Atomics International Division
Rockwell International

SUPPORTING DOCUMENT

NUMBER: FDP-704-990-003
REV LTR/CHG NO. SEE SUMMARY OF CHG

DOCUMENT TYPE: Facilities Dismantling Plan

KEY NOUNS: Dismantling Plan

PROGRAM TITLE: Decontamination and Disposition of Facilities Program

DOCUMENT TITLE: Facilities Dismantling Plan for SRE

ORIGINAL ISSUE DATE: 6-24-75

GO NO.	S/A NO.	PAGE 1 OF TOTAL PAGES
07704	10100	84

REL. DATE: 6-26-75

PREPARED BY/DATE

	DEPT	MAIL ADDR
B. Ureda	713	NB02
W. Heine	713	NB02

IR&D PROGRAM? YES ☐ NO ☒ IF YES, ENTER TPA NO. ______

SECURITY CLASSIFICATION

(CHECK ONE BOX ONLY)	AEC	DOD
UNCL	☒	☐
CONF.	☐	☐
SECRET	☐	☐

(CHECK ONE BOX ONLY)
RESTRICTED DATA ☐
DEFENSE INFO. ☐

AUTHORIZED CLASSIFIER ______ DATE ______

APPROVALS DATE

A. W. Graves A.W. Graves 6-24-75
M. E. Remley M E Remley 6-24-75
W. F. Heine W Heine
B. F. Ureda B.F. Ureda 6-25-75
P. F. Higgins P F Higgins 6/24/75

DISTRIBUTION

*	NAME	MAIL ADDR
	A. W. Graves	NB02
	W. F. Heine (10)	NB02
	P. F. Higgins	KB45
	C. R. Johler	LA38
	W. R. McCurnin	T020
	G. P. Streechon	NB02
	R. J. Tuttle	NB13
	B. F. Ureda	NB02
	J. H. Walter	T009
	J. H. Moore	NB13

mlm

* TITLE PAGE ONLY

ABSTRACT

The partially dismantled Sodium Reactor Experiment (SRE) will be further dismantled and decontaminated. Sodium systems will be removed, residual sodium chemically passivated, bulk primary sodium stored, components scrapped or buried. Reactor vessels will be removed remotely. Other contaminated components will be cleaned or buried. Facilities will be decontaminated to allow unrestricted use.

RESERVED FOR

FORM 734-C REV. 2-74

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

SRE DISMANTLING PLAN

1.0 Objective

The partially dismantled Sodium Reactor Experiment (SRE) located at AI, Santa Susana, contains extensive radioactive materials in the form of activated and contaminated structures and components. The SRE is inactive and "Stored in Place." The SRE fuel, Cores I and II, is stored in the vault at the Radioactive Materials Disposal Facility (RMDF) at AI, Santa Susana.

The dismantling of the SRE will culminate in the removal of all significant reactor originated radioactivity from the site and the release of the facility from all requirements for radiological control, licensing, or monitoring. The surfaces of all areas which remain following completion of dismantling, and of all material released for unrestricted use, will be decontaminated to levels which are as low as practicable but, in all cases, below those in Table 1. Acceptable levels of residual induced radioactivity, in terms of specific activities, are being developed for application to the excavation requirements. These levels will be described in the appropriate Activity Requirements documents. Materials which are contaminated or activated to levels greater than the acceptable levels will be packaged and transported to a licensed burial site for disposal by land burial.

The contamination limits shown in Table 1 were developed by AI. In some cases these limits are lower than current NRC and ERDA guidance due to an attempt to make the limits compatible with the requirements of the State of California Bureau of Radiological Health and the AI State of California Broad Radioactive Materials License, and overall AI internal operating limits. The SRE is sited on property owned by Rockwell International Corporation, which ERDA has an option to purchase. As a facility located on ERDA-optioned land, the SRE is exempt from the licensing requirements and regulations of NRC or the State of California. However, should ERDA at some future date relinquish the option, the property would be subject to the jurisdiction of the

State of California Bureau of Radiological Health and the requirements of the AI State of California Broad Radioactive Materials License.

All excavations resulting from the decontamination and dismantling will be filled to grade with clean earth or with clean rubble topped with clean earth.

TABLE 1

CONTAMINATION LIMITS FOR DECONTAMINATION AND DISPOSITION OF THE SRE

	Total	Removable
Beta Gamma Emitters	0.1 mrad/hr at 1 cm with 7 mg/cm^2 absorber	100 dpm/100 cm^2
Alpha Emitters	100 dpm/100 cm^2	20 dpm/100^2

2.0 SRE History

2.1 Plant Description

The SRE plant is located at the Atomics International Nuclear Development Field Laboratory, in the southeastern portion of Ventura County, California, about 30 miles from downtown Los Angeles. The field laboratory is situated in rugged terrain on an elevated rocky plateau nearly 1,000 ft above the surrounding valleys and is bordered on three sides by the high surrounding Simi Hills. The reactor is sited on property owned by Rockwell International Corporation, which the U. S. Government (ERDA) has the option to purchase. Figures 1 and 1A show the location of the SRE in the Field Laboratory site, and Figure 2 shows the SRE Reactor Building Plan.

The SRE was a sodium cooled, graphite moderated, thermal reactor using slightly enriched uranium fuel in Core I and 93% enriched fuel in Core II. During operation of the plant, two core loadings of fuel were employed: Core I of uranium-metal, and Core II of uranium-thorium metal

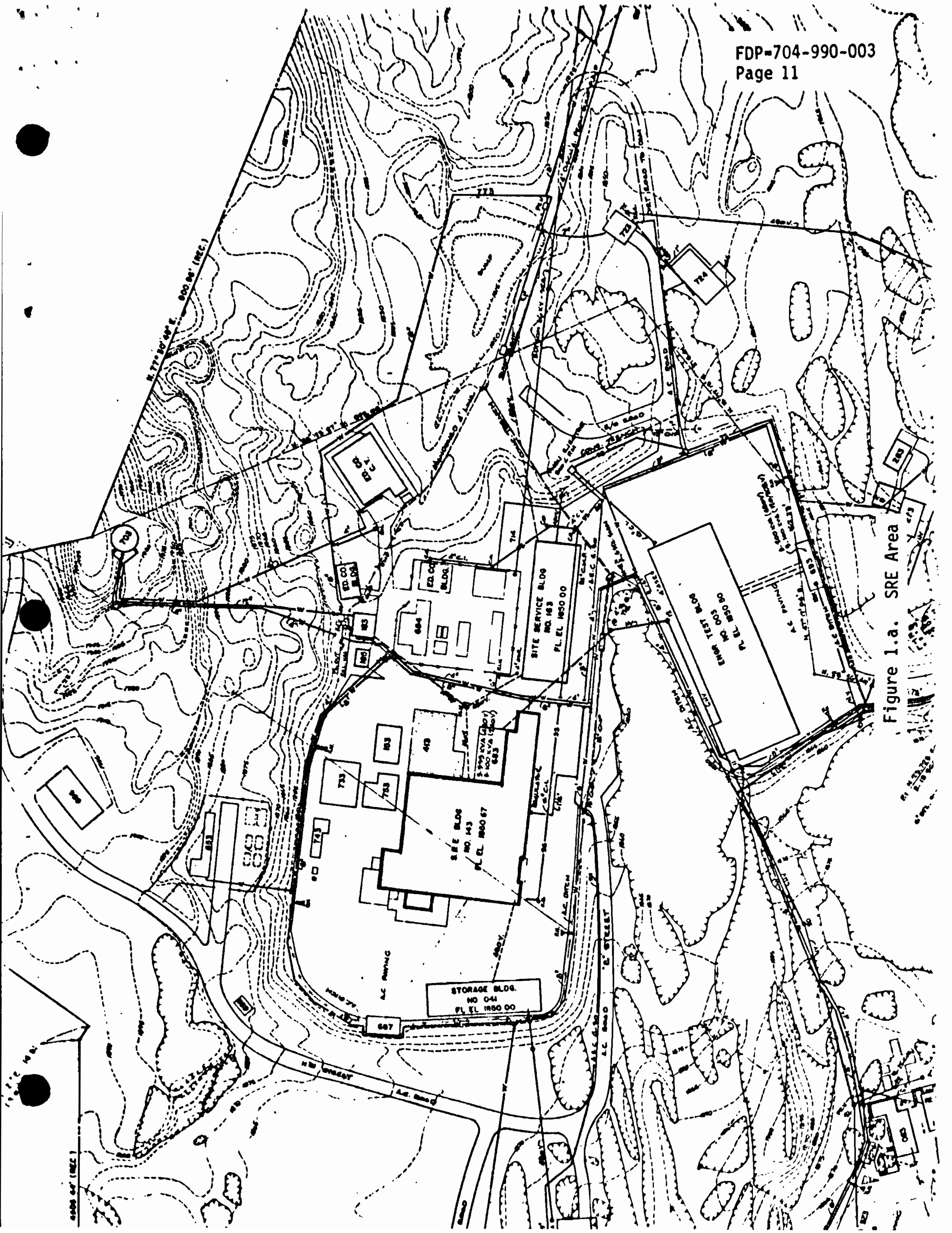


Figure 1.a. SRE Area

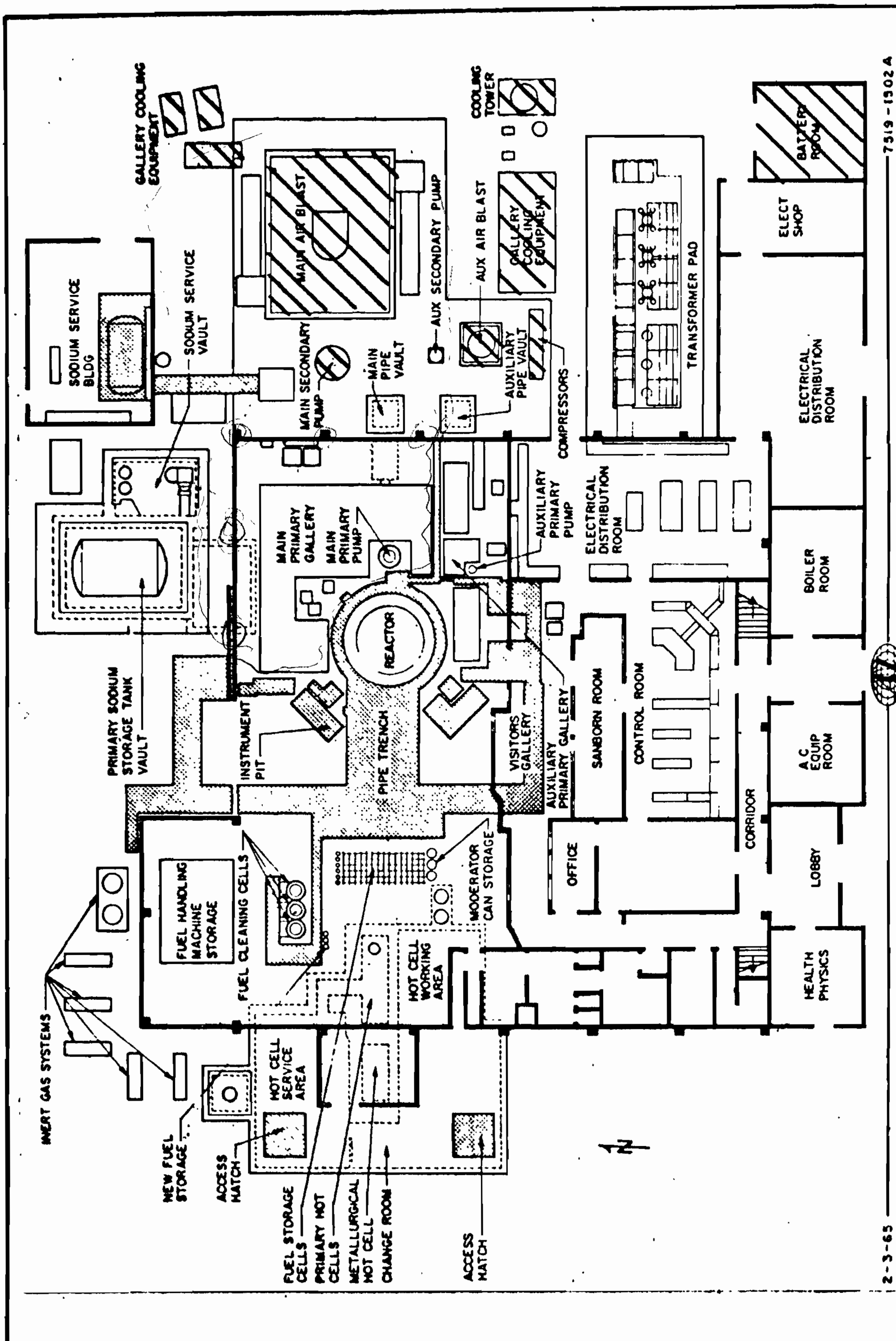


Figure 2. SRE Reactor Building Plan

= Equipment Removed

alloy. The fuel was in the form of stainless steel clad rods with NaK bonding in the annulus between the fuel and cladding. The active core length was 6 ft. Fuel elements hung from the plugs in the reactor top shield in channels at the center of hexagonal zirconium clad graphite moderator elements.

2.2 Operating History

Initial operation of the SRE began in April 1957. The power output histogram, with significant events noted, is presented in Figure 3 for the span of time through February 15, 1964. The SRE primary system hot leg thermal history for several temperature ranges is accumulated in Table 2 for Core I and Core II. Most of the sodium system was in service through both core operations. In Figure 4, the 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 May 15, 1965, to September 1967, included operation of the Main Primary Sodium for 4,386 hours at approximately 700°F, and 13,196 hours at approximately 350°F.

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

The SRE generated more than 37 million kilowatt hours of electrical energy in over 27,300 reactor operating hours. A summary of the more important operating statistics is presented in Table 3.

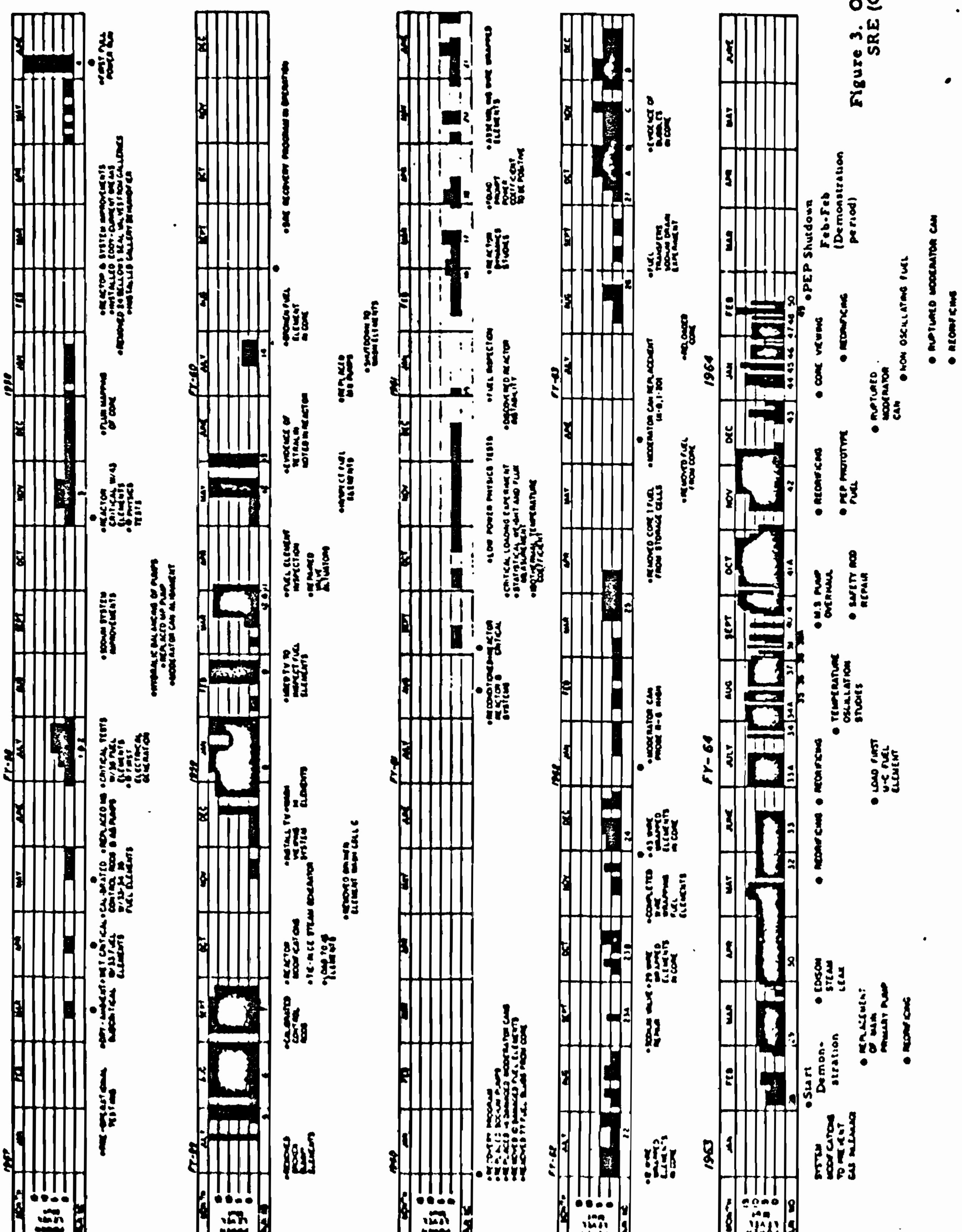


Figure 3. Operating History of SRE (Core I and II)

TABLE 2

SRE HOT LEG OPERATIONAL TIME AND TEMPERATURE*

Temperature Range (°F)	Time (hr) Core I	Time (hr) 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

*Core I - May 4, 1958 to November 10, 1959
Core II - July 22, 1960 to February 15, 1964

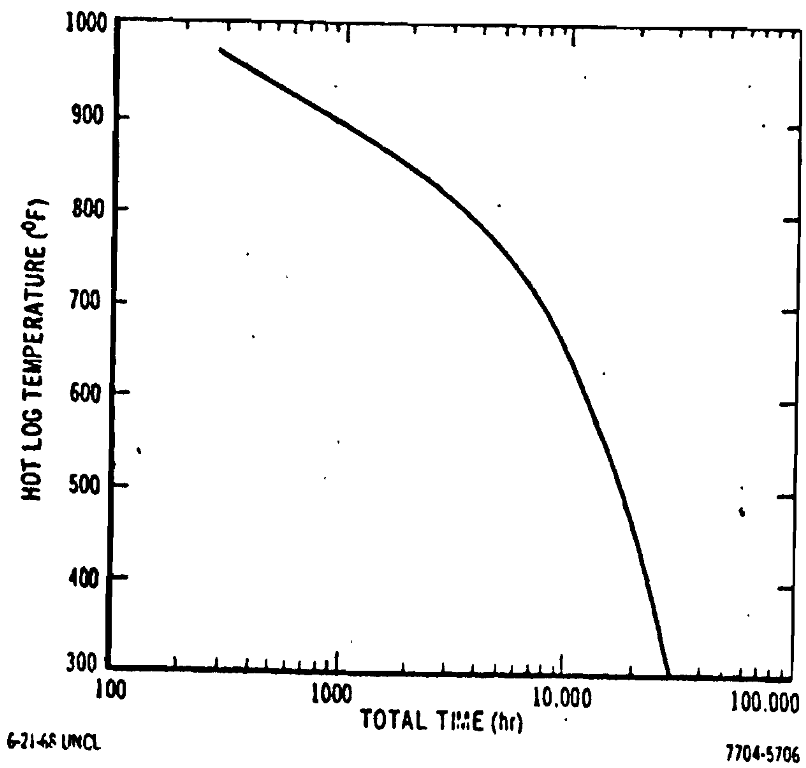


Figure 4. Total Time At or Above a Hot Leg Temperature

TABLE 3

SRE OPERATING STATISTICS

Reactor Critical (hr)			27,300
Integrated Thermal Reactor Power (Mwd)			6,700
Integrated Electric Output (kw-hr)			37,174,200
Primary Pumps (hr)			
	Main	Original (Freeze Seal)	37,060
		"PEP" (Free Surface)	17,582
	Auxiliary	Original (Freeze Seal)	37,060
		"PEP" (Free Surface)	15,241
Secondary Pumps (hr)			
	Main	Original (Freeze Seal)	24,760
		"PEP" (Free Surface)	11,442
	Auxiliary	Original (Freeze Seal)	41,152
		"PEP" (Free Surface)	17,881
Intermediate Heat Exchanger (hr)			
	Main	Original	37,060
		"PEP"	17,582
	Auxiliary		55,642
Steam Generators Sodium Filled (hr)			63,000
Steam Generators Steaming (hr)			30,392
PEP Operation (Primary and Auxiliary Na System Flow at ~350°F (hr)			17,582

2.3 Deactivation

A plan for the deactivation of the SRE (Reference 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 SIR steam generator and uncontaminated support facilities were not maintained. The principal activities of that deactivation are described below.

2.3.1 Removal of property requested by other AEC contractors.

2.3.2 Transfer of Core III fuel from the SRE to Building 064.

2.3.3 Draining of primary sodium to the fill tank.

2.3.4 Removal of secondary sodium from the SRE.

2.3.5 Modification of the inert gas system to combine the helium and nitrogen gas systems.

2.3.6 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 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.

2.3.7 Decontaminating the external surfaces of the fuel and moderator handling machines.

2.3.8 Shutting down control and instrumentation air and power.

2.3.9 Decontaminating the main and portable hot cells, and shutting down the ventilation systems.

2.3.10 Prepared the batteries, m-g sets, and diesel generator for inactive period.

2.3.11 Providing power for the emergency paging system and perimeter lights.

2.3.12 Shutting down heating, ventilating, and plant air systems.

The above activities, reported in Reference 2, were completed in 1968. A surveillance program conducted by the Facilities Engineering Department was initiated. Surveillance 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.

3.0 Present Status of the Site

A physical inspection of the site and a review of all the SRE deactivation documents was made. The present condition of the facility is described in the following paragraphs. Radiological surveys were performed in 1966 about two years after the last nuclear operation. An updated survey of the critical areas will be made prior to initiation of the SRE dismantling.

3.1 Fuel Assemblies

The shield plug/hanger assemblies have been disassembled from the unirradiated Core III fuel. Core III fuel has been shipped from the site. All fissile material from Cores I and II is stored in the fuel storage vault at the Radioactive Materials Disposal Facility (RMDF), also located at the Santa Susana site. Disposition of this fuel will be discussed in another planning document (Reference 3).

3.2 Core Components

The core components described in Table 4 are stored in the core positions of the reactor indicated on the core loading face diagram, Figure 5.

The following items are stored in the Safety and Control Rod Storage Rack in the Building 143 high bay.

(1) Safety rod tower and drive assembly, 5 units

(2) Control rod tower and drive assembly, 4 units, plus 2 spare drive units.

The long shield plugs from Core III fuel are stored in the storage cells in the SRE high bay floor area. The numbers are marked on the floor cover plates.

3-inch diameter shield plugs: Nos. 4, 9, 11, 16, 17, 23, 24, 36, 40, 41, 46, 60, 61, 66, 73, 77, 84.

3.5-inch diameter shield plugs: Nos. 5, 26, 27, 32, 33.

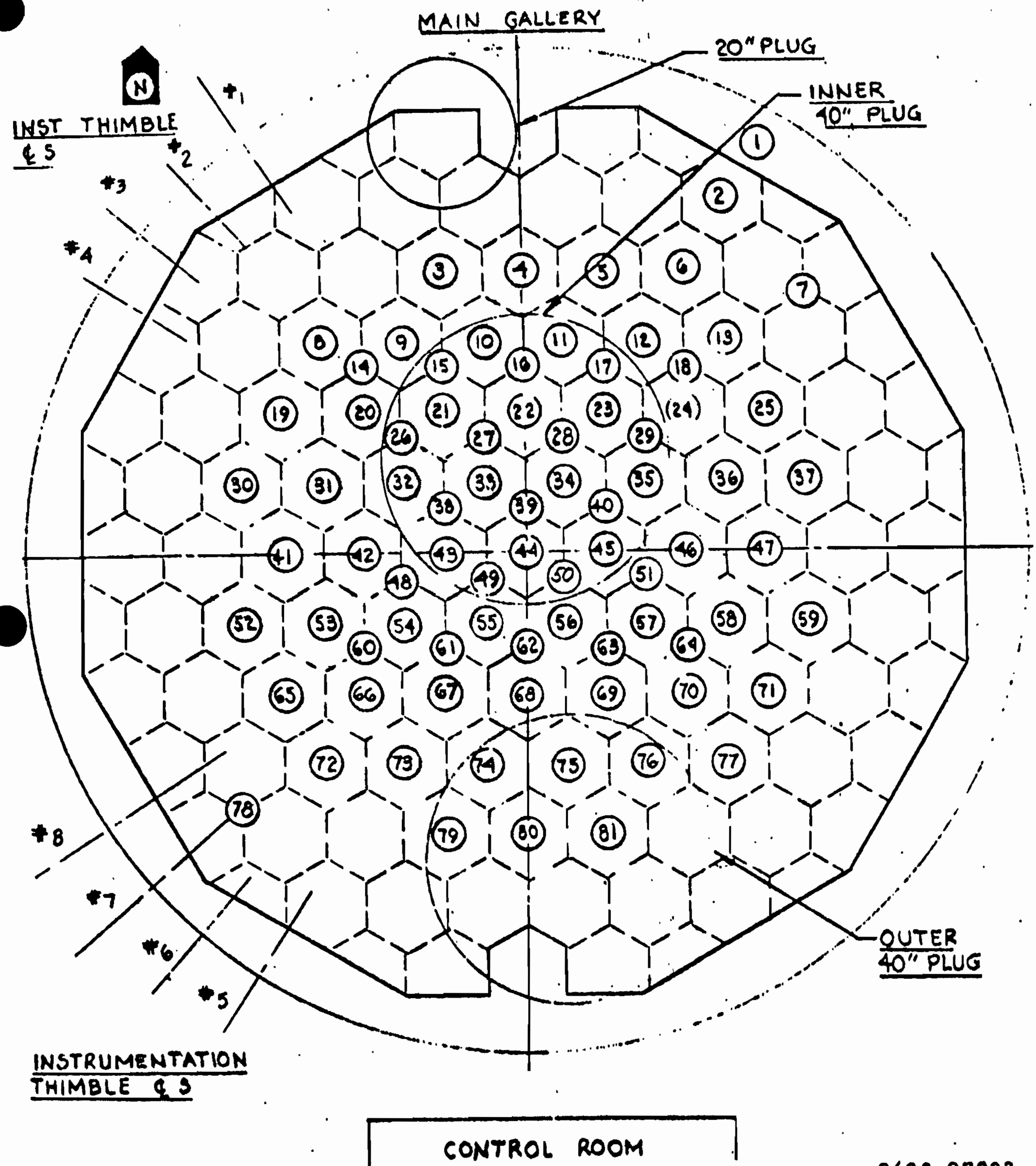
During the SRE-PEP program most core components (Table 4) were replaced with new components. Some were modified and some are the originals. The upper portion of the Dummy Fuel Elements is original. The lower sections were modified by replacing the fuel simulating portion. The Pile Oscillator is an original component but the thimble was replaced. The Fission Monitor Plug and the Core II Shield Plug and Hanger Assemblies are original components. All of the components listed in Table 4 have been in direct contact with radioactive sodium and are therefore contaminated. In addition, original components have been exposed to the reactor flux and are activated. During the SRE deactivation program when the fuel plugs were removed from the core, radiation levels of 100 mR/hr were measured on the plugs.

TABLE 4

STORAGE LOCATION OF CORE COMPONENTS IN THE REACTOR

Item	Location
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 & 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

* Numbers refer to reactor (R) locations in Figure 5.



9693-97003

Figure 5. SRE Reactor Loading Face

3.3 Hot Cells

Contamination levels in the permanent A and B hot cells are below 500 dpm/100 cm^2 beta except for the two fuel storage thimbles. Beta contamination levels in the thimbles are less than 2500 dpm/100 cm^2.

Contamination levels on the internal surfaces of the demountable maintenance shielding assembly (DMSA, also known as the portable hot cell) in the fuel handling machine storage area are less than 5000 dpm/100 cm^2.

Contamination smear surveys on the interior areas of the ventilation ducts adjacent to each filter in the filter room show beta-gamma contamination levels as given in Table 5.

TABLE 5

ACTIVITY LEVELS OF SRE VENTILATING SYSTEM (1966)

Sample No.	Description and Location	β-γ Activity (dpm/100 cm^2)
1	West duct - upstream of filter (final)	2,334
2	West duct - downstream of filter (final)	129
3	Center duct - upstream of filter (final)	10,181
4	Center duct - downstream of filter (final)	1,293
5	East duct - upstream of filter (final)	756
6	East duct - downstream of filter (final)	423
7	East plenum floor (under filter)	1,953
8	Center plenum (under filter)	1,479
9	West plenum floor (under filter)	2,118

3.4 Sodium Systems

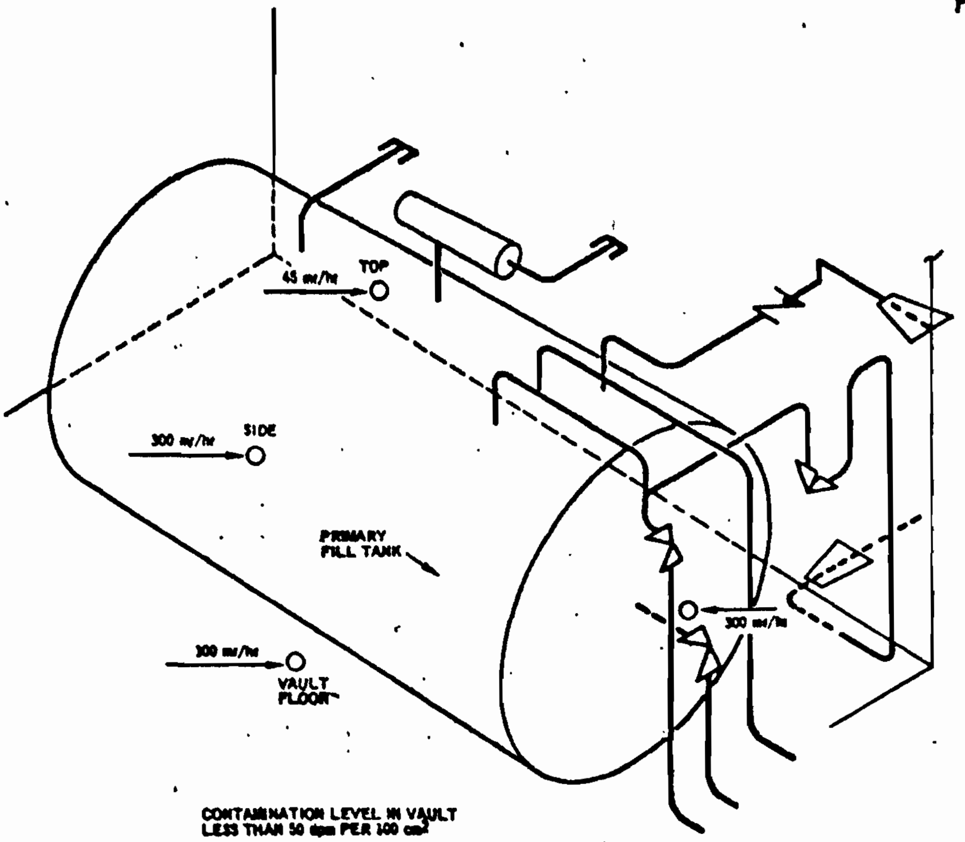
The sodium systems are drained, except for residual sodium heels throughout the system, and are at ambient temperature. All heaters have been turned off, but remain in operating condition.

Approximately 55,000 lb of primary sodium remains in the primary fill tank at ambient temperature. A map of the radiation levels from the tank and associated piping is shown in Figure 6. The contamination levels on the surfaces of the vault are less than 50 dpm/100 cm^2. (β) Figures 7 and 8 are the radiation maps for the main and auxiliary sodium systems. Figure 9 is a radiation map for the sodium services piping in the sodium service vault. The man-way plugs have been removed in the following areas: primary fill tank vault; sodium service vault; and primary drain vault, to allow access for surveillance. The main and auxiliary galleries have recently been opened by removing the shield plugs to permit a radiological resurvey and to prepare for dismantling.

3.5 Radioactive Liquid Waste System

The contamination and radiation levels present in several locations within this liquid waste system and in areas associated with this system are indicated in Figure 10. All liquid waste, except inaccessible heels, was removed.

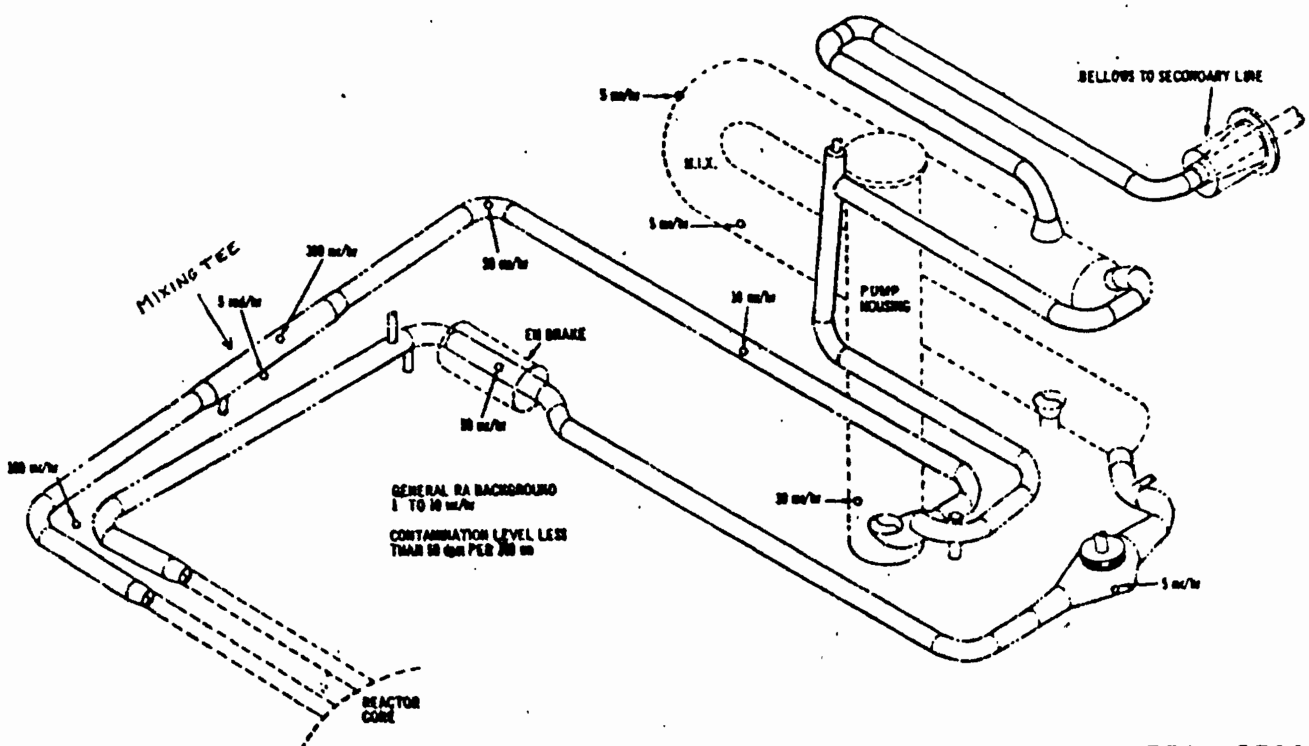
The two 5,000-gallon storage tanks in the obsolete liquid waste system are contaminated to levels of less than 5,000 dpm/100 cm^2. Figure 11 shows these tanks and the radioactive gaseous waste tanks during the construction stages. Figure 12 shows the mild steel pipe-runs to these tanks (both liquid and gaseous waste). Figure 9 presents contamination levels in the various liquid waste lines. Sometime during the operating history of the SRE, the obsolete liquid waste tanks apparently developed leaks, which resulted in soil contamination in the immediate vicinity of the tanks and along rock outcroppings in the cut bank below them.



5-14-68

7704-5705

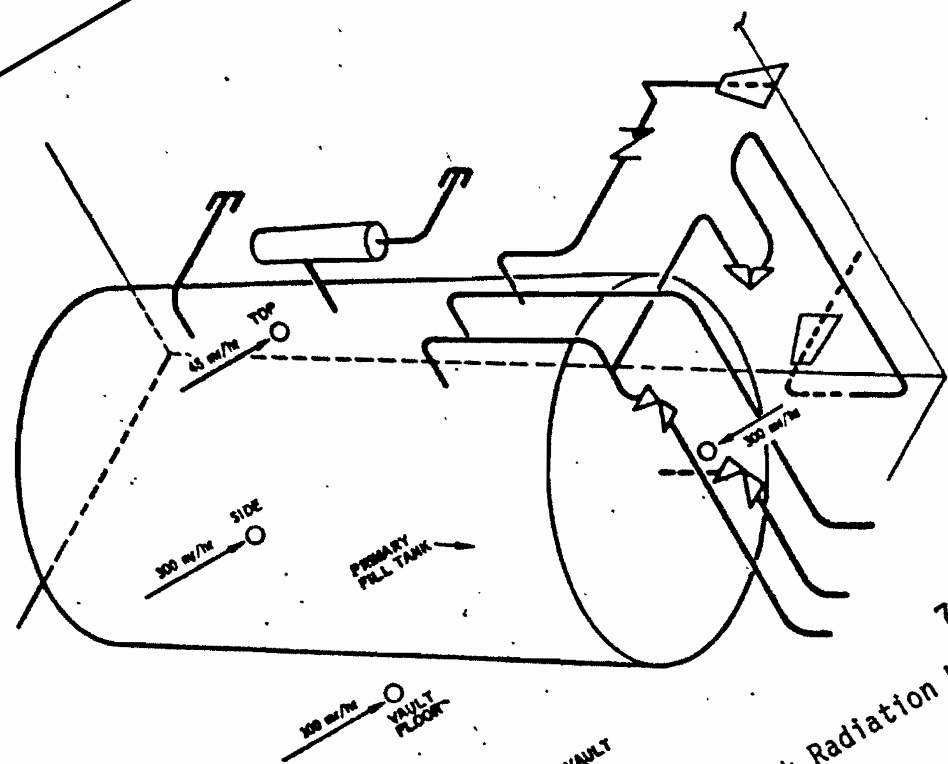
Figure 6. Primary Fill Tank Radiation Map



5-14-68

7704-5703

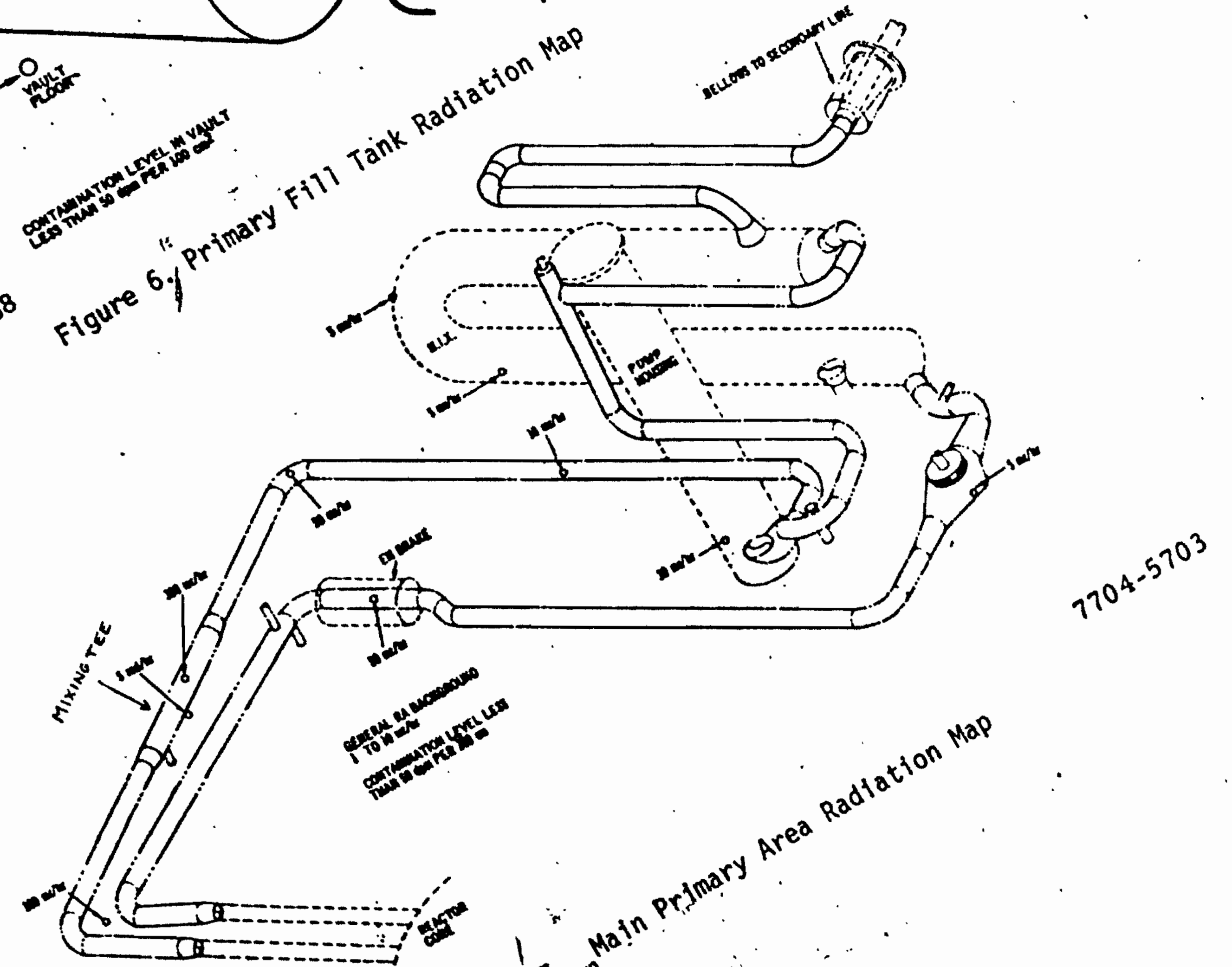
Figure 7. Main Primary Area Radiation Map



7704-5705

Figure 6. Primary Fill Tank Radiation Map

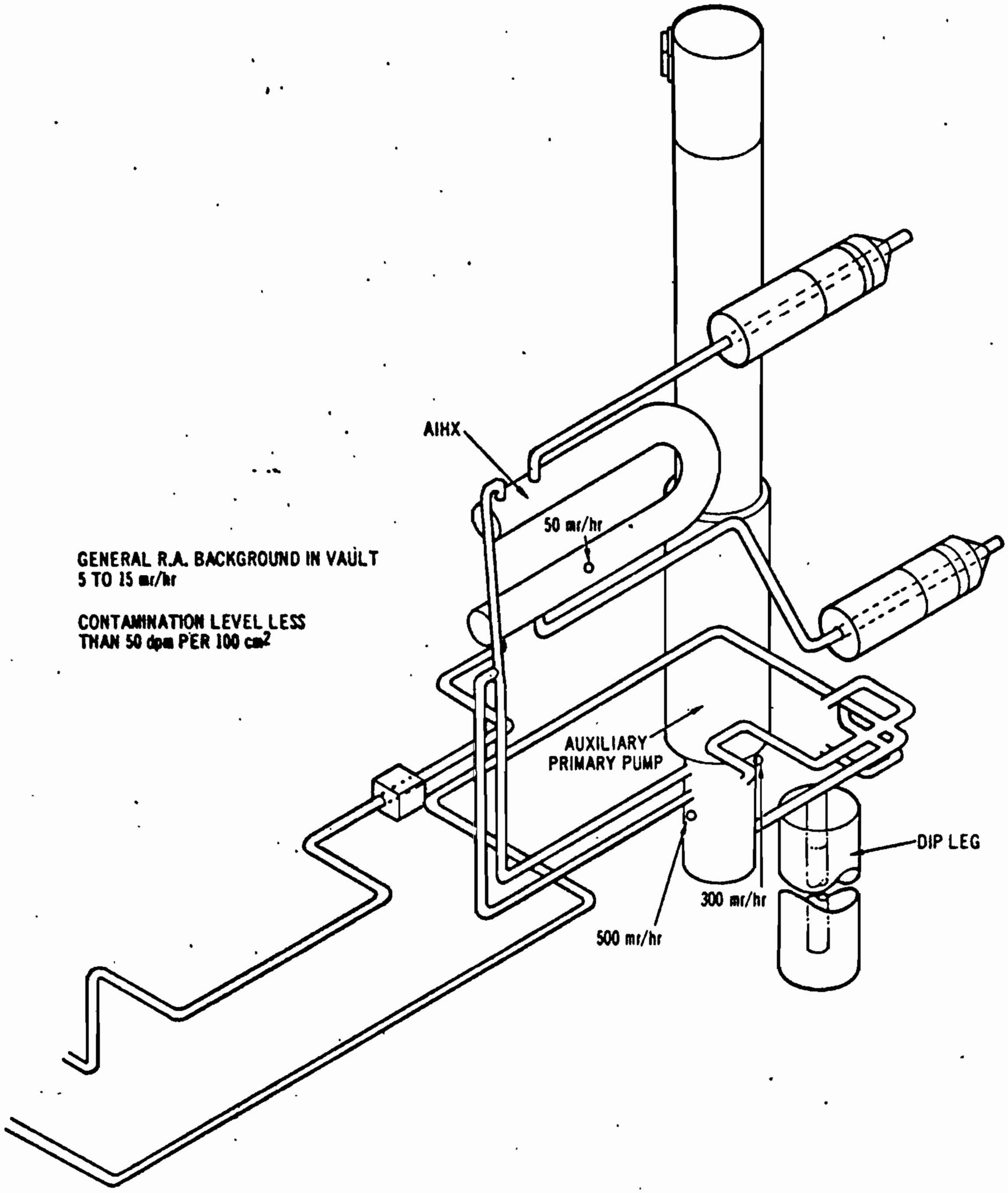
5-14-68



7704-5703

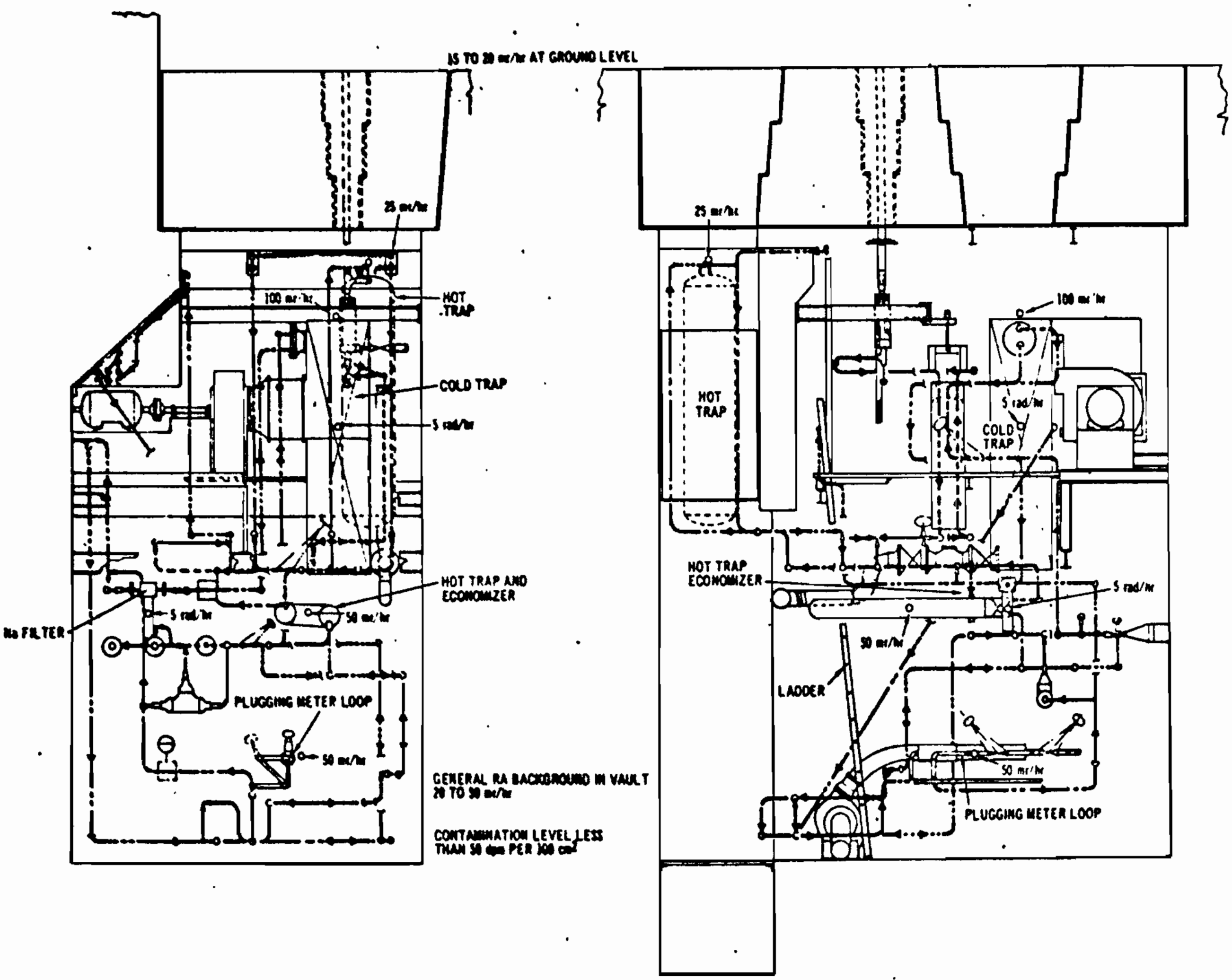
Figure 7. Main Primary Area Radiation Map

5-14-68



5-14-68 7704-5702

Figure 8. Auxiliary Primary Area Radiation Map



5-14-68 7704-5701

Figure 9. Primary Sodium Service Vault Radiation Map

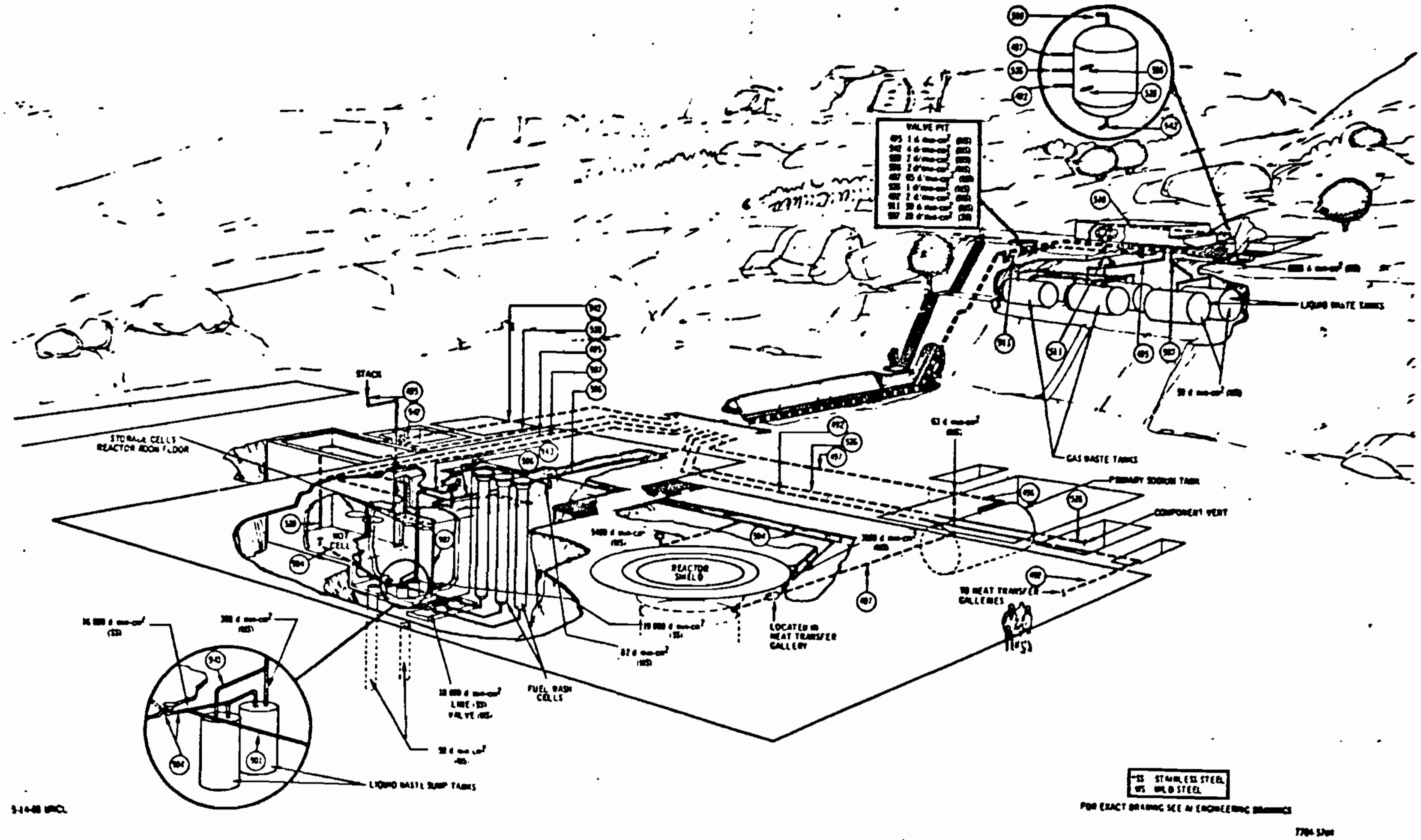
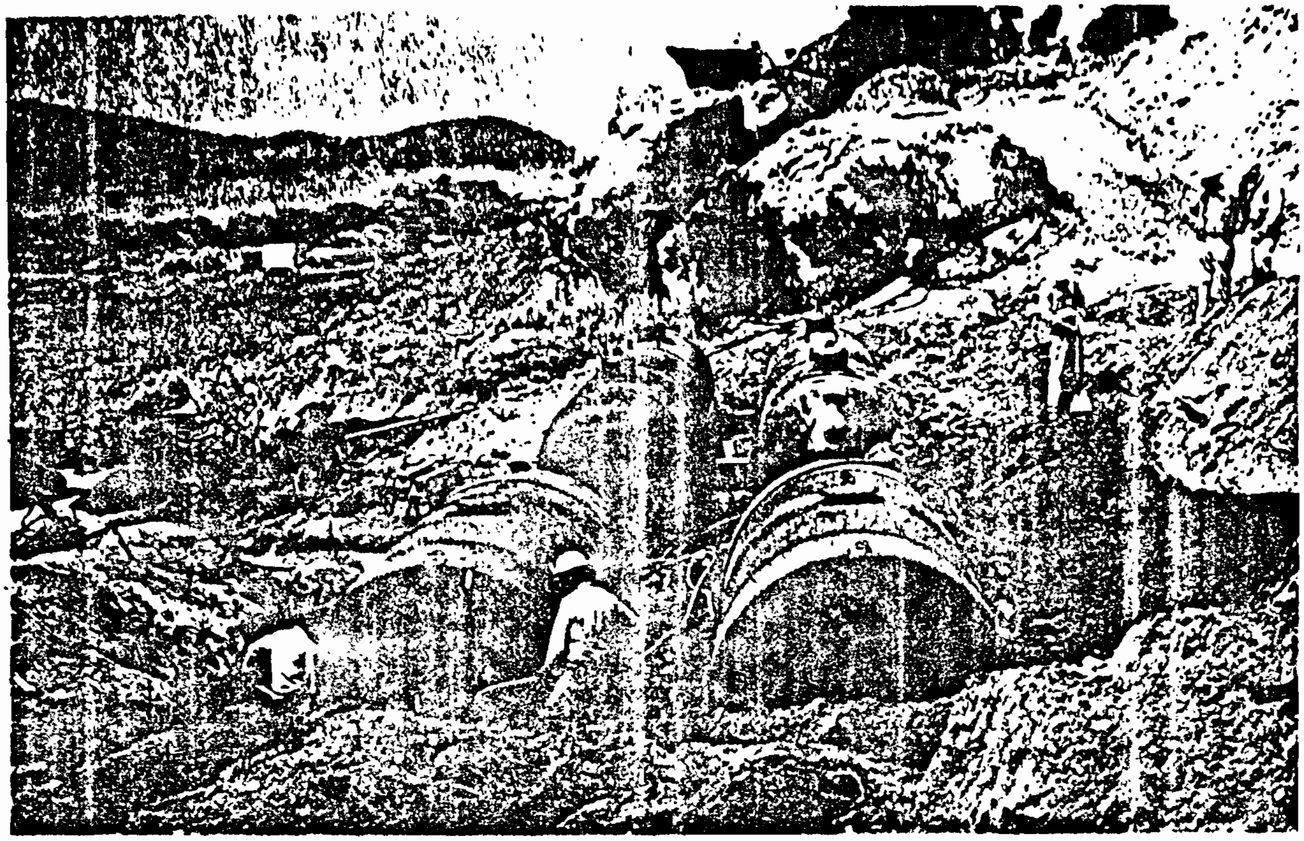
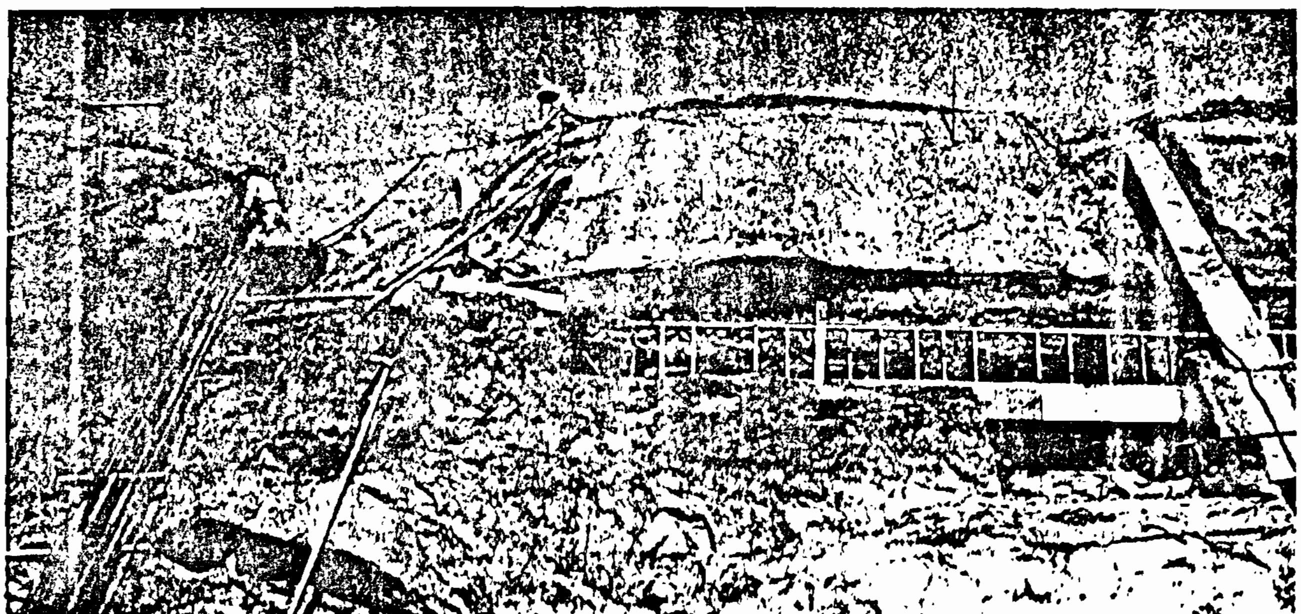


Figure 10. R/A Liquid Waste and Vent Systems Contamination Levels



6-20-56 9693-57131A

Figure 11. Arrangement of R/A Liquid Waste and Vent Systems Tanks (Construction Phase)



11-6-56 9693-58216B

Figure 12. R/A Liquid Waste and Vent Systems Pipe Runs (Construction Phase)

The change room holdup tank is contaminated to levels of less than 500 dpm/100 cm^2. (β)

3.6 Radioactive Gaseous Waste System

This system was purged with fresh nitrogen gas to reduce the contamination levels. The levels of activity present are indicated in Figure 10. The compressor vault is radiologically clean. The four gaseous waste storage tanks, which are shown in Figure 11 during construction, are leak tight and were sealed off under a nitrogen pressure of ¼ psig. The pipe-runs to these tanks are shown in Figure 12.

3.7 Component Handling Machines

The Mark I and II fuel handling machines are stored in the Fuel Handling Machine Storage Bay. The β-γ contamination present on this equipment at the exposed surfaces is shown in Table 3. The internal surfaces of the Handling Machines are highly contaminated.

The Moderator Handling Cask and support equipment are stored in the SRE high bay. Contamination levels on the outside surfaces of the cask are shown in Table 6. The internal surfaces are highly contaminated.

The equipment necessary to reactivate the component handling machines is stored nearby. Planning has been included for reactivation of those machines, if required for removing the various core components stored in the reactor.

3.8 Fuel Moderator and Pump Storage Cells

These cells are empty except for the plugs used to seal the entrances. The plugs in the fuel storage cells are the plugs removed from the Core III fuel assemblies. The cells were exposed during the reactor operation to a number of ruptured fuel assemblies and, except for the reactor vessels, represent the most contaminated area on the site. Table 7 is a tabulated record of the contamination levels at the time the cells were sealed.

The fuel cleaning cells are sealed. The floor trenches surrounding the cleaning cells are equipped with temporary lead shielding. This area is reported to have a removable contamination level of about 1000 dpm/100 cm^2, and fixed contamination levels of greater than 200 mrad/hr. During the nuclear operation period of the reactor, the center cell experienced an explosion while a spent fuel assembly was being cleaned. The cell and the surrounding floor area were contaminated. The earth surrounding the lower regions of the cell may also have been contaminated, as indicated by the presence of slightly contaminated water in the wash cell valve pit.

3.9 Peripheral Areas

The west end of Building 163, the Contaminated Equipment Repairs Facility (CERF), is contaminated to levels shown in Table 8. Most of this contamination is imbedded in the floor and wall construction materials and has been fixed in place by painting over the contaminated surfaces. Table 8 indicates the level of contamination of the painted surfaces. Radiation measurements of the floor area range from .5 to 5 mR/hr.

Building 724, the SRE Oil Cleaning Facility, was used extensively to remove contaminated sodium from pipes and miscellaneous sodium equipment. This facility has been cleaned to the contamination levels shown on Table 9.

TABLE 6

CONTAMINATION LEVELS OF FUEL HANDLING MACHINES (1966)

Sample No.	Description and Location	Removable β-γ Contamination (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 - Bio-Shield	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

TABLE 7

CONTAMINATION LEVELS (1966) IN SRE BUILDING (143)

Storage Cells No.	β-γ Activity (dpm/100 cm^2)	No.	Removable β-γ Contamination (dpm/100 cm^2)
-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	β-γ Activity (dpm/100 cm^2)
A	1,980
B	1,440
C	900

Pump Storage Cells	β-γ Activity (dpm/100 cm^2)
East	789
West	500

High Bay Floor

Maximum β-γ level is 75 dpm/100 cm^2 with an average of 50 dpm/100 cm^2.

TABLE 8

CONTAMINATION LEVELS OF CERF, BUILDING 163 (1966)*

Description and Location	Removable β-γ Contamination (dpm/100 cm^2)	Description and Location	Removable β-γ Contamination (dpm/100 cm^2)
South Floor (west)	< 30	West Wall (south)	30
South Floor (center)	< 30	West Wall (center)	30
South Floor (east)	< 30	West Wall (north)	87
Center-Floor (west)	< 30	Light Fixtures (N-E)	187
Center-Floor (center)	112	Light Fixtures (N-W)	112
Center-Floor (east)	< 30	Light Fixtures (W)	70
North Floor (west)	< 30	Light Fixtures (E)	87
North Floor (center)	< 30	Light Fixtures (S-E)	30
North Floor (east)	87	Light Fixtures (S-W)	30
East Wall (north)	87	Top of Supply Room	
East Wall (center)	87	Overhead Crane Rails	300
East Wall (south)	< 30	(south)	266
North Wall (west)	< 30	(south)	252
North Wall (center)	< 30	(north)	294
North Wall (east)	130	(north)	185
South Wall (west)	< 30		
South Wall (center)	< 30		
South Wall (east)	< 30		

*Fixed contamination levels on the floor surface of this facility range up to 5 mrad/hr.

TABLE 9

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

Description and Location	Removable β-γ Contamination (dpm/100 cm^2)
Outside Areas	<30
Inside Areas	
- Floor (N-E)	150
- Floor (N-W)	<30
- Floor (S-E)	110
- Floor (S-W)	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

3.10 Reactor Cavity

The most highly contaminated part of the facility is the reactor vessel. No records of measured radioactivity are available. It is anticipated that several feet of water shielding will be required during the removal operation. This facility was left intact with approximately two-inch radioactive sodium heel on the bottom of the vessel. Sodium is also adhering to the sides and top of the moderator cans and other equipment stored in the vessel. Since heaters were turned off, the vessel and cavity are now at room temperature.

4.0 Activated Structures and Components Radioactivity Inventory

4.1 Introduction

The level of radioactivity and significant radionuclides present in the neutron activated structures and components including all the reactor structural components and the biological shielding is required for definition of the decontamination and dismantling work scope. A detailed study was therefore performed to evaluate the levels of radioactivity and to identify the significant radionuclides which remain in these components and structures. The SRE internals are contained as shown in Figure 23 in concentric structures consisting of:

1.	Stainless steel core tank liner	1/4" thick
2.	Stainless steel core tank vessel	1 1/2" thick
3.	Steel thermal rings	5 1/2" thick
4.	Steel outer tank	1/4" thick
5.	Insulation	12" thick
6.	Steel core cavity liner	1/4" thick
7.	Concrete shielding	48" thick

The fuel has been removed from the reactor. The moderator elements were replaced with new elements during the SRE-PEP program and thus have not been exposed to neutron irradiation. Certain dummy elements, control elements, and instrument elements which remain in the core may have been exposed to neutron irradiation. No evaluation of the radioactivity of the removable core components was performed.

4.2 Assumptions

4.2.1 Nuclear Power History

A reactor operating history of one year at 9.2 Mwt, ending February 1959 for Core I, and one year at 9.2 Mwt, ending July 1964 for Core II, was developed by integrating the power levels in the power output histogram as shown in Figure 3. The amount of induced radioactivity present was evaluated for a point in time 16 years after completion of the first period of operation, which corresponds roughly to April, 1975. Neutron activation fluxes were derived from Figures 13 through 16, which were taken from Reference 4. The material, thickness, total weight, assumed activation fluxes, and thickness averaging factors are described in Table 10 for each reactor component or structure considered. The radioactivity present in objects as a result of neutron activation may be determined from the following equation:

$$A = \frac{\phi\sigma N}{K} (1 - e^{-\lambda T})\, e^{-\lambda t}, \qquad (1)$$

where

A = Radioactivity of product isotope (curies)
ϕ = Activation flux n/cm^2/sec),
σ = Nuclear cross section (cm^2/atom),
N = Number of atoms of target isotope,
T = Reactor operating time (years),
t = Decay time following shutdown (years),
K = dis/sec/curie, and
λ = Decay constant for product isotope (yr^{-1}).

In the evaluation of the activation of components and structures during operation of the SRE reactor for the second one-year period of operation, Equation 1 may be reduced to the following:

$-\lambda I$ II

$$A = 1.6 \times 10^{-5} \frac{\phi E I \sigma(b)}{W} (1 - e^{-\lambda 1})\, (e^{-\lambda 11}) \qquad (2)$$

where

A = Specific activity (μCi/gm),
E = Abundance of target element in irradiated object (fraction),
I = Isotopic abundance of target isotope (fraction),
W = Atomic weight of target isotope (grams/gram-atom),
σ(b) = Nuclear cross section (barns/atom), and
ϕ = Neutron flux (n/cm^2/sec).

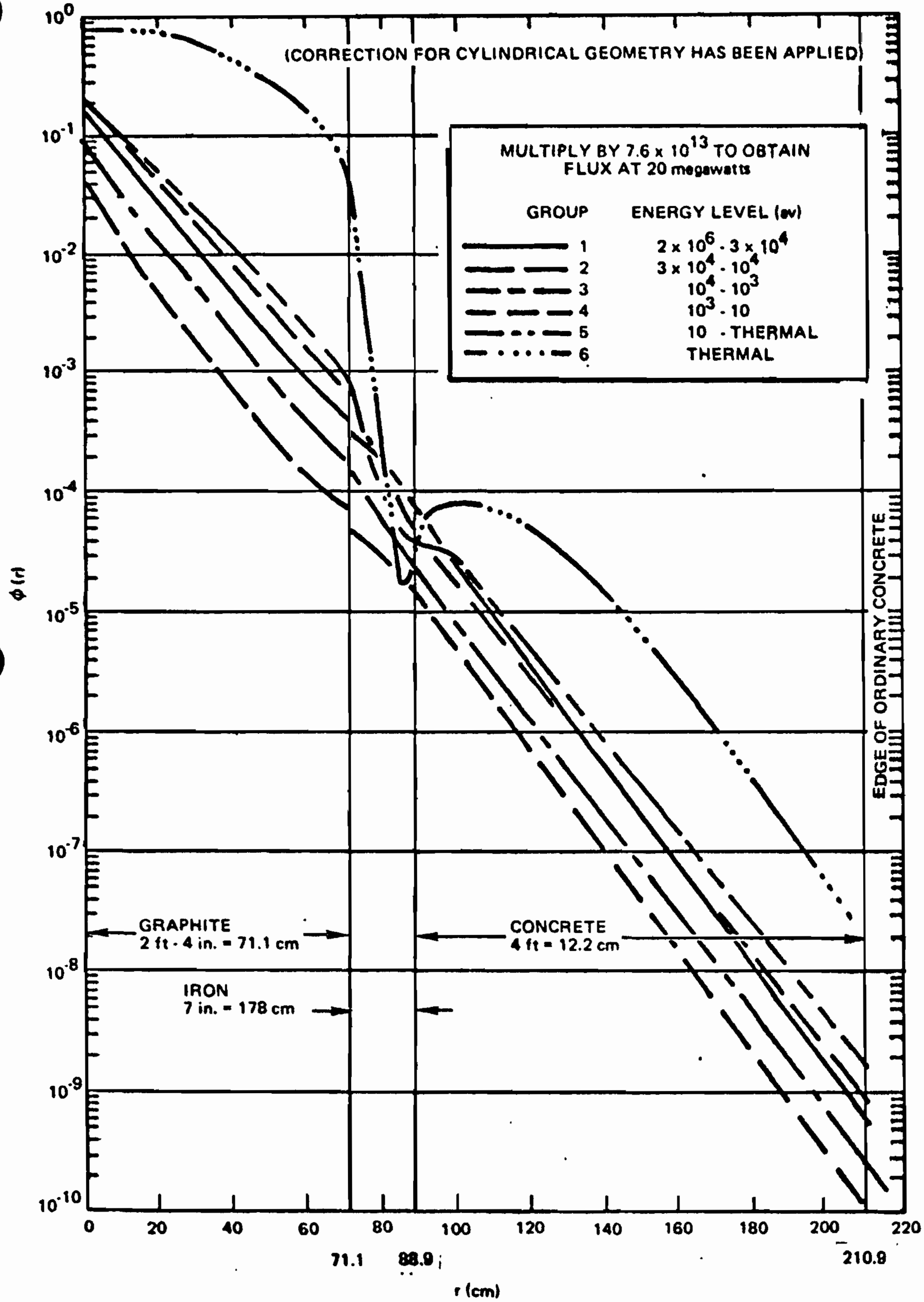


Figure 13. Calculated Radial Flux Normalized to Unit Thermal Flux at Center of Core

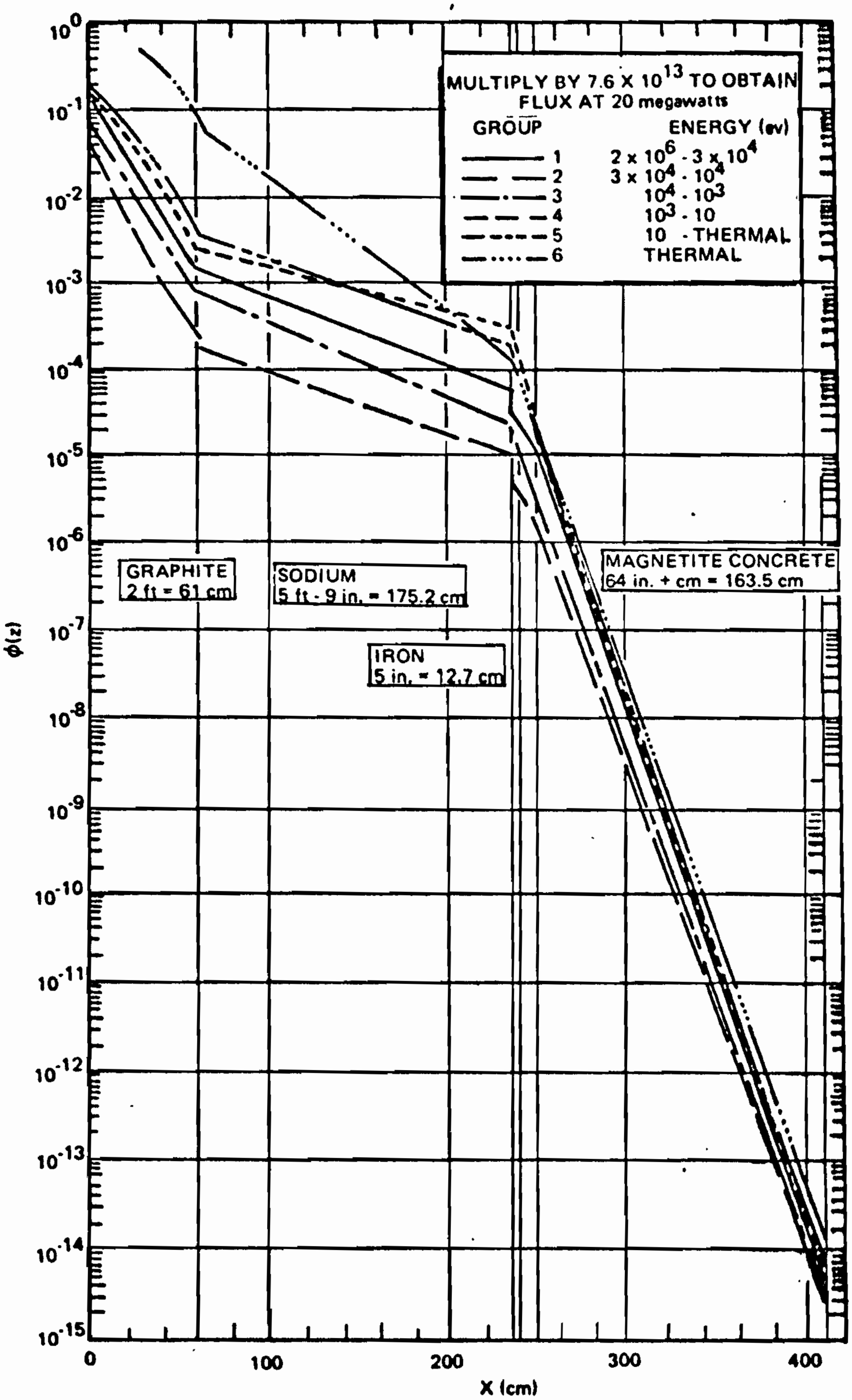


Figure 14. Calculated Axial Flux Above Reactor Normalized to Unit Thermal Flux at Center of Core

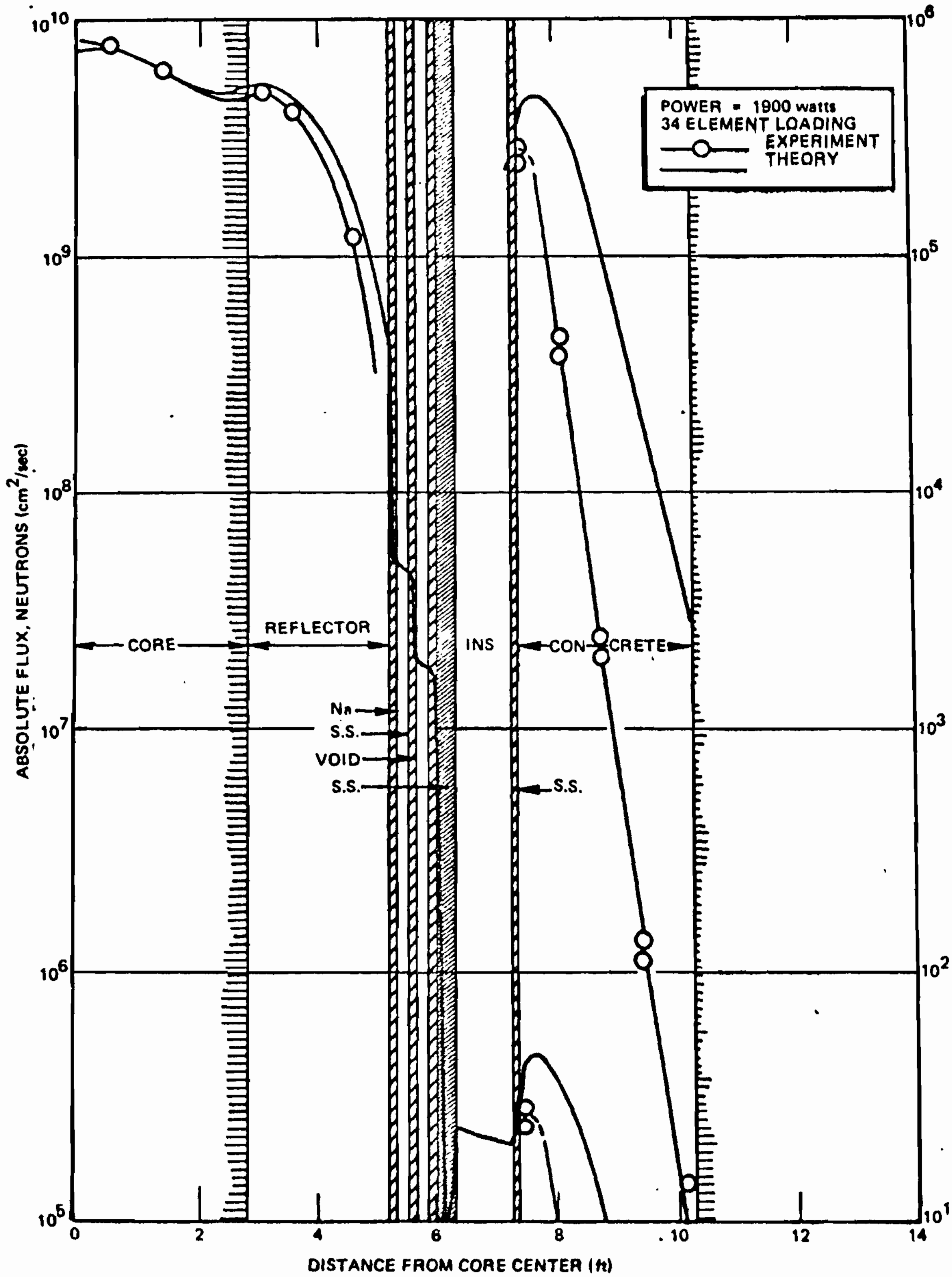


Figure 15. Radial Thermal Flux in SRE

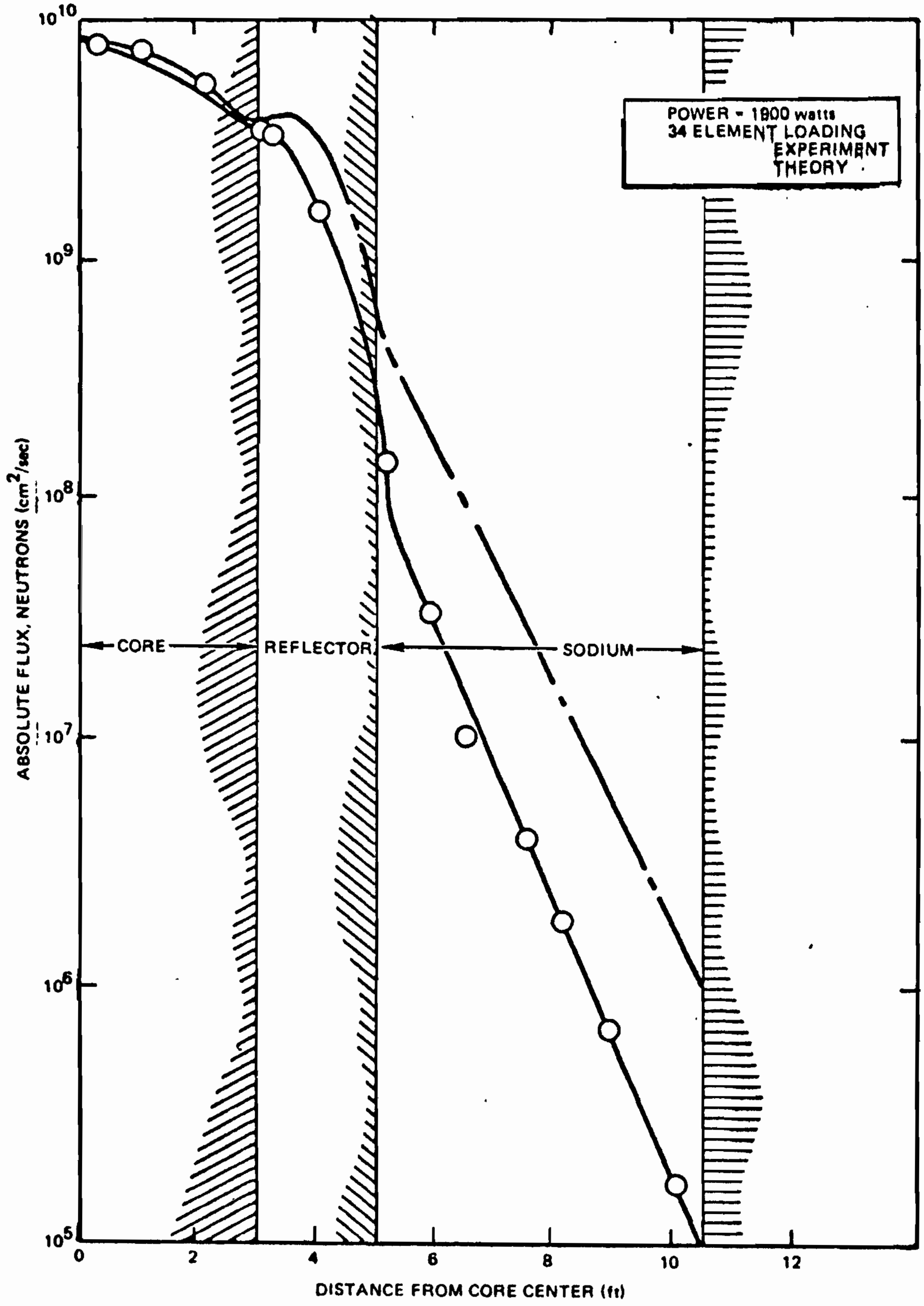


Figure 16. Axial Thermal Flux in SRE

The activation resulting from the first period of operation may be evaluated similarly by substituting 16 years cooling for the eleven years cooling.

4.2.2 Materials

Irradiated SRE components and structures are composed of Type 304 stainless-steel, carbon-steel, Superex insulation, ordinary concrete, and magnetite concrete.

TABLE 10

ASSUMED ACTIVATION FLUXES FOR SRE COMPONENTS AND STRUCTURES

	Material	Thickness (inches)	Weight (gm)	Max th Ø (n/cm^2-sec)	Max Total Ø (n/cm^2-sec)	Thickness Averaging Factor*
Internal Piping	304 SS	1/8 to 1/4 wall	1.7×10^6	2.2×10^{12}	2.6×10^{12}	0.89
Vessel Liner	304 SS	1/4	2.7×10^6	2.2×10^{12}	2.6×10^{12}	0.89
Grid Plate	304 SS	1 1/2	2.5×10^6	1.1×10^{12}	1.3×10^{12}	0.60
Support Columns	304 SS	3 Dia	6.9×10^5	4.5×10^{11}	5.0×10^{11}	0.60
Core Tank	304 SS	1 1/2	2.2×10^7	4.5×10^{11}	5.0×10^{11}	0.60
Thermal Shield	C Steel	5 1/2	2.6×10^7	1.5×10^{11}	2.0×10^{11}	0.24
Outer Tank	C Steel	1/4	1.3×10^7	6.0×10^8	8.4×10^9	0.89
Insulation	Superex	12	8.7×10^6	6.9×10^8	1.0×10^{10}	1.00
Cavity Liner	C Steel	1/4	8.6×10^6	6.9×10^8	1.0×10^{10}	0.89
Cooling Coils	C Steel	1/8 wall	1.9×10^6	3.0×10^9	6.6×10^9	0.89
Reinforcing Bars	C Steel	7/8	2.3×10^5	3.0×10^9	6.6×10^9	0.84
Top Shield Insulation	304 SS	1/2	8.4×10^5	5.4×10^9	2.6×10^9	0.84
Top Shield Plate	304 SS	1	1.8×10^6	3.6×10^9	2.4×10^{10}	0.69
Top Shield Plate	Lead	1 1/4	3.2×10^6	3.0×19^9	1.0×10^{10}	0.74
Top Shield Plate	C Steel	1	1.7×10^6	1.0×10^9	1.0×10^{10}	0.69
Top Shield	Magnetite Concrete	60	1.0×10^8	2.0×10^9	1×10^{10}	0.05
Side and Bottom Shield	Ordinary Concrete	48	3.0×10^8	3.0×10^9	6.0×10^9	0.09

*Approximation based upon Σ_t = 0.3 cm^{-1} for steel, 0.2 cm^{-1} for lead, 0.09 for ordinary concrete, and 0.13 for magnetite concrete.

4.2.2.1 Alloys

Table 11 describes the composition of alloys making up the irradiated material.

TABLE 11

ASSUMED COMPOSITION OF ALLOYS IN IRRADIATED SRE STRUCTURES AND COMPONENTS
(%)

Type Alloy	C	Mn	S_i	P	S	Cr	Ni	Fe	Co	Zr	Other
304 SS	0.08	2	1	0.04	0.04	20	10	70	0.1	-	-
C Steel	0.01	0.37	0.01	0.02	0.04	-	-	100	0.1	-	-

4.2.2.2 Insulation

The thermal insulation contained in the annulus between the cavity liner and the reactor outer vessel is a fibrous material (density 21 lb/ft^3) called "Superex." The composition of this material is described in Table 12.

TABLE 12

COMPOSITION OF "SUPEREX" INSULATION

Compound	Compound in Insulation (%)	Oxygen in Compound (%)	Oxygen in Insulation (%)	Element in Compound (%)	Element in Insulation (%)
CO_2	10	73	7.3	27	2.7
H_2O	10	89	8.9	11	1.1
SiO_2	58	47	27	53	31
MgO	13	60	8	40	5
Fe_2O_3	6.0	70	4	30	2
Al_2O_3	2.6	53	1.4	47	1.2
CaO	0.4	71	0.3	29	0.1

4.2.2.3 Ordinary Concrete

4.2.2.3.1 Concrete Mixture

A density of 2.4 gm/cm^3 was assumed for the ordinary concrete. A nominal gravel-cement-water mixture which approximates a typical mixture as described in Reference 5 was assumed. The nominal mixture is described in Table 13.

TABLE 13

NOMINAL CONCRETE MIXTURE

Constituent	Weight	Weight % of Mixture
6 bags cement	564 lb	11.5
7 gallons water/bag	336 lb	6.5
1.5 yd^3 sand and gravel	4000 lb	82.0

4.2.2.3.2 Composition of Cement

The composition of Portland cement as described in Reference 5 was utilized to calculate the content of oxygen and other specific elements in the cement. The content of these elements is described in Table 14.

TABLE 14

COMPOSITION OF PORTLAND CEMENT

Compound		Oxygen (%)		Element (%)	
Formula	(%)	in Compound	in Cement	in Compound	in Cement
CaO	64.1	28.5	18.20	71.5	45.6
SIO_2	22.9	53.4	12.20	46.6	10.7
Al_2O_3	4.5	47.0	2.12	53.0	2.38
Fe_2O_3	3.11	32.0	0.99	68.0	2.12
MgO	0.79	40.0	0.315	60.0	0.48
TiO_2	0.24	40.0	0.096	60.0	0.14
Na_2O	0.54	25.8	0.14	74.2	0.40
K_2O	0.64	17.0	0.11	83.0	0.53
SO_3	2.37	60.0	1.42	40.0	0.95

4.2.2.3.3 Composition of Sand and Gravel

The assumed chemical composition of the sand and gravel in the concrete was based upon the composition of gravels of igneous origin as described in References 6 and 7. The content of each major element was maximized across the granite-rhyolite to gabbro-basalt series, which classifies all igneous rocks on the bases of grain size and acidity (Si content). The content of each major element in these rocks is described in Table 15, as reproduced from Reference 6. Trace elements in the sand and gravel were also maximized across the granite-rhyolite to gabbro-basalt series. Trace element content is described in Figure 17, which has been reproduced from Reference 6.

Maximizing the content in the various igneous rocks of the elements described in Table 15 and Figure 17 results in the assumed composition for the gravel in the SRE ordinary concrete as described in Table 16. After summing the content of the individual elements, the remaining fraction was assigned to oxygen.

TABLE 15

AVERAGE CONTENT OF MAJOR ELEMENTS IN ROCKS
(In Percent By Weight)

Element	Igneous										Sedimentary		
	Coarse-Grained						Fine-Grained						
	Granite	Grano-Diorite	Quartz Diorite	Diorite	Gabbro	Diabase	Rhyolite	Dacite	Andesite	Basalt	Shale	Sand Stone	Lime Stone
Si	32.79	31.25	18.77	24.23	22.59	23.58	34.01	28.70	25.32	22.92	27.16	36.59	2.42
A	7.66	8.29	8.58	8.68	8.91	8.12	7.14	8.82	9.09	8.31	8.15	2.52	0.43
Fe	2.48	2.94	4.71	7.33	7.94	8.74	1.69	3.90	6.70	8.71	4.71	0.99	0.40
Mg	0.53	0.95	1.69	3.69	4.86	3.49	0.23	1.28	2.63	3.72	1.47	0.70	4.76
Ca	1.42	2.54	3.85	6.00	7.91	6.39	0.86	3.95	5.66	6.40	2.22	3.93	30.42
Na	2.58	4.05	2.50	2.49	1.68	2.28	2.51	3.06	2.72	2.31	0.96	0.33	0.04
K	3.41	4.91	1.74	1.10	0.46	0.81	3.70	3.93	0.72	1.26	2.69	1.10	0.27
Ti	0.23	0.35	0.40	0.90	0.79	0.87	0.20	0.10	0.79	0.82	0.39	0.15	0.04
P	0.08	0.08	0.11	0.15	0.10	0.11	0.03	0.03	0.12	0.20	0.07	0.03	0.02
Mn	0.09	0.09	0.08	0.14	0.14	0.15	0.06	0.04	0.12	0.24	0.08	0.19	0.04

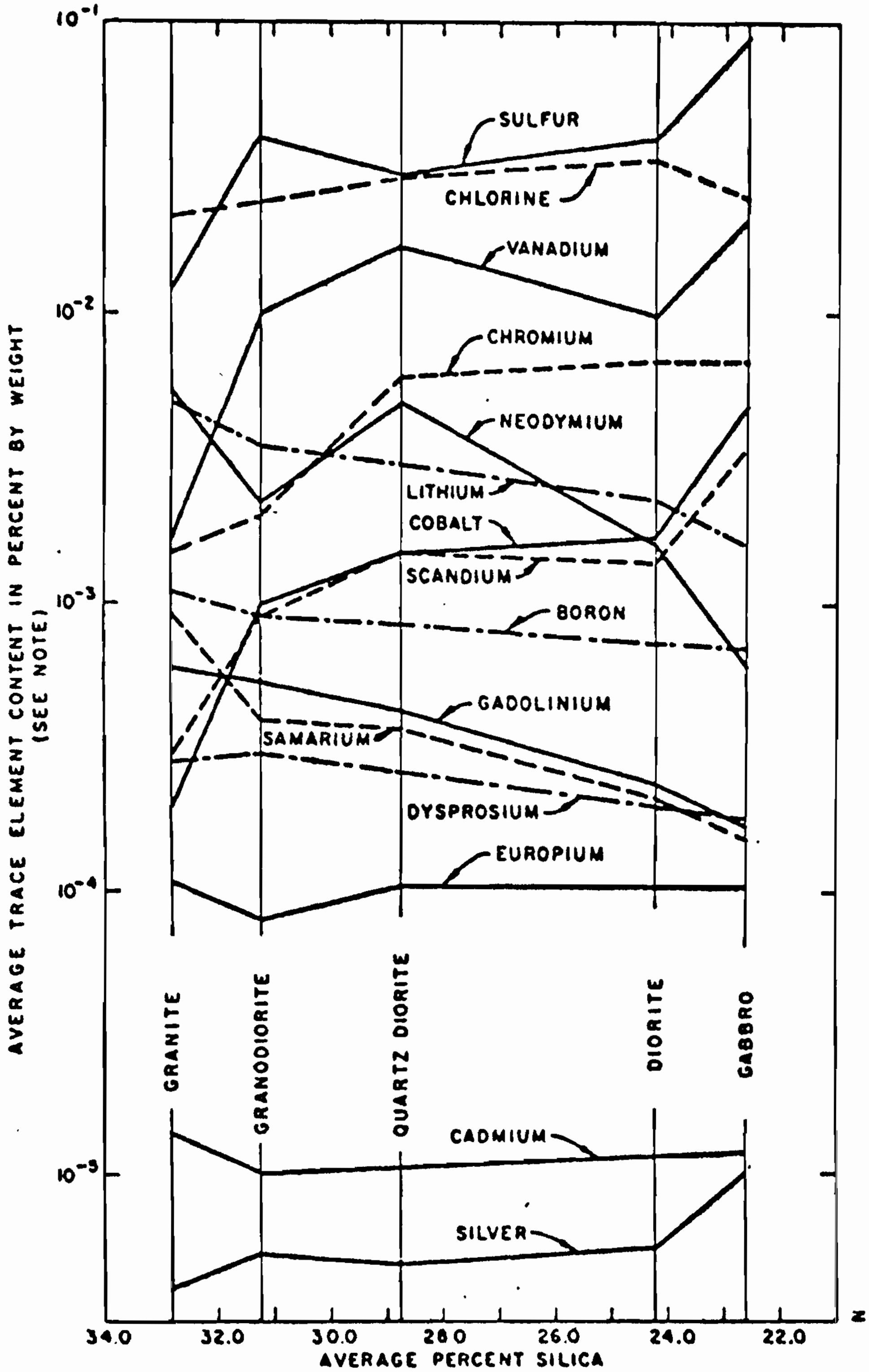


Figure 17. Variation of Trace Element Content Through Granite-Gabbro Series

Note: Data plotted on condensed scale for convenience; do not interpolate between rock types.

(From PNE 5006)

TABLE 16

MAJOR AND TRACE ELEMENT CONTENT OF IGNEOUS ROCKS MAXIMIZED ACROSS THE GRANITE-RHYOLITE TO GABBRO-BASALT SERIES

Major Element Content (%)				Trace Element Content (%)			
Si	34.01	Na	4.05	Sm	1.0×10^{-3}	B	1.0×10^{-3}
Al	9.09	Ti	0.90	Eu	1.0×10^{-4}	Sc	3.0×10^{-3}
Fe	8.74	P	0.20	Dy	4.0×10^{-4}	Co	5.0×10^{-3}
Mg	4.86	Mn	0.24	Gd	7.0×10^{-4}	Li	5.0×10^{-3}
Ca	7.91	O	26.00	Cr	7.0×10^{-3}	Nd	5.0×10^{-3}
K	4.91			V	2.0×10^{-2}	Ag	1.0×10^{-5}
				Cl	4.0×10^{-2}	Cd	1.0×10^{-5}
				S	1.0×10^{-1}		

4.2.2.4 High Density Concrete

4.2.2.4.1 Concrete Mixture

The concrete contained in the reactor loading face shield is a magnetite-type in which magnetite aggregate was mixed with Portland cement. A density of 3.4 gm/cm^3 and a typical magnetite aggregate-Portland cement-water mixture, as described in Reference 8, were assumed for the concrete. The assumed mixture appears in Table 17.

4.2.2.4.2 Composition of Magnetite Aggregate

The assumed elemental composition of the magnetite aggregate was based upon the typical composition of such aggregate as described in Reference 9. This assumed composition appears in Table 18.

TABLE 17

MAGNETITE CONCRETE MIXTURE

Constituent	Weight % of Mixture
Magnetite Coarse Agregate	41
Magnetite Fine Aggregate	34
Portland Cement	17
Water	8

TABLE 18

ASSUMED COMPOSITION OF MAGNETITE AGGREGATE

Element	Weight %
Fe	73
O	27

4.2.2.4.3 Composition of Portland Cement

The assumed composition of the Portland cement is the same as that contained in the ordinary concrete surrounding the reactor vessel.

4.2.2.4.4 Elemental Composition of Concrete Mixture

The elemental composition of the magnetite concrete in the assumed mixture appears in Table 19.

TABLE 19

ASSUMED ELEMENTAL COMPOSITION OF MAGNETITE CONCRETE

Element	Weight %	Element	Weight %
Ca	7.7	Na	0.1
Si	1.8	K	0.1
Al	0.4	S	0.1
Mg	0.1		
Fe	56.0	O	33.4
Ti	0.1	H	0.1

4.3 Results

4.3.1 Alloys and Insulation

Evaluation of the neutron activation of these components reveals that the significant radionuclides present following 11 years decay are limited to Fe-55 in the carbon steel; Fe-55, Co-60, and Ni-63 in the stainless steel; and Fe-55 in the Superex insulation. Table 20 describes the maximum specific activity, average specific activity, and total activity of each of these radionuclides present in major irradiated components of the reactor system and in the thermal insulation. Average specific activities were determined on the basis of the thickness averaging factors described in Table 10. The average specific activities were applied to the component weights described in Table 10 to determine total activities.

4.3.2 Ordinary Concrete

Tables 21 and 22 describe, for elements in the gravel and the cement, respectively, the neutron reactions considered in the calculation of product isotopes. Of the elements in the concrete, ten were found to have no radioactive daughters with half-lives greater than one year as a result of the neutron reactions considered (n,p; n,d; n,2n; and n,γ). These ten elements are:

Silicon	Chronium
Magnesium	Vanadium
Titanium	Sulfur
Phosphorous	Scandium
Maganese	Neodynium

Twelve of the radionuclides described in Tables 21 and 22 were found to be present in significant quantities following a decay period of 11 years.

Although not identified as present in even trace quantities, certain elements, if present in the SRE concrete, would result in significant daughter product radioactivity. To this end, all naturally occurring isotopes for which neutron reactions could result in daughter radionuclides with half-lives of greater than one year were examined. Those which might result in significant activity following 11 years cooling are described in Table 23 on the basis of the activity which would result if they were present in the concrete in concentrations of 1 ppm.

The specific activities and total activities for the twelve most significant radionuclides are described in Table 24. Total activities were determined on the basis that 5.2×10^7 cm^3 of concrete are contained in the first 11 cm of the shield, which is one relaxation length for fission spectrum neutrons, based upon a macroscopic removal cross section of $\Sigma_t = 0.09$ cm^{-1}, as described in Reference 10.

4.3.3 High Density Concrete

Table 25 describes, for elements in the high density concrete of the reactor loading face shield, the neutron reactions considered in the calculation of the radioactivity in the concrete and the resulting specific activities of product isotopes. The total activities may be determined on the basis that 1.5×10^6 cm^3 of concrete are contained in the first 8 cm of the shield, which is one relaxation length for fission spectrum neutrons, based upon a macroscopic removal cross section of $\Sigma_t = 0.126 cm^{-1}$.

TABLE 20

INDUCED ACTIVITY IN SRE REACTOR COMPONENTS (ALLOYS AND INSULATION)

Component	Maximum Specific Activity (μci/gm)			Average Specific Activity (μci/gm)			Total Activity (μci)		
	Fe^{55}	Co^{60}	Ni^{63}	Fe^{55}	Co^{60}	Ni^{63}	Fe^{55}	Co^{60}	Ni^{63}
Internal Piping	1.0×10^{3}	9.6×10^{2}	4.6×10^{2}	9.2×10^{2}	8.6×10^{2}	4.2×10^{2}	1.6×10^{9}	1.5×10^{9}	7.2×10^{8}
Vessel Liner	1.0×10^{3}	9.6×10^{2}	4.6×10^{2}	9.2×10^{2}	8.6×10^{2}	4.2×10^{2}	2.4×10^{9}	2.2×10^{9}	1.1×10^{9}
Grid Plate	5.2×10^{2}	4.8×10^{2}	2.3×10^{2}	3.1×10^{2}	2.9×10^{2}	1.4×10^{2}	7.7×10^{8}	7.2×10^{8}	3.5×10^{8}
Support Columns	2.1×10^{2}	2.0×10^{2}	9.6×10^{1}	1.3×10^{2}	1.2×10^{2}	5.8×10^{1}	9.0×10^{7}	8.3×10^{7}	4.0×10^{7}
Core Tank	2.1×10^{2}	2.0×10^{2}	9.6×10^{1}	1.3×10^{2}	1.2×10^{2}	5.8×10^{1}	2.9×10^{9}	2.7×10^{9}	1.3×10^{9}
Thermal Shield	1.0×10^{2}			2.4×10^{1}			6.3×10^{8}		
Outer Tank	6.6×10^{-1}			5.9×10^{-1}			7.7×10^{6}		
Insulation	9.2×10^{-2}			9.2×10^{-2}			8.0×10^{5}		
Cavity Liner	4.3×10^{-1}			3.8×10^{-1}			3.3×10^{6}		
Cooling Coils	2.0×10^{0}			1.8×10^{0}			3.4×10^{6}		
Reinforcing Rods	2.0×10^{0}			1.7×10^{0}			3.9×10^{5}		
Top Shield Insulation	2.5×10^{0}	2.4×10^{0}	1.1×10^{0}	2.1×10^{0}			1.8×10^{6}		
Top Shield Lower Plate	1.8×10^{0}	1.5×10^{0}	7.4×10^{-1}	1.2×10^{0}	1.0×10^{0}	5.1×10^{-1}	2.2×10^{6}	1.8×10^{6}	9.2×10^{5}
Top Shield Lead Plate									
Top Shield Upper Plate	6.6×10^{-1}			4.5×10^{-1}			7.6×10^{5}		
TOTAL							8.4×10^{9}	7.1×10^{9}	3.5×10^{9}

TABLE 21

MAXIMUM SPECIFIC ACTIVITY OF SRE ORDINARY CONCRETE RESULTING FROM ACTIVATION OF MAJOR AND TRACE ELEMENTS IN GRAVEL

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	E(0.82) (fraction)	I (fraction)	Activity From Core I_3 ($\mu ci/cm^3$)	Activity From Core II_3 ($\mu ci/cm^3$)	Total Maximum Specific Activity ($\mu ci/cm^3$)
H^3	1.2×10^1	Li^6	n,α	9.5×10^2	4.1×10^{-5}	7.4×10^{-2}	1.3×10^0	1.7×10^0	$3{,}0 \times 10^0$
Be^{10}	2.7×10^6	B^{10}	n,p	2×10^{-1}	8.2×10^{-6}	2.0×10^{-1}	9.8×10^{-10}	9.8×10^{-10}	2.0×10^{-9}
C^{14}	5.7×10^3	O^{17}	n,α	4.0×10^{-1}	2.1×10^{-1}	3.7×10^{-4}	2.5×10^{-5}	$2.5 \times 1\text{-}^{-5}$	5.0×10^{-5}
Na^{22}	2.6×10^0	Na^{23}	n,2n	7×10^{-6}	3.3×10^{-2}	1.0×10^0	3.9×10^{-6}	1.5×10^{-5}	1.9×10^{-5}
Al^{26}	7.4×10^5	Al^{27}	n,2n	4×10^{-6}	7.5×10^{-2}	1.0×10^0	1.2×10^{-9}	1.2×10^{-9}	2.4×10^{-9}
Cl^{36}	3.0×10^5	K^{39}	n,α	6×10^{-3}	4.0×10^{-2}	9.3×10^{-1}	1.5×10^{-6}	1.5×10^{-6}	3.0×10^{-6}
		Cl^{35}	n,γ	4.4×10^1	3.3×10^{-4}	7.6×10^{-1}	8.3×10^{-5}	8.3×10^{-5}	1.7×10^{-4}
		Cl^{37}	n,2n	1.1×10^{-4}	3.3×10^{-4}	2.4×10^{-1}	6.2×10^{-11}	6.2×10^{-11}	1.2×10^{-10}
Ar^{39}	2.7×10^2	Ca^{42}	n,α	2.4×10^{-3}	6.5×10^{-2}	6.4×10^{-3}	7.1×10^{-6}	7.1×10^{-6}	1.7×10^{-5}
		K^{39}	n,p	2.4×10^{-2}	4.0×10^{-2}	9.3×10^{-1}	6.9×10^{-3}	6.9×10^{-3}	1.4×10^{-2}
K^{40}	1.3×10^9	K^{39}	n,γ	2.2×10^0	4.0×10^{-2}	9.3×10^{-1}	1.3×10^{-7}	1.3×10^{-7}	2.6×10^{-7}
		Ca^{40}	n,p	4.7×10^{-2}	6.5×10^{-2}	9.7×10^{-1}	4.6×10^{-9}	4.6×10^{-9}	9.2×10^{-9}
		K^{41}	n,2n	1.5×10^{-4}	4.0×10^{-2}	6.9×10^{-2}	6.5×10^{-13}	6.5×10^{-13}	1.3×10^{-12}
Ca^{41}	7.7×10^4	Ca^{40}	n,γ	2×10^{-1}	6.6×10^{-2}	9.7×10^{-1}	3.3×10^{-4}	3.3×10^{-4}	6.6×10^{-4}
		Ca^{42}	n,2n	4×10^{-5}	$6{,}5 \times 10^{-2}$	6.4×10^{-3}	4.1×10^{-10}	4.1×10^{-10}	8.2×10^{-10}

TABLE 21 (Continued)

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	E(0.82) (fraction)	I (fraction)	Activity From Core I, ($\mu ci/cm^3$)	Activity From Core II, ($\mu ci/cm^3$)	Total Maximum Specific Activity ($\mu ci/cm^3$)
Fe^{55}	2.7×10^{0}	Fe^{54}	n,γ	2.9×10^{0}	7.2×10^{-2}	5.8×10^{-2}	9.3×10^{-2}	3.4×10^{-1}	4.3×10^{-1}
		Fe^{56}	n,2n	7×10^{-5}	7.2×10^{-2}	9.2×10^{-2}	3.3×10^{-6}	1.2×10^{-5}	1.5×10^{-5}
Co^{60}	5.3×10^{0}	Co^{59}	n,γ	3.7×10^{1}	4.1×10^{-5}	1.0×10^{0}	4.4×10^{-2}	8.4×10^{-2}	1.3×10^{-1}
Pd^{107}	7.0×10^{6}	Ag^{107}	n,p	9×10^{-4}	8.2×10^{-8}	5.2×10^{-1}	7.1×10^{-2}	4.1×10^{-15}	0.2×10^{-15}
		Cd^{110}	n,α	1.6×10^{-5}	8.2×10^{-8}	1.2×10^{-1}	1.6×10^{-17}	1.6×10^{-17}	3.2×10^{-17}
Ag^{108}	1×10^{2}	Ag^{107}	n,γ	4×10^{1}	8.2×10^{-8}	5.2×10^{-1}	1.3×10^{-5}	1.3×10^{-5}	2.6×10^{-5}
		Ag^{109}	n,2n	1.1×10^{-3}	8.2×10^{-8}	4.8×10^{-1}	3.2×10^{-10}	3.1×10^{-10}	6.4×10^{-10}
		Cd^{108}	n,p	7.6×10^{-3}	8.2×10^{-8}	8.8×10^{-3}	4.0×10^{-11}	4.0×10^{-11}	8.0×10^{-11}
Cd^{109}	1.3×10^{0}	Cd^{108}	n,γ	1×10^{1}	8.2×10^{-8}	8.8×10^{-3}	6.7×10^{-10}	9.5×10^{-9}	1.0×10^{-8}
		Cd^{110}	n,2n	7×10^{-4}	8.2×10^{-8}	1.2×10^{-1}	6.4×10^{-13}	9.1×10^{-12}	9.7×10^{-12}
Cd^{113}	1.4×10^{1}	Cd^{112}	n,γ	3×10^{-2}	8.2×10^{-8}	2.4×10^{-1}	1.4×10^{-8}	1.8×10^{-8}	3.2×10^{-8}
		Cd^{114}	n,2n	1.2×10^{-3}	8.2×10^{-8}	2.9×10^{-1}	6.5×10^{-10}	8.3×10^{-10}	1.5×10^{-9}
Pm^{147}	2.7×10^{0}	Sm^{147}	n,p	1.1×10^{-4}	8.2×10^{-6}	1.5×10^{-1}	3.8×10^{-10}	1.4×10^{-9}	1.8×10^{-9}
Sm^{146}	1.2×10^{8}	Sm^{147}	n,2n	2.4×10^{-2}	8.2×10^{-6}	1.5×10^{-1}	1.3×10^{-13}	1.3×10^{-13}	2.6×10^{-13}

TABLE 21 (Continued)

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	E(0.82) (fraction)	I (fraction)	Activity From Core I_3 ($\mu ci/cm^3$)	Activity From Core II_3 ($\mu ci/cm^3$)	Total Maximum Specific Activity ($\mu ci/cm^3$)
Sm^{151}	9×10^{1}	Sm^{150}	n,γ	1×10^{2}	8.2×10^{-6}	7.4×10^{-2}	3.3×10^{-4}	3.4×10^{-4}	6.7×10^{-4}
		Sm^{152}	n,2n	5.8×10^{-3}	8.2×10^{-6}	2.7×10^{-1}	6.8×10^{-8}	7.0×10^{-8}	1.4×10^{-7}
		Eu^{151}	n,p	1×10^{-4}	8.2×10^{-7}	4.8×10^{-1}	2.1×10^{-10}	2.2×10^{-10}	4.3×10^{-10}
		Gd^{154}	n,α	1.8×10^{-6}	5.7×10^{-6}	2.2×10^{-2}	1.1×10^{-12}	1.2×10^{-12}	2.3×10^{-12}
Eu^{150}	5×10^{0}	Eu^{151}	n,2n	4.8×10^{-3}	8.2×10^{-7}	4.8×10^{-1}	2.0×10^{-8}	4.0×10^{-8}	6.0×10^{-8}
Eu^{152}	2.4×10^{0}	Eu^{151}	n,γ	8.4×10^{3}	8.2×10^{-7}	4.8×10^{-1}	5.9×10^{-3}	2.5×10^{-2}	3.1×10^{-2}
		Eu^{153}	n,2n	4.4×10^{-3}	8.2×10^{-7}	5.2×10^{-1}	3.3×10^{-9}	1.4×10^{-8}	1.7×10^{-8}
		Gd^{152}	n,p	1.4×10^{-4}	5.7×10^{-6}	2.0×10^{-3}	2.8×10^{-12}	1.2×10^{-11}	1.5×10^{-11}
Eu^{154}	1.6×10^{1}	Eu^{153}	n,γ	3.2×10^{2}	8.2×10^{-7}	5.2×10^{-1}	2.3×10^{-3}	2.8×10^{-3}	5.1×10^{-3}
		Gd^{154}	n,p	2×10^{-5}	5.7×10^{-6}	2.2×10^{-2}	4.0×10^{-11}	5.0×10^{-11}	9.0×10^{-11}
Eu^{155}	1.8×10^{0}	Gd^{155}	n,p	8×10^{-5}	5.7×10^{-6}	1.5×10^{-1}	3.0×10^{-11}	2.1×10^{-10}	2.4×10^{-11}
Tb^{158}	1.5×10^{2}	Dy^{158}	n,p	1×10^{-4}	3.3×10^{-6}	9.0×10^{-4}	1.0×10^{-12}	1.0×10^{-12}	2.0×10^{-12}

TABLE 22

MAXIMUM SPECIFIC ACTIVITY OF SRE ORDINARY CONCRETE RESULTING FROM ACTIVATION OF PORTLAND CEMENT

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	E(0.12) (fraction)	I (fraction)	Activity From Core I, (μci/cm^3)	Activity From Core II, (μci/cm^3)	Total Maximum Specific Activity (μci/cm^3)
Ca^{41}	7.7×10^{4}	Ca^{40}	n,γ	2×10^{-1}	5.5×10^{-2}	9.7×10^{-1}	2.8×10^{-4}	2.8×10^{-4}	5.6×10^{-4}
		Ca^{42}	n,2n	4×10^{-5}	5.5×10^{-2}	6.4×10^{-3}	3.6×10^{-10}	3.6×10^{-10}	7.2×20^{-10}
Ar^{39}	2.7×10^{2}	K^{39}	n,p	2.4×10^{-2}	6.4×10^{-4}	9.3×10^{-1}	1.1×10^{-4}	1.1×10^{-4}	2.2×10^{-4}
		Ca^{42}	n,α	2.4×10^{-3}	5.5×10^{-2}	6.4×10^{-3}	6.0×10^{-6}	6.0×10^{-6}	1.2×10^{-5}
K^{40}	1.3×10^{9}	Ca^{40}	n,p	4.7×10^{-2}	5.5×10^{-2}	9.7×10^{-1}	3.8×10^{-9}	3.8×10^{-9}	7.6×10^{-9}
		K^{39}	n,γ	2.2×10^{0}	6.4×10^{-4}	9.3×10^{-1}	2.0×10^{-9}	2.0×10^{-9}	4.0×10^{-9}
		K^{41}	n,2n	1.5×10^{-6}	6.4×10^{-4}	6.9×10^{-1}	9.9×10^{-14}	9.9×10^{-14}	2.0×10^{-13}
Al^{26}	7.4×10^{5}	Al^{27}	n,2n	4×10^{-6}	2.9×10^{-3}	1.0×10^{0}	4.7×10^{-11}	4.7×10^{-11}	9.4×10^{-11}
Fe^{55}	2.7×10^{0}	Fe^{54}	n,γ	2.9×10^{0}	2.5×10^{-3}	5.8×10^{-2}	3.2×10^{-3}	1.2×10^{-2}	1.5×10^{-2}
		Fe^{56}	n,2n	7×10^{-5}	2.5×10^{-3}	9.2×10^{-2}	1.2×10^{-7}	4.3×10^{-7}	5.5×10^{-7}
Na^{22}	2.6×10^{0}	Na^{23}	n,2n	7×10^{-6}	4.8×10^{-4}	1.0×10^{0}	6.0×10^{-8}	2.2×10^{-7}	2.8×10^{-7}
Cl^{36}	3.0×10^{5}	K^{39}	n,α	6×10^{-3}	6.4×10^{-3}	9.3×10^{-1}	2.4×10^{-8}	2.4×10^{-8}	4.8×10^{-8}
C^{14}	5.7×10^{3}	O^{17}	n,α	4.0×10^{-1}	4.2×10^{-2}	3.7×10^{-4}	5.1×10^{-6}	5.1×10^{-6}	1.0×10^{-5}

TABLE 23

MAXIMUM SPECIFIC ACTIVITY WHICH WOULD RESULT FROM PRESENCE IN SRE ORDINARY CONCRETE OF CERTAIN ADDITIONAL ELEMENTS IN CONCENTRATIONS OF 1 ppm

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	Assumed E (fraction)	I (fraction)	Activity From Core I (μci/cm^3)	Activity From Core II (μci/cm^3)	Total Maximum Specific Activity (μci/cm^3)
Be^{10}	2.7×10^{6}	Be^{9}	n,γ	1.0×10^{-2}	1.0×10^{-6}	1.0×10^{0}	3.3×10^{-11}	3.3×10^{-11}	6.6×10^{-11}
C^{14}	5.7×10^{3}	N^{14}	n,p	1.8×10^{0}	1.0×10^{-6}	1.0×10^{0}	1.0×10^{-6}	1.8×10^{-6}	3.6×10^{-6}
Ni^{59}	8.0×10^{4}	Ni^{57}	n,γ	4.4×10^{0}	1.0×10^{-6}	6.8×10^{-1}	5.3×10^{-8}	5.3×10^{-8}	1.1×10^{-7}
Ni^{63}	9.2×10^{1}	Ni^{62}	n,γ	1.5×10^{1}	1.0×10^{-6}	3.7×10^{-2}	7.3×10^{-6}	7.3×10^{-6}	1.5×10^{-5}
		Cu^{63}	n,p	9.0×10^{-3}	1.0×10^{-6}	6.9×10^{-1}	8.6×10^{-3}	8.6×10^{-8}	1.7×10^{-7}
Co^{60}	5.3×10^{0}	Cu^{63}	n,α	7.6×10^{-4}	1.0×10^{-6}	6.9×10^{-1}	1.5×10^{-8}	2.9×10^{-8}	4.4×10^{-8}
Nb^{94}	2.0×10^{4}	Nb^{93}	n,γ	1.1×10^{0}	1.0×10^{-6}	1.0×10^{0}	4.9×10^{-8}	4.9×10^{-8}	9.8×10^{-8}
Mo^{93}	1.0×10^{4}	Mo^{92}	n,γ	3.1×10^{-1}	1.0×10^{-6}	1.6×10^{-1}	4.3×10^{-9}	4.3×10^{-9}	8.6×10^{-9}
Sn^{121}	2.5×10^{1}	Sn^{120}	n,γ	1.0×10^{-3}	1.0×10^{-6}	3.3×10^{-1}	8.0×10^{-9}	9.2×10^{-9}	1.7×10^{-8}
		Sn^{122}	n,2n	1.9×10^{-3}	1.0×10^{-6}	4.7×10^{-2}	2.1×10^{-9}	2.4×10^{-9}	4.5×10^{-9}
		Sb^{121}	n,p	3.3×10^{-4}	1.0×10^{-6}	5.7×10^{-1}	4.4×10^{-9}	5.1×10^{-9}	9.5×10^{-9}
Cs^{134}	2.1×10^{0}	Cs^{133}	n,γ	3.1×10^{1}	1.0×10^{-6}	1.0×10^{0}	3.8×10^{-5}	2.0×10^{-4}	2.4×10^{-4}

TABLE 23 (Continued)

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	Assumed E (fraction)	I (fraction)	Activity From Core I ($\mu ci/cm^3$)	Activity From Core II ($\mu ci/cm^3$)	Total Maximum Specific Activity ($\mu ci/cm^3$)
Ba^{133}	7.2×10^0	Ba^{132}	n,γ	7.2×10^0	1.0×10^{-6}	9.7×10^{-4}	1.2×10^{-7}	1.9×10^{-7}	3.1×10^{-7}
Ho^{166}	1.2×10^3	Ho^{165}	n,γ	1.0×10^0	1.0×10^{-6}	1.0×10^0	4.2×10^{-7}	4.2×10^{-7}	8.4×10^{-7}
Pt^{193}	5.0×10^2	Pt^{192}	n,γ	1.6×10^1	1.0×10^{-6}	7.8×10^{-3}	1.3×10^{-7}	1.3×10^{-7}	2.6×10^{-7}
Ti^{204}	3.8×10^0	Ti^{203}	n,γ	1.1×10^1	1.0×10^{-6}	3.0×10^{-1}	1.1×10^{-5}	3.4×10^{-5}	4.5×10^{-5}
Sr^{90}	2.8×10^1	U^{235}	Fission	5.8×10^2	1.0×10^{-6}	7.0×10^{-3}	4.5×10^{-5}	5.1×10^{-5}	9.6×10^{-5}
Y^{90}	Equilib Sr^{90}	U^{235}	Fission	5.8×10^2	1.0×10^{-6}	7.0×10^{-3}	4.5×10^{-5}	5.1×10^{-5}	9.6×10^{-5}
Cs^{137}	3.0×10^1	U^{235}	Fission	5.8×10^2	1.0×10^{-6}	7.0×10^{-3}	4.5×10^{-5}	5.1×10^{-5}	9.6×10^{-5}

TABLE 24

MAXIMUM SPECIFIC ACTIVITY AND TOTAL ACTIVITY FOR TWELVE MOST SIGNIFICANT RADIONUCLIDES IN THE SRE ORDINARY CONCRETE BIOLOGICAL SHIELDING

Radionuclide	Maximum Specific Activity ($\mu ci/cm^3$)	Total Activity (μci)
H-3	3.0×10^{0}	1.6×10^{8}
Fe-55	4.5×10^{-1}	2.3×10^{7}
Co-60	1.3×10^{-1}	6.8×10^{6}
Eu-152	3.1×10^{-2}	1.6×10^{6}
Ar-39	1.5×10^{-2}	7.8×10^{5}
Eu-154	5.1×10^{-3}	2.7×10^{5}
Ca-41	1.2×10^{-3}	6.2×10^{4}
Sm-151	6.7×10^{-4}	3.5×10^{4}
Cl-36	1.7×10^{-4}	8.8×10^{3}
C-14	6.0×10^{-5}	3.1×10^{3}
Ag-108	2.6×10^{-5}	1.4×10^{3}
Na-22	1.9×10^{-5}	9.9×10^{2}
Total	3.6×10^{0}	1.9×10^{8}

TABLE 25

INDUCED RADIOACTIVITY IN THE CONCRETE IN THE SRE REACTOR LOADING FACE SHIELD

Product Isotope	Half Life (Years)	Parent	Reaction	Cross Section (barns)	E (fraction)	I (fraction)	Specific Activity From Core I_3 ($\mu ci/cm^3$)	Specific Activity From Core II_3 ($\mu ci/cm^3$)	Total Maximum Specific Activity ($\mu ci/cm^3$)
Ca^{41}	7.7×10^{4}	Ca^{40}	n,γ	2.0×10^{-1}	7.7×10^{-2}	9.7×10^{-1}	3.8×10^{-4}	3.8×10^{-4}	7.6×10^{-4}
		Ca^{42}	n,2n	4.0×10^{-5}	7.7×10^{-2}	6.4×10^{-3}	1.9×10^{-9}	1.9×10^{-9}	3.8×10^{-9}
Ar^{39}	2.7×10^{2}	K^{39}	n,p	2.4×10^{-2}	1.0×10^{-3}	9.3×10^{-1}	6.0×10^{-4}	6.6×10^{-4}	1.3×10^{-3}
		Ca^{42}	n,α	2.4×10^{-3}	7.7×10^{-2}	6.4×10^{-3}	3.2×10^{-5}	3.2×10^{-5}	6.4×10^{-5}
K^{40}	1.3×10^{9}	Ca^{40}	n,p	4.7×10^{-2}	7.7×10^{-2}	9.7×10^{-1}	2.0×10^{-8}	2.0×10^{-8}	4.0×10^{-8}
		K^{39}	n,γ	2.2×10^{4}	1.0×10^{-3}	9.3×10^{-1}	3.1×10^{-9}	3.1×10^{-9}	6.2×10^{-9}
		K^{41}	n,2n	1.5×10^{-4}	1.0×10^{-3}	6.9×10^{-1}	7.9×10^{-13}	7.9×10^{-13}	1.6×10^{-12}
Al^{26}	7.4×10^{5}	Al^{27}	n,2n	4.0×10^{-6}	4.0×10^{-3}	1.0×10^{0}	2.5×10^{-9}	2.5×10^{-9}	5.0×10^{-9}
Fe^{55}	2.4×10^{0}	Fe^{54}	n,γ	2.9×10^{0}	5.6×10^{-1}	5.8×10^{-2}	6.5×10^{-1}	2.4×10^{0}	3.1×10^{0}
		Fe^{56}	n,2n	7.0×10^{-5}	5.6×10^{-1}	8.7×10^{-1}	9.5×10^{-4}	3.5×10^{-3}	4.5×10^{-3}
Na^{22}	2.6×10^{0}	Na^{23}	n,2n	7.0×10^{-6}	1.0×10^{-3}	1.0×10^{0}	5.7×10^{-7}	2.1×10^{-6}	2.7×10^{-6}
Cl^{36}	3.0×10^{5}	K^{39}	n,α	6.0×10^{-3}	1.0×10^{-3}	9.3×10^{-1}	1.4×10^{-7}	1.4×10^{-7}	2.8×10^{-7}
C^{14}	5.7×10^{3}	O^{17}	n,α	4.0×10^{-1}	3.3×10^{-1}	3.7×10^{-4}	3.8×10^{-5}	3.8×10^{-5}	5.6×10^{-5}

5.0 Project Management

5.1 Organization

The Program Office organization and its relationship to other AI operations is described in the Decontamination and Disposition of Facilities Program Plan, PP-704-990-002.

5.2 Cost and Schedule Control

Plans for Cost and Schedule Control are also presented in the Program Plan.

5.3 Health and Safety

Program Health and Safety requirements are as designated in the Operational Safety Plan for the Decontamination and Disposition of Facilities Program, SRR-704-990-001. In addition, specific Health and Safety requirements will be delineated in the Activity Requirements and in the applicable steps of the Detailed Working Procedures.

5.4 Quality Assurance

The Quality Assurance requirements are presented in the Quality Assurance Program Plan for the Decontamination and Disposition of Facilities, PP-704-990-001.

6.0 Planning

6.1 Activity Requirements

The magnitude of the SRE dismantling requires that the dismantling activities be subdivided into separate manageable tasks designated as "Activities." Each of these tasks will be described and the requirements for accomplishment identified in a specific document termed "Activity Requirements." A listing of the currently proposed Activity Requirements is presented in Table 26.

6.2 Detailed Working Procedures

Detailed working procedures will be prepared to describe the step-by-step actions required to accomplish a specific activity. The detailed working procedure will relate to a specific Activity Requirements document. More than one procedure may be necessary to support each Activity Requirements document.

6.3 Activity Network Schedule

The overall D&D schedule was presented in the Program Plan, along with the Second Level Activity Network Schedule for the SRE dismantling. The second level contains estimated performance duration of each Activity and the sequence and interrelationship between Activities. Third Level Activity Network Schedules will be developed for each major Activity having sufficient complexity to warrant interface planning. These schedules will be used for in-house progress monitoring.

TABLE 26

SRE DISMANTLING AND DISPOSITIONING ACTIVITY REQUIREMENTS

Activity Number	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 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 Fuel Handling Machine
19.0	Decontamination and Dismantling of Mark II Fuel Handling Machine
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
28.0	Dispositioning of Electrical Support Systems: Diesel Engine, M-G Sets, Switch Gear
29.0	Final Closeout of SRE Facility

7.0 Tooling and Equipment

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

An Activity Requirement document has been written which identifies the tasks and technical approach that will be utilized to develop 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 No. 1 - Remote Tooling for Removal of SRE Vessels," 10-2-74). This document identifies fourteen task requirements and presents the Activity Network Schedule for performing the tooling effort. Each of these tasks will require a Task Requirement document to define the associated purpose, scope, requirements, and technical approach. Additionally, a Task Requirement document has been written which describes the consulting effort provided by ORNL to AI for the manipulator development and operations.

The manipulator design and fabrication, mockup design and fabrication, and plasma torch/control console checkout will be performed during GFY 1975. These subsystems will be installed in the mockup facility and the vessel cutup parameters will be developed in GFY 1976. The SRE fixed internals will also be mocked up and special tooling and procedures developed. The vessels and fixed internals in the SRE will be cut out and disposed of during GFY 1977 and 1978.

8.0 Decontamination and Dismantling

8.1 Peripheral Systems Removal

As noted in the "Present Status of the Site" section of this document, portions of the SRE support systems external to the SRE, Building 143, were removed during GFY 1975 as a part of the overall AI D&D Program.

The systems and equipment removed are as follows:

8.1.1 The Kerosene Cooling System located on the pad north of Building 143. The items removed are the pump drive, filter, storage tank, heat exchanger, piping, and fittings.

8.1.2 The Nitrogen Gallery Cooling Systems, including equipment and piping, up to the trenches entering Building 143 and up to the point of penetration into the sodium service and the primary sodium tank vaults.

8.1.3 The Secondary Sodium System external to Building 143. The sodium brake, the cold trap, the piping, and the sodium coils in the air blast heat exchanger were removed.

8.1.4 The air blast heat exchangers (main and auxiliary) and the associated cooling tower.

Continuation of the peripheral systems removal will include removal of the SRE process water tank and piping, the nitrogen gallery and vault cooling system piping in the trenches outside Building 143, the nonradioactive sodium piping to and between the primary sodium vaults, the sodium service building (153), the kerosene system piping in the trenches outside Building 143, and the sodium nitrogen and kerosene piping support structures. Secondary sodium-containing components and piping are being stripped of asbestos insulating material in the SRE area. The piping and equipment are then transferred to the AI sodium burn pit where the residual sodium is reacted with water. The asbestos is packaged in plastic bags and disposed of as waste.

Removal of the peripheral systems, except for items which require cutting into sodium piping, is being accomplished by a salvage contractor. Standard AI working conditions for construction workers on AI sites apply. These conditions are described in the Program Plan and in AI Document 511-C, Rev. 4-70, "Plant Rules and Regulations for Construction Workers." The contractual arrangement with the salvage contractor essentially is a no-cost purchase order. The contractor receives the salvage material in exchange for the labor of removal. Wherever the salvage value of special items is significantly greater than the labor costs, the salvage contractors are requested to submit bids. Equipment and material usable on other AI programs or potentially usable on this program are being set aside. Health, Safety, and Radiation Services department personnel will survey all equipment and materials for radioactive contamination prior to release from the site.

8.2 Facility Reactivation; Buildings and Handling Systems

Support of the SRE D&D activities will require that the Contaminated Equipment Repair Facility in Building 163 and the Component Cleaning Facility, Building 724, be reactivated. These facilities will be used for removing and passivating the residual sodium from the secondary and primary sodium systems components. An Activity Requirements document will be prepared for each facility. The Activity Requirements will document the need for the facility, describe the existing facility conditions, delineate the steps required to make the facility functional, delineate the criteria for acceptable completion, and briefly describe the method of accomplishment.

The SRE radioactive waste handling systems, both gaseous and liquid, are operational and will be maintained until no longer needed to support the SRE dismantling operations. The SRE Fuel Handling Machines, Mark I and Mark II, and the Moderator Handling Machine, may require refurbishing to make them operational. The use of the handling machines is a complicated operation, requiring rotation of the reactor loading face shield plug in addition to other involved operational procedures. Since the operation is complex and time consuming, it may be advantageous to avoid using the machines entirely. A simpler method of removal of components will be considered. After the residual sodium in the reactor is passivated, the reactor will be filled with water and the core components supported from the loading face shield will be lifted using the building crane and the plug-lifting attachment. These core items will be lifted slowly and bagged while a continuous monitoring of radiation levels is performed. The moderator elements are new and therefore not activated. They may be contaminated by the sodium and other reactor materials but not to levels requiring special shielding. To remove the moderator elements, the large 140" plug will be removed and the elements grappled in the water, again using the building crane and a modified grapple.

8.3 Disposal of Primary Sodium

The primary sodium has been drained from the reactor cooling system into the primary sodium fill tank. This sodium is radioactive. Figure 6 shows the maximum radiation levels on the surface of the tank.

The tank has a 1,204 cu ft capacity and presently contains 55,000 lb of sodium. The primary sodium will be drained into 55-gallon drums and stored for future disposition. Based on current considerations, the SRE sodium will be used in the CRBR Program or will be passivated and shipped to burial. A passivation process for such a significant amount of radioactive sodium will require development.

Activity Requirements Document 2.0, Disposal of Primary Sodium, has been prepared and released. A Detailed Working Procedure has been prepared, reviewed and approved. The Activity has begun. A chemical and radioactivity analysis will be made of representative samples and the data included with the storage documentation. The sodium will be stored in drums meeting DOT requirements for shipment.

8.4 Removal of Sodium Components in Main and Auxiliary Pipe Galleries and the Sodium Service Vault

Primary and secondary sodium components in the main gallery will be cut out and removed. The residual sodium in these components will be passivated in the sodium component cleaning facilities located in Buildings 724 and 163. Activity Requirements documents will be prepared for the removal of the primary sodium components from the pipe galleries, sodium service vault, and the sodium service building.

Activity Requirements Documents 12.0 and 11.0 will establish the criteria for operations in Facilities 724 and 163. The asbestos insulation will be removed from the components and the component will be cut out, lifted from the gallery, and sent to the cleaning facility. Health, Safety, and Radiation Services will provide radiological and safety monitoring during the operations and will review the Detailed Working Procedures for health and safety requirements. The principal sodium components shown in the P&I diagram (Figure 18) are listed in Table 27.

TABLE 27

PRINCIPAL PRIMARY SODIUM COMPONENTS

	Component	Identity No.
A.	Main Intermediate Heat Exchanger	X-1
B.	Auxiliary Intermediate Heat Exchanger	X-3
C.	Main Primary Pump	P-1
D.	Auxiliary Primary Pump	P-3
E.	Primary Electromagnetic Brake	BR-1
F.	Condensate Trap	T-3
G.	Reactor Drain Pump	P-6
H.	Primary Drain Pump	P-7
I.	Valve (V103) Scram Flow Control	
J.	Flowmeters	
K.	Hot Traps	HT-1A, 1B
L.	Hot Trap Economizer	X-5
M.	Main Primary Cold Trap	CT-1
N.	Primary Plugging Meter (PM) Economizer	X-6
O.	Primary PM Cooler	X-7
P.	Miscellaneous Piping, Valves, Fittings	X-7

After reacting the residual sodium, the primary system components will be surveyed for radioactive contamination. A determination will then be made whether to decontaminate or to ship for burial. Large complicated components, such as traps and heat exchangers, will likely be shipped for burial.

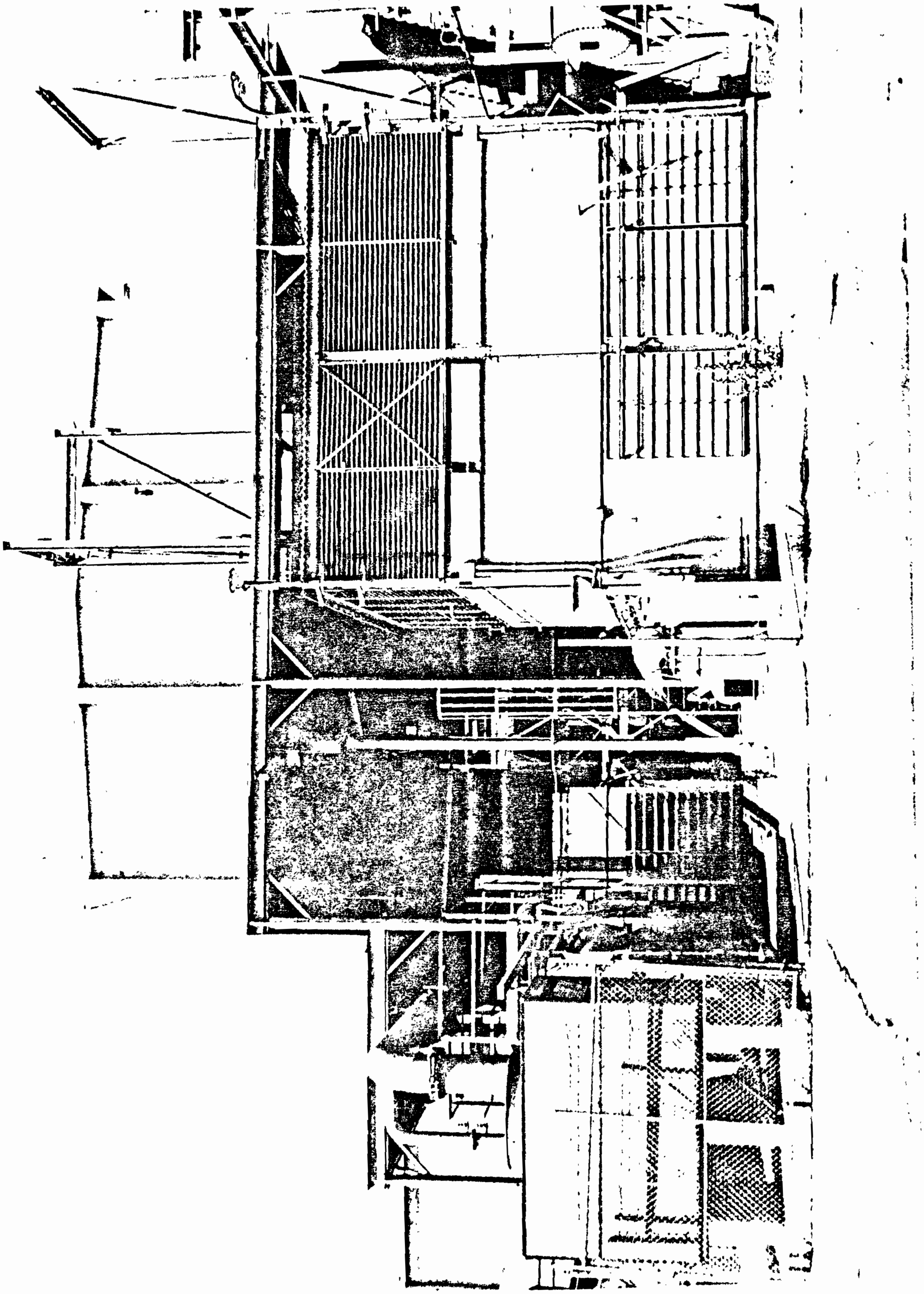


Figure 19. SRE: Air Blast Heat Exchanger, Cooling Tower, Gallery Cooling

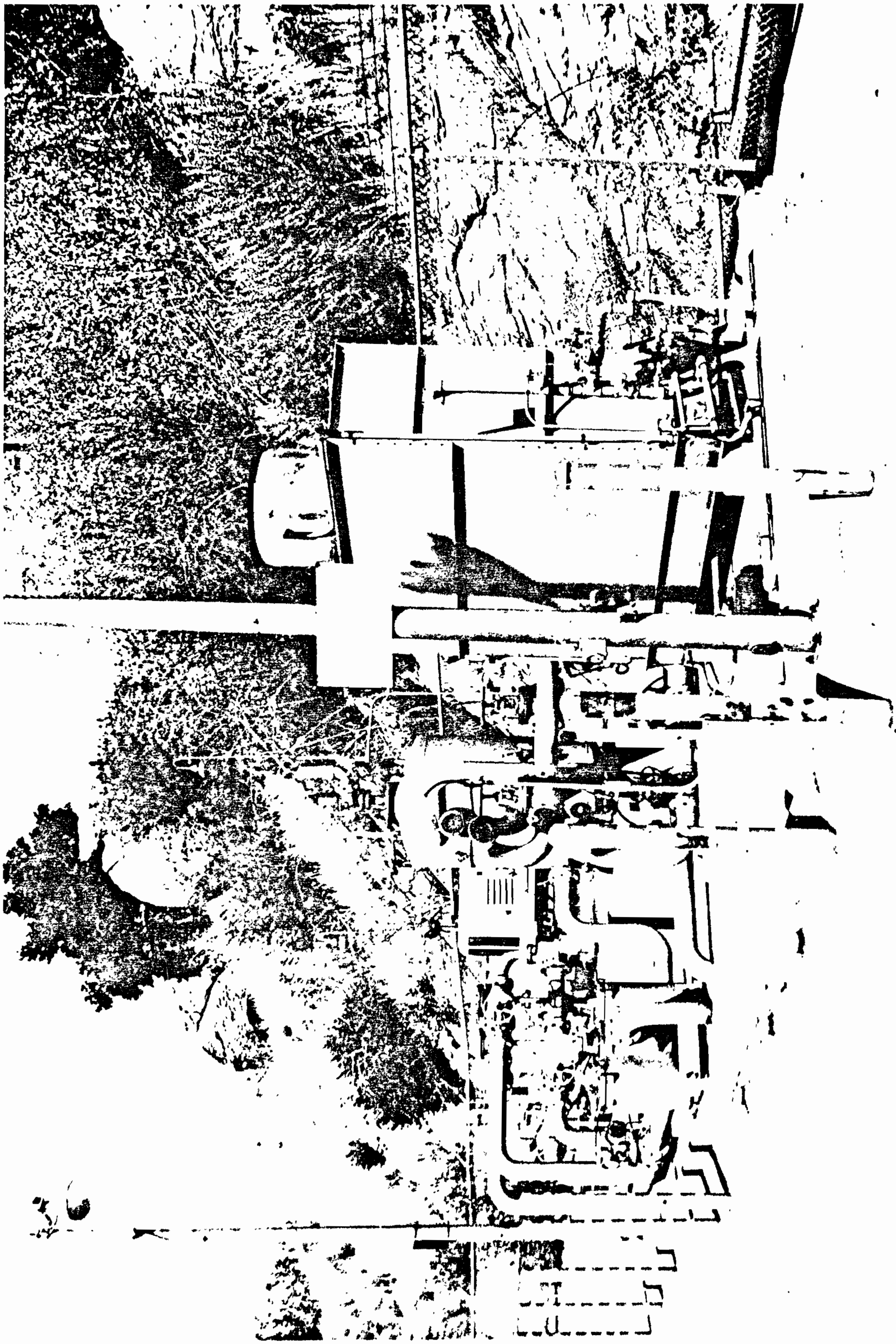


Figure 20. SRE: Kerosene Supply System

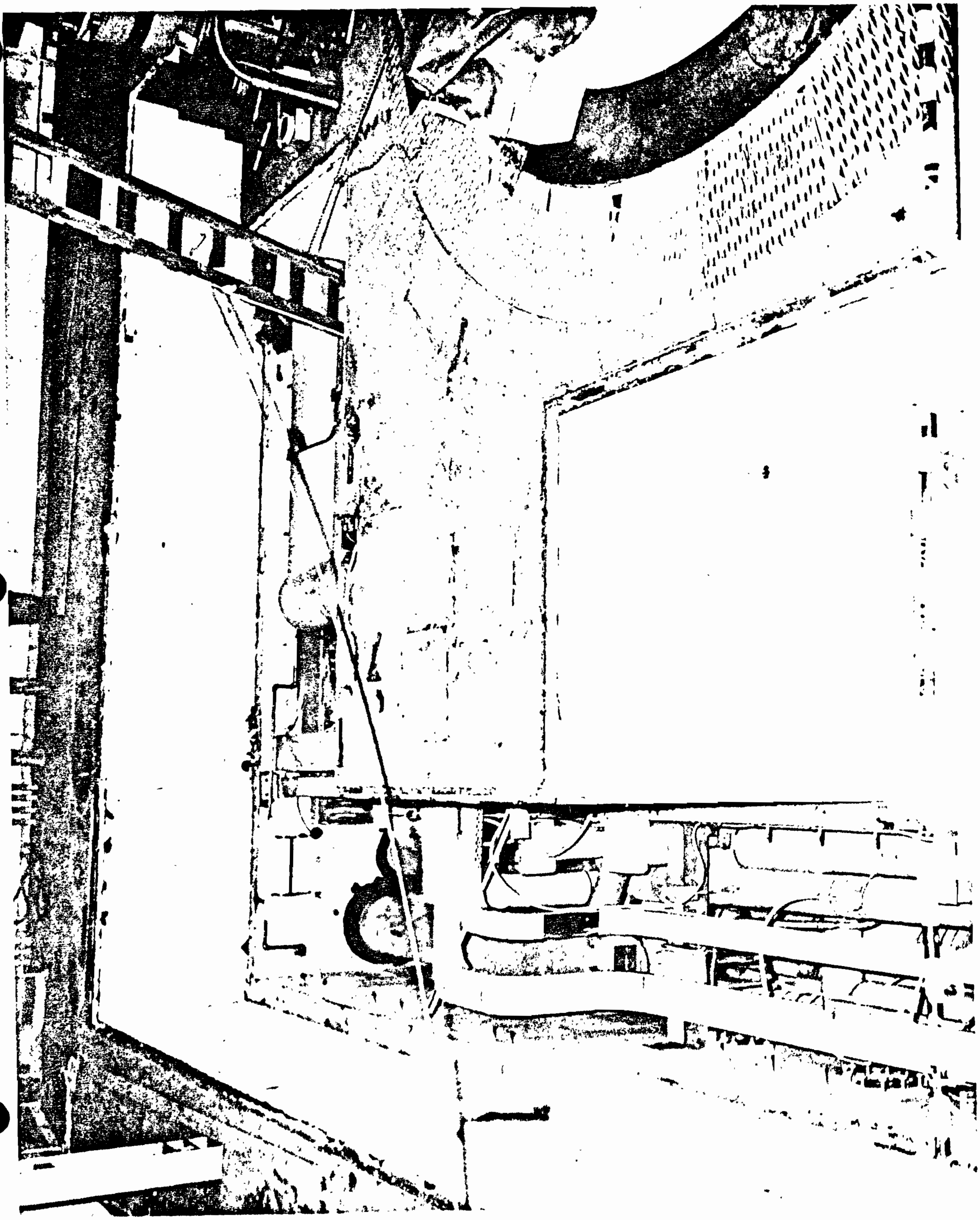


Figure 21. SRE: Main Pipe Gallery

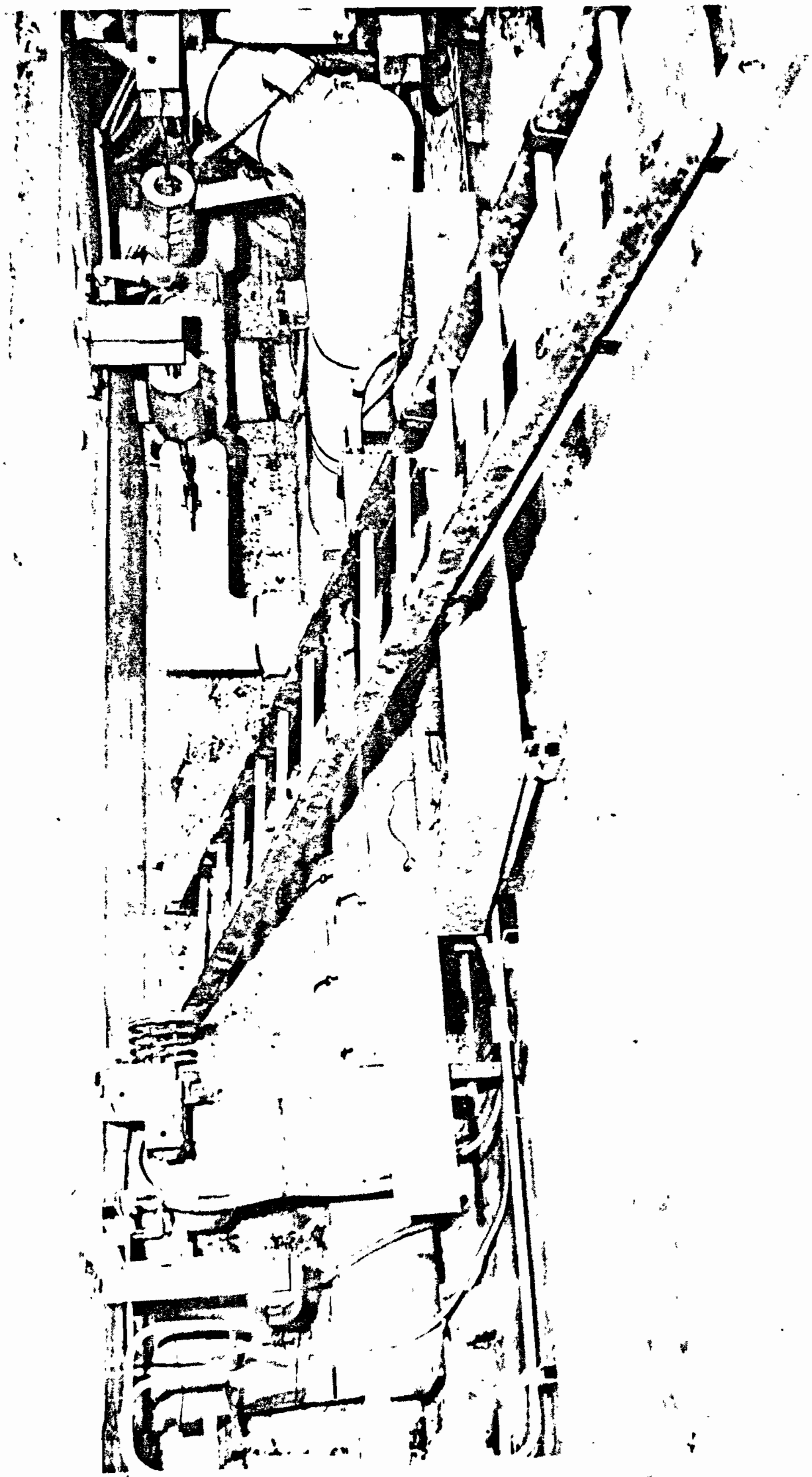


Figure 22. Auxiliary Pipe Gallery

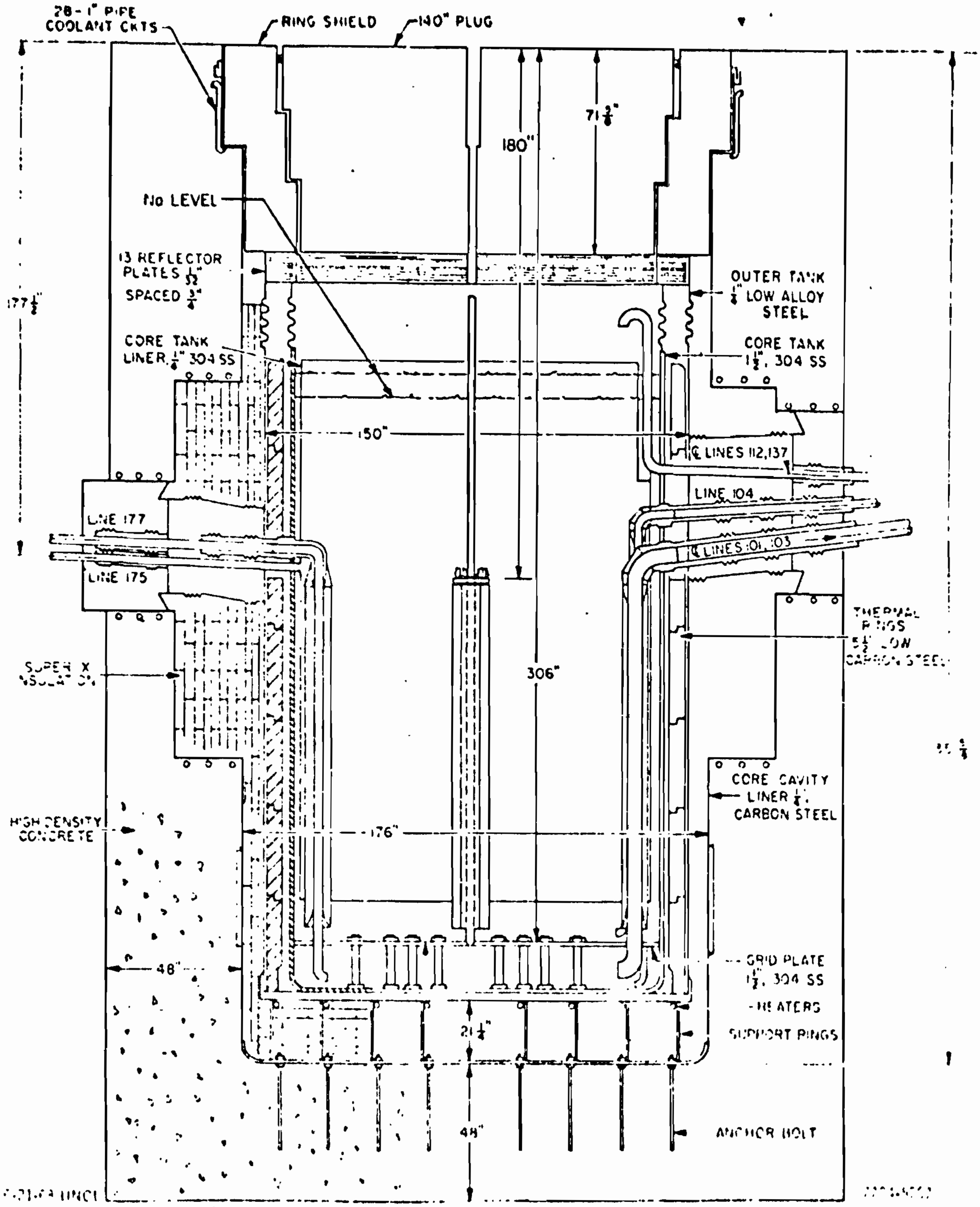


Figure 23. SRE Core Vessel Vertical Section

8.5 <u>Passivation of the Residual Sodium in the Reactor Vessel</u>

The reactor vessels and internals are shown in Figure 23. The reactor vessel has been drained of primary bulk sodium. However, a residual heel approximately two-inch deep exists and pockets of sodium, sodium frost and films are present on the moderator cans and reactor vessel internals. Passivation of this residual sodium will be accomplished by an alcohol or a steam-nitrogen process system. Upon completion of the passivation the reactor vessel will be filled with water. Activity Requirements Document 9 will be prepared to define the requirements for the design, fabrication, installation, and operation of the process system.

8.6 <u>Sodium Service System Dismantling</u>

The sodium service system in Building 153 will be dismantled. The main elements of this system are listed in Table 28.

TABLE 28

SODIUM SERVICE SYSTEM COMPONENTS

1. 2,620 gal Secondary Fill Tank
2. Diffusion Cold Trap Attached to the Bottom of the Secondary Fill Tank
3. 80 gal Transfer Tank
4. Sodium Melt Station
5. Piping Valves
6. Freeze Trap
7. Vapor Trap
8. Electrical Controls and Switch Gear

The secondary fill tank has been drained. However, the diffusion cold trap welded to the underside of the tank contains 5 to 15 cu ft of solid sodium. This trap will be cut loose and sodium passivated in Building 163. After all the sodium systems are removed and the electrical and gas services are deactivated, Building 153 will be razed by the salvage contractor.

8.7 Decontamination of Primary Sodium Fill Tank Vault and the Sodium Service Vault

The primary sodium vault and the adjacent sodium service vault will be decontaminated to acceptable levels. If necessary, all or portions of the concrete walls or floors will be removed to accomplish the decontamination. After decontamination is completed, the above grade concrete structure will be demolished. The rubble will be bulldozed into the vault and sufficient clean fill to level the primary fill tank vault to the existing grade will be placed over the rubble. The sheetmetal house over the sodium service vault will be removed and the walls and floors of the vault decontaminated. The vaults will be backfilled.

8.8 Pipe Galleries Decontamination

The pipe galleries (main and auxiliary) have been previously decontaminated to < 50 dpm/100 cm^2 (Ref. 1). The dismantling activities may add to this contamination; therefore, upon completion of the sodium components removal, a radiological survey will be conducted to determine the need for decontamination. If necessary, surface concrete will be removed to accomplish the decontamination.

8.9 Hot Cell Decontamination and Dismantling

The hot cell area has been deactivated and partially decontaminated (Ref. 1). The hot cell internal areas have been decontaminated to levels of less than 1,000 dpm/100 cm^2. The service and gallery working area has been sealed to exclude dirt. The Demountable Maintenance Shielding Assembly (DMSA) has been decontaminated to a level less than 500 dpm/100 cm^2.

The hot cell area including the hot changeroom, the sump pit, the wash cell valve pit, exhaust system, inert gas system, the CO_2 system, the DMSA, and working and service galleries will be dismantled. Components will be decontaminated if practical; otherwise, they will be shipped for burial. The cell areas will then be decontaminated to acceptable levels. If washing, foaming, or etching prove ineffective for decontamination, the contaminated concrete will be removed.

8.10 Disposal of Reactor Vessel and Internals

Disposal of reactor vessels and fixed internals will require special tooling similar to that used in the Elk River Reactor D&D, as developed by Oak Ridge National Laboratory (ORNL). The reactor vessel fixed internals include the sodium inlet and outlet downcomers and the grid plate. AI has acquired the ORNL design for the rotating mast manipulator used to dismantle the Elk River Reactor. AI will modify the design to fit the SRE reactor vessels, fabricate the manipulator, and develop the underwater cutting parameters in a mockup facility, as described in Section 7.0 above. After developing the cutting parameters, the manipulator will be installed in the SRE and the vessels will be cut up. The cut-up vessel segments from the SRE will be stored underwater in a shipping cask liner until a full cask load is ready for shipment. The activated vessel internals and segments will be disposed of by land burial in the cask liners.

Reactor components (see Table 4) suspended from plugs in the loading face shield will be removed and remotely packaged for shipment to burial. The moderator cans are not irradiated. They will be removed after the vessel is filled with water and the reactor 140" plug is removed.

Development and qualification of remote tooling for reactor fixed internals and reactor vessels removal is described in detail in the SRE Remote Tooling Task Requirements documents.

8.11 Removal of Reactor Biological Shield and Other Activated Concrete

An Activity Requirements document will be prepared describing the techniques to be used in breaking up, containing, and removing the concrete. The primary technique will be blasting, similar to that used in the Elk River reactor D&D. Detailed information was acquired from the Elk River efforts, and will be used extensively for the SRE excavation planning.

The concrete surrounding the wash cells and the fuel storage cells is contaminated and will be removed before the biological shield concrete which contains much more radioactivity. This sequence will establish and prove the techniques for fume, dust and debris control, and will establish optimum explosive loading patterns for breakout of massive concrete. The reactor biological shield demolition will begin from inside the cavity thereby effecting maximum containment of dust and debris. Activated concrete will be placed in containers and shipped to burial.

8.12 Decontamination and Removal of Storage and Wash Cells

The wash cells A, B, and C have been rinsed with water. All material was removed from the storage cells, and shield plugs or covers were installed on each cell.

Piping to the wash cells will be cut out and decontaminated or shipped for burial. If the soil surrounding the wash cells is contaminated, it will be removed and shipped for burial. The concrete surrounding the cells will be removed and the cells pulled out. The cells will be decontaminated if practical or shipped for land burial. All excavations will be backfilled.

8.13 Disposal of Radioactive Waste Systems

The contaminated residue in the two 5,000-gallon storage tanks has been removed. The change room hold-up tank has been flushed and the rinse water pumped out. Storage tanks T1, T2, and T3 have been flushed and the rinse water pumped out. Standing water in the R/A liquid waste sump pit was pumped out and the sump pit cleaned to 5000 dpm/100 cm^2.

The cooling water to the vent compressors has been drained and valved off. The vent system has been purged with air and nitrogen. The R/A vent system is filled with nitorgen to a pressure of 1/2 psig and is sealed off.

All gaseous and liquid radioactive waste system lines and tanks will be excavated and removed. Decontamination of the tanks and piping will be attempted but, if impractical, the equipment will be shipped for burial. The vent system compressors, filters, ducts, and stack will be decontaminated or removed. The stack may remain if a future use is indicated.

8.14 Removal of the Inert Gas System

Much of the inert gas distribution systems, such as the gallery nitrogen cooling system, will be removed during the dismantling of other systems. The nitrogen and the helium inert gas systems have been combined during the deactivation (Ref. 1). These will be shut off as system removal progresses and removed when no longer needed. The remaining supply and distribution system will then be removed. An Activity Requirements Document 22 will be prepared.

8.15 Removal of Remainder of the Kerosene System

The kerosene system has been drained and the portions of the system external to Building 143 removed. The internal portions, which consist of piping in the trenches to the reactor, to the wash cells, and to the storage cells, will be removed by the salvage contractor. The surge tank, the pump, and the compression water coolers in the wash cells will be removed.

8.16 Decontamination and Dismantling of Fuel Handling Machines and Moderator Element Handing Machine

The Mark I and II fuel handling machines and the moderator element handling machine will be decontaminated, dismantled, and the exposed machine components further decontaminated. When impractical to decontaminate to the acceptable levels, the components will be shipped for burial. The remaining components will be scrapped. An Activity Requirements document will be prepared for the dispositon of each machine.

8.17 Decontamination and Disposition of Peripheral Areas

Buildings 163, 724, 686, and the retention pond will be radiologically resurveyed. The walls and floor in Building 163 will be decontaminated or removed as necessary to allow unrestricted future use. Building 724 will be razed. The tanks and structure at Site 686 will be removed. The retention pond will be decontaminated if necessary by removing R/A soil. The pond will be filled with earth and the dam removed. An Activity Requirements document will be prepared for each structure.

8.18 Final Closeout of SRE Facility

This activity will complete the SRE D&D. The activity will include backfilling, repaving, concrete structure removal, and repairs resulting from demolition to the remaining structures. The scope of the work will be dictated mainly by the objective to leave the facilities in a safe, usable condition. Radiological safety is not a consideration at this point in the D&D effort since the remaining level of radioactivity will be below those specified in Table 1, and below those levels developed in the Activity Requirements documents.

9.0 Waste Disposal

Radioactive waste will be packaged at the SRE or shipped to the RMDF and processed for shipping. Liquid wastes will be evaporated at the RMDF and the remaining sludge will be solidified and packaged for burial at a licensed site.

10.0 Radiological Survey

A radiological survey of the SRE facilities has been made and reported in the "Plant Status" section of this plan. Throughout the SRE D&D, health physics surveillance of the operations will be maintained as prescribed in the Operational Safety Manual. A final survey of the site will be made to confirm that the remaining levels of radioactivity are below those specified in Table 1 and in the Activity Requirements documents, and the site will then be available for unrestricted use.

11.0 Documentation

11.1 Procedures

Detailed Working Procedures, based upon the Activity Requirements, will be prepared. The procedures will be written to conform to the requirements of the Operational Safety Plan, Quality Assurance Plan, and the Program Plan. Detailed procedures will be reviewed internally and will be controlled and released by the Engineering Data Release System.

11.2 Reporting

Progress on SRE D&D activities will be reported monthly as stated in the Program Plan.

11.3 Records

The results of radiological surveys of the areas, materials, and equipment will be recorded. A complete accounting of all radioactivity disposed of by the RMDF will be maintained. Photographic coverage of the more significant operations will be made, both in still photographs and in motion pictures.

11.4 Final Report

A final report will be prepared describing the dismantling and decontamination activities. Problem areas and solutions will be highlighted. The final report will include the results of the final radiation surveys and the still photographic information.

REFERENCES

1. Deactivation Plan TI-599-19-001, Rev. A, May 10, 1967.

2. Retirement of the SRE - AI-AEC-12572, August 30, 1968.

3. Special Nuclear Materials Control Program for SRE Fuel Decladding, AI-74-44, March 14, 1975.

4. "Analysis of Neutron Flux in the Shield of the Sodium Reactor Experiment," NAA-SR-2953, October 15, 1958.

5. "Materials for Architecture," Caleb Hornbostel, Rheinhold Publishing Corp., New York, 1961.

6. "Trace Elements in Common Rock Types and Their Relative Abundance in Neutron-Induced Radioactivity Calculations," Roger Paul and Walter Day, PNE-5006, August, 1966.

7. "Kemps Handbook of Rocks," James Kemp, Van Nostrand, Inc., New York, 1958.

8. Reactor Handbook, Vol. I, Materials, "Interscience Publishers, Inc., New York, 1960.

9. "High Density Concrete for Gamma and Neutron Attenuation," ORNL-3074, March, 1965.

10. "Nuclear Radiation Shielding," Nuclear Engineering Handbook, E. P. Blizzard, McGraw Hill, New York, 1958.