1977, resulted in a clear end to the second period in early 1977. The reason for this demarcation is the application of on-line decontamination methods, especially the cover-gas cleanup system[2] and the primary-coolant cesium trap.[3]

In anticipation that breached elements in the RBCB program would release substantial quantities of fission products to the primary system, especially fission gases, a cover-gas cleanup system (CGCS) was installed. This system processes the primary cover gas to remove fission gases in a cryogenic column. The effectiveness of this system, plus substantial reductions in the leakage rate of primary cover gas to the building containment, has sharply reduced the release of fission gases to the environment via the site stack. This effect applies, of course, to both the RBCB and RTCB programs, which are being conducted simultaneously. More recently, in April 1978, a cesium-removal trap was installed in the primary-sodium purification loop. This trap has reduced the long-lived cesium activity in the sodium by about a factor of 20.

This programmatic evolution has had little impact on industrial-waste production rates, however. The industrial-waste data are, in general, presented within a single time frame. The evolution of radioactive-liquid-waste management has also proceeded on a schedule independent of the EBR-II program.

To summarize briefly, for this report the periods of interest are: (1) to 1968, with little breached fuel; (2) 1968-1977, with an active program up to cladding breach; and (3) 1977-present with an active run-beyond-breach program and on-line decontamination of coolant and cover gas. Figure 2 displays the history of radioactive cesium in the primary coolant. The effects of the RTCB program, the RBCB program, and the installation of the cesium trap are evident on this graph.

2. Functional Description of EBR-II

a. Primary System

Figure 3 is a schematic of the EBR-II primary, secondary, and steam systems.

The primary system is contained within the reactor building, which is a cylindrical gas-tight steel shell. EBR-II uses the "pot concept," where the core, reactor vessel, and primary coolant circuit, including pumps and intermediate heat exchanger, are submerged in a tank of bulk sodium. This concept minimizes worries about leakage of primary sodium and provides a large heat sink in case of pump failure.

Approximately 325 m^3 of sodium at a normal operating temperature of 370°C are contained in the primary tank. As the sodium is pumped through the core at about 0.57 m^3/s it is heated to ~505°C. The primary sodium then circulates through the shell side of the intermediate heat exchanger, where it transfers heat to the secondary sodium. The primary sodium, which is radioactive (primarily from ^{24}Na, ^{22}Na, and ^{137}Cs) is confined to the primary tank and never mixes with the nonradioactive secondary sodium.

b. Secondary Sodium System

The secondary sodium system is an intermediate closed loop between the primary system and the steam system. Approximately 50 m^3 of sodium are contained in the secondary sodium loop. The secondary sodium flows through the tube side of the intermediate heat exchanger and is heated to 465°C. It then flows through the shell side of the evaporators and superheaters of the steam generator, where it is cooled to about 305°C. The basic purpose of the secondary system is to separate the steam system from the radioactive primary system.

The secondary system is contained within a two-wing sodium boiler building except for a few pipes that extend across the intervening yard to the intermediate heat exchanger in the primary tank of the reactor building.

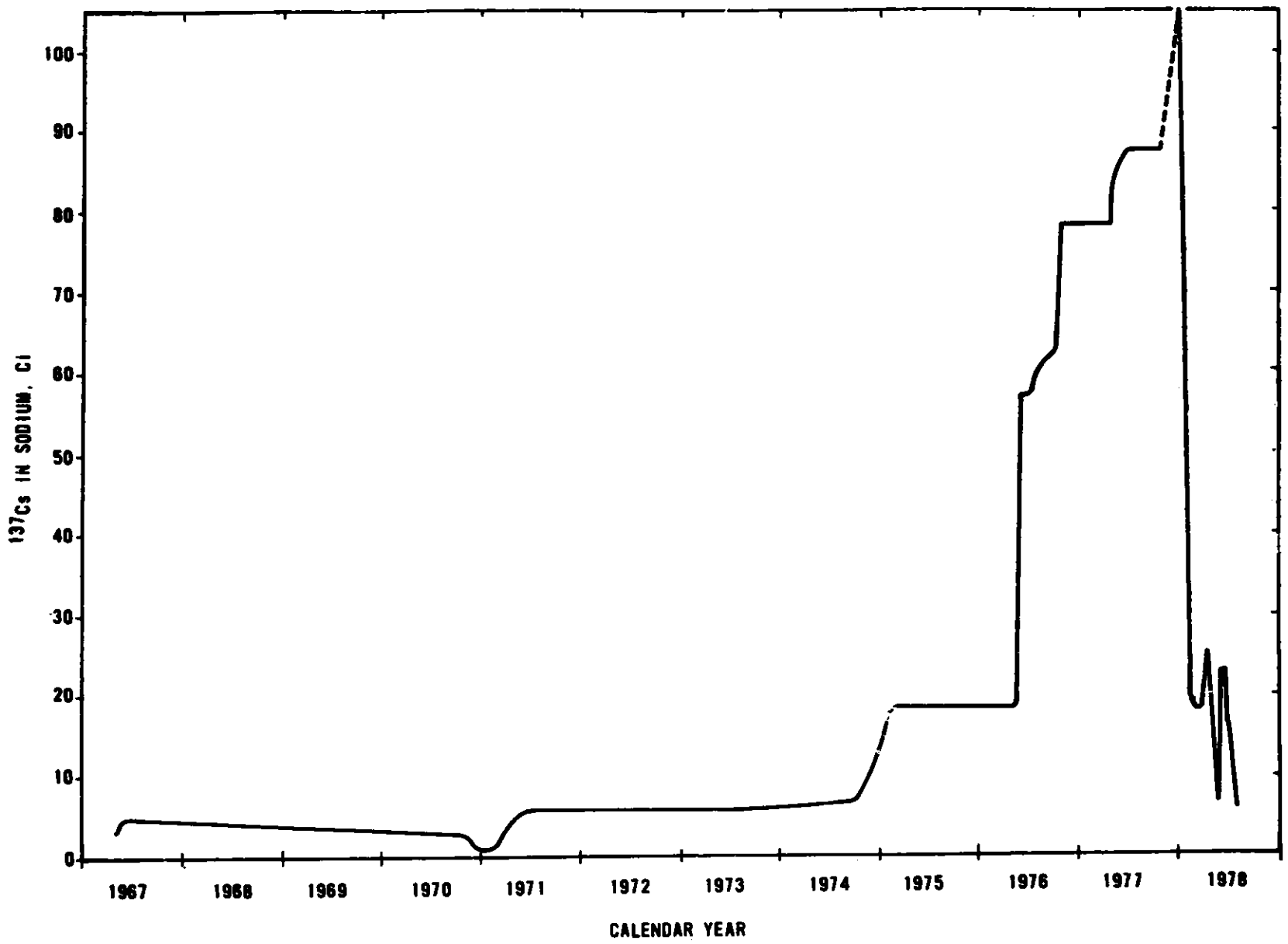

Fig. 2. Cesium in EBR-II Primary Sodium

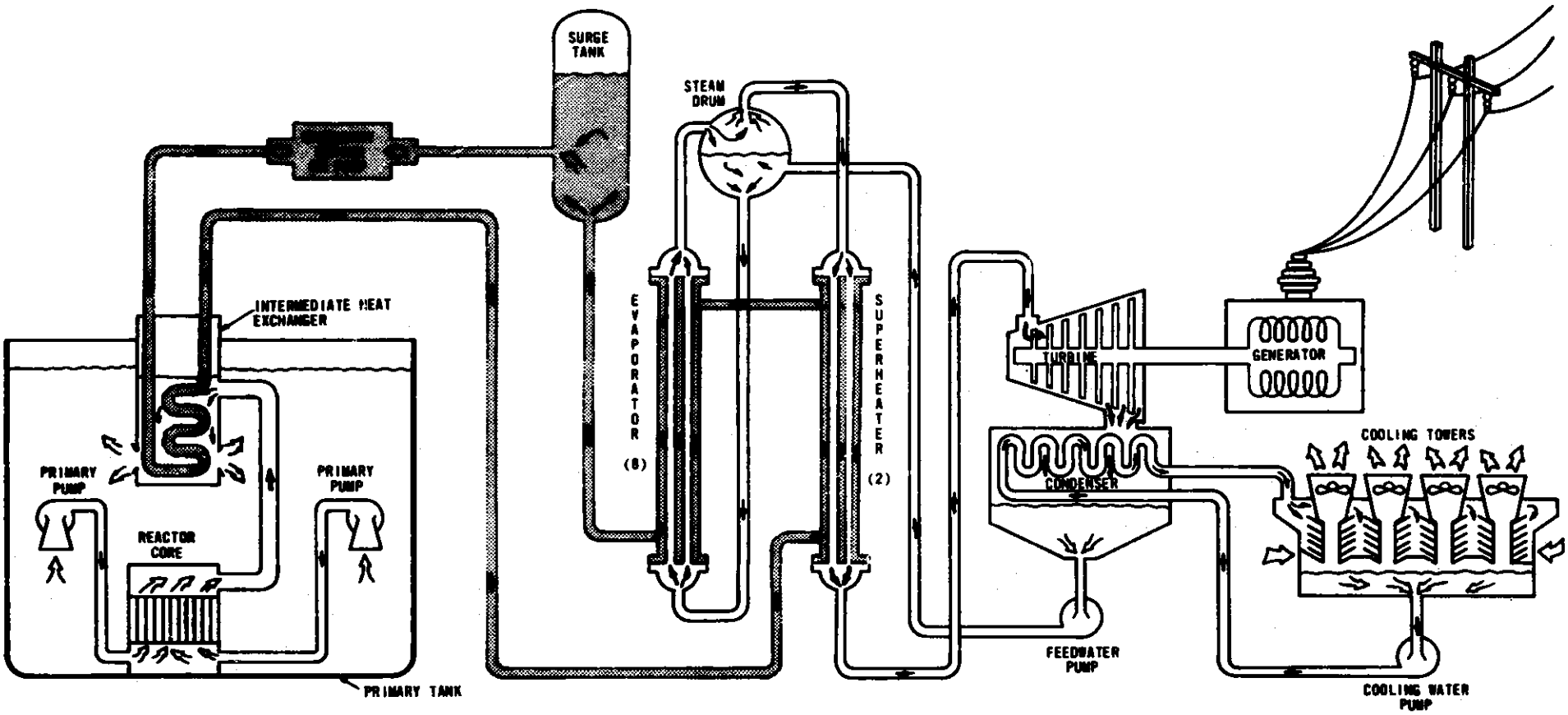

Fig. 3. Schematic of EBR-II Primary, Secondary, and Steam Systems

c. Steam System

The steam system consists of a steam generator, which absorbs heat from the secondary sodium system and converts feedwater to superheated steam; a turbine generator, which produces 19.5 MW of electric power; and cooling towers and other conventional support systems. The steam generator is contained in the boiler wing of the sodium boiler building; the main cooling tower is located west of the power plant; and an auxiliary cooling tower is mounted on the roof of the power plant.

Feedwater at a temperature of ~290°C flows through the tubes of eight parallel-connected evaporators and is heated to approximately 305°C by secondary sodium flowing in the shells. After the moisture has been removed from the resultant steam in the steam drum, the saturated steam flows through the tubes of two parallel-connected superheaters and is heated to ~440°C by the sodium. The superheated steam is then applied to the turbine generator to produce electricity, or if the generator is not operating, is bypassed to the condenser to be condensed to feedwater for recirculation through the steam system.

The steam is condensed by water circulating in the tubes of the condenser. The condenser cooling water passes through an air-cooled cooling tower.

The turbine generator consists of a standard commercial turbine of the impulse type, and a synchronous generator.

A blowdown system consisting of flash tanks, heat exchangers, and a demineralizer continuously extracts impurities from water in the steam drum. Also provided are water-treatment components to remove impurities and chemically treat the feedwater.

d. Cover-gas Cleanup System

The purpose of the cover-gas cleanup system is to continuously remove condensible gases from the EBR-II cover gas. The major components and flow paths are shown in Fig. 4. Basically, a flow of argon is extracted from the cover-gas space in the EBR-II primary tank and returned to the primary tank after treatment. The treatment removes (1) sodium vapor and aerosols, (2) xenon, krypton, and condensible impurities, and (3) any xenon tags released in-reactor. A compressor system in a building adjacent to the reactor building supplies the driving force for the flow of cover gas. When required, a bypass flow of this gas can be routed through cooled charcoal beds to remove and identify released xenon tags.

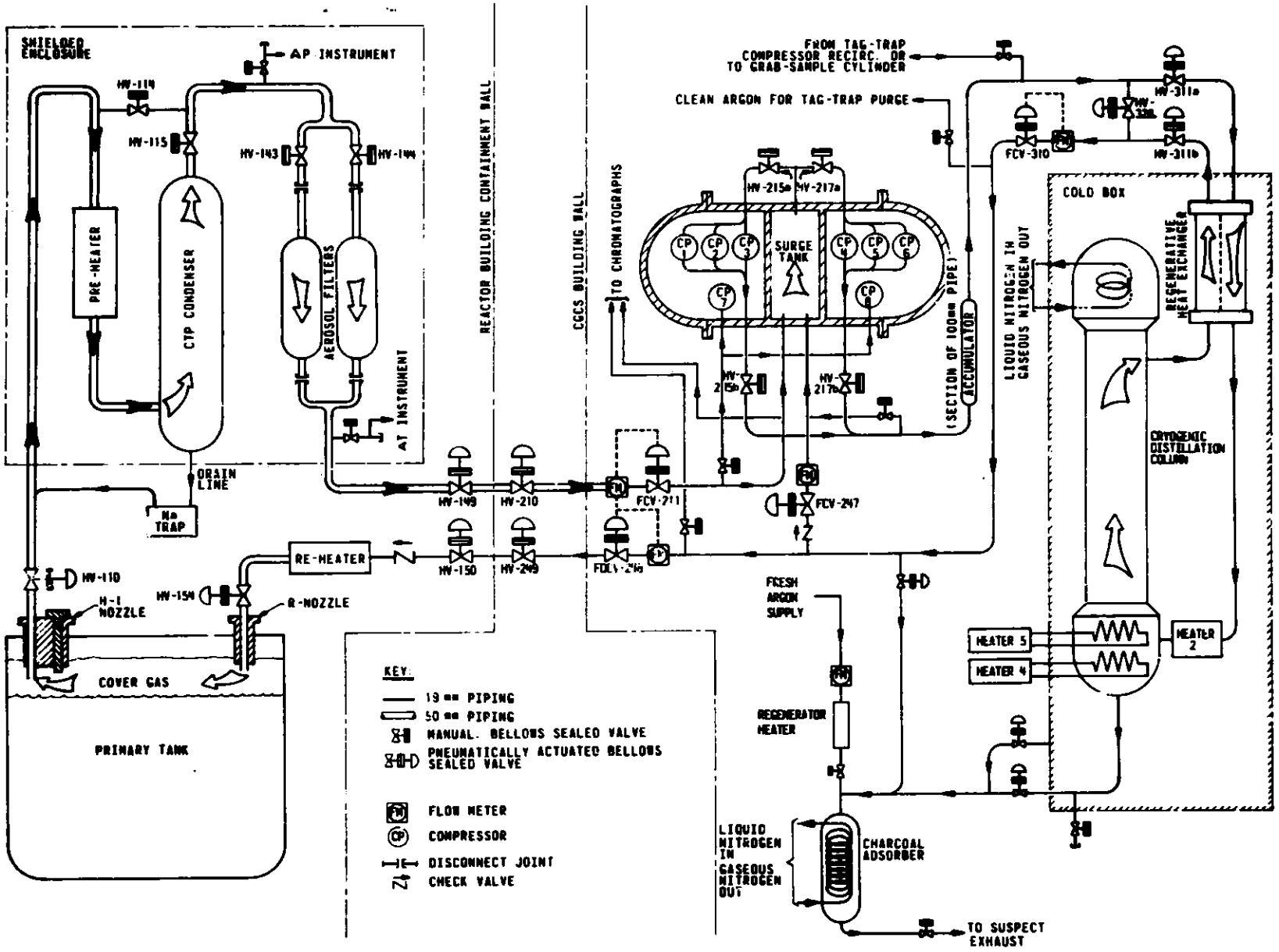

Fig. 4. Main Loop of Cover-gas Cleanup System

In the cleaning process, argon cover gas--containing xenon and krypton, sodium vapor and aerosol, sodium oxide particulate, and gaseous impurities--is removed from the primary tank through a nozzle. The gas is first passed through a preheater to vaporize entrained sodium, then through a condenser filled with Raschig rings, where the sodium is condensed for return to the primary tank. The argon, now at 170°C and containing less than 1 ppm of sodium, is passed through an aerosol filter to remove any remaining particulate.

The cooled sodium-free cover gas leaves the reactor containment building through shielded pipes and appropriate isolation valves and goes to an auxiliary process building. This building contains, among many other components, the compressor system. From the compressors, the gas is introduced into an insulation-filled containment vessel called the cold box. In the cold box, the gas passes through a regenerative heat exchanger enroute to the bottom sump of a cryogenic distillation column, where it is bubbled through liquid argon at -182°C and condensed. Liquid entering the distillation-column overflow is vaporized by heaters, and the gas passes up through the reflux portion of the column to a nitrogen-cooled condenser, where further condensation occurs. Condensate draining back to the bottom of the column provides the necessary reflux to strip xenon and krypton from the gaseous argon.

Cleaned gaseous argon flows from the distillation column through the heat exchanger to be warmed and routed to the EBR-II primary tank. A reheater raises the argon temperature to approximately 315°C before the argon enters the primary tank.

During routine operation, whenever the oxygen or methane content of the distillation-column sump exceeds 15 or 2.6 g respectively, the liquified gases in the sump are discharged to a charcoal adsorber. Operating experience shows that the xenons are removed and decay on the charcoal. Some ^{85}Kr that was previously adsorbed is released and exhausted out the 60-m-high exhaust stack. The reason for sump transfer when oxygen or methane reaches these levels is to prevent the conditions for a potentially explosive ozone reaction. Such sump transfers have been carried out at intervals of once or twice a year.

Gas chromatographs and special oxygen analyzers continuously analyze samples of the CGCS inlet gas, the compressor-discharge gas, and the returning clean argon for H_2, He, O_2, and N_2. A hydrocarbon analyzer is used to monitor methane accumulation.

B. Hot Fuel Examination Facility (HFEF)

The HFEF now consists of a north and a south building, designated HFEF/N and HFEF/S. The older of the two, HFEF/S, was built with EBR-II

and was originally designated the Fuel Cycle Facility (FCF). The FCF was intended as a process plant in which irradiated fuel elements and blanket material removed from EBR-II would be disassembled and processed by pyrochemical methods to remove fission products. The fuel was then reconstituted, refabricated into new fuel elements, and assembled into new subassemblies. It was designed as a unique facility to employ processes and procedures on a limited production scale that had previously been performed only in a laboratory. The process was successfully demonstrated during the period 1964 to 1968. The production of radioactive waste at the EBR-II site in this period was dominated by the FCF operations. The release of radioactive waste by the reactor prior to 1968 was insignificant; for this reason, this report considers only the period from 1968 to the present for EBR-II operations.

HFEF/S consists primarily of an argon-atmosphere cell (where the fuel processing used to be performed); an adjacent air-atmosphere cell, where reactor subassemblies are assembled and disassembled; and an operating area for personnel surrounding the two cells. All in-cell operations are done remotely, behind thick shielding walls and windows.

When a subassembly is transferred from the reactor tank to the fuel transfer systems after irradiation, it normally retains 5-50 g of primary sodium. Since at HFEF/S the subassembly initially enters an air-atmosphere cell, the sodium must be removed from the subassembly. This sodium-wash operation has always been performed at an HFEF/S wash station. However, this operation would be required prior to off-site fuel shipment whether HFEF/S performed the service or not; therefore, the waste generated by this step more properly should be categorized as EBR-II reactor, rather than HFEF/S, waste.

Following cessation of hot processing operations at HFEF/S in early 1969, the facility was redirected to a full-time mission of interim and final examinations of experimental and driver-fuel elements and structural materials. The EBR-II driver fuel was subsequently fabricated from unirradiated feed stocks, both at commercial vendors and at an in-house fabrication facility.

To supplement the HFEF/S capabilities for examinations, a new large facility, HFEF/N, was put in operation in 1975. HFEF/N has 60 m^2 of floor space in an air-atmosphere decontamination cell and 200 m^2 in an argon-atmosphere main cell. Special features include design for particle-tight containment of plutonium contamination and the ability for in-cell vertical handling of 9-m-long loop experiments. Much of this capability is used to support programs other than EBR-II operations. Therefore, HFEF/N waste is not considered in this report except where the waste can be specifically attributed to EBR-II.

C. Laboratory and Office Building

The Laboratory and Office (L&O) building contains analytical facilities and serves as the headquarters for the ANL-West site. The building is of conventional construction with the exception of six analytical caves located in the north wing.

The majority of the building space (except the north wing) is used for offices and other supporting facilities. The north wing, in addition to the analytical caves, contains the analytical chemistry laboratories, a nondestructive testing laboratory, and the equipment for treating the site's radioactive liquid waste.

D. EBR-II Site Description

1. Geology

EBR-II is located at the Argonne-West area of the INEL (see Fig. 5). This southeastern Idaho site is situated on a high desert plain at an average elevation of 1485 m and is surrounded by mountain ranges. The surface of much of the plain is covered by water-borne and wind-borne topsoil with an underlaying gravel bed 0.3 to 15 m deep. Lava rock beneath the gravel layer extends to considerable depths, ranging at least to the water table.

The INEL is located over the Snake River Plain aquifer. The estimated lateral water flow is not less than 14 m^3/s. Mountain streams originating in the north and disappearing into the porous soil of the area are sources of water for this aquifer. Because of the high porosity of the surface gravel, surface drainage is minimal and drainage flow is toward the northeast, opposite to the direction of the main subsurface flow.

2. Meteorology

Annual precipitation at the INEL amounts to about 18 cm; roughly half of this precipitation occurs as snow during the winter months. Rainfall during the warmer months usually occurs in the form of highly localized instability showers.

Measurements of wind speed and direction have been recorded for a 25-year period. The prevailing winds under inversion conditions (which afford the most serious hazard) are not directed toward the nearest heavily populated area. Under inversion conditions it is unlikely that contamination would travel more than 50 km from the site.

The remote INEL location was chosen for the EBR-II facility because of the limited information available at that time pertaining to breeder-reactor operation with plutonium fuel. Although located in a remote site, the plant was constructed as though it were sited in a populated area.

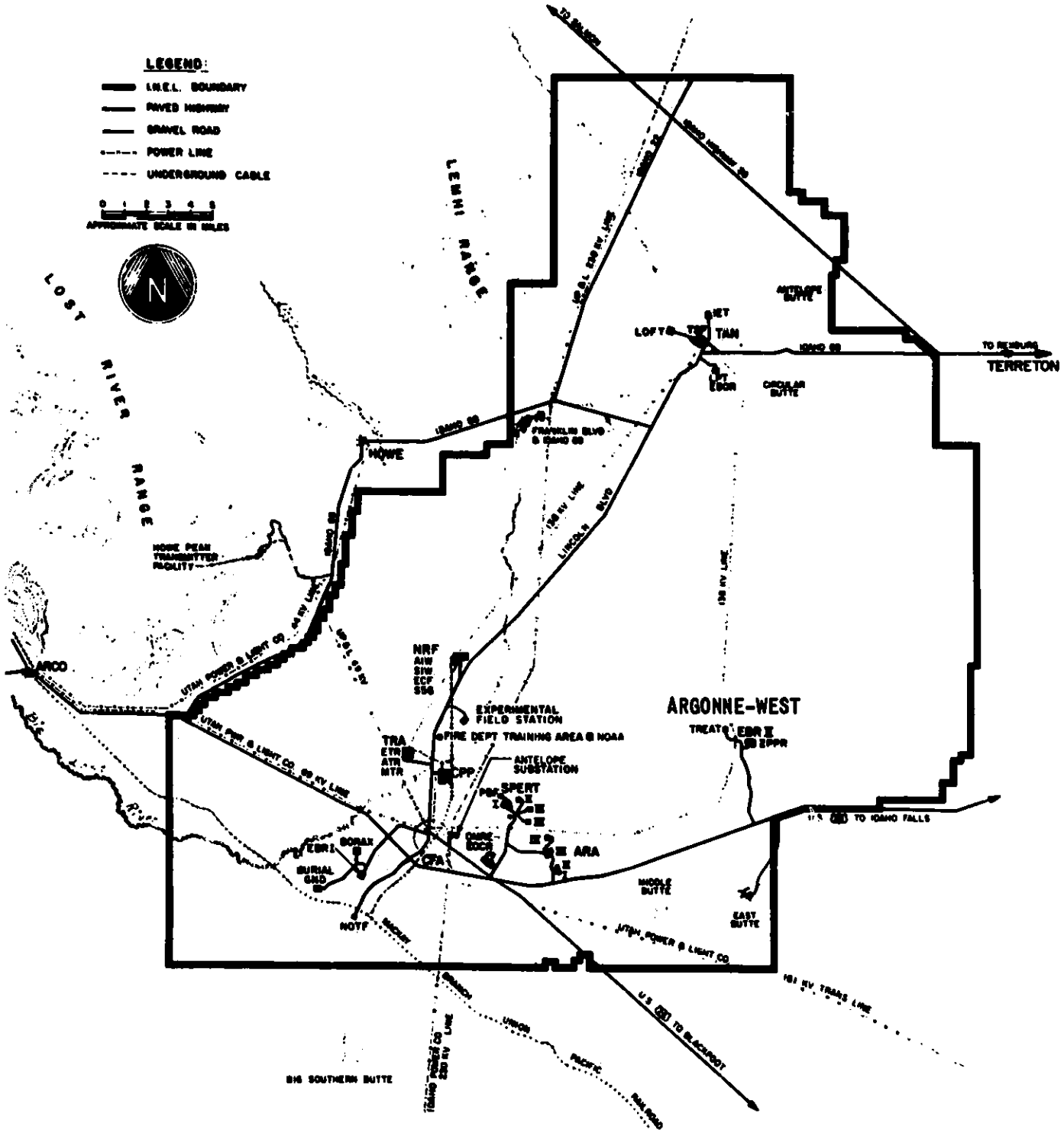

Fig. 5. Idaho National Engineering Laboratory

III. INDUSTRIAL WASTE MANAGEMENT AT EBR-II

Industrial waste, for the purposes of this report, is considered to be all the waste that is not radioactive. Although in actual practice the sanitary system is completely separate from the liquid industrial waste system, it is included in this section for simplification.

A. Solid Industrial Waste

Solid industrial waste consists of waste paper, rags, wood, and metal products associated with administrative office work and plant maintenance operations. Argonne-West (ANL-W) generates approximately 1750 m^3 of nonradioactive solid waste per year. There is no practical way to determine how much of this is directly attributable to EBR-II operations. This material consists primarily of trash (1450 m^3) and cafeteria garbage (300 m^3). The trash and garbage are collected in dumpsters and trucked to the INEL sanitary landfill operated by the INEL service contractor (EG&G Idaho).

In addition, scrap metal and scrap wood are separated and sold for salvage at a rate of approximately 45 metric tons per year and 425 m^3 per year, respectively. In the past, an attempt was also made to recover and recycle office paper. This program donated approximately 40 metric tons per year to the Southeastern Idaho Community Action Agency (SICAA) to be made into cellulose insulation for Project Winterization. This program was discontinued in 1978 when SICAA no longer expressed an interest in the paper. To date, no other outlet has been found.

B. Liquid Industrial Waste

Liquid industrial waste is generated in EBR-II by the operation of air compressors, pumping systems, auxiliary boilers, reactor-plant auxiliaries, water-chemistry laboratory, air-conditioning equipment, and cooling towers. Additional industrial waste that can, in part, be attributed to EBR-II results from the operation of the L&O building, which includes a photo laboratory. Sanitary waste is also produced at EBR-II and its support facilities and is processed separately from the industrial waste. The sanitary system also receives waste from the cafeteria.

The primary source of industrial liquid waste is the blowdown from the cooling towers. The EBR-II main cooling tower contributes about 6.4×10^7 L per year and the auxiliary cooling tower about 1.9×10^7 L per year. An additional 59 000 L is produced by the regeneration of the ion exchangers used for purifying the steam-system makeup water. The cooling towers and regeneration process are the main sources of the chemicals released: about 17 000 kg Na^+; 67 000 kg SO_4^{-2}; 400 kg Cr^{+3}; 125 kg Zn^{+2} annually. Turbine condensate contributes an additional 7.2×10^6 L per year, once-through air-conditioning and cooling water contributes about 15.5×10^6 L, and other

sources contribute about 3.8×10^6 L per year to the industrial waste stream. These wastes are discharged via piped systems to three open ditches, which drain to the 0.6-ha industrial waste pond where the wastes are disposed of by leaching and solar evaporation.

The water in the industrial waste pond is sampled monthly during ice-free periods, normally May-October. Analyses show that the water consistently meets the State of Idaho standards for stock and wildlife watering for the chemical constituents released.

The sanitary-liquid-waste system collects the sanitary waste from all the facilities of ANL-West. The sanitary waste is pumped to three lined sanitary lagoons where aerobic action (anaerobic in the winter when the ice forms) and solar evaporation dispose of the waste. The sanitary waste system disposes of about 30×10^6 L of water per year. An in-line monitor ensures that no radioactivity is released to the sewage lagoons.

C. Airborne Industrial Waste

The only significant source of gaseous industrial waste is the operation of auxiliary boilers. These boilers produce all the steam for site uses when the reactor is shut down and part of the steam during reactor operation. A secondary source of airborne waste is cooling-tower drift.

The boiler plant presently burns low-sulfur No. 2 fuel oil. Prior to 1972, No. 5 fuel oil was used. Modifications made over the past several years, which result in using reactor steam for space heating whenever possible, have reduced the amount of fuel oil used from 3.64×10^6 L in 1973 to the present level of 1.13×10^6 L in 1978. Figure 6 shows the SO_2 and particulate releases for the past 10 years. Although some of the modifications were made for energy conservation, the reduction in nonradioactive releases demonstrates the compatibility of energy conservation with reduction in environmental insults, as well as the advantages of nuclear-generated power.

Cooling-tower drift was significantly reduced by modifications made to the cooling tower in 1974. Drift now results in about 14.5 kg of hexavalent chromate released each year (Fig. 7).

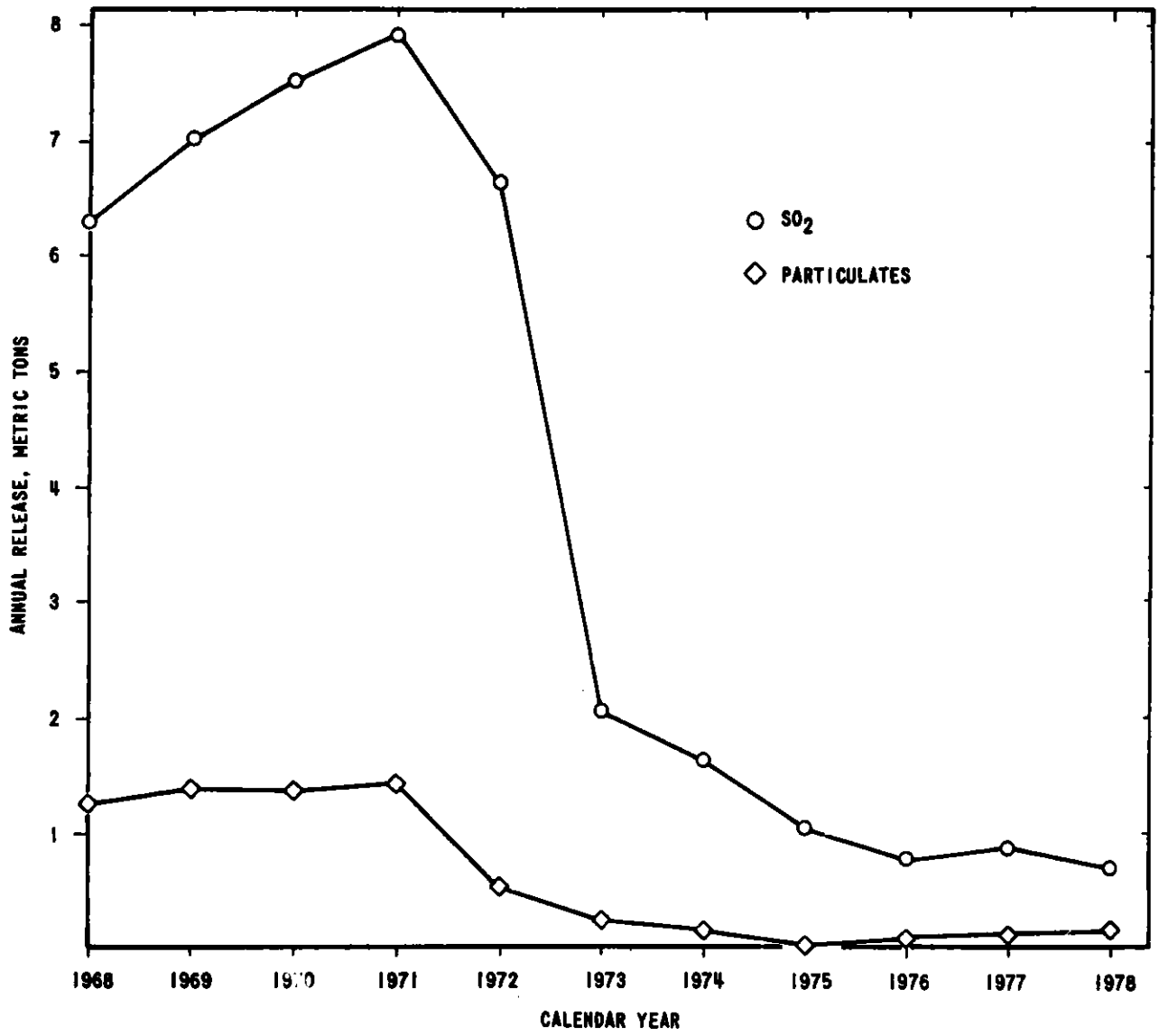

Fig. 6. Release of SO_2 and Particulates from Auxiliary Boilers

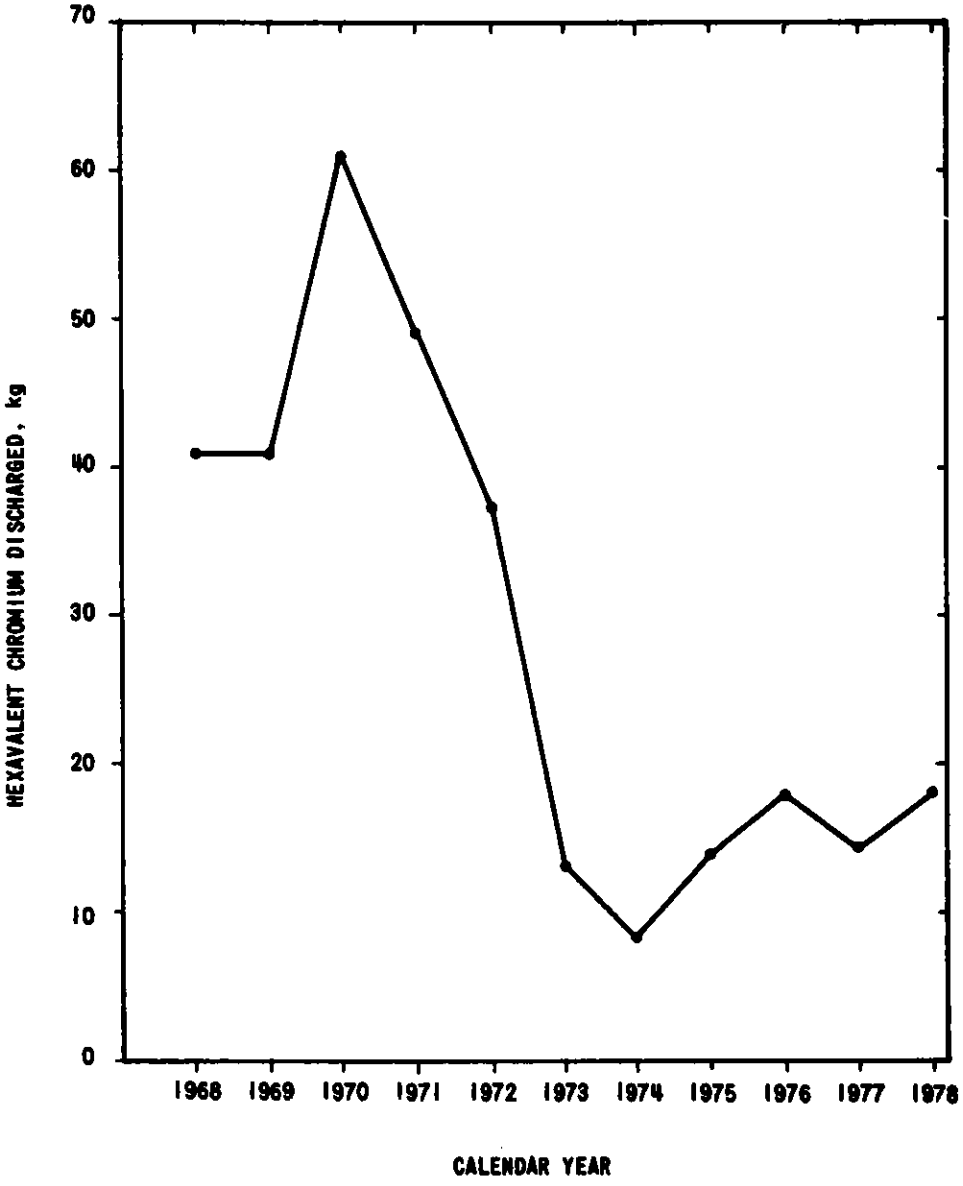

Fig. 7. Release of Airborne Chromate from Cooling Tower

IV. RADIOACTIVE WASTE MANAGEMENT AT EBR-II

A. Solid Radioactive Waste

1. Sources of Solid Waste

At EBR-II, a minor quantity of solid waste is produced. A major portion (by volume but not by radioactivity) of this solid radioactive waste is the accumulation of wipe rags, plastic containers, shoe covers, and other industrial solids associated with working with radioactive materials during maintenance. Reactor components such as thermocouples, nuts and bolts, and other hardware are disposed of as solid radioactive waste. Radiation from these components is generally low-level (less than 10 mR/h).

Another source of solid waste is elemental sodium in bulk form, which is produced in small quantities during maintenance. This sodium is collected in 3.8-L cans, which are placed in metal drums and covered with sand. Because of low radiation levels, these drums are stored in a covered trailer until the sodium can be treated in the near future.

Approximately 500 L of elemental sodium are present in cold traps that are used in the reactor coolant system to control oxygen content. These traps are removed from the system when their efficiency has dropped and are stored in a radioactive storage building if the levels are low, for example 100 mR/h. When the radioactivity is high, they are stored with shielding in a metal container underground in an interim storage facility at ANL-West designated as the Radioactive Scrap and Waste Facility (RSWF). The storage of these cold traps is temporary until a treatment process has been developed.

Most of the solid radioactive waste is generated at HFEF. This waste results from the disassembly, assembly, and inspection of EBR-II subassemblies, discarding of used experimental hardware, preparation of reactor blanket subassemblies for storage, and inspection of LMFBR-related tests that are performed by other organizations. The solid wastes are of the following categories: low-level nontransuranic, intermediate-level nontransuranic (up to 10 000 R/h), low-level transuranic, intermediate-level transuranic (up to 1000 R/h), low- and intermediate-level transuranic and nontransuranic waste with sodium, and bulk sodium in the intermediate level.

The chemistry laboratories that support EBR-II produce solid waste from maintenance and operations. These wastes comprise low and intermediate levels in similar categories to HFEF, but the volume is low and there is no elemental sodium.

Solid waste is also produced when the evaporator bottoms are solidified as discussed in Subsection B below.

2. Handling of Solid Waste

Radioactive solid waste is handled by several different methods depending upon the radiation levels and/or the content of the waste. Low-level nontransuranic compactible waste is collected in plastic bags and hand-carried to and placed in specific strategically located dumpsters. The dumpsters are then trucked to the INEL Radioactive Waste Management Complex (RWMC) operated by EG&G Idaho, where the waste is compacted prior to permanent disposal in the trenches. EG&G Idaho has reported that compaction results in a volume reduction of about 10:1.

Prior to December 1977, low-level nontransuranic noncompactible waste was placed in cardboard boxes; the boxes were placed in dumpsters and sent to the RWMC. Bulky or heavy items were boxed in wooden boxes. These boxes were then sent to the RWMC for burial. Since December 1977, this low-level waste has been collected in 208-L steel drums. The drums are held in a covered van until the van is full and then sent to the RWMC. Items too large for the drums are placed in standard-size plastic-lined wooden boxes painted with fire-retardant paint. The boxes are available in sizes up to 1.2 x 1.2 x 2.4 m.

Low-level transuranic waste is placed in Department of Transportation (DOT) specification-17C drums equipped with a 2.3-mm rigid poly liner or a fiberglassed DOT specification-19A box. The filled containers are sent to the RWMC for storage. However, no transuranic waste has yet been generated as a result of EBR-II operations.

Low-level wastes containing elemental sodium are not acceptable to the RWMC or any other disposal site because of the sodium. This material is being stored until a new facility can react the sodium. The sodium will be treated with water and water/alcohol, with all air discharges HEPA-filtered. When the sodium has been removed, the metal containers, pipes, etc., will be packaged and shipped to the RWMC. The water will be evaporated in a central liquid-processing facility, and the alcohol will be distilled and recovered in the sodium cleanup facility.

Intermediate-level waste with radiation levels up to 10 000 R/h is generated within the HFEF as a result of the disassembly of EBR-II driver subassemblies prior to inspection or shipment of fuel to the reprocessing plant. Intermediate waste from the HFEF is remotely packaged in 1.8-m-long 295-mm-ID carbon steel cans. These cans are then inserted into a stainless steel outer can and either seal-welded or gasketed closed. The cans that do not contain elemental sodium are sent to the RWMC for either interim storage if transuranic or permanent disposal if nontransuranic. Cans containing elemental sodium are sent to the ANL-West RSWF for storage. No intermediate-level transuranic waste is generated by EBR-II.

The RSWF, used for storage of solid wastes and scrap only, is a controlled-access, fenced 1.6-ha area located about 0.8 km north of EBR-II. The area was selected for drainage reasons, and was built up by banking the earth to a level about 1 m above the original land surface to eliminate chances of flooding by surface runoff. The storage site utilizes storage holes with liners of 6-mm-thick carbon steel. Most liners are 3.7 m long by 406 mm in diameter, although a few other sizes are also used. The liners are welded closed at the bottom end and have a top closure plate with integral concrete shield, which is welded on after the material has been deposited.

Only solid material is stored at the RSWF. The arrangement of storage holes provides for 27 rows on 3.7-m centers, with approximately 40 holes per row on 1.8-m centers. Prior to 1977, all high-gamma waste packed in 1.8-m-long waste cans was stored in the RSWF. Since that time, storage has been primarily limited to high-gamma waste containing elemental sodium and high-gamma material containing recoverable scrap.

In the early use of this facility, 1.8-m-long containers were placed in these liners and covered with gravel for shielding, and a plate was welded on the top. Since 1977, the double containers described above have been used. These are transported in a cask to the RSWF and lowered by cable into the 3.7-m liners. The upper end of the cable is then attached to the bottom of a 0.9-m-long shield plug, and the cable is left in place in the storage liner. Figure 8 is a drawing of this equipment. These stored containers can be retrieved by removing the concrete shield plug, securing the lifting cable, and hoisting the container into a suitable cask.

Intermediate-level waste generated within the L&O building is either transferred to HFEF for packaging (analytical-cave waste) or packaged and sent to the RWMC (evaporator bottoms). The evaporator bottoms are slurried into a 208-L drum, which is encased in a concrete-filled corrugated culvert pipe. The drum contains a disposable steam coil which is used to further evaporate the slurry to dryness. When the barrel is full, the steam coil is disconnected, the drum is capped, and concrete is poured on top for shielding. The resulting package is shipped to the INEL RWMC.

Solid waste generated by ANL-West and EBR-II operations for the period 1968-1978 is shown in Table II.

B. Liquid Radioactive Waste

1. Sources of Liquid Waste

No radioactive liquid waste is produced by operation of the EBR-II reactor or within the reactor containment building except for controlled liter-batch quantities of water/alcohol used for component decontamination. Therefore, no liquid-waste system has been installed within the containment building.

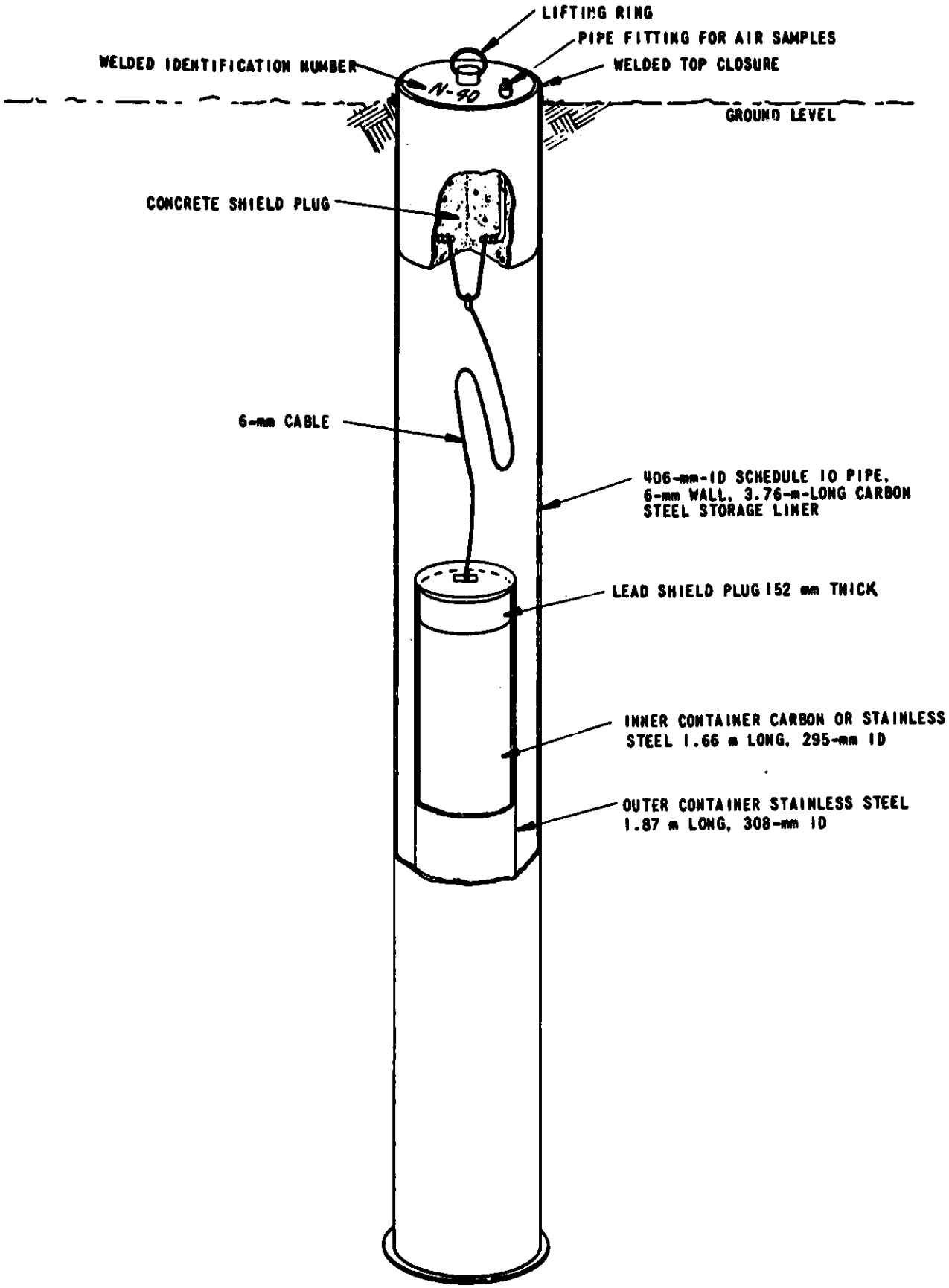

Fig. 8. RSWF Storage Liner with HFEF/N Waste Container

TABLE II. ANL-West and EBR-II Solid Radioactive Waste, 1968-1978

Year	Total Solid Waste To INEL RWMC		Total Solid Waste To ANL-W RSWF		EBR-II Solid Waste[a]	
	Curies	m^3	Curies	m^3	Curies	m^3
1968	2.1×10^4	4.8×10^2	1.9×10^6	9.1	< 1	96
1969	7.0×10^3	4.5×10^2	9.5×10^5	5.1	< 1	90
1970	2.1×10^3	3.5×10^2	6.7×10^5	1.8	< 1	70
1971	3.3×10^3	4.2×10^2	4.8×10^5	3.0	< 1	84
1972	4.0×10^1	3.4×10^2	2.7×10^5	4.1	< 1	68
1973	8.6×10^2	4.7×10^2	4.4×10^5	2.3	< 1	94
1974	7.1×10^2	2.7×10^2	9.2×10^4	1.7	< 1	54
1975	1.2×10^2	3.4×10^2	9.2×10^4	3.5	< 1	68
1976	4.3×10^2	4.8×10^2	6.2×10^4	2.5	< 1	96
1977	1.9×10^5	3.8×10^2	8.6×10^4	2.4	< 1	76
1978	1.8×10^5	4.1×10^2	1.2×10^4	2.6	< 1	82

[a]EBR-II waste volume is estimated to be 20% of volume sent to INEL RWMC.

Components and equipment that are to be repaired or discarded are cleaned to remove or react the attending sodium. Radioactive liquid is produced by the reaction of water/alcohol with the radioactive sodium that adheres to these items. This cleaning is done in the sodium-component maintenance shop (SCMS).

Tritium is produced in EBR-II and eventually released in the liquid stream.[4] Ternary fission is the major source of this tritium, which is released to the primary sodium coolant. Although tritium is also generated in boron carbide used in high-worth control rods and is present in materials studied in experimental subassemblies, it is quantitatively retained in the boron carbide at the temperatures reached in EBR-II and is not released to the coolant. Approximately 90-95% of the tritium produced in ternary fission is released from the fuel through the stainless steel cladding to the primary sodium coolant.

Distribution of tritium in the EBR-II power-plant complex has been investigated:[4] (1) a portion of the tritium remains in the primary sodium coolant, (2) a large but unknown fraction is deposited in the primary-sodium-purification cold trap, (3) a very small fraction is transferred to the primary argon cover gas, and (4) a small fraction is transferred through the intermediate heat exchanger to the secondary sodium. Of the tritium that reaches the

secondary sodium system: (1) a portion remains in the secondary sodium, (2) a large but unknown fraction is deposited in the secondary-sodium-purification cold trap, (3) a very small fraction is transferred to the secondary argon cover gas, and (4) a small fraction is transferred through the evaporator and superheater tube walls to the steam system. The level of tritium in the steam-turbine condensate averages about 10 pCi/cm^3. The makeup rate for the steam system is 38 m^3 of water per day; therefore the tritium release through the steam system is about 380 μCi per day. At a 70% plant factor, this totals about 0.1 Ci per year.

When subassemblies are removed from the reactor, they are transferred to HFEF/S. The sodium is removed from the surfaces by an argon/water wash, and the liquid is collected in a 5700-L retention tank. Maintenance operations in the hot cells result in decontamination liquids, which are also collected in the retention tank. The contents of the tank are then transferred to the central liquid-processing facility. Most of the liquid waste from HFEF/S can be directly attributed to EBR-II power plant operations as opposed to experimental program support.

The HFEF/N facility is primarily used for examination of experiments and is a source of liquid, mainly from decontamination. This liquid is collected in a retention tank and then transferred to the central liquid-processing facility. This liquid is not attributable to EBR-II power operations, since it is generated in support of EBR-II experiments and the breeder-safety-program experiments conducted at TREAT and the Sodium Loop Safety Facility.

The L&O building produces radioactive liquid waste in the analytical labs and caves in support of the EBR-II program.

2. Handling of Liquid Waste

a. 1968-1970

During this period liquid wastes were collected in retention tanks within the facilities, sampled, and discharged to a leaching pit if the activity was within 100 times the then-AEC guidelines for release to controlled areas. An evaporator system was available for use in the event the concentration exceeded 100 times the guidelines. The evaporator bottoms were then handled as solid waste (see Subsection IV.A.2). As previously noted, EBR-II itself produced very little of the liquid waste. The majority came from the secondary operations of washing subassemblies or components to remove sodium, or decontaminating HFEF/S equipment prior to maintenance.

The leaching pit which served as a receiver for this waste is 11 m long by 5.5 m wide by 3 m deep. It is covered with a 200-mm-thick concrete slab which protects it from the weather and ingress of wildlife.

b. 1971-1973

An effluent-reduction program was established in 1971, which reduced by a factor of 100 the concentration of radioactive liquid allowed to be discharged to the leaching pit. This reduction was achieved by evaporating all radioactive liquids having concentrations greater than the guidelines for release to controlled areas. Condensate was pumped to the leaching pit, or recycled through the evaporator if greater than the value allowed to be discharged to controlled areas. Figure 9 shows that a considerable reduction of effluent was achieved during this period--even though the curies generated in the facilities remained relatively constant.

c. 1974-1978

Further reductions of radioactivity in the liquid discharges occurred during this period. Administratively the discharge limit was reduced to 3×10^{-7} $\mu Ci/cm^3$ gross beta-gamma activity, based on the release of ^{90}Sr to uncontrolled areas. The use of the leaching pit was discontinued, and all treated radioactive waste was released to the industrial waste pond.

The central liquid-processing facility (evaporator) is located in the L&O building. Figure 10 is a schematic of the processing facility. The radioactive liquid waste from all facilities at ANL-West is transported either through underground pipes or by means of portable tanks to retention tanks at the evaporator. The liquid is processed through a clarifier (centrifuge) and then received in one of two 5700-L carbon steel settling tanks. When this tank is full, the waste is pumped through a welded stainless steel piping system to one of two 5700-L glass-lined evaporator feed tanks. Both tanks are equipped with high-level indicators, which activate local and remote visual and audible alarms.

The evaporator is a commercially available natural-circulation system composed of two main parts: the heat exchanger and the flash chamber. The evaporator can process a 5700-L batch at a design capacity of 980 L/hr with a measured decontamination factor of between 10^2 and 10^4, depending on the concentration of the feed. A safety feature of the evaporator is an in-line radioactivity detector for monitoring the condensate of the steam heating system. If concentrations reach a predetermined alarm point, indicating a leak through the tubes, an alarm is sounded both locally and remotely at a location that is continuously occupied, and the steam condensate is automatically diverted to the evaporator condensate tank.

The vapor from the evaporator is carried overhead into a condenser and then a cooler. The evaporator condensate flows to an evaporator condensate tank, which is also a 5700-L glass-lined carbon steel tank. It is then processed through ion-exchange columns and collected in a retention tank. The 85-L resin beds use two parts of mixed-bed resin to one part of

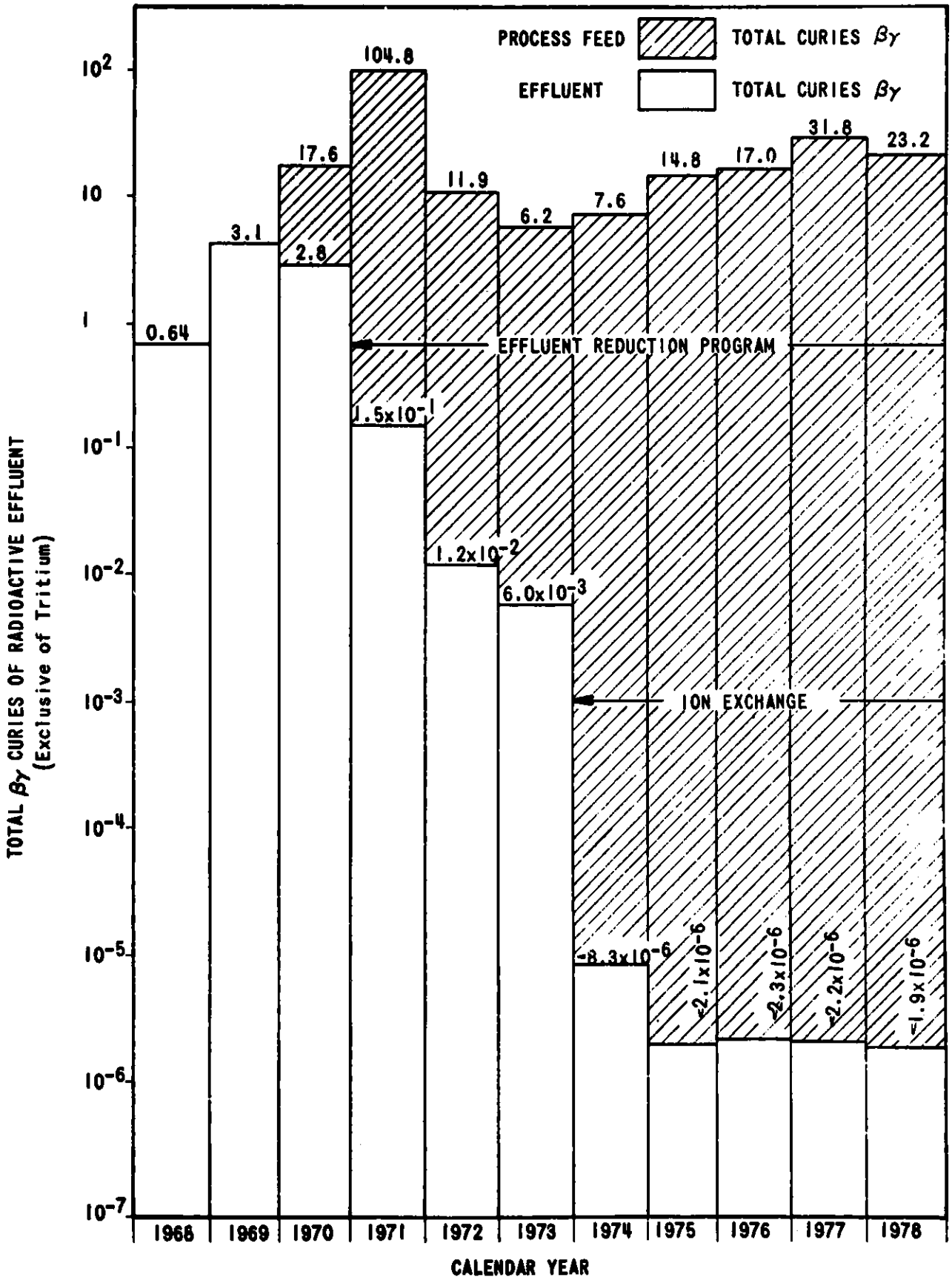

Fig. 9. ANL-West Radioactive Liquid Waste—Process Feed and Effluent

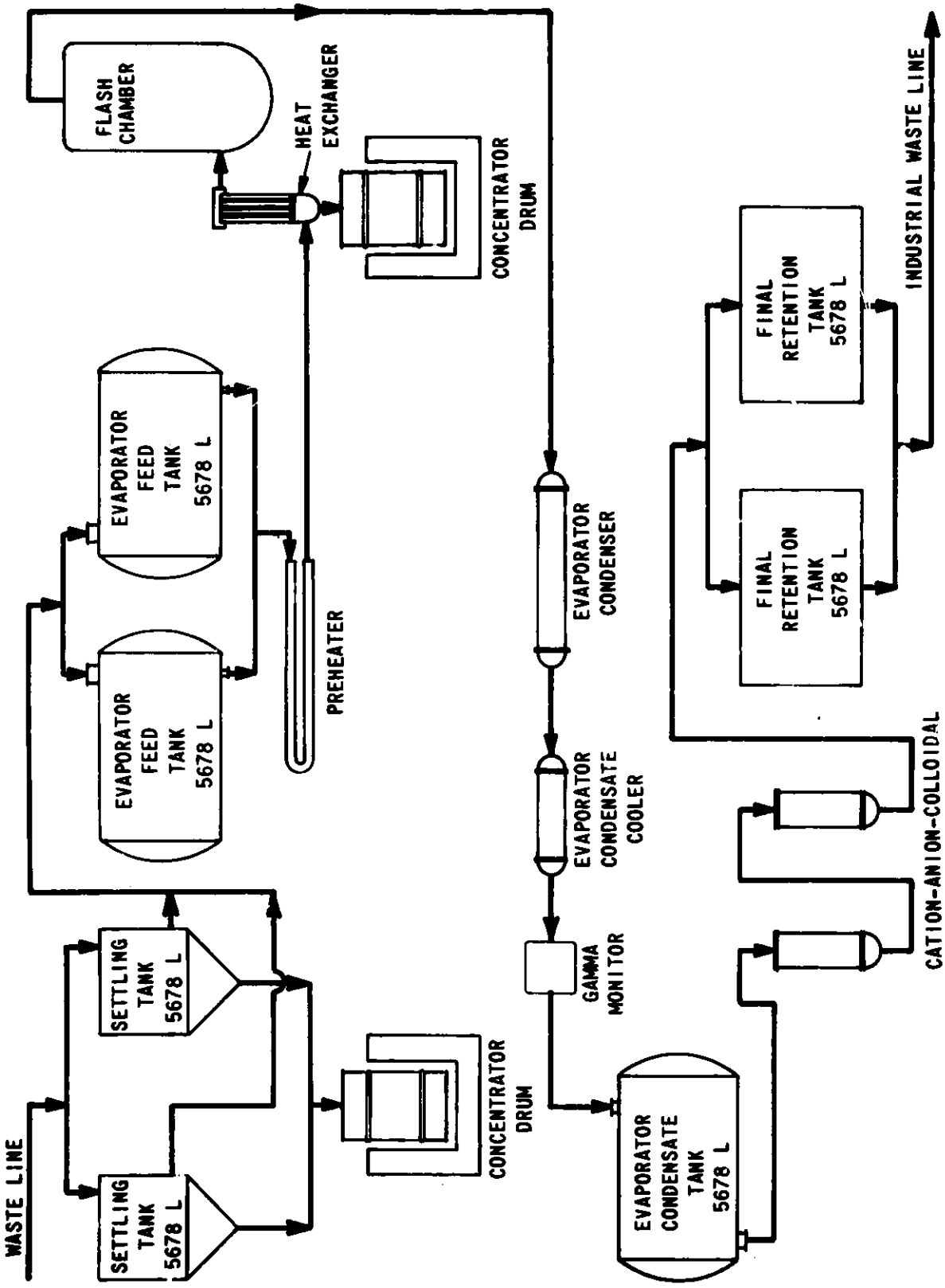

Fig. 10. Schematic of Central Liquid-processing Facility

macroreticular resin for colloid removal. The normal processing line-up consists of two ion exchangers in series between the evaporator condensate tank and the final retention tank. The resin is changed by dumping the ion-exchanger contents into a 208-L drum for disposal as solid waste. This occurs about four times per year.

Processed water is sampled and analyzed for residual radioactivity and then pumped to the industrial waste pond or recycled. All condensate discharged to the industrial waste pond is monitored continuously by an in-line monitor that alarms both locally and remotely at a location that is continuously occupied. An alarm results in notification of waste-management personnel, who institute prescribed procedures to correct the condition.

The concentrated waste in the bottom of the evaporator heat exchanger is transferred by gravity to a 208-L carbon-steel disposable concentrator drum. Any liquid remaining in the residue is evaporated with an expendable copper steam coil, which remains with the residue at the time of disposal. The vapor from the concentrator drum is passed through a moisture separator and then channeled through 20 HEPA filters in parallel to the exhaust stack. The condensate from the moisture separator is returned to the evaporator feed tank.

During this 1974-1978 period, additional emphasis was placed on removing radioactivity at the generating source. In HFEF/S, the wash station for interbuilding coffins was modified to include filters to filter the wash water before it enters the facility retention tank. An ion-exchange unit is also in-line and available.

The EBR-II Project constructed a sodium-components maintenance system for removing sodium from reactor components requiring maintenance. This system moves the sodium-reaction process indoors and provides an attendant waste-collection system. The elimination of use of an open pad and buried tank has reduced the volume of liquid processed from this source by eliminating the water collected from rain and melting snow.

Although the limit for discharging water to the industrial waste pond has been set at 3×10^{-7} $\mu Ci/cm^3$ beta-gamma, actual practice is to evaporate and ion-exchange all water that might significantly increase the annual total of radioactivity discharged. This practice is in keeping with the "as low as reasonably achievable" policy. Except for the tritium contained in the waste-treatment-facility effluent and in the EBR-II turbine condensate, ANL-West liquid discharges have contained no radioactivity above the detectable limits of 1×10^{-8} $\mu Ci/cm^3$ beta-gamma or 1×10^{-9} $\mu Ci/cm^3$ alpha in 1977 or 1978, although the radioactivity generated has remained fairly constant. Tritium liquid discharges remain in the 100-200 mCi/year range and are dependent primarily upon the plant capacity factor achieved by EBR-II.

C. Gaseous Radioactive Waste

1. Sources of Gaseous Waste

Radioactive airborne effluent from EBR-II consists of noble-gas fission products (which include isotopes of xenon and krypton and their daughter products) and activation products entrained in the argon cover gas. The argon cover gas in the primary tank is located in a plenum immediately above the surface of the primary sodium. Because of a positive pressure differential (i.e., pressure inside the tank greater than that outside it), leakage from the cover gas enters the containment building. The building atmosphere passes through HEPA filters to a 60-m-high exhaust-gas stack.

During reactor operation in which various fuels are tested, fuel-cladding failures sometimes occur. In this event, fission-product noble-gas concentrations in the primary cover gas increase. In order to identify the subassembly containing the failed element, continued reactor operation may be necessary; however, this may increase the fission-product concentrations in the argon cover gas and in the containment building. In the past it was sometimes necessary to purge the cover-gas system to reduce the fission-product concentration. The purged argon cover gas was exhausted at a rate of up to 0.08 m^3/min into the stack system. In 1977, a cover-gas cleanup system was activated, which has practically eliminated the need for cover-gas purges.

The HFEF releases little, if any, radioactive gas that is directly attributable to EBR-II operations. Radioactive gases in small quantities are released from the L&O building, primarily the analytical caves and labs.

2. Handling of Gaseous Waste

a. 1968-1969

The major portion of EBR-II gaseous releases originate from the argon-cover-gas system. The reactor core is immersed in a primary containment vessel that contains approximately 325 m^3 of molten sodium. Immediately above the surface of the sodium is an 18-m^3 plenum region filled with argon. Because of a positive pressure differential, any leakage from the cover-gas system enters the containment building. This leakage gas is combined with the atmosphere in the containment building. Approximately 2.5 m^3/s of the building atmosphere is withdrawn through the shield-cooling system and through HEPA filters, and combined with another 0.75 m^3/s which is withdrawn through the thimble-cooling system and also passed through HEPA filters. The combined flow of 3.2 m^3/s is then passed through a radiation monitor and through a blower to the 60-m-high stack. Approximately 2.1 L/min of argon cover gas from the cover-gas plenum is discharged through monitoring devices into the radioactive-gaseous-waste disposal system downstream of the HEPA filters. Annual releases of radioactive gases from the reactor containment building are shown in Table III.

TABLE III. ANL-West and EBR-II Radioactive Gaseous Releases, 1968-1978

YEAR	Total ANL-W Curies	EBR-II Curies[a]
1968	837	0.01 (0.001%)
1969	130	0.00 --
1970	84	0.1 (0.1%)
1971	74	8.6 (12%)
1972	127	20 (16%)
1973	803	674 (84%)
1974	666	515 (77%)
1975	669	482 (72%)
1976	556	379 (68%)
1977	635	461 (73%)
1978	297	143 (48%)

[a]EBR-II percentage of total ANL-W shown in parenthesis.

b. 1970-1976

This period represents the expansion of testing of fuel elements in the run-to-cladding-breach (RTCB) program. Such operations, of course, produce breached fuel elements, and in this event, fission-product noble-gas concentrations increase. In order to establish the identity of the subassembly containing the failed element, continued reactor operation was sometimes necessary, thereby increasing fission-product concentrations in the argon cover gas and the containment building. It then became necessary to purge the cover-gas system to reduce the fission-product concentrations in the reactor building. The purge-discharge rate (maximum 0.08 m^3/min) was determined by identifying the nature and concentration of the contaminant, so that discharge did not exceed limits. The purge exhaust during this time bypassed the HEPA filters and went directly into the stack.

The concentration of radionuclides in the containment building increases in proportion to the activity of the argon cover gas. Under off-normal conditions of cover-gas activity, samples of the containment building were taken and analyzed to determine the nature of the contaminants. If the activity level was within predetermined acceptable limits, the containment-building area was purged directly to the outside atmosphere through a 3-m^3/s centrifugal blower in addition to that processed through the HEPA filter system previously described.

As would be expected, the quantities of gaseous radioactivity released increased significantly during this period. These quantities are shown in Table III.

c. 1977-1978

In anticipation that breached elements in the run-beyond-cladding-breach (RBCB) program would release substantial quantities of fission products to the primary system, especially fission gases, the cover-gas cleanup system (CGCS) was installed. This system processes the primary cover gas to remove fission gases in a cryogenic column. The effectiveness of this system, plus substantial reductions in the gas leakage rate through the primary-tank cover to the building containment, has sharply reduced the release of fission gases to the environment via the site stack. This effect applies, of course, to both the RBCB and RTCB programs, which are being conducted simultaneously.

The CGCS became operational in June 1977. Releases from EBR-II in 1977 prior to CGCS operation totaled 411 Ci. Although the RBCB program results in greater fission-gas releases to the cover gas, releases after the startup of the CGCS totaled only 50 Ci (Table III). Prior to June 1977, ^{135}Xe and ^{133}Xe comprised the large majority of the radionuclides released. Since that time the effluent has consisted of primarily ^{85}Kr.

In 1978 the exhaust from the L&O analytical caves was removed from the 60-m main stack and redirected to a separate monitored exhaust stack. This not only improved the efficiency of the blowers in the main stack but allowed for more definitive evaluation of the source of airborne releases. Previously an alarm on the main stack could have been caused by a release from either EBR-II, L&O, or HFEF/S. Under the present arrangement with EBR-II's additional upstream monitor, the facility in which the release originated can be immediately determined.

V. COMPARISON OF EBR-II WITH COMMERCIAL NUCLEAR POWER PLANTS

Radioactive releases from EBR-II were compared with those of commercial power plants by using the data in the NRC publication "Radioactive Materials Released from Nuclear Power Plants (1976)" NUREG-0367, which was the latest report available at the time. That report cautions "against making simplistic comparisons of radioactive releases with the energy generated because of the many factors which affect the amount of radioactive materials released; factors such as the condition of the fuel, primary system integrity, effluent and radioactive waste treatment systems and the extent to which these systems are used." Nevertheless, the overall environmental release per unit of energy generated was considered to be a reasonable index in the present report for comparing an LMFBR against the light-water plants.

Each of the commercial plants listed was ranked according to the curies released per terawatt-hour thermal in four categories: noble gases, airborne halogens and particulates, liquid tritium, and liquid mixed fission and activation products. The data used were, in all cases, the most recent: 1976 for all plants except EBR-II, and 1978 data for EBR-II.

Since the stated purpose of EBR-II is to test the effects of fuel elements with breached cladding, consideration was given to adjusting EBR-II's 1978 releases to exclude the noble gases released as a result of the RTCB and RBCB programs. However, since light-water reactors may, at times, continue operation with breached fuel, it was decided to use the 1978 EBR-II gaseous-release data. Thus the EBR-II noble-gas data include the fission-gas releases from the 15 RTCB subassemblies that incurred cladding breach and the three subassemblies that were irradiated as part of the RBCB program.

The EBR-II data were then placed in the rankings for the four categories and the overall "good neighbor" ranking arrived at by summing the four rankings. Table IV shows the tabulation of all the plants reported in order of their ranking by this system. EBR-II, representing the LMFBR, is considered to be the number three good neighbor by this simple system. Note that the two plants having a lower index are both 10 years younger than the EBR-II plant.

Table IV also tabulates the volume of liquid waste released by each plant. This volume was adjusted to liters per terawatt-hour thermal for comparison, even though it may not be realistic. EBR-II, for example, generates liquid waste not on the basis of power produced but of the number of subassemblies removed and washed. Solid-waste volumes and radioactivity are tabulated as total cubic meters and total curies. No other meaningful comparison can be made since, generally speaking, solid waste is generated inversely with power production; i.e., when the reactor is shut down for maintenance more solid radioactive waste is generated. Since light-water reactors normally store removed fuel in the absence of any reprocessing facility, the solid waste generated by the disassembly of EBR-II driver subassemblies for reprocessing was not included in the comparison.

TABLE IV. Radioactive Releases of EBR-II and of Commercial Nuclear Power Plants

	Plant	Type	Year of Initial Crit.	[a]TWh	Noble Gas, Ci/TWh	Halogens & Particulates, Ci/TWh	Liquid ^{3}H, Ci/TWh	[b]Liquid MFP & MAP, Ci/TWh	Liquid Volume, L/TWh	Solids, Ci	Solid Volume, m^3
1.	Rancho Seco	PWR	74	6.91	18.4 (4)	<0.001 (1)	0.0 (1)	0.0 (1)	0.0	28.0	111
2.	Vermont Yankee	BWR	72	10.2	297 (18)	<0.001 (1)	0.157 (5)	0.00005 (4)	30 784	238	29.3
3.	EBR-II	LMFBR	63	0.40	358 (19)	<0.001 (1)	0.350 (9)	0.000005 (3)	435 000	< 1	82
4.	E. I. Hatch	BWR	74	13.8	203 (16)	<0.001 (1)	0.651 (10)	0.0029 (8)	1 072 463	412	291
5.	Beaver Valley 1	PWR	76	1.97	0.54 (1)	<0.001 (1)	4.37 (20)	0.086 (20)	7 208 121	0.04	43
6.	Yankee Rowe	PWR	60	4.25	6.05 (3)	<0.001 (1)	36.7 (37)	<0.0022 (7)	4 376 470	26.5	360
7.	Three Mile Island 1	PWR	74	13.9	199 (15)	<0.001 (1)	13.6 (24)	0.0072 (11)	241 727	185	406
8.	Cook 1	PWR	75	21.5	45.3 (7)	<0.001 (1)	8.93 (23)	0.087 (21)	120 930	0.26	169
9.	Palisades	PWR	71	9.66	3.1 (2)	0.0043 (20)	0.997 (14)	0.0455 (17)	1 790 890	95.8	681
10.	Millstone Pt. 2	PWR	75	15.2	103 (13)	<0.001 (1)	18.2 (29)	0.017 (13)	516 447	1.8	280
11.	Monticello	BWR	70	12.3	927 (26)	<0.014 (29)	0.0 (1)	0.0 (1)	0.0	285	3 790
12.	Duane Arnold	BWR	74	8.02	656 (23)	0.010 (28)	0.042 (4)	0.0009 (6)	66 209	595	187
13.	Haddam Neck	PWR	67	13.0	34.8 (5)	<0.001 (1)	373 (44)	0.01 (12)	4 684 615	746	767
14.	Maine Yankee	PWR	72	19.4	67.0 (9)	<0.001 (1)	18.9 (30)	0.146 (25)	6 855 670	504	184
15.	Cooper	BWR	74	11.9	3 193 (32)	<0.003 (16)	0.701 (11)	0.0059 (10)	629 412	301	320
16.	Kewaunee	PWR	74	10.8	130 (14)	<0.001 (1)	16.7 (26)	0.262 (30)	708 333	49.4	594
17.	Point Beach 1, 2	PWR	70, 72	21.8	87.6 (11)	<0.001 (1)	31.8 (35)	0.149 (26)	9 770 642	304	199
18.	Prairie Island 1, 2	PWR	73, 74	20.6	84.5 (10)	<0.0012 (15)	93.7 (43)	0.0005 (5)	16 407 766	65.3	152
19.	Fort Calhoun	PWR	73	7.15	271 (17)	0.0028 (16)	17.1 (27)	0.077 (19)	10 909 090	127	757
20.	Trojan	PWR	75	7.54	88 (12)	0.0022 (16)	4.78 (21)	0.367 (35)	2 374 005	4.32	43.7
21.	H. B. Robinson	PWR	70	15.9	40.3 (6)	0.0063 (25)	61.7 (41)	0.024 (15)	9 622 641	62.9	316
22.	Calvert Cliffs 1	PWR	74	19.8	475 (21)	0.007 (27)	13.8 (25)	0.0596 (18)	3 414 141	122	118
23.	Zion 1, 2	PWR	73, 73	31.1	3 666 (34)	0.0029 (16)	24.0 (32)	0.0051 (9)	4 855 305	68.2	2 060

Numbers in parenthesis indicate ranking within that category

TABLE IV (Contd.)

	Plant	Type	Year of Initial Crit.	[a]TWh	Noble Gas, Ci/TWh	Halogens & Particulates, Ci/TWh	Liquid ^3H, Ci/TWh	[b]Liquid MFP & MAP, Ci/TWh	Liquid Volume, L/TWh	Solids, Ci	Solid Volume, m^3
24.	San Onofre 1	PWR	67	7.75	53.7 (8)	<0.001 (1)	437 (45)	0.959 (41)	1 354 639	698	145
25.	Dresden 2, 3	BWR	70, 71	27.4	1 179 (28)	0.20 (42)	0.719 (12)	0.044 (16)	189 781	7120	4 330
26.	Browns Ferry 1, 2, 3	BWR	73, 74, 76	13.4	< 6 007 (37)	<0.005 (23)	<0.30 (7)	<0.295 (32)	4 955 223	260	103
27.	R. E. Ginna	PWR	69	6.98	791 (25)	0.0045 (21)	34.7 (36)	0.099 (23)	6 146 131	97.8	280
28.	Peach Bottom 2, 3	BWR	73, 74	37.2	5 618 (36)	0.026 (32)	1.98 (17)	0.091 (22)	1 126 344	1200	585
29.	Quad Cities 1, 2	BWR	71, 72	25.9	1 297 (29)	0.051 (34)	1.92 (16)	0.270 (31)	861 004	1000	2 350
30.	Arkansas 1	PWR	74	12.1	470 (20)	0.0047 (22)	17.5 (28)	1.08 (42)	795 041	Not reported	
31.	Nine Mile Point	BWR	69	13.1	13 435 (38)	0.168 (41)	0.188 (6)	0.163 (27)	124 427	538	2 510
32.	J. A. Fitzpatrick	BWR	74	12.6	3 500 (33)	0.054 (35)	0.333 (8)	0.477 (37)	1 928 571	619	341
33.	Dresden 1	BWR	59	3.42	132 164 (43)	0.246 (44)	0.0058 (3)	0.105 (24)	581 871	Reported with Dresden 2, 3	
34.	Brunswick 1, 2	BWR	76, 75	7.81	2 433 (31)	0.059 (36)	0.755 (13)	0.421 (36)	6 568 501	1790	646
35.	Oyster Creek	BWR	69	11.8	14 153 (39)	0.542 (45)	3.27 (19)	0.019 (14)	263 559	1200	1 290
36.	Turkey Point 3, 4	PWR	72, 73	26.8	582 (22)	0.016 (31)	28.8 (33)	0.323 (34)	4 701 492	477	1 440
37.	Oconee 1, 2, 3	PWR	73, 73, 74	39.7	1 110 (27)	0.0069 (26)	55.2 (40)	0.200 (28)	491 184	783	1 220
38.	St. Lucie	PWR	76	0.35	4 914 (35)	0.0057 (24)	38.0 (38)	0.229 (29)	12 885 714	1.57	86.8
39.	Surry 1, 2	PWR	72,73	25.1	761 (24)	0.015 (30)	31.2 (34)	1.34 (43)	13 466 135	617	700
40.	Pilgrim	BWR	72	7.6	24 079 (41)	0.089 (38)	6.14 (22)	0.307 (33)	694 737	912	36 900
41.	Big Rock Point 1	BWR	62	0.83	18 313 (40)	0.063 (37)	2.90 (18)	0.928 (40)	598 795	28.8	3.69
42.	Millstone Point 1	BWR	70	11.6	43 707 (42)	0.201 (43)	1.73 (15)	0.832 (39)	676 724	1330	1 700
43.	Indian Point	PWR	73	7.6	1 526 (30)	0.032 (33)	43.7 (39)	0.655 (38)	9 881 578	946	919
44.	Humboldt Bay 3	BWR	63	0.68	136 765 (44)	0.123 (40)	19.1 (31)	1.46 (44)	2 338 235	84.9	4.09
45.	Lacrosse	BWR	67	0.61	203 278 (45)	<0.116 (39)	67.2 (42)	< 9.48 (45)	2 032 786	Not reported	

Numbers in parenthesis indicate ranking within that category

[a]Thermal power production for year, terawatt-hours (terawatt = 10^{12} watts).

[b]Liquid mixed fission products and mixed activation products.

The overall tabulation of gaseous, liquid, and solid wastes shows that an LMFBR compares very favorably with light-water plants.

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