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SODIUM REACTOR EXPERIMENT DECOMMISSIONING FINAL REPORT

DOE Research and Development Report

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Rockwell International

Environmental & Energy Systems Division
Energy Systems Group

SUMMARY

The Sodium Reactor Experiment (SRE) located at the Rockwell International Field Laboratories northwest of Los Angeles was developed to demonstrate a sodium-cooled, graphite-moderated reactor for civilian use. The reactor reached full power in May 1958 and provided 37 GWh to the Southern California Edison Company grid before it was shut down in 1967. Decommissioning of the SRE began in 1974 with the objective of removing all significant radioactivity from the site and releasing the facility for unrestricted use.

The SRE site consisted of the main reactor building and support buildings and facilities, such as sodium storage and purification, radioactive waste storage, component cleaning, waste retention, heat exchangers, and cooling systems.

Planning documentation was prepared to describe in detail the equipment and techniques development and the decommissioning work scope. A plasma-arc manipulator was developed for remotely dissecting the highly radioactive reactor vessels. Other important developments included techniques for using explosives to cut reactor vessel internal piping, clamps, and brackets; decontaminating porous concrete surfaces; and disposing of massive equipment and structures. The documentation defined the decommissioning in an SRE dismantling plan, in activity requirements for elements of the decommissioning work scope, and in detailed procedures for each major task.

An early decision was made to retain the SRE building superstructure, primarily to provide containment for airborne contamination released by the decontamination operations. Controls to limit personnel radiation exposure and radiation release were established and maintained throughout the program. Arrangements for the collection, packaging, and burial of radioactive waste were made, first at Beatty, Nevada, and later at Hanford, Washington.

Decontamination began with the removal of peripheral nonradioactive systems, such as the process water system, kerosene cooling system, pipe gallery

nitrogen cooling system, secondary sodium systems, and heat exchangers. Remaining bulk sodium was drained and shipped offsite. Residual sodium in the reactor vessel and sodium system components were reacted with alcohol to negate a potential chemical hazard. Liquid and gaseous waste holdup systems were excavated, removed, and shipped to a burial location. The SRE retention pond was drained and decontaminated. Reactor vessel internals were cut (using explosives) into manageable sections for packaging in a shielded cask and shipping to burial. Remotely operated plasma-torch systems were used to cut the vessels. Massive equipment and structures such as the fuel handling machines, the moderator element handling machine, and the reactor vessel shield ring and plugs were removed and shipped intact to a burial location. The wash cells, dry fuel storage cells, pump dip legs, and hot cell storage thimbles were excavated and removed from the reactor room.

Removal of contaminated soil and bedrock, particularly in the northeast corner of the reactor room, was a significant task. Contamination had penetrated below building column footings, necessitating replacement of the footings after removal of contaminated soil and bedrock.

The excavations were backfilled with clean soil and rubble, and the area was paved. The reactor room walls and ceiling were decontaminated by sandblasting. The SRE interior was repainted, the floor was repoured, and lighting fixtures were replaced.

SRE decommissioning operations generated 136,411 ft^3 of radioactive waste, which was sent to a burial site.

A final radiological survey was conducted to verify that the SRE site was decontaminated to levels that allow unrestricted use of the facility. A third party, Argonne National Laboratories, conducted an independent survey also to verify that the objectives were met. An Environmental Evaluation study was prepared to further assure that the area was safe for any future use.

A continuous record of personnel radiation exposure was maintained. Film badges processed by an independent laboratory provided the legally documented record of external exposure. Internal exposures were monitored quarterly by analyzing urine samples. The cumulative group dose (in man-rem) for the project was 89 man-rem, which is well below the amount that would have accumulated had each worker received the limit of 5 rem per year. In fact, it is well below the DOE guideline for new facilities of 1 rem per year per worker. Knowledge gained from the SRE decommissioning will be applied to other decommissioning programs described in this report.

Decommissioning costs for the period 1974 through 1983 were $16.6 million. This is approximately 11% of the $150 million estimated cost to replace the SRE in 1982 dollars. Physical activities at the SRE ended in September 1982.

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1.0 BACKGROUND

1.1 FACILITY HISTORY

The Sodium Reactor Experiment (SRE) was designed by Atomics International, a division of Rockwell International Energy Systems Group (ESG), as a part of a program with the Atomic Energy Commission (AEC) to develop a sodium-cooled, thermal power reactor for civilian application. Construction was largely by subcontract, under the supervision and direction of Atomics International (AI), who also designed and manufactured special components, such as fuel elements, moderator elements, and core component handling machines, for the SRE. Southern California Edison (SCE) installed and operated the steam electric power generating plant.

The SRE was located about 30 miles northwest of Los Angeles. It had been designed and constructed by AI and the AEC to demonstrate the feasibility of a high-temperature, sodium-cooled, graphite-moderated reactor as the heat source of a central power station. It was the first nuclear reactor in the United States to produce power for supply into a commercial power grid. The SRE was a sodium-cooled, graphite-moderated, 20-MW thermal reactor using slightly enriched uranium fuel in the initial core loading. The fuel was in the form of stainless-steel-clad rods with sodium-potassium bonding in the annulus between the fuel and cladding. The active core length was 6 ft. Heat generated in the reactor was transported by the sodium to a heat exchanger and then by a secondary sodium system to a steam generator, which then powered an SCE steam turbine and generator.

Intensive design of the SRE began in June 1954. Actual construction of the plant began in April 1955. Construction was completed in February 1957, and the ambient subcritical experiment, without sodium in the core, was started on 23 March 1957. On 25 April 1957, the SRE was brought to criticality with 350°F sodium in the core. The reactor was brought to full power in early May 1958.

The SRE primary system hot leg thermal history for several temperature ranges is presented in Table 1 for Core I and Core II. Most of the sodium system was in service through both core operations. In Figure 1, thermal history of the SRE is presented as the number of exposure hours at or above any given temperature. Not included in this thermal history is the accumulated operating time of the sodium systems for sodium cleanup purposes in preparation for the SRE Power Expansion Program. This period, which extended from 15 May 1965 to September 1967, included operation of the main primary sodium for 4,386 h at 700°F and 13,196 h at 350°F.

TABLE 1

SRE HOT LEG OPERATIONAL TIME AND TEMPERATURE[a]

Temperature Range (°F)	Time (h)	
	Core I	Core II
<300	120	180
300 to 399	4,080	9,480
400 to 499	2,016	3,288
500 to 599	576	4,008
600 to 699	192	6,408
700 to 799	520	2,256
800 to 899	1,972	1,056
900 to 959	512	40
960 to 1030	356	0
Total	10,344	26,716

[a]Core I: 4 May 1958 to 10 November 1959
Core II: 22 July 1960 to 15 February 1964

The SRE generated more than 37 GWh of electrical energy in more than 27,300 reactor operating hours. A summary of the more important operating statistics is presented in Table 2.

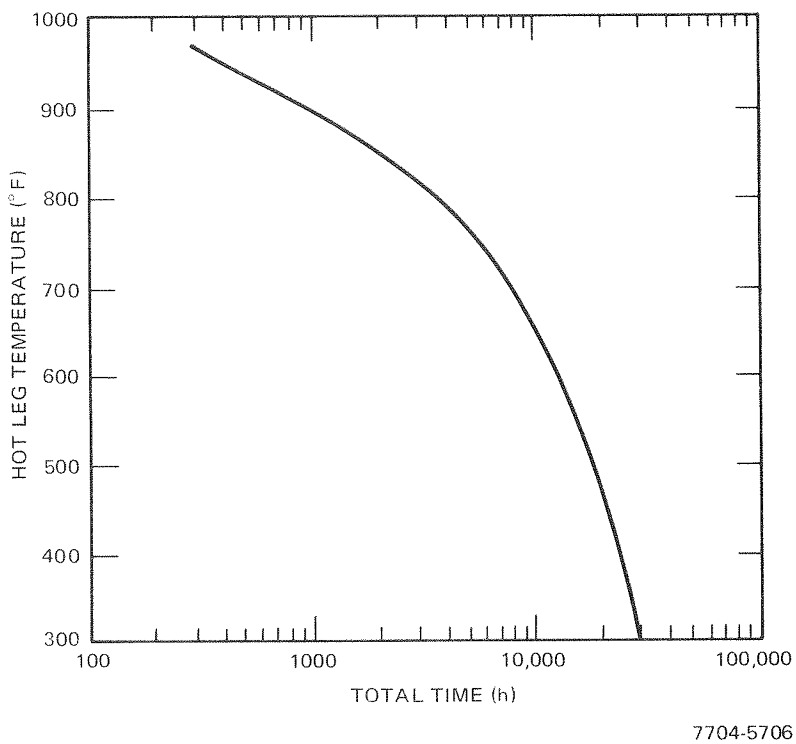

Figure 1. Total Time at or above a Hot Leg Temperature

The SRE was operated from 1957 to 1964 at sodium outlet temperatures up to 1000°F and thermal power levels to 20 MW. In February 1964, the SRE was shut down for Power Expansion Program (PEP) modifications with the objective of raising the sodium operating temperatures to 1200°F and thermal power levels to 30 MW with a stainless-steel-clad uranium carbide fuel loading. These modifications, completed in May 1965, included replacement of the primary and secondary system pumps and the intermediate heat exchanger as well as most of the other components of the primary sodium system. Moderator elements were replaced and a new fuel loading was prepared.

In September 1967, the primary sodium system was shut down and the sodium was drained into the primary fill tank; the secondary sodium was drained into drums. The SRE did not operate as a nuclear plant after 15 February 1964.

TABLE 2

SRE OPERATING STATISTICS

Reactor critical			27,300 h
Integrated thermal reactor power			6,700 MW d
Integrated electrical output			37,174,200 kW h
Primary pumps			
	Main	Original (freeze seal)	37,060 h
		PEP (free surface)	17,582 h
	Auxiliary	Original (freeze seal)	37,060 h
		PEP (free surface)	15,241 h
Secondary pumps			
	Main	Original (freeze seal)	24,760 h
		PEP (free surface)	11,442 h
	Auxiliary	Original (freeze seal)	41,152 h
		PEP (free surface)	17,881 h
Intermediate heat exchanger			
	Main	Original (freeze seal)	37,060 h
		PEP (free surface)	17,582 h
	Auxiliary		55,642 h
Steam generators sodium filled			63,000 h
Steam generators steaming			30,392 h
PEP operation (primary and auxiliary Na system flow at ∿350°F)			17,582 h

A plan for the deactivation of the SRE[1] was approved by the AEC early in 1967. The implementation of this plan resulted in a "stored-in-place" configuration, except that nonessential equipment was removed and the steam generator and noncontaminated support facilities were not maintained. Other major activities of the deactivation were:

1) Transferring Core III fuel from the SRE to storage in Building 064 at the Santa Susana Field Laboratories (SSFL)
2) Draining primary sodium to the fill tank

[1]"SRE-Deactivation Plan," TI-599-19-001 Rev. A, 25 January 1967.

3) Removing noncontaminated secondary sodium in drums from the SRE
4) Modifying the inert gas system to combine the helium and nitrogen gas systems
5) Placing the radioactive waste system in storage by: flushing and draining the liquid waste system and purging the gaseous waste system; decontaminating the sump pit and wash cell pit; replacing the stack filter; shutting down the compressor; shutting off cooling water for the compressor; installing sump pit blocks; transporting waste to the RMDF; disconnecting the electricity to the wash cells, gaseous waste, and sanitary waste systems; draining wash cell steam and water systems; and installing shield blocks in the gaseous waste vaults
6) Decontaminating the external surfaces of the fuel and moderator handling machines
7) Shutting down control and instrument power
8) Decontaminating the main portable hot cells and shutting down the ventilation system
9) Preparing the batteries, motor-generator sets, and diesel generator for an inactive period
10) Providing power for the emergency paging system and perimeter lights
11) Shutting down heating, ventilating, and plant air systems.

These deactivation activities were completed in 1968. A surveillance program, continued until decommissioning activities began in 1974, included: monitoring and servicing of the nitrogen cover gas system; inspection for water, wind, or other damage; sodium system inspection for leaks; and radiological monitoring of contaminated areas.

A decommissioning study, completed in July 1970,[2] assisted the AEC in formulating plans for the ultimate disposition of the deactivated SRE site and

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[2] "Post-Retirement Plan for Radiological Decontamination of the SRE Site," TI-599-19-103, 30 July 1970.

provided an estimate of costs to make the SRE site radiologically clean and safe so that no further surveillance or regulation of the facility would be required. A proposal to decommission the SRE site was submitted to the AEC in GFY 1973, and limited decommissioning activities began in GFY 1974.

1.2 DECOMMISSIONING PROJECT PURPOSE

The SRE site contained radioactive structures, systems, components, concrete, and soil. The SRE was decommissioned to remove all significant radioactivity from the site and to release the facility from all requirements for radiological control, licensing, or monitoring (unrestricted release). At completion of the decommissioning and release for unrestricted use, the facility was decontaminated to levels that are as low as reasonably achievable, and in all cases are below levels specified in Table 3.

TABLE 3

UPPER CONTAMINATION LIMITS FOR DECONTAMINATION AND DISPOSITION AT SRE

Surfaces	
Beta gamma emitters	Total = 0.1 mrad/h at 1 cm, with 7 mg/cm^2 absorber
	Removable = 100 dpm/100 cm^2
Alpha emitters	Total = 100 dpm/100 cm^2
	Removable = 20 dpm/100 cm^2
Soil	
Near surface	100 pCi/g gross detectable beta activity
Below 3 m	
Average	1000 pCi/g gross detectable beta activity
Maximum[a]	3000 pCi/g gross detectable beta activity
Concrete Rubble	1000 pCi/g gross detectable beta activity

[a]The maximum value may be averaged over a volume of 1 m^2 to meet the limit for the average value.

2.0 FACILITY DESCRIPTION

2.1 BUILDINGS AND SYSTEMS

The SRE was a complex of buildings, work areas, and systems (see Figure 2). In addition to the reactor building (Building 143), there were radioactive waste storage in Building 041, sodium components cleaning in Building 724, hot waste storage in Building 686, hot components repair in Building 163, sodium purification in Building 695, sandblast cleaning in Area 723, primary sodium fill tank and system in Building 753, secondary sodium fill tank and system in Building 453, liquid waste holdup system and gaseous waste holdup system in Building 653, and cask and hot component storage in Area 654. Included in the decommissioning effort were the kerosene cooling system for the biological shield, fuel storage cells, and reactor plugs; primary and secondary sodium service systems, fuel element wash cells; hot cell; fuel storage cells; absolute filtered ventilation system; retention pond and dam; vaults cooling system (nitrogen); emergency power system; water supply system and change room with accompanying holdup tanks.

2.1.1 Containment Building Layout

The reactor building layout was as shown in Figure 3. Entrance to the south side of the building was made via the visitors lobby. Adjacent to the lobby was the Health Physics office where visitors obtained film badges. The main corridor adjacent to the lobby led to the visitors gallery where the reactor bay area could be viewed through windows. Access to the control room, recorder room, shift supervisor's office, restrooms, and mezzanine was through the main corridor. The southeast portion of the building comprised the battery room, instrument shops, boiler room, and electrical distribution room.

2.1.2 Containment

The building superstructure was a low-air-leakage building with ventilation and exhaust systems designed to control leakage and air flow paths. The

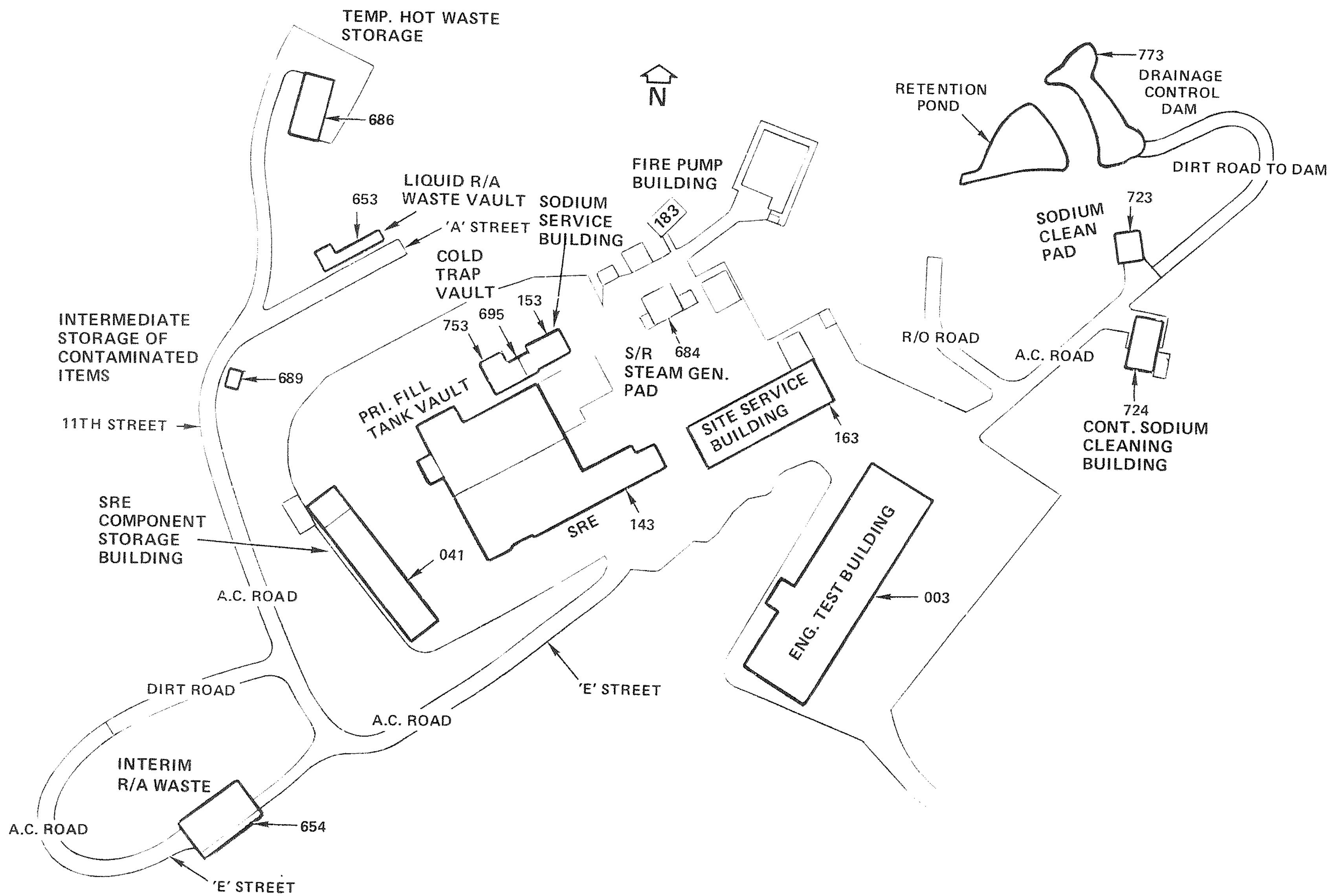

Figure 2. Plan View of SRE Site

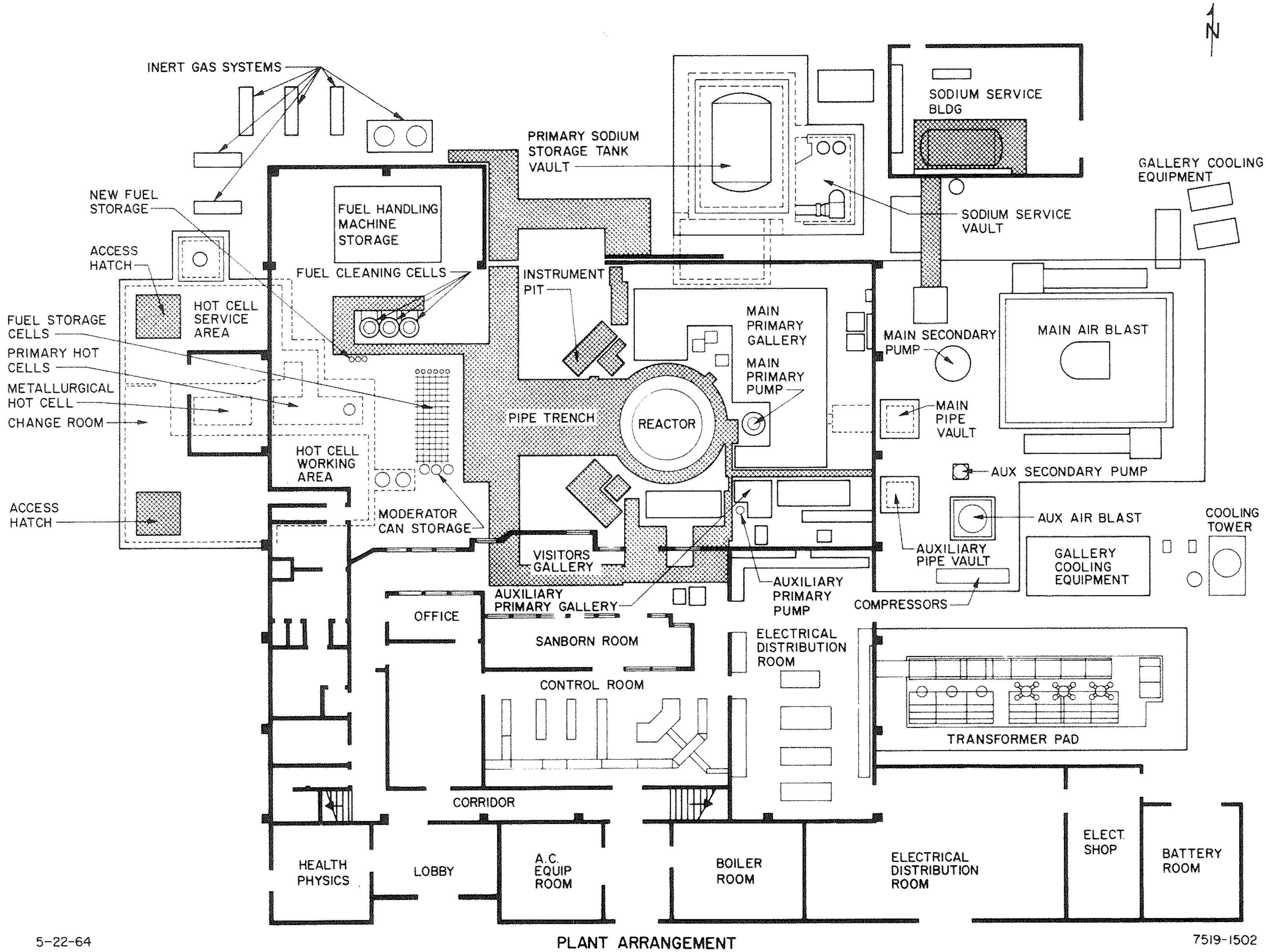

Figure 3. Building Layout

interior of the building was maintained at a lower pressure than the exterior of the building so that air flow would always be into the building. In this manner, any radioactive particulates that might have escaped would have been retained within the structure and trapped by the exhaust filters.

The reactor and components containing radioactive materials were completely enclosed in below-grade vaults and galleries sealed from the outside atmosphere. An artist's cutaway view of the facility is shown in Figure 4.

The reactor core was completely enclosed in a stainless steel cylindrical core tank, which in turn was enclosed within two additional steel enclosures, i.e., the outer tank and the core cavity liner (Figure 5). A helium cover gas blanket filled the space between the sodium and the top shield. The upper portion of the reactor containment structure included the ring shield, the loading face shield, and the various plugs within the loading face shield. A bellows connected the reactor tank to the ring shield.

Confinement of the reactor atmosphere was achieved by means of various seals. Small plugs, such as fuel element and control element plugs, were sealed by two "quad" rings. Double rings were also used to seal the moderator shield plugs.

The loading face shield was sealed to the ring shield by a frozen metal (cerrobend) seal. The cerrobend (a eutectic alloy of bismuth, lead, tin, and cadmium with a melting point of 158°F) was frozen into a trough attached to the ring shield. A 6-in.-long steel cylinder welded to the loading face shield fit into the trough. The alloy expanded during solidification to maintain the seal. When it became necessary to break the seal, built-in heaters were provided to melt the alloy.

The primary radioactive sodium system piping and equipment external to the reactor were contained within two galleries, one for the main loop and one for the auxiliary loop, and within three vaults for the primary drain pump, primary sodium service system and the primary storage tank. Large shield

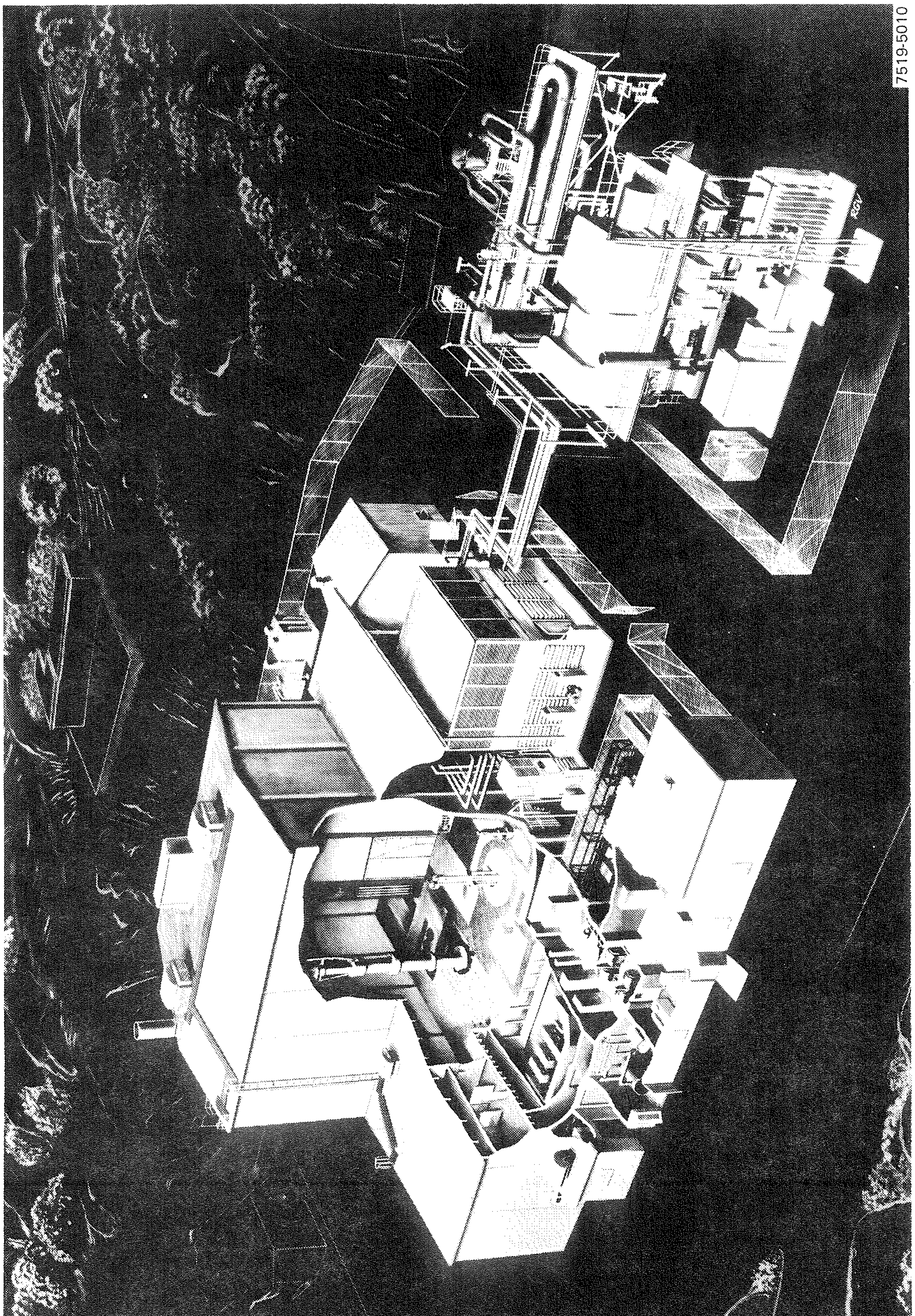

Figure 4. Artist's Cutaway of SRE

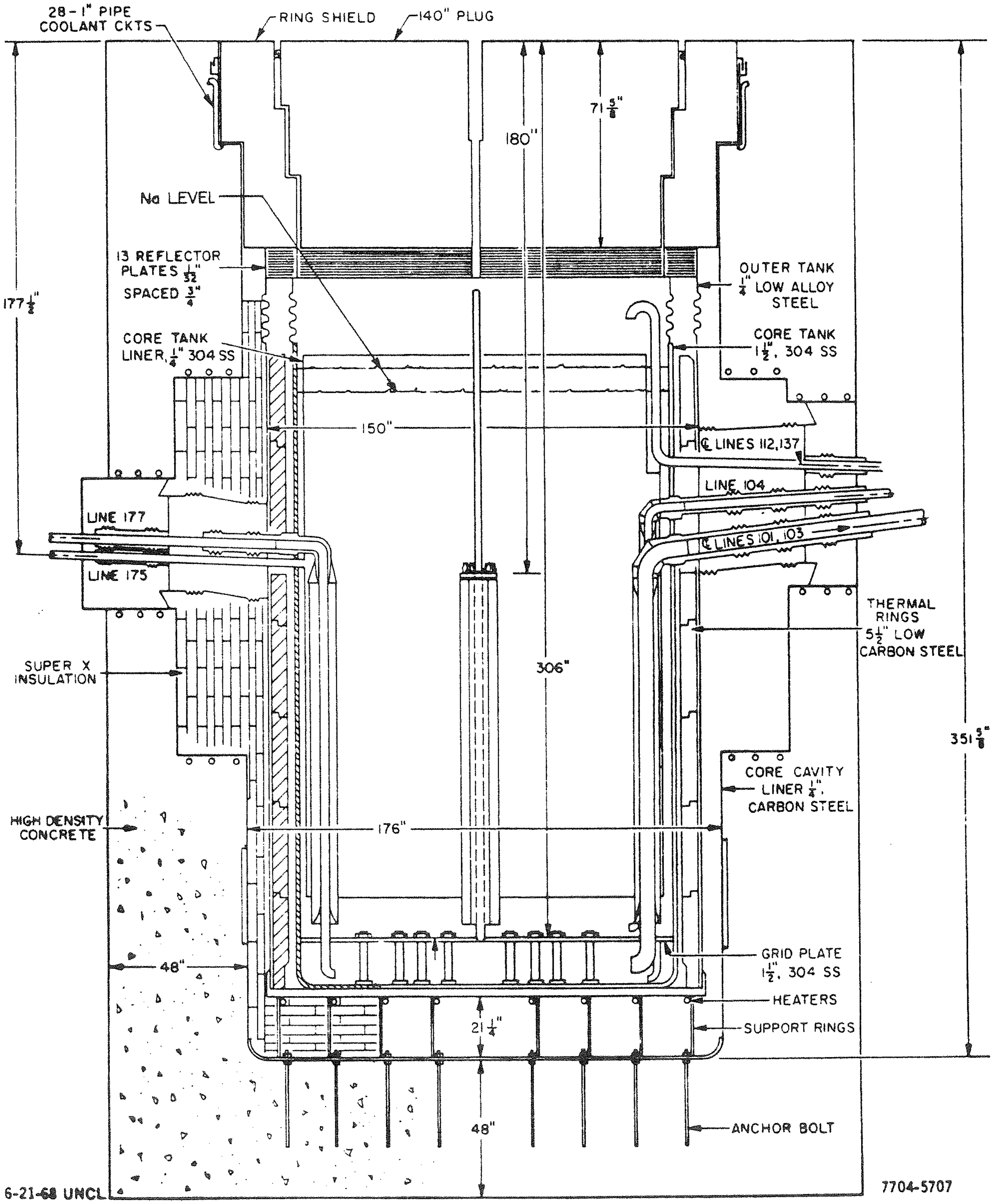

Figure 5. Vertical Cross Section of Reactor

blocks made from a minimum of 4-1/4-ft-thick dense concrete were placed over the vaults and galleries.

The building ventilation system was designed and operated so that air moved toward potentially contaminated areas. Makeup air was brought in from the outside and combined with recirculated air within the administrative areas to maintain positive pressure relative to the contaminated areas. Fresh air at the rate of five air changes per hour was supplied to the reactor room by independent supply fans. Exhaust fans and high-efficiency filters on the reactor room roof were sized to maintain the reactor room pressure below all contiguous regions, which were (1) adminstration areas, (2) hot cell, and (3) out of doors. The use of filters reduced the possibility of local contamination by the accumulation of radioactive particulates on building and equipment surfaces. The pipe and equipment vaults were maintained in an atmosphere of dry nitrogen from the recirculating nitrogen cooling system. By excluding air from the vaults, no sodium-oxygen reaction could occur in the event of a sodium leak.

Blowers, with exhausts to the dilution stack, also maintained a negative pressure in the SRE hot cells. The hot cell personnel area was maintained at a positive pressure relative to both the hot cell chambers and the reactor room. Air from the hot cell chambers was filtered before it left the cell and was filtered again by the radioactive vent system filter banks prior to dilution in the stack.

2.1.3 Reactor Structure

The SRE reactor core was a matrix of moderator elements containing the fuel elements, control rods, neutron source, and devices for measuring temperature and sodium level. Since the SRE was an experimental reactor, it was expected that the core geometry would be changed from time to time and that test elements would be used.

As shown in Figure 5, the core tank was a flat-bottomed, stainless steel vessel supported on the bottom of the core cavity liner. The main coolant inlet plenum was located between the grid plate and the bottom of the core tank. Pedestals on the bottom of the core tank supported the grid plate. The moderator elements were supported by the grid plate, with the lower fittings of the moderator element socket located in holes in the grid plate. The lower ends of the moderator elements, together with the grid plate below, formed the moderator coolant inlet plenum.

Of the 119 moderator elements in the core, 86 were hexagonal shaped and clad with Zircaloy-2. Full and partial hexagonal spaces at the core reflector periphery were occupied by one or more stainless-steel-clad graphite logs.

Of the 86 Zircaloy-2-clad moderator elements, 57 had central process channels; the remaining 29 were solid and were located in the periphery of the core under that part of the reactor loading face shield having no access plug positions. Thirty-three of the 57 process channels contained fuel elements, 8 contained shim- and safety-rod thimbles, 1 contained the neutron source, and 15 (1 of which was in the peripheral Zircaloy-2-clad element) contained instrumentation devices. All components positioned in the process channels were supported from the reactor loading face shield by individual hanger rod and shield plug assemblies.

A core tank liner, extending up from the grid plate to an elevation above the normal sodium level, was located approximately midway in an annulus between the stainless-steel-clad moderator elements and the core tank. It created an annulus of stagnant coolant adjacent to the core tank wall and thereby reduced transient thermal stresses in the core tank wall. The main and auxiliary sodium coolant inlet lines and the moderator coolant inlet line were brought through the core tank wall and the core tank liner at an elevation slightly above the top of the moderator elements. The coolant inlet lines were routed down to their respective inlet plenums through vacancies in the outer ring of moderator elements.

The grid plate was made of Type 304 stainless steel. The upper head castings of the moderator elements and the arrangement of core clamps which held them in position were made of 400 series stainless steel.

The thermal shield was located between the core tank wall and the outer tank. It was constructed of ASTM-A7 structural steel plate and was supported by the bottom of the outer tank. A structure of concentric rings above the bottom of the cavity liner supported the outer tank. The volume between the outer tank and the cavity liner was filled with block thermal insulation. Outside of the cavity liner was the high-density concrete biological shield. Overheating of the shield was prevented by cooling coils attached to the exterior of the cavity liner through which kerosene from the kerosene cooling system was passed. Both the core and outer tanks were sealed by two welded bellows to the cavity liner at an elevation near the top of the core tank. The cavity liner extended upward to the reactor floor. It was stepped to support the reactor loading face shield assembly.

The loading face shield assembly had an outer support ring and a 140-in.-diameter rotating plug, which contained two 40-in.-diameter, one 20-in.-diameter, twenty-four 3-1/2-in.-diameter, and fifty-seven 3-in.-diameter plugs. The loading face shield assembly was made of Type 405 stainless steel filled with high-density concrete and lead shielding. All plugs were supported in stepped channels. The rotatable 140-in.-diameter plug, together with the two 40-in.-diameter plugs, permitted replacement of the moderator elements.

2.1.4 Primary Coolant System

The reactor coolant system consisted of two complete circuits: the main circuit and the auxiliary circuit. The main circuit consisted of a primary and a secondary loop. Included in the primary loop were an intermediate heat exchanger (IHX), a free-surface mechanical pump, an electromagnetic (EM) brake, a valve flow controller, instrumentation, controls, and sodium service connections.

The main primary heat transfer loop removed 1200°F sodium from the reactor by means of the main primary pump, which transferred it directly to the main IHX where the thermal energy was transferred to the nonradioactive main secondary system. The major function of the secondary loop was to provide separation of the steam system from the radioactive primary system sodium.

Main primary sodium circuit piping and components were located in a concrete vault below grade to facilitate containment and shielding. A nitrogen gas atmosphere was maintained in the vault to prevent ignition of any sodium that might leak. The nitrogen also provided cooling and dehumidification of the cell.

The main primary sodium pump was a vertical, single-stage, free-surface, centrifugal unit. The case was mounted permanently in the main primary gallery, with the sodium pipes welded to the pump case. The pump shaft extended upward through the shielding to the pump motor at floor level in the reactor room.

The main IHX (Figure 6) was an all-welded, Type 304 stainless steel, U-shaped shell and tube, vertically mounted unit. Primary sodium passed through 555 tubes of 5/8-in. OD and 0.042-in. wall, spaced on a 7/8-in. triangular lattice. Secondary sodium flowed over the tubes on the shell side. The exchanger was mounted on one fixed pad and two roller pads to permit thermal expansion.

2.1.5 Secondary Systems

The main secondary pump and expansion tank were integrated into a single unit and located in the cold leg of the system. The expansion tank provided space for sodium volume changes and a free surface for liberation of entrained gas.

Sodium was pumped through the main IHX, where it was heated and then passed to the Edison Plant steam generator, where the thermal energy was used

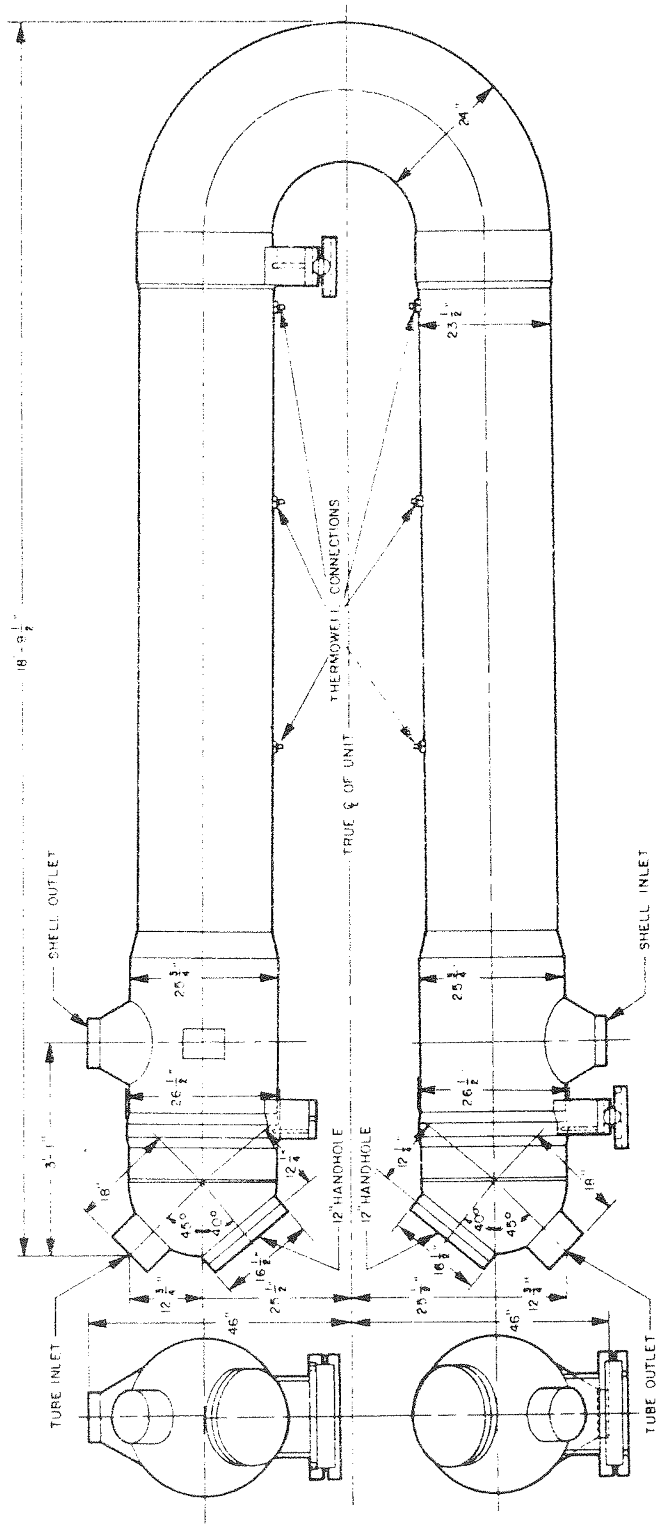

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Figure 6. Main Intermediate Heat Exchanger

to produce steam. The sodium exited from the steam generator and passed through an EM brake prior to returning to the main secondary pump and expansion tank.

The main secondary sodium pump was a vertical, single-stage, free-surface, centrifugal-type unit mounted inside the spherical secondary expansion tank. The main secondary pump internals were essentially the same as those of the main primary pump.

2.1.6 Power Conversion System

The power conversion system was an outdoor installation consisting of a 7500-kW turbine, steam generator, and other equipment common to a conventional steam-powered electricity generation station (Figure 7). The power conversion system had the capacity to remove 30 MW of reactor heat (102×10^6 Btu/h). Salient features of the plant are summarized in Table 4.

TABLE 4

STEAM GENERATOR DESIGN OPERATION CONDITIONS[a]

Parameter	Condition A	Condition B
Heat Load (Btu/h)	102×10^6	102×10^6
Sodium inlet temperature (°F)	900	1166
Sodium outlet temperature (°F)	440	616
Feedwater temperature (°F)	297	297
Steam temperature (°F)	825	825
Steam pressure (psig)	600	600

[a]The steam generator was designed for 30-MW heat removal under either condition A or B.

The steam generator was a once-through, horizontal, U-shaped, shell-and-tube heat exchanger employing 199 double-walled tubes with mercury in the tube annuli. The mercury was pressurized to a value intermediate between the

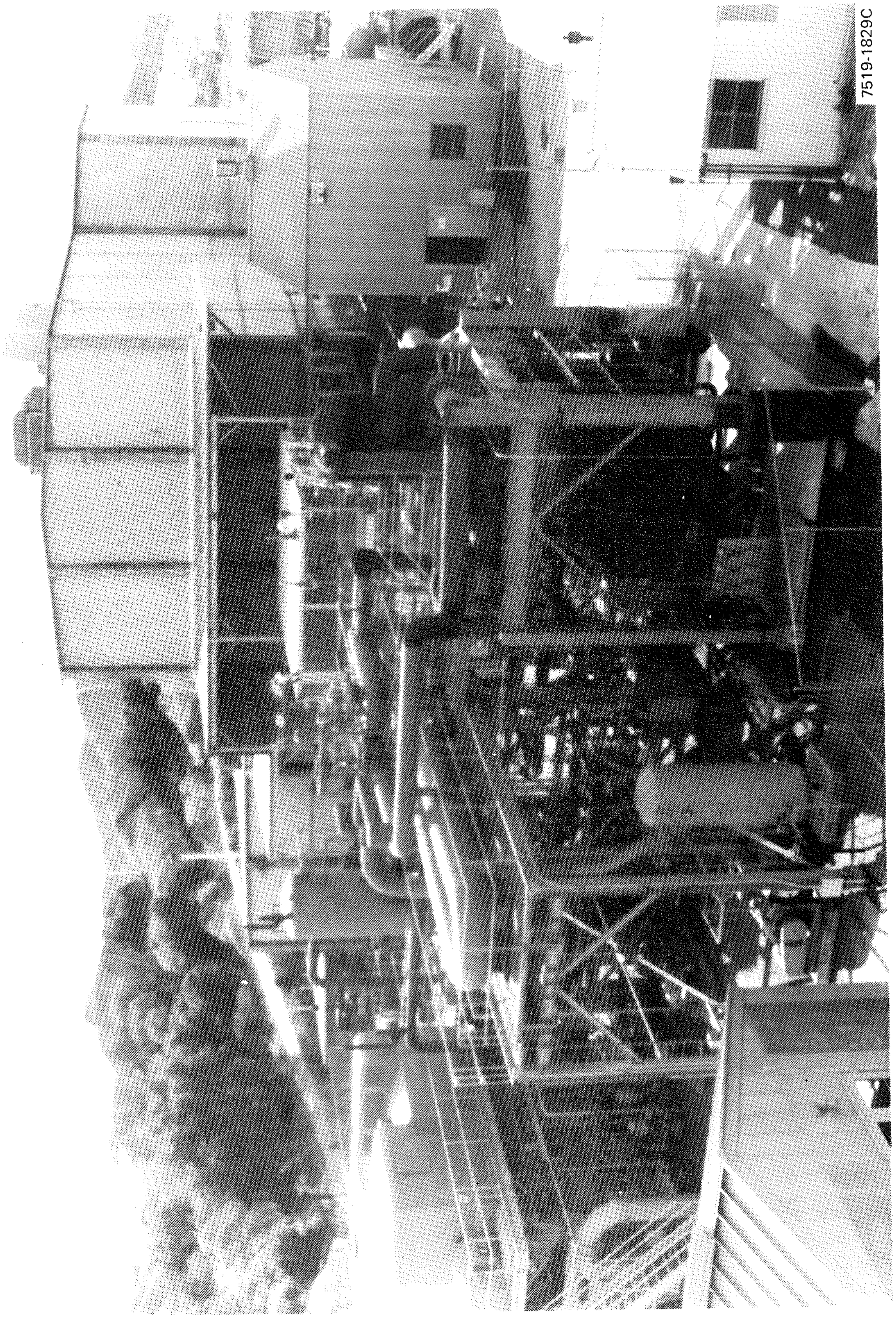

Figure 7. Steam Plant Facilities

water-side pressure of 620 psia in the tubes and sodium-side pressure of 35 psia in the shell.

The steam generator was capable of delivering 88,700 lb/h of superheated steam, with feedwater at 297°F. Controlled cooling of the steam provided 825°F and 600 psig at the turbine throttle. The low-pressure steam from the turbine was exhausted to the condenser and collected as condensate in the hot well. The condensate was pumped through the air ejector condenser and the deaerator where any entrained gas was removed. The feed pumps then took the feedwater from the deaerator through the closed heater and on to the steam generator, completing the cycle.

2.1.7 Reactor Auxiliary Systems

2.1.7.1 Fuel-Handling Machine

The two fuel-handling machines (FHMs) were large, lead-lined casks weighing 55 tons, equipped with hoisting devices to transfer fuel and other core elements.

Special equipment and procedures were provided for the safety of the operating personnel and equipment. The radiation shielding on the FHM consisted of an equivalent 9-1/2 in. of solid lead for a height of over 10 ft.

An inert atmosphere was maintained within the FHM during fuel transfer. A gas-tight seal was formed between the reactor and the FHM as a precaution against release of radioactive gases or the introduction of oxygen to the sodium in the reactor.

2.1.7.2 Sodium Coolant Purification System

The sodium coolant purification system was used to detect and remove carbon, sodium oxide, and other impurities from the sodium coolant. Formation of sodium oxide was minimized by maintaining an inert atmosphere of helium within the sodium tanks and pumps and above the core sodium pool and by purging the sodium system with helium before filling with sodium. Vapor traps and freeze traps were used for purging or venting the sodium system. The major components of the sodium coolant purification system were the circulating cold traps, plugging meters, hot traps, and sodium sampler. One cold trap was provided for the primary sodium system and one for the secondary sodium system. The primary cold trap, located in a vault, was cooled with gallery nitrogen. The secondary cold trap was cooled by air.

2.1.7.3 Kerosene Cooling System

The kerosene cooling system consisted of a main and limited-volume cooling system. The main kerosene system provided cooling for: (1) the core cavity, (2) instrument thimbles, (3) the sodium service vault, (4) fuel storage cells, (5) the wash cell, (6) the primary cold trap nitrogen cooler, and (7) the heat exchanger for the limited-volume kerosene system. It also provided backup cooling for the main primary and auxiliary primary pump barrels. The limited-volume kerosene system cooled the top shield of the reactor.

Major components of the main kerosene system were the supply tank, two circulation pumps, and two evaporative coolers. The system contained 1100 gal of kerosene. The supply tank, which had a capacity of 500 gal, also served as a surge tank.

The limited-volume kerosene cooling system was a closed circuit with a capacity of 50 gal. The primary components of this system were a 12-gal surge tank, a pump, a heat exchanger (heat was transferred to the main kerosene system), and the top-shield cooling circuits that serviced the 20-in.-diameter shield plug, the center 40-in.-diameter shield plug, the 140-in.-diameter shield plug, the ring shield, and the outer 40-in.-diameter shield plug.

2.1.7.4 Sodium Melt Station and Primary Fill Tank

The sodium melt station was used for transferring sodium from 55-gal drums to the primary or secondary fill tanks that were used to fill and drain the sodium coolant systems. The sodium melt station and the secondary fill tank were located in the sodium service building. Since the primary sodium was radioactive, the primary fill tank was located in a vault constructed of dense concrete, which served as a biological shield. The primary tank was constructed of 1/4-in.-thick Type 304 stainless steel and had an 8850-gal capacity.

2.1.7.5 Moderator-Handling Cask

The moderator-handling cask, similar to the FHM, was available for handling, encapsulating, and transporting moderator cans similar to the way the FHM was used for fuel elements.

2.1.7.6 75-Ton Crane

The 75-ton crane was used to operate the FHM and to handle various casks and other objects in the reactor bay area.

2.1.7.7 Fuel and Moderator Storage Cells

There were 99 storage cells available for storage of irradiated core elements. These cells were arranged in a 6 by 16 array and were imbedded in concrete on 1-ft centers. Three additional cells were similar in function, but were not part of the array. Each cell consisted of a carbon steel tube about 25 ft in length with a 4-in. minimum ID and a wall thickness of 1/4 in. The tube was closed except at the upper end where a gas seal and biological shielding were provided by either a special shield plug or the core element shield plug. Seventy-nine of these tubes were attached to kerosene cooling lines for removal of afterglow heat from stored irradiated core elements. A helium atmosphere was maintained in the storage cells and was supplied by either the portable purging equipment or the FHM.

Three moderator storage cells were available for handling or storage of moderator and reflector cans. Each cell was formed by a pipe approximately 22 ft long and 20 in. in diameter, with a 1/2-in. wall thickness. These cells were not cooled, but they could be supplied with an inert atmosphere.

2.1.7.8 Radioactive Vent System

Gas within the primary system was potentially radioactive. It was collected in four main vent lines which emptied into a large tank (Figure 8). A compressor drew gas from the tank, compressing the gas to 100 psig and discharged it into four 500-ft decay tanks.

2.1.7.9 Radioactive Liquid Waste System

The radioactive liquid waste system (Figure 8) consisted of three tanks (150-, 350-, and 3200-gal capacities), three pumps, associated piping and valves, and monitoring equipment. A pump was contained in each tank and was used for transfer or recirculation of liquid waste.

Two small tanks were located in the sump tank pit. Liquid waste could be gravity drained to these tanks from waste sources such as the hot cell and wash cells. Normally, the 150-gal tank would be used as the collection tank and the 350-gal tank as a holdup tank. Waste could be transferred from either of the smaller tanks to the 3200-gal storage tank.

2.1.7.10 Wash Cells

The wash cells were used to clean fuel elements by removing reactor sodium retained on the bundle surface. A cell consisted of a 5-in.-OD, Schedule 40, Type 304 stainless steel pipe vertically encased below the reactor room floor. The FHM was used to load an element into the cell. The element was secured in the cell by a breech-lock mechanism designed to withstand cell pressure as high as 300 psig. Only a few pounds of steam at 300°F were charged past the element during a 30-min interval.

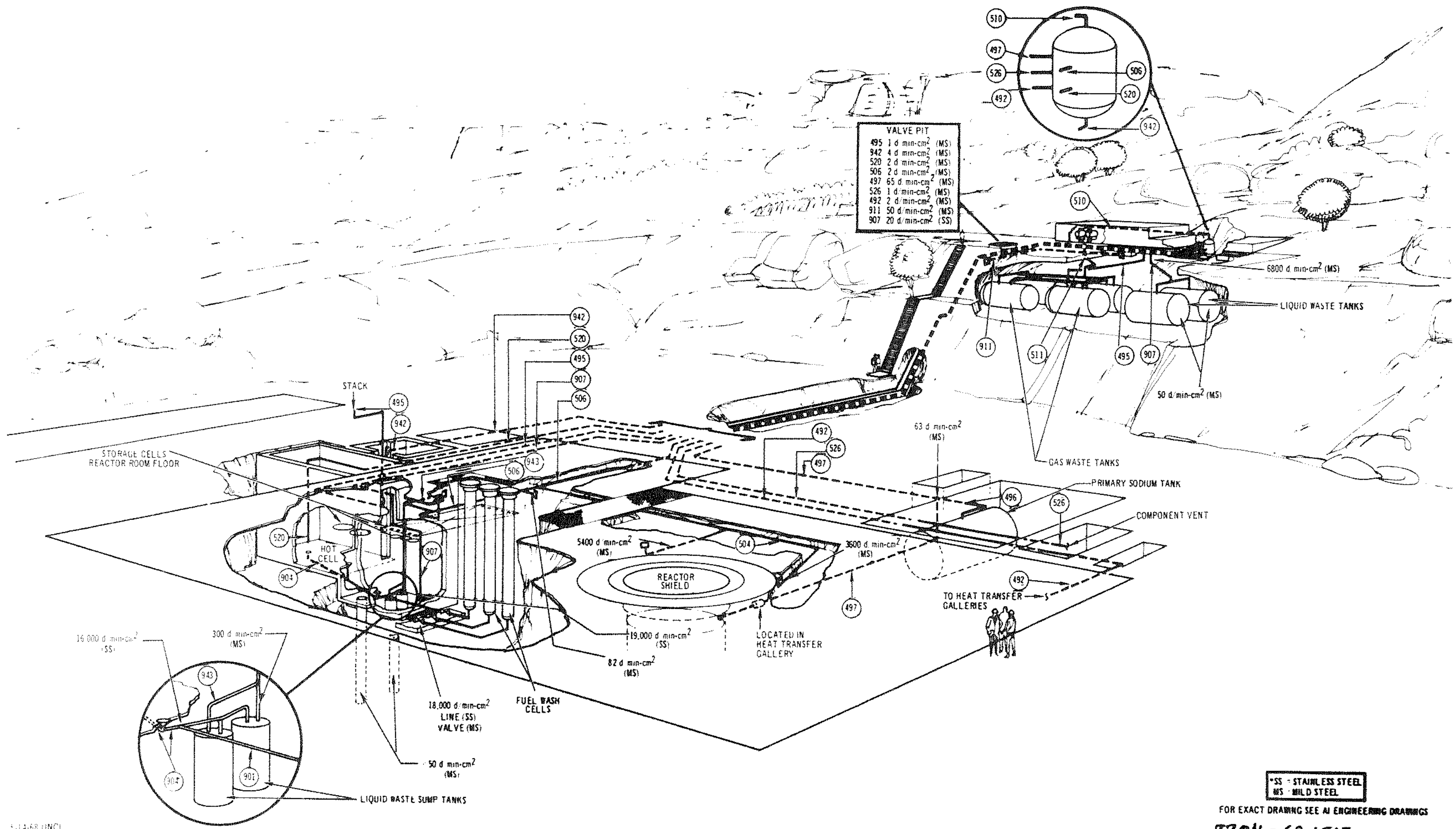

Figure 8. Radioactive Liquid Waste and Vent System

2.1.7.11 Hot Cell

The hot cell was used to remotely inspect and modify irradiated core elements and to can used fuel clusters for transfer to shielded transport casks. The facility consisted of two adjacent hot cells (metallurgical and main), an operating area, a service area, a shower, and a change room located below the level of the reactor high-bay floor, as shown in Figure 9.

Each of the areas was isolated by access doors. Access to each cell interior was through heavily shielded doors which rolled out into the service area. An L-shaped access tunnel to the main cell provided a means to place tools and small parts into the hot cell from the service area without exposure. Two airlock doors to the shower and change rooms separated the possibly contaminated service area from the clean operating area. All operations in the cell were performed with manipulators and remotely operated equipment by means of controls located in the operating area. The cell interiors were kept at a negative pressure of -0.5-in. water. The service and operating areas were maintained at 0.25-in. water pressure. Air flow directions were from these areas into the cells.

2.1.7.12 Miscellaneous Areas

Additional peripheral areas associated with the SRE that contained radioactive contamination were as follows:

Building	Description
041	SRE Component Storage Building
163	Site Service Building
653	Liquid Radioactive Waste Vault
654	Interim Radioactive Waste
686	Temporary Hot Waste Storage
724	Contaminated Sodium Cleaning Building
773	Drainage Control Dam

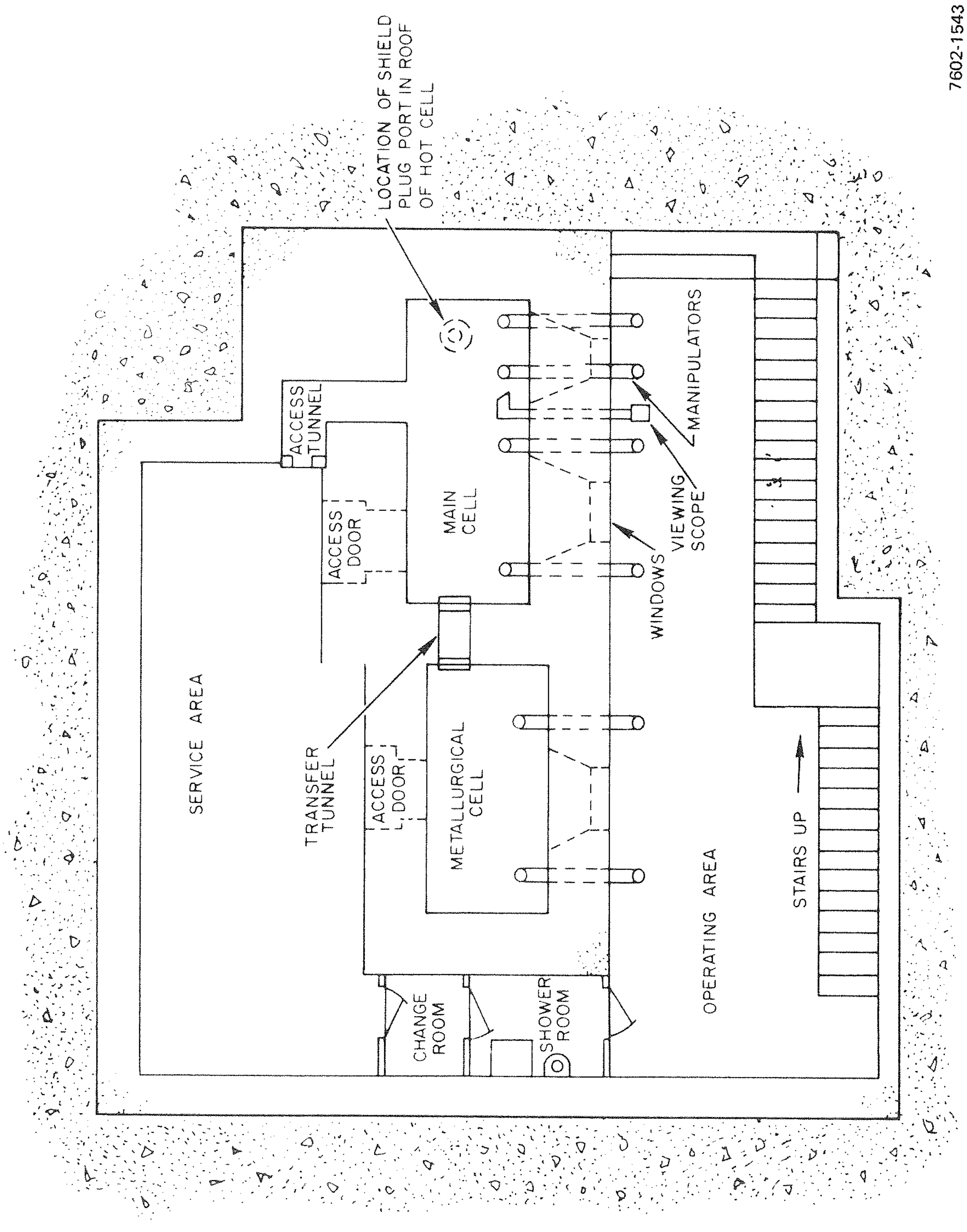

Figure 9. SRE Hot Cell Facility — Plan View

2.2 PRE-DECOMMISSIONING STATUS

The nuclear reactor operated from 1957 to February 1964. It was then shut down for major changes to increase the power rating from 20 MW to 30 MWt and to upgrade systems and components. This Power Expansion Program (PEP) was completed in 1965. Nonnuclear operations of the reactor systems were resumed in May 1965. They were terminated in September 1967, and the primary and secondary sodium systems and the kerosene cooling system were drained. A retirement program prepared the SRE plant, at minimum cost, for an indefinite period of storage prior to disassembly. However, certain equipment was removed and shipped to other DOE sites. This program was completed in June 1968, and subsequently, regular surveillance of the site was conducted. Maintenance on the remaining equipment was performed only to the extent necessary to ensure the security of the facility and stored radioactive materials.

The following facilities contained radioactive materials:

Building	Description
041	SRE Component Storage Building
143	Sodium Reactor Experiment
163	Site Service Building
653	Liquid Radioactive Waste Vault
654	Interim Radioactive Waste
686	Temporary Hot Waste Storage
724	Contaminated Sodium Cleaning Building
753	Primary Fill Tank Vault
773	Drainage Control Dam

The following systems contained hazardous or dangerous material: the secondary sodium, the sodium service, and the kerosene coolant systems.

Radiological surveys were performed in 1966, about 2 years after the last nuclear operation. These surveys primarily reflected the cesium and strontium activity which would have decayed 20% prior to the initiation of decommissioning activities.

2.2.1 Fuel Assemblies

The reactor was defueled before the start of decommissioning activities. The shield plug and hanger assemblies were disassembled from the unirradiated Core III fuel and stored in the storage cells in the high-bay floor of Building 143 (Figure 4). All fissile material from the three cores was stored in the fuel storage vault at the Radioactive Materials Disposal Facility (RMDF) under the control of the Nuclear Materials and Waste Management Organization.

2.2.2 Core Components

The core components were stored in the reactor core.

2.2.3 Sodium Systems

The sodium systems were drained, except for residual sodium heels throughout the system, and were at ambient temperature. All heaters were turned off, but remained in operating condition.

Approximately 5500 lb of primary sodium was in the primary fill tank at ambient temperature. A map of the radiation levels from the tank and associated piping is shown in Figure 10. The contamination level on the surfaces of the vault was less than 50 dpm/100 cm^2. Figures 11 and 12 are the radiation maps for the main and auxiliary sodium systems. Figure 13 is a radiation map for the sodium services piping in the sodium service vault. The manway plugs were removed in the following areas: primary fill tank vault, main gallery, sodium service vault, and primary drain vault to permit access for surveillance. However, the main shielding plugs were kept over the remainder of these areas.

2.2.4 Radioactive Liquid Waste System

The contamination and radiation levels present in several locations within the liquid waste system and in areas associated with this system were

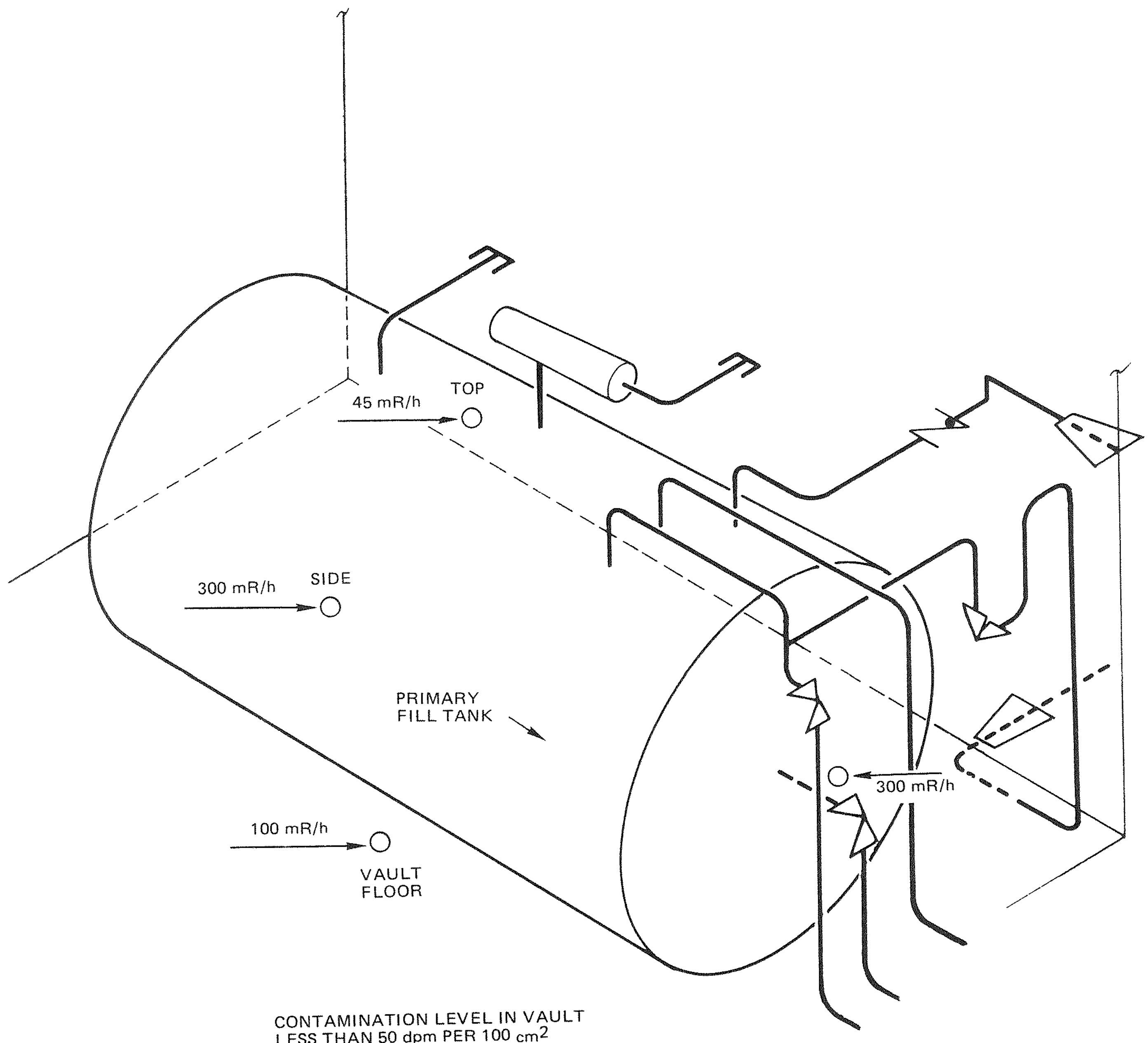

Figure 10. Radiation Levels from Tank and Associated Piping

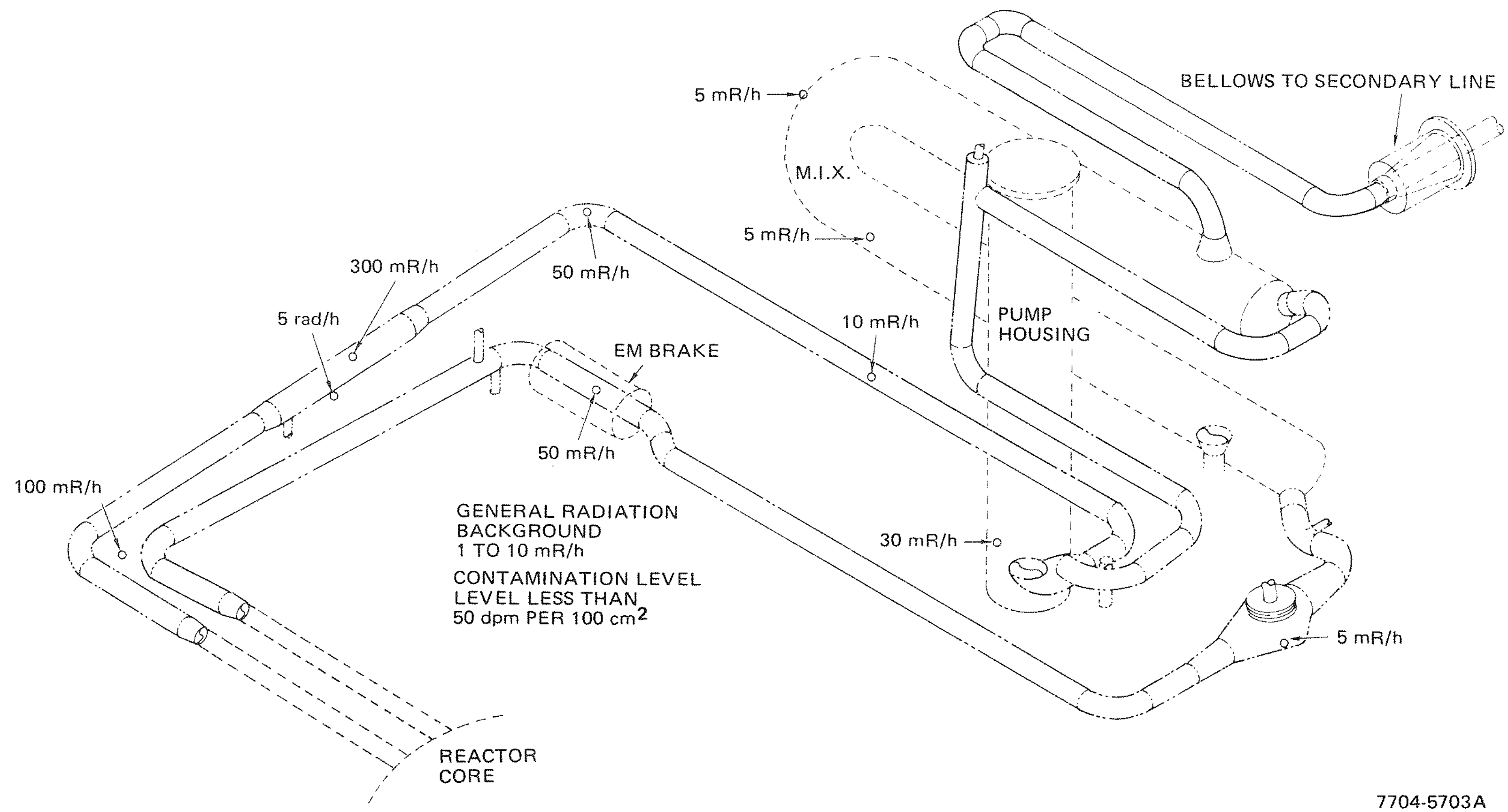

Figure 11. Radiation Map of the Main Sodium System

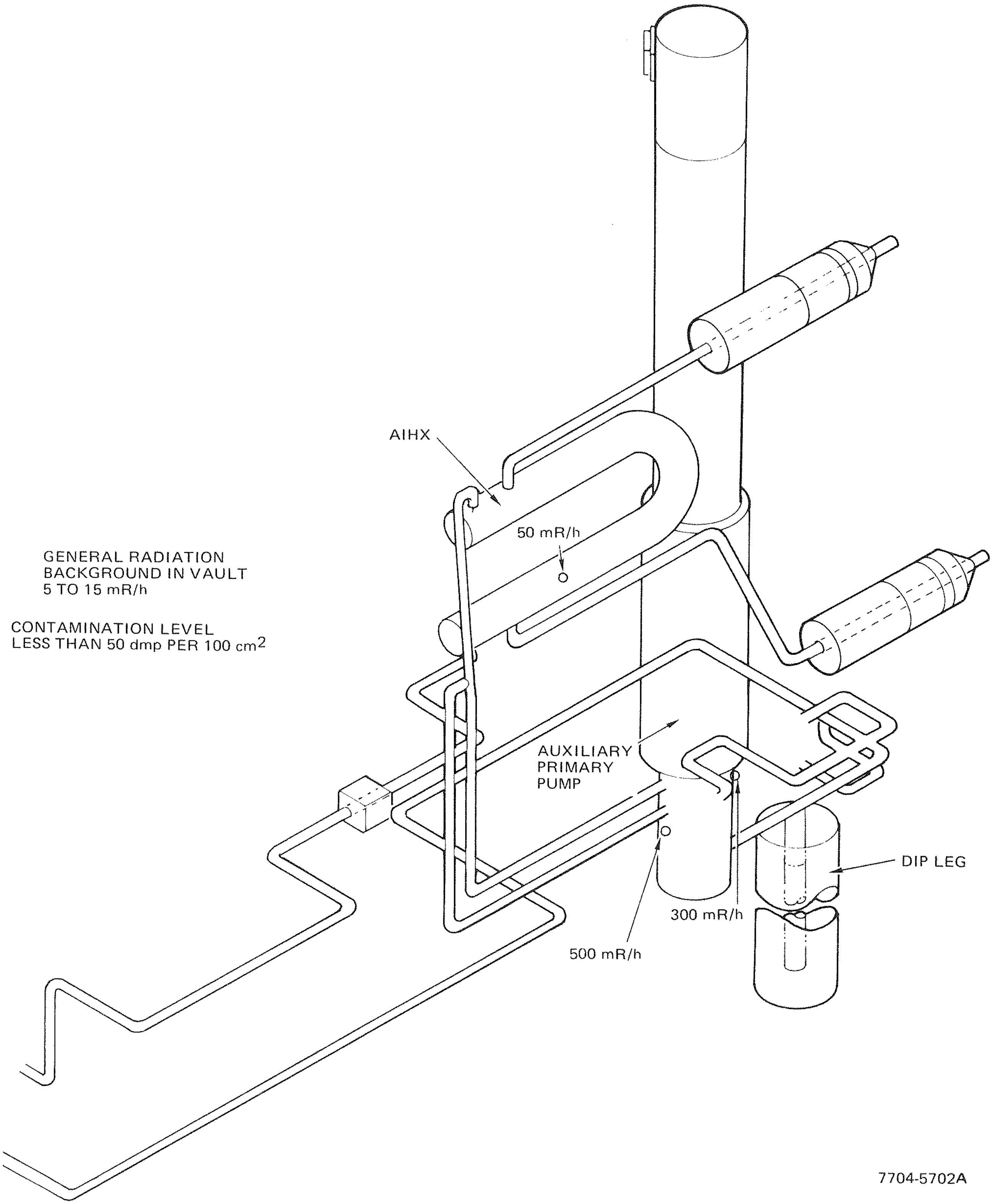

Figure 12. Radiation Map of the Auxiliary Sodium System

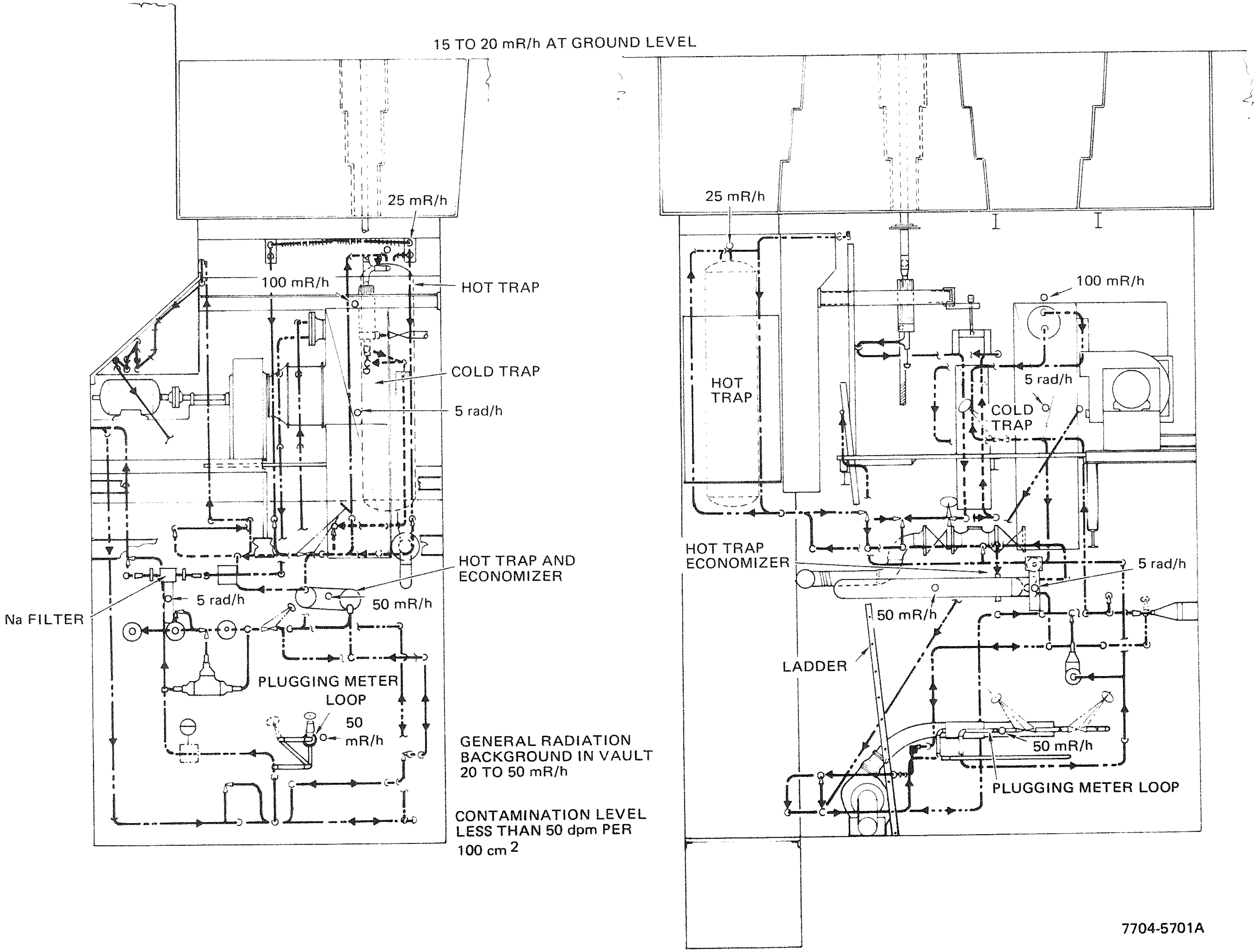

Figure 13. Radiation Map of the Sodium Services Piping

as indicated in Figure 8. All liquid waste, except inaccessible heels, was removed.

The two 5000-gal storage tanks in the obsolete liquid waste system had a radiation level of less than 5000 dpm/100 cm^2. The tanks were black iron and showed considerable internal pitting. Figure 8 presents contamination levels in the various liquid waste lines. Sometime during the operating history of the SRE, the liquid waste system produced a leak, and contaminated soil existed in the immediate vicinity of the tanks and along rock outcroppings in the cut bank below them.

The change room holdup tank contamination level was 500 dpm/100 cm^2. The activity inside the tank was 3 mR/h maximum.

2.2.5 Radioactive Gaseous Waste System

The levels of activity that were present in this system are indicated in Figure 8. The compressor vault was radiologically clean. The four gaseous waste storage tanks appeared to be leaktight and were sealed off under a nitrogen pressure of 1/4 psi.

2.2.6 Component-Handling Machines

The Mark I and II FHMs were stored in the FHM storage bay. The contamination that was present on this equipment at the exposed surfaces is shown in Table 5.

2.2.7 Fuel, Moderator, and Pump Storage Cells

The cells were empty, except for the plugs used to seal their entrances. The plugs in the fuel storage cells were the ones removed from the Core III fuel assemblies. During the reactor operation, the cells were exposed to a number of ruptured fuel assemblies and represented, other than the reactor

TABLE 5

CONTAMINATION LEVELS OF FUEL HANDLING MACHINES (1966)

Sample	Description and Location	β-γ Activity (dpm/100 cm^2)
1	Mark I FHM, bottom ledge	114
2	Mark I FHM, bottom ledge	15
3	Mark I FHM, bottom ledge	320
4	Mark I FHM, bottom ledge	69
5	Mark I FHM, bottom ledge	153
6	Mark I FHM, bottom ledge	117
7	Mark I FHM, top platform	36
8	Mark I FHM, top platform	39
9	Mark I FHM, superstructure	153
10	Mark I FHM, superstructure	138
1	Mark II FHM, control console	78
2	Mark II FHM, control platform	51
3	Mark II FHM, valve panel	Background
4	Mark II FHM, vacuum pump	36
5	Mark II FHM, O_2 analyzer panel	24
6	Mark II FHM, power panel	30
7	Mark II FHM, relay panel	24
8	Mark II FHM, bioshield	378
9	Mark II FHM, lower section	45
10	Mark II FHM, center section	105
1	Moderator cask, valve housing	81
2	Moderator cask, console	63
3	Moderator cask, lower base	27
4	Moderator cask, center section-S	57
5	Moderator cask, center section-N	33
1	Loading face spider	Background
1	Moderator cask strongback	Background
1	Loading face support bridge	Background
1	Long gas lock (lower section tagged)	Background

vessel, the most contaminated area on the site. Table 6 is a tabulated record of the radiation activity at the time the cells were sealed.

The fuel-cleaning cells were sealed, and the trenches surrounding them were equipped with temporary lead shielding. This area had a radiation level of about 1000 dpm/100 cm^2. During the nuclear operation period of the reactor, a strong chemical reaction occurred in the center cell while a spent fuel assembly was being cleaned. The cell, the surrounding floor area, and the earth surrounding the lower regions of the cell were contaminated.

The moderator top latch grapple was stored in the moderator-handling machine located in the SRE high bay.

The long shield plugs from Core III fuel were stored in the storage cells in the high-bay floor area.

2.2.8 Hot Cells and Ventilation System

The permanent A and B hot cells were below a contamination level of 500 dpm/100 cm^2 except for the two fuel storage thimbles, which were below a contamination level of 2500 dpm/1000 cm^2.

Contamination smear surveys on the interior areas of the ventilation ducts adjacent to each filter in the filter room showed beta and gamma contamination levels as given in Table 7.

2.2.9 Peripheral Areas

The west end of Building 163, Contaminated Equipment Repairs Facility (CERF), had contamination levels as shown in Table 8.

Building 724, the SRE Oil Cleaning Facility, was used extensively to remove contaminated sodium from pipes and miscellaneous sodium equipment. The contamination levels were as shown in Table 9.

TABLE 6

CONTAMINATION LEVELS IN SRE BUILDING 143 (1966)

Number	β-γ Activity (dpm/100 cm^2)	Number	β-γ Activity (dpm/100 cm^2)
Storage Cells			
1	10,200	68	40,800
2	4,200	69	1,400
3	11,400	72	9,600
42	14,600	74	6,600
43	3,600	75	15,600
44	1,600	78	1,800
45	4,000	79	2,000
48	5,400	80	3,800
49	4,200	81	7,000
50	2,500	83	7,200
51	3,800	84	7,800
53	4,200	85	6,000
54	9,600	86	45,400
55	12,000	87	11,400
56	52,500	90	15,600
57	6,600	91	15,000
60	11,400	92	30,000
61	11,400	93	5,400
62	2,500	94	6,600
63	5,000	96	7,800
64	7,200	97	8,400
66	4,000	98	4,400
67	1,600	99	11,400
Moderator Storage Cells		Pump Storage Cells	
A	1,980	East	780
B	1,440	West	500
C	990		
High Bay Floor			
Maximum β-γ level was 75 dpm/100 cm^2 with an average of 50 dpm/100 cm^2			

TABLE 7

ACTIVITY LEVELS OF SRE VENTILATION SYSTEM (1966)

Sample	Description and Location	β-γ Activity (dpm/100 cm^2)
1	West duct, upstream of filter	2,334
2	West duct, downstream of filter	129
3	Center duct, upstream of filter	10,181
4	Center duct, downstream of filter	1,293
5	East duct, upstream of filter	756
6	East duct, downstream of filter	423
7	East plenum floor, under filter	1,953
8	Center plenum, under filter	1,479
9	West plenum floor, under filter	2,118

2.2.10 Reactor Cavity

The most highly contaminated part of the facility was the reactor cavity. Radiation measurements were taken during decommissioning in 1977. Radiation levels as high as 100 R/h were recorded. It was estimated that 12 to 14 ft of water would be required to serve as shielding during the removal operation. This structure contained a 1-1/2-in. radioactive sodium heel on the bottom of the vessel. Sodium also adhered to the sides and top of the moderator cans and other equipment stored in the vessel.

TABLE 8

CONTAMINATION LEVELS OF BUILDING 163 (1966)

Description and Location	β-γ Activity (dpm/100 cm^2)
South floor, west	30
South floor, center	30
South floor, east	30
Center floor, west	30
Center floor, center	112
Center floor, east	30
North floor, west	30
North floor, center	30
North floor, east	87
East wall, north	87
East wall, center	87
East wall, south	30
North wall, west	30
North wall, center	30
South wall, east	130
South wall, west	30
South wall, center	30
South wall, east	30
West wall, south	30
West wall, center	30
West wall, north	87
Light fixtures, northeast	187
Light fixtures, northwest	112
Light fixtures, west	70
Light fixtures, east	87
Light fixtures, southeast	30
Light fixtures, southwest	30
Supply room overhead crane rails, top	300
Crane rail, south, first sample	266
Crane rail, south, second sample	252
Crane rail, north, first sample	294
Crane rail, north, second sample	185

TABLE 9

CONTAMINATION LEVELS OF SRE OIL CLEANING FACILITY, BUILDING 724 (1966)

Description and Location	β-γ Activity (dpm/100 cm^2)
Outside areas	<30
Inside areas	
Floor, northeast	150
Floor, northwest	<30
Floor, southeast	110
Floor, southwest	100
Walls, south	<30
Walls, north	<30
Walls, west	<30
Walls, doors	<30
Trench	
South	115
North	130
Angle iron, west	440
Angle iron, east	<30

3.0. DECOMMISSIONING OBJECTIVE AND WORK SCOPE

3.1 OBJECTIVE

Deactivation of the SRE facilities temporarily provided a safe storage condition. Because of the potential for release of radioactive material into the environment and because of the continuing cost for surveillance and maintenance, the government initiated a program to decommission the SRE facilities. Dismantlement was chosen as the decommissioning mode. This mode would dispose by land burial all contaminated material and would remove or decontaminate all contaminated structures, components, and areas.

Thus, the objective of the SRE decommissioning project was to remove radioactive material from the site as necessary to release the site from all requirements for radiological control, licensing, or monitoring.

3.2 WORK SCOPE

The SRE decommissioning project included planning, development and test, dismantlement operations, radiation control, waste management, quality assurance, and supporting activities as necessary to accomplish the project objectives.

3.2.1 Planning

Engineering studies were conducted to define and describe the work needed and the best method for performing and controlling the work. These studies produced the following planning documents:

1) Program Plan, which is the top-level guidance document for stating objectives and describing the manner of performance of the decommissioning program
2) Quality Assurance Plan, based on the general requirements of the AEC manual, Chapter 0820, and updated to DOE Order 5480-1

3) Operational Safety Plan, which delineated the radiation safety, industrial hygiene, and industrial safety procedures for the decontamination and disposition of the SRE reactor systems

4) Training Plan, which described the training activities to be performed to assure that all employees engaged in the SRE decommissioning received radiation and nuclear safety indoctrination, SRE facility familiarization, emergency procedure training, and specific training on the operation and use of equipment

5) SRE Dismantling Plan, which described the site conditions at the beginning of the decommissioning program, established a radiological characterization of the site based on survey data and analysis, and defined the tasks to be performed. The magnitude of the SRE dismantling required that the dismantling activities be subdivided into separate manageable tasks designated as "activity." An activity requirements document was prepared for each of 27 tasks listed in Table 10.

3.2.2 Development and Test

Engineering studies conducted in support of the planning documents revealed the need for specialized tooling and techniques to perform the decommissioning tasks safely. Disposal of the highly radioactive reactor vessels required the development or adaptation of special tooling. An existing Oak Ridge National Laboratory torch-manipulator designed for use on the Elk River reactor dismantling program was modified and tested.

The Elk River plasma-arc manipulator design was modified to fit the SRE reactor geometry. A full-scale mockup of the concentric SRE reactor vessels was constructed in the engineering test building near the SRE. A major development of the manipulator was the design, fabrication, and test of the manipulator capability to cut the reactor vessel radius located at the junction of the vessel walls and the bottom. Cutting parameters such as rate of cut, arc amperage, and arc length were determined for application on the radioactive

TABLE 10

SRE DISMANTLING AND DISPOSITIONING ACTIVITY REQUIREMENTS

Activity	Title
1.0	Remote Tooling for Removal of SRE Vessels
2.0	Primary Sodium Disposal
3.0	Reactivation of Contaminated Equipment Repair Facility, Building 163
4.0	Reactivation of Contaminated Components Cleaning Facility, Building 724
5.0	Removal of Primary Sodium Components in the Main and Auxiliary Pipe Galleries
6.0	Removal of Secondary Sodium Components in the Main and Auxiliary Pipe Galleries
7.0	Removal of Primary Sodium Components from the Service Vault
8.0	Dismantling of Sodium Service System in Building 153
9.0	Passivation of Residual Sodium in the Reactor Vessel
10.0	Removal of Reactor Internals
11.0	Component Cleaning in Building 163
12.0	Component Cleaning in Building 724
13.0	Removal of Reactor Vessels
14.0	Decontamination of Primary Fill Tank Vault
15.0	Decontamination of the Pipe Galleries
16.0	Decontamination of Hot Cell Facilities
17.0	Removal and Decontamination of the Storage and Wash Cells
18.0	Decontamination and Dismantling of Mark I FHM
19.0	Decontamination and Dismantling of Mark II FHM
20.0	Decontamination of Moderator-Handling Machine
21.0	Removal of Activated Concrete
22.0	Removal of Inert Gas System
23.0	Disposal of Radioactive Waste Systems
24.0	Decontamination of Building 163
25.0	Decontamination and Dismantling of Building 724 and Pad 723
26.0	Decontamination and Dismantling of Facilities at Site 686
27.0	Decontamination and Fill of the Retention Pond and Dam 773

material of the actual reactor vessel. Special tools such as pry bars, spacers, and grapples were developed to support the manipulator operation.

Removal of the highly radioactive reactor vessel internal piping required the development of an explosives cutting technique. This development was a joint contractor/Rockwell effort. Shaped charges for circumferential and longitudinal cuts were designed, constructed, and tested. The optimum explosives quantities for cutting specific pipe and other reactor vessel internals were established. Techniques and tools for application of the charges to the underwater vessel internals were developed.

Concrete surface decontamination required development and testing of existing commercial devices and techniques. The development primarily concerned application of these devices to the special problems of limiting the spread of contamination, working in limited access, and effectively accomplishing the decontamination. Scabblers, chipping hammers, jackhammers, sandblasters, and spalling tools were tested.

Techniques for decontaminating painted surfaces by using solvents and foams were developed. Development of the foam technique was necessary to accommodate the restrictions associated with disposal of contamiated liquid waste. The foaming technique uses very little liquid and is effective in lifting loose contamination from surfaces. The use of a vacuum system to pick up the foam after application was a significant improvement in the use of foams.

3.2.3 Dismantlement Operations

The decontamination and dismantlement work scope consisted of the following operations:

1) Removing peripheral systems, primarily noncontaminated, nonsodium-containing systems such as the kerosene cooling system, the nitrogen pipe gallery cooling system, and water tank

2) Disposing primary sodium from storage tank and the residual sodium in the reactor
3) Removing sodium system components such as the pumps, heat exchanger valves, hot traps, and cold traps
4) Decontaminating and dismantling the hot cells, components, and structure
5) Disposing of reactor vessels and internals
6) Demolishing and disposing of contaminated concrete such as the biological shield and shield plugs
7) Disposing of radioactive waste handling systems such as the gaseous and liquid waste holdup system
8) Excavating contaminated soil and bedrock
9) Packaging equipment and waste and shipping it to burial
10) Rectifying the site, including pavement and floor repairs, lighting replacement, painting, and wall repairs.

3.2.4 Radiation Control

The Radiation & Nuclear Safety unit was responsible for establishing design and operational procedures for disposing of source and special nuclear materials and byproduct radioactive material; designating and identifying areas to be radiologically posted; taking field measurements of radiation and radioactive contamination levels; evaluating internal and external personnel radiation exposures; and evaluating radioactive material concentrations in effluents and in the environment surrounding the facility. In addition, Radiation & Nuclear Safety was responsible for maintaining records necessary to demonstrate compliance with ESG standards and applicable state and federal regulations. Included was a chronological log of information dealing with daily operations, conditions, and occurrences relating to radiological safety.

Administrative and physical radiation controls were instituted to minimize both the release of contamination and the exposure of working personnel to radiation. The details of radiation control are defined in each detailed procedure and generally in the Health and Safety Operating Procedures. The procedures described the actions necessary for radiation control, including such actions as:

1) Erecting containment structures to limit the spread of contamination, particularly airborne contamination
2) Using water sprays to settle contaminated dust
3) Constructing, installing, and using radioactive exhaust systems to flame or arc cut contaminated materials
4) Designating areas as contaminated and limiting access to personnel; establishing step-off areas, change rooms, and waste holdup areas
5) Using protective clothing, air-breathing apparatus, and dosimetry for personnel
6) Continually surveying working areas, packaged equipment, and waste shipments.

3.2.5 Waste Management

The Nuclear Materials Management unit provided guidelines for packaging and shipping, based on DOE, DOT, and burial site requirements; verification, along with Quality Assurance, that guidelines were followed; maintenance of waste packaging and shipment records; and liaison with government agencies and burial sites on changing requirements in waste disposal.

The waste management work scope handled by the Radioactive Materials Disposal Facility included the activities associated with the furnishing waste containers, boxes, drums, and casks; preparing containers for waste handling; arranging for shipment and burial; and packaging and shipping. In addition to radioactive waste, asbestos and sodium wastes were processed by waste management. Radioactive liquid wastes, which could not be buried as a liquid, were processed by evaporation or by solidification in cement or a similar medium.

3.2.6 Project Management and Support

The management work scope included activities such as reporting, cost control, customer interfacing, recordkeeping, review, approval of documents, coordination of engineering, manufacturing, quality assurance, health and safety, traffic, photography, business administration, and contracts.

Project management generally defined the work scope, prepared cost estimates, schedules, expenditure plans, and designated the kind and level of support required from the participating departments. The rate of expenditures and conformance to schedule were monitored and adjusted to accommodate problems as they were encountered.

3.2.7 Quality Assurance

The Quality Assurance Program Plan was based on the requirements of the AEC Manual, Chapter 0820, and the updated DOE Order 5480-1. The primary objective of the plan was preserving the health and safety of the decommissioning personnel and the general public and protecting the environment. This objective was accomplished by reviewing all documents generated for the program, participating in all design reviews, and conducting periodic audits to verify compliance with all procedures used during the decommissioning. The Quality Assurance Department also verified that personnel had received the necessary radiation safety training prior to the commencement of work activities and, through audits, verified that radiation-detection instruments were calibrated correctly. In addition, Quality Assurance verified that all radioactive waste was properly identified and packaged according to applicable requirements. Final radiological survey sampling plans and results were reviewed and approved by Quality Assurance.

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4.0 WORK PERFORMED

4.1 PROGRAM AND PROJECT MANAGEMENT

The SRE decommissioning was administered by the SFMPO of DOE-RL working through DOE-SAN, who managed ESG's activities on the project. ESG established a program office to manage the implementation of the project beginning with the preparation of the top-level guidance and project plans and concluding with the final report and film documenting the SRE decommissioning. A document flow chart is shown in Figure 14.

A program plan described the task and delineated the objectives of the program. In addition, it described the procedures to be used for cost and schedule control and reporting, purchasing and subcontract control, and program and engineering data control. Requirements for the quality assurance plan, operational safety plan, training plan, dismantling plan, activity requirements, and detailed work procedures were also presented.

The ESG program office acted as liaison with the DOE representatives who monitored the project and with all organizations that were involved during the performance of the project. The program office was also responsible for the overall schedule and budget performance and for the submission of the schedules and budgets. A performance control system (PCS) was used to monitor progress and to initiate corrective action when necessary.

All reporting to DOE and its delegated representatives was done by the program office, including the monthly, annual, technical, and final reports.

4.2 PROJECT ENGINEERING

Project Engineering, within ESG, followed the guidance of the program plan and prepared the necessary documents to accomplish the physical decommissioning of the SRE. The top-level document prepared by Project Engineering was the "Facilities Dismantling Plan For SRE." The second-level documents

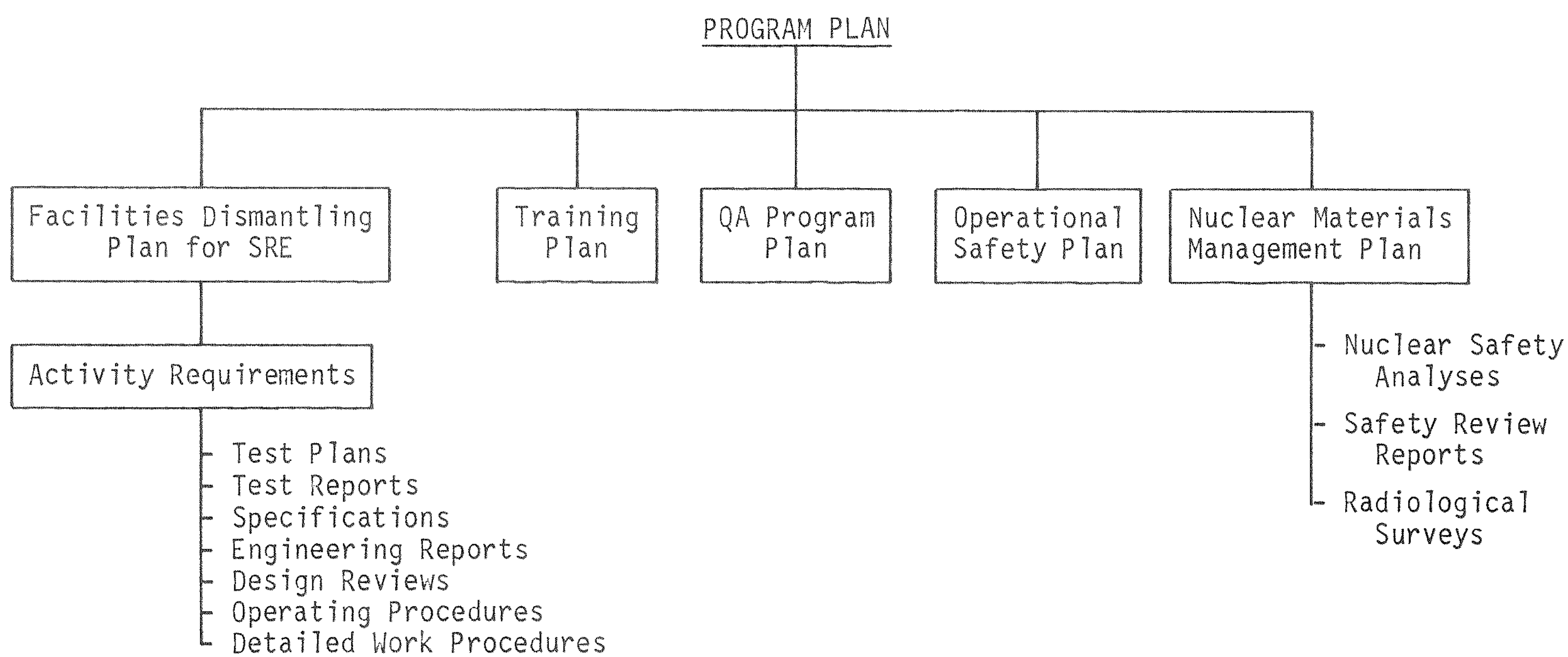

Figure 14. Document Flow Chart

were the Activities Requirements. To satisfy the activities requirements, many subservient documents were prepared, including test plans and reports, specifications, design reviews and reports, operating procedures, and detailed work procedures that were used to direct craftsmen performing the physical work.

Project Engineering was also responsible for developing techniques to be used during the decommissioning of the SRE, including adapting and refining the underwater plasma-torch segmentation technique for the reactor vessels, explosive cutting techniques for piping, and adapting the alcohol passivation technique to permit disposal of the residual sodium in the reactor system.

Project Engineering, acting through the ESG Engineering Department, was responsible for the technical adequacy and completeness of documents prepared as the program progressed. Day-to-day problems dealing with the dismantlement activity were also handled by Project Engineering.

Project Engineering acted as liaison with the Engineering Department in obtaining support for manipulator design, structural design, temporary building support design, and in obtaining support for the monitoring of subcontracted efforts such as earth moving and excavation and shoring wall construction.

4.3 SITE PREPARATION

The SRE facility had been in a maintenance and surveillance mode since September 1967. To support decommissioning activities at the site, the reactivation of various subsystems was required. In addition, new materials and equipment had to be procured prior to the start of work.

The following outline identifies the significant activities performed as part of the site preparation.

4.3.1 Equipment Reactivation and New Materials

1) Reactivate utility and convenience services to support limited office occupancy and the demolition activity
2) Inspect and reactivate the radioactive gas and liquid waste systems to support the demolition activity
3) Reactivate the liquid nitrogen gas system to restore full capacity
4) Procure special equipment needed for rotating the top shield
5) Design and procure a special water circulation and filtration unit to support the reactor cavity demolition
6) Design and fabricate one-way-approved shipping casks
7) Prepare specifications and procedures for each task
8) Procure materials and equipment necessary to begin decontamination activities
9) Reactivate facility cranes
10) Establish a health, safety, and radiological services office and analysis laboratory onsite
11) Upgrade facility as necessary (i.e., repair leaking roof, replace air conditioner, install new hot water heater)
12) Reactivate site security (i.e., repair perimeter fence, replace door and gate locks.

4.4 DECOMMISSIONING OPERATIONS

4.4.1 Noncontaminated Peripheral Systems Removal

Prior to reactor dismantlement, noncontaminated peripheral systems at the SRE were removed. These included the kerosene cooling system, nitrogen gallery cooling system, secondary sodium system, air blast heat exchangers, process water tank and piping, vault cooling system, sodium service building, and steam and electrical generation facilities (see Figure 3).

Removal of the peripheral systems, except for items that required cutting into sodium piping, was accomplished by a salvage contractor. The arrangement with the salvage contractor was no cost; the contractor received the salvage material in exchange for the labor of removal. Equipment and material usable on other programs or potentially usable on the SRE decommissioning program were set aside. Health, Safety & Radiation Services personnel surveyed all equipment and materials for radioactive contamination prior to release from the site.

4.4.2 Primary Sodium Disposal

At the start of dismantlement, approximately 7400 gal of sodium was stored in the primary fill tank (PFT) under a 1.0-psig nitrogen cover gas. This sodium was slightly radioactive. Figure 10 shows the maximum radiation levels on the surface of the tank.

A piping system was fabricated to facilitate draining the sodium from the primary fill tank into 55-gal drums. A differential pressure between the primary fill tank cover gas and the 55-gal drum cover gas was used to transfer the sodium into the drums. A total of 158 drums containing 55,000 lb of slightly radioactive sodium were shipped to Hanford, Washington, for storage and future use.

4.4.3 Residual Sodium Passivation

Sodium passivation required a reaction process that could be well controlled and easily monitored for completion. Ethyl alcohol was the reactant selected to convert sodium to passive compounds.

Alcohol was selected over the water vapor/nitrogen process for the following reasons:

1) Safer reaction of large-bulk sodium pools, eliminating the possibility of explosive reactions of water vapor condensate and sodium

2) Controllable reaction rate, depending on the alcohol temperature
3) Reduced possibility of melting the Cerrobend seal between the loading face shield and the reactor vessel during cleaning, since a hot gas was not required
4) Less expensive to design, install, and operate.

4.4.3.1 PFT Passivation

Passivation reaction parameters had previously been developed in reaction rate studies that investigated temperature, geometry, and orientation of the sodium-alcohol reaction interface. The PFT was the first sodium system component at the SRE to be passivated.

The PFT was of simple interior geometry, 119 in. ID by 170 in. long with approximately 604 ft^2 of surface area. Little sodium was visible. It was passivated by spraying the interior with alcohol through a multihole nozzle. The large reacting surface area produced a rapid temperature and pressure rise. When the pressure reached 4.5 psig, a thimble weld cracked and released much of the generated pressure. No radioactivity was released.

4.4.3.2 Reactor Vessel Passivation

Prior to passivation, a visual inspection of the reactor internals with a TV camera revealed significant sodium frost deposits — to 1 in. thick — above the previous sodium pool level. The below-pool-level surfaces were well drained and had little adhering sodium. Residual sodium in the bottom of the reactor vessel was measured and found to be 1.25 in. deep. Passivation was necessary, since the vessel would be water filled for radiation shielding during plasma cutting of the core tank and associated internals. Also, the moderator cans and loose internals had to be sodium free prior to burial since federal regulations precluded burial of metallic sodium.

Sodium had been removed from the reactor bottom by a vacuum technique several times during operation of the SRE. The vacuum system consisted of a

5-hp vacuum pump, a 170-gal stainless steel catch tank, a 28-ft-long vacuum nozzle, a sodium vapor trap, and a modified fuel plug to maintain inert atmosphere (see Figure 15). Approximately 40 gal of sodium were vacuumed off the reactor bottom with this system.

An alcohol piping system for passivating the SRE core tank was then built, using two solvent pumps and the PFT as the alcohol supply tank. All reactor lines were connected to the piping. Four lines (three directional spray nozzles and one dump line) penetrated the main 140-in.-diameter top shield plug. Pump flow rates of 35 gpm could be achieved. Figures 16 and 17 show the system during operation.

Instrumentation included five immersion thermocouples above the moderator cans, seven original core tank vessel thermocouples, two alcohol level detectors, two variable-pressure trip solenoid valves on the vent line, and a thermal conductivity-type gas chromatograph with a sampling pump, also on the vent line. Two multipoint recorders produced the thermocouple printouts.

The sodium heel and grid plate were passivated by adding small quantities of alcohol onto the sodium surface. This permitted accurate process control of heat and hydrogen generation. Additional passivation of the reactor internals was accomplished by controlled flooding of the moderator cans and spraying the upper reactor wall and hanger rods with alcohol. A total of 410 lb of sodium was reacted, using 2500 gal of alcohol in 535 h of operation.

Passivation of the reactor was completed with the following exceptions: a short rumble was heard when the moderator coolant header was rinsed with water, and bubbles were seen when both the moderator cans and grid plate standoff bolts were unseated. The latter cases were caused by hydrogen blanketing of the sodium suface.

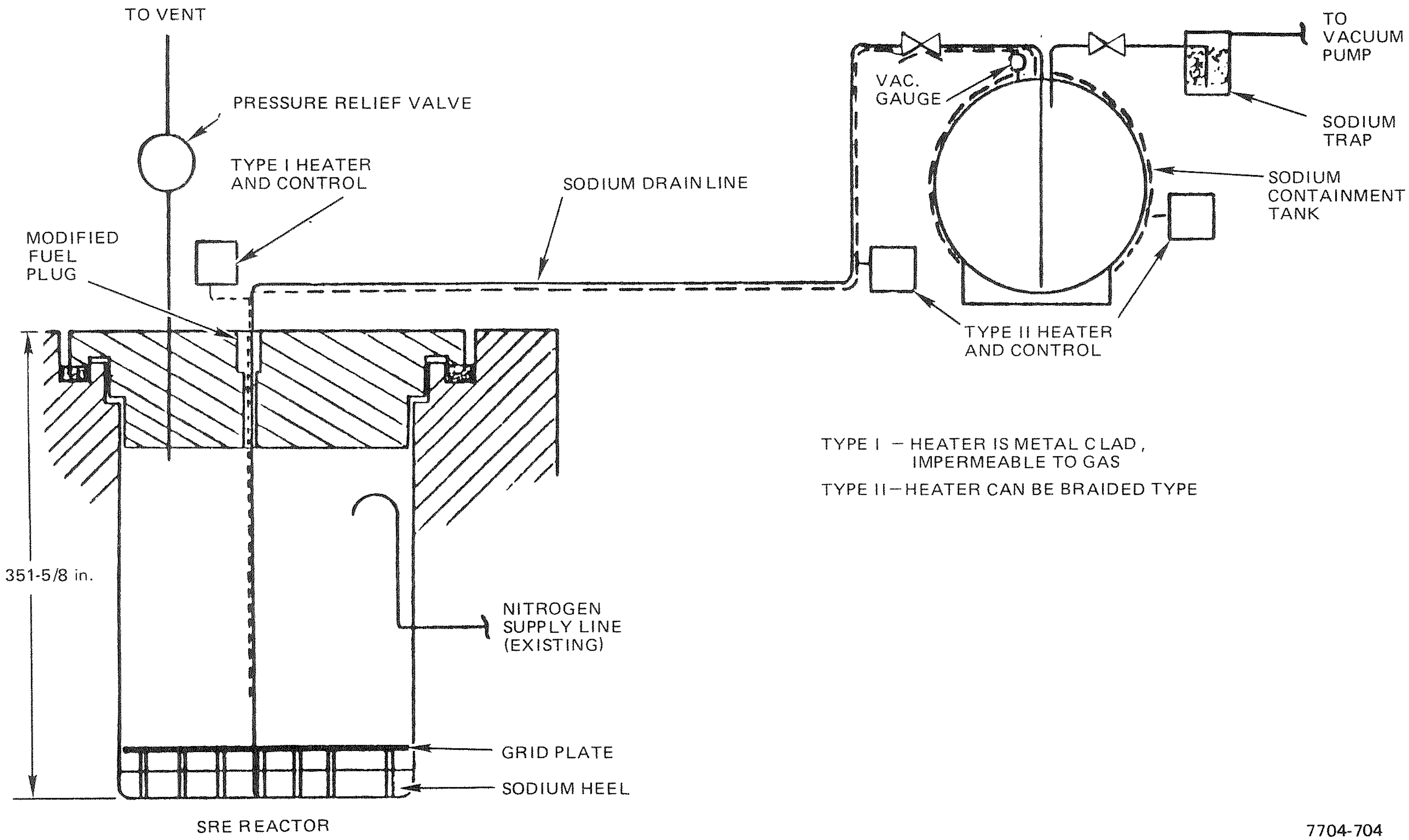

Figure 15. Sodium Vacuuming System

Figure 16. Reactor Passivation Piping During Installation

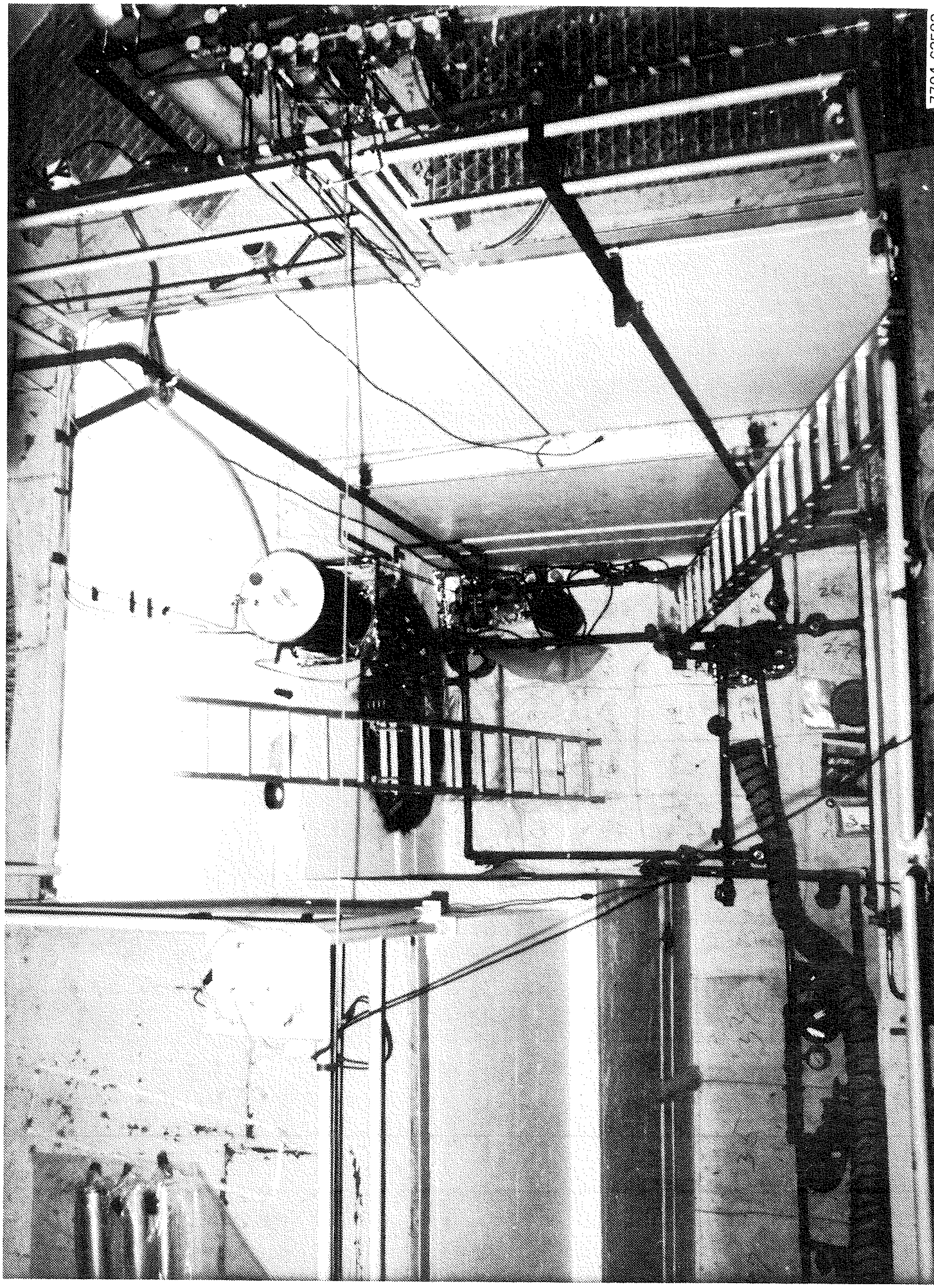

Figure 17. Reactor Passivation Piping in Primary Piping Vault

4.4.3.3 Sodium System Component Passivation

Passivation of sodium system components was accomplished with two separate systems: a small passivation system and a large passivation system.

The small passivation system consisted of a 175-gal alcohol supply tank, 3/8-in. carbon steel tubing, an air-driven pump, and Swagelok-type stainless steel valves. All system connections were metal-to-metal, compression type, which had proved to be leakproof. Figure 18 is a sketch of the system.

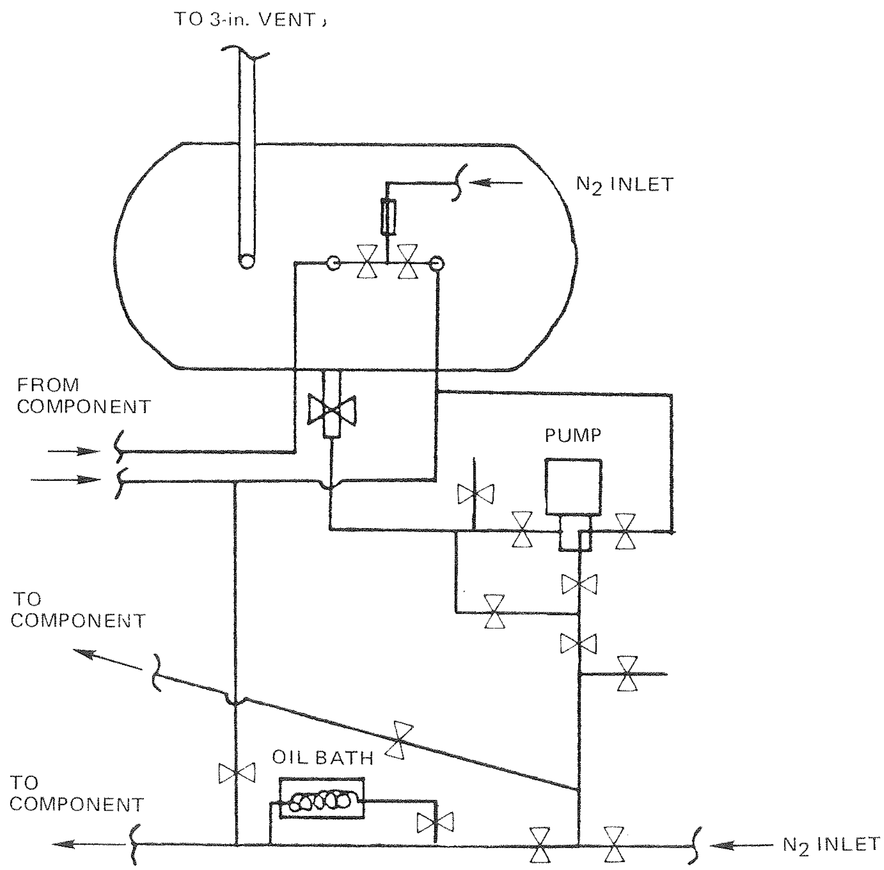

Figure 18. Small Component Sodium Passivation System

Under controlled conditions, large quantities of sodium (50 lb) could be reacted with the small-capacity system.

The large passivation system used 1-1/2-in. pipe, one of the previously used alcohol pumps, the PFT as the alcohol supply tank, and a 3-ft by 8-ft flanged vessel as an immersion tank (the passivation vessel). All generated hydrogen was entrained with the return alcohol to the PFT, which was vented by 3-in. tubing to the radioactive exhaust duct. Reaction rates could be easily observed from the temperature recorder and hydrogen concentrations in the vent gas. The system was installed in one section of the primary pipe vault (see Figure 19). The greatest amount of sodium, more than 1800 lb, was reacted during this phase.

Details of the reactor vessel and sodium system component passivation can be found in ESG Technical Report N704TR990007, "Report on Passivation of the SRE Reactor Vessel and Associated Components."

4.4.4 Radioactive Sodium System Component Removal

The heat transfer system for the SRE consisted of four loops: a main primary and a main secondary loop and an auxiliary primary and an auxiliary secondary loop. The main secondary and the auxiliary secondary loops contained no radioactivity and were removed as part of the peripheral systems dismantling. The main primary and the auxiliary primary loops were contaminated. Additional contaminated sodium system components were located in the sodium service vault.

A radiological survey was conducted in the sodium system areas. The point at which the survey was made is indicated in Figure 20 for the main primary loop, Figure 21 for the auxiliary primary loop, and Figure 22 for the sodium service vault. The survey results are tabulated below.

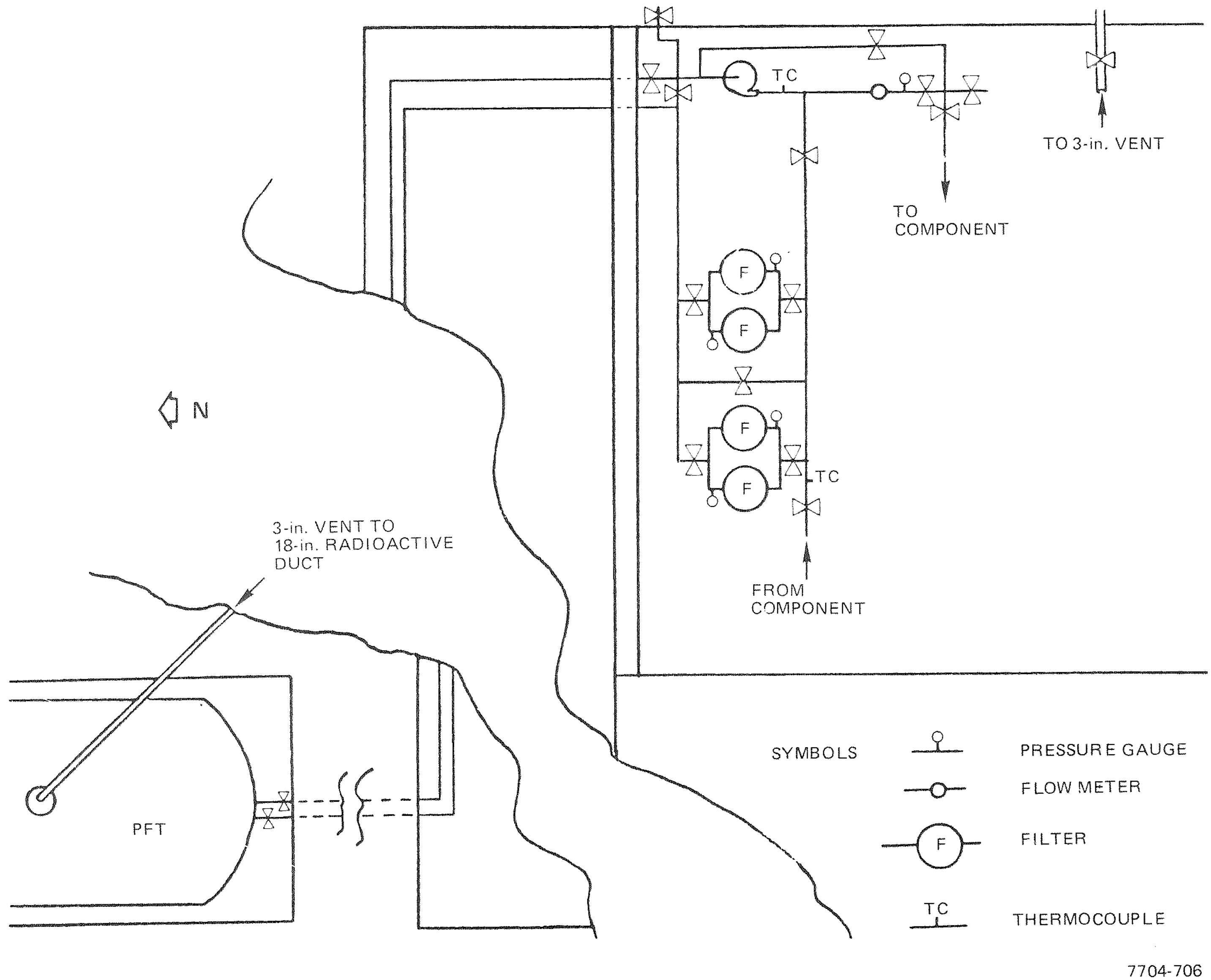

Figure 19. Large Component Sodium Passivation System

Main Primary Gallery Radiological Survey

(See Figure 20)

Area	mR/h
1	30
2	5
3	10
4	5
5	5
6	50
7	50
8	300
9	5,000 at 2 ft, 22,500 at contact
10	100
11	1 to 10

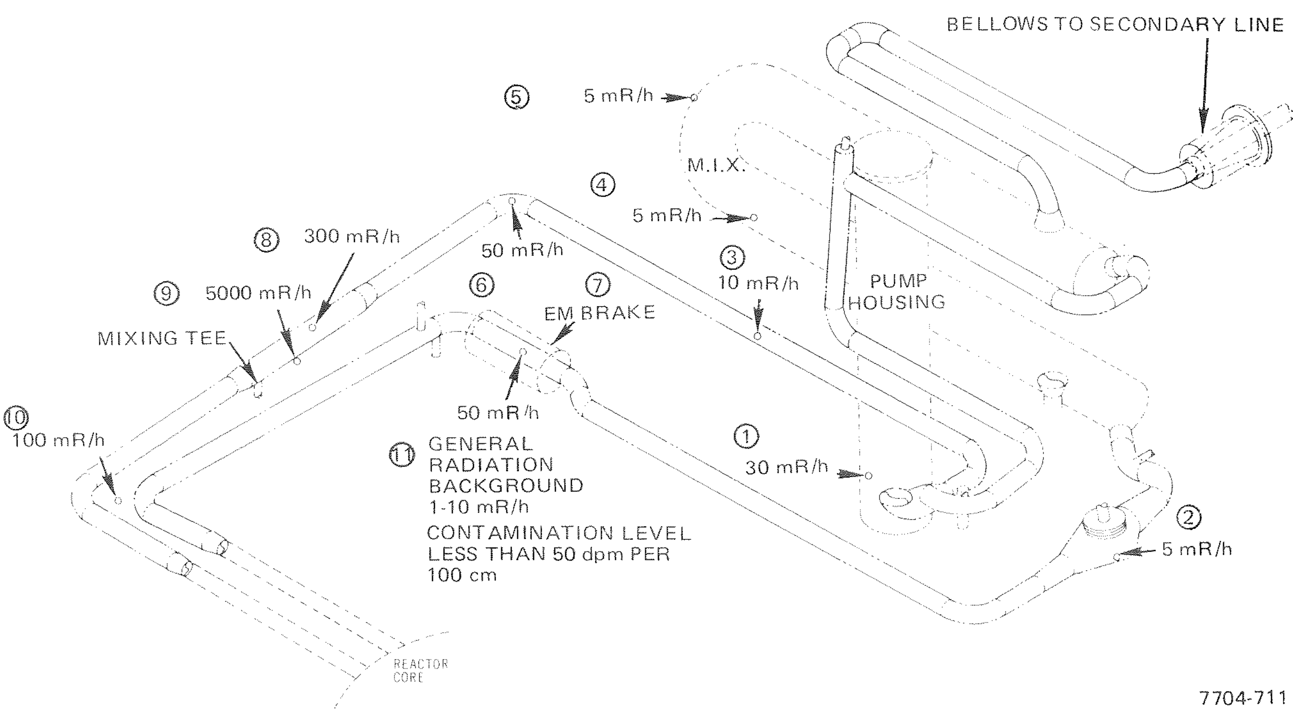

Figure 20. Main Primary Loop Radiation Levels

Auxiliary Primary Gallery Radiological Survey

(See Figure 21)

Area	mR/h
1	500
2	300
3	50
4	5 to 15

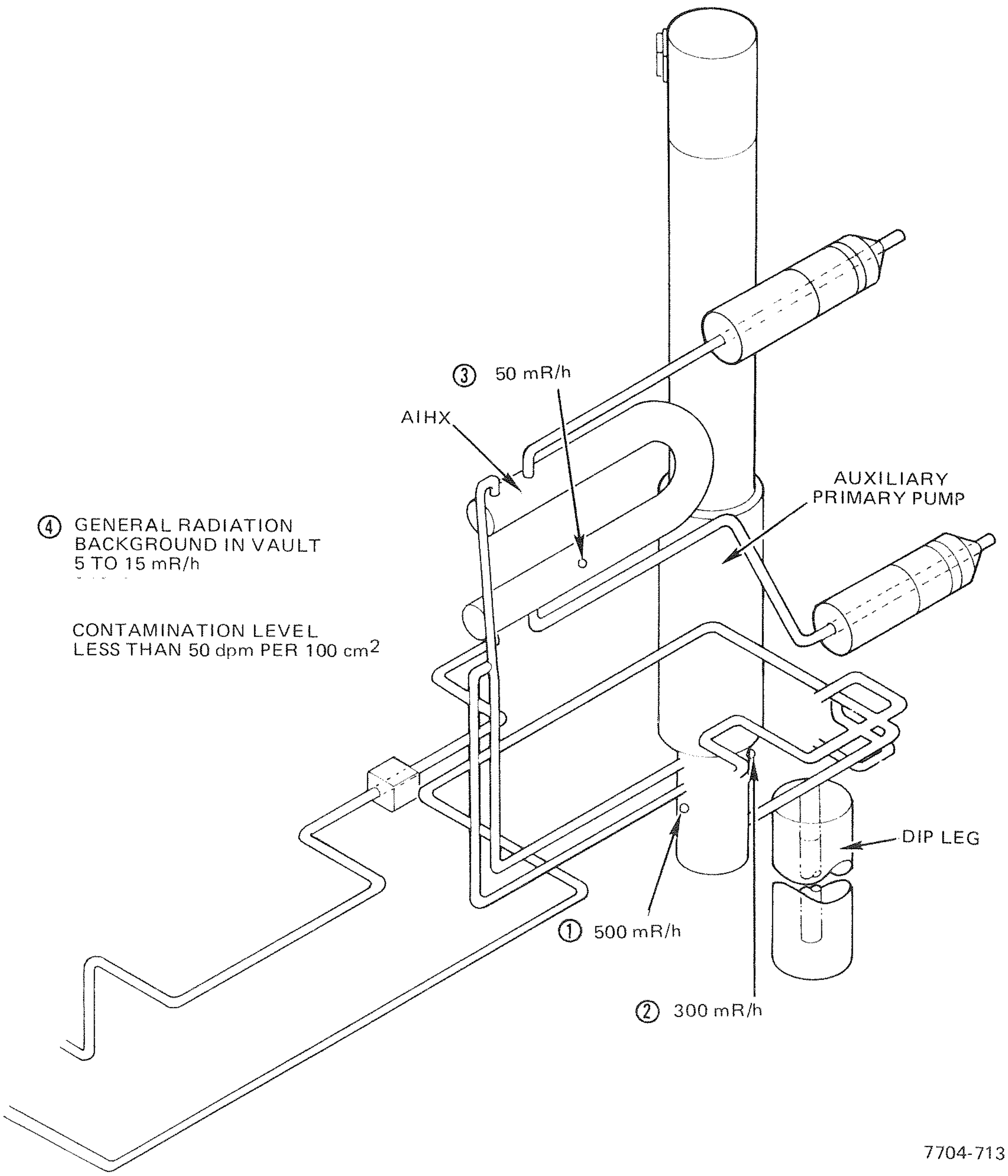

Figure 21. Auxiliary Primary Loop Radiation Levels

Sodium Service Vault Radiological Survey

(See Figure 22)

Area	mR/h
1	25
2	50
3	50
4	5,000
5	5,000
6	20 to 50

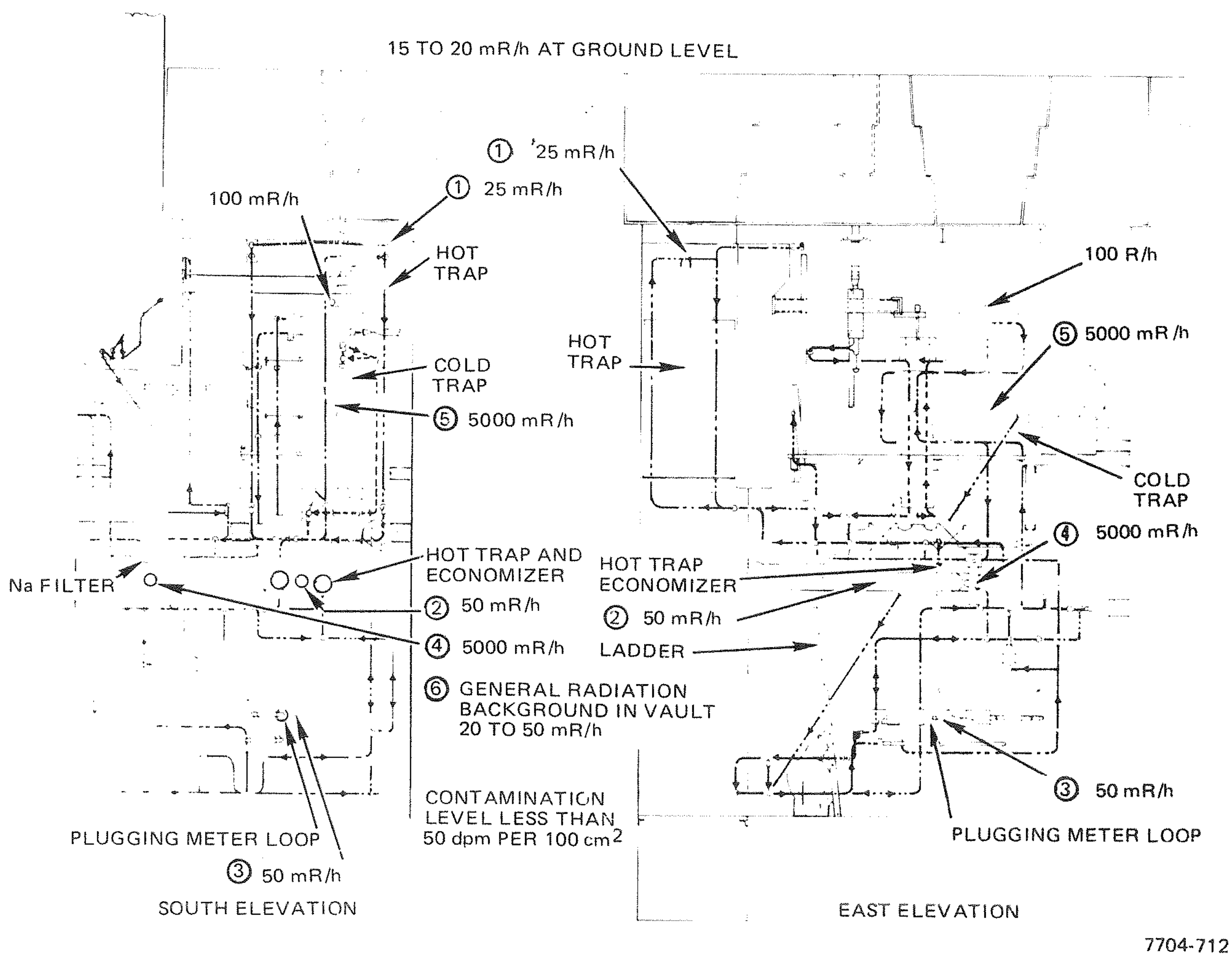

Figure 22. Sodium Service Vault Radiation Levels

Detailed working procedures were written to cover the removal of the SRE sodium system components. A separate procedure covered the sodium service vault. Before any components could be cut and removed from the vaults, insulation material containing asbestos had to be removed and packaged for disposal. The pipes were cut using a hydraulically operated, pipe-mounted cutter and a hand-held "porta-band" saw. Pipe ends were sealed, and the cut pipes were transferred to Building 163 for further size reduction. Large components within the vaults such as the main intermediate heat exchanger (MIHX) and the lower section of the main primary pump were removed and packaged for burial.

Some of the sodium system components presented special problems. The mixing tee in the main primary gallery exposed workers to a radiation field of 5 R/h at 2 ft and 22.5 R/h at contact. The cold trap and sodium filter in the sodium service vault both read 5 R/h at contact. Localized shielding was required on these components during pipe cutting and removal from the vault. Health, Safety & Radiation Services provided guidance and survey support throughout the sodium system component removal. Personnel exposure was closely monitored, and personnel were rotated from operation to operation to minimize exposure to each individual.

4.4.5 Disposal of Radioactive Waste Handling and Storage Systems

The SRE Radioactive Liquid and Gas Waste Retention System was located on a hillside north of the SRE, in and around Building 653. The system included six 5000-gal decay tanks (four for gas and two for liquid) buried in front of the building. One receiver tank was buried adjacent to Building 653, and three standing tanks (T1, T2, and T3) were located at the northwest corner of Building 143. T3 was at grade; T1 and T2 were in a 35-ft-deep pit.

Decommissioning in this area began with the removal of system components inside Building 653. Health, Safety & Radiation Services determined the extent of contamination. All contaminated piping, equipment, and machinery were packaged for burial. The six hillside waste holding tanks were excavated and removed with the assistance of an excavation contractor. The tanks were surveyed, wrapped in plastic, and shipped to a burial site. Figure 23 shows the six waste tanks during construction. Contaminated soil in the vicinity of the two liquid holding tanks was removed using a backhoe, placed in approved shipping containers, and shipped to a burial site. After all buried tanks and piping adjacent to Building 653 were excavated, the building was razed and all noncontaminated rubble was buried in the clean and surveyed tank excavation area. The area was restored to grade, and drainage control was constructed as necessary to prevent erosion.

Next, the block wall supporting T3 instrumentation was removed, exposing the tank. Concrete blocks covering the pit were removed, exposing T1 and T2. The nitrogen cover gas in the tanks and piping system was monitored, found to be contaminated, and vented to the facility radioactive exhaust system. The tanks and piping were purged with compressed air to remove any radioactive gases. Interconnecting piping, valves, valve operators, and instrumentation were removed, surveyed for contamination, and dispositioned accordingly. T1 and T2 were shipped to burial. T3 was moved to the Radioactive Materials Disposal Facility and utilized as a holding tank in a Radioactive Liquid Waste Disposal System. Contaminated liquids from the SRE were sent to the RMDF for processing and disposal.

Interconnecting piping between Building 143 and the radioactive waste handling and storage area, Building 653, was excavated and the pipe was disposed of as radioactive waste. A small amount of contaminated soil was also shipped to burial. The clean trench was backfilled with clean soil and the surface was repaved with asphalt.

4.4.6 Removal of the Inert Gas Systems

During reactor operation, the SRE facility had two inert gas systems: a nitrogen system and a helium system. These two systems were combined during initial deactivation. Much of this combined inert gas distribution system, such as the gallery nitrogen cooling system, was removed during the sodium system dismantlement. The remaining noncontaminated supply and distribution

Figure 23. Installation of the Six Radioactive Waste Decay Tanks

system was removed when sodium passivation was complete since an inert cover gas was no longer required.

4.4.7 Removal of the Kerosene Cooling System

The SRE facility was equipped with a secondary cooling system. In this system, the kerosene coolant was circulated through the reactor plug and ring shield, and around the core cavity liner, the wash cells, and the storage cells. A portion of the system external to Building 143 was removed as part of the noncontaminated peripheral dismantlement. The internal portions, which consisted of piping in the trenches to the reactor, to the wash cells, and to the storage cells, were removed during the excavation of the high bay. Residual kerosene coolant in the piping posed a hazard because of its flammability and toxicity. Care was taken when working around components cooled with kerosene to ensure adequate ventilation.

4.4.8 Disposal of Fuel- and Moderator-Handling Machines

Two FHMs and one moderator-handling machine were used at the SRE. The FHMs, designated Mark I and Mark II, were gas-tight, lead-shielded cylinders weighing approximately 52 tons each. They consisted of a hoisting assembly (dome), shielded body section, a viewing section, a lower adapter assembly with a vacuum-tight gate valve, and internal mechanisms for fuel pickup and guidance.

The SRE moderator-handling machine was also a gas-tight, lead-shielded cylinder weighing approximately 25 tons. All three machines were stored upright in the high bay of the SRE.

Before any decommissioning activities were begun, a radiological survey of the machines was conducted by Health, Safety & Radiation Services. Based on the results of the survey, a trade study was made to determine the best method of disposing the three units. The decision was made to ship the machines to burial intact. All noncontaminated attachments to the units were removed and sent to salvage. External surface contamination was removed to meet shipping requirements. A contractor licensed to receive and transport radioactive equipment was selected. The facility crane was used to load each unit on special transport vehicles. The three machines, one per truck, were packaged and transported to the burial site in Beatty, Nevada.

4.4.9 Removal and Disposal of the Reactor Vessel and Internals

The removal and disposal of the reactor vessel and fixed internals required special tooling similar to that developed by Oak Ridge National Laboratory and used in the Elk River reactor decommissioning. AI acquired the ORNL design for the rotating mast manipulator used to dismantle the Elk River reactor. AI modified the design to fit the SRE reactor vessel, fabricated the manipulator, and developed the underwater cutting parameters in a mockup facility. After developing the cutting parameters, the manipulator was installed in the SRE, and the vessel was cut into manageable sections. The vessel segments were stored under water in a shipping cask liner until a full cask load was ready for shipment. The activated vessel internals and segments were disposed of by land burial inside the sealed cask liners.

The SRE reactor vessel consisted of three concentric tanks, the core tank, the outer tank, and the core cavity liner. The innermost tank was the 1-1/2-in.-thick stainless-steel core tank. The open-top right cylindrical tank was 20 ft high and 11 ft in diameter. Its top was located 10 ft below the level of the SRE high-bay floor. The top of the core tank was connected to the core tank bellows assembly. The bellows provided a flexible seal between the tank top and the ring shield (see Figure 5).

Immediately outside the core tank were seven 5-1/2-in.-thick thermal rings. These rings rested on each other and were not welded together. Two of the rings had cutouts for piping penetrations. The top surface of each ring had four equally spaced tapped holes that were used to install and remove rings.

Immediately outside the thermal rings was the 1/4-in.-thick carbon steel outer tank. This 20-ft-high tank was 12-1/2 ft in diameter and was set 10 ft below the level of the high-bay floor. The top of the tank was welded to a bellows assembly that provided a flexible seal between the top of the tank and the core cavity liner. Two bellows assemblies intersected the walls of the outer tank at 180° from each other. These bellows provided a flexible seal between the core cavity liner and the outer tank walls and also encased reactor piping.

Immediately outside the outer tank wall was Super X insulation. This 9-in.-thick layer of insulation was held to the outermost containment vessel (the core cavity liner) by wires connected to studs. The studs were welded to the inside of the core cavity liner. The outer tank was supported at the bottom by four rings that were interspaced with insulation.

The core cavity liner was the final metal containment for the reactor core. It was a 1/4-in.-thick carbon steel tank backed by high-density concrete.

Inside the core tank was a 1/4-in. stainless-steel core tank liner, an open-ended cylinder. The upper and lower halves of the liner were held in place by a liner attachment ring located midway between the top and bottom of the tank.

Inside the tank liner were the main and auxiliary inlet pipelines that carried sodium coolant to the core. The pipelines were constructed of an outer guard pipe and inner coolant pipe. Also inside the core tank liner were the core clamps and band. There were 12 core clamps.

Located between the liner and the core tank were three pipelines: one for the moderator coolant, one for the tank drain, and one for the tank vent.

The grid plate was located 18 in. above the core tank bottom. This plate was used to channel the sodium coolant through the fuel elements and moderator cans. The grid plate was supported by a ledge that was welded to the ID of the core tank. Twenty-four stud and nut combinations secured the grid plate to the ledge. The grid plate was also supported by 20 staybolts that rested on the bottom of the core tank. The bolts were attached to the grid plate with nuts.

4.4.9.1 SRE Mockup Facility and Operations

A remote manipulator system was used to cut up the reactor vessels while submerged under water for radiation shielding. A mockup of the various vessels was used to develop and check the remote manipulator operating parameters prior to installation in the SRE. An existing ORNL manipulator design used for the Elk River dismantling program was modified to meet the SRE geometric requirements. The underwater portion of the manipulator was fabricated from stainless steel for ease of decontamination and for corrosion resistance. The vessel mockup facility was designed and then fabricated in place in Building 003 at the AI, Santa Susana site (see Figure 24). An existing manipulator control console was obtained from ORNL and modified for added versatility of operation.

An Activity Requirement document was written that identified the tasks and technical approach for developing the special remote tooling required to effect a safe and timely removal of the SRE reactor internals and vessels. (TI-704-990-001, "Activity Requirement 1 — Remote Tooling for Removal of SRE Vessels," 2 October 1974.) This document identifies 14 task requirements and presents the Activity Network Schedule for performing the tooling effort. Each of these tasks required a Task Requirement document to define the associated purpose, scope, requirements, and technical approach. Additionally, a Task Requirement document was written that described the consulting effort provided by ORNL to AI for the manipulator development and operations.

Mockup operations are described in documents N704TR990003, N704TR990004, and N704TR990005.

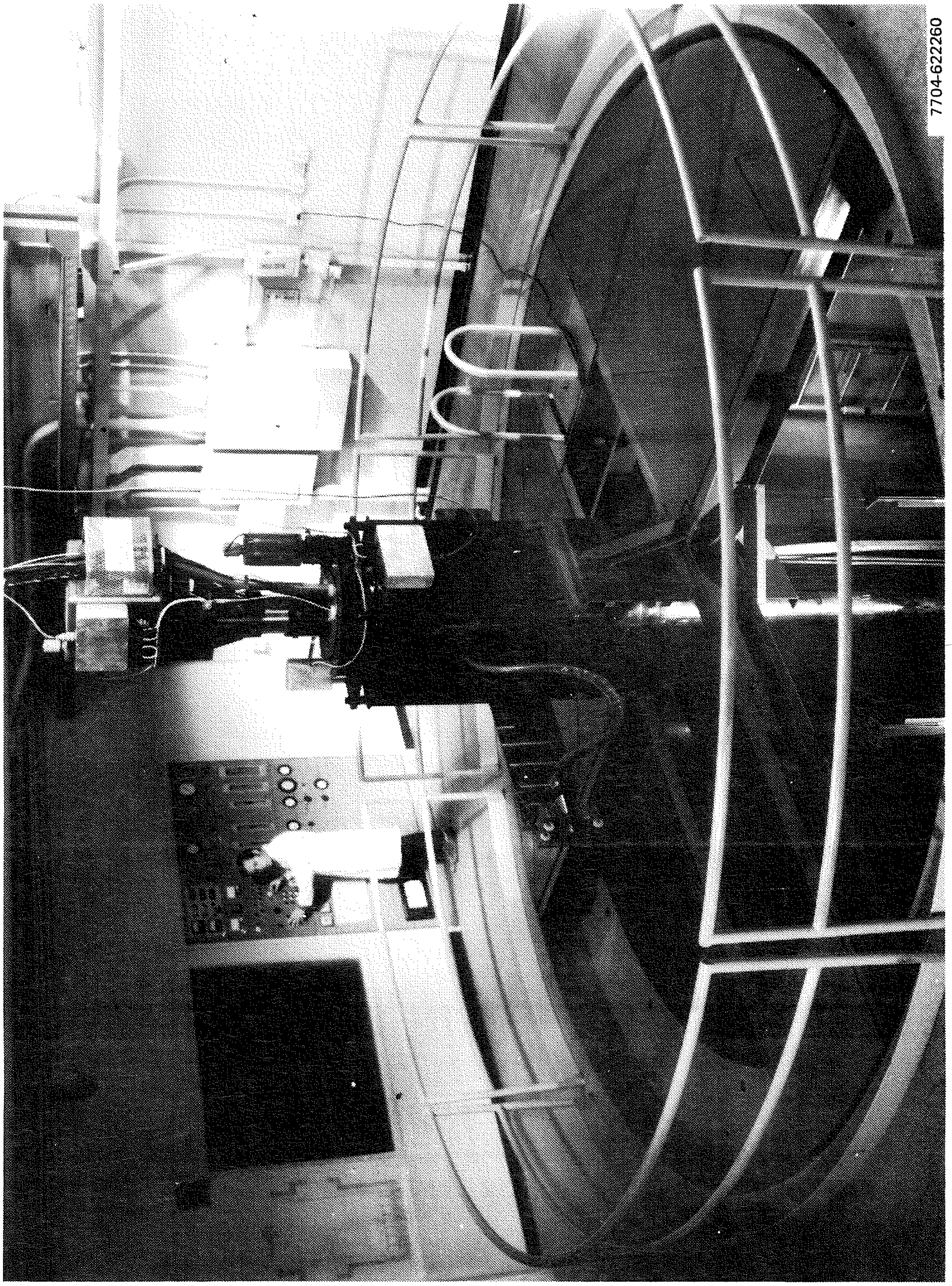

Figure 24. SRE Reactor Vessel Mockup Facility

4.4.9.2 Reactor Vessel Shield Plug and Core Component Removal

After passivating the residual sodium in the SRE reactor vessel, all of the 3-1/2-in.-diameter shield plugs and the core components, except the moderator and reflector cans, were removed. Table 11 itemizes the core components and their location in the reactor.

TABLE 11

LOCATION OF CORE COMPONENTS IN THE REACTOR

Item	Location[a]
Control rods (4) (new)	R-21, -23, -67, -69
Safety rods (4) (new)	R-32, -35, -54, -57
Core heaters (10) (new)	R-4, -7, -14, -16, -25, -41, -49, -50, -62, -78
Dummy fuel elements (6) (canned graphite)	R-42, -43, -44, -45, -46, -47
Sodium level probes (4) (new)	R-60, -61, -63, -64
Pile oscillator (inner assembly) and spare safety rod thimble	R-68
Core exposure facility (new)	R-2
Moderator temperature probes (2) (new)	R-18, -39
Fission product monitor plug	R-17
Core II shield plug and hanger assembly (25)	R-3, -5, -6, -9, -10, -11, -13, -19, -30, -31, -36, -53, -59, -65, -66, -71, -72, -73, -74, -75, -76, -77, -80, -81
Spare safety rod boron assembly (new)	R-52
Experimental thimbles (3) (new)	R-8, -52, -79
Neutron source (antimony oxide-beryllium) (new)	R-37
PEP moderator cans (new)	91 central core positions
Graphite reflector cans (new)	28 outer core positions

[a]Numbers refer to reactor (R) locations in Figure 25.

During the SRE-PEP program, most of the core components were replaced with new components. Some of the remaining components were modified and some were the originals. The upper portion of the dummy fuel elements was original. The lower sections had been modified by replacing the fuel simulating portion. The pile oscillator was an original component, but the thimble had been replaced. The fission monitor plug and the core II shield plug and hanger assemblies were all original components. All of the components listed in Table 11 had been in direct contact with radioactive sodium and were contaminated. In addition, original components had been exposed to the reactor flux and were activated. During the SRE deactivation program completed in 1968, radiation levels of 100 mR/h were measured on the fuel plugs when they were removed from the shield plug.

Each core component was disconnected from its 3- or 3-1/2-in.-diameter shield plug, surveyed radiologically, and dispositioned. The 20-in.- and both 40-in.-diameter shield plugs (Figure 25) were then removed and transferred to the RMDF where the bottom reflector plates were removed. The remaining 3- and 3-1/2-in.-diameter shield plugs were also removed for disposal.

The 140-in.-diameter shield plug assembly was about 82 in. high (Figure 26). The bottom 10 in. contained a series of reflector shields suspected of being contaminated with sodium. This bottom reflector shield package consisted of 13 layers of overlapping shields. It was supported by fusion welds at the two 40-in.-, one 20-in.-, and fifty 3.5-in.-diameter penetrations. Spacers were stacked around each penetration to separate the shields. A 10-in.-high skirt surrounded the entire circumference of the 140-in.-diameter plug. All of the penetrations and the perimeter skirt had to be severed to separate the reflector shield package from the rest of the plug.

To accomplish the bottom reflector shield package removal, the 140-in.-diameter plug was removed from its core position and was mounted temporarily on support legs constructed from I-beams. A nitrogen-purge system was installed to provide an inert atmosphere and to control sodium oxidation. The

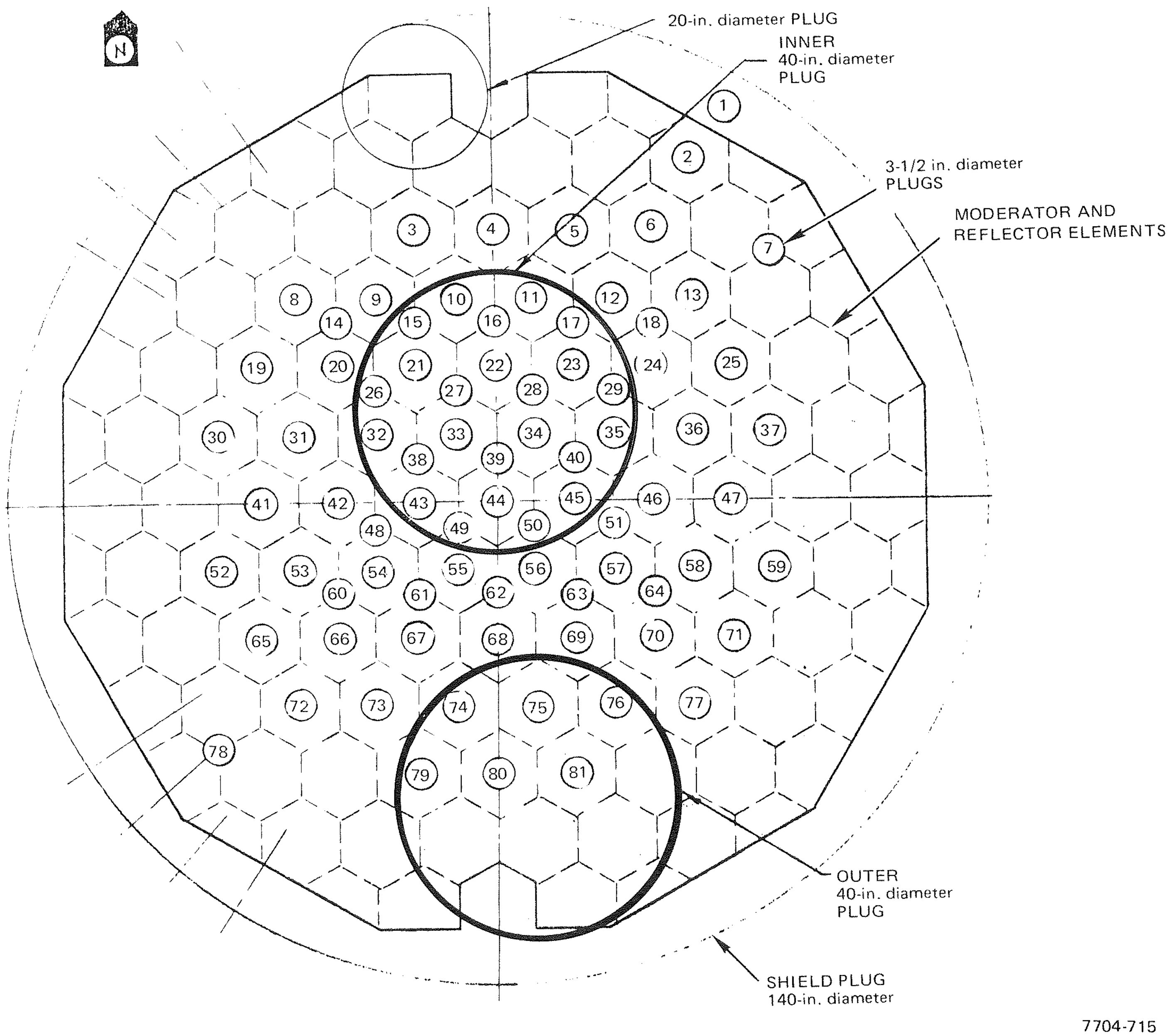

Figure 25. SRE Reactor Loading Face

shields were then removed, boxed, and transferred to the RMDF for further disassembly and to react residual sodium (Figure 27). The plug was then decontaminated, painted, and wrapped for disposal without further size reduction. Figure 28 shows the 140-in.-diameter shield plug being shipped for disposal.

Figure 26. 140-in.-diameter Plug Removal

7704-62973

Figure 27. Reflector Shield Segments at RMDF

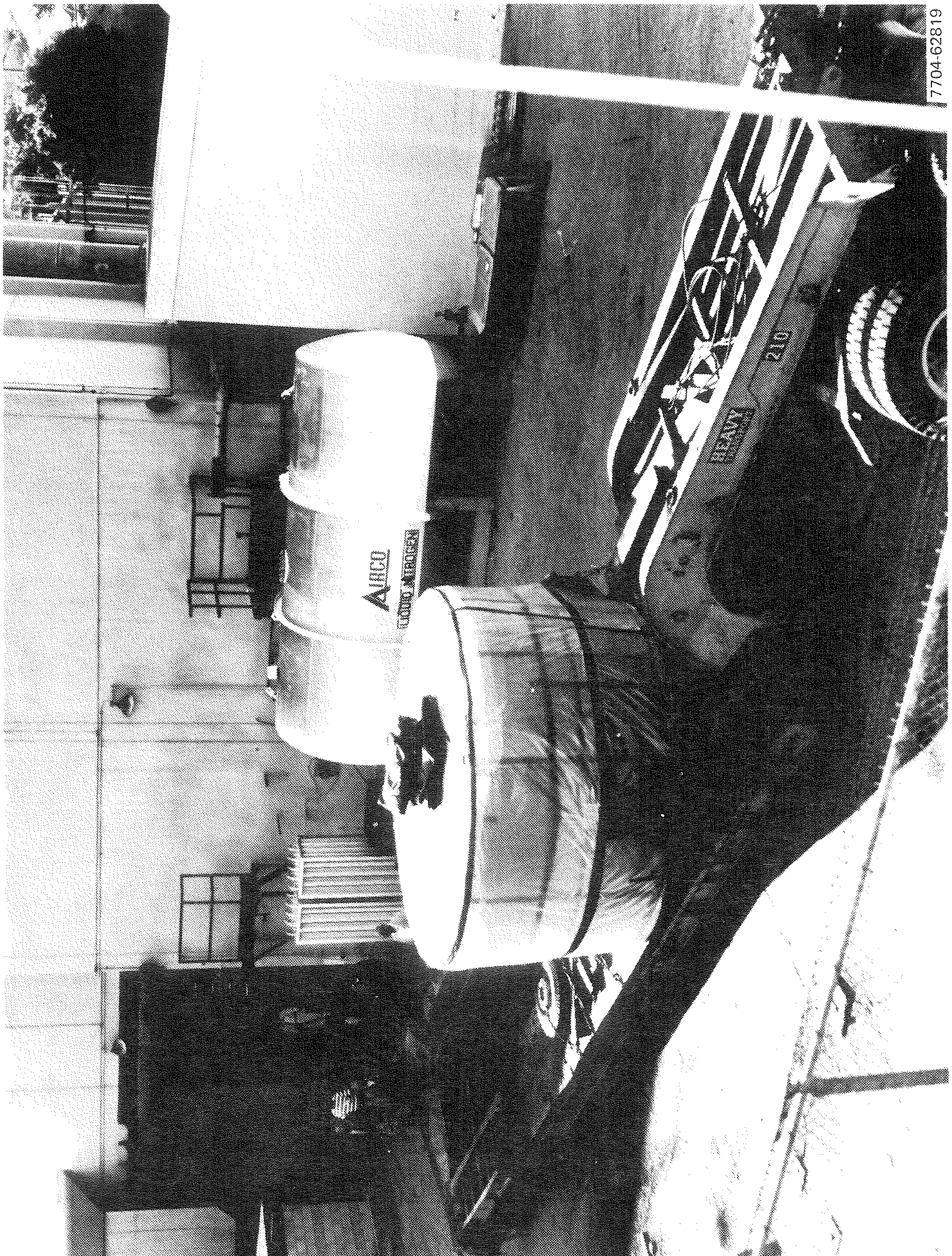

Figure 28. 140-in.-diameter Plug Packaged for Disposal

The 91 central-core-position moderator cans and the 28 outer-core-position graphite reflector cans were installed in the reactor during the SRE PEP. Since the PEP program did not reach the critical state, the moderator and reflector cans were only contaminated and not irradiated. These cans (Figure 29) were removed and inserted into plastic sheathing in a one-step operation. The snorkel tubes were saw cut from the cans as close to the top of the element as possible. Snorkel/vent tubes were packaged separately from the moderator/reflectors which were packaged six elements per box. These elements were then shipped for land burial.

4.4.9.3 Removal of the Ring Shield and Core Tank Bellows

The SRE tank bellows assembly provided a flexible sodium seal between the core tank and the ring shield (see Figure 30). It was a 4-ft-high, 13-in.-diameter corrugated cylinder. The 4-in.-deep corrugations were fabricated from 0.06-in. stainless steel. The cylinder was attached to the core tank and ring shield using 0.12-, 0.38-, and 0.7-in.-thick stainless-steel sections. The ring shield was 6 ft high, donut shaped, with a 194-in. OD and a 132-in. ID. It was constructed of reinforced magnetite concrete and lead and was jacketed with stainless steel. The 60-ton ring was used as shielding, and it also supported the 140-in.-diameter rotating shield plug. Below the ring shield and external to the bellows were reflector plate assemblies, which were stacks of ten 0.06-in. plates spaced 1 in. apart. A vent pipe and monitor tube shroud partially obstructed access for the bellows to core tank cut.

The vent pipe and monitor tube shroud were cut to provide torch access to the core tank bellows. A bellows cutter (see Figure 31) was then installed and used to cut the bellows at the top of the core tank and just below the ring shield. The ring shield was removed, followed by the bellows and the reflector plate assemblies. All plasma torch cutting for removal of the ring shield and core tank bellows was performed in air.

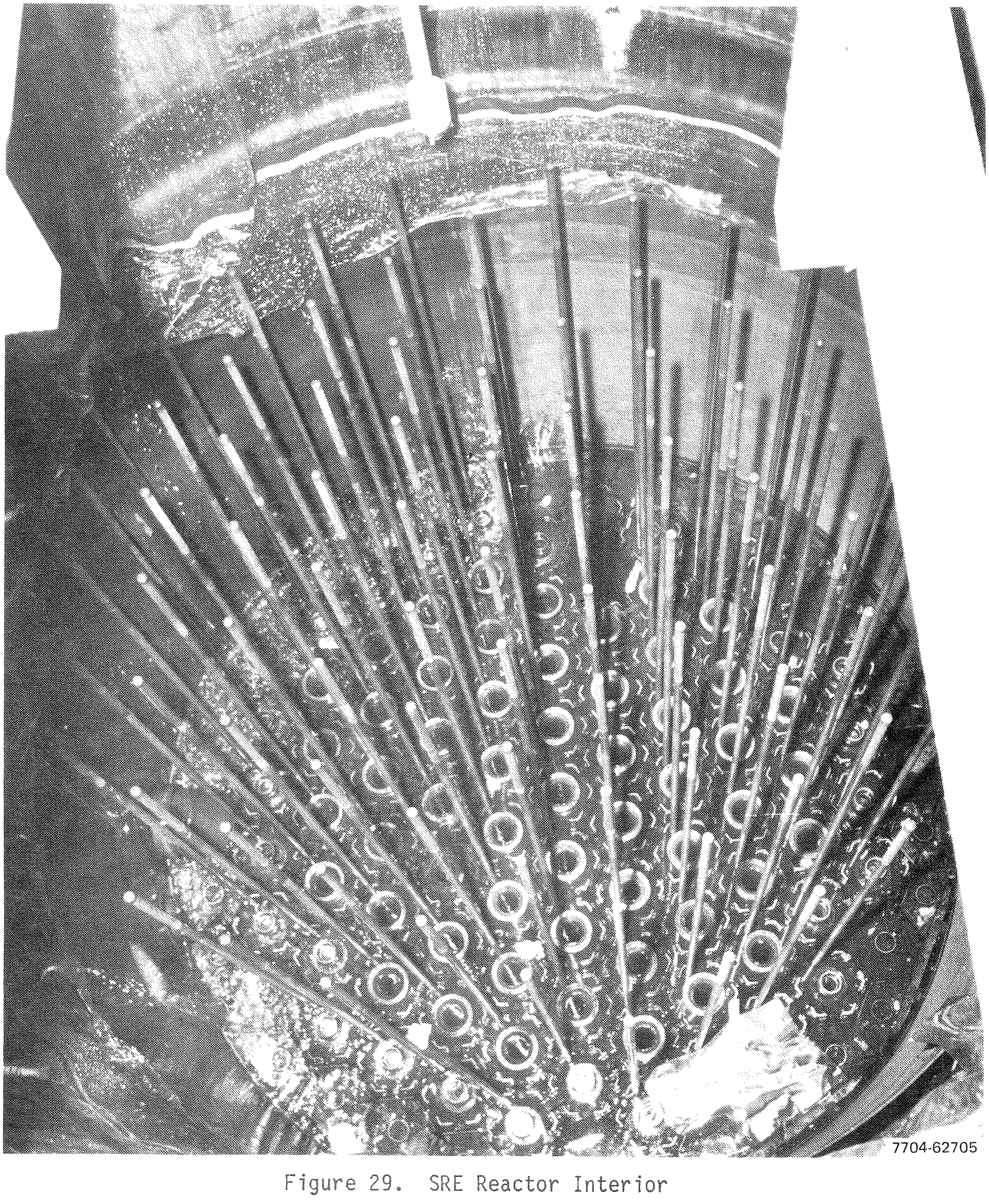

Figure 29. SRE Reactor Interior

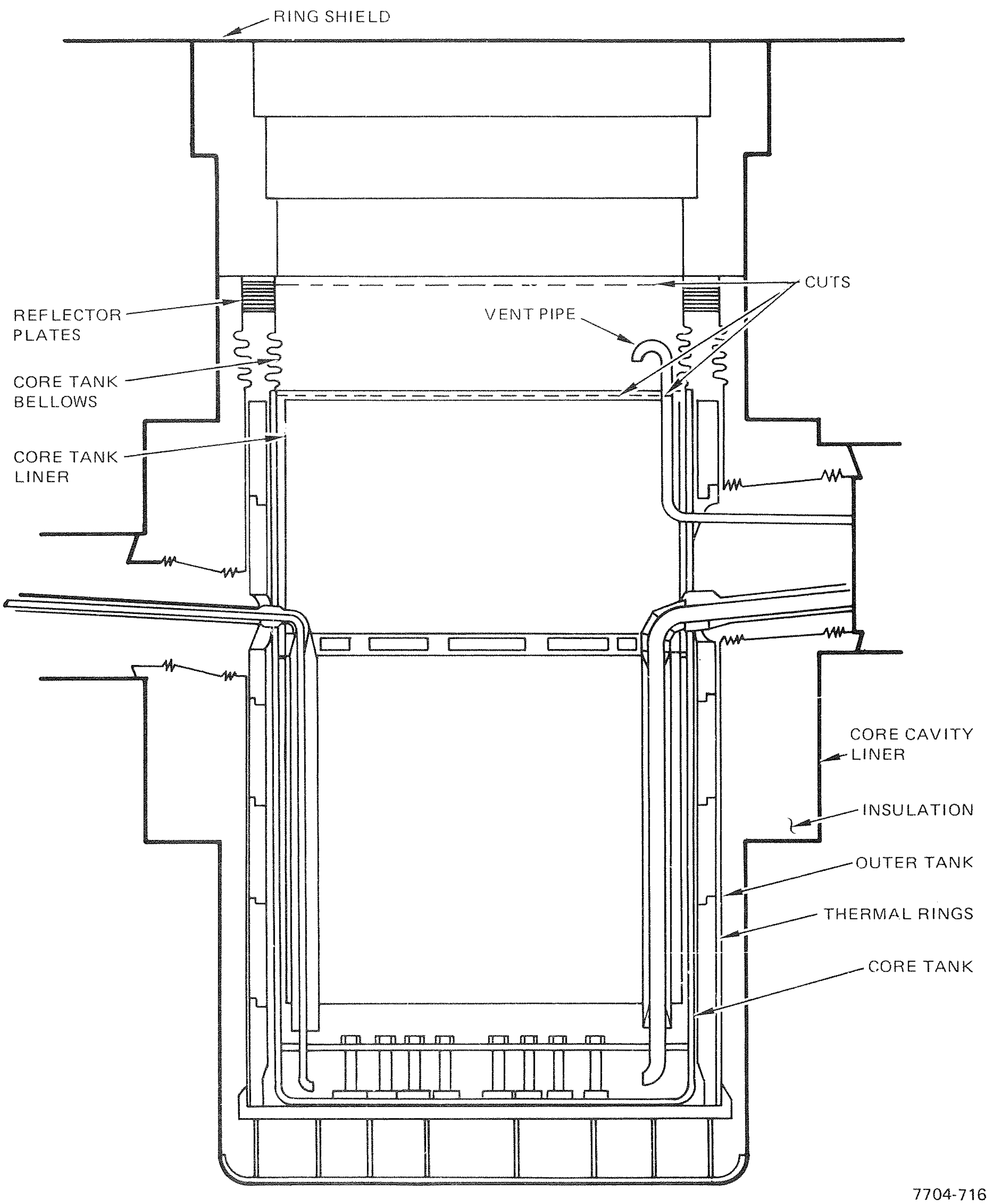

Figure 30. Bellows and Vent Pipe Cuts

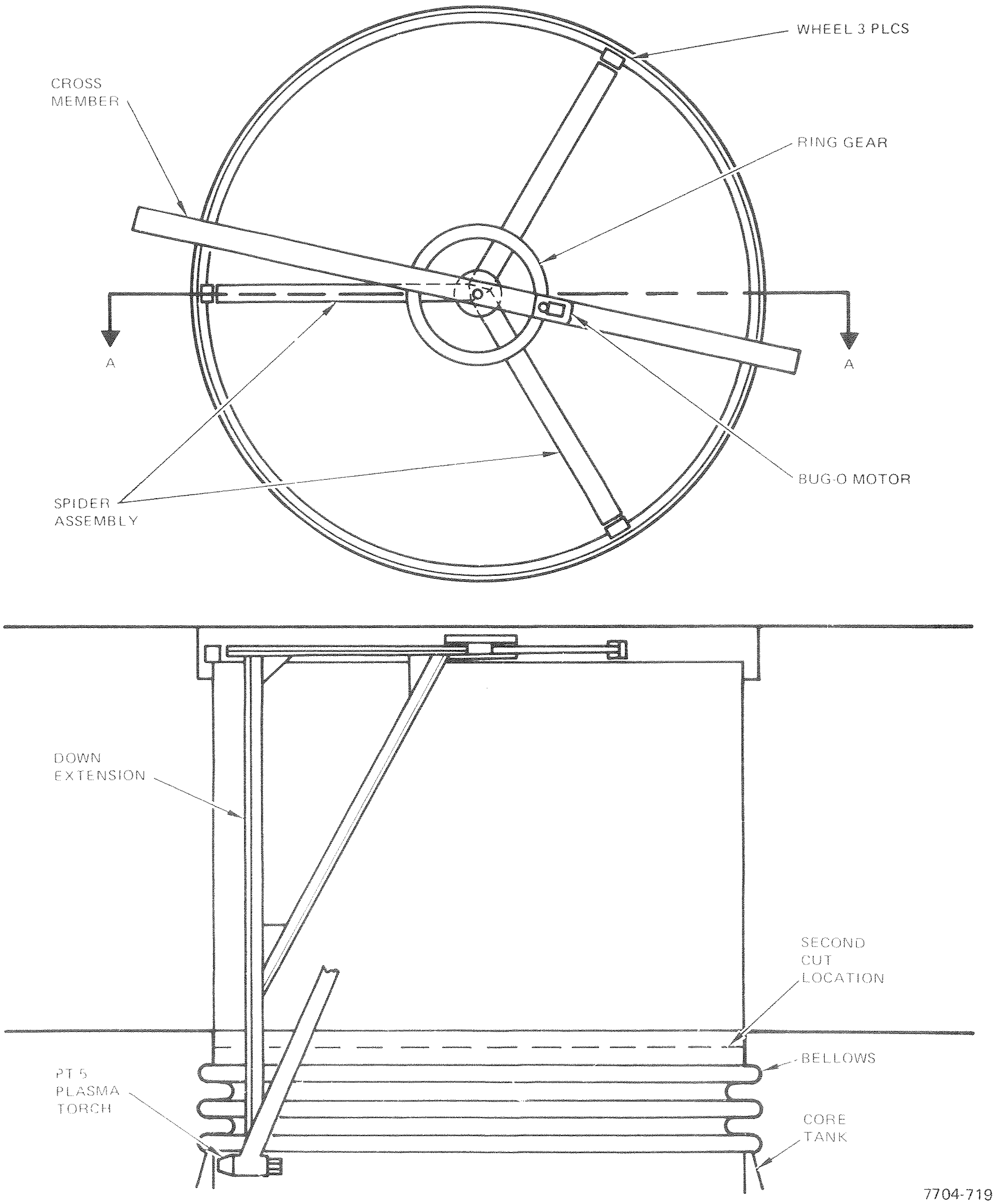

Figure 31. Bellows Cutting Fixture

Before implementing the removal of the ring shield and bellows, cutting parameters were developed in the SRE mockup. The mockup operations were divided into three tasks:

1) Develop cutting parameters for the Linde PT-5 plasma torch on materials representative of the bellows assembly
2) Check bellows cutting fixture
3) Develop remote tools and techniques to cut the vent pipe and monitor tube.

The vent pipe was removed by placing a temporary wooden platform over the water-filled reactor vessel, and the radiological exhaust system of the building was connected to the volume under the platform. The torch hose bundle was sheathed in plastic. The torch operator wore protective clothing and a full face mask. The pipe was secured with a nylon line which was tied off to the platform railing. A high-volume air sample taken at the operator's level detected no airborne activity when the pipe was cut.

The same equipment and techniques used to remove the vent pipe were used to remove the upper 3 in. of the instrument monitor tube. A lifting grip was used to remotely grapple the tube. Again, no significant airborne activity was detected.

The bellows was removed by making two circumferential cuts to free it from the ring shield as well as to free it from and provide access to the core tank. Torch movement was erratic because the guide wheel encountered surface irregularities. The guide wheel was removed and the torch was restrained so that the torch stood off the workpiece. The torch was moved over the core tank while the standoff distance was maintained constant.

A temporary wooden platform placed over the reactor pit was connected to the radiological exhaust duct and covered with plastic. The cut at the core tank level severed the bellows from the core tank. The cut of the bellows from the ring shield was made using the same equipment setup.

The ring shield was removed, loaded onto a 100-ton (gross vehicle weight) truck, and shipped intact to an approved burial site. The reflector plates were then removed using six plate grapples modified for remote placement. The plates were removed, cut up, and loaded into a low-level waste container. The bellows assembly was removed using four lifting grips that had been attached to the thermal ring-lifting spider. As the bellows was removed, it was wrapped in plastic. The bellows was cut into shippable sections in the RMDF Decon Room, Building 21.

4.4.9.4 Fixed Internals Removal

To gain access and to segment the reactor vessels, it was first necessary to remove the fixed internals from the SRE reactor vessels. Fixed internals consisted of the internal piping, protective shrouds, core clamps, and core clamp band.

The main and auxiliary core clamps were removed first to facilitate access to the main and auxiliary sodium inlet pipe shrouds. These clamps were removed by free-hand cutting with the plasma torch.

Five pipe lines and three shrouds (half sections of pipe welded into the liner to permit clearance for pipes between the core tank and liner) had to be cut before the core tank liner could be segmented. These tasks were initially performed by explosive cutting and finished by plasma torch. Jet Research Corporation (JRC) was awarded a contract to cut the pipes and shrouds. They performed the explosive cutting work during two separate visits.

During the first visit, the main and auxiliary pipes plus the liner shrouds were cut. The main inlet pipe was a 6-in. pipe concentric with a 10-in. pipe. The auxiliary inlet pipe was a 2-in. pipe concentric with a 10-in. pipe. These pipes were cut just above the grid plate and at the elbow (see Figure 32). The cuts just above the grid plate were made with a clip-on circular cutter (see Figure 33). A series of cuts was made to sever the main inlet pipe at the elbow. These cuts were made to remove a kneecap section

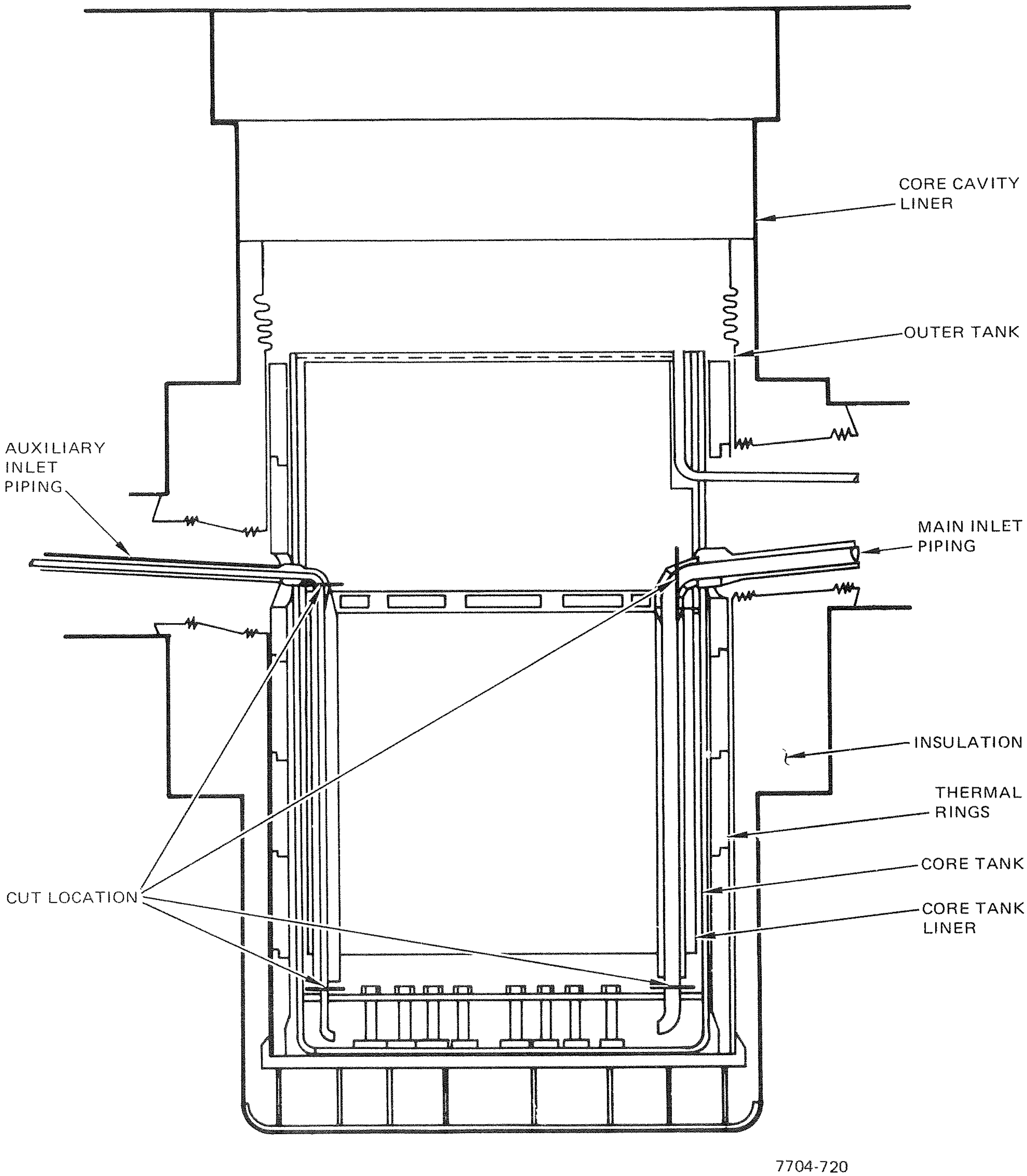

Figure 32. Main and Auxiliary Piping Cuts

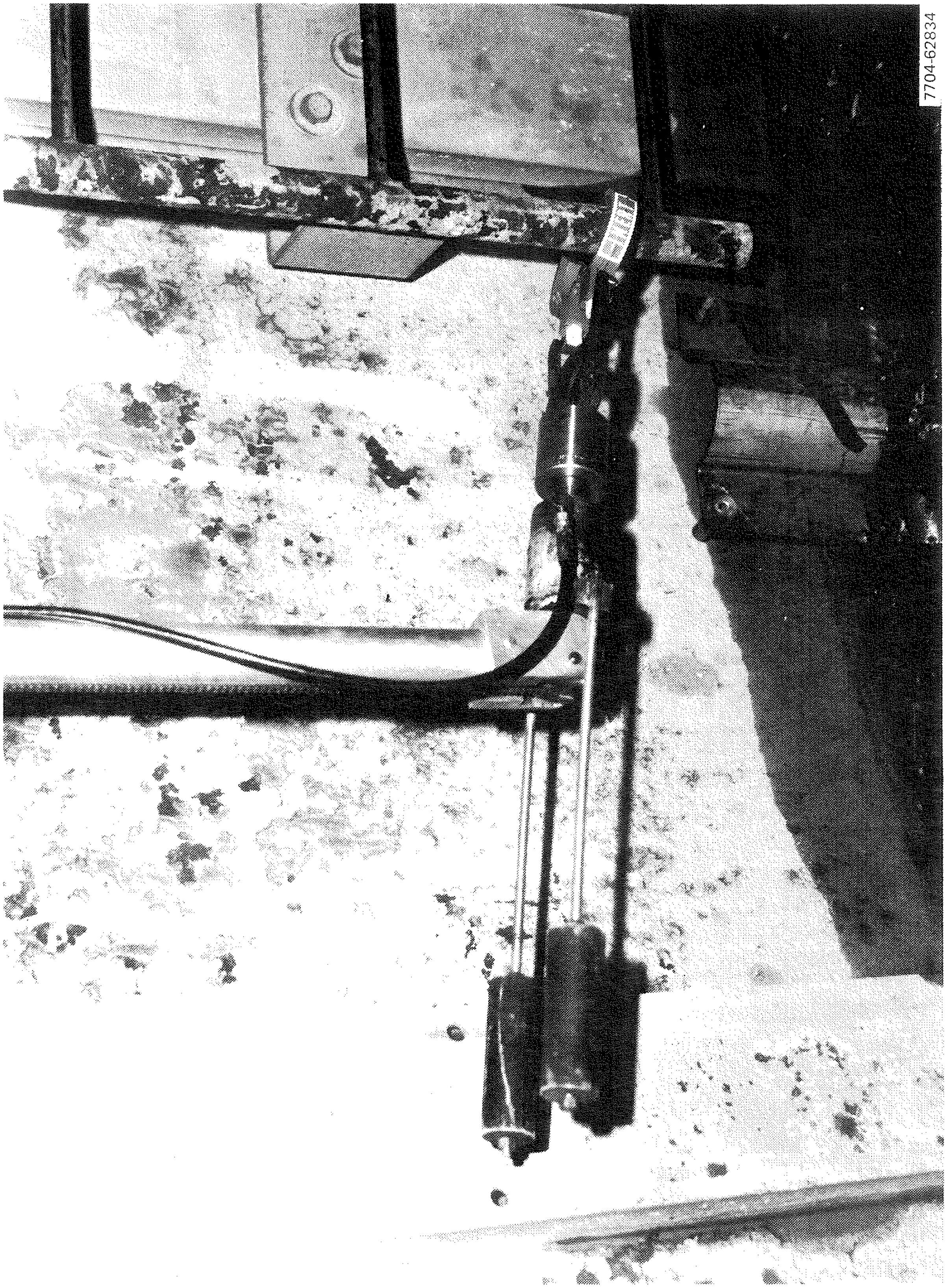

Figure 33. JRC Manipulator for Remote Installation of Explosive Charge

from the inner and outer pipe elbows. The remaining pipe "ligaments" were cut using circular outside diameter cutters. The auxiliary inlet was cut using a clip-on cutter on the horizontal section of the pipe. The cut pipes were grappled, transferred to the water-filled storage pit, and then cut into shippable lengths. Several cuts were made across the core tank liner shrouds. These cuts were linked to plasma-torch cuts in the liner.

During the second visit, the following pipe lines were cut:

1) Reactor vent (2-in., schedule 40)
2) Moderator coolant (2-in., schedule 40 concentric with 4-in., schedule 40)
3) Reactor drain (1-1/2-in., schedule 40).

The remaining core clamps and core clamp band were removed by the plasma torch after the pipe and shrouds were removed.

The SRE mockup in Building 003 was used extensively to set up the underwater plasma-cutting parameters, to check out the remote tooling for placement of the explosive charges, and to develop the underwater television camera system. Mockup operations are described in documents N704TR990003, N704TR990004, and N704TR990005.

Using the parameters developed in the SRE mockup, the main and auxiliary core clamps were removed. These clamps fell to the grid plate during removal where they were retrieved using a magnet.

Explosive cutting techniques and the plasma torch were used to cut the main and auxiliary pipes plus the liner shrouds, reactor vent, moderator coolant, and reactor drain pipes. Cutting proceeded as follows.

The JRC manipulator was installed directly at the center of the reactor core tank. The camera and pool lights were located on each side of the manipulator. A primed charge was installed on the JRC manipulator grips, and the manipulator was lowered into the work area. The spring-loaded charges were attached to the pipe by a clothes-pin-type spring action. After the charge was installed, the camera, lights, manipulator, and water filtration sump pump were lifted out of the water. The charge was armed, platform openings were covered with plastic sheeting, the high bay and surrounding hallway were cleared, and the charge was detonated. Before personnel were allowed to re-enter the high bay, a high-volume air sample was analyzed.

The main inlet pipe required 26 explosive cuts (shots) to free it from the reactor vessel. In 2 weeks, 24 shots were required to cut the pipe elbow, 13 more shots than had been anticipated. The ligament cuts were difficult to complete. The inside diameter of the pipe had been severely deformed by the previous cuts. This resulted in a larger than normal cutter-to-workpiece distance and caused the cutting jet to be attenuated by the intervening water. The ligament cuts were finally completed by using 155-g straight-line cutters. Two shots were made on the inlet pipe just above the grid plate. The cuts did not sever the 6-in. pipe. The elbow section of pipe below the grid plate was removed using special grappling operations.

The auxiliary inlet pipe was removed in two cuts as planned. The moderator coolant header pipe was severed just above the grid plate with just one shot.

Cuts across the tank liner shrouds were unsuccessful. The charges were held against the liner by a long pole. Six shots were required to make one cut. The remaining three cuts were incomplete.

The plasma torch and manipulator system were used to complete the liner shroud cuts. During the cutting, several torch nozzles and retaining nuts were consumed. It took about 2 h to cut the shroud.

An attempt to remove the reactor drain line by explosive cutting was not successful. The next eight explosive cuts were made on the coaxial pipe moderator coolant line elbow. The cutting sequence was similar to that for the main inlet (e.g., remove a half-section of the outer and inner pipes, remove a section of the 2-in. pipe ligament, and cut the 4-in. pipe ligament). An attempt to remove a knee cap section of the 2-in. inner pipe was unsuccessful because of difficult cutter-to-workpiece geometries. Two additional attempts using large gram cutters were also unsuccessful. At this point in the operation, it was found that the water level of the storage pit had fallen 4 in. The explosive cutting was stopped and the plasma torch was used to complete the removal of the piping and to cut the access slot.

Explosive cutting operations were hampered by poor water clarity, water plume, debris, and by the large number of operations required to make an explosive cut. Each detonation stirred up fine particulate matter that clouded the water. To obtain a clear view, the camera and lights were moved closer to the work site, which then obstructed the JRC manipulator and increased the time required for charge placement. Water plume and splashes from the detonations contaminated the work platform. The maximum number of cuts that could be made each day was limited to four because of the time required to (1) remove the camera, lights, JRC manipulator, and sump pump from the reactor vessel; (2) cover the platform openings with plastic sheeting; and (3) test for high airborne radioactivity. The main drawback to explosive cutting was the large amount of debris produced that had to be removed remotely. Each cutter exploded into as many as six pieces of shrapnel, each of which had to be individually grappled for removal.

The ESG manipulator and plasma torch were used to complete pipe cutting operations. The manipulator arm was replaced with an arm that moved the torch radially and rotated it from a horizontal to a vertical position (Figure 34).

Two methods of plasma-arc cutting of the pipe were tried. The first method consisted of piercing the pipe wall using the torch to create a series of interconnecting holes until the pipe was severed. The major difficulty with this method was that it left small tangs of material between the pierced holes. These tangs prevented torch access to the backside of the pipe. To remove the tangs, the torch had to be precisely located over them. This was time consuming.

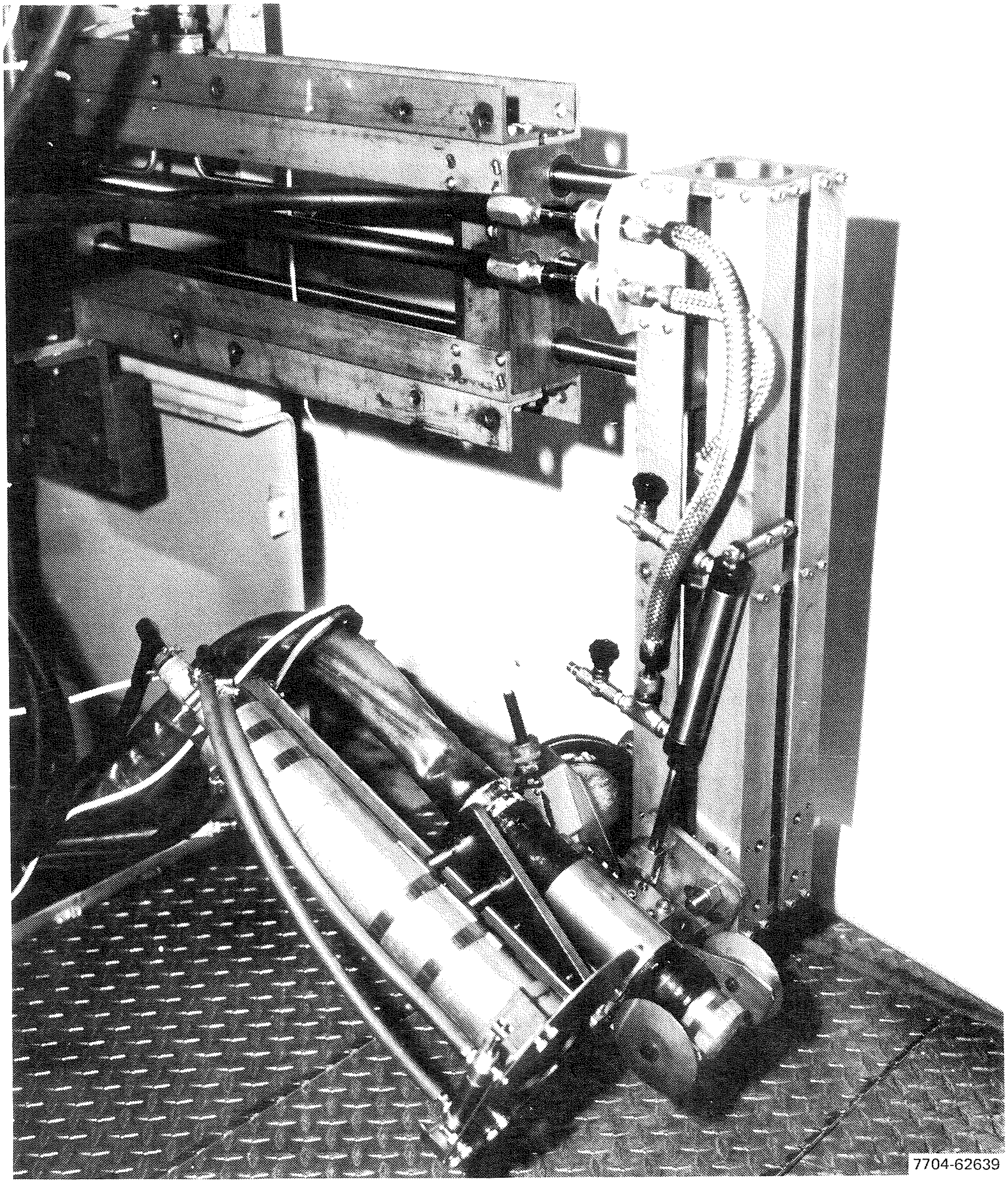

Figure 34. Plasma-Arc Cutting Torch Radial Arm

The second method consisted of making a series of vertical cuts on the pipe. The cuts were made with a large-diameter nozzle using hole-piercing torch parameters. Cutting speed was 30 in./min. These parameters made wide cuts on the pipe. A series of cuts (as shown in Figure 35) was made to provide torch access to the backside of the pipe. Because each cut was made on undisturbed material, the arc was easier to start and fewer torch stoppages resulted. It took about 1 h and required 14 cuts to sever the moderator pipe.

The moderator pipe was cut into three pieces: two 4-1/2-ft sections and one 2-ft section. The tank drain line was cut into three pieces: one 3-ft section and two 5-ft sections. The drain line pipe section located below the grid plate was removed later during core tank bottom cutup operations. The vent line was removed in one piece by cutting just above the elbow and removing the support bracket bolts. The pipe sections above the grid plate were dropped onto the grid plate as they were cut, where they were grappled and removed. All sections of pipe were cut to a length acceptable to the shipping cask liner without further size reduction.

The bolt heads for the brackets supporting the tank vent line were removed by using the plasma torch. The TV camera was used to position the torch over the bolt heads. With the torch set to operate with hole-piercing parameter settings, the arc removed most of the bolt heads. To finish removing them, the torch had to be moved several times.

Internal piping was removed from the reactor vessel in two steps: first the main and auxiliary piping, then the small-diameter piping. The main and auxiliary pipes were removed to facilitate cutting up the core tank liner. The TV camera system and placement device were used to locate the pipes and to position the grapple. The grapple-actuating cylinders were connected to a regulated, 200-psig nitrogen supply. The grapple was positioned just above the center of gravity of the pipe. The pipes were lifted vertically out of the reactor vessel and transferred in air to the storage pit. Three persons effected the transfer: a crane operator, a health physicist, and a mechanic

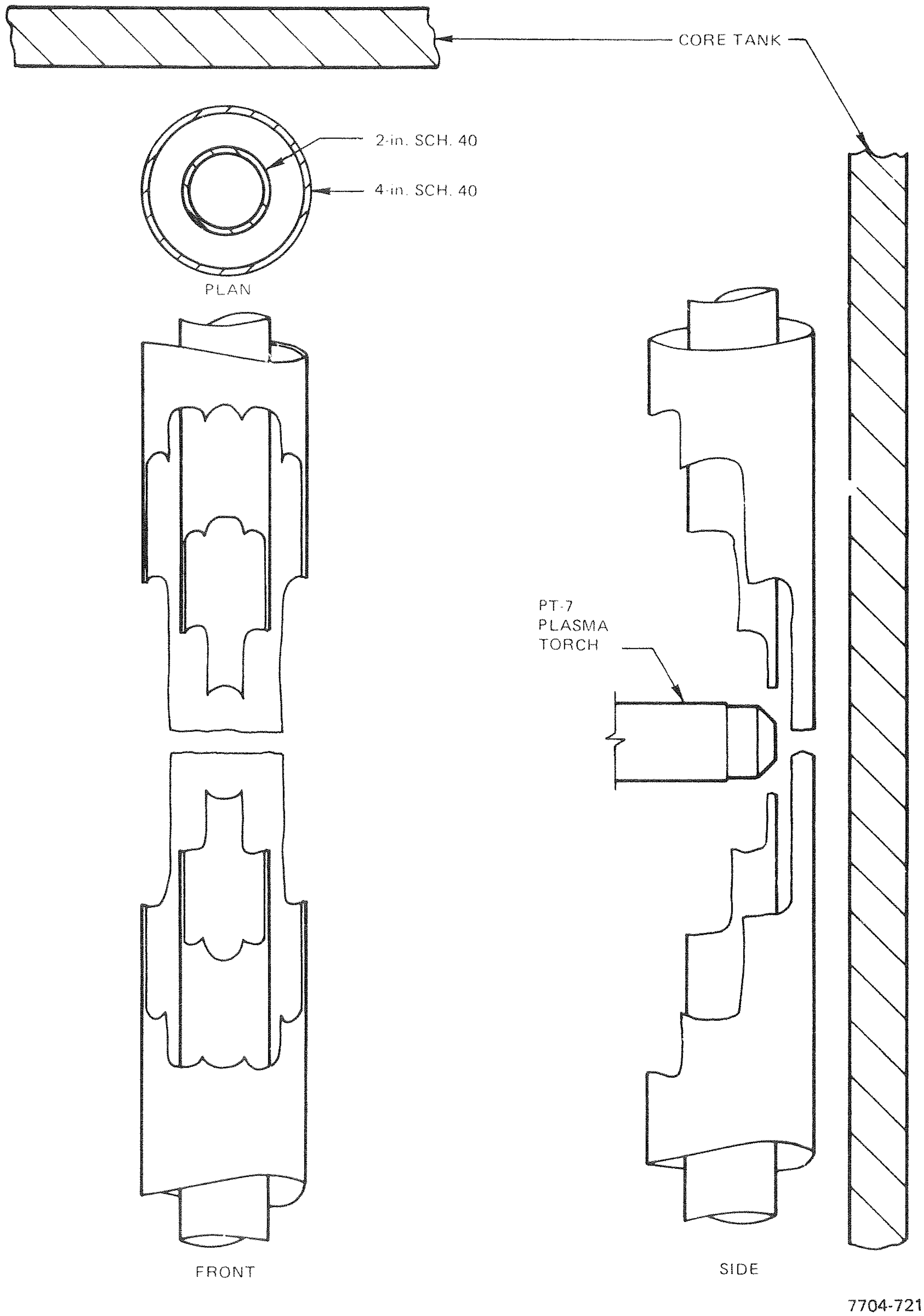

Figure 35. Pipe Cutting Using Plasma Torch

who guided the pipes through the platform openings. Maximum exposure received during the transfers was 40 mrem. Each pipe read 12 R/h at 3 ft.

The small-diameter pipes were removed from the reactor to facilitate cutting up the grid plate. A right-angle adapter was added to position the clamp for grappling horizontal pipes. A section of hose was added on the vent port of the clamp actuation valve to exhaust air and contaminated water from under the platform. The TV camera system was used to help locate the pipes and install the clamp. Each section of pipe was lifted to within 1 or 2 ft of the surface of the water and secured with a nylon line. Each pipe section was then transferred to the storage pit, and each line was tied to the pit railing for easy retrieval. The moderator coolant header was unbolted, grappled, and transferred to the storage pit to be cut under water.

Pipe grappling operations were hindered by poor water clarity. Consequently, the TV camera and placement device were needed to position the grapple. This step added to the operating time.

Eight core clamps were intact when removal operations began; the main and auxiliary core clamps had been removed previously. One clamp was dislodged when other internals not attached to the tank structure were removed. The fourth clamp was displaced as the sump pump was being removed. Sections of the band that had only one pin were displaced from the liner by jiggling the housing clamp lifting tool. Four sections of the band had two or more pins in them and required the stud displacement tool for removal. These band sections were a result of missing clamps. These double sections read only 50 mR/h, which permitted disposal in an unshielded wooded shipping container. The core clamp next to the moderator coolant line could not be grappled with the housing clamp lifting tool. Instead, a nylon line was used to catch and secure it for removal. The clamps that had been dislodged were retrieved from the grid plate by using a magnet and line.

4.4.9.5 Core Tank Liner and Liner Attachment Ring Removal

The core tank liner, a 1/4-in.-thick, Type 304 stainless steel, open-ended cylinder (see Figure 36), was held in position inside the core tank by a 2-3/4-in.-wide, 3/4-in.-high liner attachment ring. The ring had been welded to inside diameter of the the core tank at the core top level. The core tank liner removal method was to cut the liner into 44 segments (4 rows of 11 segments) using the manipulator and the plasma torch under water. The segments had grappling slots cut in them and were grappled before the final cut was made. The segments were then transferred to the underwater storage pit, where they were loaded into storage racks to await selective loading into a shipping cask liner.

The top and bottom halves of the core tank liner were welded to the attachment ring. To faciliate vertical cuts across the ring, eleven 6-in.-long sections of the band were removed with a plasma torch. Each section was removed by making two radial cuts and an intersecting cut parallel to the tank wall (see Figure 37). This reduced the ring to a 0.5-in.-thick stub and gave sufficient torch-to-stub clearance to permit using the plasma torch to make vertical cuts on the core tank through the access slots.

The following development work was performed in the mockup prior to removal of the core tank liner:

1) Cuts were made using the manipulator plasma-torch system (see Figure 38)
2) Candidate grapples and removal rigging were tested, and the best were selected
3) Improvements were made in the test plasma-torch/manipulator system.

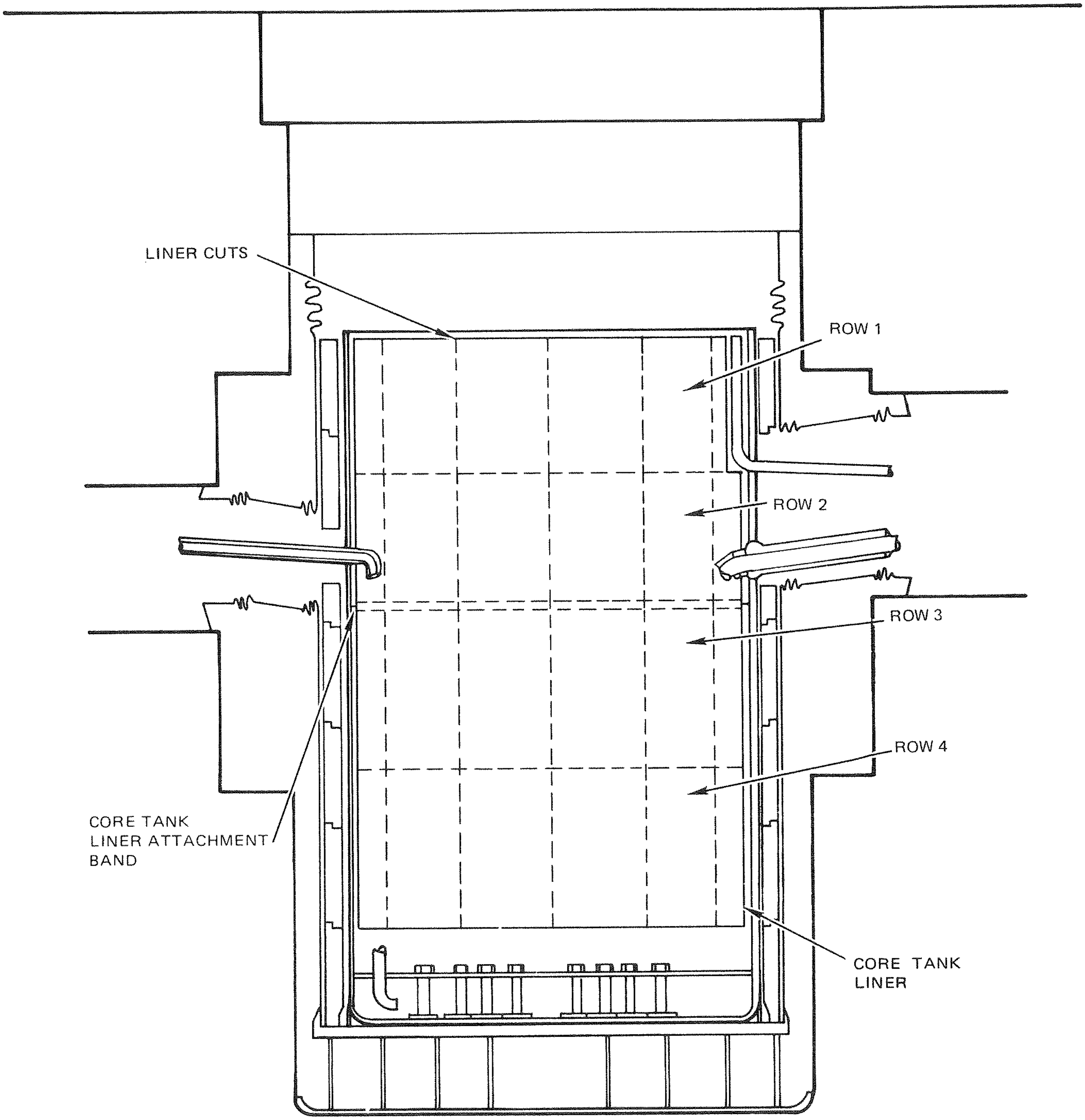

Figure 36. Core Tank Liner Cuts

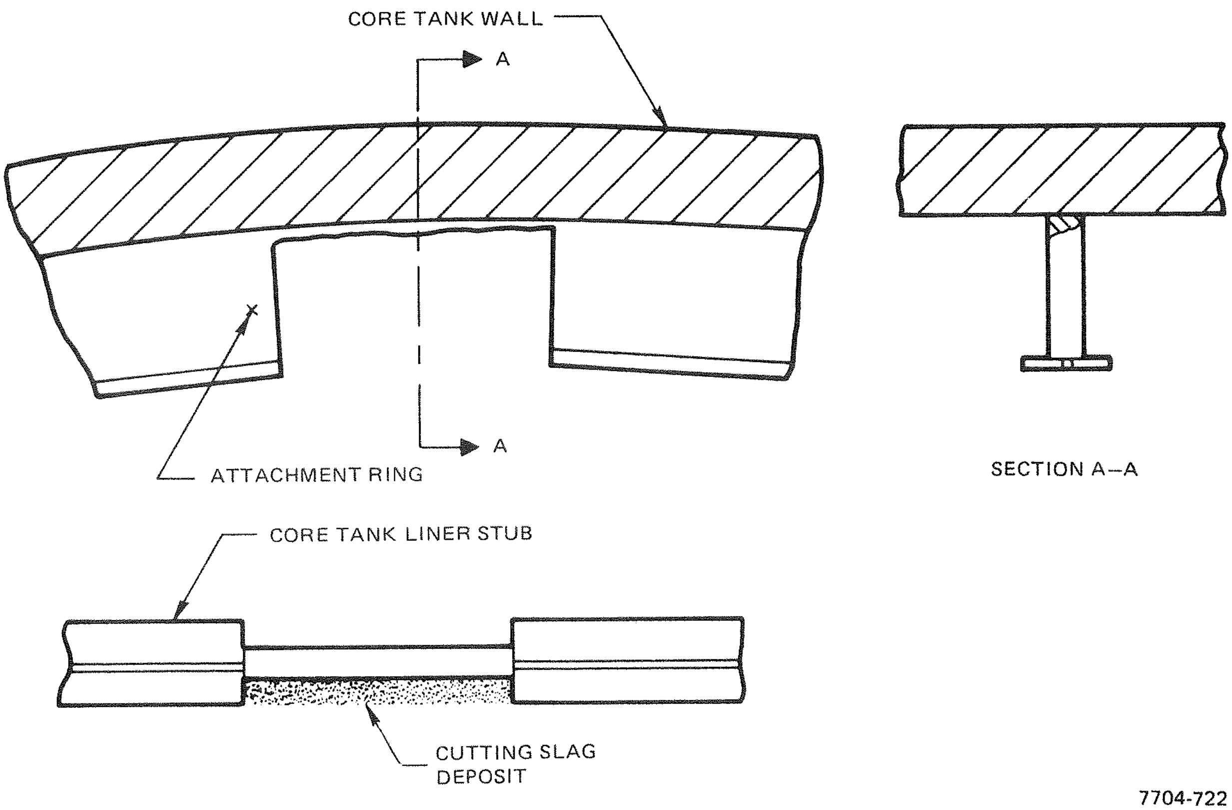

Figure 37. Attachment Ring Cuts

Removing the core tank liner required 15 working days and presented few problems. The SRE manipulator and plasma torch were used under water to segment the liner into 44 sections. Although explosive cutting of the internal piping and shrouds had deformed the liner, cutting was accomplished by changing the torch-to-liner standoff distance and by diligently observing the cutting operation with the underwater TV camera.

The first row of segments read 40 mR/h at 3 ft and were placed in wooden shipping containers for disposal. The remaining three rows of segments were transferred to the storage pit and loaded into a shipping cask liner. The second row of segments averaged 400 mR/h at 1 ft, while the last two rows ranged from 4.2 to 7.5 R/h at 3 ft.

To gain access to the core tank wall, 6-in.-long slots were cut in the core tank liner attachment ring with the underwater plasma torch. The cut pieces were retrieved from the tank by using grappling devices.

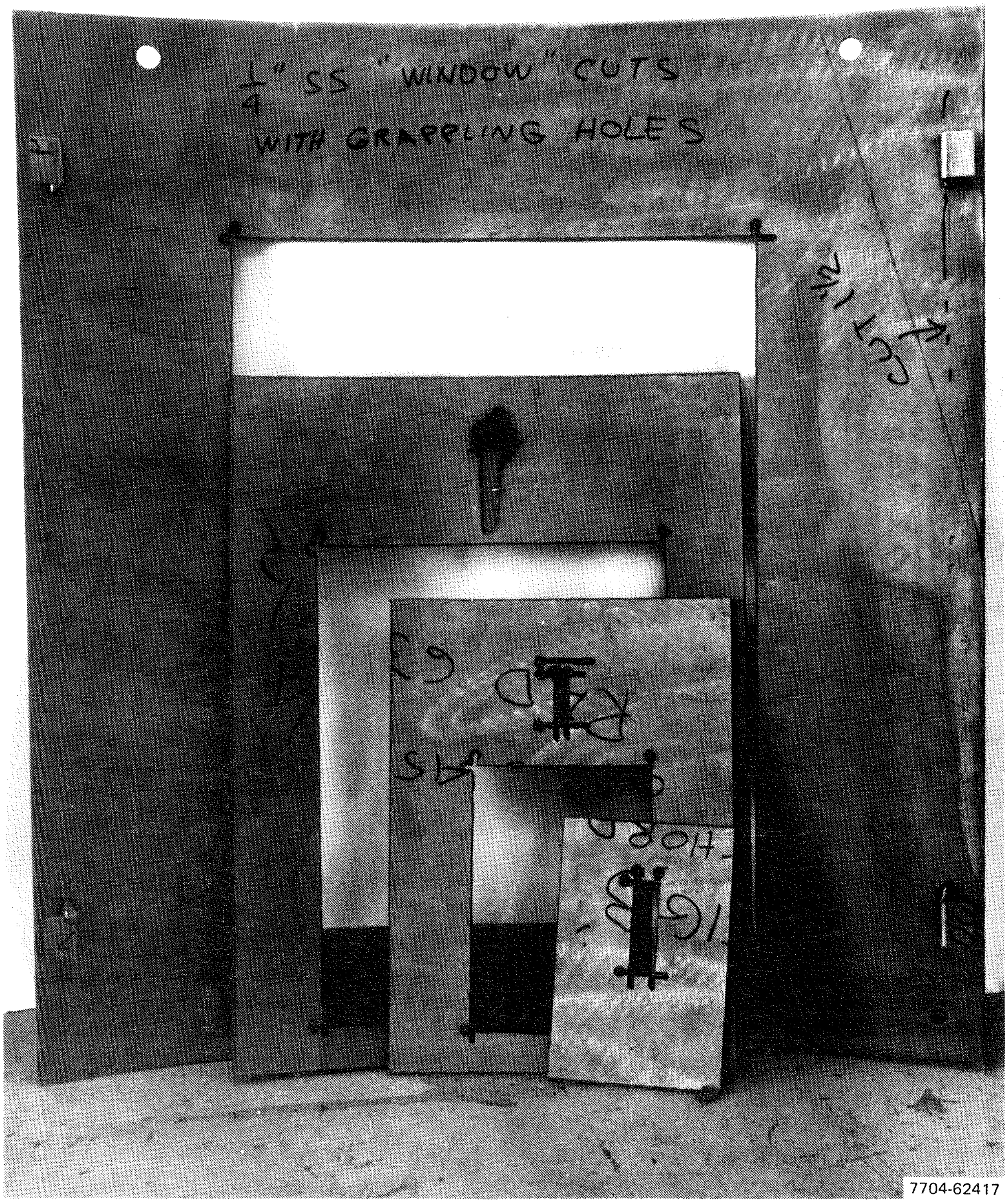

Figure 38. Core Tank Liner Mockup Cutting Test Results

4.4.9.6 Grid Plate Removal

The SRE grid plate was a 125-in.-diameter plate of 1-1/2-in.-thick, Type 304 stainless steel. Its perimeter was supported by a 5-in.-wide ledge welded to the core tank inner wall. The ledge and grid plate were bolted together with 24 stud and nut assemblies. The grid plate was also supported by 3-in.-diameter bolts threaded into pads that rested on the bottom of the core tank. The bolts were arranged in three circles: the inner circle had 6 bolts; the other two had 12 each. The pads of the middle bolt circle were welded to the tank bottom.

The underwater plasma-torch process was used to remove the grid plate. It was cut into two concentric rows of segments with a center segment. The staybolt and perimeter nuts were removed by a manually powered long extension wrench. The radial cuts on the outer row of segments were extended to within 3/4 in. of the wall of the core tank. A cutting insert consisting of a 3/4-in. section of stainless steel was installed to prevent the arc from extinguishing in the gap between the grid plate and the ledge. A circumferential cut 3/4 in. from the wall of the tank completed the segmentation (see Figure 39).

Grid plate removal operations were divided into three tasks:

1) Staybolt nut removal
2) Grid plate cutting
3) Segment removal.

The grid plate staybolt nut wrench was used to remove the 3-in. hex nuts. The wrench was a 30-ft-long, 2.5-in.-diameter tube with a T-handle on the top. The nuts were grappled using the pipe clamp with jaw inserts and placed in underwater storage cans. About 3 days were required for removal.

Inserts were to be installed in selected coolant holes to facilitate grid plate cutting and to prevent guide wheels from dropping into the coolant

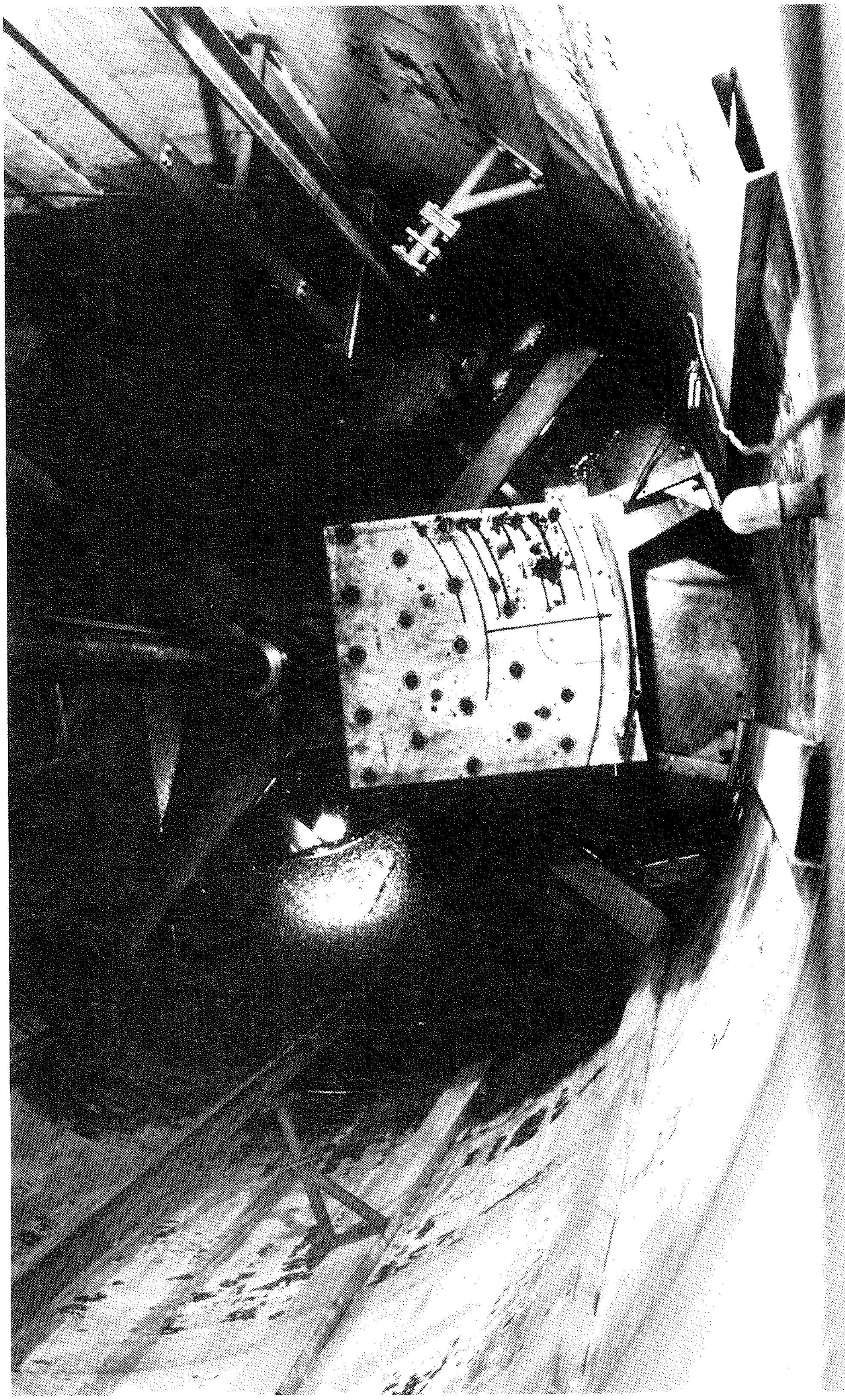

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Figure 39. Mockup of SRE Reactor Grid Plate

holes. It was determined that the grid plate could be cut with the underwater plasma torch without using guide wheels, eliminating the need for coolant hole plugs. Torch-to-workpiece distance was set by the torch-touch system described previously.

The staybolts were caught and secured with 1/2-in. nylon line as the segments were transferred to the storage pit. The hook was removed from the segment, and the nylon line was rigged to the overhead crane to free the staybolt. The segments in the middle row were grappled using three hooks of the grid plate grapple. The center segment was left in the vessel until after the thermal rings had been removed. The staybolts that were threaded into welded pads were removed using the staybolt wrench and the staybolt removal tool. As the first staybolt was being removed, a sodium water reaction occurred. To avoid this condition, the remaining bolts were loosened eight turns and then soaked for 24 h. Only a few gas bubbles were observed during removal of the remaining 11 staybolts. The bolts were grappled using the pipe clamp with pipe jaw inserts.

4.4.9.7 Core Tank Removal

The core tank was a 132-in.-diameter, 18-ft-deep, 1-1/2-in.-thick stainless-steel tank (Figure 40). Main and auxiliary inlet pipe stubs protruded through the walls of the tank. All other internal piping had already been removed, the external piping had been severed to within 18 in. of the outside of the core tank, and access slots had been cut in the core tank liner attachment ring.

Core tank removal consisted of two separate operations. First, the tank walls were removed from the tank top to a level 27 in. above the grid plate; second, the remaining wall sections and core tank bottom were removed after removal of the grid plate.

The first operation involved cutting the tank walls into 44 segments using the manipulator and plasma-torch system. The segments, approximately 37

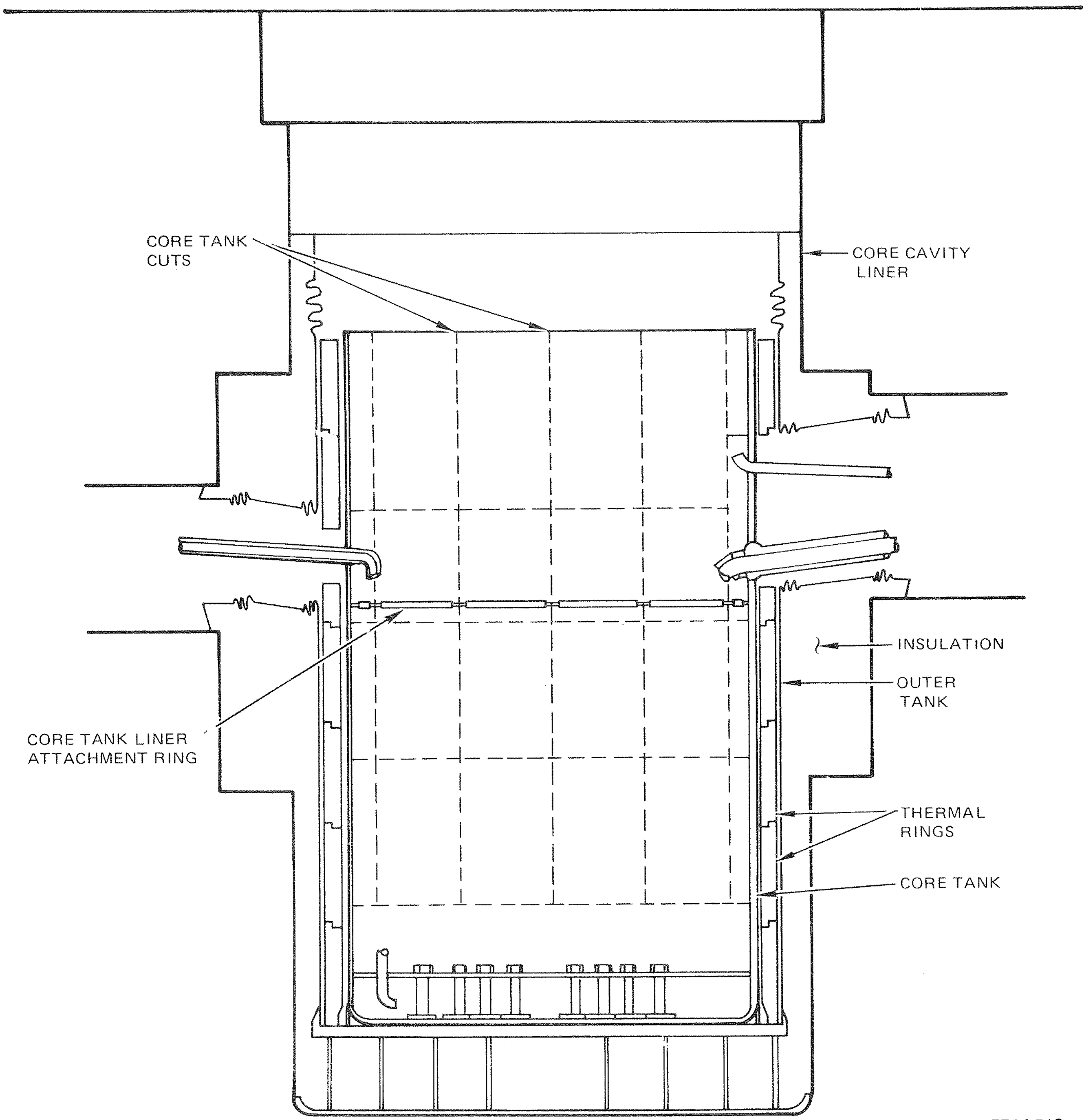

Figure 40. Core Tank Cuts

by 42 in., were cut from four rows containing 11 segments in each row. The segments were removed by grappling and tensioning an entire row of segments before the horizontal severing cut was made. A row of segments was then transferred in air and loaded into an underwater storage rack.

The second operation, removal of the core tank bottom, consisted of removing the remaining 40.-in.-high wall section; the 5-in.-wide, 1-1/2-in.-thick grid plate support ledge; and the tank bottom, as illustrated in Figure 41. The core tank bottom was actually removed after thermal ring removal. Twenty-four 3/4-in.-diameter studs had been threaded into the grid plate support ledge. Welded to the outside of the tank next to the bottom was a 3/4-in.-thick skirt that had been used for tank alignment. A 4-1/2-in.-radius section joined the wall to the tank bottom. Twelve 8-in.-diameter, 1-1/2-in.- thick pads had been welded to the tank bottom in a 68-in.-diameter circle as support for the grid plate.

Two new operations were identified in the tank bottom removal sequence: (1) cutting the tank wall through the ledge and (2) cutting the tank bottom and leaving a 1/2-in. gap between it and the outer tank. Tests directed toward solving these conditions were conducted in the SRE mockup demonstration tank and led to the following removal sequence:

1) The tank bottom was lifted 6 in. and shimmed in place. Segment tensioning devices were used to lift the tank bottom.
2) To provide torch/arm access to the lower tank walls, 3 in. of the ledge was cut off.
3) The remains of the tank drain line were removed.
4) Twelve access slots were cut through the ledge.
5) The tank wall section was cut into 12 segments and removed.
6) The skirt attachment weld was removed by cutting the tank at the weld elevation.
7) The skirt was removed from the tank.
8) The tank knuckle was cut and moved inward 20 in.

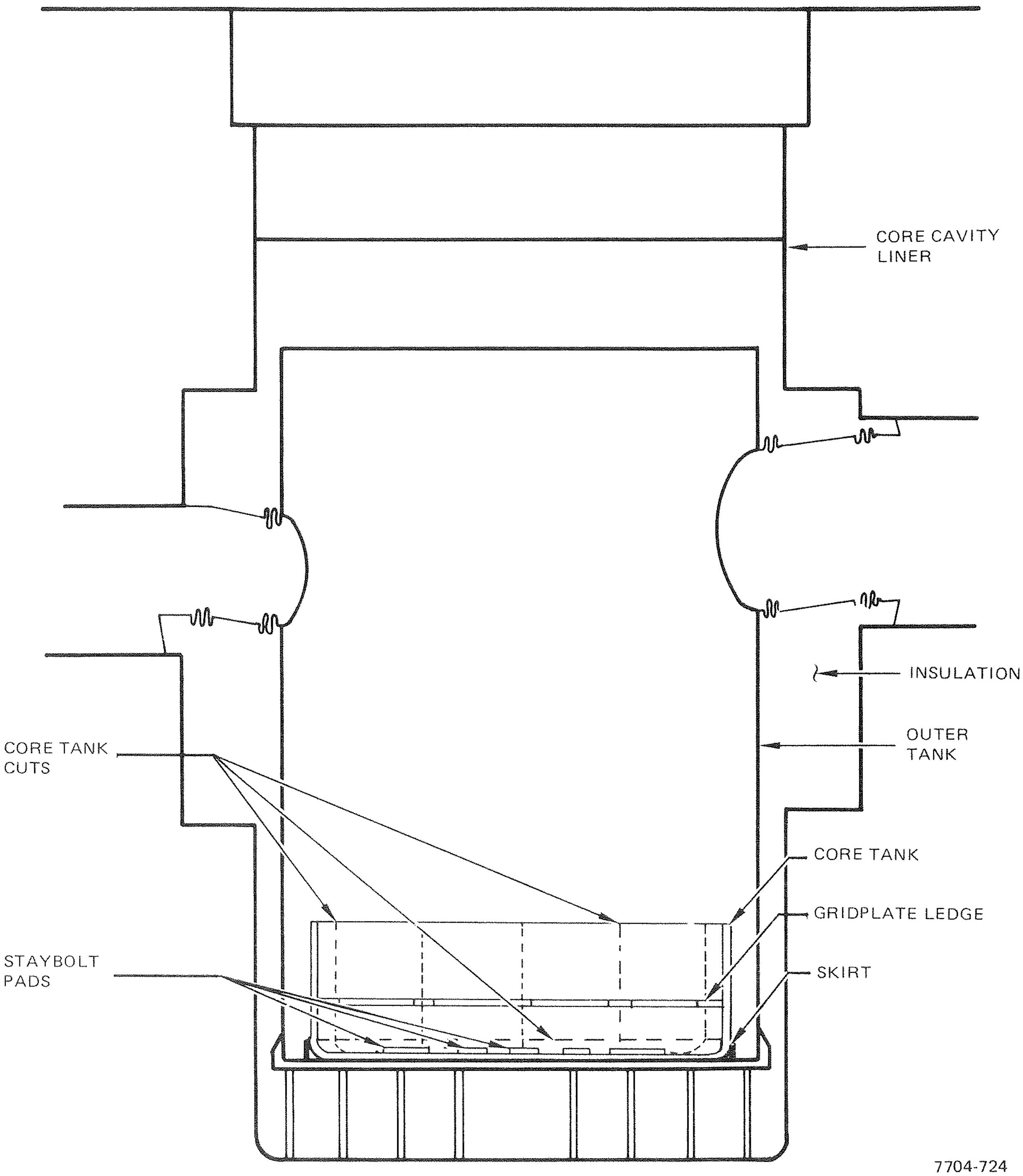

Figure 41. Core Tank Bottom Cuts

9) The outer row of segments supported by shims was cut and removed.
10) Shims were installed under the center section, and the remaining outer row segments were removed.
11) The manipulator arm was adjusted to cut between every other staybolt.
12) The staybolt segments not supported by shims were cut and removed.
13) Shims were placed under the center segment, and the remaining staybolt segments were removed.
14) The manipulator was removed from the guidepost.
15) The guidepost and center segment were removed.

No sigificant problems occurred during the removal of the core tank walls. Replacement of nozzles and electrodes were routine, underwater visibility was adequate, and cutting around the main and auxiliary inlet presented nominal challenges. Maximum segment dose rate was 45 R/h at 1 ft. Typical exposure per row of transferred segments was 25 mrem per man (see Figure 42).

The core tank bottom removal was started by placing it on shims. Cutting debris was removed to permit the installation of the manipulator and guidepost. Three days were required to lift the tank and install the shims, while it took 1 day to install the manipulator.

Cutting the grid plate ledge required removal of the grid plate perimeter studs and nuts and the tank drain line. The tank drain line was cut flush with the top surface of the ledge to permit torch clearance. Sections of the ledge surrounding the drain line were removed. Acceptable circumferential cuts were obtained by using an 8 x 12 nozzle and slowly indexing the torch.

A 2-in.-thick layer of debris on the bottom of the tank interfered with completing vertical cuts on the tank wall. Debris was scraped to the center using an arm-mounted scraper. This permitted the cuts to be completed. The

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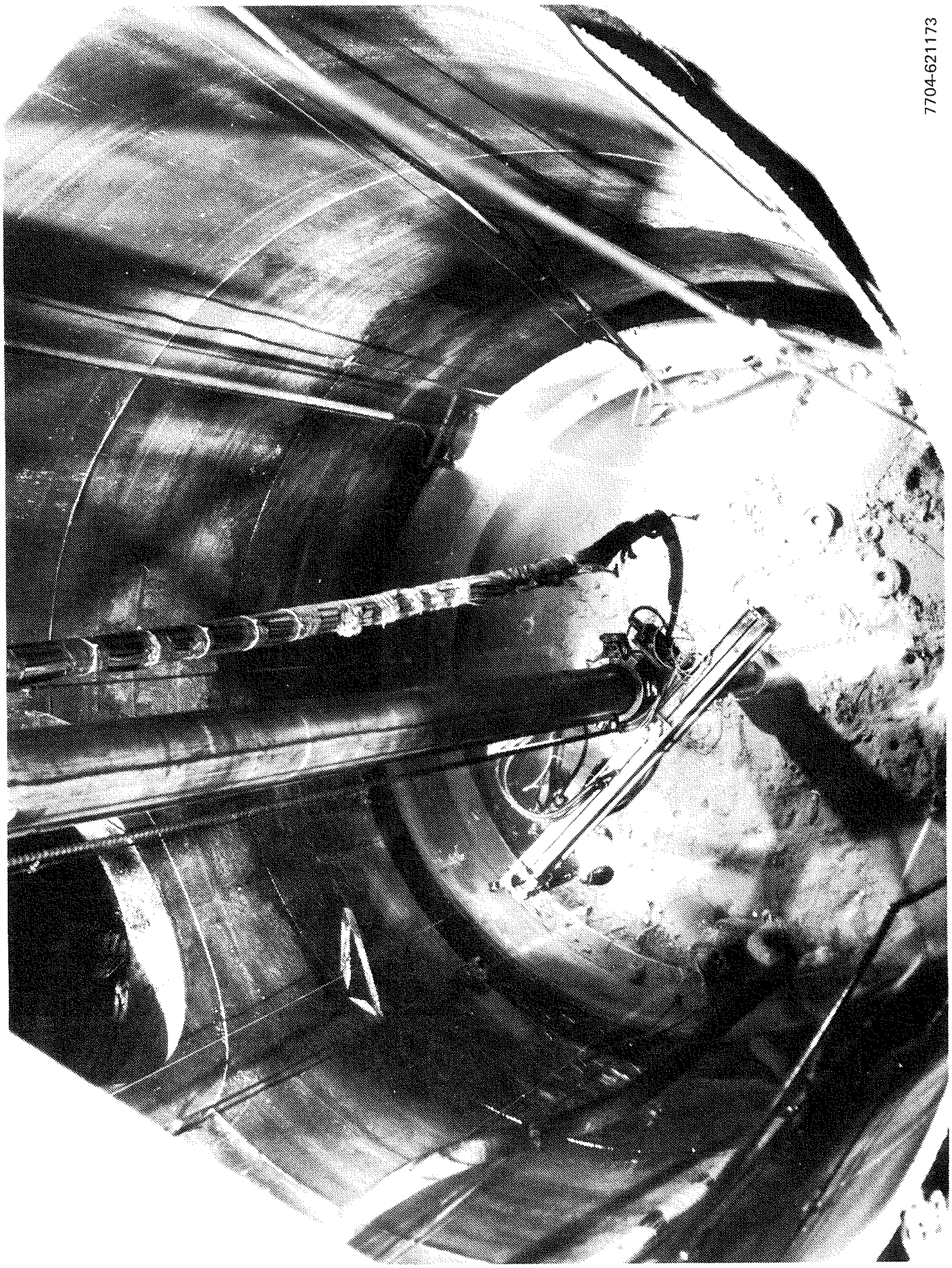

Figure 42. SRE Reactor Core Tank Wall Removal

actual skirt weld elevation was 2 in. lower than anticipated. Thus, 12 additional vertical cuts were required, which resulted in two additional small (2- by 35-in.) segments. Horizontal cuts at the skirt weld freed the 2.5-R/h segments. These were loaded into a shipping cask liner.

Several 360° scarfing cuts were made on the core tank edge to skirt weld. A 2- by 2-in. grappling hole was cut in a transition segment, and a grid plate grapple was installed. Radial, knuckle, and skirt-weld cuts permitted the segment to be peeled off of the skirt.

With the skirt out of the way, the tank transition (knuckle) segments were easy to cut and grapple. Removal of the transition segments permitted torch access to the skirt. The skirt was grappled in eight places and suspended 10 in. above the bottom of the outer tank during plasma-arc cutting. Cut sections were placed in wooden disposal containers.

The second circle consisted of six staybolt segments. The staybolts had been unthreaded and removed previously. The segments were grappled and loaded into a shipping cask liner. Typical segment dose rates were 400 mR/h at 3 ft.

The 38-in.-diameter bottom center segment of the core tank was grappled with a vertical lifting grip and placed on the floor adjacent to the reactor. The guidepost was removed and the 2.0-R/h center segment was loaded into a submerged shipping cask liner completing the core tank bottom removal operation.

4.4.9.8 Outer Tank Bellows Removal

The outer tank bellows (Figure 43) provided a flexible seal between the outer tank and the core cavity liner. It consisted of a 150-in.-diameter top cylinder welded to a 12-in.-high, 150-in.-diameter bellows assembly, and also to a 12-in.-wide flange that joined it to the cavity liner. Three conduit bellows extended from the cavity liner to the cylinder. To prevent water from contacting the insulation, all plasma-torch cuts were made in air.

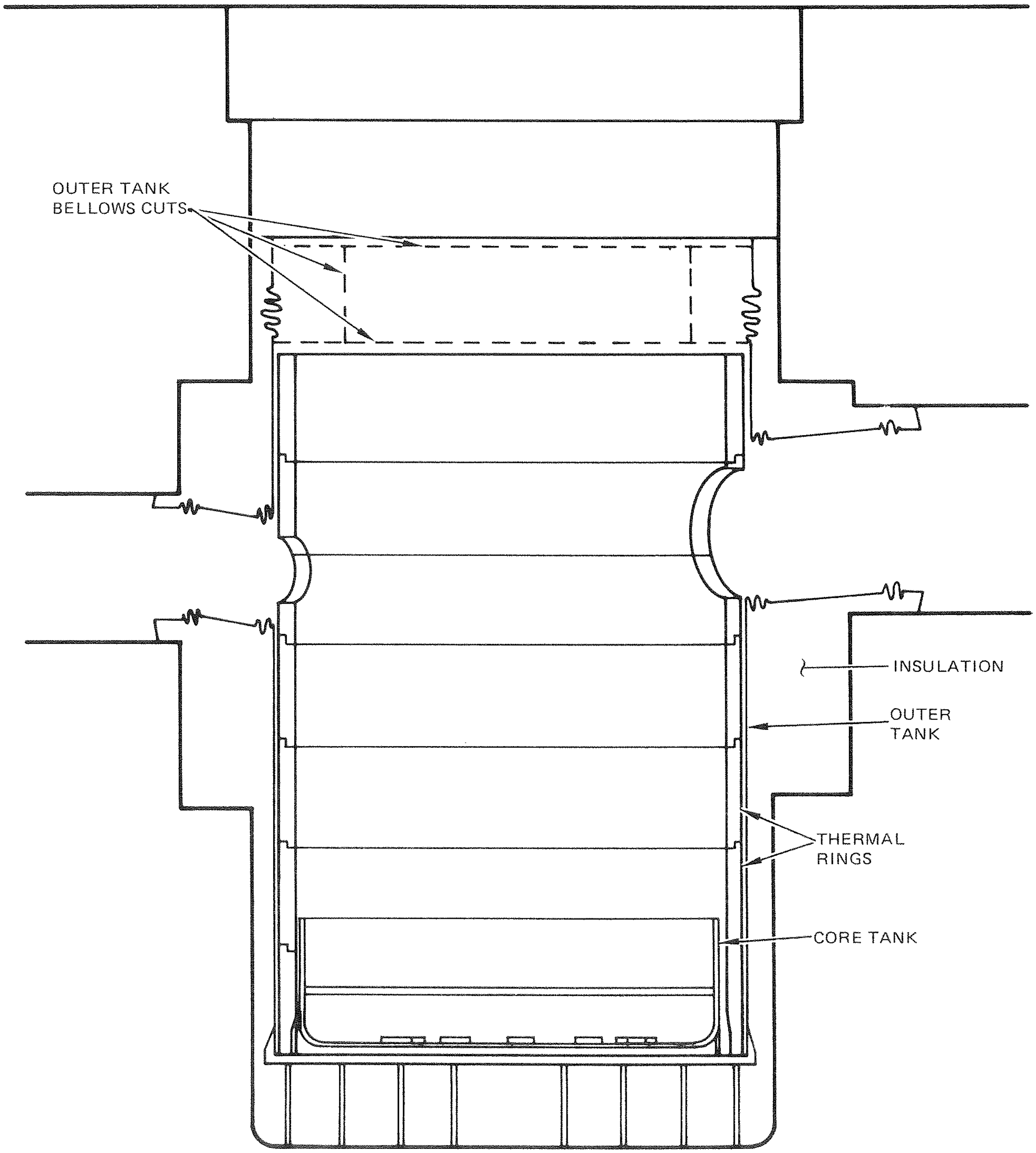

Figure 43. Outer Tank Bellows Cuts

The removal approach was to:

1) Cut the bellows free from the cylinder
2) Cut the outer tank just below the bellows
3) Cut the bellows flange
4) Cut the bellows assembly into quarters.

The radial arm with the torch-rotate mechanism was used with the plasma torch in air with slightly modified hole-piercing parameters to cut the bellows free from the cylinder. Hole piercing was repeated until all connections between the bellows and the cylinder were severed.

Next, the outer tank and flange cuts were made, then vertical cuts were made through the bellows cylinder and flange.

The four sections of the bellows were removed after the platform and manipulator had been removed. The grid plate hook sling was installed in a 3/4-in. hole in the bellows flange. A rope was installed around the end of each segment to stabilize the load. Several small connecting sections of material were found during removal but were broken by flexing the material. The segments were loaded into a wooden shipping box for disposal.

4.4.9.9 Thermal Ring and Debris Removal

The thermal rings — a stack of seven rings external to the core tank — were separated from the core tank by a 1-1/2-in. annulus. The rings, made of low-carbon steel, were 5-1/2 in. thick, 128 in. in diameter, and 33 to 37 in. high. Some rings had cutouts for vessel piping clearance.

The rings were removed by:

1) Remotely bolting the ring to the lifting fixture
2) Removing the ring using the 75-ton overhead crane
3) Placing the ring in a specially prepared cutting area

4) Installing mechanized oxyacetylene cutter and containment hood
5) Segmenting the ring into four sections.

Dose rate calculations showed that the exposure per man would be within acceptable limits. Calculations also showed that unshielding shipping of the segments would be within acceptable limits.

Debris generated during explosive and plasma-arc cutting operations was remotely removed from the thermal ring annulus, core tank, and outer tank. The debris consisted of:

1) A mixture of particles, ranging in size from 0.13 in. to 0.001 in. in diameter, generated by the plasma-arc cutting of stainless steel
2) Strips of agglomerated cutting particles (1 in. wide) generated during the plasma-arc cutting of the core tank
3) Odd-shaped strips of metal scraps generated by explosive cutting
4) Fine particulate dirt, which probably was introduced when the reactor vessel was rinsed with industrial water.

The removal operations involved:

1) Vacuuming debris using a jet-pump-type nozzle plumbed to a screen-lined collector basket
2) Using the filtration system to remove particles too small to be collected by the basket
3) Using a grapple to remove pieces too large to be vacuumed. (The grapple was designed and fabricated during SRE operations, when the nature of the removal problem was understood.)

The rings were painted with zinc chromate primer to fix surface contamination. The IDs of the upper three thermal rings were painted using a manipulator-mounted spray gun; during removal, the remaining surfaces of each ring were painted using a 6-ft-long extension wand.

To install the ring-lifting fixture, 1 h was required; 1/2 h was required to lift the ring to floor level; and 8000 lb of force was required to separate the rings. The only cutting problems involved small sections of steel or slag joining the two cut faces. These sections were either recut with the plasma torch or broken when the segments were transferred to the storage area. Two highly activated segments were loaded into the center of the shipping box and were flanked by two less activated segments which acted as shielding to meet DOT regulations (see Figure 44).

A pipe clamp/lifting tool was used to remove a fuel slug and the debris caused by explosive operations from the core tank. Most of the debris was pushed onto the grid plate ledge and then, after grid plate removal, onto the bottom of the core tank. After thermal ring removal, the dredge was used to remove cutting debris from the thermal ring annulus. Approximately three to four times a day the hose that connected the nozzle to the collector basket would become jammed with chunks of slag particles. The dredge had to be disassembled to remove the chunks. The annulus dredging operations were discontinued because of the high personnel exposures resulting from hands-on disassembly of the dredge.

A filter system was used to remove fine particulates. This system consisted of a submersible pump connected to a 3-in.-diameter flexible hose that led to the cyclone separators. The water then flowed through a booster pump, through an aggregate bed Culligan filter, and back to the reactor vessel. The system could filter water from either the reactor vessel or the storage pit. Particulates were removed from the Culligan filter by reversing the water flow, backflushing, and diverting the flow to the 500-gal radiological transfer tank. Backflushing through 10-micron-thick disposable cartridge filters did not adequately clarify the water.

To remove debris from the core tank, the first method tried was to remove the fine particulates and then scoop up the debris. A box was installed on the inlet of the submersible pump and connected to a flexible hose and nozzle. This worked well for removing particulates from the top sections of the

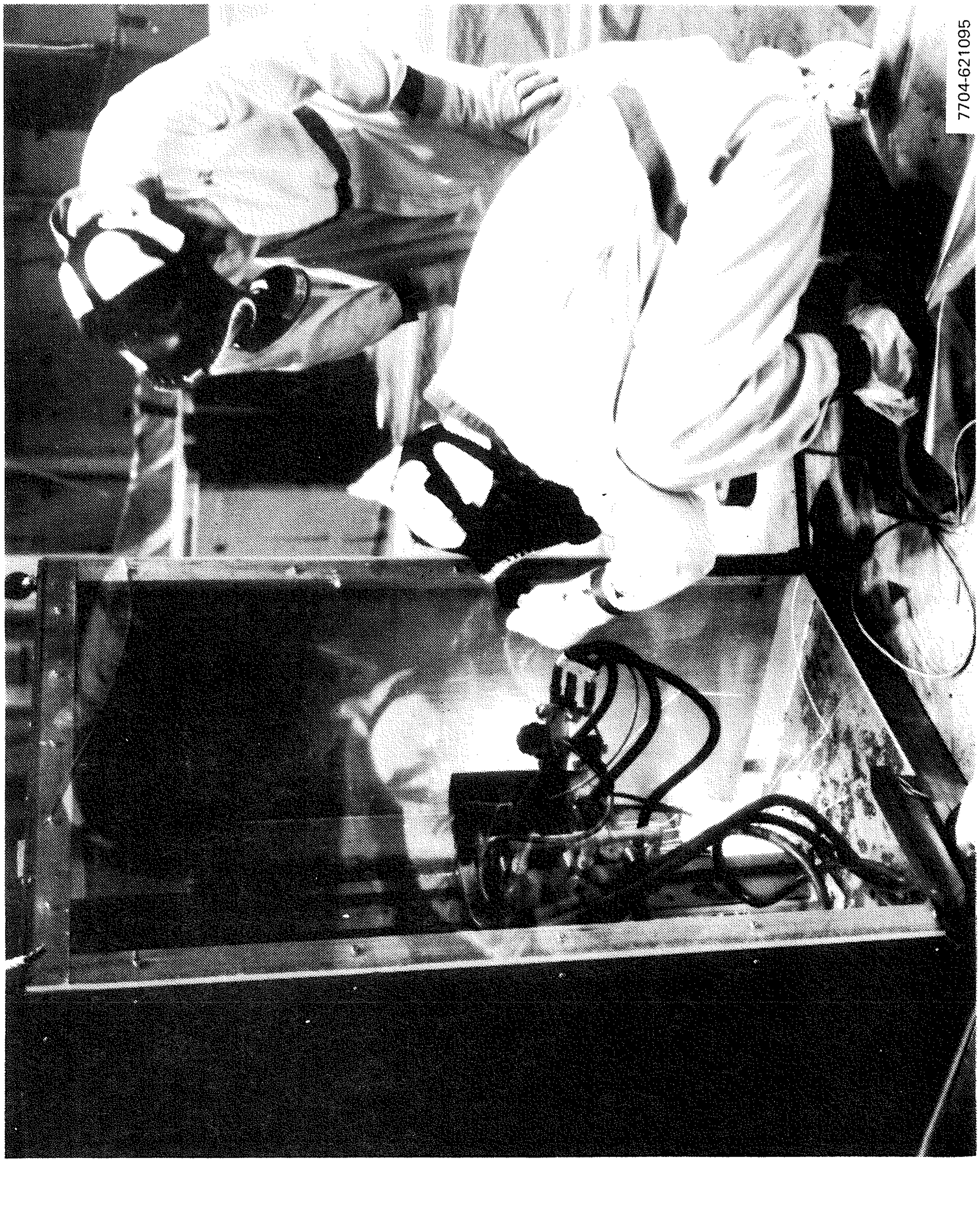

Figure 44. Thermal Ring Being Cut

slag piles, but failed to remove particulates from within the piles. A 6-in.-wide blade mounted on the down extension of the arm and the water jet from the pump were used to push the debris into a basket. Three full baskets (600 lb) were removed. This debris was visually inspected for fuel elements, but none was found.

4.4.9.10 Outer Tank Removal

The outer tank was a 15-ft-diameter, 19-ft-high, open-top vessel installed 8 ft below floor level and surrounded by thermal insulation (see Figure 45). The 1/4-in.-thick carbon steel tank was used for cover gas containment. Two horizontal corrugated cylinders (bellows) intersected the tank at 14 ft below floor level. A 3/4-in.-thick doubler plate reinforced the bellows-to-tank junction.

Removing the bottom of the outer tank consisted of removing all sections of the outer tank below the line where the tank flange was 1-3/4 in. thick. The lower half of the flange was 2 in. thick; the upper half tapered to a 1/2-in.-thick section. The carbon steel flange was welded to a 150-in.-diameter, 2-1/2-in.-thick carbon steel plate. Eight equally spaced locating cleats (2 by 2 by 9 in.) were fastened to the bottom of the tank.

To prevent water from saturating the thermal insulation, the outer tank was cut in an air environment. The guidepost was installed in the locating hole previously drilled into the bottom of the tank. The manipulator was installed on the guidepost with the radial arm bolted to the carriage in the outermost location. An asbestos cover installed over the hose bundle prevented damage from sparks. Contamination was fixed to the inside diameter of the tank by painting with zinc chromate primer. An airless paint sprayer attached to the manipulator was used to apply the paint. All platform openings were covered with plastic, and the radiological exhaust duct was connected to the platform.

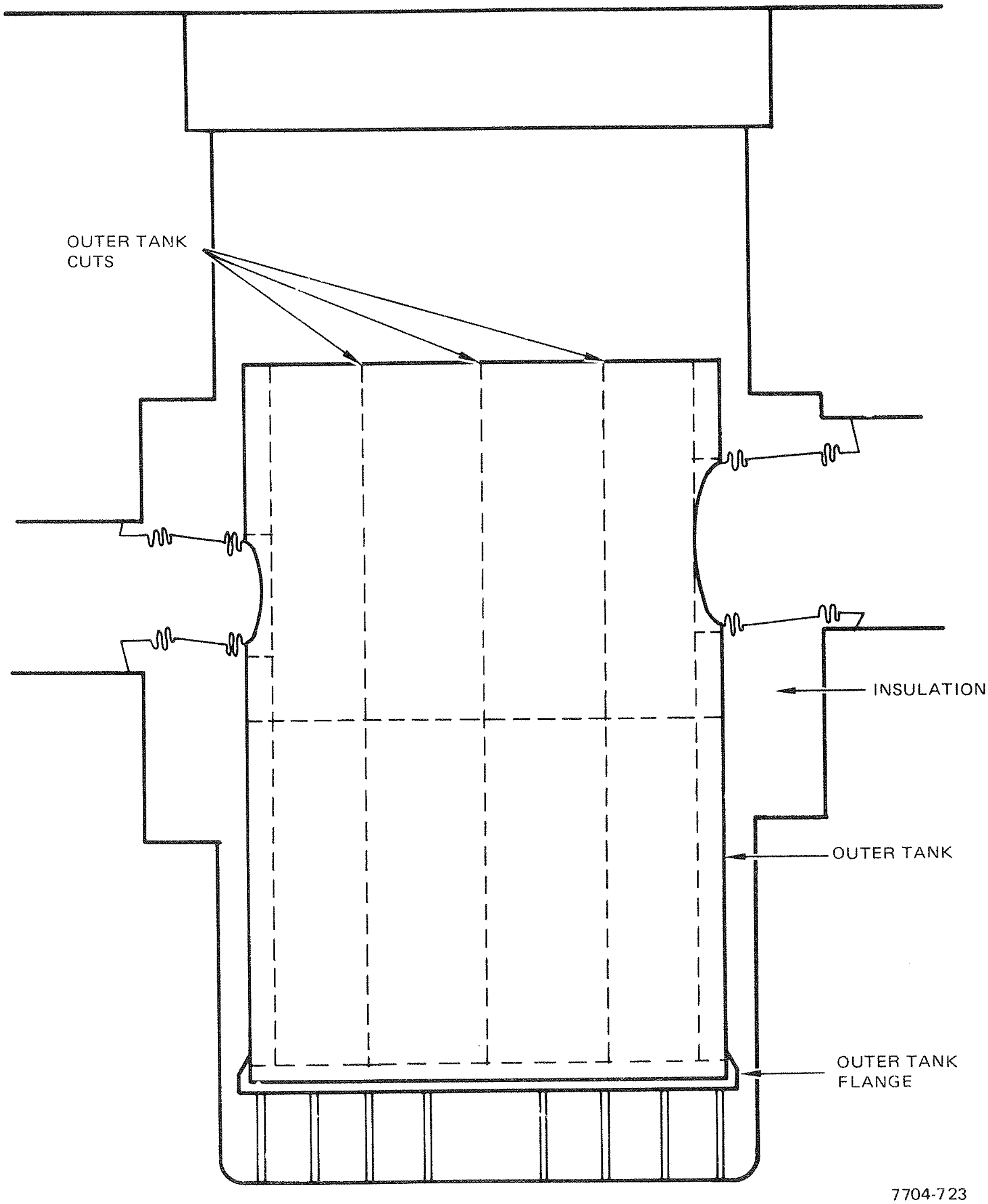

Figure 45. Outer Tank Cuts

The cutting pattern specified four rows of 55-in.-high segments, but since low activity levels permitted segments to be shipped in a wooden box, the height was doubled. Segments from the top row read 25 mR/h at 1 ft, and segments from the bottom row read 45 mR/h at 1 ft. Cutting the doubler plates around the bellows required using cutting parameters for 1-1/2-in.-thick stainless steel.

The outer tank bottom flange taper dimensions were not the same as indicated by the drawing dimensions. As a result, the horizontal cut severed the flange where its thickness was too great for lifting grips to be installed easily. Several grips were inadvertently installed in the wrong locations and could not be remotely removed. This caused the cutting pattern to be revised and produced two double and several half-size segments.

Cutting and removal operations were almost flawless. Changing the arm to carriage location produced a few uncut tabs at the cut start/stop points. The segment removal sequence was simplified by removing all but one segment. The 46-in.-diameter center section with an attached segment was loaded into an extra large shipping box. The maximum segment dose rate was 1 R/h at 1 ft. This concluded remote dismantling operations.

4.4.9.11 Insulation Removal

A 9-in.-thick layer of insulation covered the inside diameter of the core cavity liner. It was held in place by wires connected to studs welded to the cavity liner. Remote removal using the plasma-torch manipulator system was considered, but manual operation was selected.

Insulation tiedown wires were cut manually, and the 3- by 8- by 36-in. blocks were stacked in a wooden shipping container. Removal of 1100 ft^3 of wall insulation required 3 days. Removal of the floor insulation required 2 days.

4.4.9.12 Core Cavity Line Removal

The reactor core cavity liner formed the inside surface of the biological shield. The 1/4-in.-thick steel liner was attached to the biological shield by the kerosene cooling coils which were welded to the liner and embedded in the concrete.

After all of the core components, tanks, and insulation were removed, the reactor core cavity liner was removed. The upper 3/4 section of the 1/4-in.-thick steel liner was cut into sections using a hand-held oxygen-acetylene torch. The lower 1/4 section of the core cavity liner was cut with a torch and pryed loose from the biological shield using a hydraulic Hy-Ram. Removal of the liner was phased with the biological shield removal to take advantage of having the concrete removed from behind the liner. The sections were packaged in wooden crates for disposal.

4.4.10 Excavation of Contaminated Soil

The primary sodium fill tank vault and the sodium service vault were located directly north of the reactor building high-bay area. During reactor operation, the concrete surfaces of these vaults became contaminated. Decontamination was best accomplished using the Hy-Ram and working inward from the outside of the vault. This required that an access ramp be excavated at the northeast corner of the SRE high bay. Figure 46 shows the initial excavation to expose the outside wall of the sodium service vault. Figure 47 shows a portion of the east exterior wall of the primary fill tank vault. This was a common wall between the two vaults. The numbered grids defined the interior wall of the sodium service vault.

During this excavation, contamination above the Table 3 limits was found in the soil surrounding the footing at the northeast corner of the high bay. Further excavation uncovered contamination as high as 13,000 pCi/g located in cracks in the bedrock east and north of the high bay (see Figure 48). Figure 49 shows the excavation along the east face of the building. Contaminated

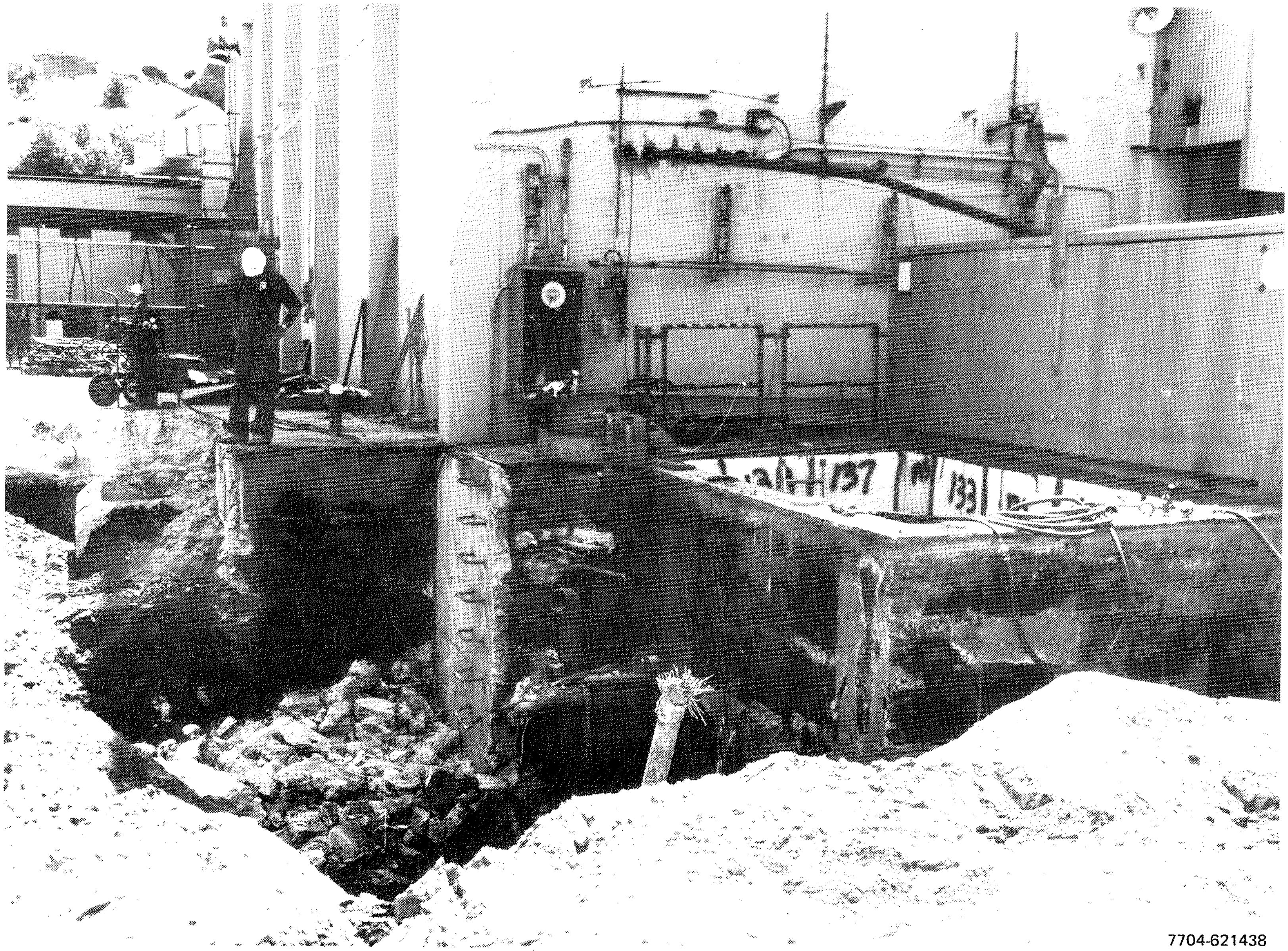

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Figure 46. Initial Excavation at Northeast Corner of the SRE High Bay

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Figure 47. Excavation East of Primary Fill Tank Vault

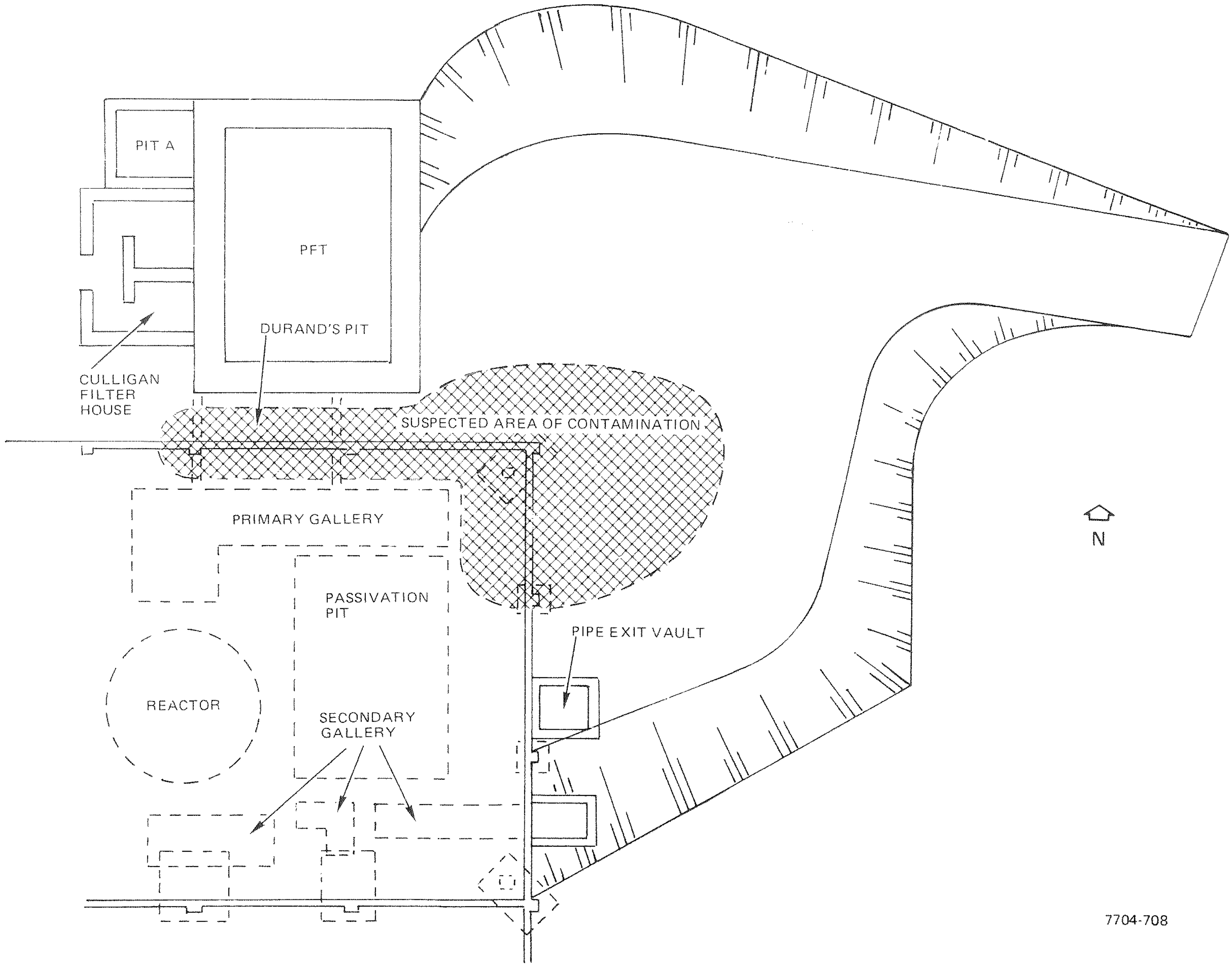

Figure 48. Access Ramp to High-Bay Excavation and Suspected Area of Contamination

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Figure 49. Excavation East of SRE High Bay

soil and bedrock were packaged for disposal in tri-wall cardboard "King-Pac" containers. The containers, which were mounted on a plywood skid, held approximately 2000 lb each. All contamination above the release limit was removed from the area. The area was backfilled to provide machine access for further decontamination activity within the high bay.

Decontamination of the sodium service vault and the primary fill tank vault was completed using the Hy-Ram. Whenever possible, selective, rather than bulk, removal approaches were used to minimize the waste to be packaged and sent to burial. It was shown that the Hy-Ram could be equipped with chisel-like tools that could peel off selected layers of contaminated concrete in vaults and walls. Figure 50 shows the Hy-Ram in operation.

After the sodium service vault was demolished, access was provided to the interior of the primary fill tank vault. Contaminated surfaces were removed with the Hy-Ram. The east, south, and west walls were demolished and the clean rubble was stored onsite for future use as backfill material. The floor and the north wall remained and were subseqently buried during backfilling of the access ramp. Figure 51 shows the demolition of the primary fill tank south wall and the extent of the excavation along the east side of the high bay.

Access to the north and east exterior surfaces of the reactor biological shield was provided by demolishing the main and auxiliary primary vault walls. Prior to this time, it was decided to excavate the entire SRE high bay. This would provide equipment access to the remaining below-grade components that had to be removed. These included the fuel storage cells, wash cells, moderator element storage cells, and fuel element enclosure sleeves. Prior to the start of soil excavation, a consulting civil engineering firm was hired to evaluate the impact of the future excavation on the structural integrity of the building. The recommendations from the consultant were as follows:

1) Excavation work should not begin until completion of facility improvements.

Figure 50. Hy-Ram in Operation

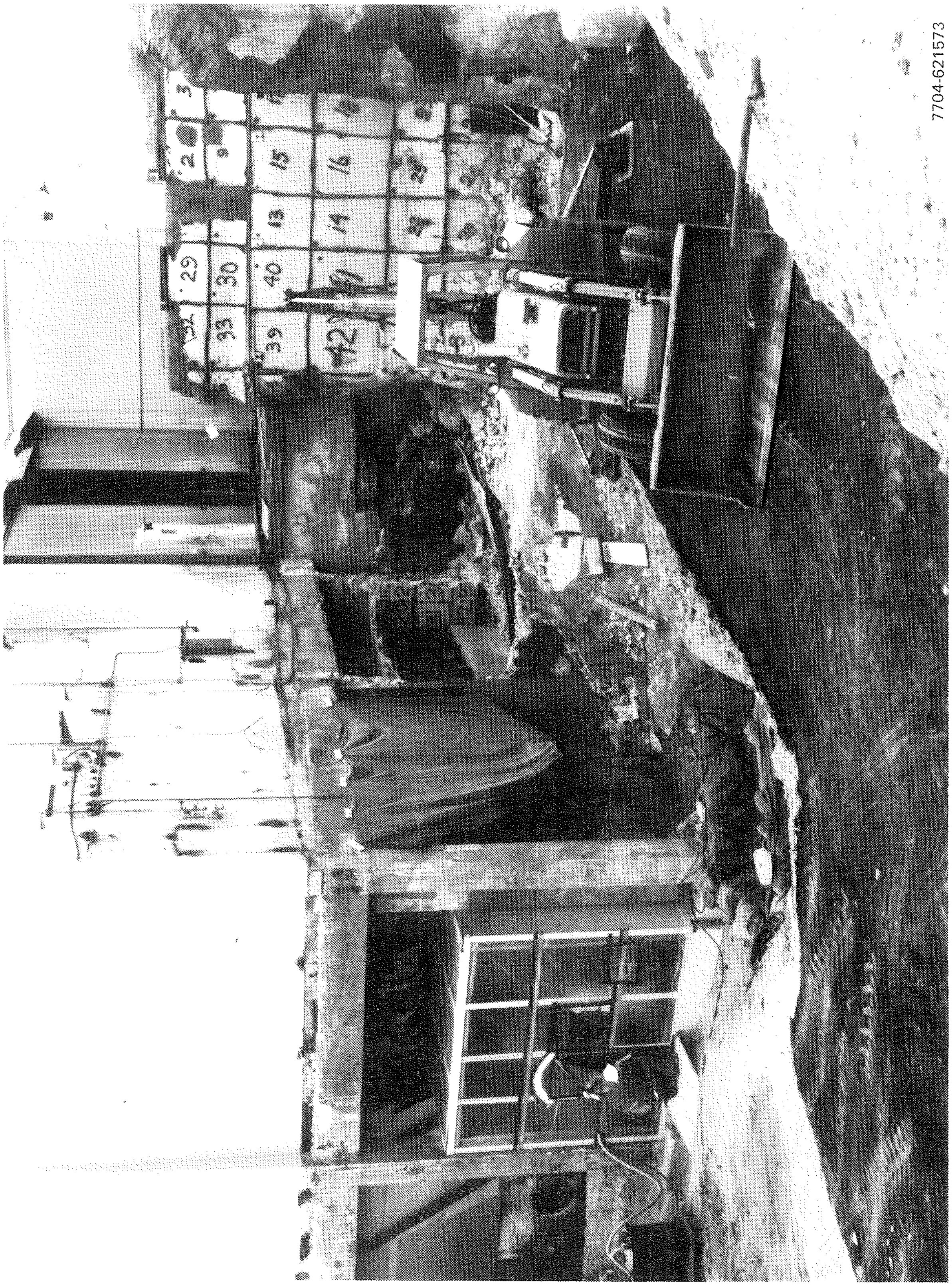

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Figure 51. Demolition of Primary Fill Tank Vault

2) The 75-ton bridge crane should be down rated to 5 tons and stored at the far west end of the high bay when not in use.

3) A retaining wall should be built at the southern limit of the excavation in the high bay between the hot cell wall on the west and the auxiliary secondary gallery wall on the east (see Figure 52). The retaining wall was to consist of 50-ft-deep, steel-reinforced concrete pilings with 4 by 12 wooden shoring secured in channels imbedded in the piles. Figure 53 shows the installation of the wooden shoring. As the excavation deepened, additional shoring could be added.

4) Seismic bracing had to be added when the excavation was completed.

5) Additional bracing had to be added if excavations went below the bottom of a footing.

The pilings for the retaining wall were installed while the excavation of the access ramp was in progress. The ramp provided equipment access for the concrete vault demolition and the high-bay excavation. The SRE high-bay excavation lowered 90% of the high-bay floor area an average of 20 ft. Six seismic braces were added. Four 8-in.-diameter steel pipes tied the grade beam under the north wall to the floor of the primary fill tank vault. A single 8-in. pipe bisected the northeast corner from the grade beam to a new footing down in the excavation. Another 8-in. pipe tied the grade beam under the east wall to an existing concrete footing, located at grade, east of the high bay.

Contamination was found in the bedrock near the primary fill tank vault and the main primary vault. The bedrock in this area consisted of multilayered sandstone. Numerous cracks and slip planes were visible. Individual cracks could be traced for 10 ft or more along the surface. Contaminated liquids had come into contact with this particular rock formation and traveled along numerous pathways. The source of the contaminated liquid was probably surface and ground water and water leaks from a temporary storage pool in the primary pipe vault.

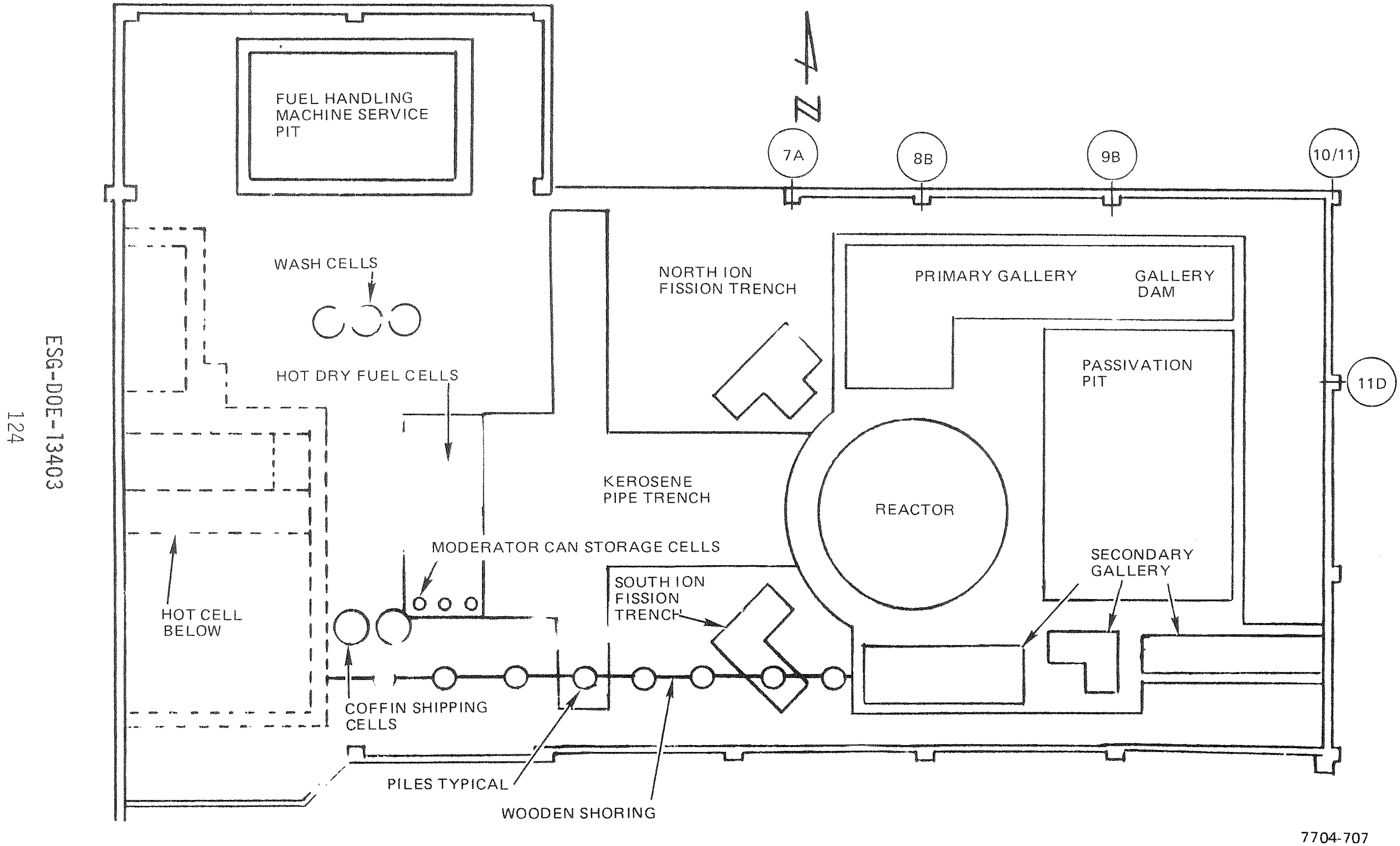

Figure 52. Location of High-Bay Retaining Wall and Column Identification

Figure 53. Installation of Wooden Shoring in High-Bay Retaining Wall

The main primary pipe vault was used for underwater storage during reactor vessel disassembly. It was constructed of concrete with a few pipe penetrations. In preparing for the work, the vault was sealed watertight. The surfaces were cleaned, and the cracks were filled. The pipe penetrations were sealed, and the entire surface was coated with an epoxy material. Despite the preparation, a water leak developed during the segmentation project. It may have resulted from the explosive cutting operation. The leak resulted in water containing fission and activation product contamination being released into the soil at the north and east quadrant of the building. As soon as the loss of shielding water was noted, a steel tank was fabricated to fit inside this vault, and contaminated material and water were transferred to the tank.

Contaminated rocks and soil were found around five facility support column footings. Three footings were along the north wall (7A, 8B, and 9B). One was along the east wall (11D). And one footing was at the corner where the north and east walls meet (10/11) (see Figure 52). The structural consultant previously contacted was again brought in on contract. A series of temporary footings, cross bracing, and support columns needed to be installed if excavations were required near or below these five column footings. Exploratory excavations in this area were performed using jackhammers and hand tools. Rock and soil samples were analyzed, and the results were compared with previous samples. Extrapolation was used to estimate the depth of radioactive contamination in several visible cracks. Based on this information, it was apparent that the only solution was to remove the contaminated soil from beneath the footings.

Three levels of effort were suggested by the consultant:

1) A footing could be partially undermined to remove local soil contamination
2) The footing could be completely undermined to remove deep soil contamination
3) The column and footing could be removed to provide access for deep excavation.

The level chosen for a specific column footing was based on the extrapolated amount of contamination expected. One column was partially undermined (10/11). Two columns were totally undermined (7A and 11D). And two columns and footings were removed and later replaced (8B and 9B). The installation of temporary footings and columns was subcontracted. Only one column was worked on at a time, and a full, 28-day cure was permitted on a foundation or column before work was started on an adjacent footing. Figure 54 shows the column supports and the seismic braces anchored to the floor of the primary fill tank vault. Figure 55 shows the column bracing inside the high bay. Column 7A, on the left, already has a new foundation and reinforced footing in place. Note that Column 9B has been removed.

The excavation east of Column 11D was opened once access through this area was no longer required. The contaminated bedrock directly under the column footing was removed and a new concrete foundation was installed. Excavation of contaminated soil was continued east of the facility. A 6- by 10- by 12-ft-deep hole was excavated below Column 11D to remove all of the contaminated bedrock. Figure 56 shows a part of this excavation.

The column support bracing remained in place until the high-bay excavation backfill and compaction was started. The seismic bracing, which was anchored above the lower level of the excavation, remained in place until the backfill soil reached the level of the primary fill tank floor.

4.4.11 Biological Shield Demolition

The SRE biological shield was a solid, high-density concrete structure surrounding the core cavity liner. The shield was 4 ft thick, with the exception of the pipe bellows area (see Figure 5). Prior to the start of demolition activities, a biological shield removal methods trade study was performed. Three methods were evaluated: (1) using the large hydraulic Hy-Ram, (2) using standard jack hammers and scabbling tools, and (3) using conventional explosives. To aid the study, six core samples were taken of the biological shield and analyzed for activation. The samples were obtained

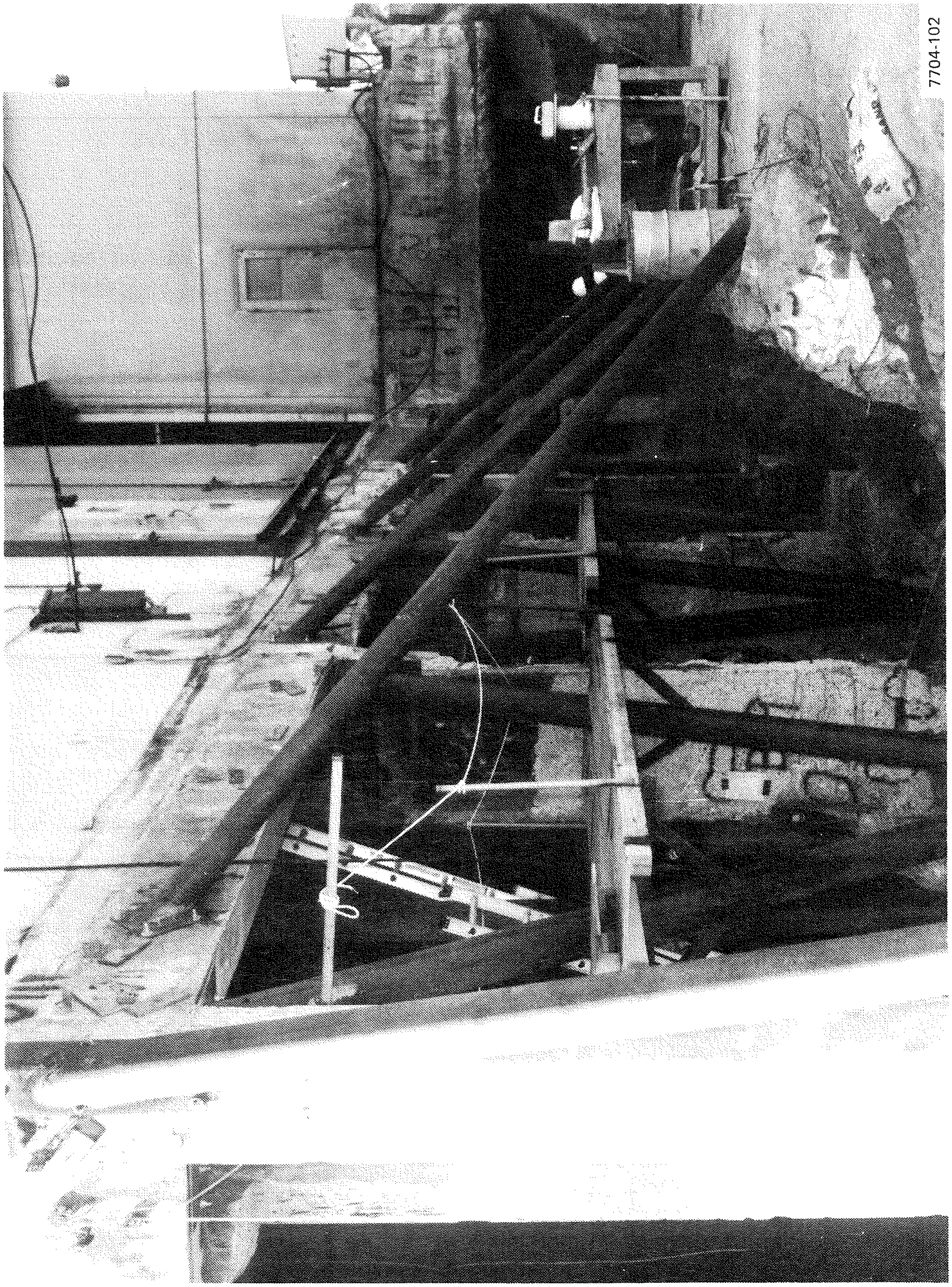

Figure 54. Column Supports and Seismic Bracing Installed under North Wall of SRE High Bay

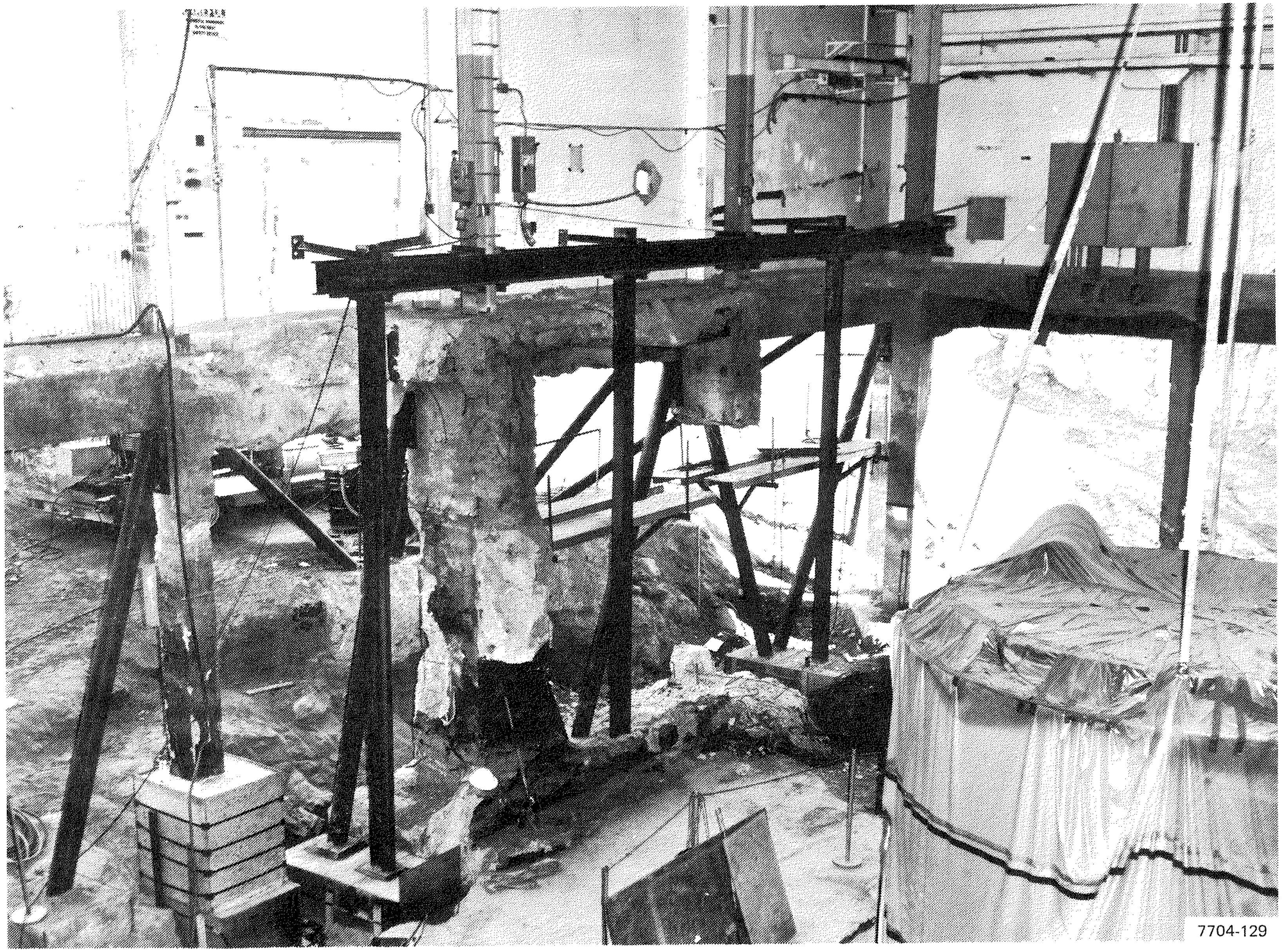

Figure 55. SRE High-Bay North Wall Column Supports Interior View

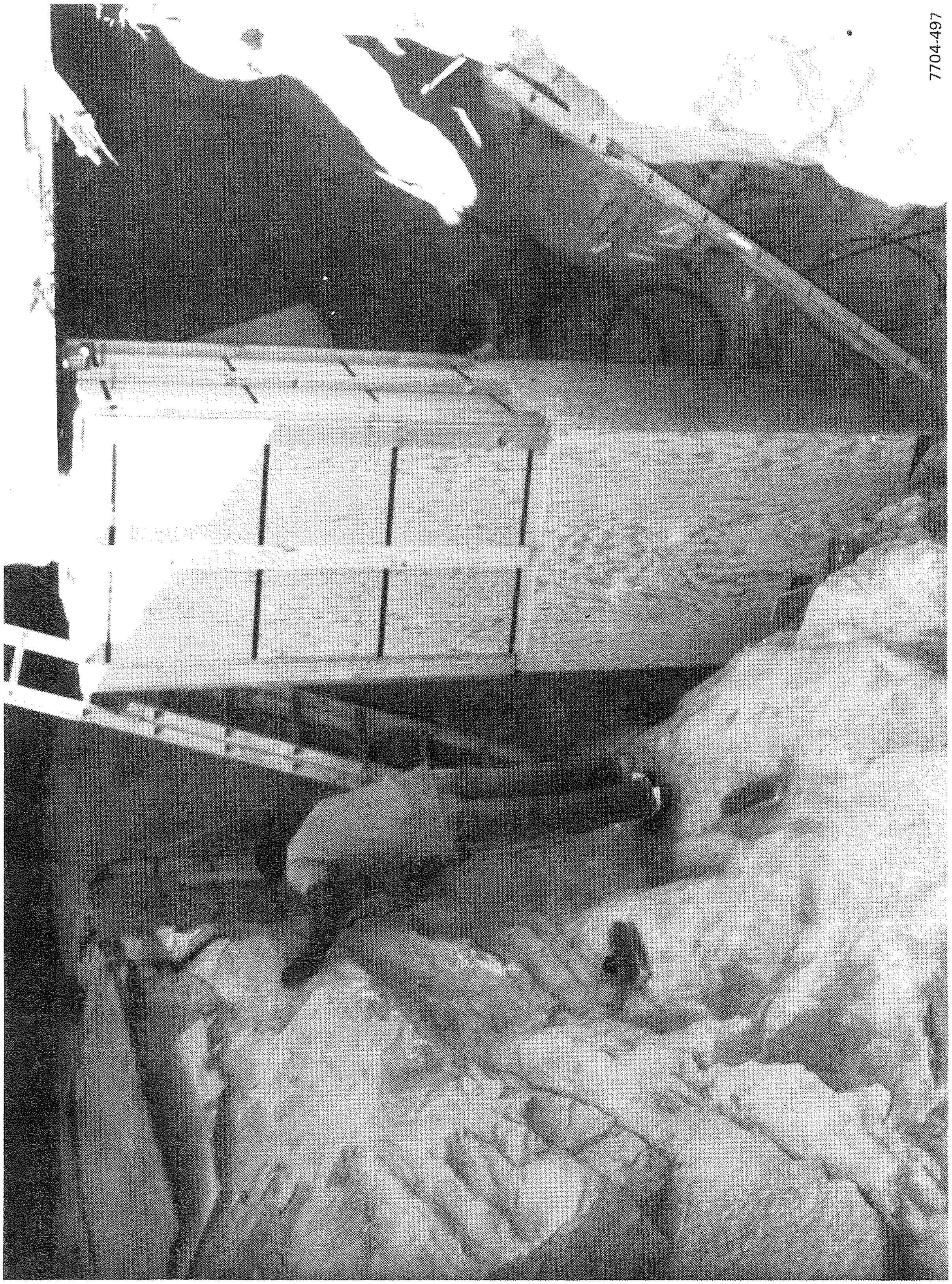

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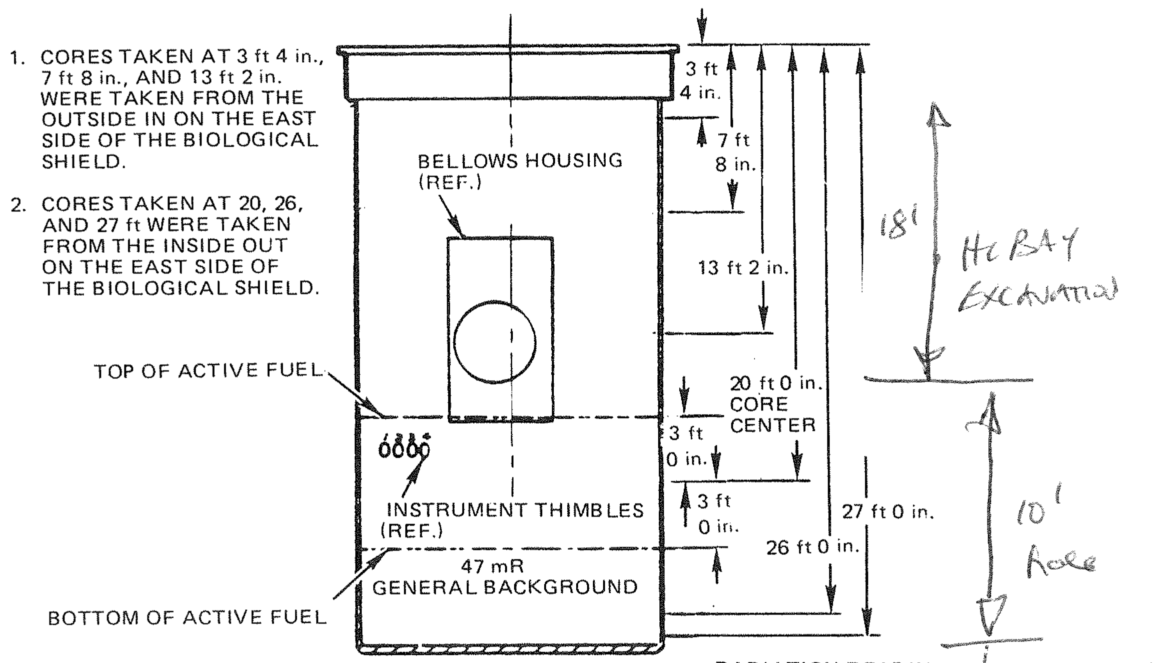
Figure 56. Construction of Form for Replacement Foundation under High-Bay Column 11D

using a Milwaukee heavy-duty drilling rig with a 2-in.-diameter by 12-in.-long, diamond-impregnated core drill.

The first three samples were taken at 3 ft 4 in., 7 ft 8 in., and 13 ft 2 in. below-grade level. The drilling was accomplished from the outside of the biological shield to the inside. The next three samples were removed from 20, 26, and 27 ft below grade and were drilled from the inside of the biological shield out. The sample locations and their relationship to the reactor core are shown in Figure 57.



RADIATION READINGS AT CORE HOLES

ELEVATION (ft)	AT SUPPORT (mR)	AT 1 meter (mR)	AT CENTERLINE (mR)
20	200	100	47
26	65	65	75
27	30	45	47

RADIATION READIN AT INSTRUMENT THIMB

INSTRUMENT THIMBLE	1	2		
NORTH SIDE (mR)	250	400	250	200
SOUTH SIDE (mR)	250	400	250	250

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Figure 57. Biological Shield Core Sample Locations

Samples for radiometric analysis were removed from the concrete cores using a hand hacksaw with a tungsten carbide blade. Radiometric analysis was performed using a thin window gas proportional counter having a background of approximately 20 counts per minute (cpm). A 1-g KCl standard was used to determine counter efficiency. The data generated from the analysis are presented in Table 12.

The accumulated data taken from the six core samples indicated that the removal of approximately 24 yd^3 of rubble was required to achieve the 100-pCi/g limit in both the concrete and steel reinforcing bar. This volume of concrete is small compared with the total volume of the biological shield, which would represent approximately 1418 yd^3.

The excavation of contaminated soil from the northeast corner of the high bay provided access to the exterior surface of the biological shield. Thus, it was decided to use the hydraulic Hy-Ram to remove the concrete from the biological shield. First, the clean soil was removed from around the outside surface to a depth of 18 ft below grade. Starting at the top, the Hy-Ram removed the nonactivated concrete from the outside in. Health, Safety & Radiation Services provided concrete sample analysis as new concrete surfaces were exposed. Prompt sample analysis combined with the core drilling data enabled the Hy-Ram to selectively remove nonactivated concrete from the biological shield without exposing the activated material close to the core cavity liner. This resulted in considerable cost savings in the packaging, transportation, and burial of contaminated waste.

Removal of contaminated material required some modifications to the concrete removal procedure. A plastic tent was suspended from a plywood platform around the entire biological shield. The plywood platform was held by an overhead bridge crane. The facility radioactive exhaust duct was connected to the tent to provide a negative pressure. A high-pressure, low-volume water spray was used to help control dust when the Hy-Ram was operating. With this arrangement, it was possible to demolish the activated concrete with a minimum release of airborne contamination.

TABLE 12

SRE BIOLOGICAL SHIELD RADIOMETRIC ANALYSIS

Core Location (Below Grade)	Sample	Sample Location (In Inches From Core Cavity Liner)	pCi/g (Gross Beta)
3 ft 4 in.	1	1/2	1.5
	2	3	NAD[a]
	3	6	7.6
	4	12	NAD[a]
	5	18	NAD[a]
	6	24	7.6
7 ft 8 in.	1	0	125.0
	2	6	32.2
	3	12	17.6
	4	18	27.9
	5	24	24.8
13 ft 2 in.	1	1/2	220.0
	2	1-1/2	348.0
	3	3-1/2	118.0
	4	5-1/2	128.0
	5	9-1/2	9.2
20 ft 0 in.	1	1	3300.0
	2	2-3/4	4780.0
	3	6	1350.0
	4	12	228.0
	5	15	93.9
	6	18	30.5
	7	23	18.2
26 ft 0 in.	1	1	503.0
	2	6	164.0
	3	8	72.0
	4	11	40.4
	5	18	14.8
27 ft 0 in.	1	1	150.0
	2	6	63.0
	3	11	29.7
	4	18	27.7
	5	23	27.7

[a]NAD = No activity detected.

Once the biological shield was removed to the level of the high-bay excavation (approximately 18 ft below grade), the core cavity liner was removed (see Figure 58). The remaining portion of the biological shield continued below the bottom of the excavation for another 16 ft. Activated concrete in this region was removed from the inside out. Figure 59 shows the remaining biological shield just prior to backfill and compaction. Archive samples were taken from the walls and floor and remain at the Santa Susana Field Laboratories.

4.4.12 Below-Grade Component Removal

Within the SRE high bay were several operational components that were located below grade. Among them were the hot dry fuel cells, the moderator can storage cells, and the wash cells (see Figure 52). The fuel element enclosure sleeves, located in the primary hot cell, and the main and auxiliary primary pump dip legs, located in their respective vaults, were also below grade. The two fuel element enclosure sleeves and the two dip legs were the deepest components since their top surface was approximately 20 ft below grade.

4.4.12.1 Fuel Storage Cells

There were 96 fuel storage cells located at the SRE. They were used to store dry reactor fuel elements. The cells consisted of steel tubes, 5-1/4 in. in diameter by 22 ft 6 in. long. The tubes were arranged in 6 rows with 16 tubes in each row, all encased in a concrete monolith. Each tube had its own kerosene cooling line. The fuel storage cells internals were removed and packaged for burial early in the SRE decommissioning. The tubes remained in place until concrete surface decontamination could be completed in that area of the high bay. Once access was available, the Hy-Ram was used to fracture the concrete encasing the tubes. A steel plate anchored each storage tube to the concrete at 5 ft below grade level. Once this plate was cut, all of the tubes were easily removed. The fuel storage cells were cut into 6-ft lengths, packaged in boxes, and shipped to land burial. Figure 60 shows the Hy-Ram fracturing the concrete around the upper portion of the fuel storage

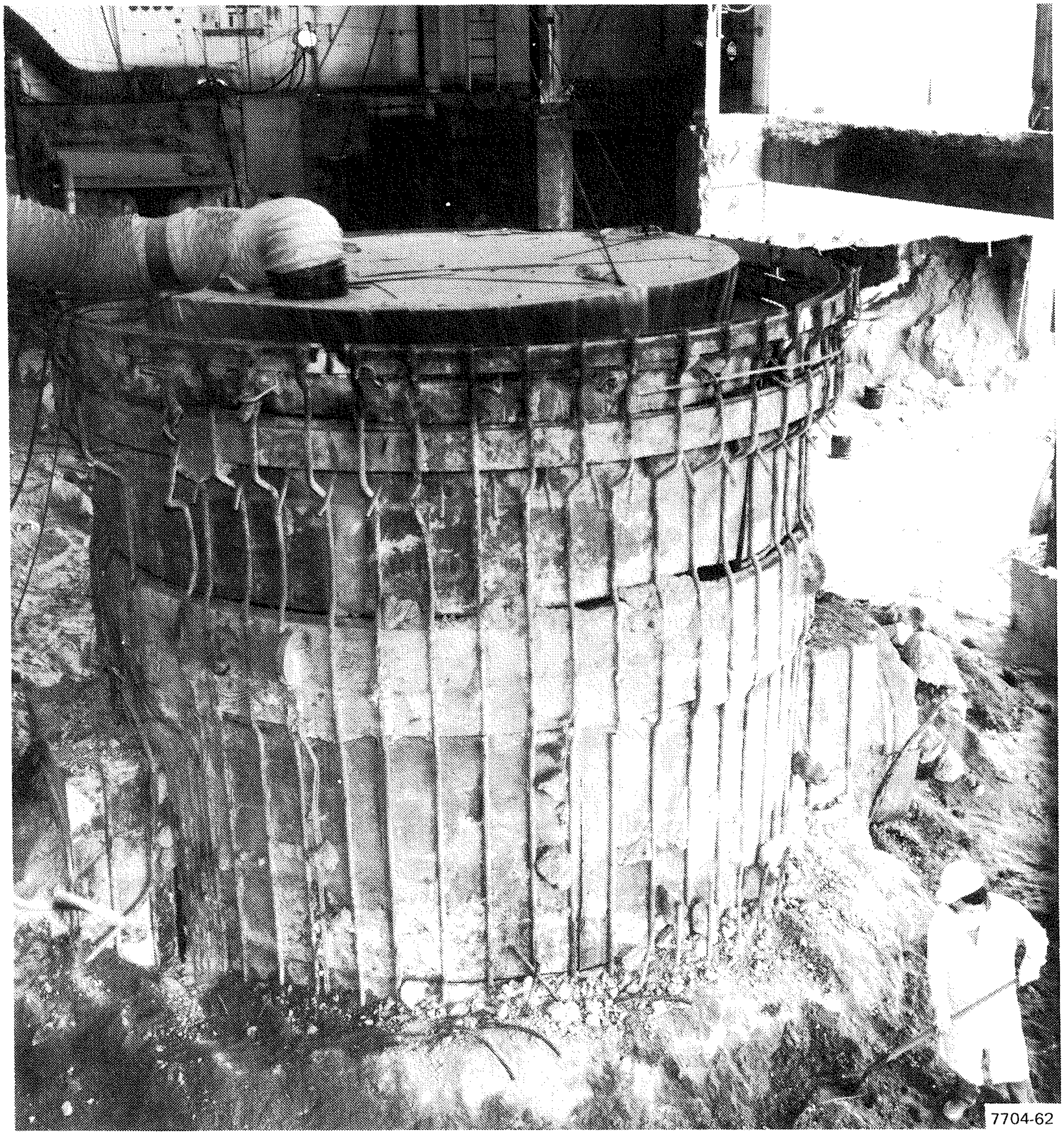

Figure 58. SRE Core Cavity Liner after Demolition of Biological Shield

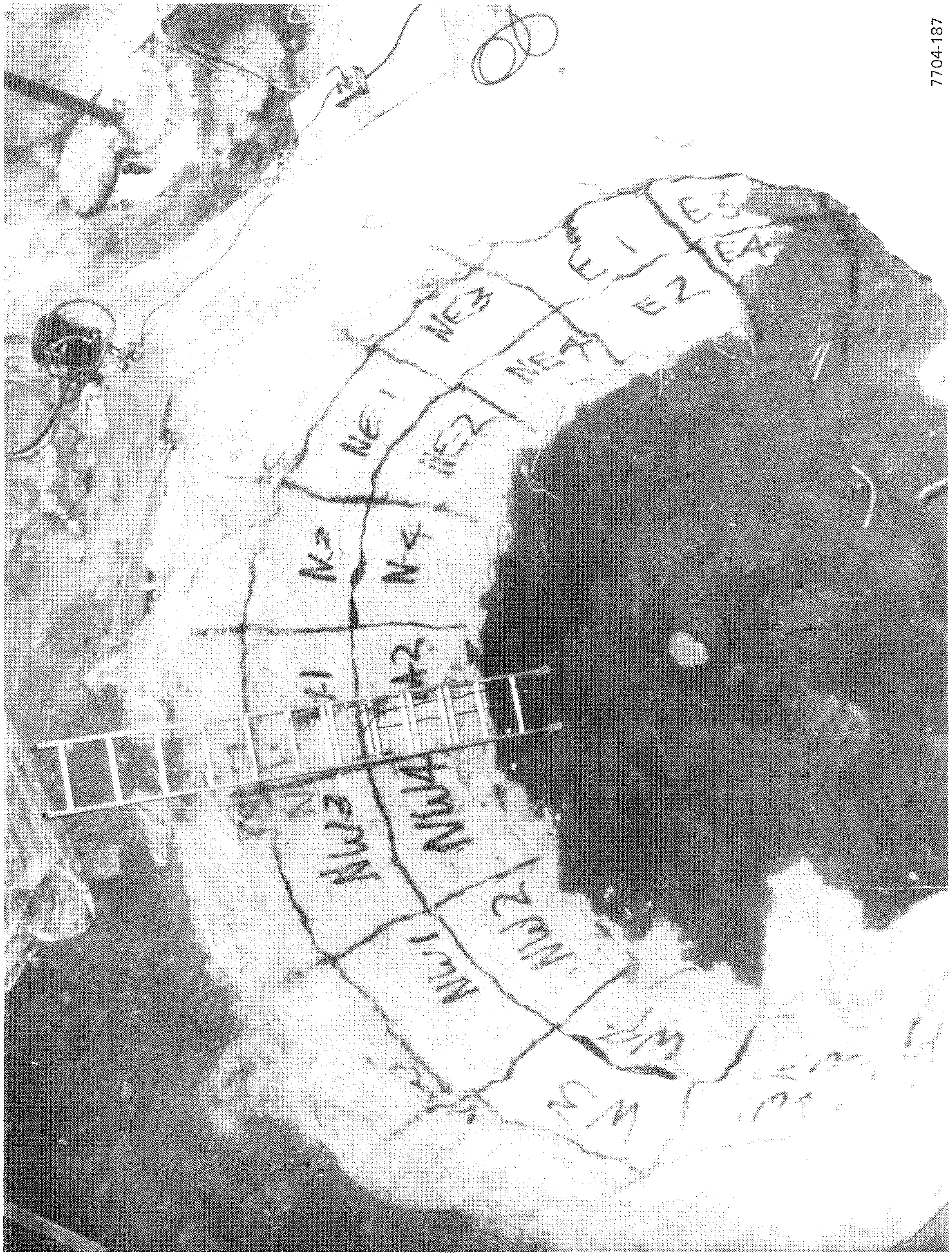

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Figure 59. Biological Shield Cavity

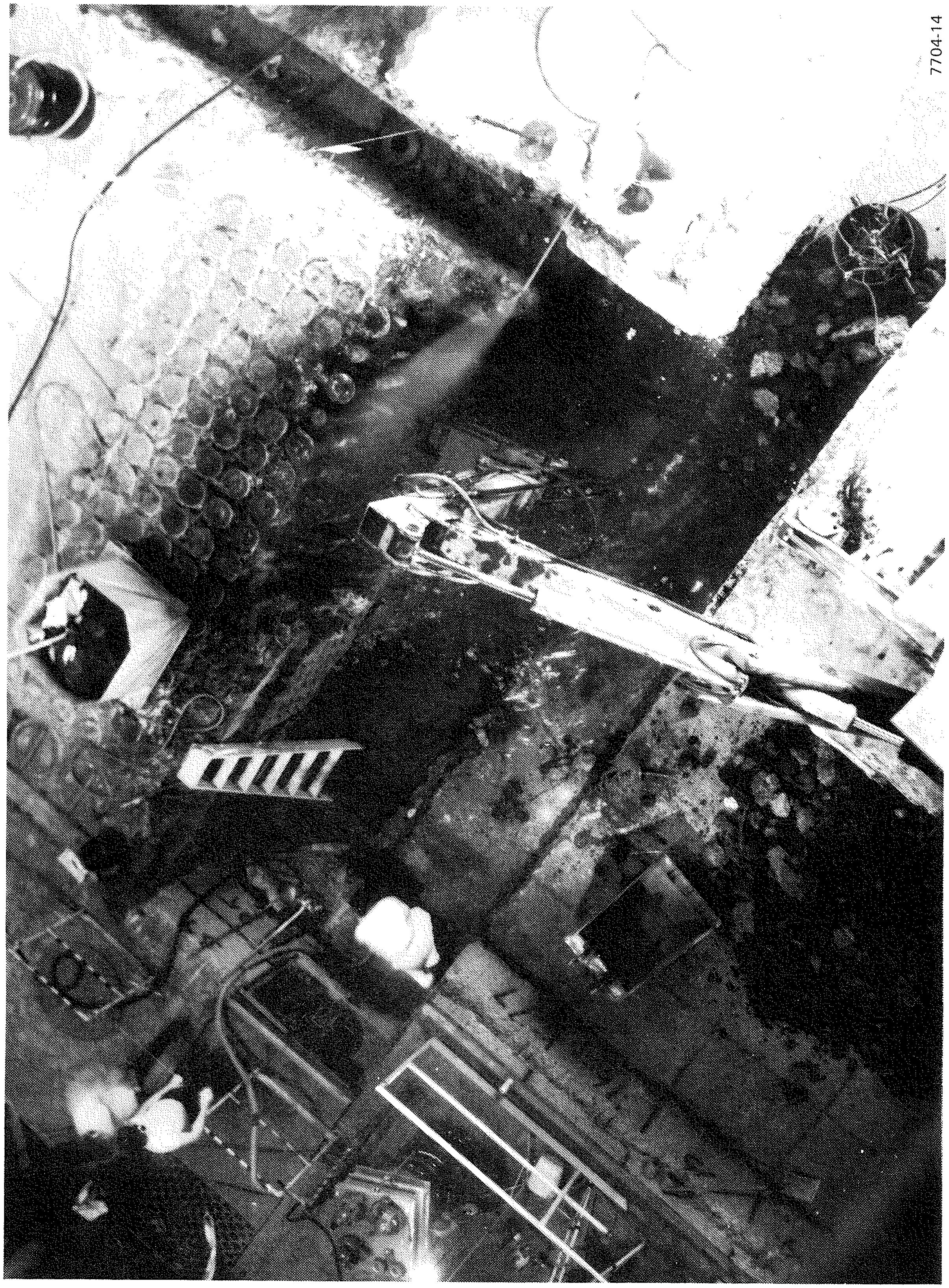

Figure 60. Hy-Ram Demolishing Concrete around Fuel Storage Tubes

cells. The high-pressure, low-volume water spray was used to control dust. Figure 61 shows a health physicist surveying a fuel storage tube as it is removed. The kerosene cooling line can be seen along the side and looping around the bottom of the tube. Figure 62 shows the demolition of the remaining fuel storage cell concrete structure after the tubes were removed. One tube can be seen next to the moderator element storage cells. Note the detail of the retaining wall construction.

4.4.12.2 Moderator Element Storage Cells

The three moderator element storage cells were located adjacent to the fuel storage cells. They were larger (20 in. in diameter by 22 ft long). Like the fuel storage cells, the moderator cells internals were removed and properly disposed of early in the SRE decommissioning activity. The Hy-Ram was used to fracture the concrete around the cells, and the facility bridge crane was used to remove each cell from the rubble. The cells were reduced and packaged in Building 163. Figure 63 shows the concrete removal around the tops of the cells.

4.4.12.3 Fuel Cleaning Cells

The three fuel cleaning cells (or wash cells) were used to remove residual sodium from a fuel assembly that had been removed from the reactor core. The wash cells were 24 in. in diameter and 25 ft long. The top 2 ft 8 in. of the cells were encased in concrete with the remainder of the units surrounded by a sandy soil backfill. As with the previous cells, the wash cells internals were removed and disposed of as radioactive waste early in the decommissioning. A unique feature of the wash cells was the presence of drain lines running from the bottom of each cell to the valve pit adjacent to the hot cell service gallery. The drain lines precluded the use of the facility crane to pull the cells as had been done with the fuel storage tubes. Instead, access had to be provided to the bottom of the cells. A pit was dug and shoring installed along the walls of the excavation before employees could be allowed

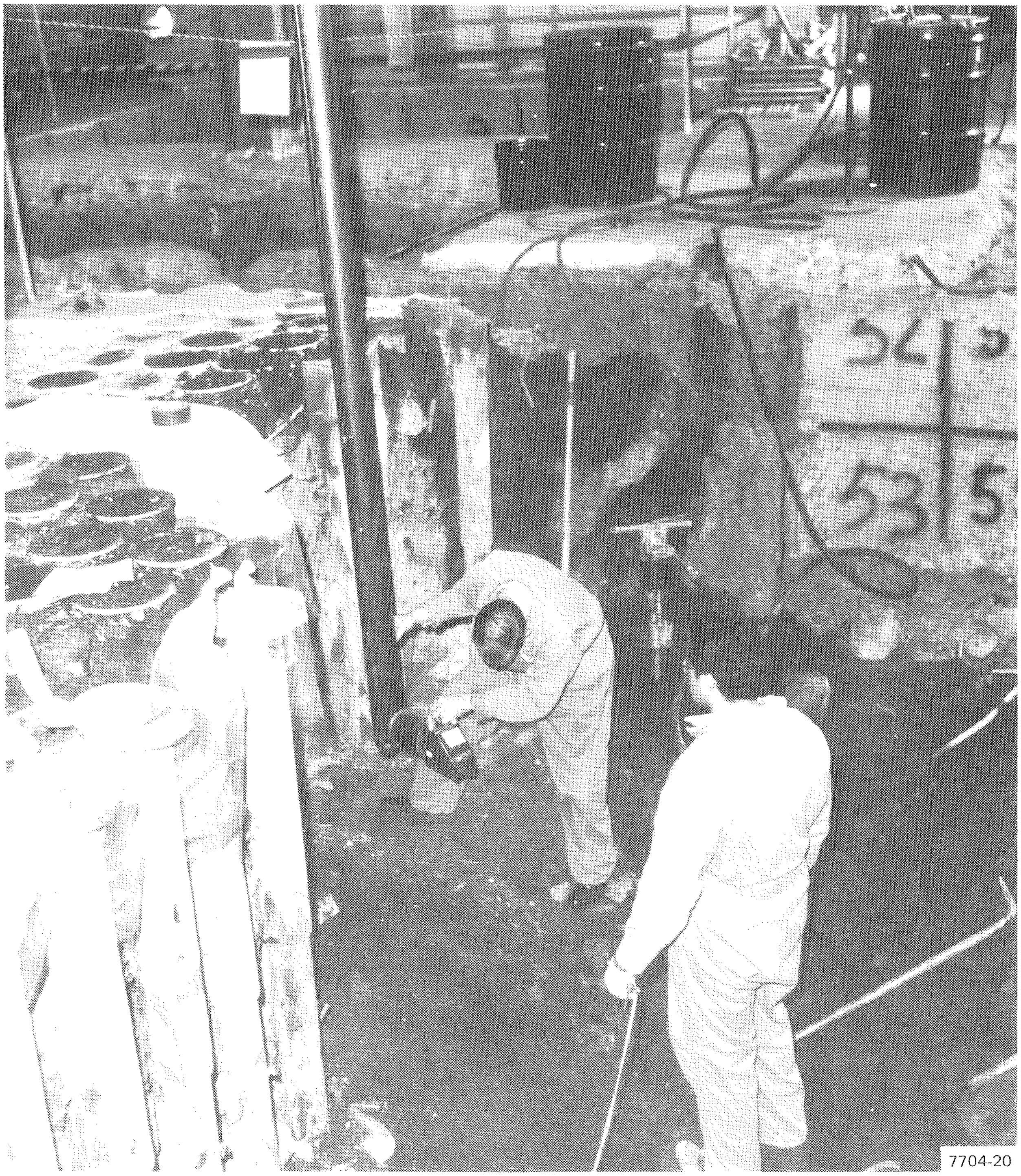

Figure 61. Fuel Storage Tube Removal

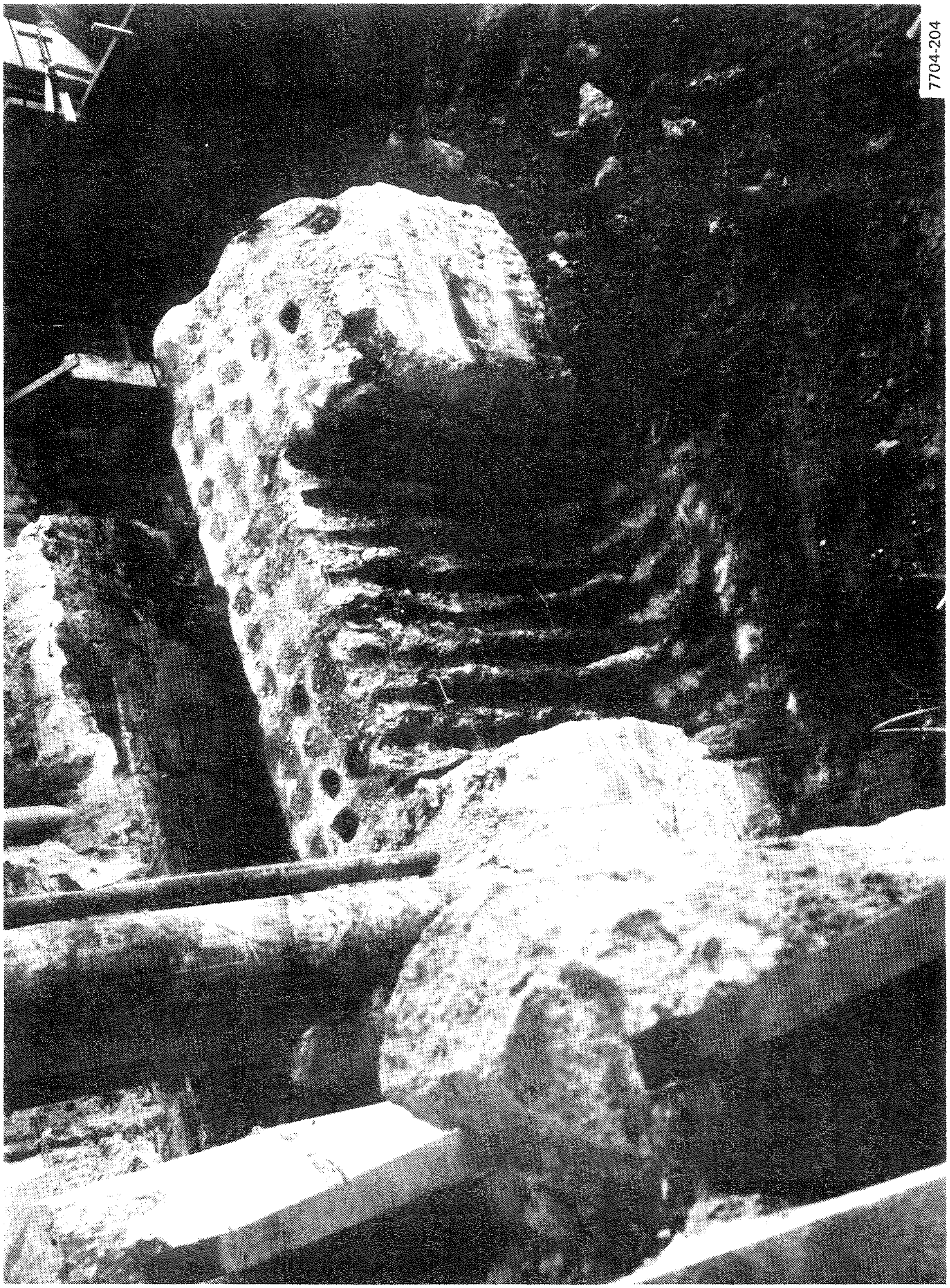

Figure 62. Fuel Storage Cell Concrete Demolition

Figure 63. Moderator Element Storage Cells

to remove the drain pipes. The upper right corner of Figure 62 shows a portion of the shoring installed for the wash cell drain line removal.

The wash cells themselves were removed in sections. As the upper concrete collar was removed, the upper 3 ft of the cells was cut free. This was accomplished by using a large-diameter pipe cutter. During the high-bay excavation, another larger section of each cell was removed in the same manner. Immediately following a cut, the top of the cell was sealed with a cover to prevent the introduction of foreign material into the wash cell. Figure 64 shows the removal of the wash cell drain lines. Caution was used to prevent the spilling of any contaminated liquids onto the surrounding soil. Once the drain pipes were removed, the remaining portions of the wash cells were removed and transferred to Building 163 for size reduction and packaging for disposal.

4.4.12.4 Hot Cell Fuel Element Enclosure Sleeves

The fuel element enclosure sleeves were the deepest elements in the SRE facility. The two sleeves were located within a 6-ft-deep pit at the bottom of the primary hot cell. Figure 65 is a sketch of the SRE hot cell area. The hot cell floor was 20 ft below the high-bay floor. This meant that the tops of the sleeves were 26 ft below grade. The sleeves themselves were 8 in. in diameter by 25 ft 6 in. long. Thus, the bottom of the sleeves were over 50 ft below the original high-bay floor.

When the sleeves were uncapped, they were found to contain 2 to 4 ft of water. Analysis of samples indicated levels above the maximum permissible concentration for mixed fission products. The small side diameter of the sleeves made decontamination of the interior surface virtually impossible. It was decided to remove the sleeves. The Hy-Ram was used to expose the top section of the two units. It quickly became apparent that the small backhoe/Hy-Ram combination would not be capable of excavating a hole deep enough to reach the bottom of the sleeves. The preliminary excavation also showed that the two units were most likely encased in concrete for their entire length.

Figure 64. Wash Cell Drain Line Removal

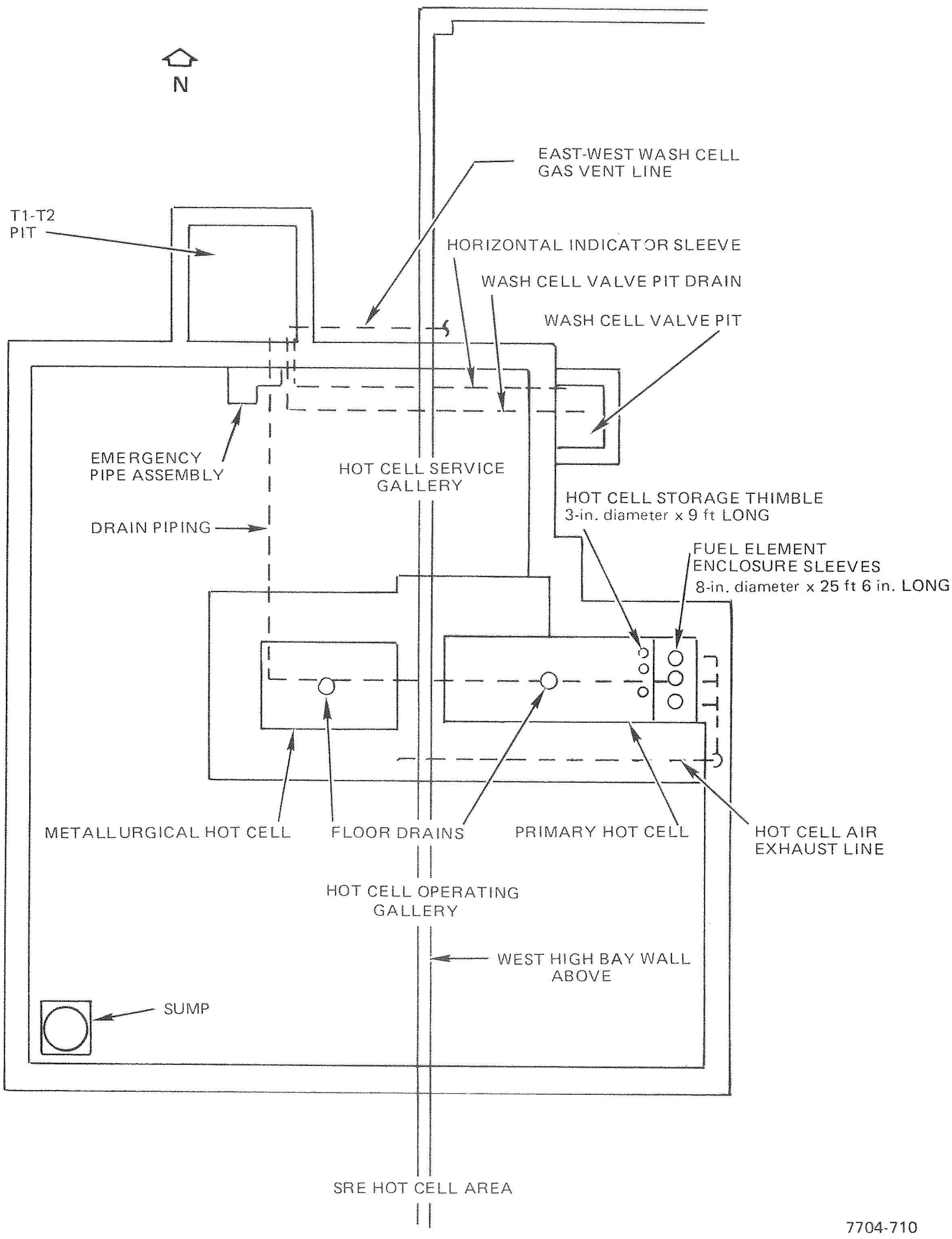

Figure 65. SRE Hot Cell Area

Meetings with various contractors and consultants were held to explore methods of excavating the bedrock adjacent to the sleeves. The following combination was selected to provide the needed access:

1) A low-strength concrete mix was placed in the excavation dug by the small backhoe. This provided a smooth, solid surface for a rock drill, adjacent to the sleeves.
2) A rock drill was used to drill a pattern of 93 holes in approximately a 10-ft by 10-ft area. The holes were a minimum of 30 ft deep.
3) A large backhoe capable of excavating 35 ft straight down was brought onsite. The large backhoe was also equipped with a Hy-Ram to fracture the bedrock which had been weakened by the rock drilling.
4) The two sleeves were pumped dry and then filled with an expanding polyurethane foam to fix contamination. If the Hy-Ram or backhoe had inadvertently punctured a sleeve, the foam would have minimized the spread of contamination.

Figure 66 shows the rock drill in operation. Figure 67 shows the large backhoe in operation inside the high-bay excavation. Figure 68 shows one of the two sleeves being removed from the excavation.

Samples for analysis were taken every 6 ft during the rock drilling as well as the excavation. No contamination was found in the 10- by 12- by 32-ft-deep excavation. Archive samples were taken and the hole was backfilled with 100 yd^3 of low-strength concrete. This proved to be cost effective compared with the labor charge of backfilling and compacting by hand, since access was limited for machine compactors. The two sleeves were transferred to Building 163 for size reduction and packaging for disposal. Figure 69 shows the size reduction booth set up in Building 163.

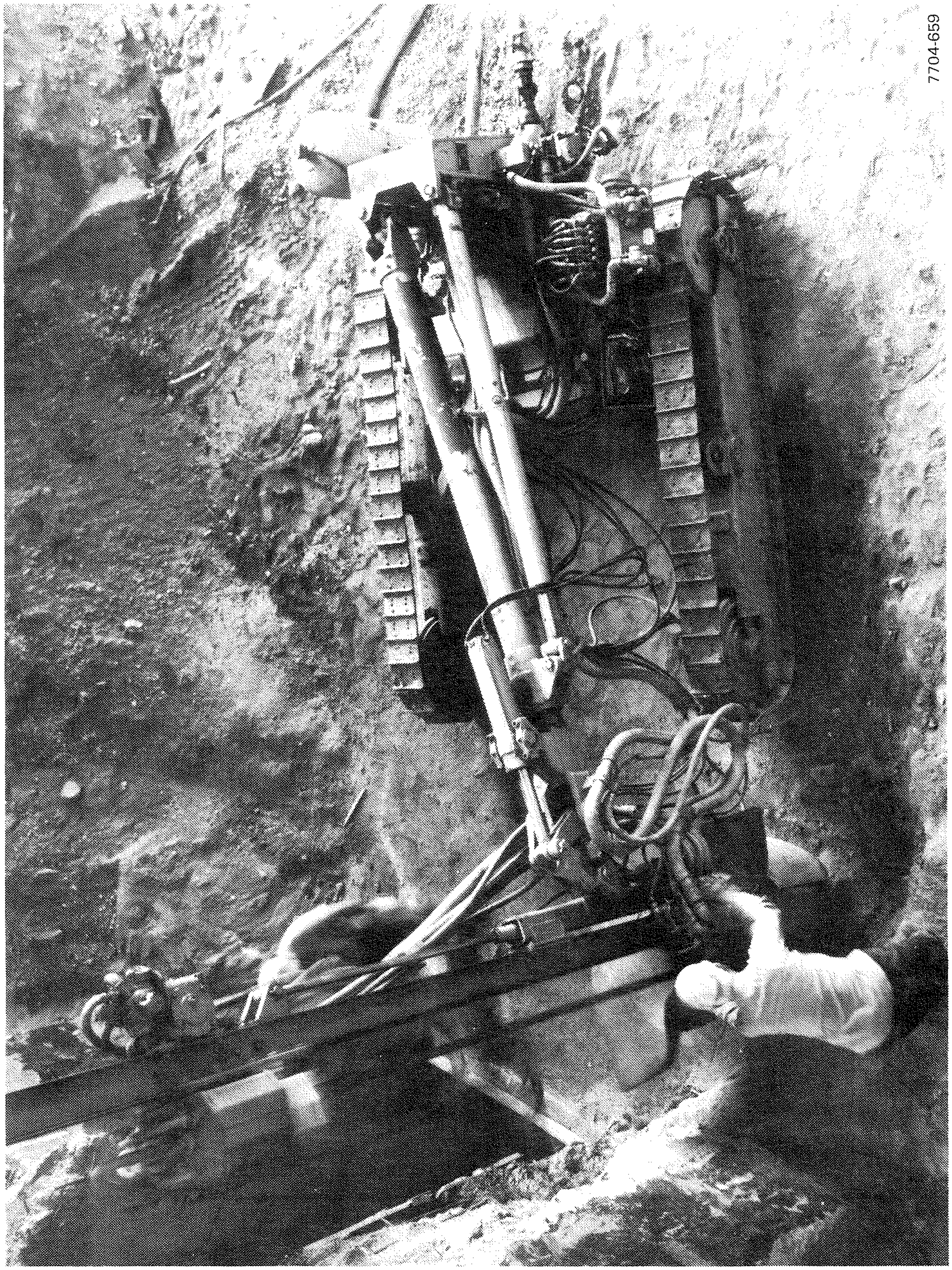

Figure 66. Rock Drill in Operation

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Figure 67. Large Backhoe in SRE High-Bay Excavation

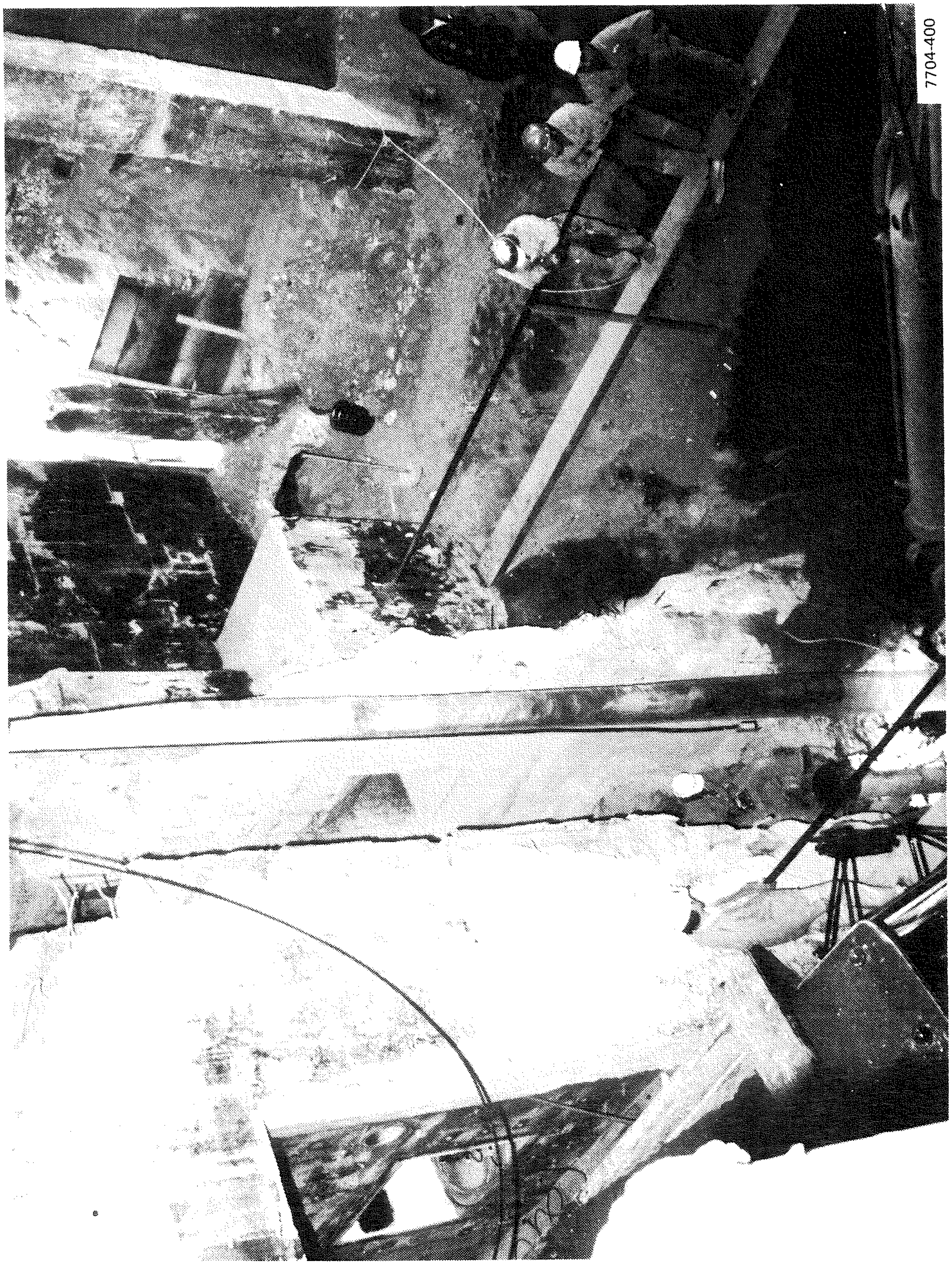


Figure 68. Fuel Element Enclosure Sleeve Removal

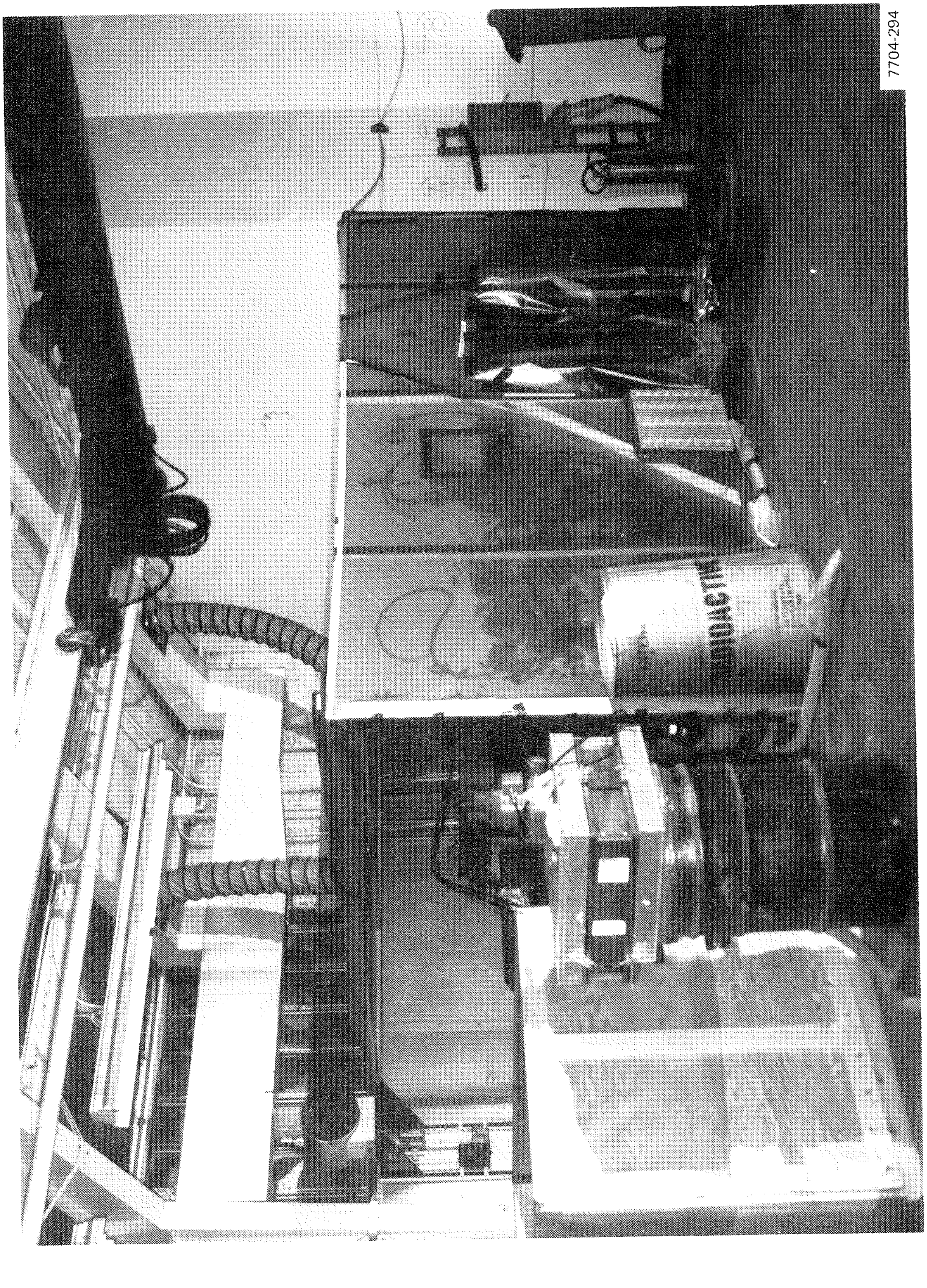


Figure 69. Building 163 Size-Reduction Booth

4.4.12.5 Sodium System Dip Legs

The main primary and auxiliary primary sodium loops each contained a dip leg. The sodium pumps were designed to accommodate a certain amount of leakage past the hydrostatic bearing and labyrinth seals into the upper part of the pump case. An overflow line permitted sodium collected in the pump case to be transferred back to the suction side of the pump through the dip leg. The dip leg provided a low point in the loop.

After the sodium systems were removed and main and auxiliary primary vaults were demolished, the dip legs remained as 24-in.-diameter by 25-ft-deep pipes at the bottom of the excavation. The two units were surveyed internally by lowering instrument probes to the bottom. The auxiliary unit was found to be clean. It was filled with concrete and a steel plate was welded to the top. The primary dip leg was internally contaminated and work was begun to remove it. First, a series of holes were drilled around the circumference of the pipe to a 24-ft depth. (The top 2 to 3 ft of the pipe had been removed after the vault floor was demolished.) During this operation, the dip leg was found to be embedded in concrete for its entire length. Several methods were tried to lift the dip leg out of the cored hole. The facility bridge crane, onsite mobile equipment, and a 25-ton hydraulic jack were all tried without success. The dip leg was removed only after the Hy-Ram was used to excavate a shaft along side the pipe. Figure 70 shows the Hy-Ram working on the excavation. The dip leg was found to have three equally spaced 1/2-in.-diameter by 4-in.-long horizontal anchors. The bottom plate was 2 in. larger in diameter than the pipe, which prevented the dip leg from being pulled after the core drilling was completed. (There was no indication of any type of anchor on the facility construction drawings.) The dip leg was transported to Building 163 and cut up and packaged for disposal.

4.4.13 Decontamination and Dismantling of the Hot Cell Area

The SRE facility had its own onsite hot cell. It was located below grade at the west end of the high bay (see Figure 3). There were two cells — a

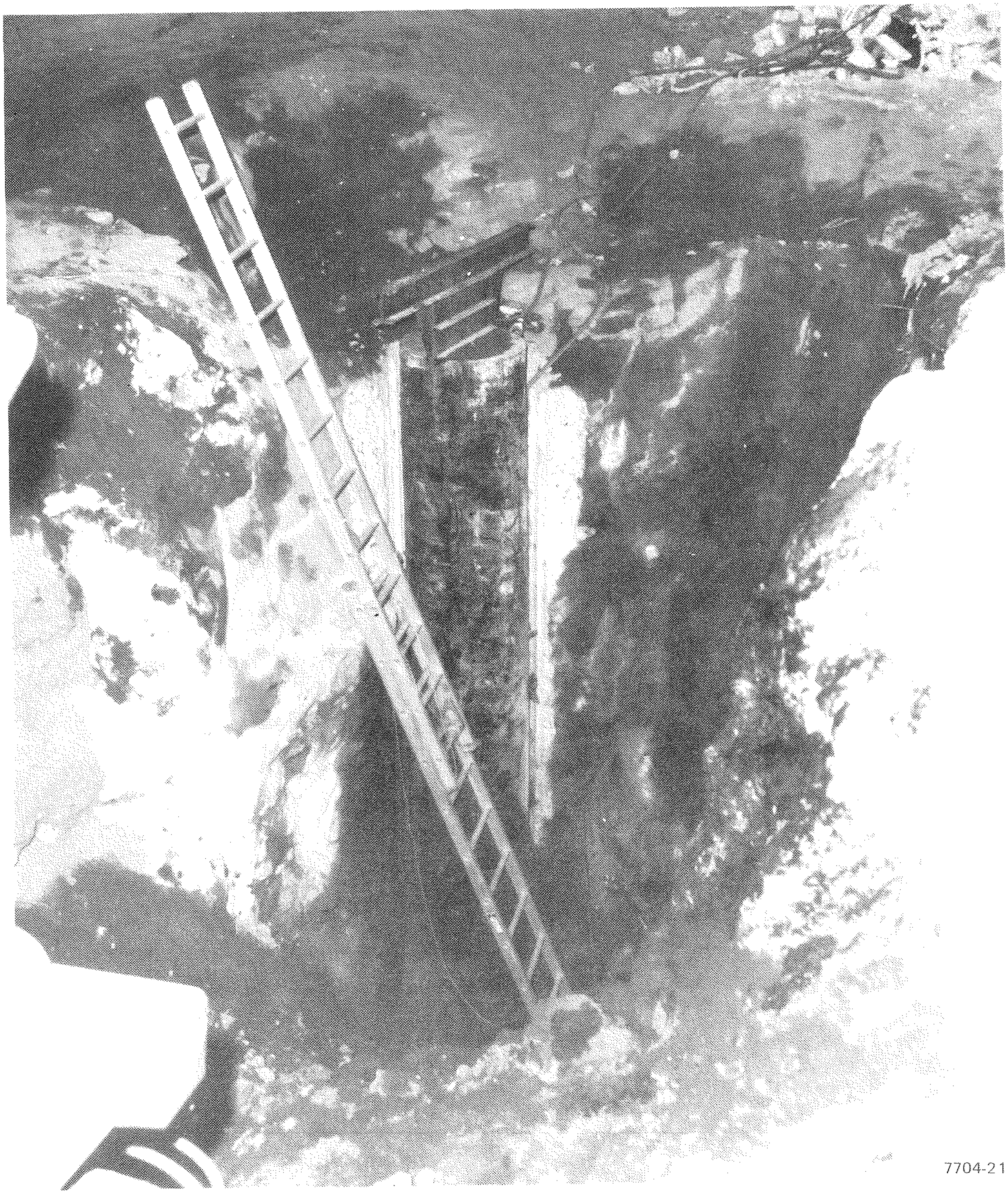


Figure 70. Dip Leg Excavation

primary cell with two windows and two remote manipulators and a metallurgical cell with one window and one manipulator. Access to the hot cell operating and service galleries was by a stairway from the west end of the high bay or through two access hatches from the outside.

The first step in decontaminating the hot cell area was to remove the cell doors, windows, and manipulators. The operating and service gallery walls were stripped of all unnecessary piping and electrical components. Next, a thorough radiological survey was performed. Surface contamination found on the concrete walls and floor was removed. The walls of the cells were 3-ft-thick, high-density magnetite concrete covered inside and out with a 1/8-in.-thick carbon steel plate, except for the floor pans in the cells which were 1/8-in.-thick stainless steel. Since the cells had never been exposed to a neutron field, there was no activation, only surface contamination. Since a considerable amount of contamination could be masked by the steel plates, they were removed. Figure 71 shows the removal of the steel plates from the interior of the primary cell. Carbon steel plates were removed with a conventional oxyacetylene cutting torch. The stainless steel floor pans were cut using the plasma-arc torch.

The in-cell floor drains and the hot cell air exhaust line were removed. The exhaust line was an 8-in.-diameter, thin wall tube used to discharge the primary cell atmosphere to the above-grade radioactive exhaust system. The tubing was embedded in the concrete wall between the primary hot cell and the hot cell operating gallery, inches away from the grade beam supporting the west wall of the high bay. A method was found to remove the tubing without endangering the structural integrity of the building. A combination of concrete sawcutting and jackhammering by hand was successful. The exhaust line was removed intact without damage to the facility. Figure 72 shows the primary cell wall and the exposed exhaust line.

Hot cell drain line removal posed a different challenge. The high point of the drain system was below the 6-ft-deep pit at the east end of the primary

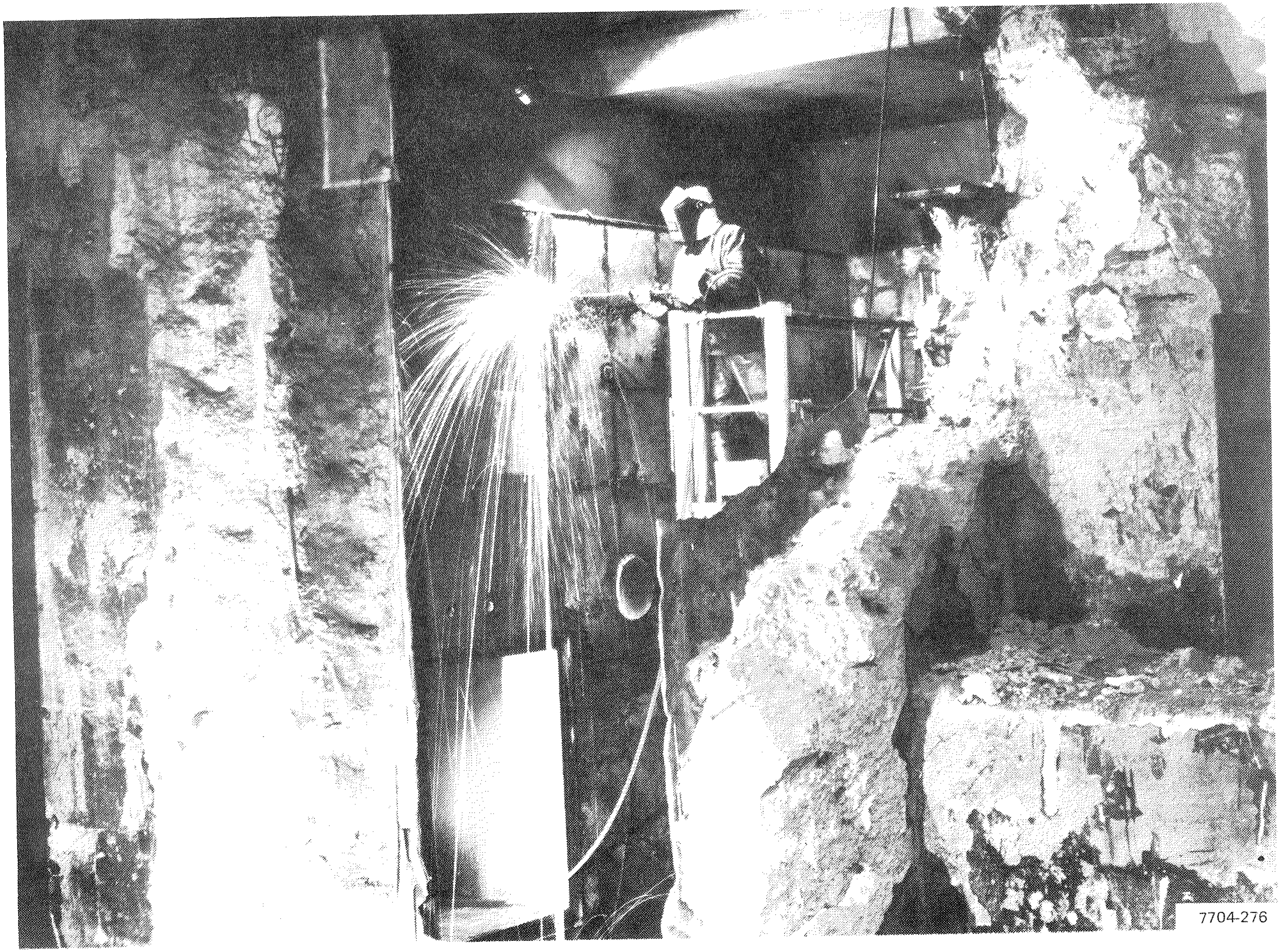


Figure 71. Cutting Steel Plates from Interior of SRE Primary Hot Cell

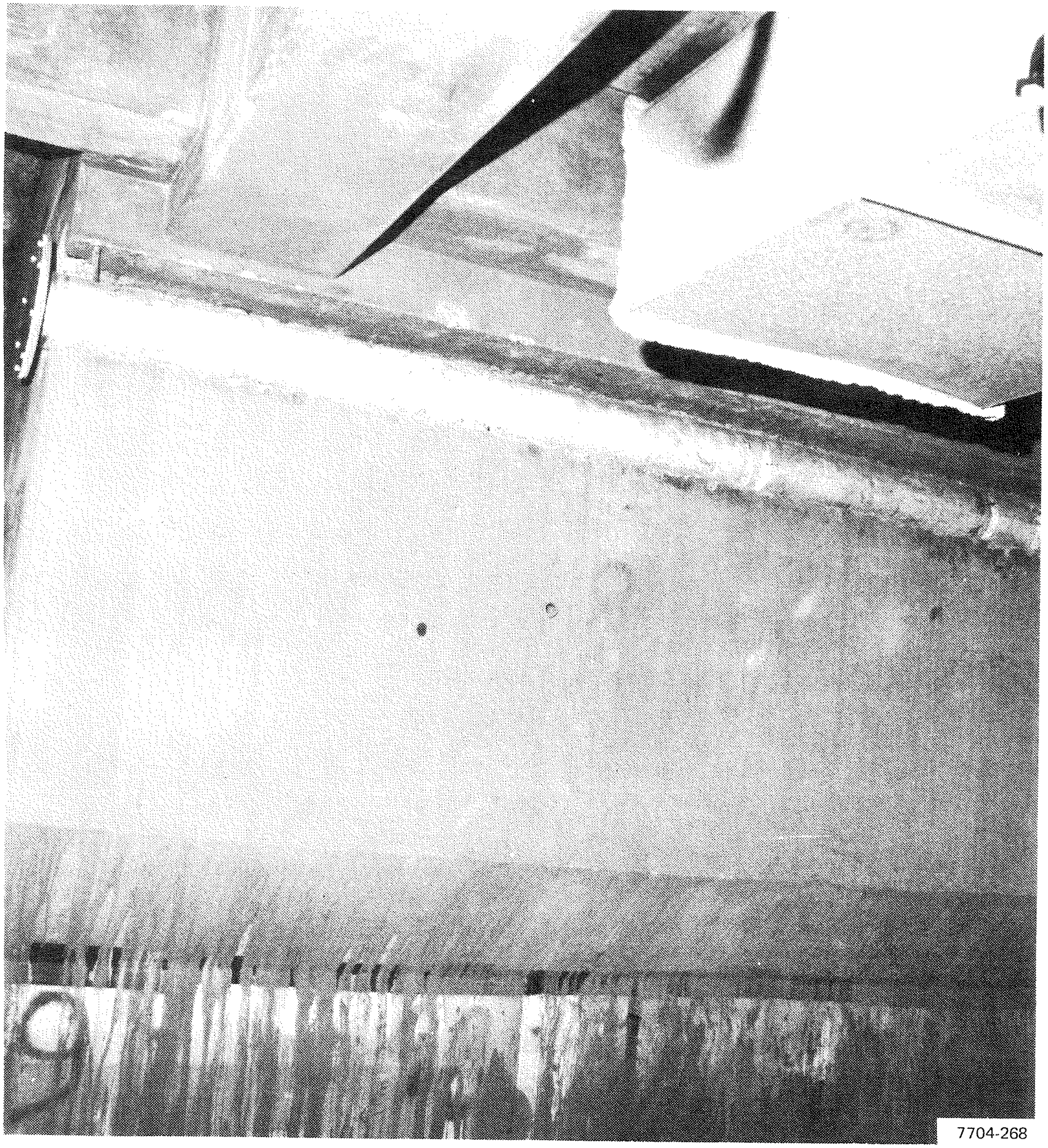


Figure 72. Exposed Primary Hot Cell Exhaust Line

cell. This put the drain line almost 7 ft below the floor of the metallurgical cell and close to 9 ft at the point where it entered the T1-T2 pit. Excavating the pipe by hand would have been costly; machine access was needed. Since the high-bay excavation had exposed the east wall of the hot cell, a doorway was made through this wall. A concrete wall saw was used to cut an opening in the 18-in.-thick wall into the operating gallery. The Hy-Ram was used to remove the concrete. Access into the primary cell was provided by demolishing the east wall of the cell. Access into the metallurgical cell was provided by demolishing the west wall of that cell. A concrete saw was used to cut the floor in the cells and the service gallery. The backhoe was used to excavate above the drains. The final few inches of soil were removed by hand, so as not to risk damaging the pipe. The contaminated drain line was removed and packaged for burial. No contamination was found in the soil. The excavation in the service gallery also provided access to the wash cell valve pit drain. Some additional concrete sawcutting and backhoe operation uncovered this drain line, which entered the T1-T2 pit at approximately the same elevation as the hot cell drain. In the same vicinity was the wash cell gas vent line which also entered the T1-T2 pit. A hole was found in the gas vent line and contaminated soil up to 21,000 pCi/g was found adjacent to the pipe. Some hand excavation was required to remove all of the contamination. No contamination was found near the wash cell valve pit drain. The wash cell valve pit itself was found to be contaminated. The Hy-Ram was used to demolish the concrete, and the rubble was packaged for burial.

Since the hot cell was below grade, it was subject to the effects of ground water at the site. Several temporary sumps were installed in the high-bay excavation, but the hot cell had its own permanent sump. Leach lines installed under the concrete floor channeled the ground water to the sump during the rainy season. A sump pump was used to pump the water up and away from the building. During the operation of the reactor, the drain from a radioactive sink was plumbed into the sump. The pump discharge was connected to a radioactive holdup tank adjacent to the discharge line. The sump, sump pump, piping, and a small portion of the leach lines were contaminated. All of the contaminated components were removed, and a new sump was installed. A portion of the leach lines was replaced. A new sump pump and some piping were required. The radioactive holdup tank was excavated and shipped to burial. No additional contaminated soil was found.

Numerous penetrations existed between the operating and service galleries and the insides of the two hot cells. There were also penetrations between the two cells. Most of these consisted of small-diameter (2-in. or less) pipe and electrical conduit. Some of these pipes were surveyed and found to be internally contaminated. Those that could not be decontaminated easily were removed by core drilling a 4-in.-diameter hole around the pipe and removing the solid core with the pipe intact. Located in the floor of the primary cell were three storage thimbles. They were 3 in. in diameter and 9 ft long. They were easily removed by the Hy-Ram as part of the cell floor demolition.

The T1-T2 pit adjacent to the hot cell originally contained radioactive liquid waste holdup tanks. They were removed early in the SRE decommissioning. The pit was constructed of concrete and required surface decontamination over a large area. Jackhammers and hand scabblers were used as machine access was limited.

All of the trenches and excavations in the floor of the hot cell were backfilled and compacted. The floor was patched at the completion of hot cell decontamination.

4.4.14 Excavation Backfilling

Backfill and compaction of the SRE high-bay excavation began immediately following the completion of the below-grade final survey. A procedure was written and approved, and the excavation contractor was retained to provide the labor and equipment necessary to complete the backfill. The procedure specified the compaction density to be a minimum of 90% of maximum density as determined by the Uniform Building Code (UBC) Standard No. 70-1 (ASTM D1557). Field density testing in accordance with UBC Standard No. 70-2 (ASTM 1556) was also required. A geotechnical engineering consultant provided the field tests and a final report verifying the compaction density of the high-bay backfill. This report is on file with the Facilities & Industrial Engineering Department of Rockwell International, Energy Systems Group.

The wooden shoring wall was removed from the bottom up, as the backfill soil raised the level of the high-bay excavation (see Figure 73). The column support bracing was removed when the soil level reached the tops of the temporary footings. Seismic bracing installed outside the high bay was also removed when the backfill soil reached the footings. Figure 74 shows the high-bay excavation during the early phase of backfill and compaction.

Approximately 7000 yd^3 of material was used to backfill the SRE excavation. About half was available onsite and consisted of soil from the excavation and clean concrete rubble. Large pieces of concrete were reduced in size with a mobile crane and a wrecking ball. The remaining soil was purchased from a nearby land development.

While the high-bay backfill was in process, Energy Systems Group's Facilities Design engineers were designing a replacement floor for the high bay. Construction of the new floor started shortly after the completion of backfilling. The excavation surface area outside the building was paved with asphalt. Since the high-bay walls had yet to be decontaminated, precautions were taken to prevent the new floor from being contaminated. Two coats of a concrete sealer were applied. Next a coat of strippable latex paint was applied to prevent liquid contaminates from getting into the porous concrete. In the event a spill occurred, a second coat of paint would have been applied over any residual contamination that could not be removed. Then both coats of paint would be stripped off and fresh paint reapplied. The theory was not tested since the floor was not cross contaminated during any remaining decontamination activities.

Figure 73. High-Bay Excavation Retaining Wall with Lower Wooden Shoring Removed

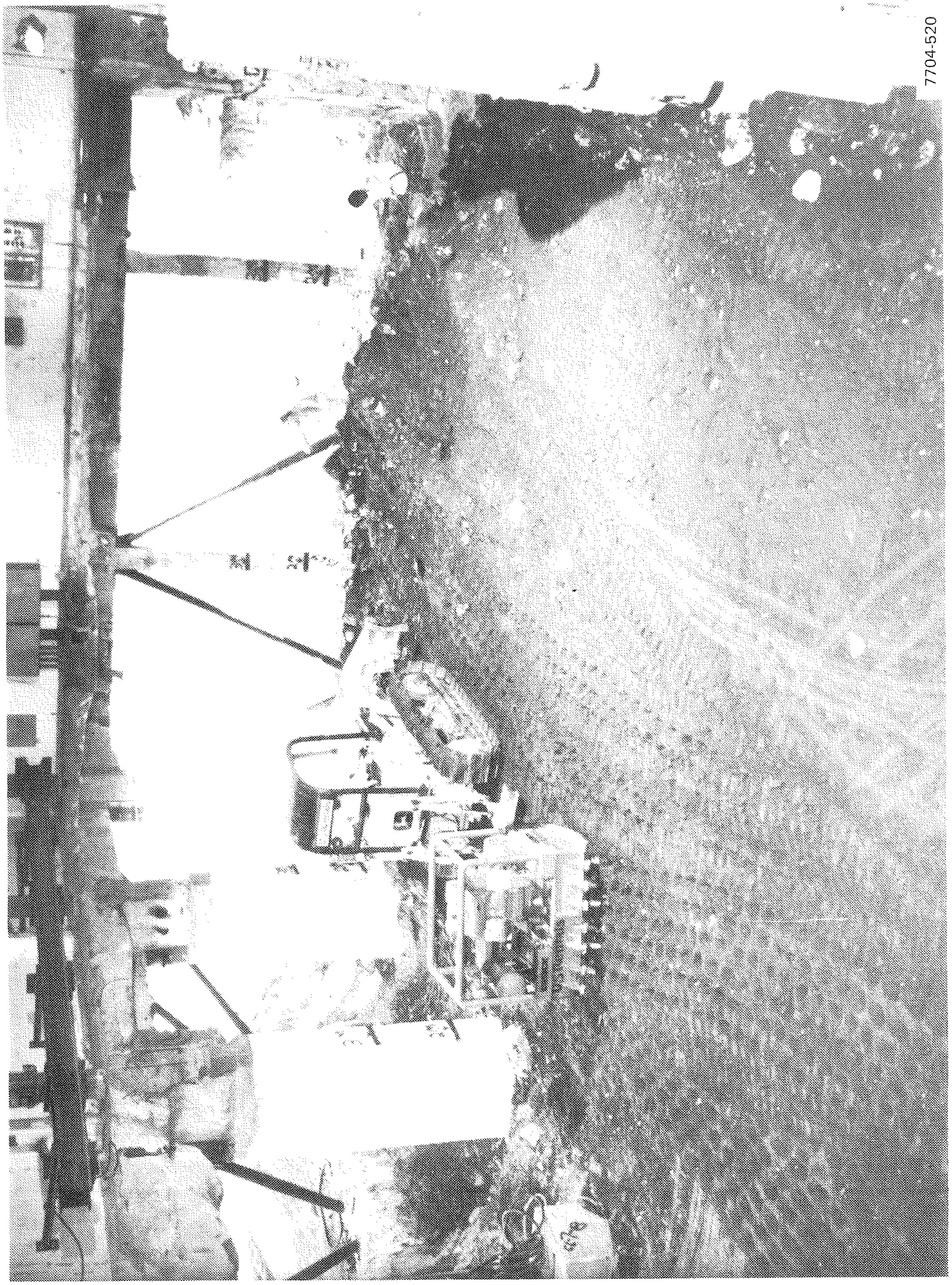


Figure 74. SRE High-Bay Excavation Backfill and Compaction in Progress

4.4.15 Decontamination of High-Bay Walls, Ceiling, and Cranes

With the replacement of the high-bay floor, below-grade decontamination activities at the SRE were completed. Only the high-bay interior and the radioactive exhaust system remained to be decontaminated. Since the radioactive exhaust would be needed for high-bay decontamination, it was left for last. A characterization survey was performed on the high-bay interior surfaces to determine the extent of the contamination. Locations of high instrument readings were identified on the walls. Smear samples indicated no removable contamination. Samples of the paint and the concrete behind the paint were obtained and analyzed. In all cases, contamination was found in the paint but not in the concrete. An evaluation of different paint removal methods was initiated. Sandblasting was selected over mechanical or chemical means. A sandblasting contractor who could work to our procedures was selected and a contract was awarded. Concurrently, a contract to install scaffolding in the high bay was awarded. The scaffolding provided access to 100% of the high-bay walls, the perimeter of the ceiling, and a portion of the three bridge cranes. Access to the remainder of the ceiling and cranes was accomplished by using a manlift. Figure 75 shows the scaffolding along the east wall of the high bay.

Prior to the start of sandblasting, the high bay was sealed and the radioactive exhaust units on the roof were activated. A negative pressure was maintained in the high bay to prevent the release of airborne contamination. The rooftop exhaust system had been designed for this purpose since the high bay was at a negative pressure during reactor operation. Sandblasting was very effective in removing the many layers of paint on the concrete and steel surfaces. The large amount of dust generated necessitated a much higher than normal frequency of filter changes in the exhaust system, and the large volume of sand deposited on the floor each day necessitated regular floor cleanup after the contractor had left for the day.

For the decontamination by sandblasting method to be effective, the paint has to be removed as completely as possible, especially from the horizontal

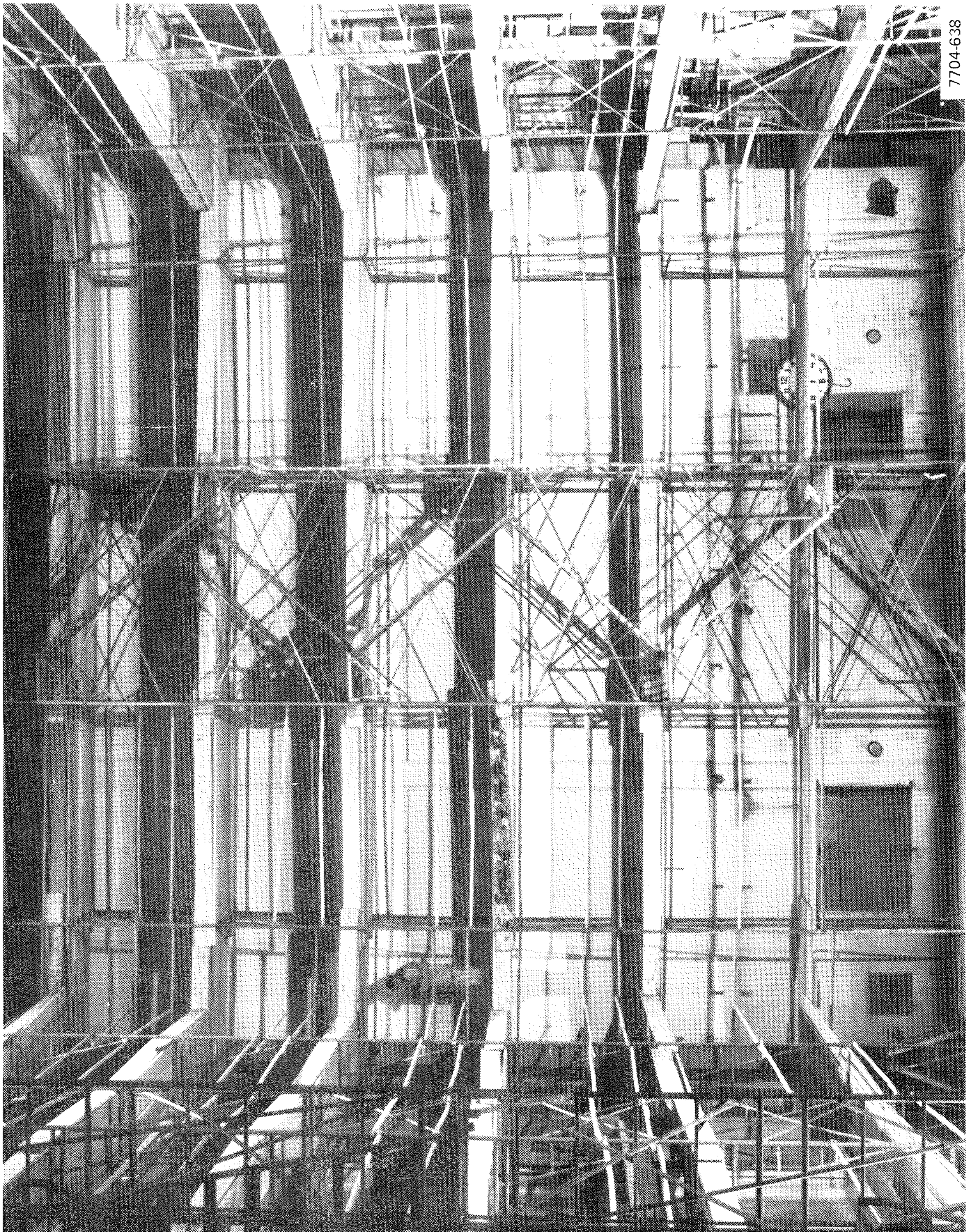


Figure 75. High-Bay Scaffolding

surfaces near the ceiling. Normally, a sandblaster prepares a surface for painting. It is usually acceptable if a certain amount of the original paint is not removed, particularly on surfaces that are difficult to reach (e.g., tops of beams or trusses, or backside of support columns). On a decontamination job, these are the surfaces that must be cleaned as well as possible. Consequently, some of the structural steel surfaces had to be resandblasted.

In addition to the walls, ceiling, and structural steel, three bridge cranes were located in the high bay. A self-propelled, 60-ft manlift was required to reach the cranes for sandblasting and surveys. The two 5-ton cranes were relatively easy to decontaminate. The large, 75-ton crane had many surfaces that were not cleaned by sandblasting. Some surfaces required several applications of a chemical paint stripper followed by hand scrubbing with a wire brush to remove contaminated paint. More contamination was found in and behind electrical control panels mounted on the crane itself. Decontamination of the panels proved ineffective. They were removed and disposed as waste. The braided steel cable on the hoist was contaminated in several places and had to be removed and shipped to burial.

A final survey verified the removal of all contamination from within the high-bay interior.

4.4.16 Disposal of Radioactive Exhaust Systems

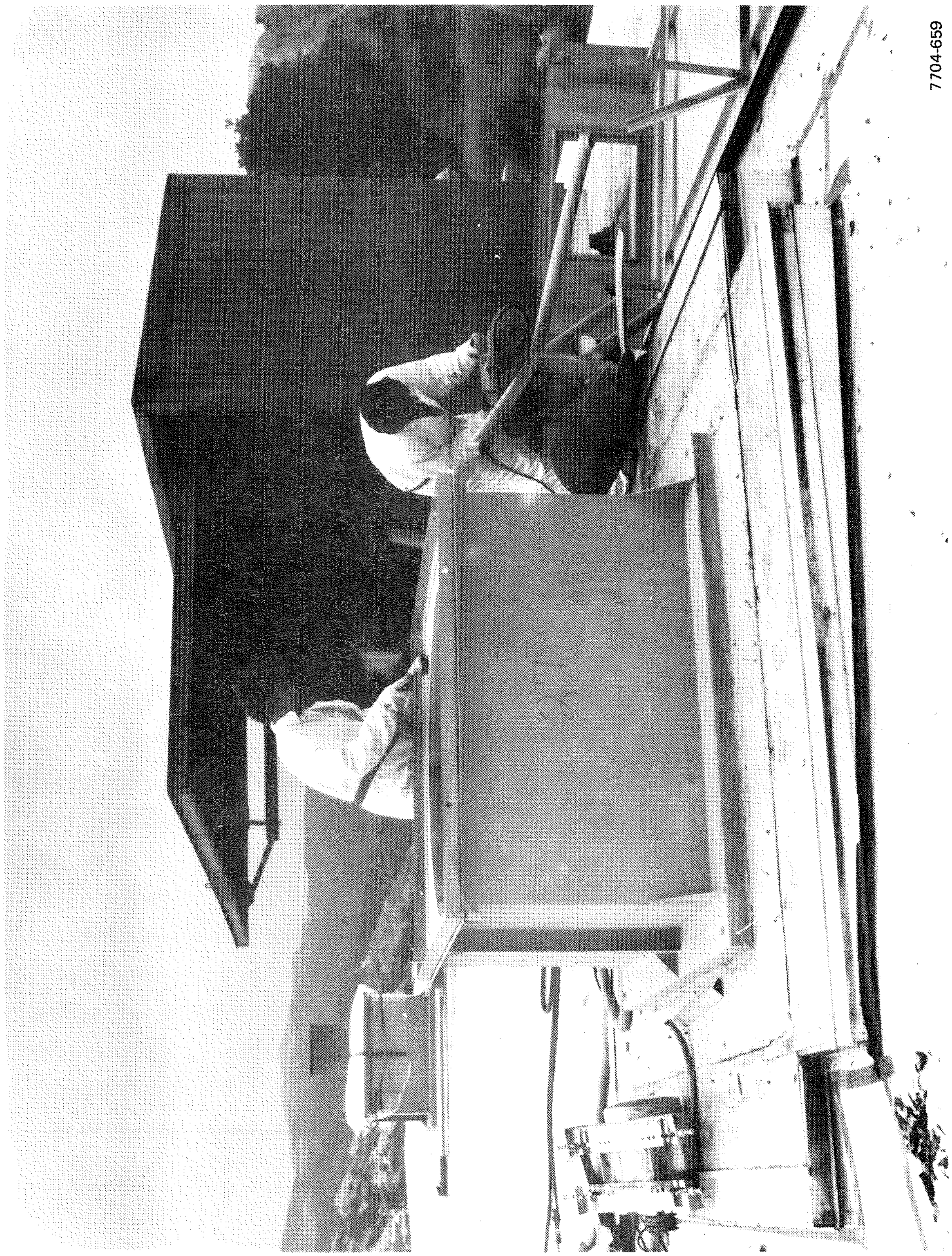
The SRE was equipped with three independent radioactive exhaust systems. Two identical units were mounted on the roof and exhausted air from the high-bay area. A separate single exhaust system for the hot cells was located directly above the metallurgical cell west of the high bay. During the decontamination activity, the roof units were used primarily during the sandblasting operation. The hot cell exhaust was disconnected from the two cells early in the decommissioning, and a new 18-in.-diameter exhaust line was routed into the high bay. The new duct provided a header to which several smaller diameter flexible or rigid exhaust lines could be connected. This radioactive exhaust system was used frequently during the SRE decommissioning. It was a vent for the sodium passivation, provided a negative pressure during reactor dismantlement and biological shield demolition, and was used during high-bay decontamination.

A radiological survey of the two roof units indicated they could be dismantled in place if the removable contamination could be "fixed" to the interior surfaces. This was accomplished with spray paint. Figure 76 shows the dismantling effort on the east roof exhaust. Both units were cut up into manageable pieces, wrapped in plastic, and lowered to the ground using an existing roof hoist. Figure 77 shows the removal of a portion of the roofing material adjacent to the roof penetration. Contaminated material was packaged for burial. The hot cell exhaust was dismantled starting with the ducting farthest from the blowers and moving toward the plenum. The contaminated components were packaged and shipped to burial. Noncontaminated items were sent to salvage. The exhaust stack was internally surveyed, found to be clean, and left in place for possible future use.

4.4.17 Decontamination of Buildings 163 and 041

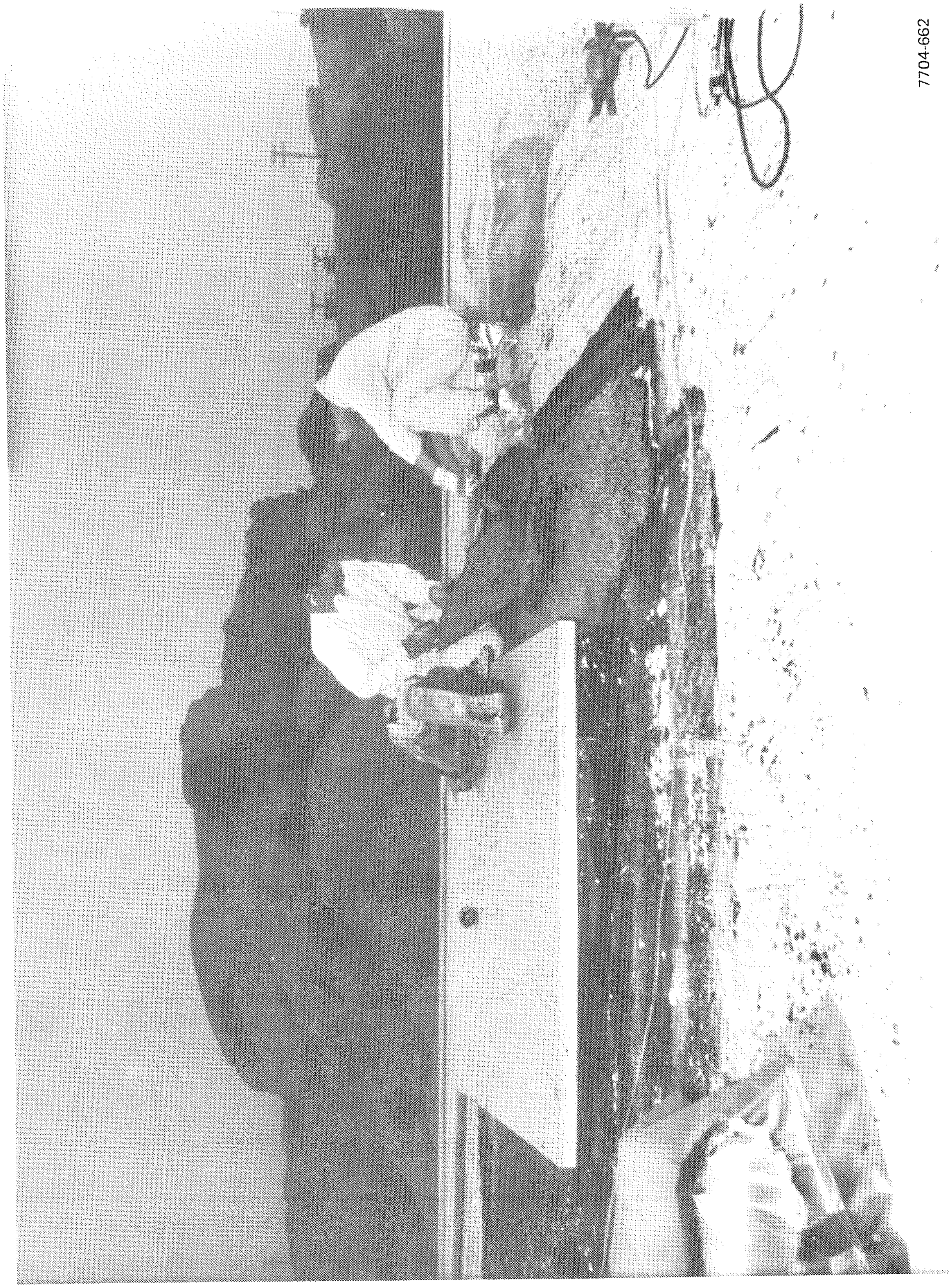
Building 163 is a prefabricated, sheet metal building located just east of the SRE. The west end of Building 163 has served as a pipe spool fabrication shop, a machine shop, a radioactive waste staging area, and a radioactive waste size reduction and packaging facility. Decontamination of the west end of Building 163 was included in the SRE D&D activity.

A radiological survey indicated that most of the contamination in the building was located in the wall insulation material and in the concrete floor. All of the insulating material was removed from the ceiling and walls and disposed of as radioactive waste. A 2-ton bridge crane and its supporting structure were removed. The hoist was packaged for burial. The structural steel was decontaminated and sent to salvage. The concrete floor was scabbled to remove fixed contamination, and a small radioactive exhaust system was dismantled and shipped to burial.



7704-659

Figure 76. Dismantling East Roof Exhaust



7704-662

Figure 77. SRE High-Bay Roof Decontamination

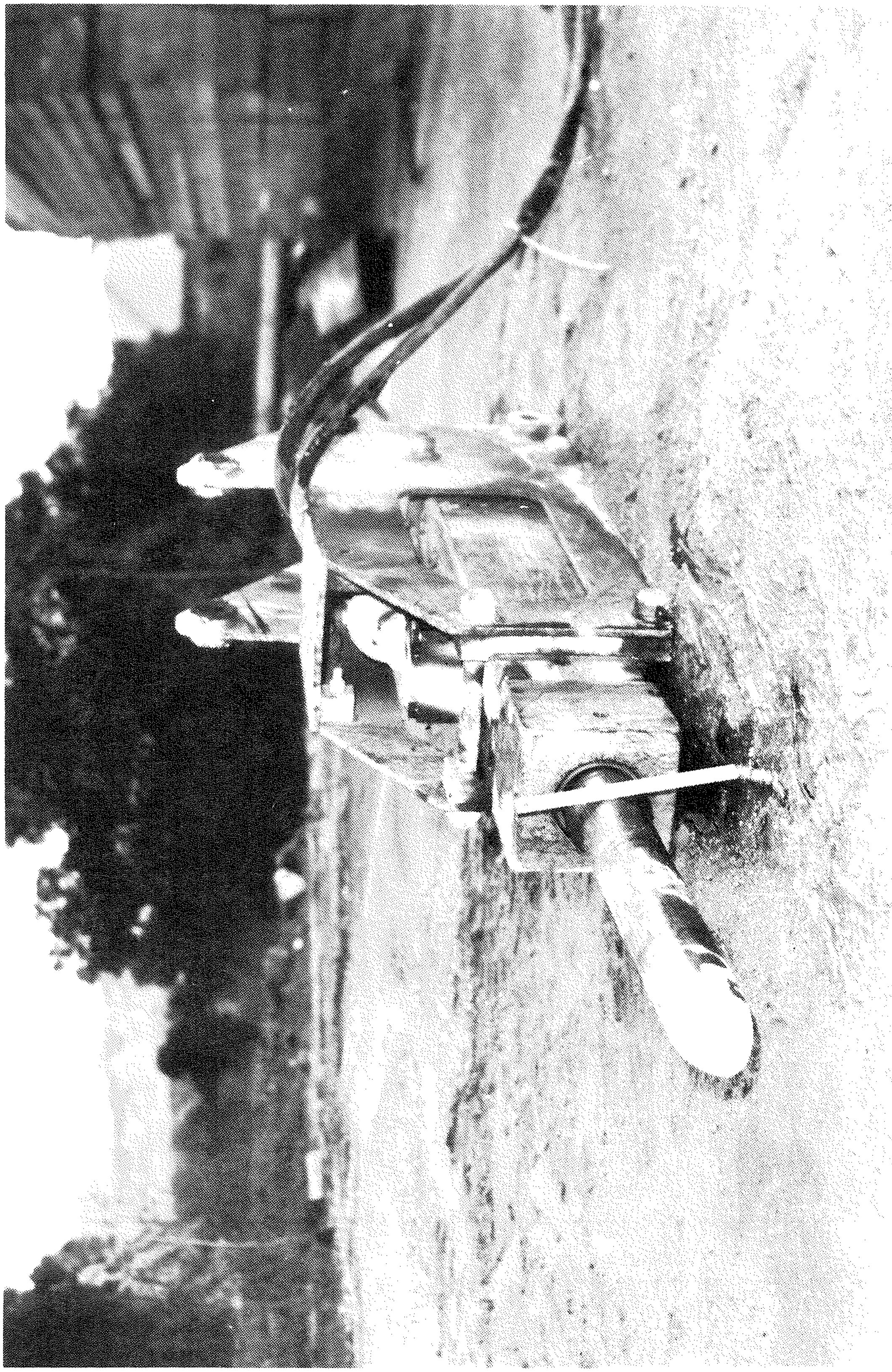
Building 041 was located within the SRE complex, west of Building 143. It had primarily been used for storage of radioactive waste prior to shipment. A small amount of fixed contamination was found on the concrete floor. Scabbling was used to remove the contamination.

4.4.18 Decontamination Equipment

Several pieces of equipment were found to be extremely useful throughout the SRE decomissioning project. A time-and-materials contract was established to provide concrete demolition services. The contractor had a large assortment of equipment available for use at the site, as well as a ready supply of experienced operators and laborers. The workhorse of the project was the backhoe/Hy-Ram combination. Figure 78 shows the Hy-Ram, a hydraulic jackhammer that could be mounted onto a backhoe in place of a bucket. The Hy-Ram could demolish concrete or bedrock many times faster than hand-held jackhammers. It could operate in any position, including 20 ft below the surface where the backhoe was parked. Despite its size, the Hy-Ram was easily maneuvered and could be used for selective material removal. The backhoe was used regularly at the SRE site. Various size buckets enabled the backhoe to excavate all but the deepest components at the site. The large backhoe shown in Figure 67 was used for the fuel element enclosure sleeves removal. It had the capability to excavate a hole 35 ft deep.

Concrete surface decontamination was a large part of the SRE D&D project. If a concrete surface could be decontaminated, the remaining material could be disposed of as clean rubble with a significant waste disposal cost savings. Many different methods and various pieces of equipment were tried with varying degrees of success.

The best and most versatile tool was found to be a scabbler. The scabbler is an air-driven, multiheaded concrete scarifying tool. Figure 79 shows two units: a three-headed hand-held model and a five-headed floor model. In actual use, the smaller hand-held unit was preferred. A guard was installed over the sides, and the hose from a portable radioactive vacuum system was



7704-621590

Figure 78. Hy-Ram Used at SRE

7704-621601

Figure 79. Floor Scabblers Used at SRE

attached. With the vacuum on, very little airborne dust escaped from the work surface. A single-headed "potato masher" scabbler was used in places the larger unit could not reach. Pneumatic tools, from 15-lb chipping hammers to 90-lb jackhammers, were used extensively for concrete and bedrock removal.

Portable vacuum cleaners for the removal or control of radioactive particulates were equipped with absolute filters and were used regularly at the SRE. Figure 69 shows a vacuum in use in Building 163.

4.5 WASTE DISPOSAL

4.5.1 Radioactive Waste Packaging and Handling

Most of the radioactive waste from the SRE was packaged onsite. Material or rubble identified as contaminated was packaged for burial as soon as possible. This was done for several reasons: first, it reduced the chance of cross contaminating a clean or recently cleaned surface; second, it provided access for further decontamination activity; and third, it reduced the radioactive field in a work area.

The selection of a container to package waste was based on several factors including type of material, volume to be packaged, and radiation level. Low specific activity contaminated soil, bedrock, and concrete rubble were packaged in cardboard, tri-wall King-Pac* containers. These containers were mounted on a plywood skid, held approximately 1 yd^3 each, and when loaded, had an average weight of 2000 lb. The King-Pacs had several advantages: they were inexpensive, the skids could be fabricated in advance and stored at the site, a box could be assembled in minutes, and even a loaded container was easy to handle. The primary disadvantage was the fragility of the cardboard. Great care was required when packaging or handling King-Pac containers. For example, no metal could be disposed of in a King-Pac. Large concrete pieces had to be packaged with a sufficient quantity of soil to prevent the rubble

*Registered trademark.

from shifting during transport and rupturing the sidewall of the container. Even scratches to the outer layer of cardboard were sufficient to cause a container to be rejected for shipment. A rejected box would have to be repackaged.

Low specific-activity contaminated steel or wood was packaged in strong, tight, wooden boxes. Each box held approximately 100 ft^3, and gross weight was limited to 6000 lb. Care was still exercised when packaging and handling a wooden box. If not packaged properly, the sidewall of a box could be split. Both King-Pac and wooden containers required indoor storage. Onsite, Building 041 was used for waste storage with overflow containers going to the Radioactive Material Disposal Facility.

Standardized shipping containers, like the King-Pacs and strong, tight, wooden boxes, helped organize the waste for shipment. Occasionally, oversize or custom packaging was required. The reactor vessel shield plug and the ring shield were both shipped to burial intact. The two FHMs and the one moderator-handling machine were also shipped to burial with only minor disassembly. With some large components, it was cost effective to perform a size-reduction operation prior to shipment. Figure 69 shows a size-reduction booth set up in Building 163. Items such as the fuel element enclosure sleeves and the dip leg were segmented in the booth, with a cutting torch or portable band saw, and placed directly into a shipping container.

Components having high levels of radioactivity required special packaging and handling. Lead or concrete shielding was used to reduce the surface dose rate on a container when shipping high-level waste. Dedicated shipping casks were used for the disposal of items that could not be packaged conventionally, even with shielding. The reactor vessel segments were an example of waste that required special containers.

Radioactive liquids were generated as part of the SRE decommissioning. Early in the project, it was permissible to ship low-level liquids in bulk quantities directly to a burial site. As transportation and burial guidelines became more restrictive, this was no longer permitted. Initially, contaminated liquids were acceptable if they were mixed with an absorbent medium, such as diatomaceous earth. This is how the 2500 gal of contaminated alcohol used in the sodium passivation process were disposed of. Eventually, the restrictions also prohibited this process. To comply with the new requirements, it became necessary to construct an evaporator. The unit was installed at the RMDF. Contaminated liquids were transported to the RMDF and processed in the evaporator. After processing, the remaining sludge was mixed with the proper quantity of cement powder and permitted to solidify in drums. The drums were acceptable at the burial site.

4.5.2 Radioactive Waste Transport and Burial

Transportation of radioactive waste from the SRE site was contracted to carriers licenced to receive and transport radioactive materials. Normally, a shipment was transported in a covered trailer 40 ft or 44 ft long. The quantity of each shipment was restricted by a 40,000-lb weight limit. Special transport vehicles were required to ship oversize or overweight loads. Figure 28 shows the reactor vessel shield plug being shipped to burial. Figure 80 shows the ring shield on a special transport vehicle. Radioactive waste from the SRE was shipped to two burial sites.

4.6 POST-DECOMMISSIONING RADIOLOGICAL SURVEY

The SRE facility was divided into several regions on the basis of past use, contamination history, and decommissioning operations. These regions, marked on Figure 81, were treated as geographical units in releasing the facility.

- Region I — This area contained the Hot Oil-Sodium Cleaning Facility, Building 724, and related structures, roadways, and drainage paths. Building 724 was relocated to Region IV and is now identified as Building 133. This building has been

7704-62810

Figure 80. Ring Shield on Transport Vehicle

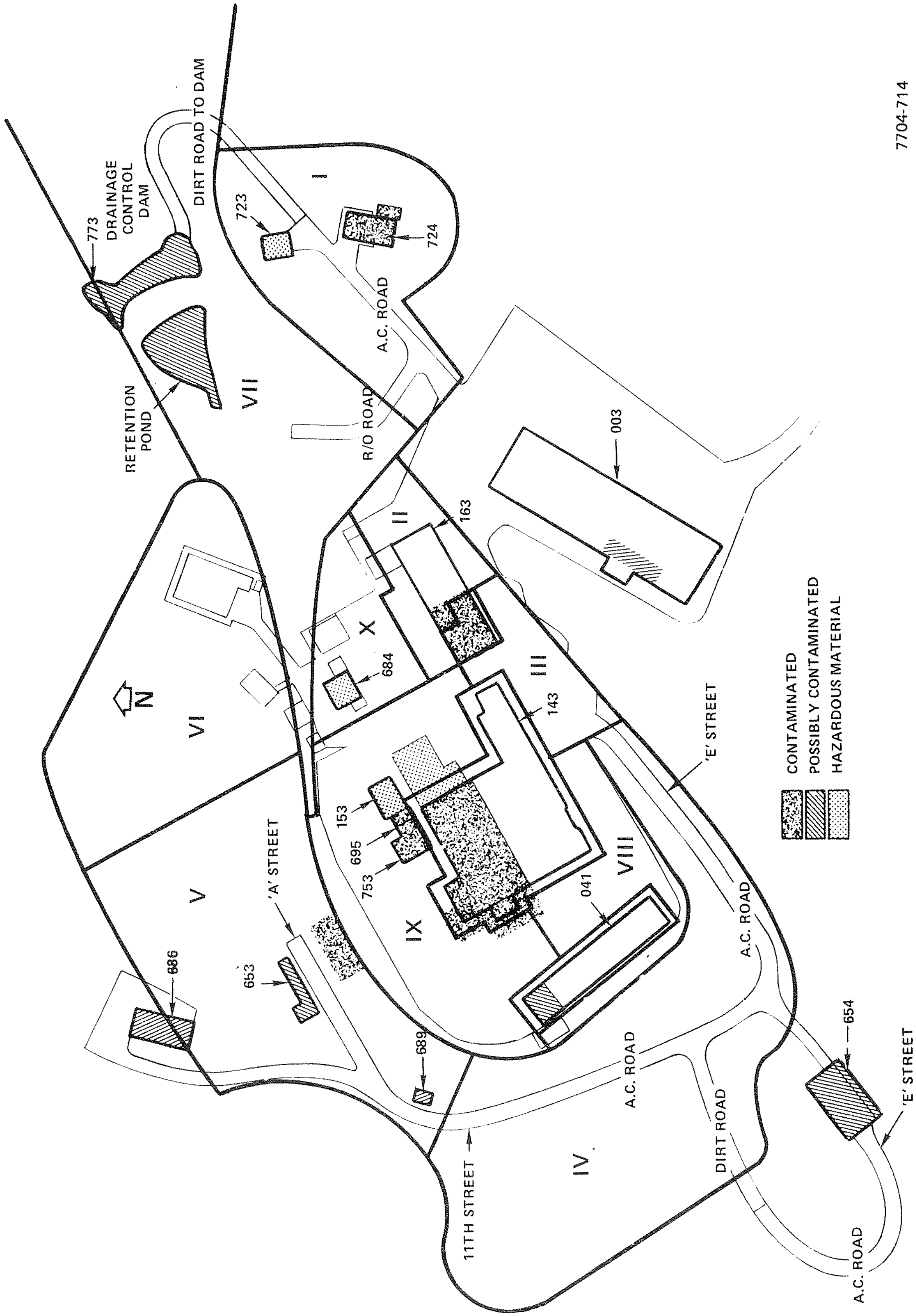


Figure 81. Radiological Survey Regions

released for unrestricted use. Survey data for this building is included in the report for Region IV.

- Region II — This area contains the Box Shop in the east end of Building 163 and also includes the surrounding paved surfaces.
- Region III — This area adjoins the contaminated work area of Building 163 and comprises the entrance approaches to the SRE and Region IV.
- Region IV — This area consists of the roadway from the SRE west parking lot and the slope to the west of the SRE and includes the building moved from Region I.
- Region V — This area contains the gas storage vault (Building 653) and the temporary radioactive waste storage area.
- Region VI — This area contains the water supply storage tank and some Southern California Edison Company structures.
- Region VII — This area contains the retention pond, the old leach field, the sanitary sewer pumping system, and the SRE drainage channel back to the fence line. It includes the retention pond overflow channel downstream for a distance of about 200 ft.
- Region VIII — This area consists of paving to the south and west of Building 143 to approximately the enclosure for the T1/T2 and T3 pits. It includes the drainage channel along the southwest to south edge of the paved area.
- Region IX — This area consists of the remainder of the paved area around Building 143 and includes the drainage path along the north side to the fence line at the northeast corner.
- Region X — This area was in use as a parking lot and includes the natural ground to the east of the parking lot.
- Building 163 — This area was the contaminated work area of the building and included the change room and the concrete ramp at the west entrance.
- Building 143 — This was the major reactor building and consisted of nonradioactive areas and areas with surface and/or

distributed contamination. The ground level and certain below-grade rooms are shown in Figure 3.

- Building 041 – The north portion of this building was used for interim storage of radioactive waste prior to shipment for disposal. The south portion was used for storage of controlled items.

As decontamination work in a region or building was completed, final survey activity began. A technical information (TI) report was prepared for each region and building as they were certified for final release to unrestricted use. The reports identify the area covered by the survey, the types of monitoring instruments used in the survey, and the results of the survey. In all cases, the level of radioactivity in the region or building was found to be below the acceptance criteria used in Table 3.

These reports have document numbers as follows:

N704TI990027, Region I
"Radiological Survey Results – Release to Unrestricted Use, SRE Region I (Building 724 Area)"

N704TI990028, Region II
"Radiological Survey Results – Release to Unrestricted Use, SRE Region II (Building 163, Box Shop)"

N704TI990029, Region III
"Radiological Survey Results – Release to Unrestricted Use, SRE Region III (SRE Entrance)"

N704TI990030, Region IV
"Radiological Survey Results – Release to Unrestricted Use, SRE Region IV (West Parking Lot)"

N704TI990031, Region V

"Radiological Survey Results – Release to Unrestricted Use, SRE Region V (Gas Storage Vault)"

N704TI990032, Region VI

"Radiological Survey Results – Release to Unrestricted Use, SRE Region VI (Water Tank Area)"

N704TI990033, Region VII

"Radiological Survey Results – Release to Unrestricted Use, SRE Region VII (Retention Pond)"

N704TI990034, Region VIII

"Radiological Survey Results – Release to Unrestricted Use, SRE Region VIII (SRE Front Lot)"

N704TI990035, Region IX

"Radiological Survey Results – Release to Unrestricted Use, SRE Region IX (SRE Back Lot)"

N704TI990036, Region X

"Radiological Survey Results – Release to Unrestricted Use, SRE Region X (SRE Parking Lot)"

N704TI990037, Building 041

"Radiological Survey Results – Release to Unrestricted Use, SRE Building 041"

N704TI990038, Building 043

"Radiological Survey Results – Release to Unrestricted Use, SRE Building 143"

N704TI990039, Building 163

"Radiological Survey Results – Release to Unrestricted Use, SRE Building 163"

5.0 SCHEDULE AND COST

5.1 SCHEDULE

Operation of the SRE as a nuclear power plant ended 15 February 1964. Modifications for a planned power expansion were completed 15 May 1965. At that time testing of the non-nuclear systems began. These systems were shut down in September 1967 after a decision was made to decommission the SRE.

A deactivation plan was implemented to put the facility in a "stored-in-place" condition. Deactivation of the facility was completed in 1968, and it was placed in a surveillance and maintenance mode.

An initial proposal for decommissioning the SRE was prepared in 1974. This proposal was prepared to get the project under way and was issued without having site characterization radiation data available (surveillance data from 1966 were utilized). The proposal requested funding for a 3-year period with a dismantlement plan to be prepared during the first year.

The dismantlement plan was prepared in 1975. It defined the methods to be used in removing contaminated equipment, systems, and materials, including cutting and removing the reactor vessels using a plasma torch. It also identified the need for development work and a test facility to establish and demonstrate cutting methods and parameters. This development work was accomplished in the 1975 through 1977 time period while auxiliary systems and structures were being removed. Reactor vessels were cut and disposed of in 1977 and 1978.

Removal of basement vaults, equipment embedded in the below grade structures, and substructure rock and soil extended from 1977 through 1981. Extensive contamination found deep in the bedrock added to the work and required extended time to complete.

The final schedule for decommissioning the SRE is shown in Figures 82, 83, and 84. The original 3-year funding period (1975 through 1978) was established to initiate the program. Subsequent schedules evolved to accommodate funding limitations and to meet the needs of changed conditions which expanded the work scope.

5.2 COST

The proposal submitted in 1974 to initiate the dismantlement covered a 3-year period and requested $6.9 million in 1975 dollars. The budget request was prepared before the dismantlement plans were prepared and without a radiation characterization survey.

The dismantlement plan had not been prepared when the original estimate was made. It was one of the first activities funded by the original budget request. The dismantlement plan identified the need for a test facility and a 2-year program to develop, test, and demonstrate plasma arc remote cutting capabilities to segment and remove the reactor vessels and internals.

Changing conditions encountered during the decommissioning added complexity and additional work to the project. As the work progressed, radiation surveys detected more contamination than anticipated in areas surrounding the reactor building, in radioactive waste systems, in the reactor building equipment vaults, and in the bedrock beneath the vaults. Contaminated water leaked into the bedrock. Extensive excavation was required to follow and remove contamination which seeped into cracks in the bedrock. Building foundations had to be temporarily braced and shored to prevent collapse.

As-built conditions differed from drawing information, resulting in added cost and time to perform additional work. Spills of radioactive material during reactor operations had been cleaned to less stringent criteria, resulting in additional cleaning and/or removal of material for burial in order to meet current criteria for release for unrestricted use.

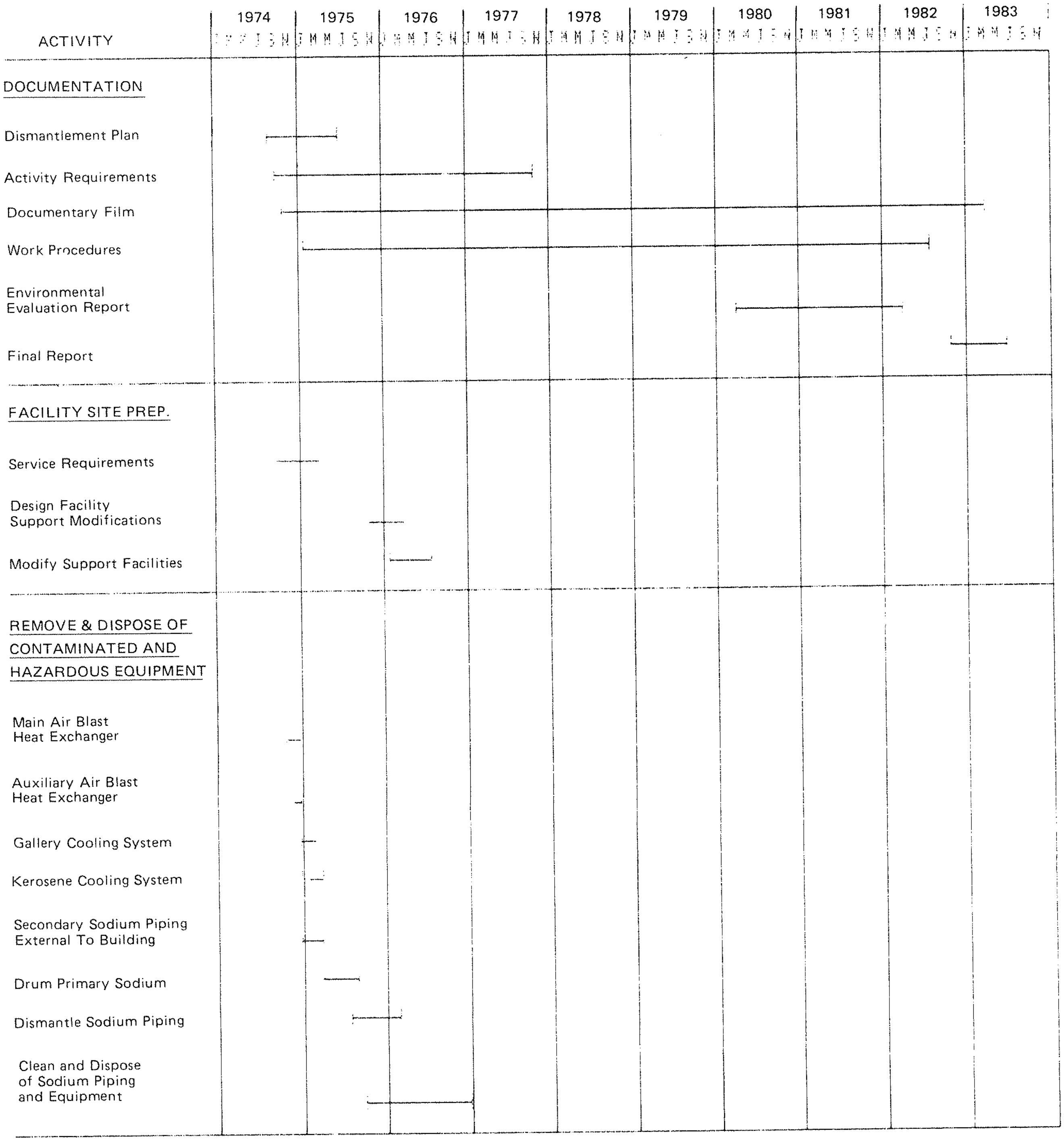


Figure 82. SRE Decommissioning Schedule

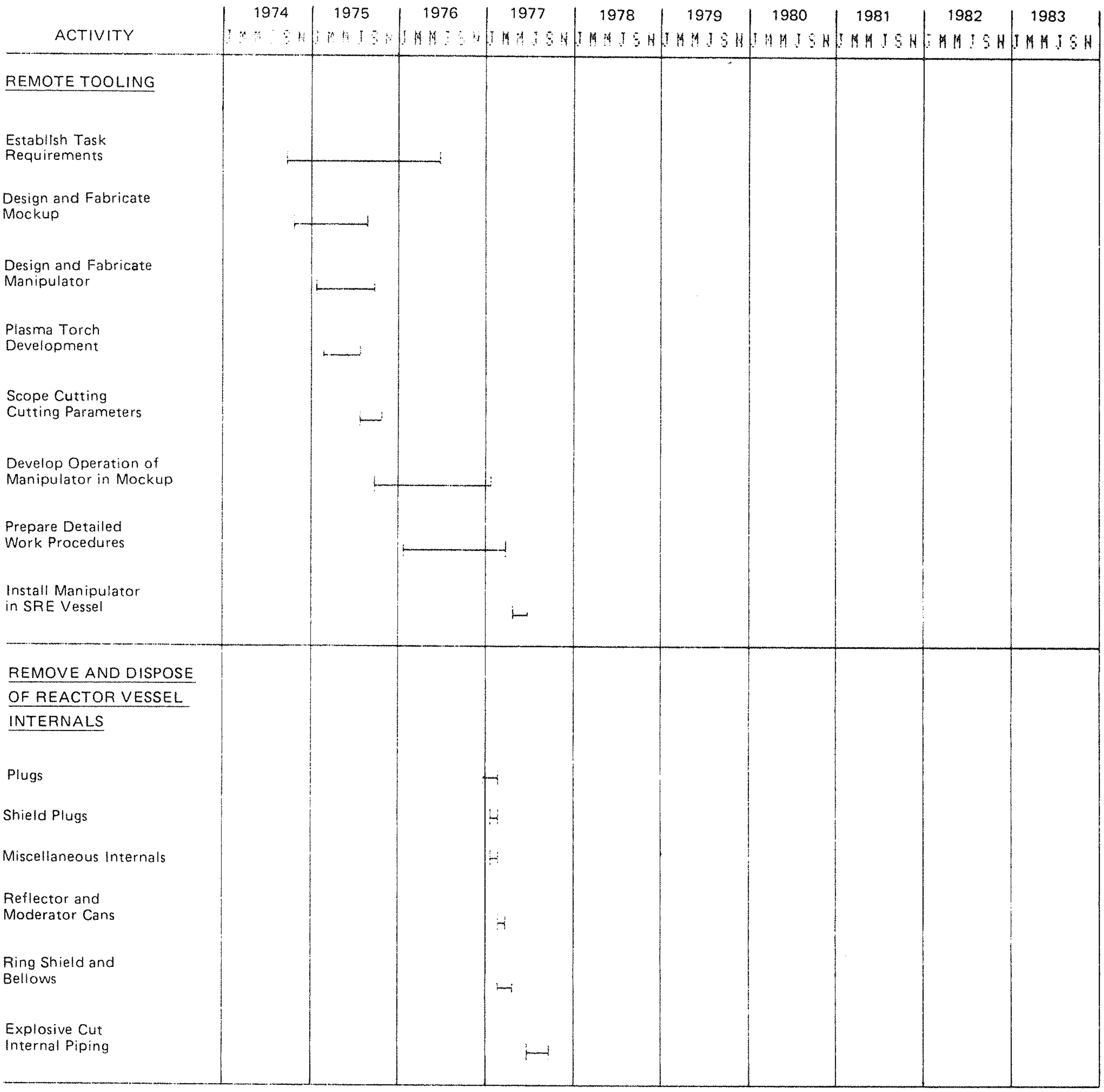


Figure 83. SRE Decommissioning Schedule

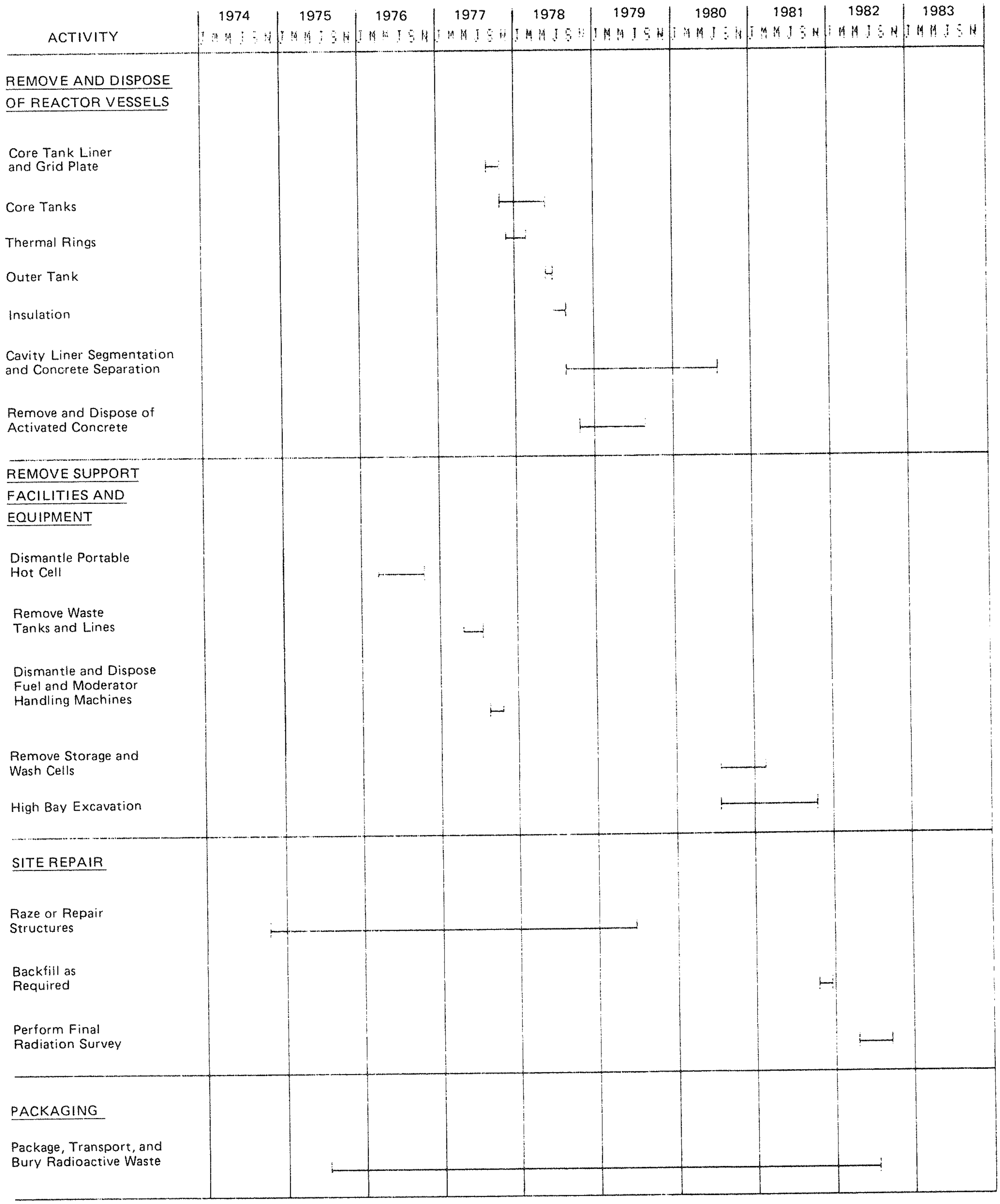


Figure 84. SRE Decommissioning Schedule

Requirements for radwaste disposal changed significantly during the contract period, and waste shipping and burial costs increased by a factor of 2-1/2 from 1974 to 1982.

An abnormally high inflation rate during the extended decommissioning period increased the costs. The original estimate of $6.9 million was in 1975 dollars. The final cost included approximately $4.3 million attributable to inflation.

The combined effects of changing conditions, increased work scope, increased waste burial costs, extended schedule, and the abnormally high inflationary period resulted in a final cost of $16.6 million in 1982 compared with the 1974 estimate of $6.9 million in 1975 dollars.

The SRE decommissioning project has shown that thorough and complete engineering studies, planning, and site survey data are needed at the onset of a major decommissioning project to allow more accurate development of cost/budget/schedule requirements.

The total SRE dismantlement costs, summarized by major cost elements, are listed below:

		Cost ($K) (Including Fee)
1.	Program Management, QA, and Planning	$2,410
2.	Development	1,465
3.	Reactor Structure	1,370
4.	Radioactive Heat Transfer Systems	1,175
5.	Reactor Building Substructure	2,805
6.	Reactor Building Superstructure	940
7.	Radwaste Systems	1,055
8.	Support Buildings and Structures	985
9.	Nonradioactive Auxiliary Systems	95

		Cost ($K) (Including Fee)
10.	Waste Shipment and Burial	2,650
11.	Rectification	465
12.	Health & Safety	1,220
	Total	16,635

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6.0 WASTE VOLUMES GENERATED

Types of waste and quantities generated in SRE D&D are shown below.

Type of Waste	Shipping Container Volumes, ft^3					Total Volumes (ft^3)
	King-Pac®	Boxes	Casks	Drums	Unboxed	
Activated vessel components		10,611	698		645	11,954
Contaminated components		51,490	1,729	1,025	595	54,839
Contaminated soil and concrete	61,874			1,481		63,355
Absorbed alcohol and other solidified liquids				4,993		4,993
Disposed liquid				1,270		1,270
Subtotal	61,874	62,101	2,427	8,769	1,240	
Total						136,411

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7.0 OCCUPATIONAL EXPOSURE TO PERSONNEL

At the beginning of the Decommissioning Facilities Program, an Operational Safety Plan[3] was developed. It provided the basic requirements and implementation for radiation safety, industrial safety, and industrial hygiene and was used throughout all the projects in the decommissioning program. All exposure to radioactive materials was under the surveillance of the Radiation and Nuclear Safety Group. The radiation protection standards established by the DOE Manual, Chapter 0524 were used. These standards limit individual occupational exposure to radiation to 3 rem in a calendar quarter and 5 rem in a calendar year for whole-body exposure. For the extremities (hands and feet), the limits are 25 rem per quarter and 75 rem per year; for the skin, the limits are 5 rem per quarter and 15 rem per year. In addition, a limit of 3 rem per year was used as a guide for controlling individual radiation exposures.

Film badges, processed by an independent laboratory, provided the legally documented record of external exposure. These badges were processed monthly during the early part of the project and quarterly later.

Direct-reading pocket dosimeters were used daily by each employee to monitor external radiation doses on a task. The dosimeters were charged at the beginning of each shift, and the indicated doses were recorded at the end of work that day. These records were summarized weekly and plotted for each employee by the health physics staff to provide a current and visible indication of exposure control.

Surface contamination in the work areas was monitored frequently by smear surveys, with the smears counted on automatic gas flow proportional counters.

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[3] J. D. Moore and E. L. Rody, "Operational Safety Plan for the AI Decontamination and Disposition of Facilities Program," SRR-704-990-001 Rev. B (October 1975).

Establishment of well-defined work areas with step-off lines aided in minimizing the spread of contamination. Airborne contamination was continuously monitored by use of continuous air monitors with automatic alarms. Airborne contamination was rarely a problem because of the control afforded by use of local containment and ventilation.

Application of special techniques to jobs that would normally create severe airborne contamination was very satisfactory. The plasma-arc cutting of the core vessel under water released a negligible amount of radioactive material. The outer tank was cut with the plasma arc in air. Airborne radioactivity from this operation was controlled by use of a temporary plastic enclosure exhausted by the facility radioactive exhaust system. Attachment of a vacuum cleaner hose to a concrete scabbling tool essentially eliminated any spread of dust from a very dusty operation. The vacuum cleaner exhaust was passed through a HEPA filter before release.

Internal exposures to radioactive material were determined by quarterly analysis of urine samples. Following suspected exposures, samples were also analyzed by an independent laboratory. Only one quarterly dose exceeded 1.25 rem, and that was allowable on the basis of prior exposure records.

Ambient radiation, particularly in areas subject to change in exposure rate, was monitored by a remote area monitor with an automatic alarm. Surveys were also performed by the health physics staff, and exposure rates were posted on a facility layout plan.

The number of the operating personnel varied in response to the type and amount of work. Radiation doses received at the SRE have been identified and recorded. The cumulative group dose for this project was 89 man-rem. It was well below the amount that would have been accumulated had each worker been exposed at a rate that would result in receiving the limit of 5 rem per year. In fact, it compares well with the cumulative group dose that would result from exposures at a rate consistent with the current DOE guideline of 1 rem per year to be used for the design of new facilities. This is shown in Figure 85.

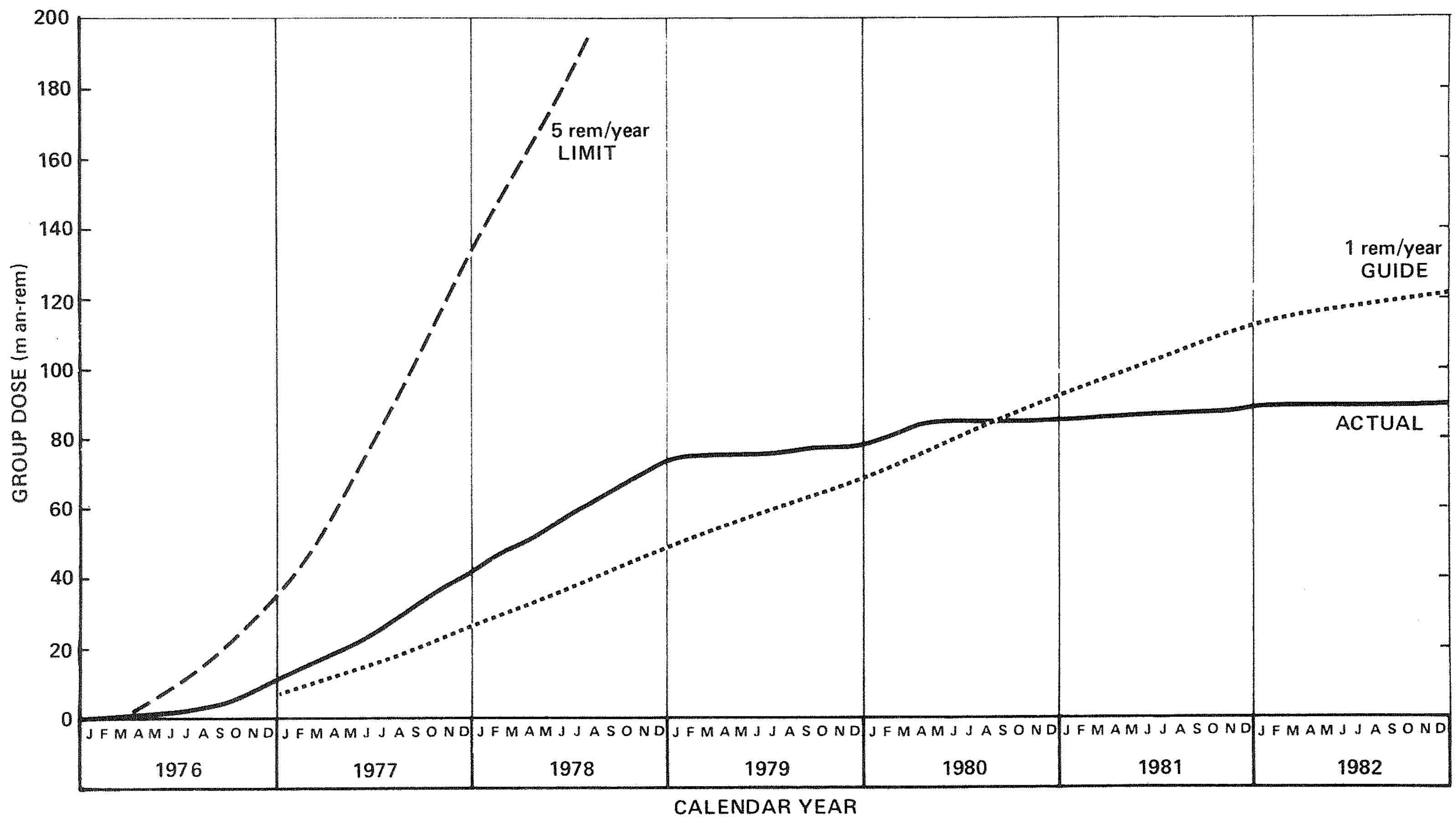

Figure 85. Cummulative Group Dose for All Personnel in Decommissioning the SRE

Much of the group dose was received in specific high-exposure operations. Disposal of the primary sodium auxiliary systems required considerable hands-on work, cutting and removing piping and components containing residual sodium. Removal of the core tank and core tank bottom involved transfers between the water-filled reactor vessel and a water-filled storage pit of components with exposure rates of 10-30 R/h at 1 m. Doses were controlled primarily by minimizing time and maximizing distance between the active pieces and the operators. The thermal rings produced a field of 4 R/h inside the ring, but operators were kept outside where the self-shielding of the 5-in.- thick steel reduced this to 400 mR/h. Decontamination of the water-filled storage pit after disposal of the activated material was another relatively high-exposure task.

Much of the work dealt with highly dispersible radioactive material: contaminated sodium and sodium compounds, contaminated water and alcohol, slag and dust from cutting activated steel, and contaminated or activated concrete and soil. In spite of this, controls were successful in restricting internal contamination of personnel by inhalation or ingestion. While the occurrence of positive bioassay results clearly suggested exposure episodes (and most of these clusters of positive indications were associated with incidents), no long-term depositions have resulted. Supplemental whole-body counts for gamma radiation showed Cs-137 burdens of 5 to 24 nCi, the largest value indicating less than 0.1% of an allowable body burden for this isotope. Assuming an equal amount of Sr-90 to be associated with this, but undetectable externally, the fraction of a body burden would be 1.2%.

All results of the radiation monitoring program showed that exposure was controlled well below established limits and, in fact, can be held to levels that are consistent with current design criteria for operation of new plants.

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8.0 FINAL FACILITY CONDITION

The SRE site, at the initiation of decommissioning, is shown in Figure 86, and the SRE site, at completion of decommissioning, is shown in Figure 87. Figure 88 is a plan view sketch of the SRE site.

The total site has been decontaminated and hazardous materials, primarily sodium, have been removed from the site. The disposition of the SRE support facilities or their present condition is as follows:

1) Temporary Hot Waste Storage 686. The above-ground facility was razed and the contaminated materials were packaged and shipped to a burial site. The area was surveyed and reported acceptable for unrestricted use in "Radiological Survey Results Release to Unrestricted Use, SRE Region V," Document N704TI990031.
2) Liquid and Gaseous Radwaste Waste Holdup and Decay System 653. The compressors, tanks, and piping were excavated and removed for shipment to burial. The remaining vaults were decontaminated and partially demolished. Contaminated rubble and soil were packaged as waste and sent to burial. The area was backfilled. The radiological survey of the area was also reported in Document N704TI990031.
3) Retention Pond and Drainage Control Dam 773. The retention pond was drained and contaminated silt and soil were removed. Recent rains have partially filled the pond. The water is noncontaminated. Water from the pond will be drained into the Simi valley as as soon as Rockwell's request for this action is approved by the State of California. The Radiological survey results are reported in "Release to Unrestricted Use, Region VII," Document N704TI990033.
4) Sodium Cleaning Pad 723. This facility was used to clean noncontaminated sodium from equipment and materials. The area is clean. Survey results are reported in "SRE Region I," "Document N704TI990027.

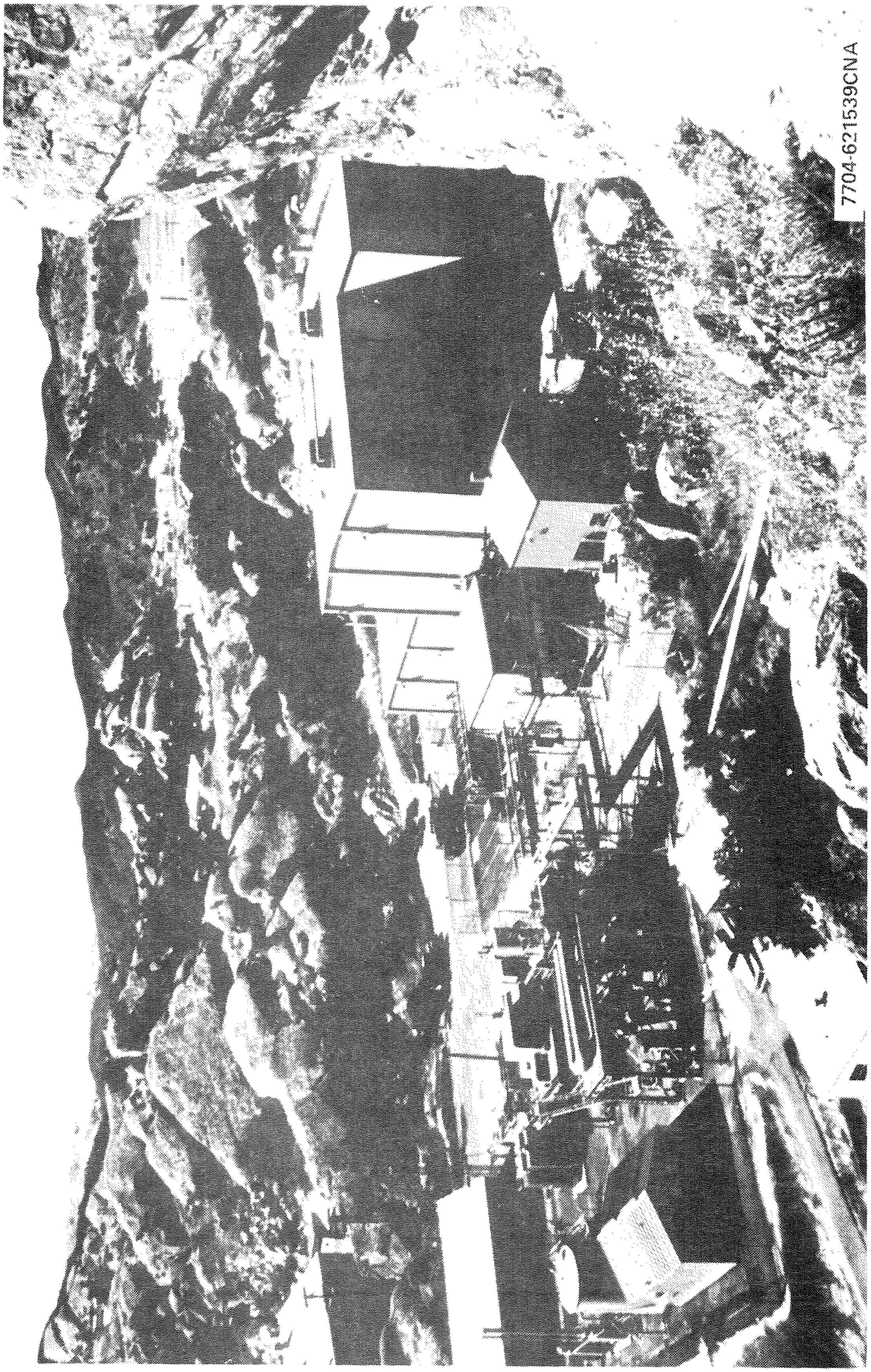

Figure 86. SRE at Initiation of Decommissioning

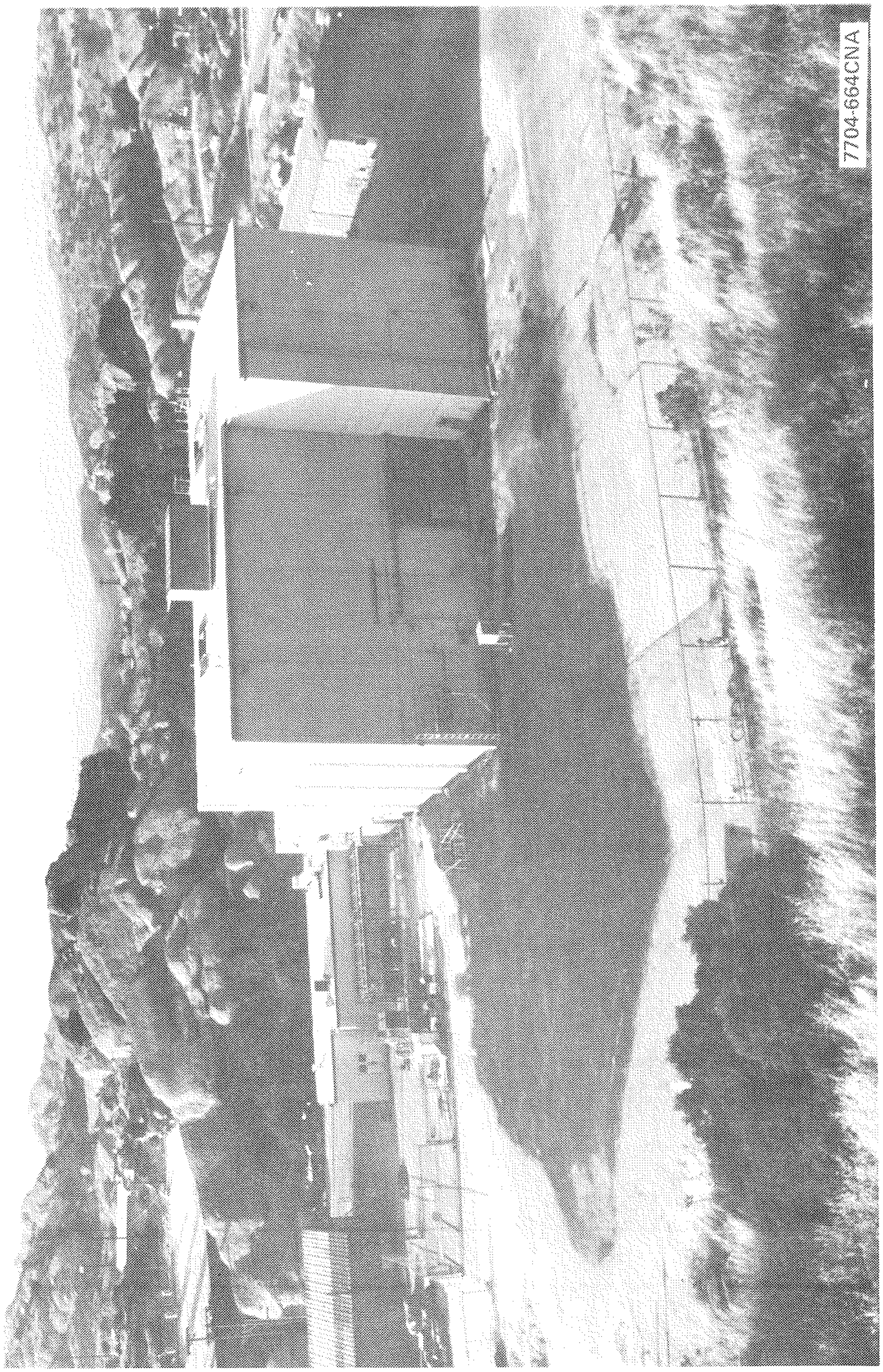

Figure 87. SRE at Completion of Decommissioning

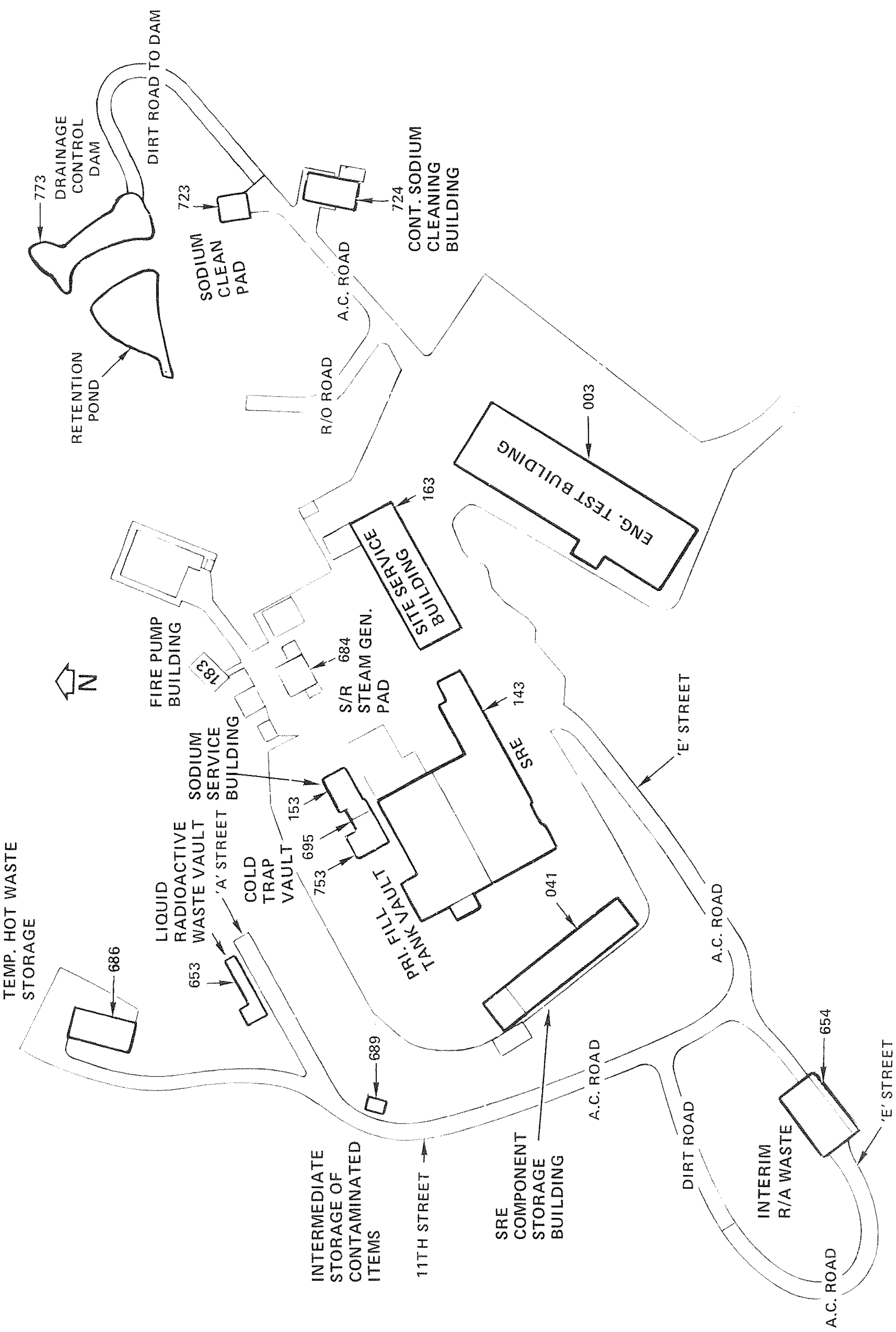

Figure 88. Plan View of SRE Site

5) Contaminated Sodium Cleaning Building 724. The building was decontaminated and transferred to another Santa Susana field laboratory site for use in cleaning sodium components. The remaining pad and surrounding area were decontaminated. Survey results are reported in "SRE Region I," Document N704TI990027.

6) Engineering Test Building 003 was used to support SRE and other programs activities. The decontamination of Building 003 was reported in a separate final report.

7) Interim Radioactive Waste Facility 654. This facility was operated in conjunction with the Radioactive Materials Disposal Facility (RMDF). All above-grade SRE materials and equipment have been removed, and the surface area has been decontaminated. The final disposition of the Interim Radioactive Waste Facility will be associated with the RMDF disposition.

8) SRE Component Storage Building 041. This facility was used to store waste during the SRE decommissioning. This decontaminated facility remains and will be used for clean storage. Radiological surveys of Building 041 are reported in "SRE Building 041," Document N704TI990037.

9) Steam Generator Pad 684. The steam generator pad and all the nearby concrete support structures for the nonradioactive systems associated with the production of electricity by Southern California Edison were demolished. The resulting clean rubble was used in the backfilling of the SRE building excavations. Radiological survey results of the steam generator area is reported in "SRE Regions II, VII, and X," Documents N704TI990028, N704TI990033, and N704TI990036.

10) Site Service Building 163. This facility was used to repair contaminated components. In recent years, the east end of the building was used for constructing shipping containers. The facility was decontaminated, and the basic structure remains for continued use as a box shop. Radiological survey results for Building 163 are reported in "SRE Region II," Document N704TI990028.

11) Intermediate Storage of Contaminated Items 689. This facility was totally removed prior to the SRE decommissioning. The contaminated blacktop in the area was removed and the area was repaved. The radiological survey results are reported in "SRE Region IV," Document N704TI990035.

12) The Primary Fill Tank Vault 753 and Cold Trap Vault 695. These were contaminated. The total below-grade structure was removed and the area was backfilled and paved. The radiological survey results are reported in Document N704TI990038.

13) Sodium Service Building 153. This building was razed. The concrete pad and footings were excavated to provide access for excavation equipment into the main SRE building. This area has been backfilled and paved. The radiological survey results are reported in Document N704TI990035.

14) SRE Reactor Building 143. The north portion of the SRE building contained the reactor and supporting systems. The contaminated systems were removed, contaminated soil and bed rock were removed, the excavations were surveyed, then backfilled and paved. After the building superstructure was decontaminated by sandblasting, it was painted. Lighting and fire detection systems were restored. The south portion of the building housed the control rooms, offices, restrooms, and electrical services. These areas were decontaminated as necessary. The final survey results of Building 143 are reported in Document N704TI990038.

15) SRE Complex. Only the reactor building, 143; the storage building, 041; the site service building, 163; and a fire pump building on the northeast end remain. The total area surrounding the building has been repaved.

Personnel access restrictions have been removed, and Buildings 041 and 143 are being used as warehouses.

The release of the SRE complex to unrestricted use will be contingent on the acceptance and approval of this final report and the report to be issued by Argonne National Laboratories survey team who conducted independent surveys at the site. In addition, the release to unrestricted use will be based on the considerations presented in the "Sodium Reactor Experiment Decommissioning Environmental Evaluation Report," DOE-SF-4, ESG-DOE-13367. This document, prepared by Rockwell ESG, states that the proposed action – release for unrestricted use – does not present any significant impact on the environment. Ground water will not be contaminated. Decontamination on various areas of the SRE has effectively reduced radioactivity in the soil to better than allowable limits. Contamination of water moving through these soils is highly improbable. Samples of surface and subsurface water has indicated radioactivity that is less than levels allowed for unrestricted areas. Radioactivity will not be introduced into the food chain from this pathway. Resuspension of radioactive materials into the atmosphere will not occur since material near the surface, which could become airborne, contains very low levels of radioactivity and no significant dose will result from inhalation of this material. Direct external exposure from the SRE will not reflect an increase above that naturally occurring at the site.

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9.0 LESSONS LEARNED

A review of the technical and management approaches taken to accomplish the SRE decommissioning indicates that, as in any program, the total performance could be improved if the project were to be repeated. This review has produced recommendations that will be useful in future programs. The SRE project has been divided into 11 major tasks. Each task is reviewed and analyzed, and recommendations are presented.

9.1 PLANNING AND ENGINEERING

The overall program incorporated plans for health and safety, training, and quality audit and assurance as well as development of approaches, responsibilities, and schedules. Beginning with a review of the facility design and operating history, a dismantling plan was prepared, followed by activity requirements and detailed working procedures for each task. Engineering was performed and control documents were developed for the tasks and for the development and special tooling designs. This intensive planning effort proved beneficial, both in technical performance and in effective control of costs. As decommissioning efforts proceeded, differences between design drawings and actual site conditions were found because facility drawings were not always updated. Available radiological survey data were insufficient for planning purposes, necessitating site characterization surveys to aid the planning effort.

<u>Conclusions and Recommendations</u>

1) Comprehensive radiological surveys should be performed prior to the establishment of schedules and cost estimates. Adequate work scope definition is impossible without thorough knowledge of the extent of radiological contamination.
2) Thorough engineering studies of alternatives and then comprehensive planning should be prepared as early as possible to adequately define the work to be done and permit accurate estimates and schedules to be prepared.

3) During reactor operations, records of incidents, radiological surveys, repairs, reconstruction, and any alterations should be maintained in sufficient detail to aid the decommissioning effort.
4) Facility drawings should be kept current.
5) Construction photographs should be taken and retained.
6) The design of nuclear facilities should incorporate decommissioning procedures.

9.1.1 Disposition of SRE Structures and Systems

The selected SRE decommissioning mode was the complete dismantlement of contaminated structures and systems. However, in the initial planning, it became apparent that the specific SRE facilities and buildings could be useful during the decommissioning procedure. Consequently, facilities such as the radioactive exhaust system and the personnel change room with its showers and radioactive waste holdup system were kept operable throughout the decommissioning procedure. In addition, the SRE reactor building provided a containment structure for airborne contamination generated by the decommissioning operations. The building also provided a health physics laboratory, office space for supervisory personnel, and restrooms and lunch rooms.

When the excavation of below-grade structures and systems began, the difficulty of excavating around building support columns and the need for shoring walls required a decision regarding the advisability of retaining the SRE superstructure. An engineering study showed that a schedular delay of several months would be necessary to dismantle the building, provide temporary containment structures for the decontamination operations, and relocate the health physics laboratory, personnel services, and supervisory offices, if the building was not retained.

9.2 DISPOSAL OF NONCONTAMINATED SRE SYSTEMS AND SUPPORT FACILITIES

The secondary sodium system, steam and electric generation facilities, airblast heat exchangers, water supply system, and external portions of the kerosene cooling system were the principal elements of the noncontaminated systems and facilities removed as the first decommissioning steps. A demolition and salvage contractor was hired to perform the removal. Because the work areas were physically separate from the contaminated areas of the SRE, the removal required no special techniques, other than industrial safety controls for the sodium and kerosene involvement and monitoring by Health Physics and Safety personnel. The cost for salvage contractor services was offset by the value of the materials removed.

By performing this work first, access to the contaminated systems of the plant was improved, and interference between work on contaminated and noncontaminated systems was prevented. The time spent removing noncontaminated systems was also used for planning and development work for the contaminated systems. It should be noted that safety precautions used by outside salvage contractors vary greatly, and therefore their operations must be monitored closely by the project onsite supervisor.

Conclusions and Recommendations

1) Maximize the use of conventional salvage contractors for noncontaminated work and provide separation from contaminated work areas.
2) Consider salvage value to offset decommissioning costs.
3) Schedule removal as early as possible to simplify access.
4) Assess contractor's safety procedures and safety record.

9.3 DISPOSAL OF CONTAMINATED SODIUM

During the deactivation process, 54,950 lb of slightly radioactive sodium (4 mR/h at the surface of each 55-gal drum of sodium) had been drained into the storage tank from the reactor and primary piping system. This sodium was later put into drums for shipment to Hanford, Washington, for possible use in other programs.

Residual sodium coating the interior of system components, piping, and the reactor vessel required special procedures for disposal. Sodium deposits and heels in piping, etc., were heated and drained into drums. Then, alcohol was added to react with the residual sodium films in piping, components, and the reactor vessel system. The alcohol was added slowly into the particular configuration, with control of the rate of flow based on the quantity of hydrogen generated by the reaction. Explosive mixtures were prevented by diluting the hydrogen with nitrogen. About 8,000 gal of alcohol were used. The resulting contaminated liquid was pumped into 55-gal drums containing diatomaceous earth, which absorbed the liquid and made it acceptable for burial. The drums were shipped to the licensed commercial burial site at Beatty, Nevada. (Current regulations do not allow such disposition of liquid wastes.)

Several sodium chemical passivation options were studied. The alcohol reaction was selected because it is comparatively slow and permits greater control. The sodium passivation procedure was carefully planned and conducted. Particular care was exercised to exclude air from the reactor vessel. Air contact with heated sodium can result in a fire.

Conclusion and Recommendation

Alcohol passivation of sodium films, although an effective and safe method, results in problems with alcohol disposition. Other sodium passivation methods such as nitrogen/steam reaction should be considered.

9.4 DISPOSAL OF CONTAMINATED SRE PERIPHERAL SYSTEMS AND FACILITIES

The next step in the decommissioning sequence was the disposal of the peripheral systems and facilities. Performing this step next in the sequence had the advantages of decreasing the area of operation in the shortest time, improving the familiarity of the crews with decontamination work, and providing better access for subsequent activities.

The peripheral systems and facilities disposed of were: (1) the gaseous and liquid waste holdup system on the hillside north of the reactor building, (2) the hot waste storage facility on the same hillside, (3) the secondary sodium storage handling and purification systems, and (4) the drainage retention pond. The vault walls of the liquid waste holdup system and the soil below the system components were contaminated. A hydraulically operated Hy-Ram was used to peel off several inches of concrete from the walls and floors. The soil below the tanks and several feet downslope was also removed and packaged for burial.

The capability and versatility of the Hy-Ram was developed in the concrete decontaminated operations. An early objective of the SRE decommissioning project was to minimize the total costs associated with the generation of contaminated waste. Whenever possible, selective rather than bulk removal techniques were used to minimize the waste to be packaged and sent to burial. The Hy-Ram could be equipped with chisel-like tools that could selectively peel off layers. It was also used to demolish the bulk noncontaminated concrete in vaults and walls. Consequently, its simplicity, versatility, accessibility, and cost effectiveness led to its evolution as the workhorse for the material removal operations.

The hot waste storage facility, essentially concrete and steel tank structures mounted on a concrete pad, were easily removed. The storage structures were above grade, and radiological surveys indicated that the soil had not been contaminated.

The secondary sodium systems, consisting of a sodium storage tank, sodium system loading facility, pumps, hot and cold traps, piping, and valves, were dismantled. Residual sodium in these components was reacted with steam. The cleaned components were free of contamination and were sent to salvage as scrap.

A radiological survey of the water in the drainage retention pond showed no contamination. The water was pumped into area drainage channels, and the silt in the pond was surveyed and sampled. Several slightly contaminated areas were detected; after these were removed, the area was resurveyed and found to be free of contamination.

Conclusions and Recommendations

1) Work on slightly contaminated peripheral systems should be used to train the work force and establish procedures.
2) Selective removal of contaminated materials is cost effective and should be used.
3) The Hy-Ram should be considered for both selective and bulk concrete removal.

9.5 DISPOSAL OF SUPPORT SYSTEMS

The support systems consisted of the machines for handling the fuel and moderator elements; the fuel storage, fuel wash, and examination cells; and the inert gas system. The large fuel and moderator handling machines were stripped of external appurtenances and then shipped to a burial site. Dismantling and decontaminating the basic structures would have been too time consuming and costly. This approach — intact shipment to burial — was used for disposing of all the large system components.

Fuel storage cells, fuel wash cells, and other cells located in the floor of the SRE high bay were excavated.

Some of the cells were located close to key building support structures and were deeply imbedded in bedrock. Removing these required techniques that would not damage the building structure or present a hazard to the personnel involved. To provide this protection, it was necessary to install shoring to prevent cave-ins. A consulting firm was hired to design and install the shoring. Holes were drilled from the grade-level floor to a depth of 60 ft to provide for the pouring of 2-ft-diameter reinforced concrete pilings. The pilings were slotted to accept timber spanners as excavation progressed.

To control contamination during dismantling and packaging for shipment, the storage and wash cells were filled with a solidifying foam and capped before being removed and later cut into manageable lengths.

The Hy-Ram was used to free the cells from their concrete support structures and to demolish the bulk concrete.

The fuel examination hot cell complex was also located below grade at a depth of about 20 ft. To decontaminate this system required removing hot cell windows, overboring pipe penetrations, cutting through concrete to free piping, removing stainless steel cell lining, and spalling off contaminated concrete surfaces.

The following procedure was used to remove the two storage tubes located in the floor of the examination cell (bottom of tubes, about 50 ft below grade):

1) A flat concrete base was poured from which a rock drill could operate.
2) Approximately 100 deep holes were drilled to weaken the bedrock.
3) The weakened bedrock was excavated using the Hy-Ram and a backhoe.
4) The storage tubes were pulled out.
5) The excavation was backfilled with grout to offset the effects of ground water.

Once set, the grout sealed the excavation and provided a firm base for further backfilling. Figure 89 shows the storage tubes imbedded in bedrock and the extent of the excavation necessary for their removal. The examination cells were partially dismantled, the inner cell liner was removed, and the drain lines were excavated.

The noncontaminated concrete structure of the hot cell complex was not completely demolished.

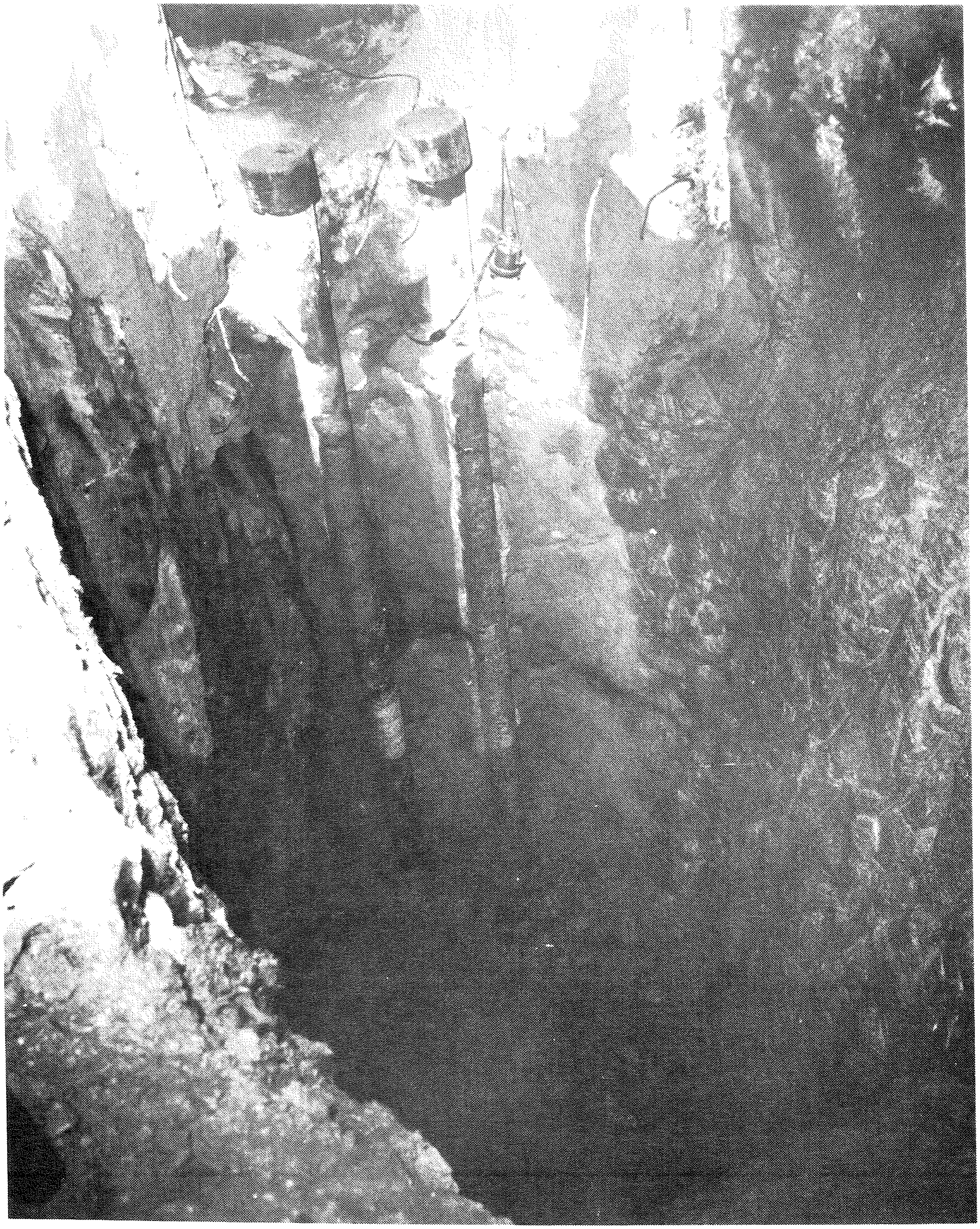

Figure 89. Storage Tubes Imbedded in Bedrock

Conclusions and Recommendations

1) Remove large components without major disassembly (fuel handling machines, plugs, shields, etc.) and transport to burial sites in one piece wherever possible since this may be most cost effective.
2) Use a solidifying foam internally to fix contamination in components so that they can be sectioned without spread of contamination.
3) Include provisions for ease of removal and for avoiding spread of contamination when installing system components into the bedrock.

9.6 DISPOSAL OF REACTOR VESSELS

The disposal of the reactor vessels, because of the significant induced radioactivity present in the metal, required development and use of remotely operated equipment and tooling. A manipulator with a plasma-arc torch-cutting head was used to remotely cut the vessels under water into manageable sections. These sections were remotely transferred to a water storage basin and then placed in cask liners for later disposal. Reactor vessel internals (such as downcomers and core clamps) were removed using shaped-charge explosives. Although explosives were used in removing the internals and proved to be very effective, it was shown later than the plasma-arc cutting technique could have been used also.

After the highly activated inner reactor vessels were removed, the thermal shield rings were lifted out. Since these were only slightly radioactive, they could be cut up for disposal safely with automated, remote, and oxy-acetylene cutting on the reactor room floor.

Removing asbestos insulation from between the vessels was a hands-on operation. After removal, it was sent to a burial site as low-level radioactive waste.

The engineering, tooling design, and cutting parameter development that preceded the use of the polar manipulator and plasma torch greatly facilitated this portion of the work. A mockup of the reactor vessel was constructed early in the project to permit development and training. The operators were adept at cutting before beginning work on the SRE. The vessel segmentation was conducted under water to cut 1-yd^2 sections from the vessel, remotely move these to a water-filled vault, and selectively stack the sections into cask liners.

The success of the vessel segmentation project was highly dependent on maintaining good underwater viewing. The TV camera attached to the manipulator was helpful as long as water clarity was maintained. A pump and filter system was used, and appropriate radiation shielding was provided. With proper maintenance, the unit produced good results.

Ground water was found in primary system vaults when they were first opened. A piping vault was also used for underwater segment storage. It was constructed of concrete with a few pipe penetrations. In preparing for the work, the vault was sealed water tight. The surfaces were cleaned, and the cracks were filled. The pipe penetrations were sealed, and the entire surface was coated with an epoxy material. Despite the preparation, a water leak developed. It may have resulted from shocks caused by the explosive cutting operation. As soon as the loss of shielding water was noted, a steel tank was fabricated to fit inside this vault and contaminated material and water were transferred to the tank. Water containing fission and activation products migrated into the soil and bedrock at the north and east quadrant of the building.

Samples were analyzed to determine the degree of activation of the concrete biological shield surrounding the reactor. This was found to be 10 in. into the shield from the inside. The lower portion of the shield was not contaminated and was buried in place. To minimize the generation of contaminated waste, the Hy-Ram was again found useful in accomplishing selective removal. In this case, the outside annulus of the concrete structure was removed (with close radiological surveillance) as noncontaminated material and saved for later use as backfill material. The inside contaminated concrete was reduced to rubble and packaged for waste disposal.

Conclusions and Recommendations

1) Careful planning, engineering, tooling development, mockup trial runs, and training are recommended for the successful completion of the task and to simplify the work.
2) Plasma-torch segmentation is recommended as a successful technique for remotely cutting highly contaminated reactor vessels.
3) Explosive cutting is feasible for remote underwater pipe cutting, but plasma-torch techniques should be considered.
4) The use of existing plant structures for critical operations such as storing of contaminated water should be avoided. New watertight structures should be engineered, fabricated, and installed for decommissioning processes.
5) Water clarity for viewing capability is essential for underwater operations. Therefore, an adequate filtration system is necessary.
6) Where access permits, the Hy-Ram should be used for selective removal of layers of the biological shield.

9.7 EXCAVATIONS

The SRE high bay was excavated to an average depth of 26 ft. The extensive excavation was necessary to remove the below-grade reactor structure, primary piping system vaults, fuel storage and cleaning cells, and the hot storage tubes.

In addition, an equivalent volume of soil and rock was removed from outside the north and east sides of the building. This excavation was necessary to remove the primary sodium storage vault, service system vault, and piping systems. Additional excavation was required to remove contaminated soil in the northwest quadrant of the building. The excavation extended beneath grade beams and column footings. Temporary bracing and shoring were installed to support local areas while the contaminated soil was being removed.

Two sources contributed to the extent of the soil contamination below the vaults. When the contaminated primary service vault was first opened, considerable water was found in the vault. This water either seeped in through the walls and floor slab from a ground water source or came in through the roof during rainy periods. The second source was the leak from the pipe vault pool used to store contaminated pieces from the reactor vessels.

Localized contaminated areas were decontaminated using manually operated tools. Large areas were excavated using the Hy-Ram. Since the SRE structure and some of the systems extended into underlying bedrock, some of the contamination in the soil extended into fractures in the bedrock. This necessitated some laborious machine and manual removal operations. After contaminated material from the below-grade areas had been removed, the area was surveyed and found to meet acceptance criteria, and the excavation was backfilled. Clean material removed from the excavation was returned. Additional backfill material was purchased from a nearby land development operation.

A consulting A&E firm was employed to prescribe and design bracing for the building columns during the period when they were extensively exposed by the excavation.

Conclusions and Recommendations

1) Carefully segregate noncontaminated materials to provide backfill and to reduce the volume of contaminated waste.
2) Give constant attention to safety considerations for personnel and remaining structures for the duration of the project.
3) Carefully assess engineering detail for all operations (including minor ones) during the progression of the job.

9.8 DECONTAMINATION OF STRUCTURES

Surface decontamination activities preceded most of the dismantling and removal operations. This decontamination was performed to reduce radiation exposures to personnel and to enable the bulk of structures or equipment to be disposed of as noncontaminated. In other cases, decontamination of the exterior of large items (such as the fuel handling machines) facilitated packaging and handling for disposition.

In a program to develop decontamination techniques, we used available literature, experiences of ourselves and others, and demonstration to produce an optimum set of techniques. Generally, we found that vacuuming, followed by applications of a foam containing decontaminating agents, was the best way to clean painted and metal surfaces and to initially clean various other surfaces. The approaches used in decontaminating concrete surfaces are described in Section IV. A scabbler device that spalls the concrete surface and is equipped with local vacuum and air cleaning proved quite effective.

On walls, ceilings, and the two high-bay cranes in the main building, contamination was fixed in the paint. The areas were decontaminated in two steps. First, paint, particularly from horizontal surfaces, was removed by sandblasting. Then, the few remaining spots were decontaminated by scrubbing or surface removal. Before sandblasting, all extraneous equipment, piping, and ducts were removed. No attempt was made to provide local aerosol control; instead, the entire high-bay area was sealed to contain contaminated dust and sand. All personnel in the area wore protective clothing and respiratory protection.

<u>Conclusions and Recommendations</u>

1) Pay careful attention to decontamination techniques to optimize contamination and exposure control, reduce waste volumes, and reduce cost.

2) Consider a foam decontamination approach for removal of loose contamination on walls. It minimized the generation of contaminated water and provided other benefits.
3) Decontaminate large surface areas of concrete floor with the scabbler.
4) Use sandblasting as an effective method for decontaminating large areas where contamination is fixed in paint. (A commercial contractor was used; however, since their usual objective is to clean in preparation for painting, and they neglect non-visible areas, the contractor's work required constant supervision.)

9.9 WASTE HANDLING

All materials leaving the SRE site were radiologically surveyed. Clean materials were sent to the salvage yard or removed by a salvage contractor. Contaminated materials were packaged in accordance with approved procedures and shipped on special trailers for land disposal. At first, these materials were shipped to a commercial site in Beatty, Nevada. Later in the program, such wastes were shipped to a DOE site at Hanford, Washington. Contaminated water was processed in an evaporator at the Radioactive Materials Decontamination Facility at Santa Susana. Sludges were solidified by mixing with concrete and then were shipped to burial.

With the large volume of slightly contaminated concrete and soil to be sent to burial from this project, the choice of suitable packaging was crucial to the achievement of optimum cost effectiveness. Tri-Wall Chemical King Pak™ containers (Tri-Wall Containers, Inc.) were found to be the most economical packaging for this category of waste. This is a fiberboard assembly about 1 yd^3 in volume. A typical loaded weight was about 2000 lb. The package was fastened to a shipping pallet. It met DOT requirements and was, until recently, acceptable to the Beatty site. It remains acceptable at some other disposal sites.

A total of 5050 yd^3 of contaminated solid waste was shipped for burial.

Conclusions and Recommendations

1) Uncertainty of commercial disposal site requirements and of state and federal waste transportation requirements greatly affect the cost, schedule, and technology of the waste disposition and should be carefully considered in early planning activities.

2) Radioactive waste generation must be carefully controlled and contained. Labor-intensive decontamination of components and materials is costly and leads to increased volumes of waste being sent to burial. However, the cost for burial is escalating, which changes the economics of decontamination practices. Trade studies are recommended.

9.10 RADIOLOGICAL SAFETY AND SURVEY CERTIFICATIONS

The SRE decommissioning operations were monitored and controlled by the Rockwell International Health and Safety Department. Decontamination and surveying were repeated until the radiological cleanliness criteria had been met. Radiological surveying and analysis were performed according to DOE, NRC, and California guidelines.

A quality assurance (QA) program, independent of Health Physics and the performing engineering department, supported the decommissioning program. In addition to being responsible for current calibration of instrumentation and inspections of waste shipments, QA established procedures for the statistical sampling and analysis of all decontaminated areas.

The Argonne National Laboratory's radiological survey team, assigned to the project by DOE, conducted surveys at various stages of the decommissioning and will provide overview and certification of the decontamination of the facility and site.

The commitment to ALARA principles was incorporated into the overall SRE operational safety plan and into the detailed working procedures. As a result, the total man-rem exposure for this project was controlled to approximately 75 man-rem. There were no individual exposures above the guidelines; most were considerably less than 3 rem/year.

The criteria for acceptable residual contamination levels that were applied are shown in Table 1. These values are slightly more conservative than those of Regulatory Guide 1.86 for surface measurements. In the absence of consensus or regulatory criteria for contamination, such as activation products distributed in materials, the project-specific values shown in Table 1 were developed. An environmental report was prepared that examined the possible effects of the residual levels and found them to be acceptable.

In the area of industrial safety, it is notable that no lost-time injuries occurred during the SRE decommissioning effort.

Conclusions and Recommendations

1) The radiological criteria for decontamination and for contamination and exposure control must be developed and formalized at the project planning stage.
2) Close operational surveillance by health physics personnel is essential to control contamination and exposure, to optimize decontamination work, and to assist in controlling the volume of contaminated waste.
3) Radiological control and certification planning is essential.
4) Quality assurance input is necessary in radiological status evaluation.
5) The participation of an overview and certification agency should be established early and the participation should be integrated into the project schedule.
6) An intensive industrial safety program is required for decommissioning activities.

9.11 FACILITY DESIGN

Decommissioning of any nuclear facility can be accomplished with existing technology. This has been demonstrated in the successful decommissioning of the SRE. It is apparent, however, in examining the technology and procedures employed in the decommissioning, that an in-depth review of the experience gained should be made by facility designers of future nuclear facilities so that problems can be avoided or at least mitigated. The designers can also incorporate facility features that will simplify the decommissioning procedure.

The more significant possibilities are considered and presented below.

9.11.1 Decontamination of Concrete Surfaces

Decontamination of concrete surfaces, particularly surfaces with cracks, expansion joints, and porous concrete, is difficult. Many techniques were tried, ranging from washing with solvents to total removal of the surfaces. None was wholly satisfactory. They all required an extensive use of manpower. Physical removal of the surfaces was usually accomplished with a concrete spalling tool (scabbler) that could remove surfaces to about 1/2 in. depth. Grit blasting was used chiefly for paint removal. Surfaces where permeation of the contamination was very deep (several inches) were removed by a hydraulically powered and positioned ram (Hy-Ram).

The difficulty of decontaminating concrete surfaces in areas where contamination is likely to occur, such as in vaults, trenches, pits, building columns, and walls, indicates a need for protection of surfaces. Metal liners with welded joints, continuous plastic material covers, paint, or hard, smooth coatings should be used to cover the concrete. In addition to permeability, the covering material selected should be sufficiently strong to withstand the potential for damage from maintenance traffic in the area. Metal liners, if used, should provide for ease of removal. Welded joints should be constructed to allow mechanical cutting and effect containment of cutting debris. The use of concrete expansion joints should be avoided unless an effective flexible sealer or cover can be specified for protection over the joints. Surface geometries should be optimized for cleaning. Rounded corners and edges should be used where possible.

9.11.2 Disposal of Massive Concrete Structures

Handling or demolishing of massive contaminated or activated concrete biological shields and support structures can be difficult and costly. These structures are usually only partially radioactive, and the radioactive materials are not easily separated from the nonradioactive materials because of the selective demolition required, the care necessary to keep from cross-contaminating, and the tedious radiological assessment procedures required. At the SRE, the other biological shield was demolished using the Hy-Ram. The upper portion of the shield was not significantly contaminated by induced radioactivity. However, the section near the reactor core line was activated, requiring packaging and shipment of the rubble to burial. Separation of the clean concrete from the activated concrete was not attempted for the concrete from the center section. Although this increased the volume of waste sent to burial, the cost for burial was less than the estimated separation costs.

The use of massive monolithic concrete designs in biological shields or other areas where concrete activation will occur should be reconsidered. Other shielding materials, for example, lead, steel shot, or iron pellets, in conjunction with hydrogenous shield material, might be used. Interlocking, modular concrete structures could be used. Where integrated concrete structures are necessary for containment or strength, means for easily removing or spalling the concrete surfaces should be included in the design. A two-layer biological shield structure could be used. The structure could consist of a physically independent outer layer for containment and an inner layer constructed to provide the major shielding requirements and contain all the induced activation expected during the reactor lifetime.

9.11.3 Disposal of Reactor Vessels

The disposal of the reactor vessel will be a major activity of decommissioning. If a plasma-torch manipulator is to be used to cut up the vessel, then design considerations of the following would simplify the torch operations.

1) Locate piping, grid plates, and other reactor internal structure with optimum access for cutting with the torch.
2) Provide for the installation of the manipulator; include manipulator support structure where possible.
3) Select material for vessels, pipe, and support structure with a concern for induced radiation effects.

9.11.4 Disposal of Waste Holdup Tanks

Excavation and removal of waste holdup tanks was relatively easy. However, leaky valves, tank fittings, and piping contaminated adjacent soil.

Consideration should be given to containing the radioactive waste tanks in isolated vaults with provisions for detecting and containing leaks. Tracks or guide structures should be considered for use on the tank outer walls to permit remote cutting and radiological surveys of the tanks.

9.11.5 Removal of Process Equipment

Process equipment such as heat exchangers, pumps, sodium purification traps, waste handling, and their support structures were removed without major difficulty.

Contamination in this equipment tends to collect in low spots and in cold spots. Design and installation of this equipment should enhance the tendency to concentrate the contamination. Dismantling of process equipment would be simplified if the larger system components were designed for ease of disassembly and for isolation of sections that are likely to be contaminated. Process system equipment installations should include adequate access for removal operations.

9.11.6 Disposal of Handling Machines

The 50- to 60-ton fuel moderator and plug-handling machines were stripped of clean, reusable exterior equipment and then shipped, intact, to the burial site. The cost for labor to dismantle the machines and decontaminate components was too great and was, therefore, not attempted.

Handling machine designs, which would permit easy disassembly and isolation of contaminated sections, would permit a greater salvage of material and a decrease in the waste volume for burial. Handling machines are usually at the center of most reactor operations and, consequently, are vulnerable to airborne contamination. An airtight shroud constructed of easy-to-wipe materials, covering the exterior wiring controls and instrumentation, would simplify decontamination of the exterior.

9.11.7 Decontamination of Facility Structures

Decontamination, rather than removal, of facility structures such as open beams and columns, electrical cable trays, conduits, exhaust ducts, etc., may be desirable to save the facility for future use. However, decontamination costs, in some instances, can exceed replacement costs.

Facility designs should recognize that radioactive spills will occur and that plant interiors, and especially horizontal surfaces, will become contaminated. Easily cleaned containment should be provided, and the routing of services through areas where spills may occur should be avoided.

9.11.8 Contamination Assessment

The radiological assessment of soil and concrete rubble, when working to essentially background levels, is a time-consuming process. Consequently, many cubic yards of soil with marginal contamination may be packaged and shipped to burial.

A well-kept record of radioactive spills and other plant construction and operating history is necessary. Radiological instrumentation-access spaces throughout the plant in concrete structure, shielding material, and soil would facilitate the plant contamination assessment.

9.11.9 General Recommendations

Perform comprehensive radiological surveys, engineering studies, and extensive planning prior to initiating the decommissioning work on site. The accuracy of the schedule and cost estimates are absolutely dependent on complete understanding and definition of the full scope of work. This scope can only be adequately defined after the completion of these activities.

The processes described can be anticipated before the decommissioning of many nuclear facilities. Solutions to problems may be made more difficult by the increasingly restrictive regulatory requirements, such as the contemplated decrease in the permissible radiation dosage for workers; the more restrictive requirements for packaging, transport, and burial of waste; and the growing emphasis on greater usage of waste volume reduction techniques.

Considerations for simplifying decommissioning not directly related to SRE experiences are presented as follows:

1) A decommissioning plan should be prepared concurrently with the conceptual design of the facility and should be revised in parallel with the development of the facility design.
2) The design of maintenance equipment and facilities should include capabilities for use in the decommissioning operations. Waste volume reduction capability such as arc cutters, shears, compactors, electropolishers, and liquid waste solidification should be provided.

3) Plant air exhaust systems should have the capability to handle airborne contamination generated during decommissioning. The plant should also have capability for containment of the liquid and solid generated during decommissioning.

4) Adequate space should be provided near the facility for decontamination, dismantling, packaging, storage. Also, adequate personnel and equipment access to the potentially contaminated areas is required.

These recommendations assume that decommissioning problems, as experienced today, will be similar to those 40 years from now when the facilities currently being designed will be ready for decommissioning. The projected physical size of plants is much larger now; the reactor designs, particularly the reactor internals, are much more complex; regulatory requirements for radiation worker radiation dosages are becoming more stringent; onsite burial of waste is being considered; and the possibility of a site being satisfactorily decontaminated to levels for unrestricted use is decreasing. The basic requirements for reducing contamination are not expected to change, but more of the decontamination work may be transferred to remotely operated machines.

10.0 LIST OF ABREVIATIONS

SRE — Sodium Reactor Experiment
PEP — Power Expansion Program
IHX — Intermediate Heat Exchanger
EM — Electromagnetic
FHM — Fuel Handling Machine
RMDF — Radioactive Materials Disposal Facility
CERF — Contaminated Equipment Repairs Facility
SFMPO — Surplus Facilities Management Program Office
DOE-RL — U.S. Department of Energy, Richland, Washington Area Office
DOE-SAN — U.S. Department of Energy, San Francisco, California Area Office
ESG — Energy Systems Group of Rockwell International Corporation
PCS — Performance Control System
PFT — Primary Fill Tank
MIHX — Main Intermediate Heat Exchanger
ORNL — Oak Ridge National Laboratory
JRC — Jet Research Corporation
DOT — U.S. Department of Transportation
UBC — Uniform Building Code
HEPA — High Efficiency Particulate Air
ALARA — As Low As Reasonably Achievable
DOE — U.S. Department of Energy
AEC — Atomic Energy Commission

APPENDIX

SUPPORTING DOCUMENTS

Document No.	Title
N704ACR990003	SRE Activity Requirement No. 005. Removal of Primary Sodium System Piping and Components
N704ACR990004	SRE Activity Requirement No. 009. Passivation of Residual Sodium in the Reactor Vessel
N704ACR990005	SRE Activity Requirement No. 008. Removal of Sodium Components from Sodium Service Building 153
N704ACR990006	SRE Activity Requirement No. 16. Decontamination and Dismantling of Demountable Maintenance Shield Assembly
N704ACR990007	SRE Decontamination and Removal of Storage and Wash Cells. SRE Activity Requirement No. 17
N704ACR990008	SRE Activity Requirement No. 10. Removal of SRE Reactor Vessel Loose Internals
N704ACR990010	SRE Activity Requirement No. 7. Removal of Primary Sodium Components from the Sodium Service Vault
N704ACR990011	SRE Activity Requirement No. 11. Cleaning of Sodium Components
N704ACR990012	SRE Activity Requirement Nos. 18 and 19. Decontamination and Dismantling of Mark I and Mark II Fuel Handling Machines
N704ACR990014	SRE Activity Requirement No. 10A - Removal and Disposal of the SRE Ring Shield
N704ACR990015	SRE Activity Requirement No. 20. Decontamination of Moderator Handling Machine
N704ACR990016	SRE Activity Requirement No. 16. Decontamination and Dismantling of the Hot Cell Facilities
N704ACR990017	SRE Activity Requirement No. 23 - Disposal of Radioactive Waste Systems Exterior to Building 143
N704ACR990020	SRE Activity Requirement No. 24. Decontamination of Building 163 (CERF)
N704ACR990021	SRE Activity Requirement No. 25. Decontamination & Dismantling of Building 724 and Pad 723

Document No.	Title
N704AACR990002	SRE Activity Requirement No. 14. Decontamination of Primary Fill Tank, Pipe & Sodium Service Vaults
N704ACR990023	SRE Activity Requirement No. 21. Removal of Activated and Contaminated Concrete Structure from the SRE
N704ACR990024	SRE Activity Requirement No. 27. D&D of Building 143 Retention Pond and Sanitary Sewer
N704ACR990025	Disposal of Building 143 (SRE) Stack and Vent System Activity Requirement
N704ACR990027	SRE Activity Requirement No. 29. Final Closeout of the SRE Facility
N704DP990001	Start-Up Safety Review for Plasma Torch System
DRR-704-990-001	Design Review Minutes for the ORNL Rotating Mast Manipulator Final Design
DWP-704-990-003	Procedure for Transferring Radioactive Sodium from Primary Drain Tank into Approved Cabinets
N704DWP990004	Procedures for Installing SRE Manipulator in SRE Vessel Mock-Up
N704DWP990008	Detailed Working Procedure for Removal of SRE Sodium System Components
N704DWP990009	Passivation of Remaining Sodium in SRE Core Vessel. Building 143 Detailed Working Procedure
N704DWP990010	Passivation of Remaining Sodium in SRE Primary Sodium Tank Detailed Working Procedure
N704DWP990011	Decontamination and Dismantling of SRE Demountable Maintenance Shield Assembly (DMSA) Detailed Working Procedure
N704DWP990012	Detailed Working Procedure for the Removal of SRE Reactor Vessel Loose Internals
N704DWP990013	DWP for Loading & Shipping SRE Declad Fuel
N704DWP990014	Removal of Primary Sodium Components from the Sodium Service Vaults
N704DWP990015	Detailed Working Procedure for the Removal of Sodium Systems & Components in Building 153

Document No.	Title
N704DWP990016	Cutting and Disposition of Primary Sodium Piping
N704DWP990017	SRE Core Liner Removal - Detailed Working Procedure
N704DWP990018	Core Clamp Band Removal - Detailed Working Procedure
N704DWP990019	Detailed Working Procedure for Explosive Removal of SRE Vessel Internal Piping
N704DWP990021	Detailed Working Procedure for the Removal of SRE Storage and Wash Cells Internals
N704DWP990025	Detailed Work Procedure for the Removal of the SRE Gridplate
N704DWP990026	Core Tank Removal Detailed Working Procedure
N704DWP990028	Detailed Working Procedure for the Passivation of Remaining Sodium in the SRE Components
N704DWP990029	Manipulator and Platform Installation and Alignment
N704DWP990031	Core Tank Bellows Removal
N704DWP990032	Main & Aux. Core Clamp Removal from the SRE Core Tank
N704DWP990033	SRE Core Tank Bottom Drilling
N704DWP990034	SRE Gridplate Nut Removal
N704DWP990035	Detailed Working Procedure for the Removal of the SRE Moderator Coolant Header
N704DWP990036	Guide Post Installation and Removal
N704DWP990037	Detailed Work Procedure for Removal, Transfer, and Reassembly of SRE Rotating Mast Manipulator and Platform
N704DWP990038	Detailed Working Procedure to Transfer SRE Internal Piping to Storage Pit
N704DWP990039	Detailed Working Procedure for the Cut-Up of the SRE Internal Piping
N704DWP990040	Detailed Working Procedure for Core Tank-To-Liner Attachment Ring Removal
N704DWP990044	Detailed Working Procedure for Removal of SRE Gridplate Perimeter Nuts

Document No.	Title
N704DWP990045	Detailed Working Procedure for Outer Tank Bellows Removal
N704DWP990046	Detailed Working Procedure for SRE Thermal Rings Removal and Cut Up
N704DWP990047	Detailed Working Procedure for Removal of Cutting Debris
N704DWP990048	Detailed Work Procedure for the Removal of the SRE Core Tank Bottom
N704DWP990049	Detailed Working Procedure for Outer Tank Removal
N704DWP990050	Detailed Working Procedure for Outer Tank Bottom Removal
N704DWP990051	Detailed Working Procedure for Super X Insulation Removal
N704DWP990053	Detailed Working Procedure for Scoring the Core Cavity Liner
N704DWP990054	Disposal of Radioactive Waste Systems at Building 653 and Building 143
N704DWP990055	Detailed Work Procedure for Operation of the SRE Shielding Water Filtration System
N704DWP990057	Decontamination and Dismantling of Building 724 Detailed Working Procedure
N704DWP990058	Removal and Shipment of Two Fuel and One Moderator Handl-ing Machines
N704DWP990060	Deactivation of Building 143 Retention Pond and Sanitary Sewer System
N704DWP990062	Decontamination and Dismantling of the Hot Cell Facilities at SRE
N704DWP990063	SRE Site Survey Plan for Release to Unrestricted Use
N704DWP990065	Building 143 Detailed Working Procedure for Excavation of the SRE to Remove Below-Grade Contaminated and Activated Material and Structures
N704DWP990070	Final Radiological Inspection Detailed Working Procedure
N704DWP990071	Backfill and Compaction of the SRE High Bay Excavation, Detailed Working Procedure

Document No.	Title
TI-704-990-011	Remote Tooling Task Requirement 1.9. Core Tank Liner Removal Tooling
TI-704-990-012	Remote Tooling Task Requirement No. 1.10. Core Tank Removal Tooling
TI-704-990-013	Remote Tooling Task Requirement No. 1.11. In-Vessel Thermal Rings Removal Tooling
TI-704-990-014	Remote Tooling Task Requirement No. 1.12. Outer Tank Removal Tooling
TI-704-990-015	Remote Tooling Task Requirement No. 1.13. In-Vessel Thermal Insulation Removal Tooling
TI-704-990-016	Remote Tooling Task Requirement No. 1.14. Core Cavity Liner Removal
TI-704-990-017	Plan of Action for Technical Assistance Provided by ORNL in Support of Remote Tooling for Removal of SRE Vessels
TI-704-990-020	Soil and Concrete Activation Limits for Unrestricted Use of Former SRE Site
TI-704-990-021	Disposal of Primary Sodiuum
TI-704-990-022	Support Calculations for Rotating Mast Manipulator
TI-704-990-023	Passivation of Remaining Sodium in SRE Core Vessel
N704TI990025	Contamination Confinement Systems for Decontamination and Disposition of the SRE Core Vessel
N704TI990026	Technical Information on Decontamination of Radioactive Materials by Foam Application
N704TI990027	Radiological Survey Results - Release to Unrestricted Use, SRE Region I (Building 724 Area)
N704TI990028	Radiological Survey Results - Release to Unrestricted Use, SRE Region II (Building 163, Box Shop)
N704TI990029	Radiological Survey Results - Release to Unrestricted Use, SRE Region III (SRE Entrance)
N704TI990030	Radiological Survey Results - Release to Unrestricted Use, SRE Region IV (West Parking Lot)
N704TI990031	Radiological Survey Results - Release to Unrestricted Use, SRE Region V (Gas Storage Vault)

Document No.	Title
N704TI90032	Radiological Survey Results - Release to Unrestricted Use, SRE Region VI (Water Tank Area)
N704TI990033	Radiological Survey Results - Release to Unrestricted Use, SRE Region VII (Retention Pond)
N704TI990034	Radiological Survey Results - Release to Unrestricted Use, SRE Region VIII (SRE Front Lot)
N704TI990035	Radiological Survey Results - Release to Unrestricted Use, SRE Region IX (SRE Back Lot)
N704TI990036	Radiological Survey Results - Release to Unrestricted Use, SRE Region X (SRE Parking Lot)
N704TI990037	Radiological Survey Results - Release to Unrestricted Use, SRE Building 041
N704TI990038	Radiological Survey Results - Release to Unrestricted Use, SRE Building 143
N704TI990039	Radiological Survey Results - Release to Unrestricted Use, SRE, Building 163
N704TI990045	Schedule to Complete Excavation of SRE in FY 1979 First Year Schedule of 2-Year Plan for Excavation of SRE in FY 1979
N704TI990047	Disassembly and Examination of SRE Sodium Heated Steam Generator
N704TI990050	References for Decontamination and Dispositioning Criteria
N704TI990051	Decontamination of Surfaces by Foam Cleaning
N704TI990057	Final Radiological Inspection of the Below-Grade Areas in the SRE Prior to Release for Unrestricted Use
N704TNP990001	Training Plan for the Decontamination and Disposition of Facilities
N704TP990005	SRE Mockup Operations Test Plan
N704TP990006	Test Plan for the Explosive Cutting Demonstration of SRE Vessel Internal Piping
N704TP990007	Manipulator Checkout and Test Procedure

Document No.	Title
N704DWP990087	SRE Radioactive Vent System Removal Detailed Working Procedure
N704ER990001	Evaluation of Cutting Methods for Removal of the SRE 5-1/2 Inch Carbon Steel Thermal Rings
N704ER990003	Radiation Levels Associated with Irradiated Metallic Components of the SRE
N704ER990005	Dose Rates Outside the ATCOR AL-33-90 Cask during the Shipment of SRE Stainless Steel Components
N704ER990008	SRE Biological Shield Removal Methods Trade Study
N704ER990009	Engineering Report for the Core Drilling and Radiometric Analysis of the SRE Reactor Biological Shield
FDP704990003	Facilities Dismantling Plan for SRE
N704DP990001	High Level Radioactive Waste Transfer
N704DP990004	Sodium Disposal Facilities Operating Procedure
N704DP990005	SRE Primary Piping Sodium Disposal
PP-704-990-001	Quality Assurance Program Plan for the Decontamination & Disposition of Facilities
PP-704-990-002	Decontamination & Disposition of Facilities Program
RPA-704-990-001	Release Plan of Action for the Decontamination & Disposition of Facilities Program
SRR-704-990-001	Operational Safety Plan for the AI Decontamination & Disposition of Facilities Program
SRR-704-990-005	Minutes of Isotopes Committee Review of the SRE Dismantling Plan
N704SRR990007	Minutes of Isotopes Committee Review of Decontamination and Disposition Program Detailed Working Procedures for STIR and SRE facilities.
N704SRR990008	Minutes of Isotopes Committee's Review of Application to Passivate Remaining Sodium in SRE Vessel Building 143 Detailed Working Procedure and an Operating Procedure for Calibration of Radiation Survey Meters in Building 100, Santa Susana

Document No.	Title
N704SRR990009	Minutes of the Isotopes Committee Review of the Detailed Working Procedure for the Proposed Sodium Passivation in the SRE Primary Tank
N704SRR990011	Minutes of Isotope Committee Review of the Detailed Working Procedures (DWP's) for Imminent D&D Operation
N704SRR990012	Minutes of Isotope Committee Review of the Detailed Working Procedure for Removal of SRE Reactor Vessel Loose Internals
N704SRR990014	Minutes of Isotope Committee Review of the Proposed Plasma-Torch and Explosive Cutting Operations at SRE
N704SRR990019	Summary and Comparison of ESG and ANL Radiometric Analyses at the SRE
TI-599-19-103	Post Retirement Plan for Radiological Decontamination of the SRE Site
TI-704-990-001	Activity Requirement No. 1. Remote Tooling for Removal of SRE Vessel
TI-704-990-002	Sodium Disposal Processes
TI-704-990-003	Remote Tooling Task Requirement No. 1.1. Initial In-Vessel Radiation Survey
TI-704-990-004	Remote Tooling Task Requirement No. 1.2. In-Vessel Viewing & Photo Survey
TI-704-990-005	Remote Tooling Task Requirement No. 1.3. SRE Core Cavity Liner Mock-Up
TI-704-990-006	Remote Tooling Task Requirement No. 1.4. Plasma-Torch Manipulator
TI-704-990-007	Remote Tooling Task Requirement No. 1.5. Plasma-Arc Cutting System
TI-704-990-008	Remote Tooling Task Requirement No. 1.6. Manipulator-Torch Control Console
TI-704-990-009	Remote Tooling Task Requirement No. 1.7. Underwater Cut-Up Tank
TI-704-990-010	Remote Tooling Task Requirement 1.8. Vessel Internals Removal Tooling

Document No.	Title
N704TP990008	Radiological Survey Plan, Support of D&D Program Operations at T-143 (SRE)
N704TR990003	Manipulator Console Systems Checkout and Calibration Test Report - Development
N704TR990004	SRE Vessel Internal Piping Explosive Cutting Development
N704TR990005	SRE Underwater Plasma Arc Cutting Development Test Report
N704TR990006	Report on Passivation of the SRE Hot Trap
N704TR990007	Report on Passivation of the SRE Reactor Vessel and Associated Components
N704TR990008	SRE Reactor Vessel Cutup and Disposal

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