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SODIUM REACTOR EXPERIMENT DECOMMISSIONING FINAL REPORT

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Rockwell International

Environmental & Energy Systems Division
Energy Systems Group

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REMEDIAL ACTION AND
DECOMMISSIONING PROGRAMS

SODIUM REACTOR EXPERIMENT DECOMMISSIONING FINAL REPORT

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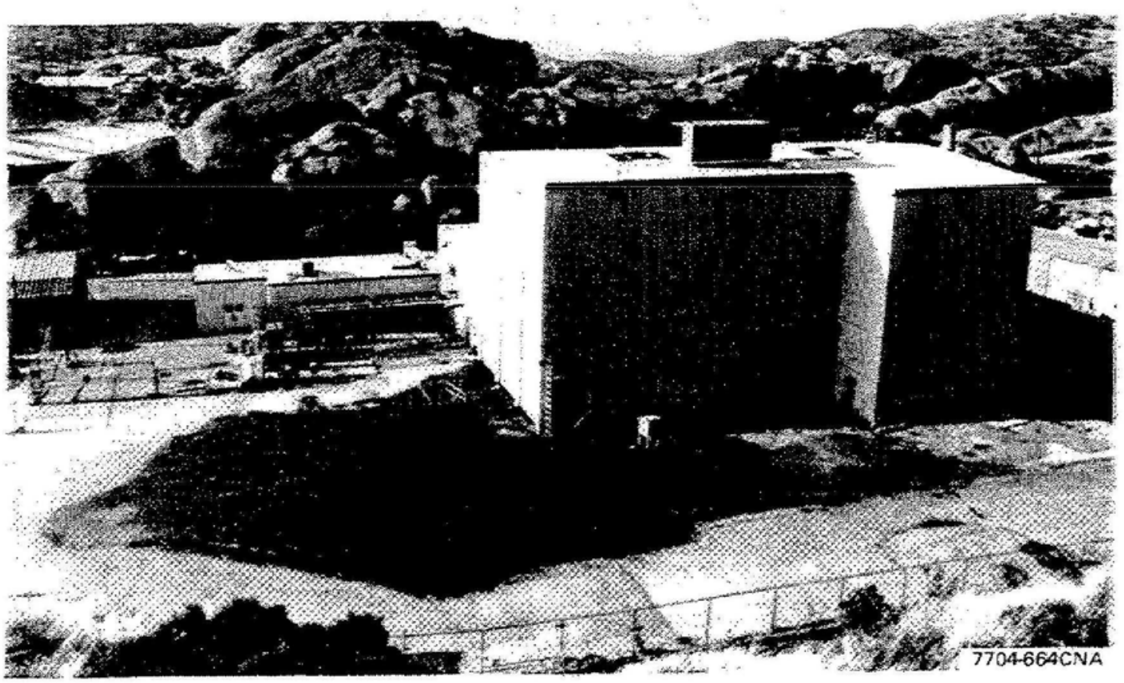
SRE During Operation



SRE After Decommissioning

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.

CONTENTS

		Page
1.0	Background	1
1.1	Facility History	1
1.2	Decommissioning Project Purpose	6
2.0	Facility Description	7
2.1	Buildings and Systems	7
2.1.1	Containment Building Layout	7
2.1.2	Containment	7
2.1.3	Reactor Structure	13
2.1.4	Primary Coolant System	15
2.1.5	Secondary Systems	16
2.1.6	Power Conversion Systems	18
2.1.7	Reactor Auxiliary Systems	20
2.2	Pre-decommissioning Status	27
2.2.1	Fuel Assemblies	28
2.2.2	Core Components	28
2.2.3	Sodium Systems	28
2.2.4	Radioactive Liquid Waste System	28
2.2.5	Radioactive Gaseous Waste System	33
2.2.6	Component-Handling Machines	33
2.2.7	Fuel, Moderator, and Pump Storage Cells	33
2.2.8	Hot Cells and Ventilation System	35
2.2.9	Peripheral Areas	35
2.2.10	Reactor Cavity	37
3.0	Decommissioning Objective and Work Scope	41
3.1	Objective	41
3.2	Work Scope	41
3.2.1	Planning	41
3.2.2	Development and Test	41
3.2.3	Dismantlement Operations	44

CONTENTS

	Page
3.2.4 Radiation Control	45
3.2.5 Waste Management	46
3.2.6 Project Management and Support	47
3.2.7 Quality Assurance	47
4.0 Work Performed	49
4.1 Program and Project Management	49
4.2 Project Engineering	49
4.3 Site Preparation	51
4.4 Decommissioning Operations	52
4.4.1 Noncontaminated Peripheral Systems Removal	52
4.4.2 Primary Sodium Disposal	53
4.4.3 Residual Sodium Passivation	53
4.4.4 Radioactive Sodium System Component Removal	60
4.4.5 Disposal of Radioactive Waste Handling and Storage System	65
4.4.6 Removal of Inert Gas Systems	66
4.4.7 Removal of the Kerosene Cooling System	68
4.4.8 Disposal of Fuel- and Moderator-Handling Machine	68
4.4.9 Removal and Disposal of the Reactor Vessel and Internals	69
4.4.10 Excavation of Contaminated Soil	115
4.4.11 Biological Shield Demolition	127
4.4.12 Below Grade Component Removal	134
4.4.13 Decontamination and Dismantling of the Hot Cell Area	150
4.4.14 Excavation Backfilling	156
4.4.15 Decontamination of High Bay Walls, Ceiling, and Cranes	160
4.4.16 Disposal of Radioactive Exhaust Systems	162
4.4.17 Decontamination of Buildings 163 and 041	163
4.4.18 Decontamination Equipment	167

CONTENTS

		Page
4.5	Waste Disposal	170
	4.5.1 Radioactive Waste Packaging and Handling	170
	4.5.2 Radioactive Waste Transport and Burial	171
4.6	Post-Decommissioning Radiological Survey	171
5.0	Schedule and Cost	177
5.1	Schedule	177
5.2	Cost	178
6.0	Waste Volumes Generated	185
7.0	Occupational Exposure to Personnel	187
8.0	Final Facility Condition	191
9.0	Lessons Learned	199
9.1	Planning and Engineering	199
	9.1.1 Disposition of SRE Structures and Systems	200
9.2	Disposal of Noncontaminated SRE Systems and Support Facilities	200
9.3	Disposal of Contaminated Sodium	201
9.4	Disposal of Contaminated SRE Peripheral Systems and Facilities	202
9.5	Disposal of Support Systems	204
9.6	Disposal of Reactor Vessels	207
9.7	Excavations	209
9.8	Decontamination of Structures	211
9.9	Waste Handling	212
9.10	Radiological Safety and Survey Certifications	213
9.11	Facility Design	215
	9.11.1 Decontamination of Concrete Surfaces	215
	9.11.2 Disposal of Massive Concrete Structures	216
	9.11.3 Disposal of Reactor Vessels	217
	9.11.4 Disposal of Waste Holdup Tanks	217
	9.11.5 Removal of Process Equipment	217

CONTENTS

	Page
9.11.6 Disposal of Handling Machines	218
9.11.7 Decontamination of Facility Structures	218
9.11.8 Contamination Assessment	218
9.11.9 General Recommendations	219
10.0 List of Abbreviations	221
Appendix	223
Supporting Documents	223

TABLES

1.	SRE Hot Leg Operational Time and Temperature	2
2.	SRE Operating Statistics	4
3.	Upper Contamination Limits for Decontamination and Disposition at SRE	6
4.	Steam Generator Design Operation Conditions	18
5.	Contamination Levels of Fuel Handling Machines (1966)	34
6.	Contamination Levels in SRE Building 143 (1966)	36
7.	Activity Levels of SRE Ventilation System (1966)	37
8.	Contamination Levels of Building 163 (1966)	38
9.	Contamination Levels of SRE Oil Cleaning Facility, Building 724 (1966)	39
10.	SRE Dismantling and Dispositioning Activity Requirements	43
11.	Location of Core Components in the Reactor	73
12.	SRE Biological Shield Radiometric Analysis	133

FIGURES

		Page
1.	Total Time at or above a Hot Leg Temperature	3
2.	Plan View of SRE Site	8
3.	Building Layout	9
4.	Artist's Cutaway of SRE	11
5.	Vertical Cross Section of Reactor	12
6.	Main Intermediate Heat Exchanger	17
7.	Steam Plant Facilities	19
8.	Radioactive Liquid Waste and Vent System	24
9.	SRE Hot Cell Facility – Plan View	26
10.	Radiation Levels from Tank and Associated Piping	29
11.	Radiation Map of the Main Sodium System	30
12.	Radiation Map of the Auxiliary Sodium System	31
13.	Radiation Map of the Sodium Services Piping	32
14.	Document Flow Chart	50
15.	Sodium Vacuuming System	56
16.	Reactor Passivation Piping During Installation	57
17.	Reactor Passivationj Piping in Primary Piping Vault	58
18.	Small Component Sodium Passivation System	59
19.	Large Component Sodium Passivation System	60
20.	Maint Primary Loop Radiation Levels	62
21.	Auxiliary Primary Loop Radiation Levels	63
22.	Sodium Service Vault Radiation Levels	64
23.	Installation of the Six Radioactive Waste Decay Tanks	67
24.	SRE Reactor Vessel Mockup Facility	72
25.	SRE Reactor Loading Face	75
26.	140-in.-diameter Plug Removel	76
27.	Reflector Shield Segments at RMDF	77
28.	140-in.-diameter Plug Packaged for Disposal	78
29.	SRE Reactor Interior	80
30.	Bellows and Vent Pipe Cuts	81

FIGURES

Page

31. Bellows Cutting Fixture.................................... 82
32. Main and Auxiliary Piping Cuts.................................... 83
33. JRC Manipulator for Remote Installation of Explosive Charge....... 86
34. Plasma-Arc Cutting Torch Radial Arm.................................... 90
35. Pipe Cutting Using Plasma Torch.................................... 92
36. Core Tank Liner Cuts.................................... 95
37. Attachment Ring Cuts.................................... 96
38. Core Tank Liner Mockup Cutting Test Results.................................... 97
39. Mockup of SRE Reactor Grid Plate.................................... 99
40. Core Tank Cuts.................................... 101
41. Core Tank Bottom Cuts.................................... 104
42. SRE Reactor Core Tank Wall Removal.................................... 105
43. Outer Tank Bellows Cuts.................................... 107
44. Thermal Ring Being Cut.................................... 111
45. Outer Tank Cuts.................................... 113
46. Initial Excavation at Northeast Corner of the SRE High Bay........ 116
47. Excavation East of Primary Fill Tank Vault.................................... 117
48. Access Ramp to High-Bay Excavation and Suspected Area of Contamination.................................... 118
49. Excavation East of SRE High Bay.................................... 119
50. Hy-Ram in Operation.................................... 121
51. Demolition of Primary Fill Tank Vault.................................... 122
52. Location of High-Bay Retaining Wall and Column Identification..... 124
53. Installation of Wooden Shoring in High-Bay Retaining Wall......... 125
54. Column Supports and Seismic Bracing Installed under North Wall of SRE High Bay.................................... 128
55. SRE High-Bay North Wall Column Supports Interior View............. 129
56. Construction of Form for Replacement Foundation under High-Bay Column 11D.................................... 130
57. Biological Shield Core Sample Locations.................................... 131

FIGURES

		Page
58.	SRE Core Cavity Liner after Demolition of Biological Shield	135
59.	Biological Shield Cavity	136
60.	Hy-Ram Demolishing Concrete around Fuel Storage Tubes	137
61.	Fuel Storage Tube Removal	139
62.	Fuel Storage Cell Concrete Demolition	140
63.	Moderator Element Storage Cells	141
64.	Wash Cell Drain Line Removal	143
65.	SRE Hot Cell Area	144
66.	Rock Drill in Operation	146
67.	Large Backhoe in SRE High-Bay Excavation	147
68.	Fuel Element Enclosure Sleeve Removal	148
69.	Building 163 Size-Reduction Booth	149
70.	Dip Leg Excavation	151
71.	Cutting Steel Plates from Interior of SRE Primary Hot Cell	153
72.	Exposed Primary Hot Cell Exhaust Line	154
73.	High-Bay Excavation Retaining Wall with Lower Wooden Shoring Removed	158
74.	SRE High-Bay Excavation Backfill and Compaction in Progress	159
75.	High-Bay Scaffolding	161
76.	Dismantling East Roof Exhaust	164
77.	SRE High-Bay Roof Decontamination	165
78.	Hy-Ram Used at SRE	167
79.	Floor Scabblers Used at SRE	168
80.	Ring Shield on Transport Vehicle	172
81.	Radiological Survey Regions	173
82.	SRE Decommissioning Schedule	179
83.	SRE Decommissioning Schedule	180
84.	SRE Decommissioning Schedule	181

FIGURES

		Page
85.	Cumulative Group Dose for All Personnel in Decommissioning the SRE	189
86.	SRE at Initiation of Decommissioning	192
87.	SRE at Completion of Decommissioning	193
88.	Plan View of SRE Site	194
89.	Storage Tubes Imbedded in Bedrock	206

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.

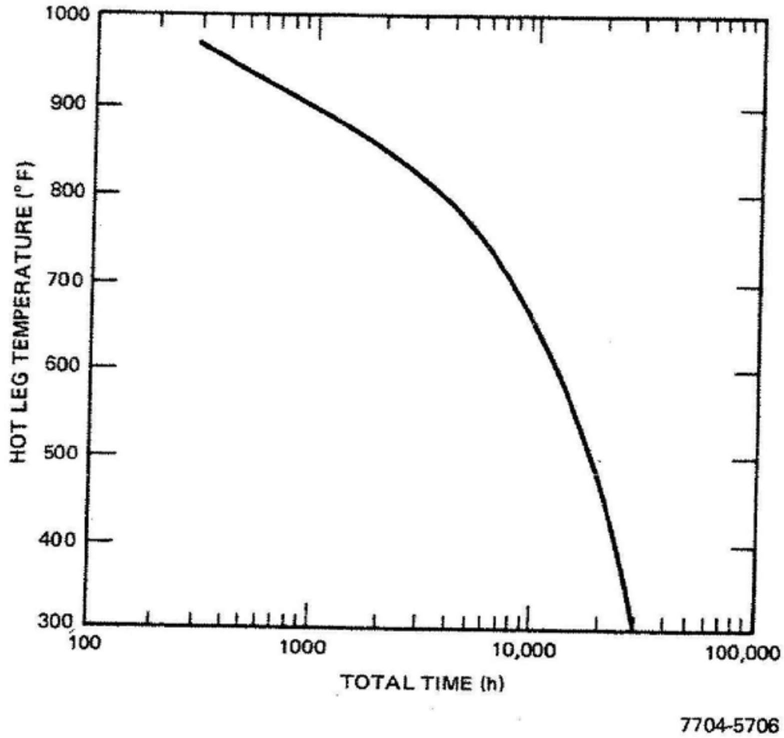


Figure 1. Total Time at or above a Hot Leg Temperature

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

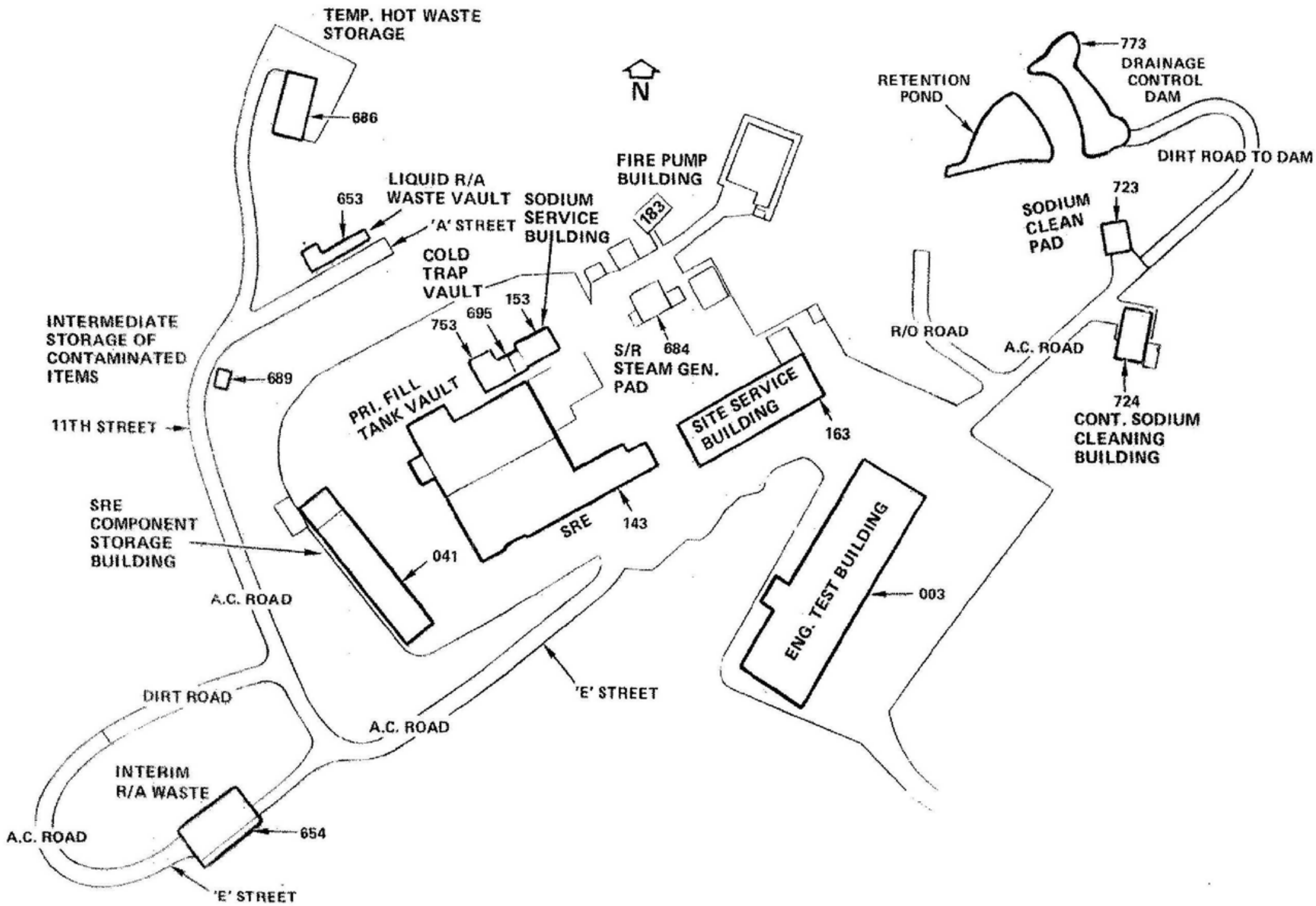

Figure 2. Plan View of SRE Site

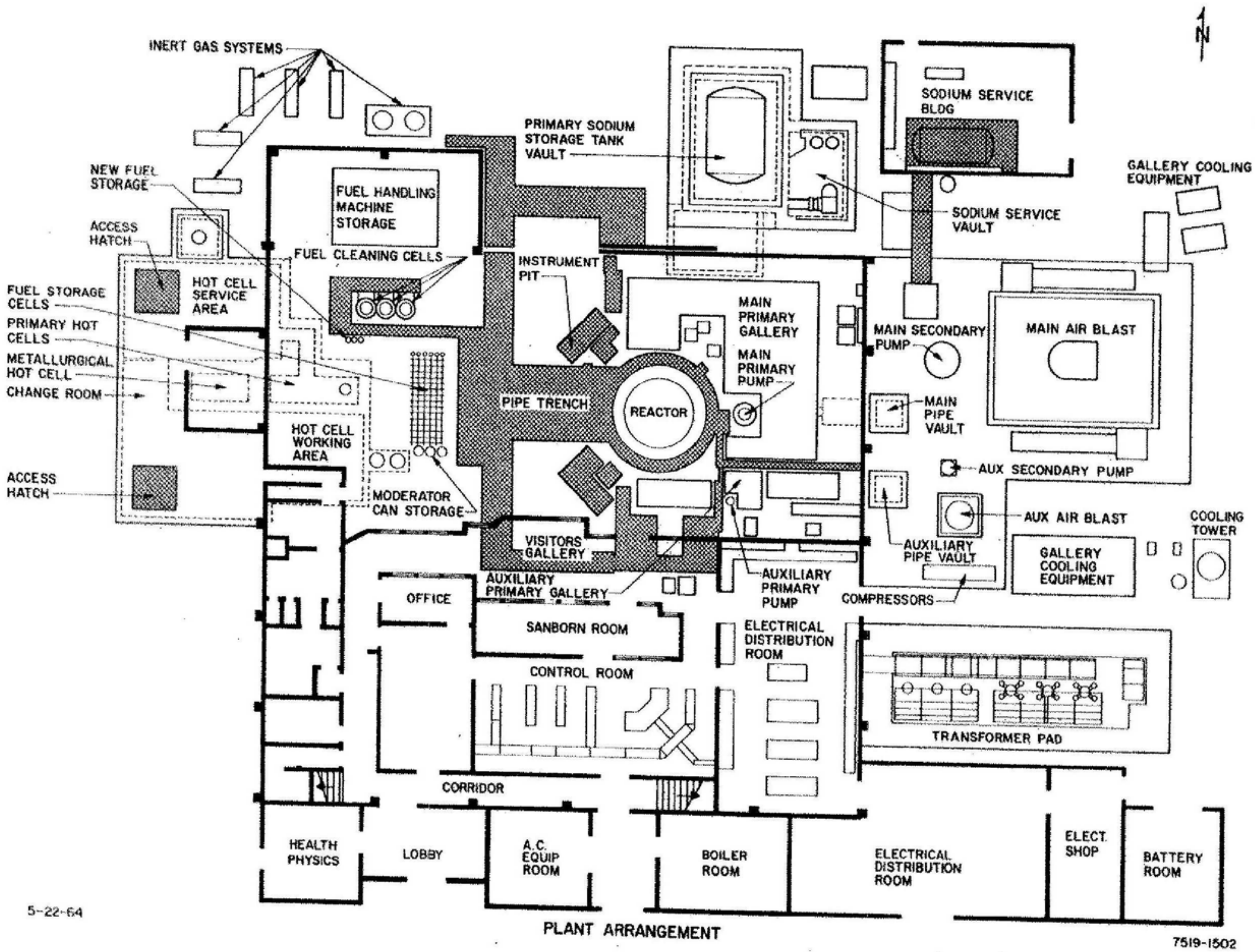

Figure 3. Building Layout

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

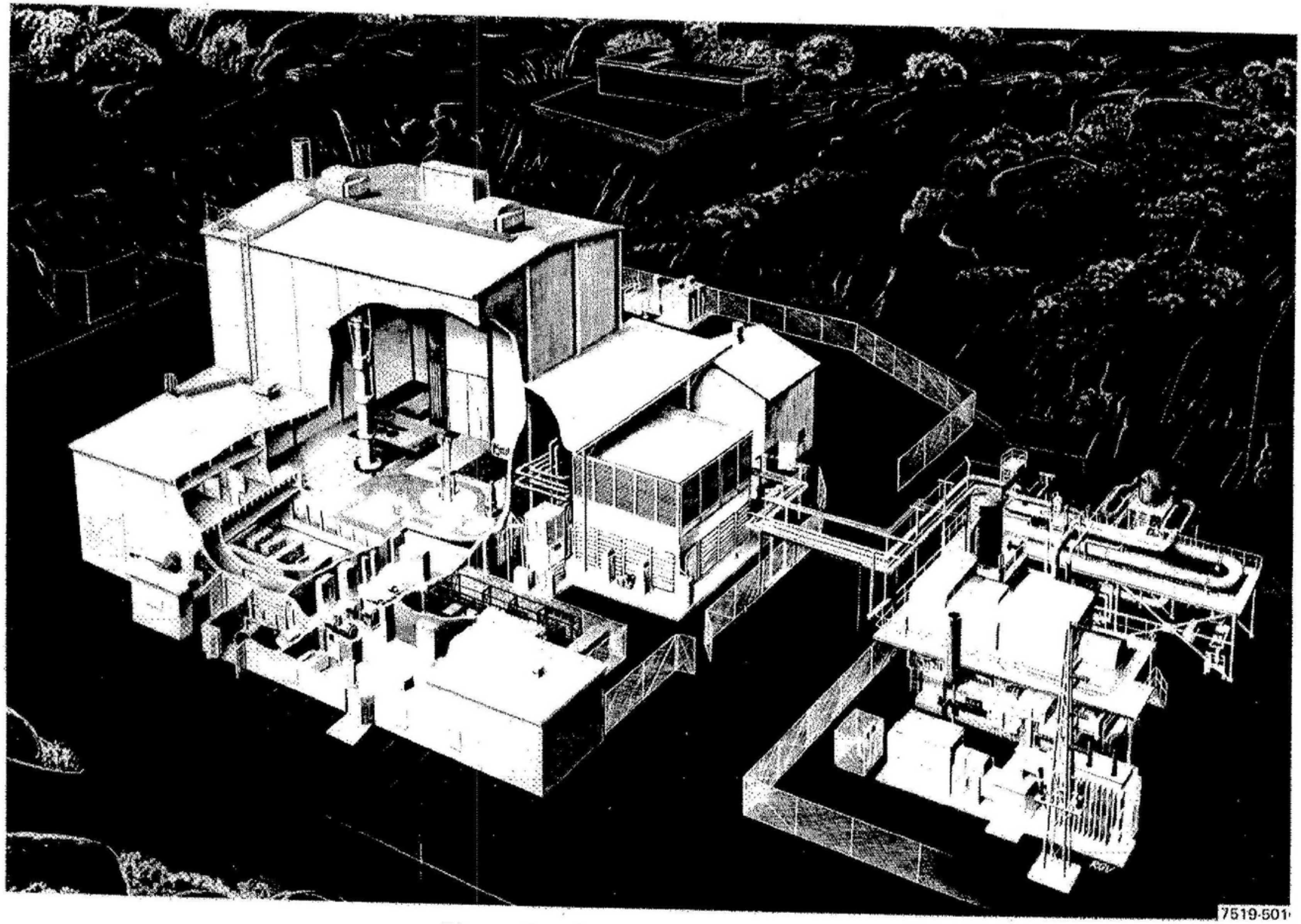

Figure 4. Artist's Cutaway of SRE

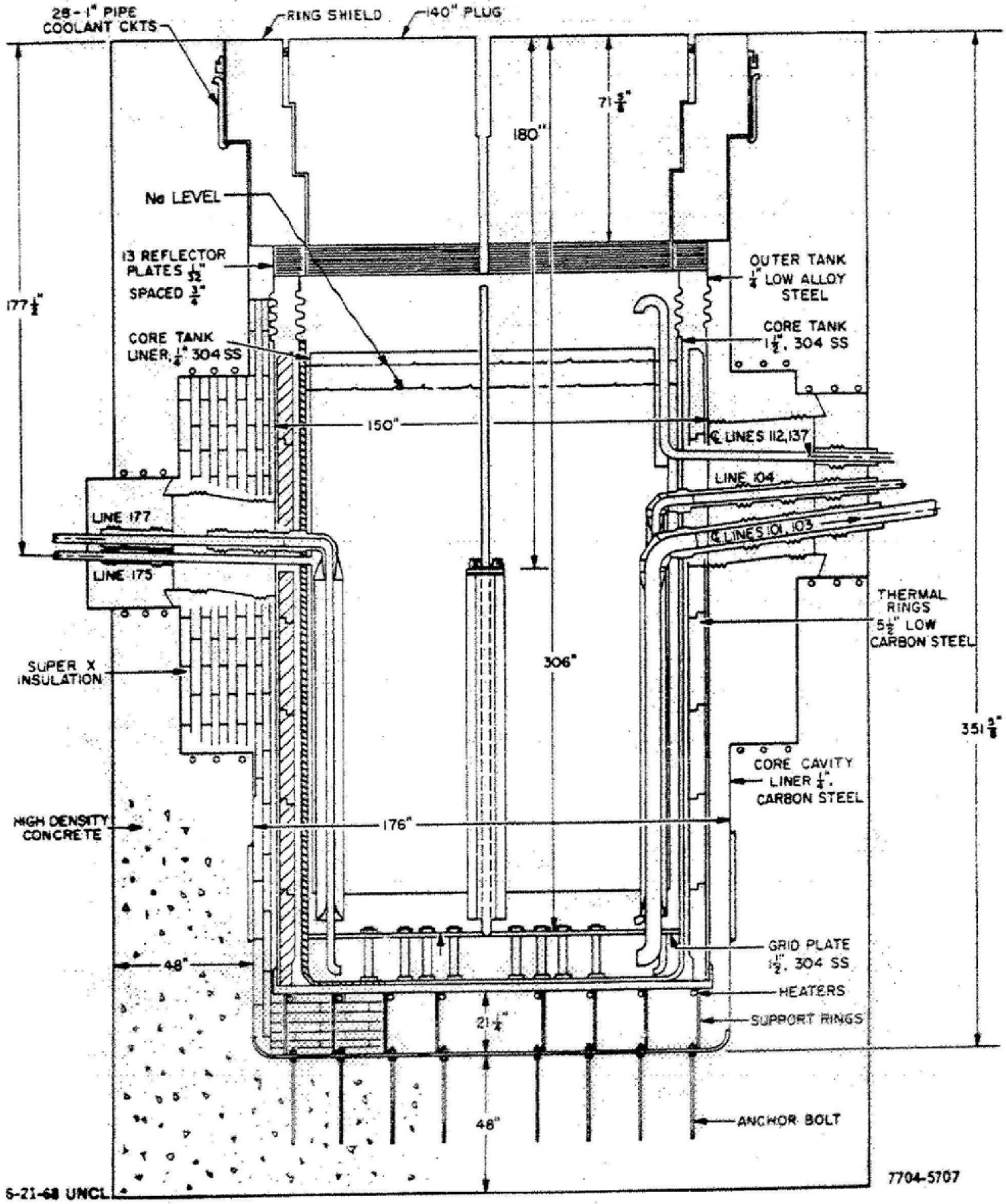

Figure 5. Vertical Cross Section of Reactor

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

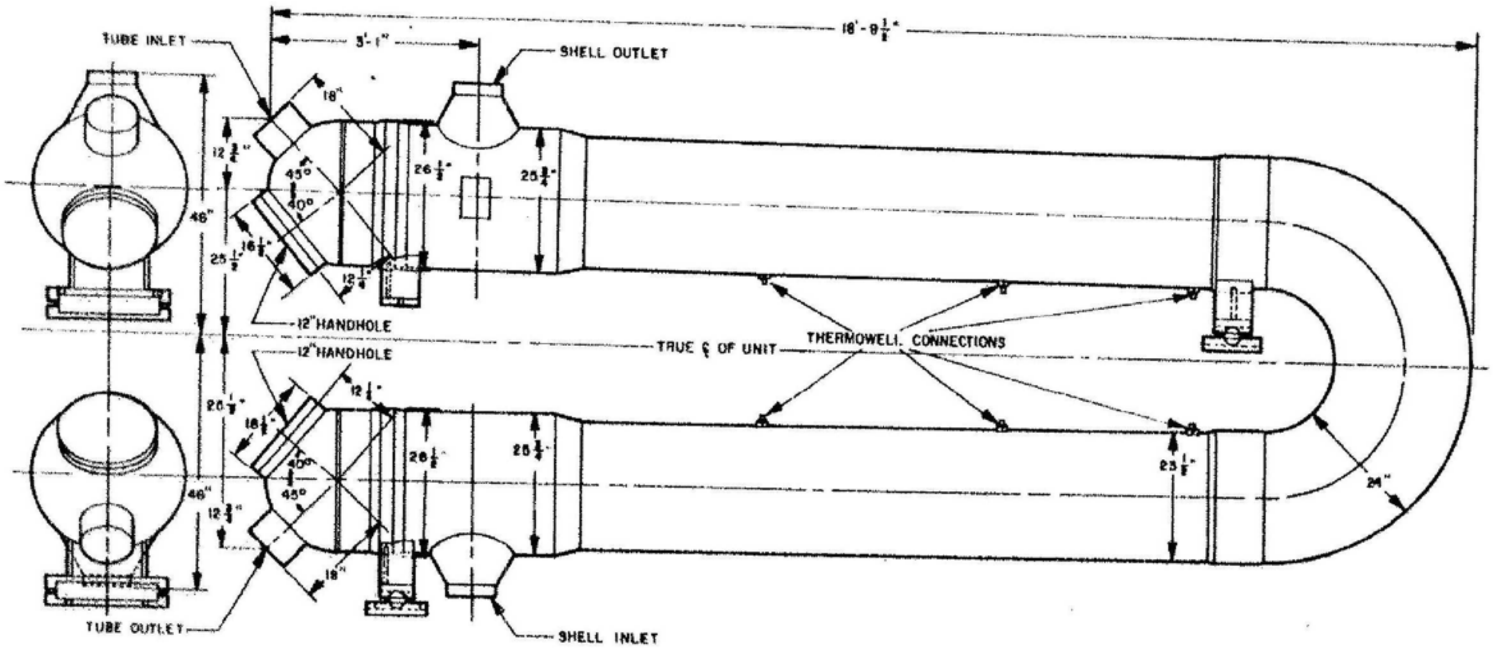

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Figure 6. Main Intermediate Heat Exchanger

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

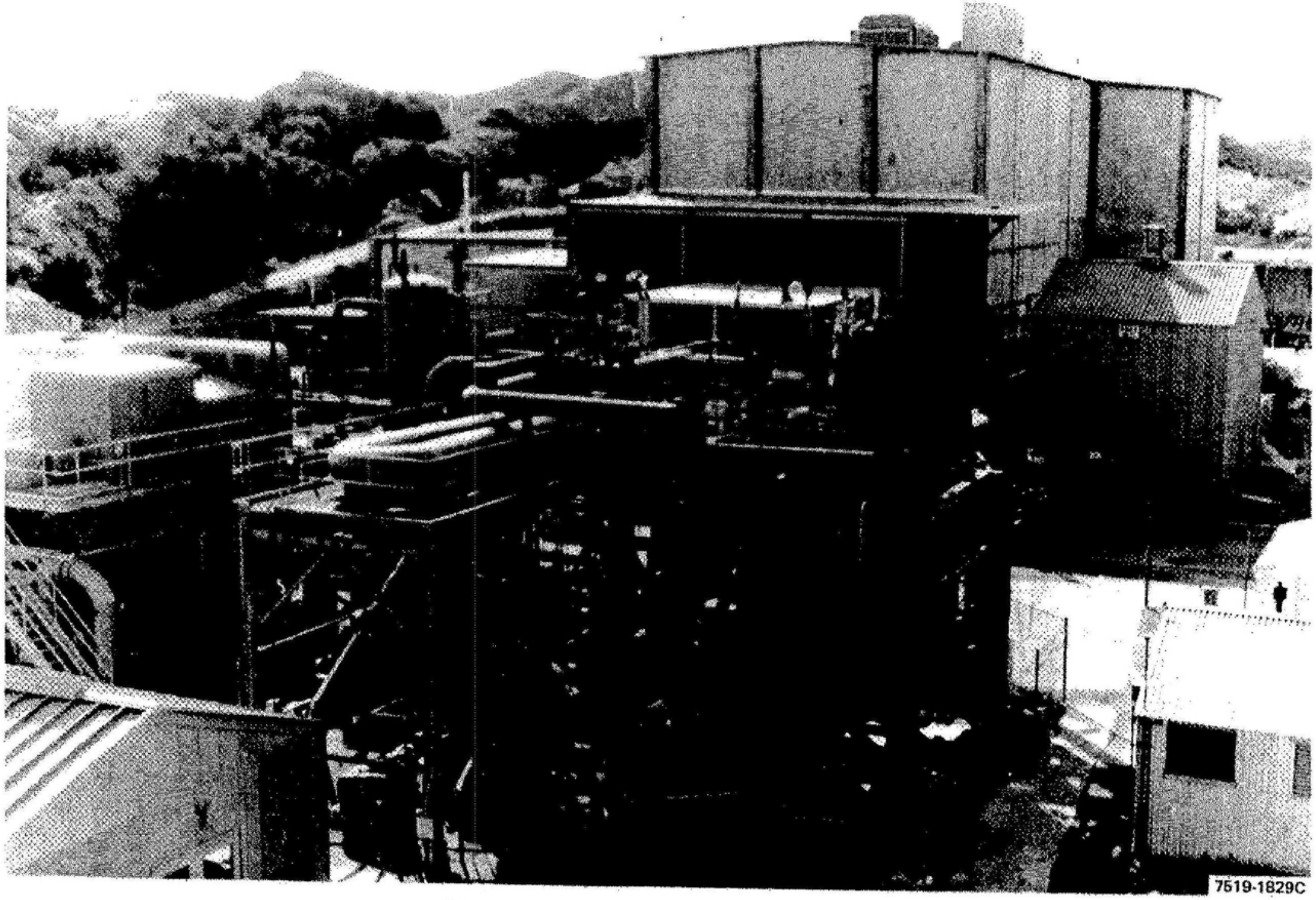

Figure 7. Steam Plant Facilities

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 with 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.

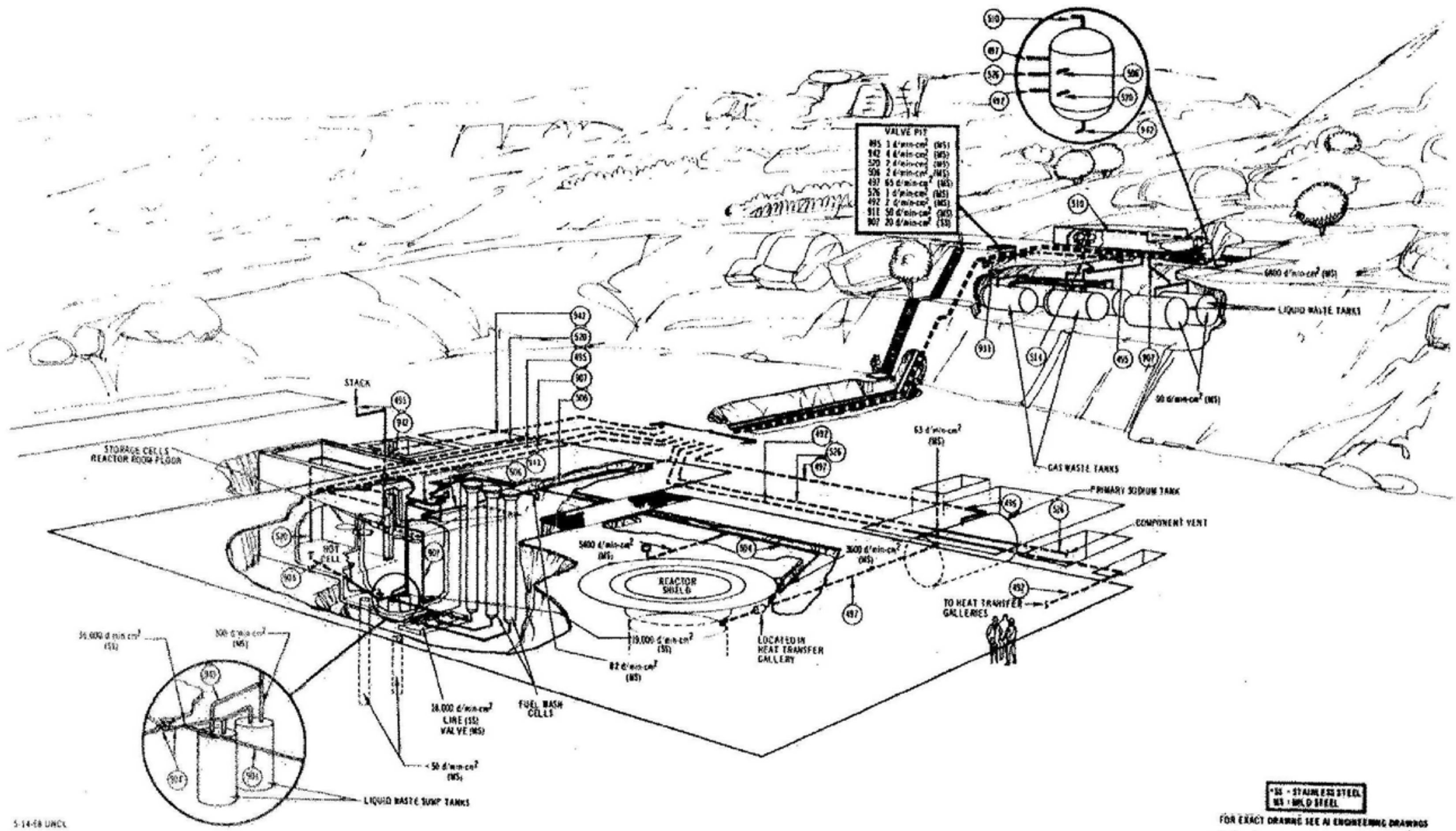

Figure 8. Radioactive Liquid Waste and Vent System

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

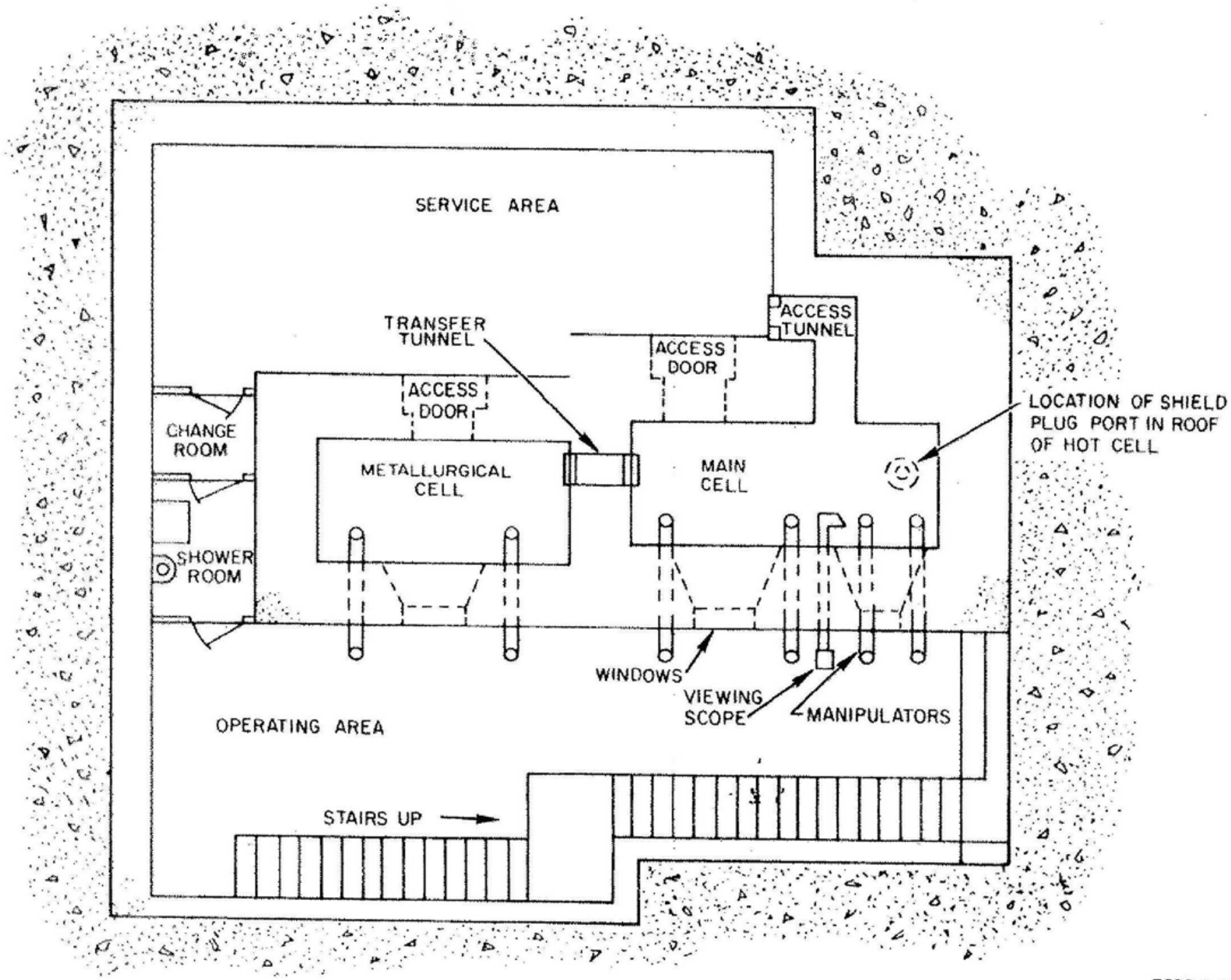

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Figure 9. SRE Hot Cell Facility — Plan View

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 decommissior ing 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

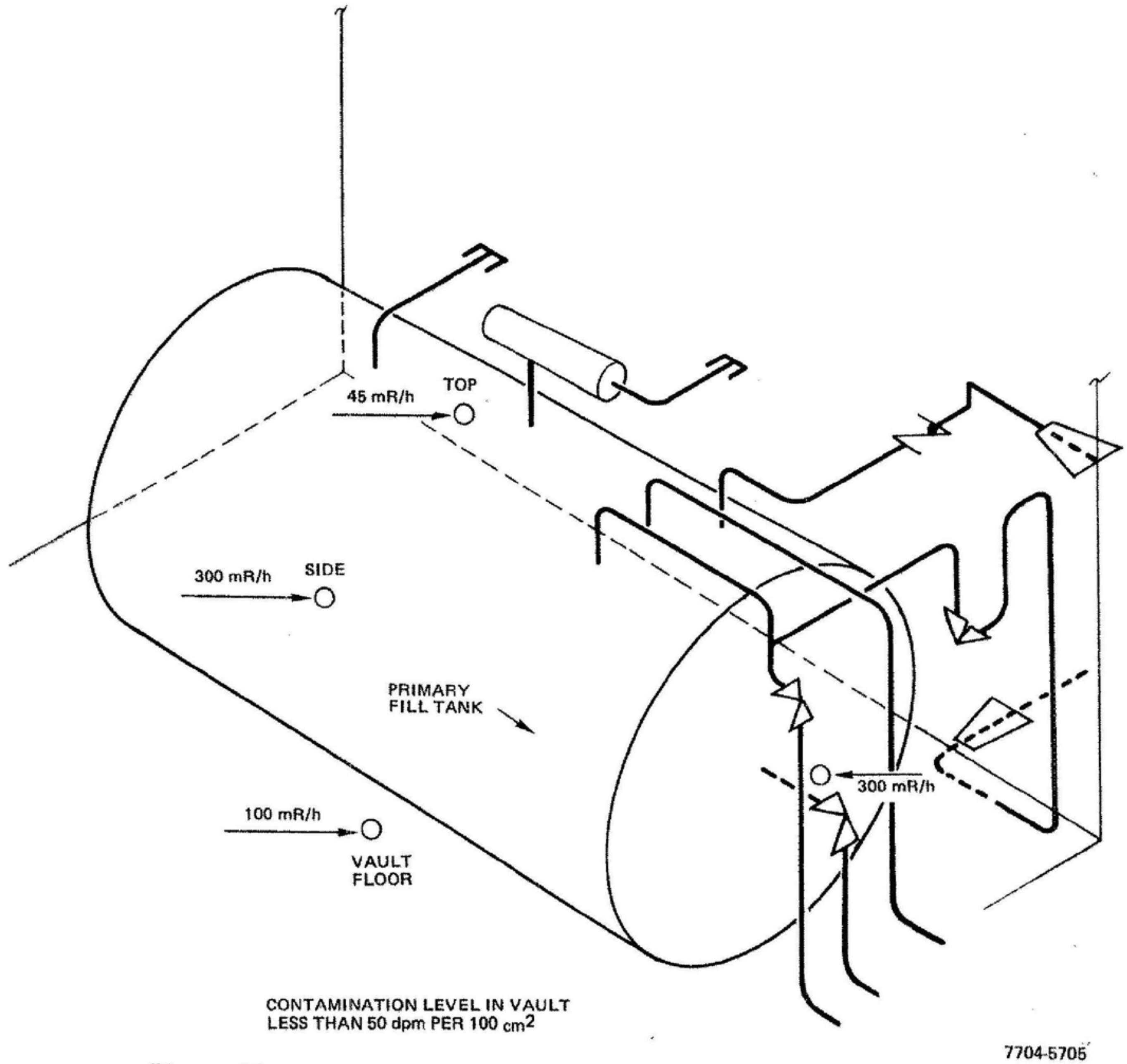

Figure 10. Radiation Levels from Tank and Associated Piping

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

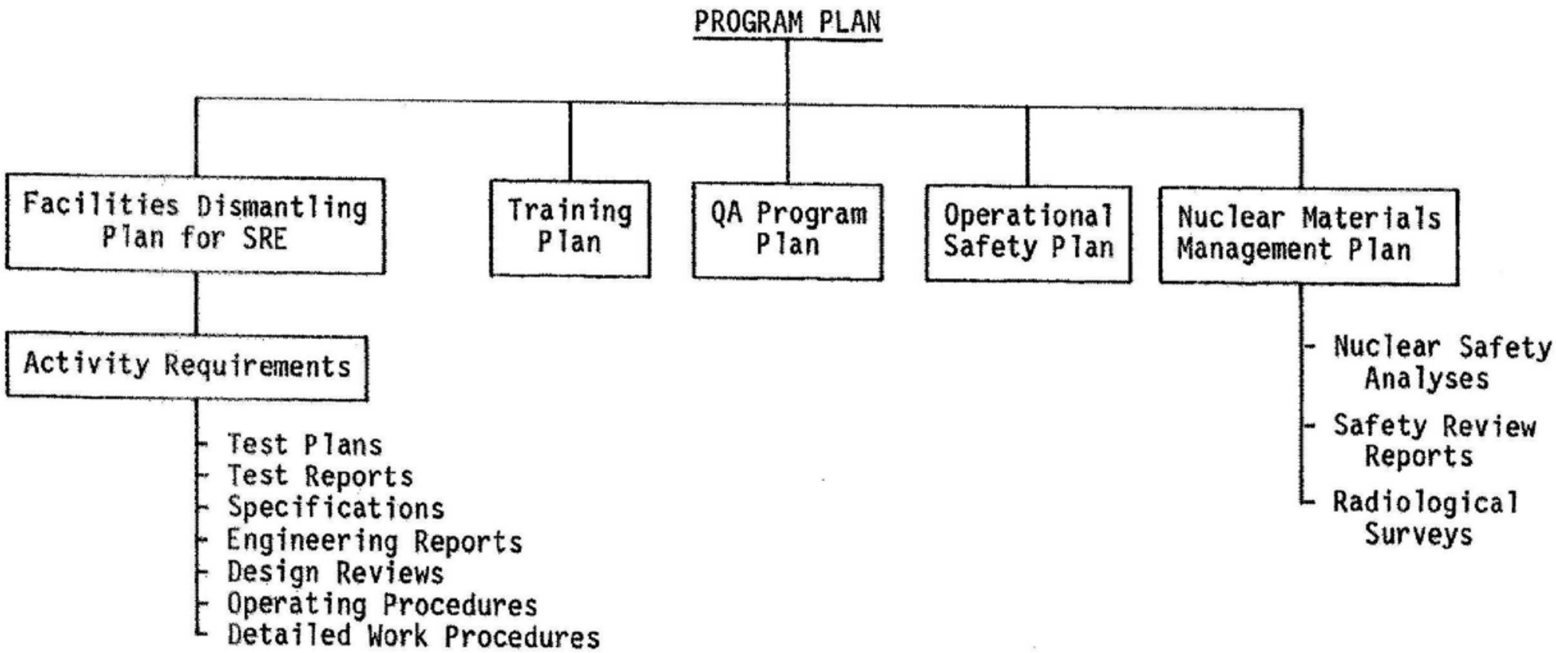

Figure 14. Document Flow Chart

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.

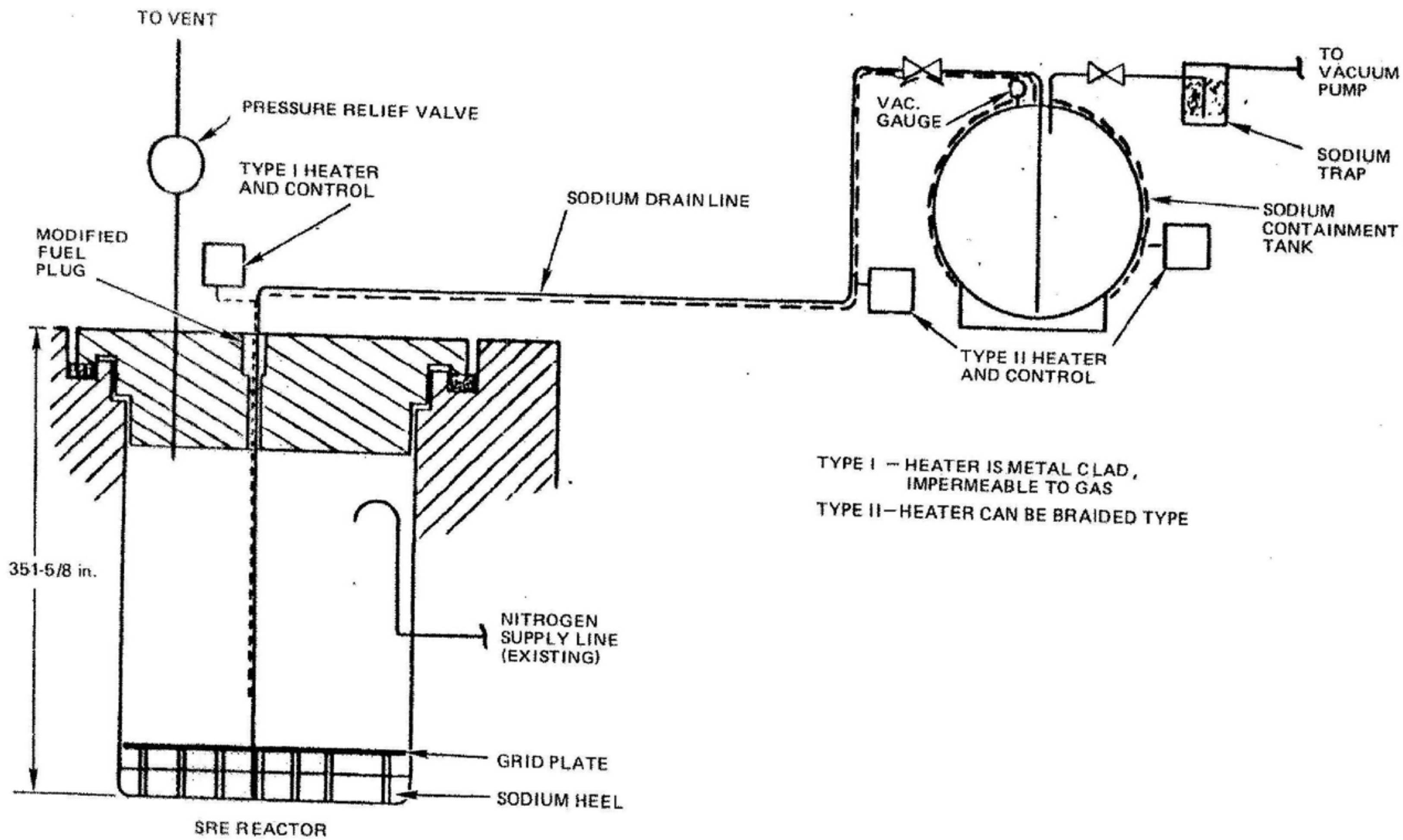

Figure 15. Sodium Vacuuming System

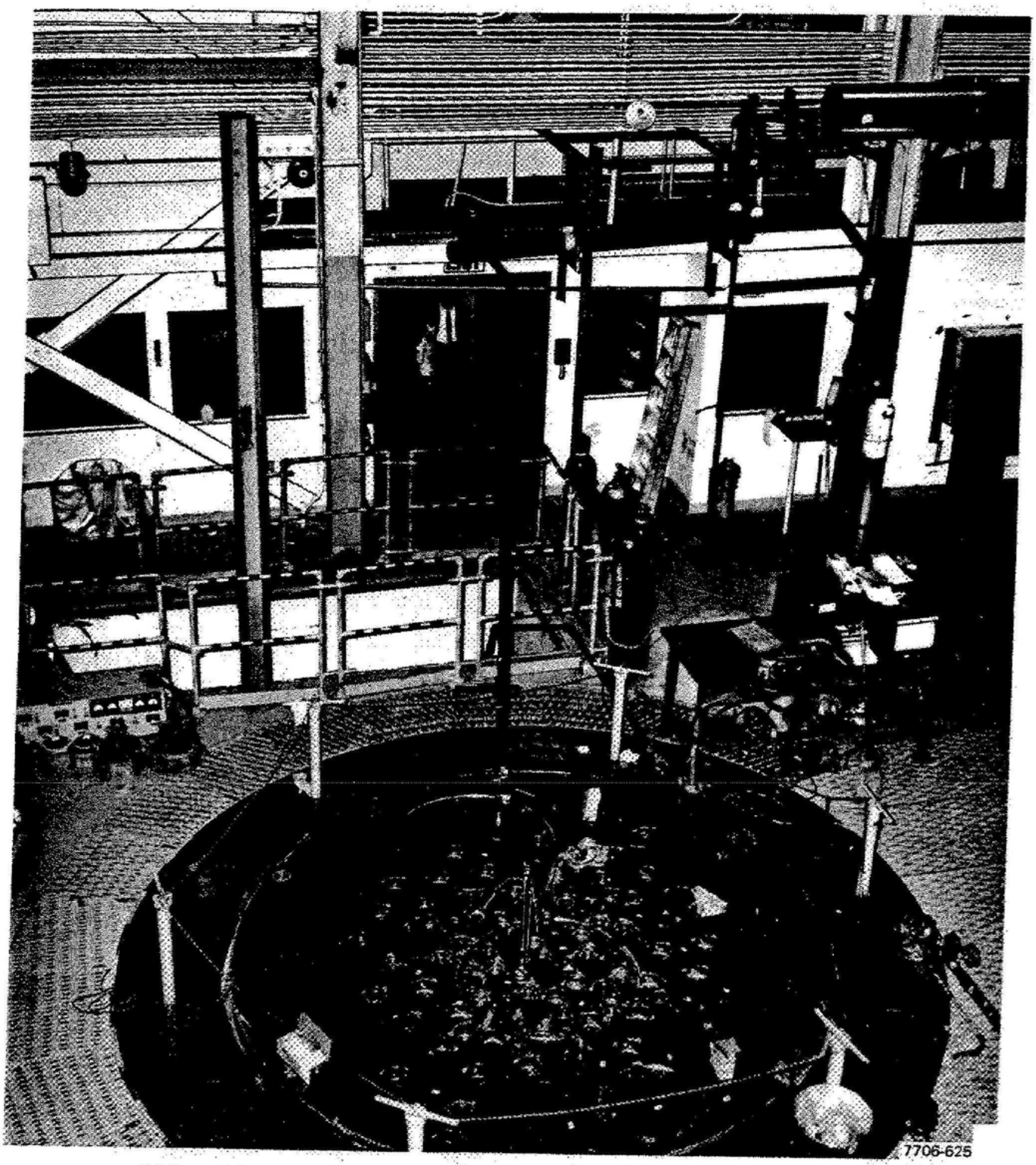

Figure 16. Reactor Passivation Piping During Installation

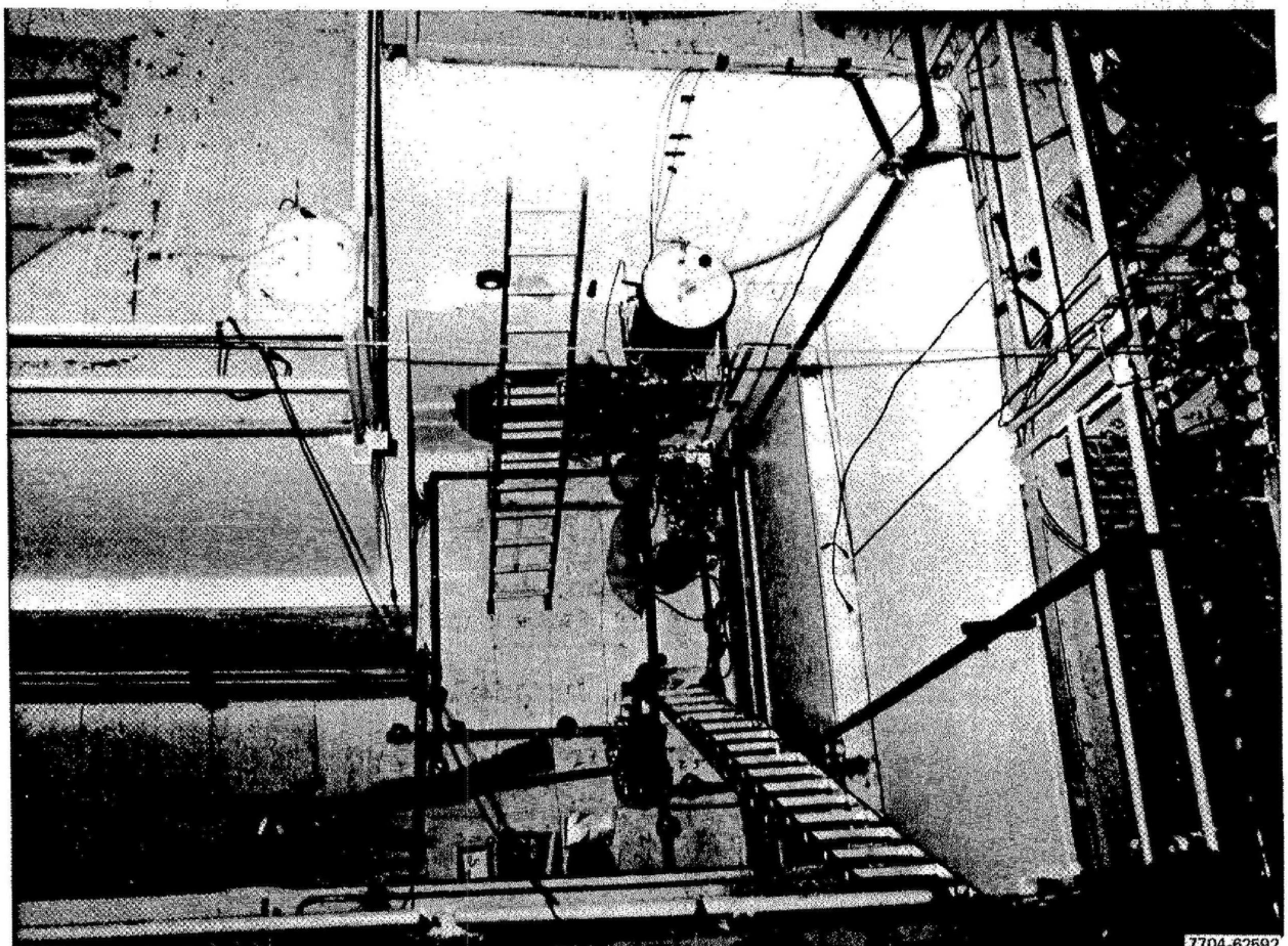

Figure 17. Reactor Passivation Piping in Primary Piping Vault

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.

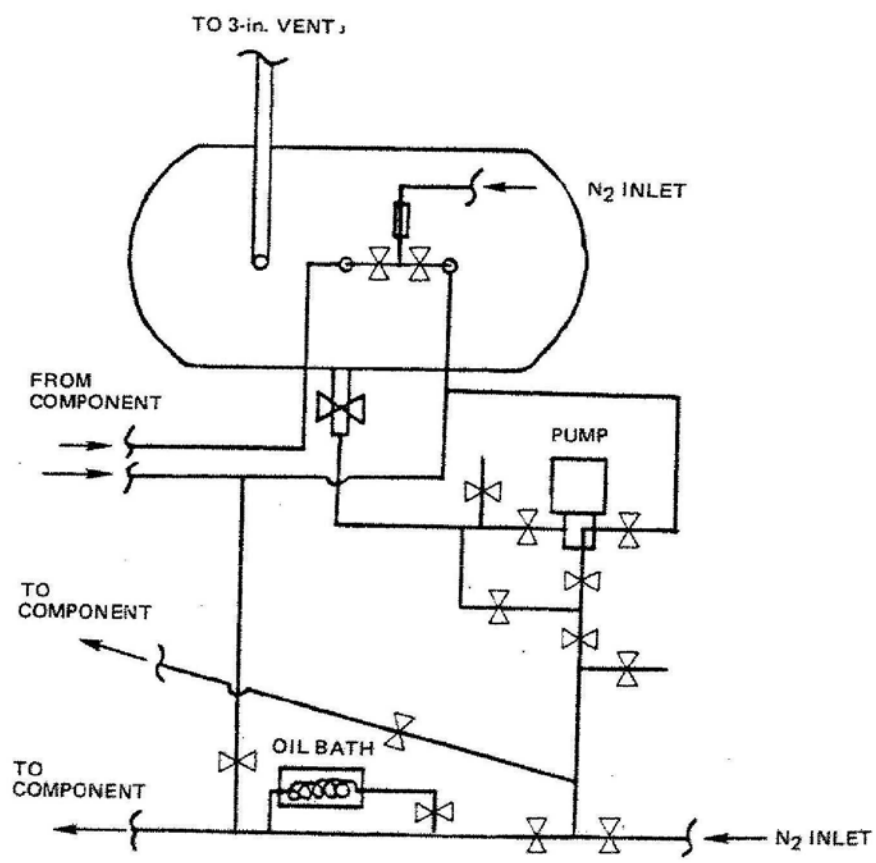

Figure 18. Small Component Sodium Passivation System

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.

9693-57131A

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.

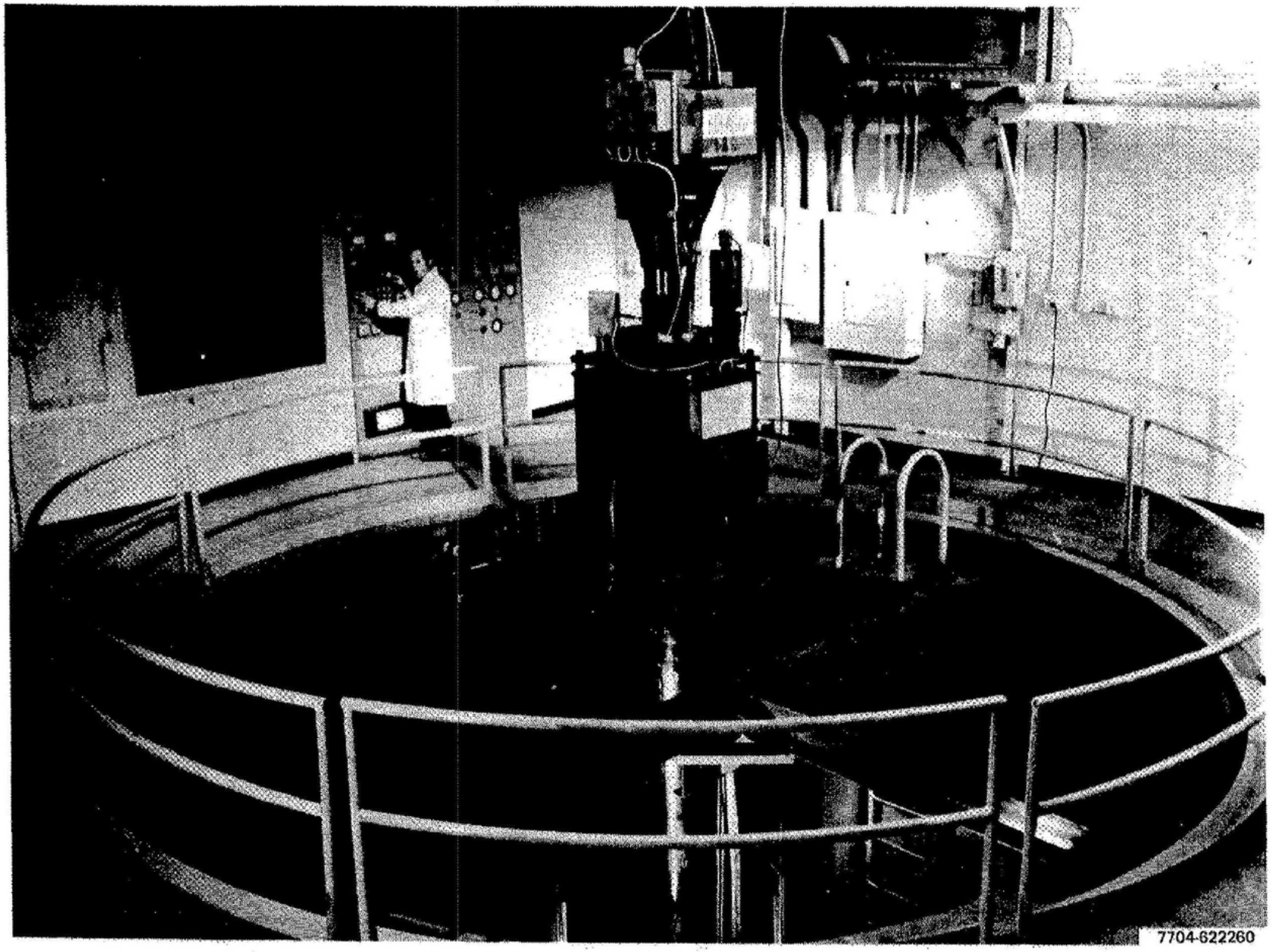

Figure 24. SRE Reactor Vessel Mockup Facility

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

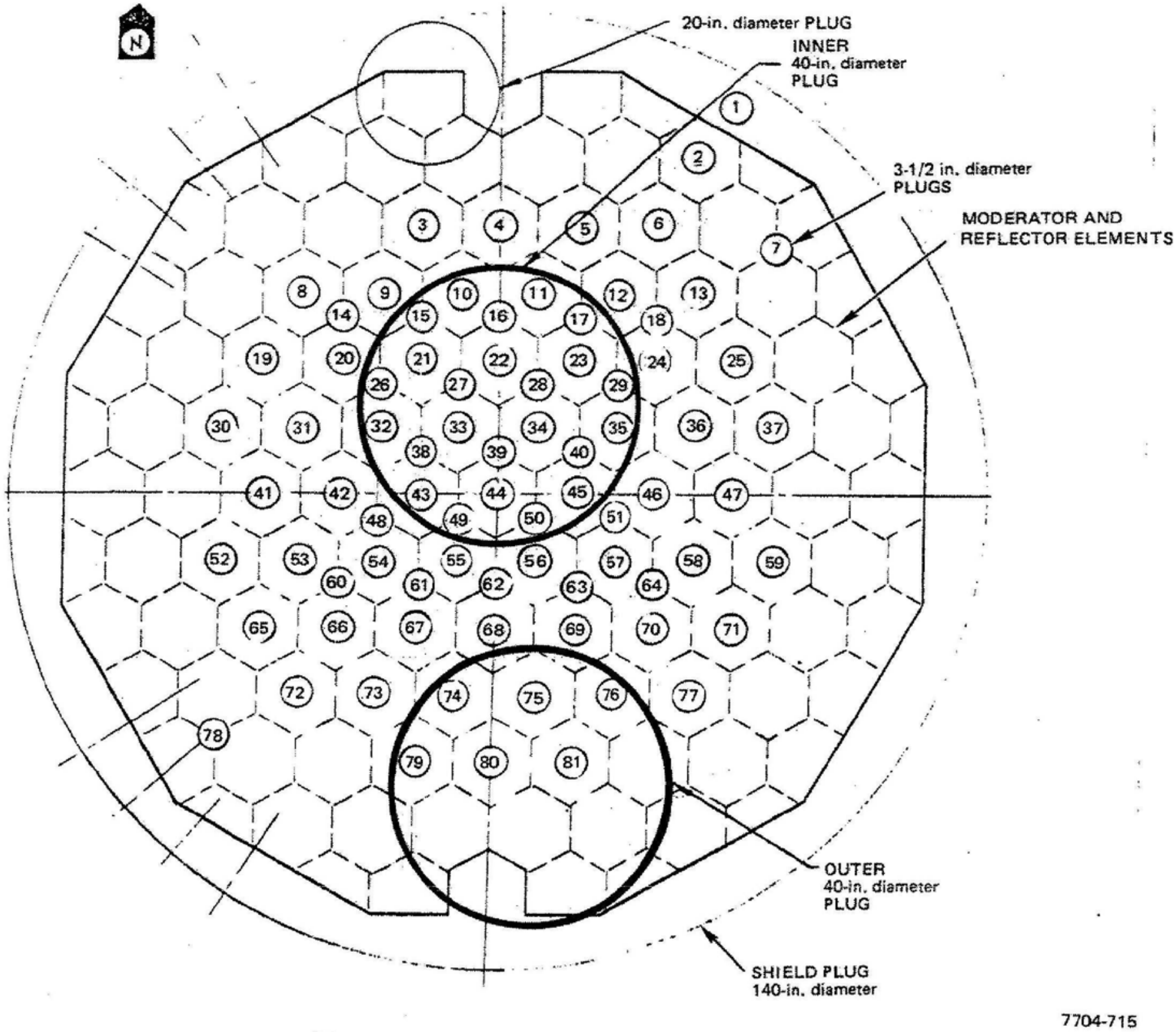

Figure 25. SRE Reactor Loading Face

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.

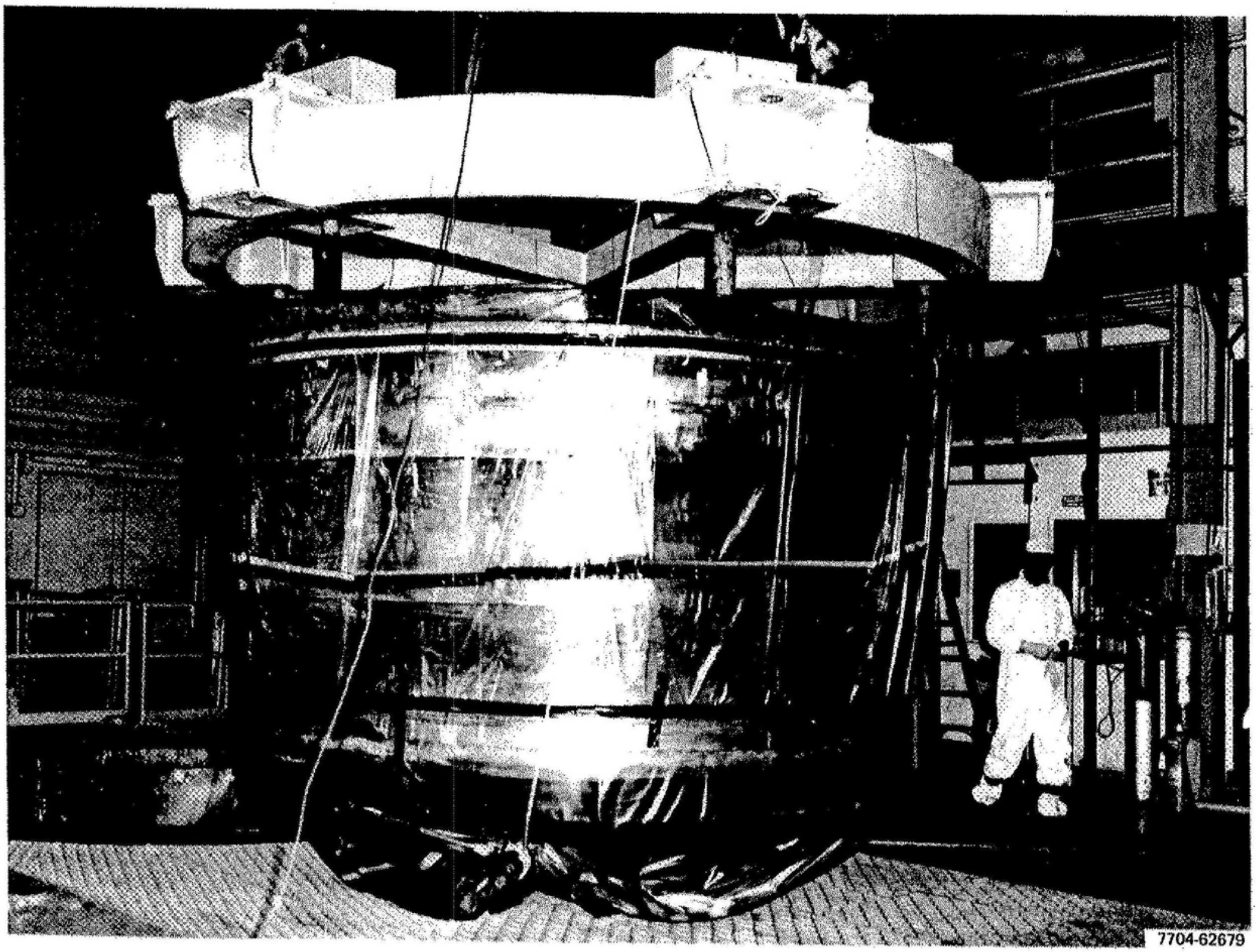

Figure 26. 140-in.-diameter Plug Removal

Figure 27. Reflector Shield Segments at RMDF

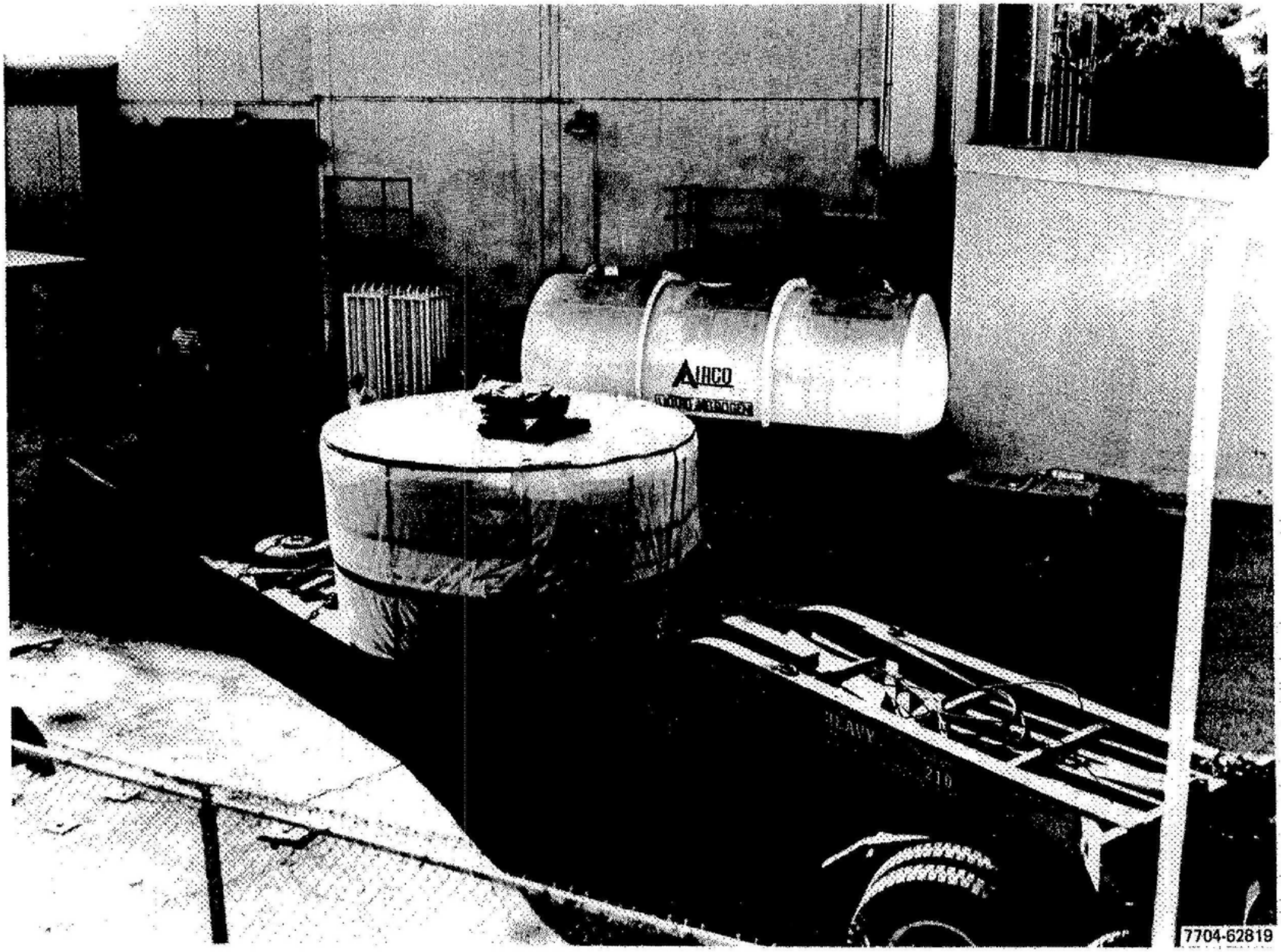

Figure 28. 140-in.-diameter Plug Packaged for Disposal

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.

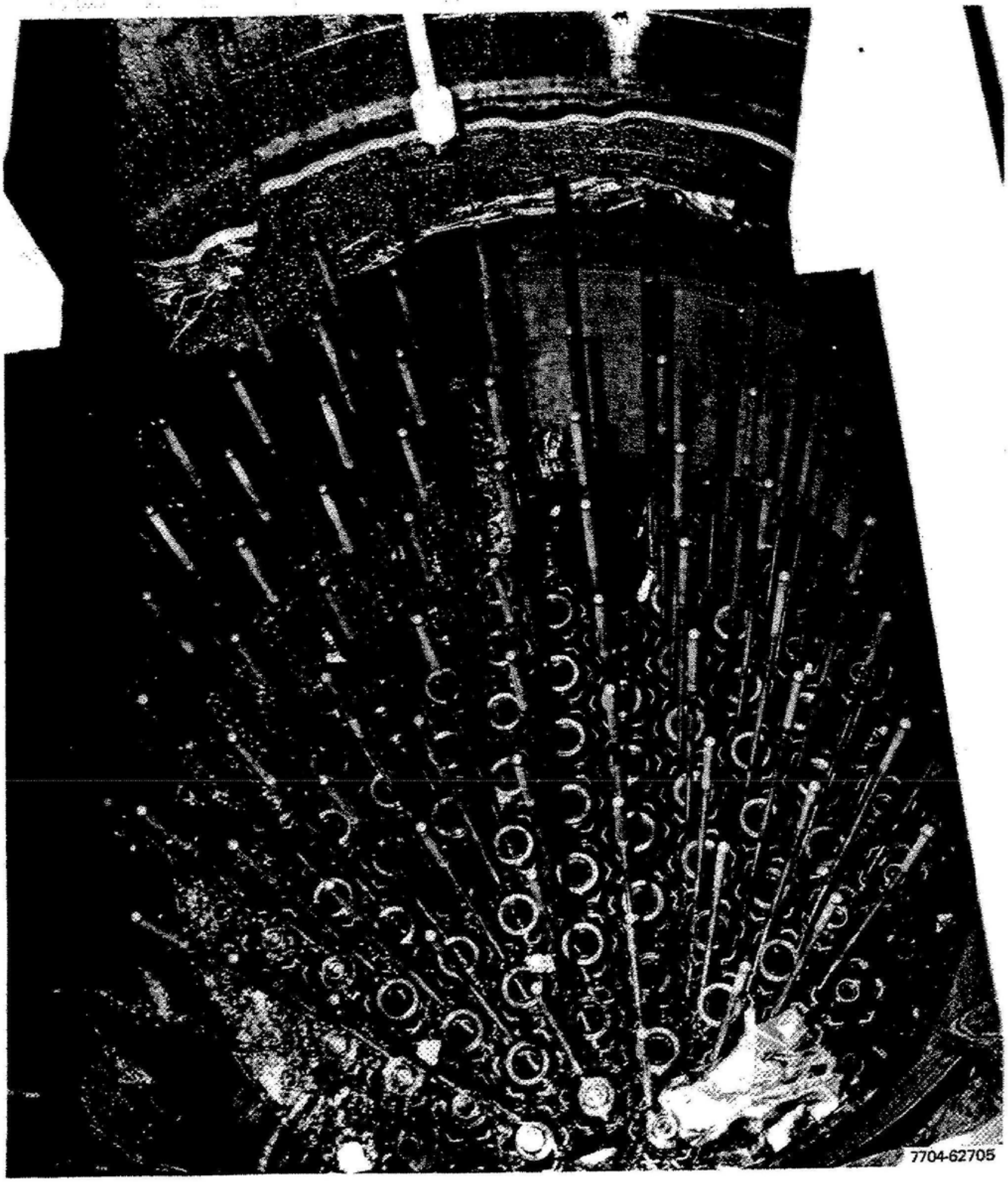

Figure 29. SRE Reactor Interior

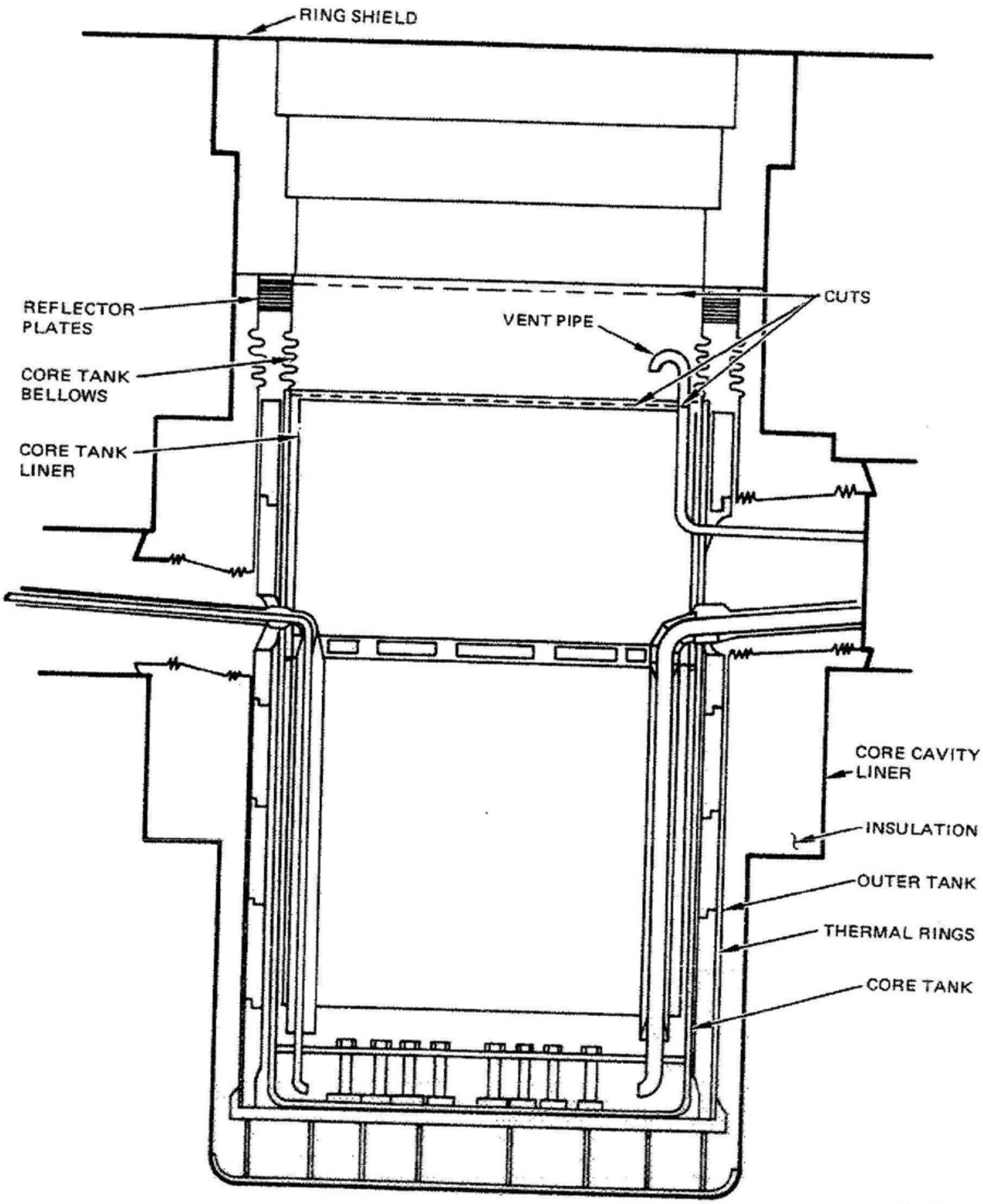

Figure 30. Bellows and Vent Pipe Cuts

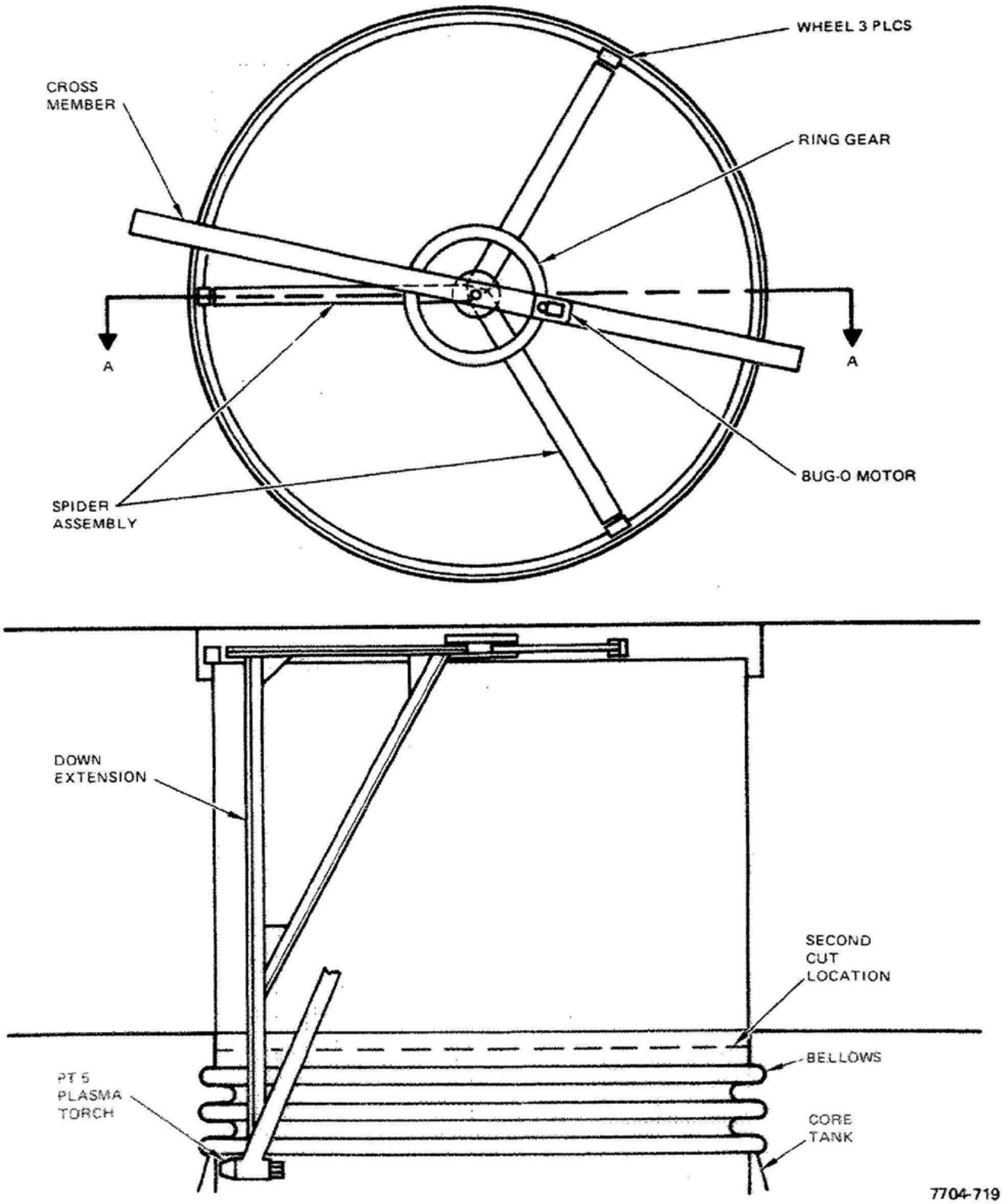

Figure 31. Bellows Cutting Fixture

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

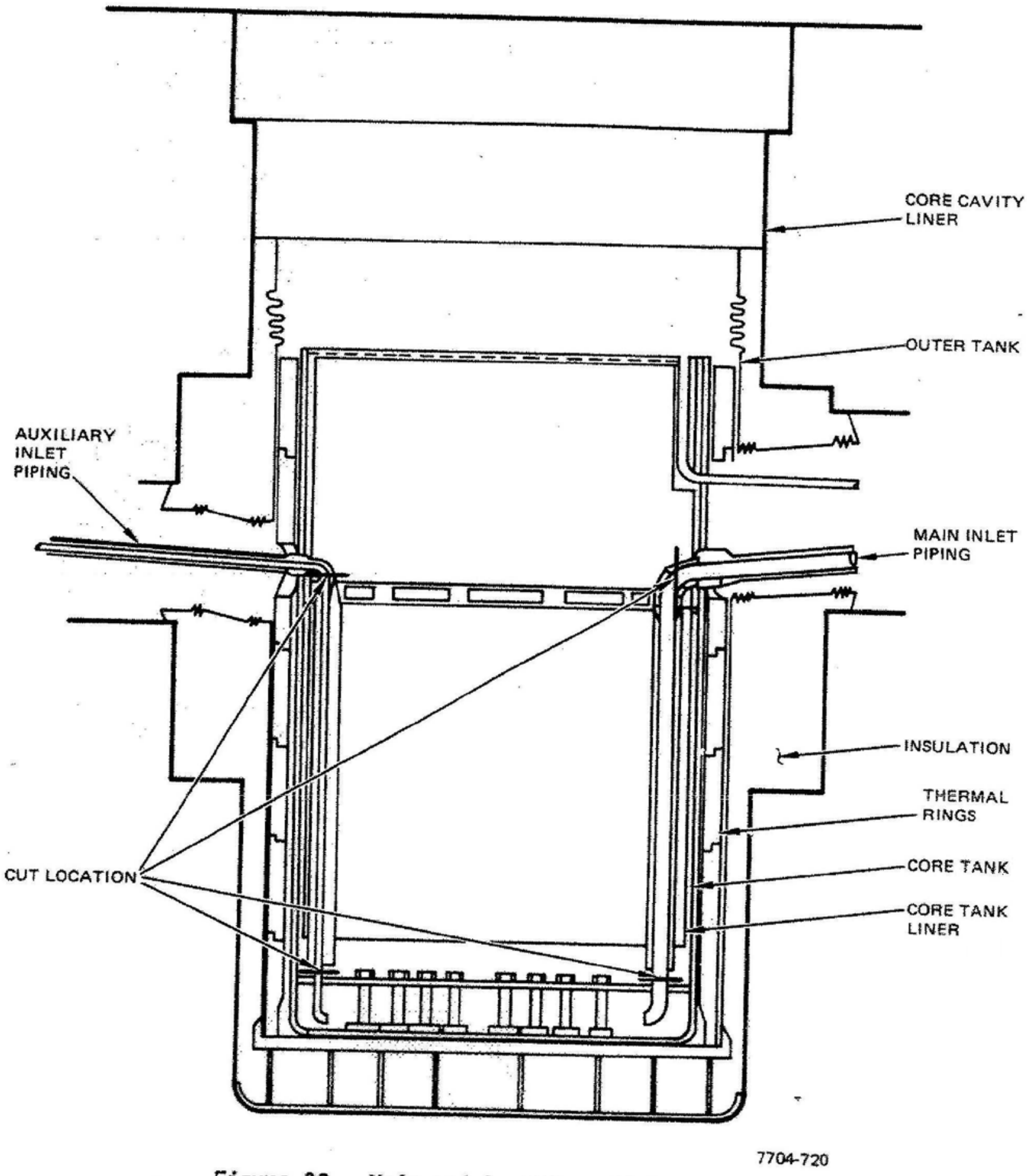

Figure 32. Main and Auxiliary Piping Cuts

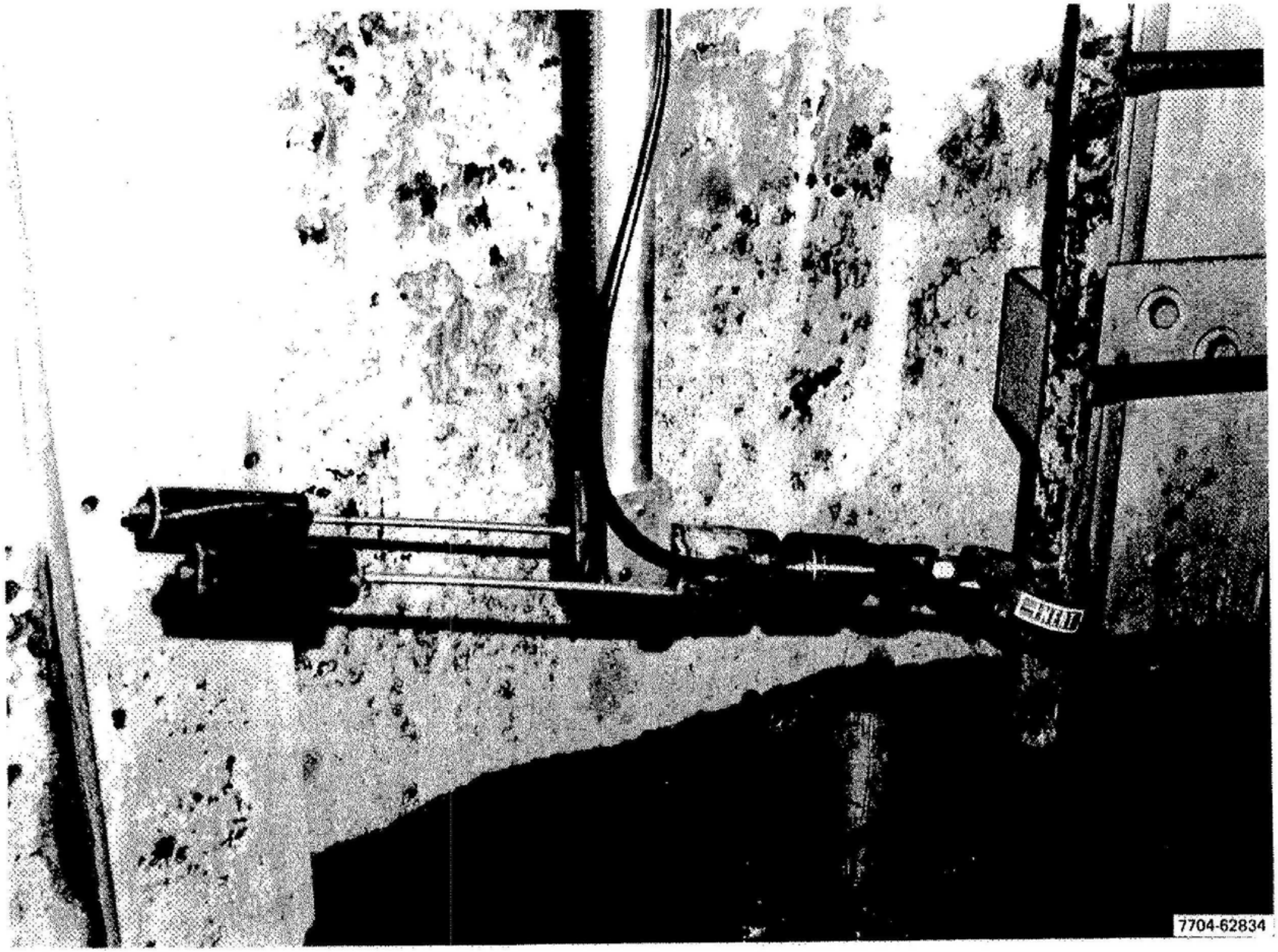

Figure 33. JRC Manipulator for Remote Installation of Explosive Charge

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.

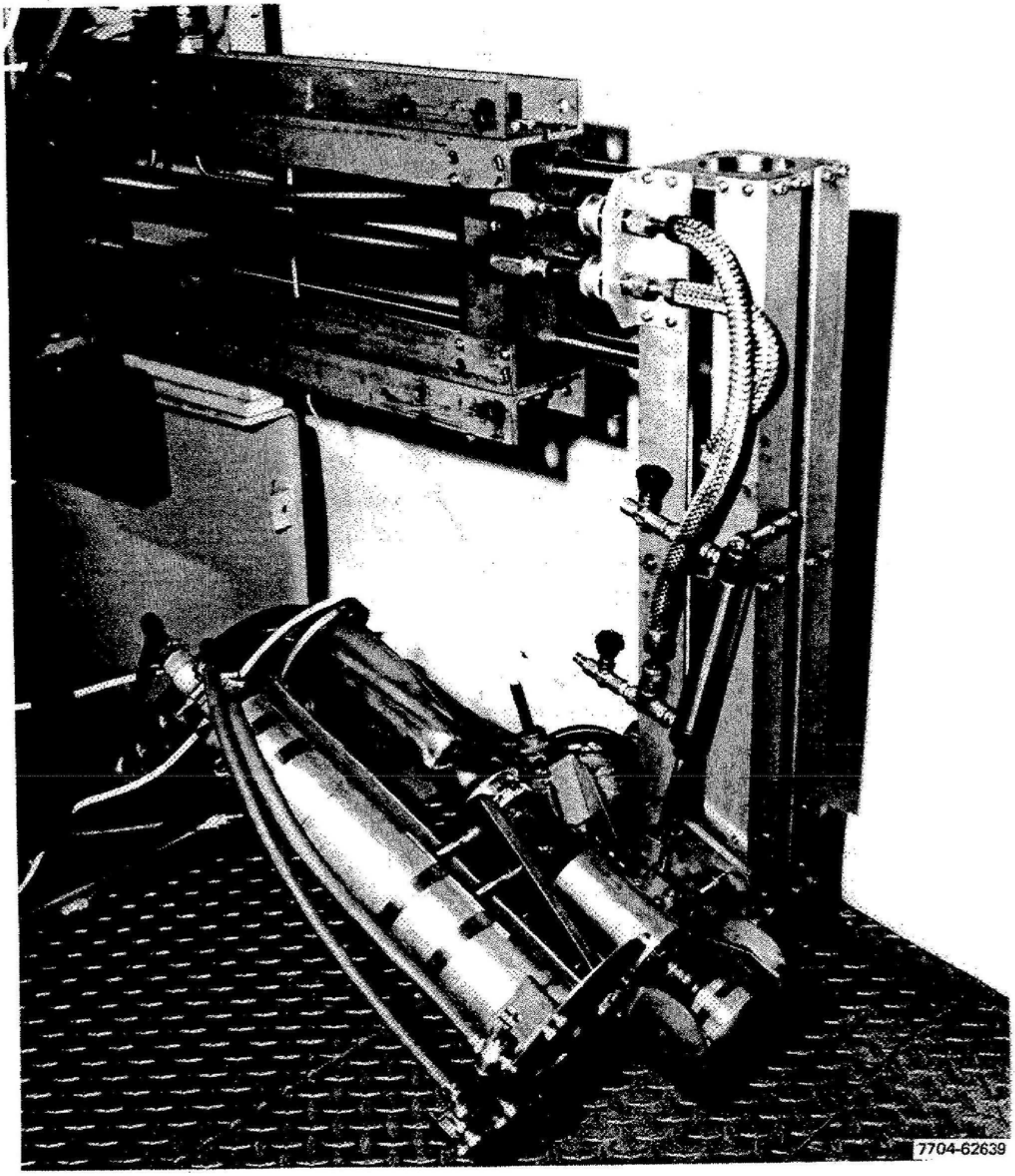

Figure 34. Plasma-Arc Cutting Torch Radial Arm

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

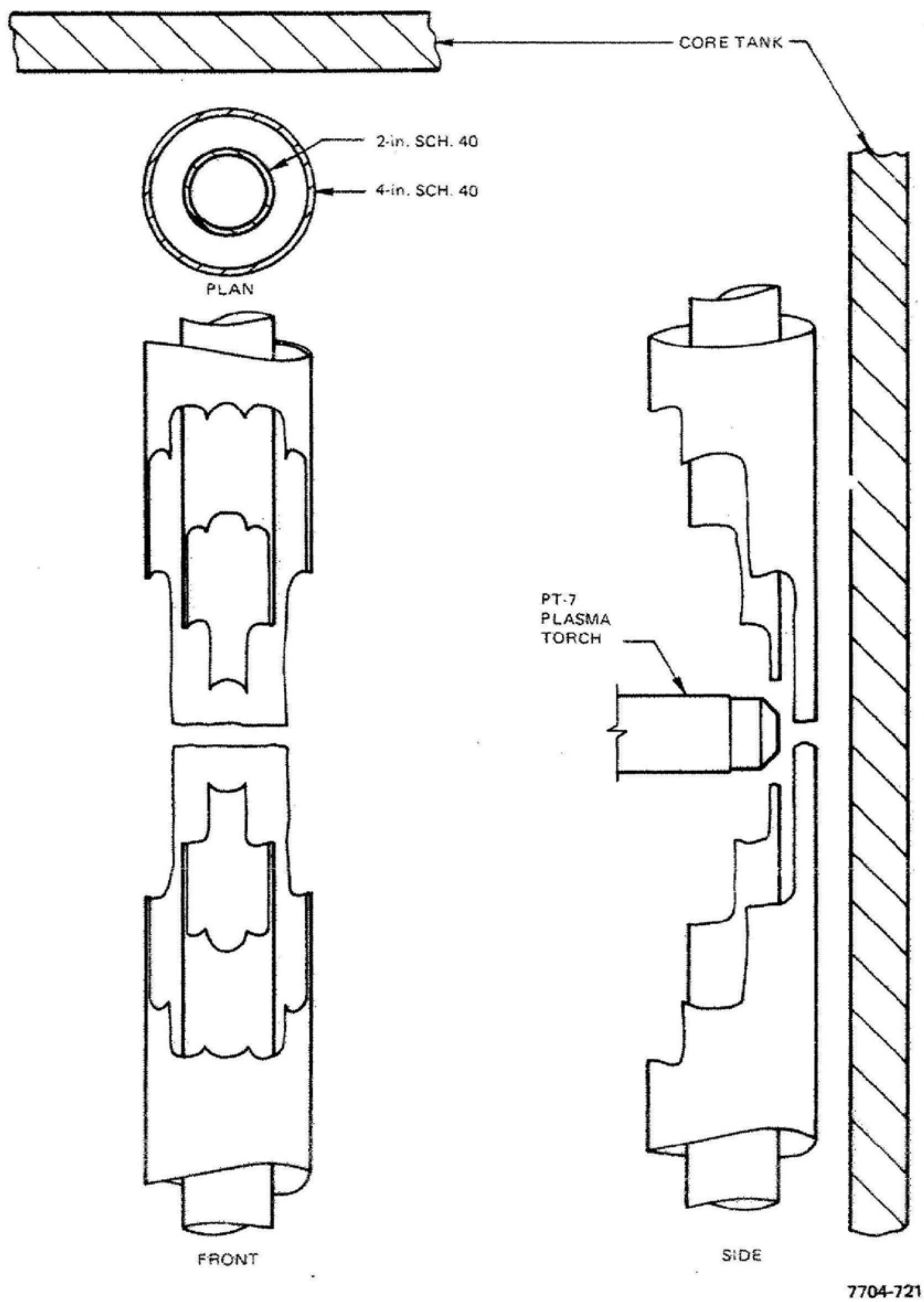

Figure 35. Pipe Cutting Using Plasma Torch

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.

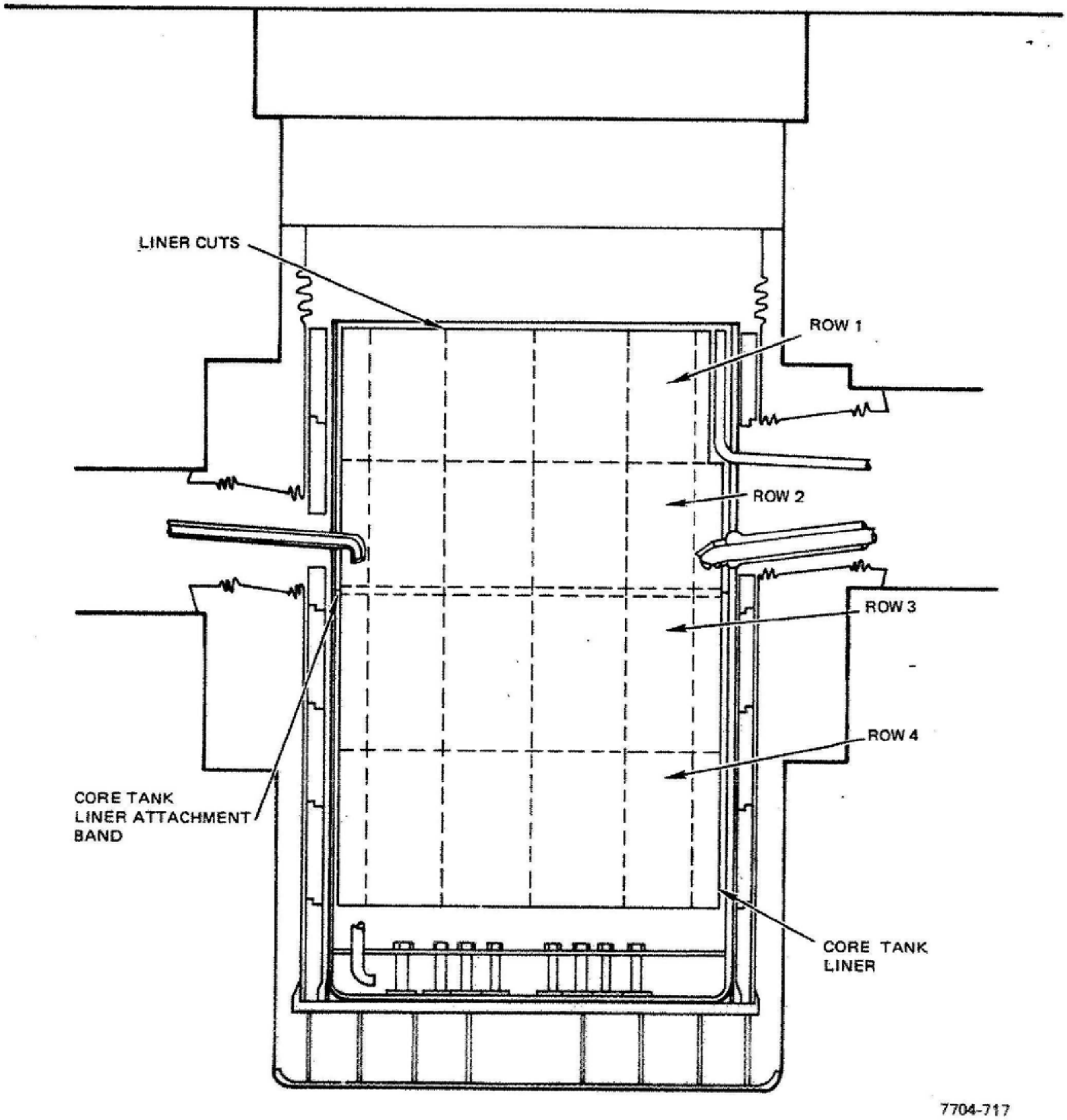

Figure 36. Core Tank Liner Cuts

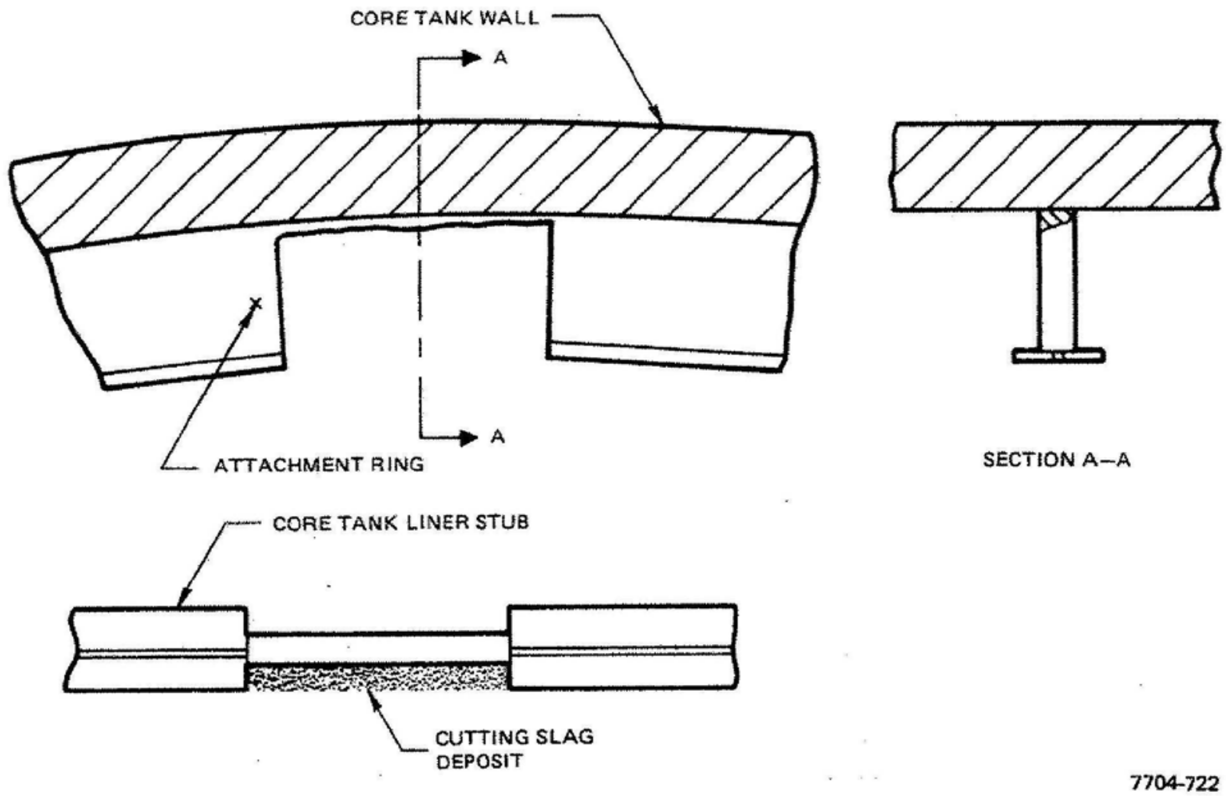

Figure 37. Attachment Ring Cuts

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.

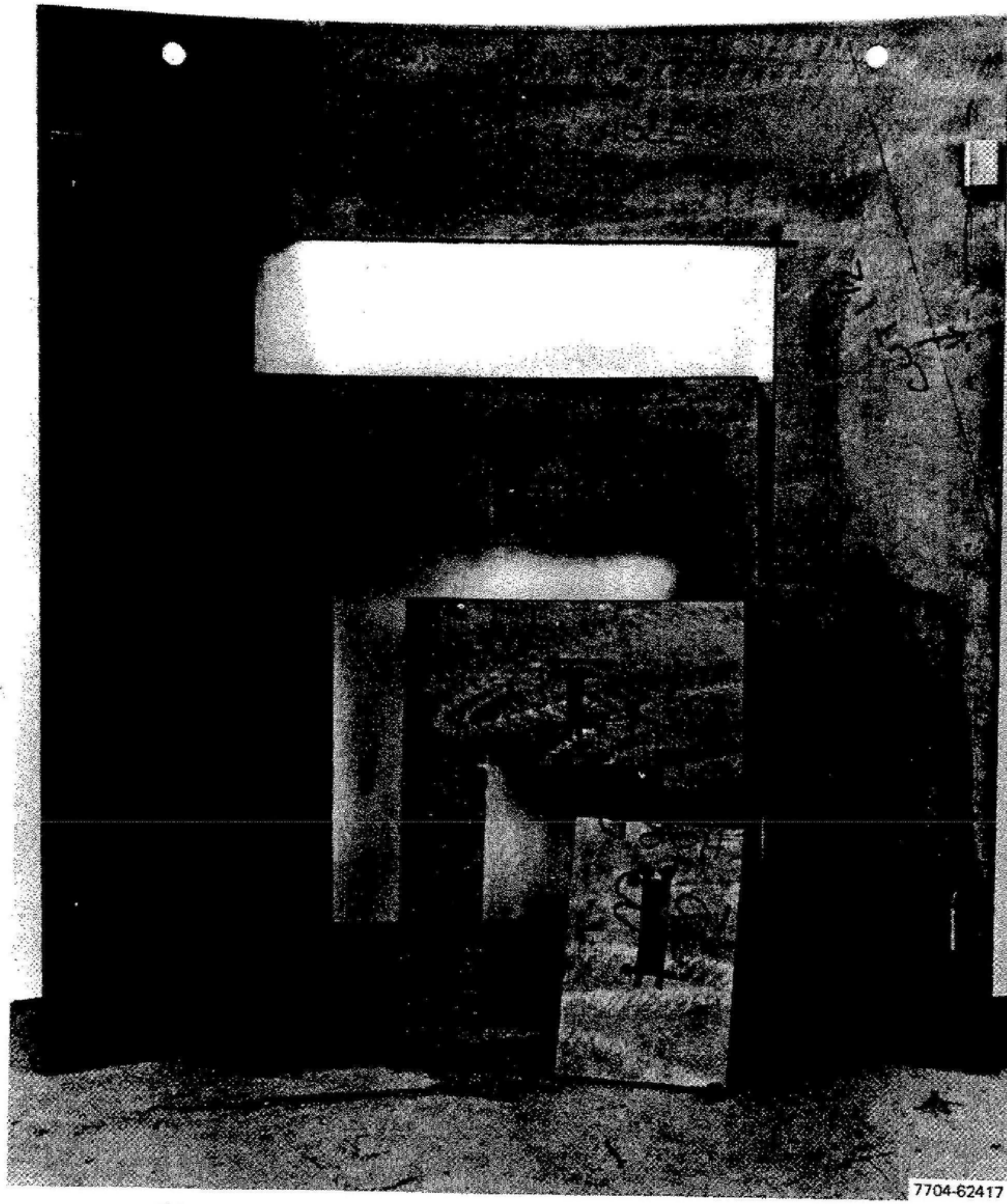

Figure 38. Core Tank Liner Mockup Cutting Test Results

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

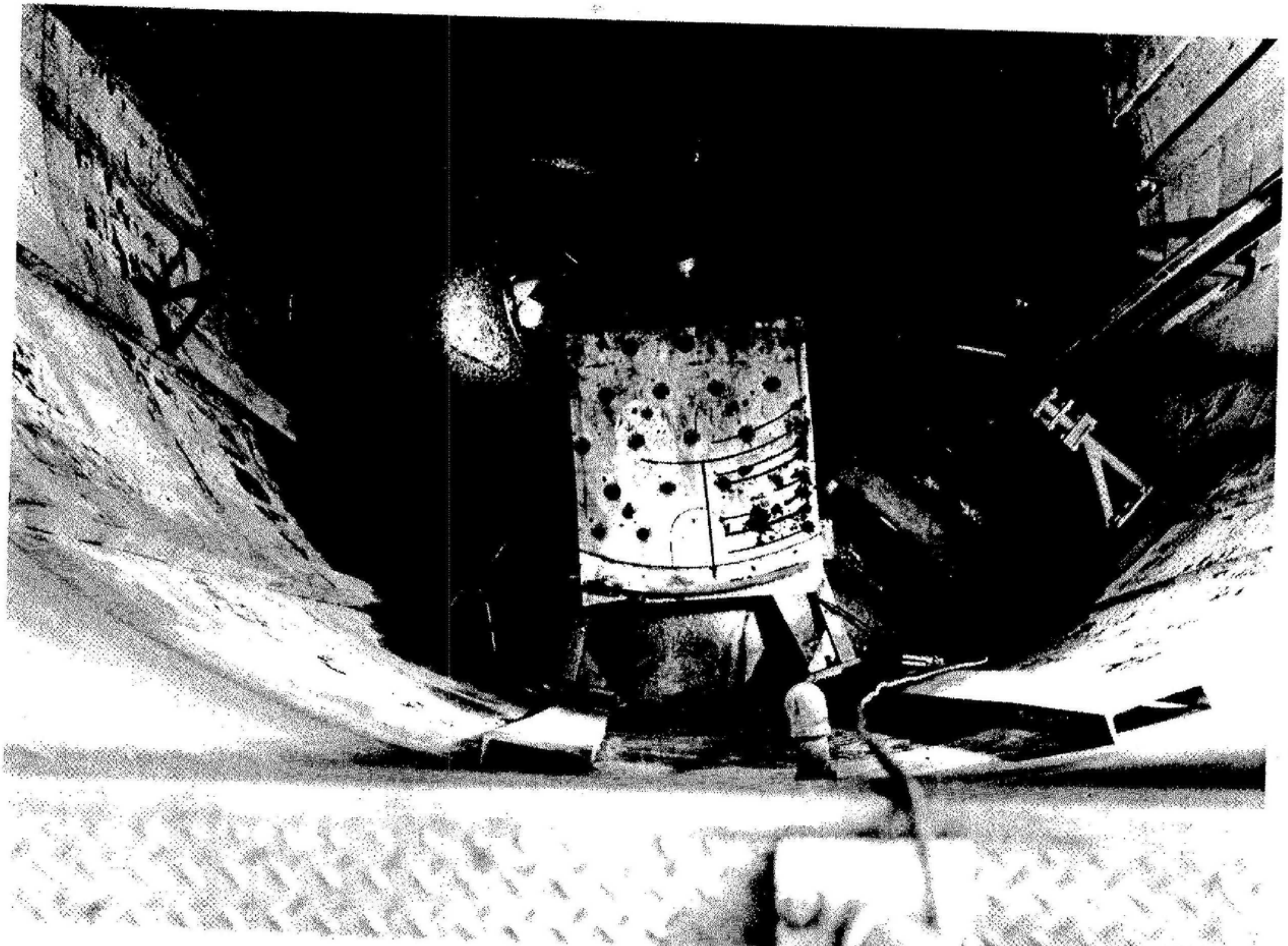

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Figure 39. Mockup of SRE Reactor Grid Plate

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

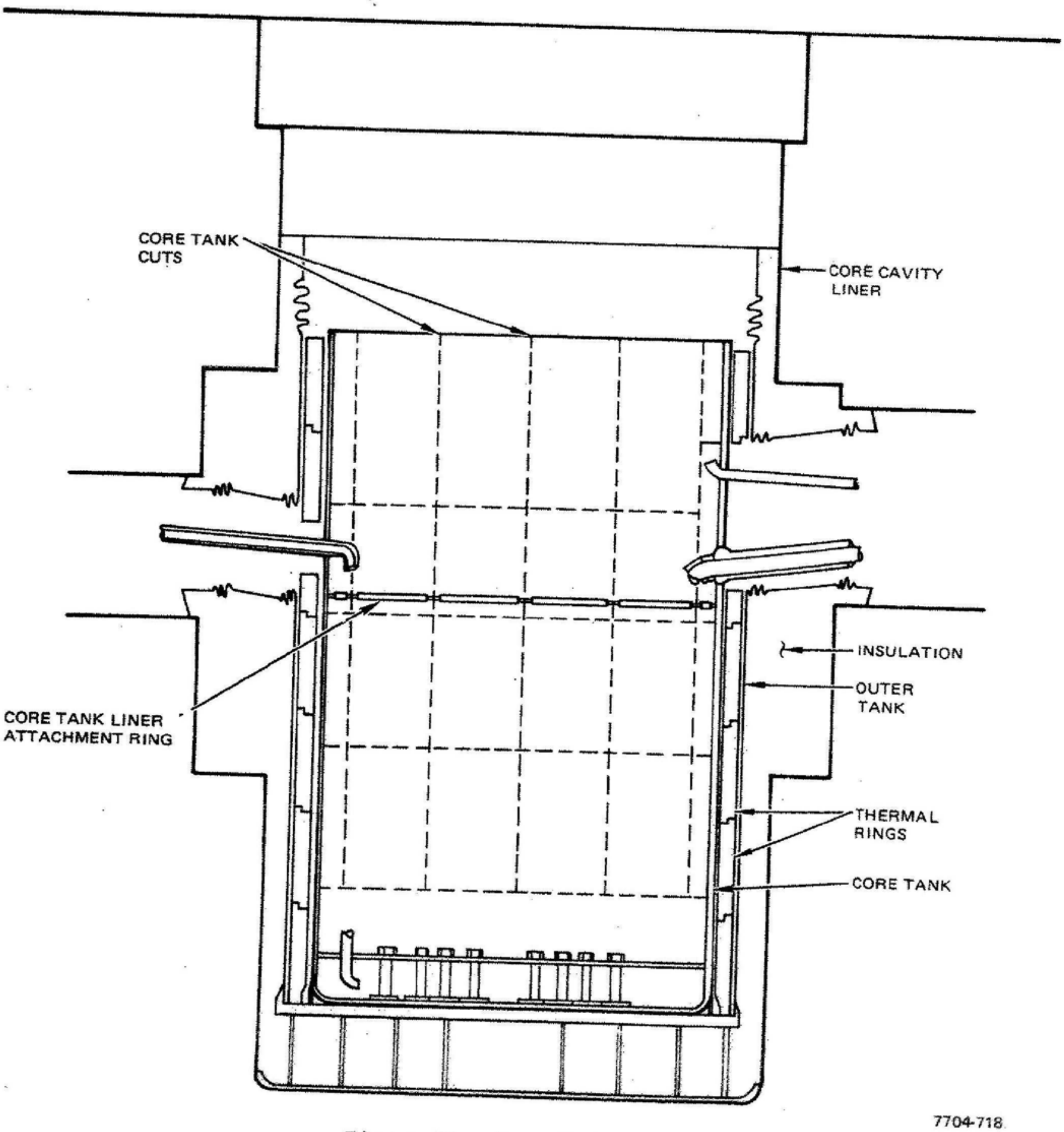

Figure 40. Core Tank Cuts

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.

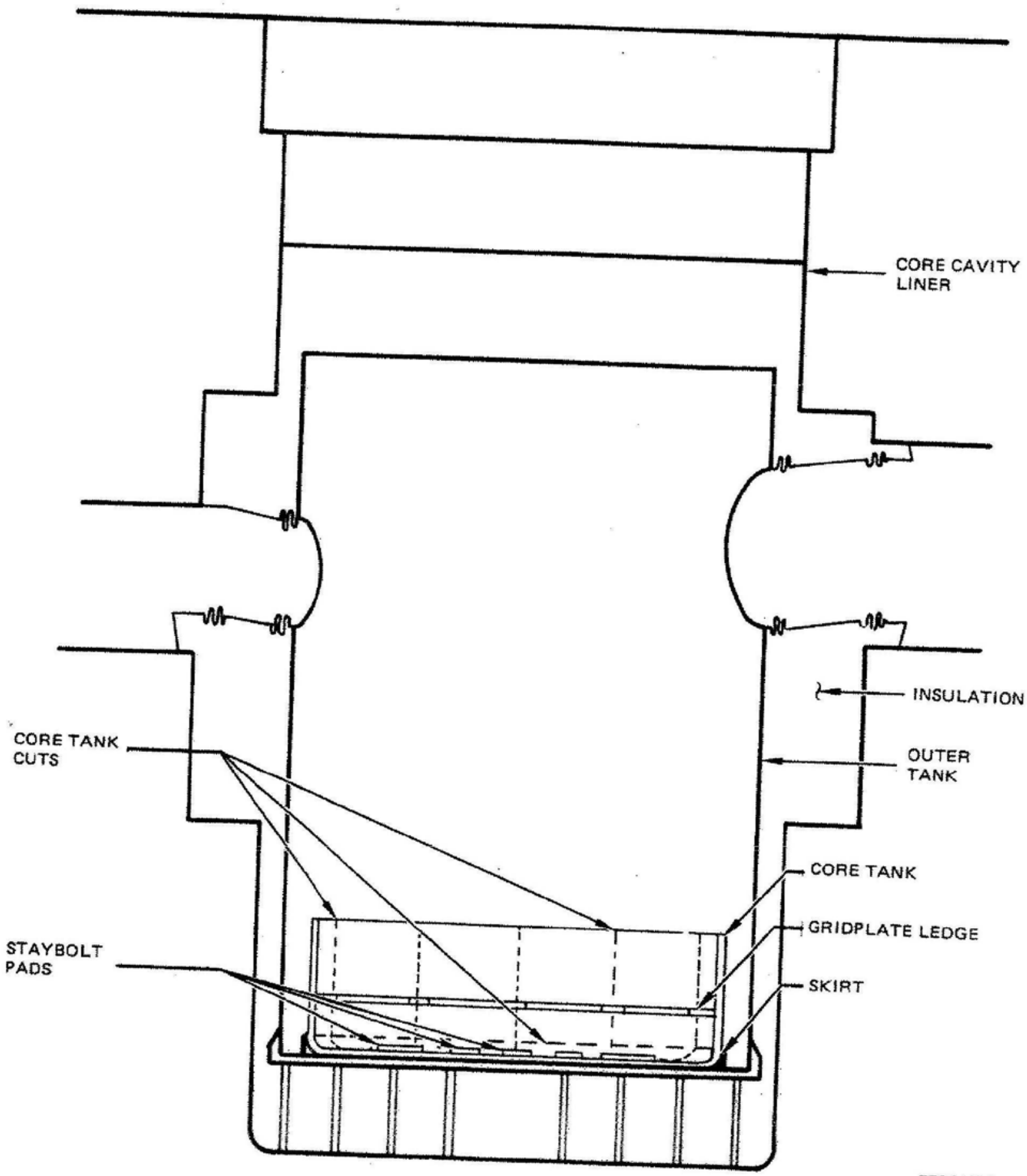

Figure 41. Core Tank Bottom Cuts

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

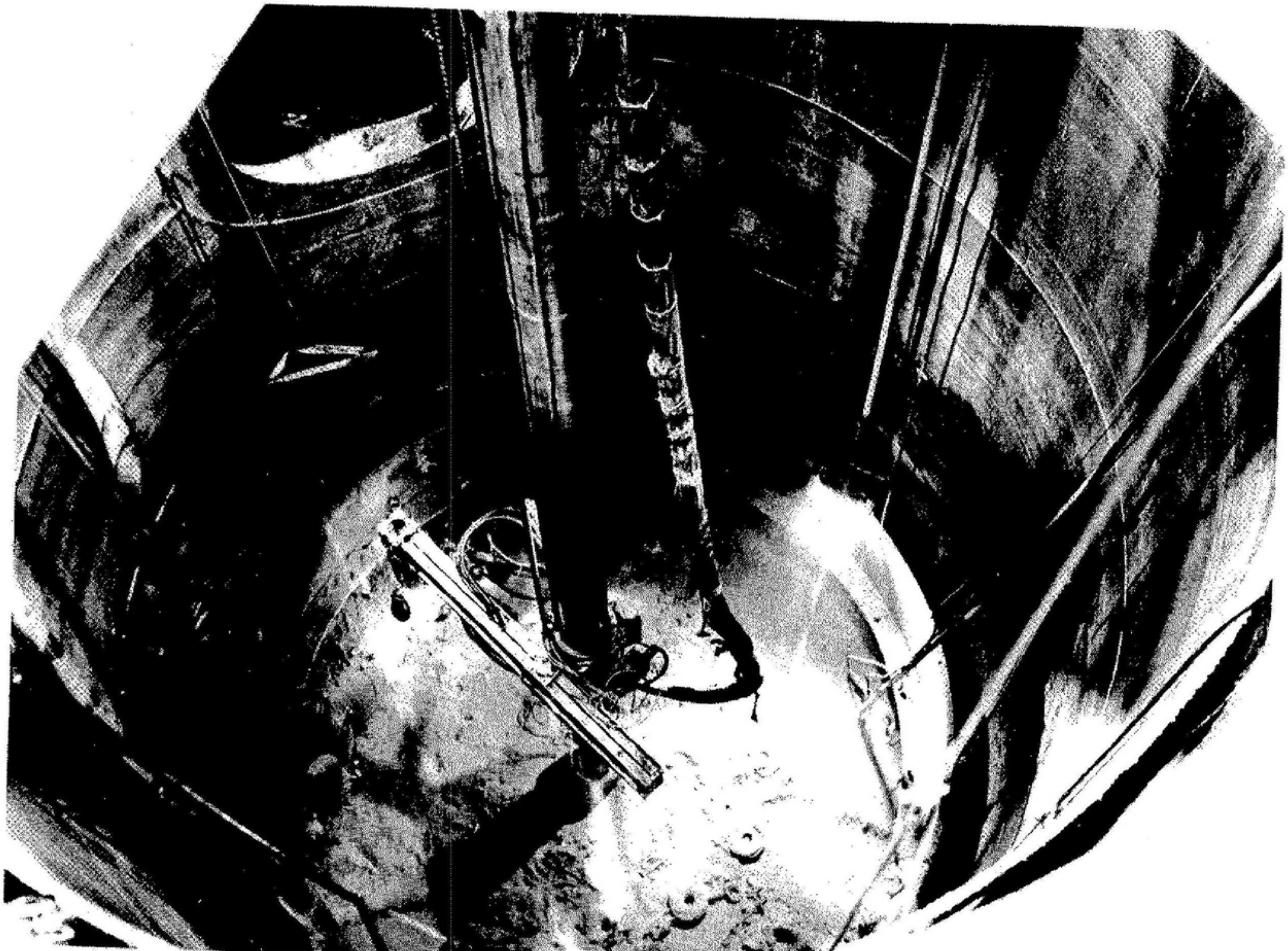

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Figure 42. SRE Reactor Core Tank Wall Removal

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.

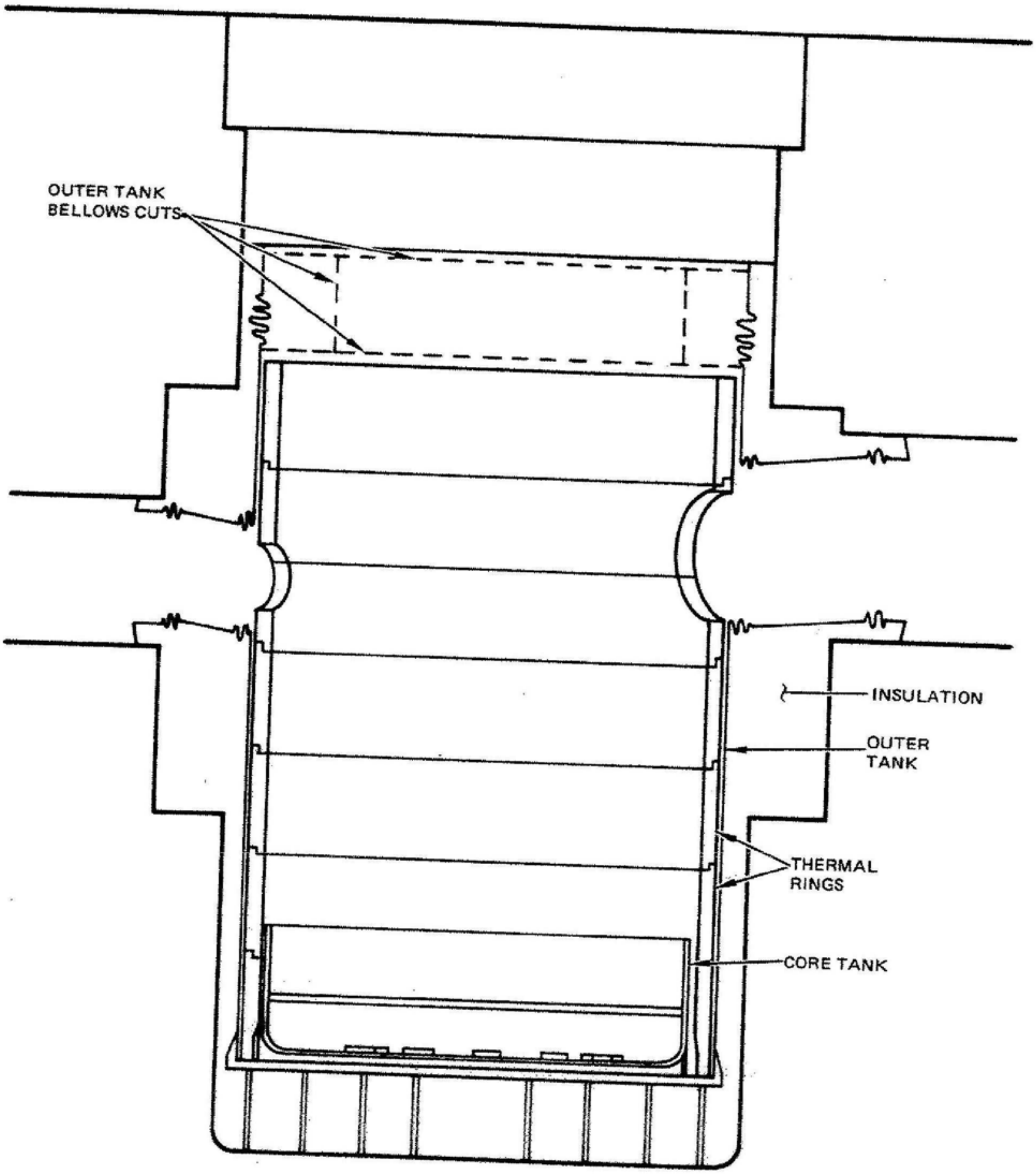

Figure 43. Outer Tank Bellows Cuts

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

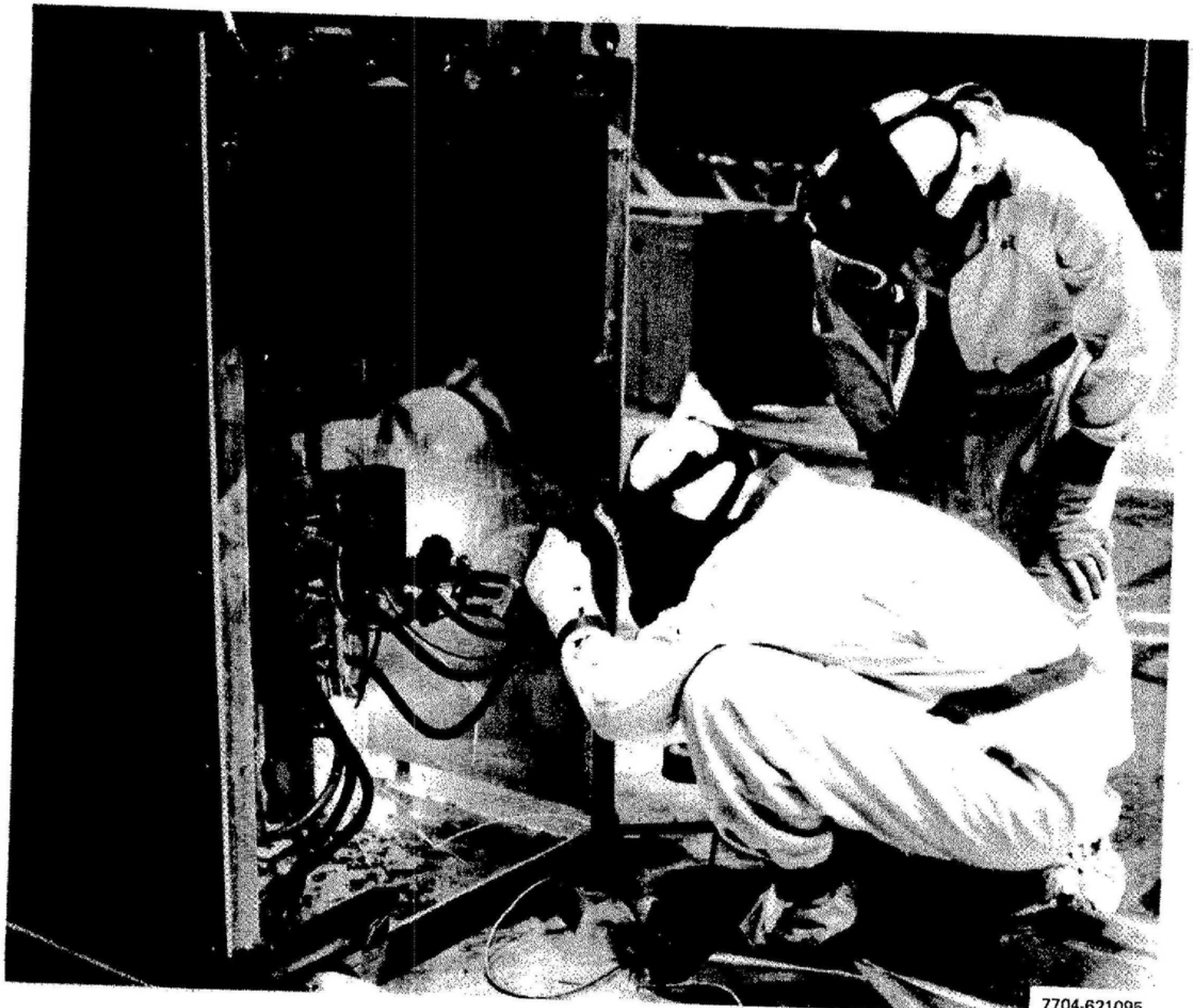

Figure 44. Thermal Ring Being Cut