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RETIREMENT OF THE SODIUM REACTOR EXPERIMENT

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ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

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REACTOR TECHNOLOGY

RETIREMENT OF THE SODIUM REACTOR EXPERIMENT

SRE STAFF
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ATOMICS INTERNATIONAL
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ABSTRACT

The SRE plant and post-operational programs are briefly described. The retirement program is described in detail together with the alternative plans that were considered. The current status of the plant is described and the means presented whereby the plant will be maintained in its present condition. The time and cost schedule to accomplish the retirement is included. Finally, recommendations are made that would be useful to future retirement efforts.

I. DESCRIPTION

The Sodium Reactor Experiment (SRE) plant is located at the Atomics International Nuclear Development Field Laboratory, in the south-eastern 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 1000 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 North American Rockwell Corporation which AEC has the option to purchase as indicated in Figure 1 of the Reference 1 document.

The SRE is a thermal sodium-cooled reactor using graphite as a moderator and a slightly enriched uranium as a fuel. During operation of the plant two core loadings of fuel were employed; Core I of uranium-metal and Core II of uranium-thorium metal alloy. The fuel is in the form of stainless steel clad rods with NaK or a sodium bond in the annulus between the fuel and cladding. The active core length was 6 ft. Fuel elements hang from plugs in the reactor top shield in channels at the center of hexagonal, zirconium clad graphite moderator elements. A detailed description of the plant as it was during initial operation to February 1964 is available in Reference 2. At this date modifications were initiated for the Power Expansion Program (PEP). The objective of this program was to increase the power by 50% (to 30 Mwt) and increase the outlet temperature 240F° (to 1200°F). These modifications were completed in May 1965. In this program improved components were installed in certain critical areas. Included were new sodium pumps in the primary sodium system, an intermediate heat exchanger, moderator elements, and fuel. The SRE reactor building arrangement is shown in Figure 1, and the Site Plot Plan is shown in Figure 2. A more complete description is available in Reference 1.

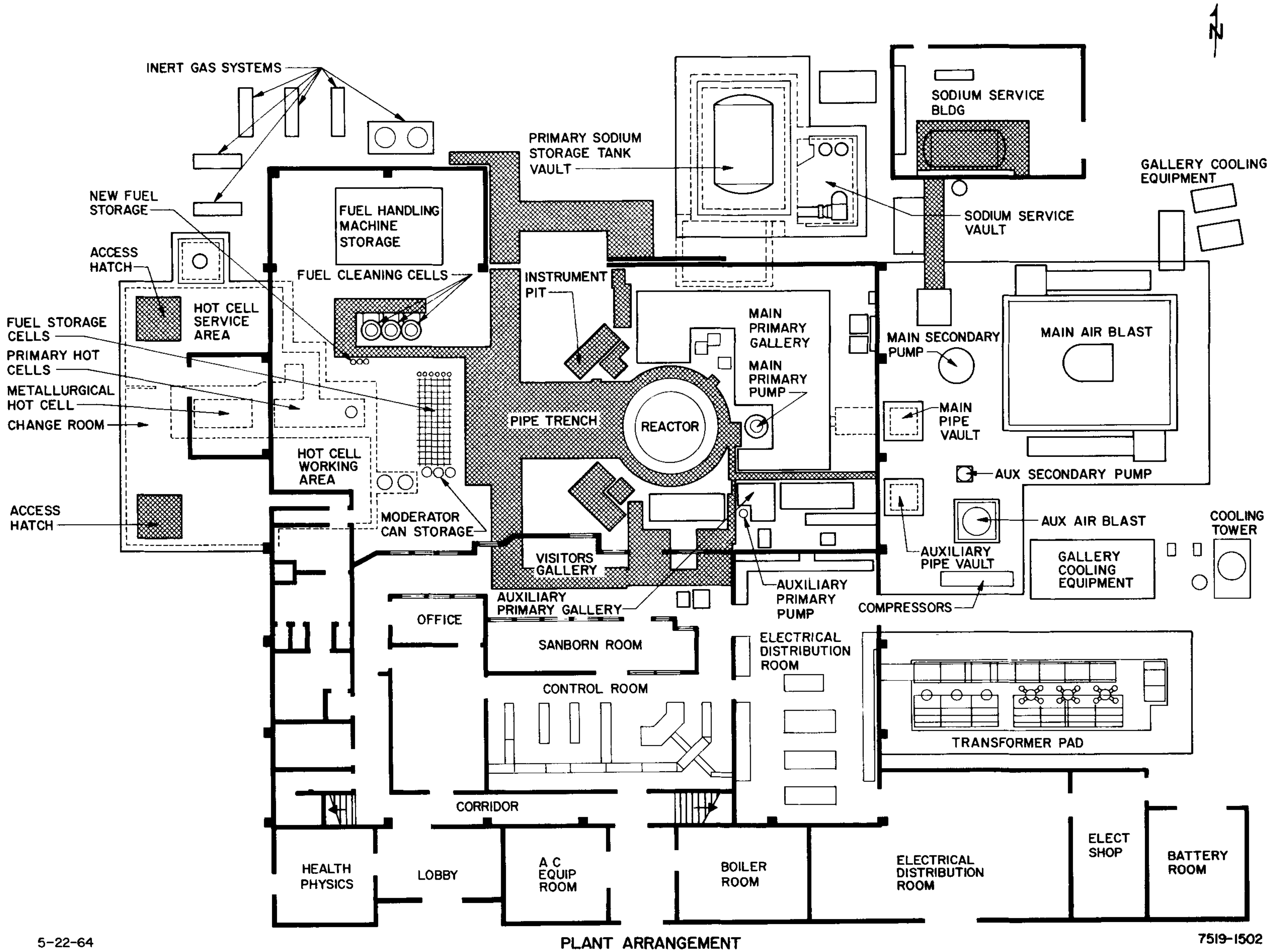


Figure 1. SRE Reactor Building Plan

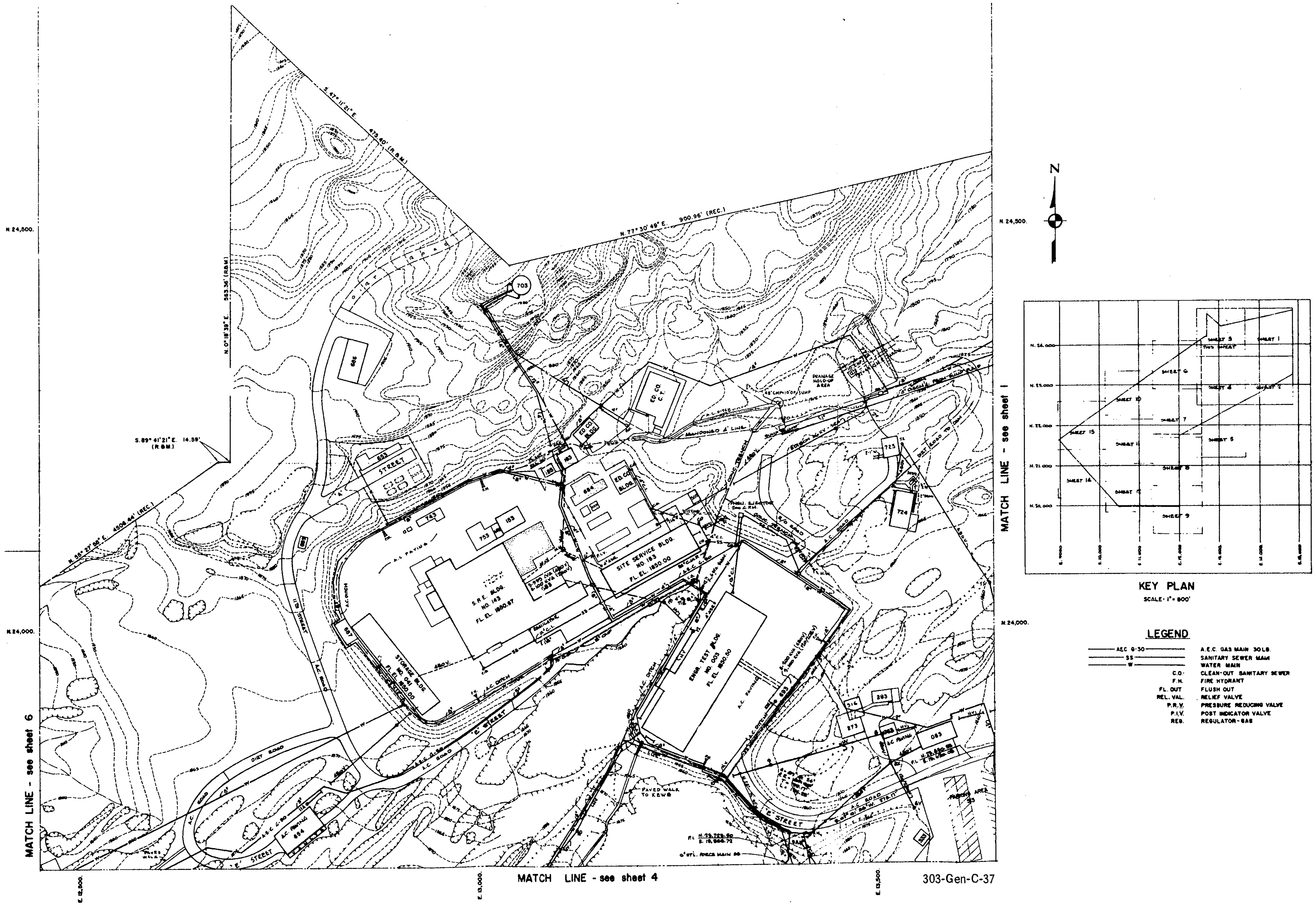


Figure 2. SRE Site Plot Plan

II. 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 1 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. The effects of this long time sodium exposure on the materials in the SRE have been evaluated and reported in the References 3 and 4 reports. Not included in this thermal history is the accumulated operating time during the SRE-PEP program for Na cleanup purposes, this span of time was from May 15, 1965 to September 1967. The time and temperature were:

Operating Time, Main Primary

at $\sim$ 700°F = 4,386 hr

at $\sim$ 350°F = 13,196 hr.

In September 1967, the primary sodium system was shut down and drained to the primary fill tank, and the secondary to 55 gal drums. The plant did not operate as a nuclear plant after February 15, 1964.

The SRE generated more than 37-million kilowatt hours of thermal energy in over the 37,000 reactor operating hours. A summary of the more important operating statistics is presented in Table 2.

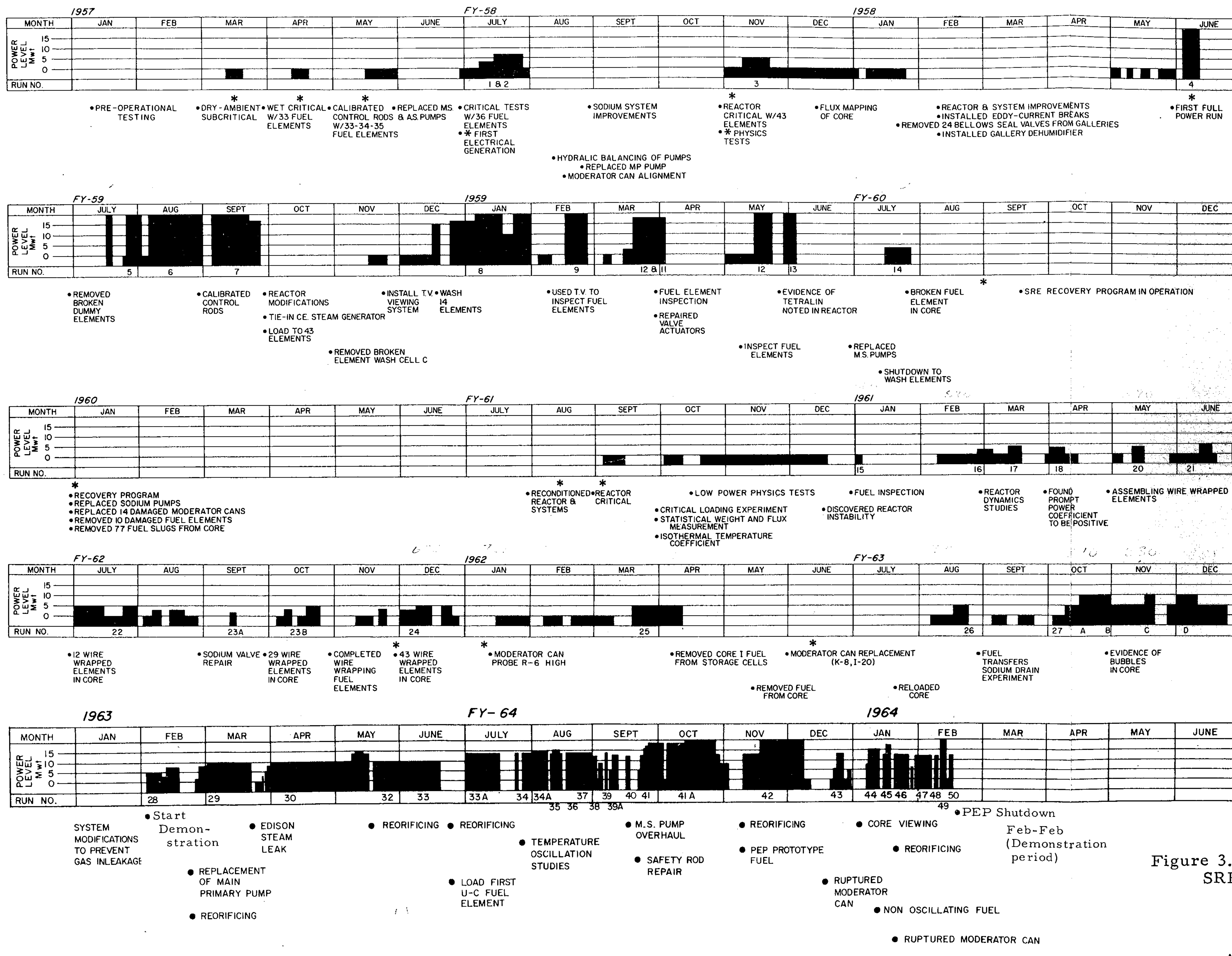


Figure 3. Operating History of SRE (Core I and II)

TABLE 1

SRE HOT LEG OPERATIONAL TIME AND TEMPERATURE*

Temperature Range (°F)	Time (hr)	
	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

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

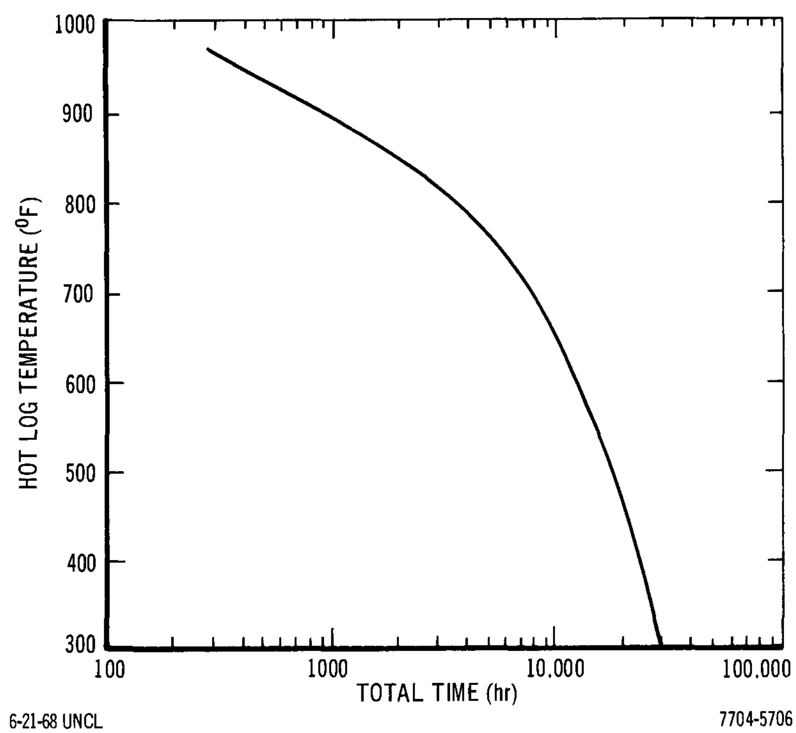


Figure 4. Total Time At or Above a Hot Leg Temperature

TABLE 2

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 $\sim$350°F) (hr)			17,582

III. TERMINATION CONSIDERATIONS

The decision to deactivate the SRE was made in the Fall of 1966. Early in 1967, the plan for retirement of the facility was approved by the AEC after a thorough evaluation of the many alternatives available.

The major decision in evaluating these alternatives was whether to maintain the SRE as a continuing AEC responsibility or to completely rehabilitate the plant site. In the latter case the State of California, via the Bureau of Radiological Health (BRH), would become the regulatory body. If existing regulations were applied, complete removal of radioactive material and contamination from the site would be necessary, which would mean removal of the R/A waste system, the reactor vessel, insulation, thermal shielding, part or all of the concrete reactor shielding, and possibly some surrounding earth.

The BRH personnel recognize that disposal of all radioactive and contaminated material from a reactor will be costly and, possibly, impractical. They also recogize that there are other deactivated plants in California (VBWR and EVESR)* and that others will become obsolete. They anticipate that provisions will eventually be resolved and appropriate legislation enacted to allow on-site burial of reactor components. However, at this time the issue is a sensitive one on which the California Legislature has not taken action.

Between the initial "stored-in-place" status of the SRE and a clean uncontrolled site, there were several alternate plans which could have been followed.[5,6,7] Basically all of them would have delayed the ultimate disposition of the facility which generally would have had the effect of increasing the total cost to the same end point. In all, there were seven basic plans developed for the SRE.

A "Transition Plan" was in effect during the transitory period of evaluation of the several alternative plans. The seven plans were:

1) Radiologically Clean
2) Complete Disassembly, Confined R/A Source
3) Partial Disassembly, Confined R/A Source

*General Electric Reactors at Vallecitos; the Vallecitor Boiling Water Reactor, and the ESADA Vallecitor Experimental Superheat Reactor.

4) Minimal Disassembly
5) Stored-in-Place
6) Standby
7) Retirement

In all cases, it was assumed that the work would be performed under AEC control and prior to BRH jurisdiction. BRH acceptance of the retired facility would have been requested for Plans 1 and 2. It was intended that Plan 1 would have been acceptable under current BRH practices and that Plan 2 might have been eventually acceptable. Continuing AEC responsibility for the retired facility is assumed for Plans 2 through 7. Plan 7, the Retirement Plan, was adopted. Each of these plans will be summarized after the description of a transition phase of the effort.

A. TRANSITION PHASE

The directive to plan the retirement of the SRE came during the course of executing a plan to place the SRE in a "stored-in-place" condition. At that time, most of the specifications and procedures had been completed and reviewed, and the sodium systems were being drained. Shift operation was discontinued on September 15, 1967 after draining the sodium and kerosene cooling systems. In accordance with the current guidance at that time, work plans had been redirected so that only those activities were being continued which were necessary in order to avoid continuing operator attention, or which were required for all of the retirement plans. The change of objective from "storage-in-place" to "retirement" generally represented a reduction in work scope; however, some of the systems were not being prepared for indefinite storage, and additional work was necessary to avoid long-continuing surveillance costs. The activities during the first several months of each plan considered were similar, and large differences would not have appeared until GFY 1969. The activities undertaken in this transitional period were:

1) Inventory revision, circulation, inspection, and negotiation
2) Deactivation of sodium systems including transfer of sodium to drums
3) Deactivation of reactor support systems.

B. PLAN 1, RADIOLOGICALLY CLEAN

All radioactive and contaminated equipment and materials would be removed from the site. Reactor systems would be removed, useful components cleaned, packaged and salvaged, and other items scrapped. The reactor vessel, insulation, thermal shielding, cavity liner, piping, core components, grid structures, and part of the concrete core shielding would be removed and transported to an approved burial site. Contaminated tanks and piping would be removed for either decontamination or disposal as R/A waste. Surfaces would be decontaminated and radioactive soil removed to the extent necessary to allow the facility to be released as a clean facility without a requirement for further surveillance or regulation.

The most difficult problems in this approach are the removal of the reactor vessel — including grids and other internals — and the concrete biological shield. Such removal could probably be done with 12 to 14 feet of water in the cavity to serve as shielding and to control airborne contamination. The reactor vessel would be cut circumferentially about 2 to 4 feet above the top of the core and removed to the high bay area for additional cutting. Highly radioactive material would be cut and loaded into canisters and then into casks for shipment and burial. The thermal shield consists of rings of mild steel 5-1/2 inches thick which would be cut vertically for removal. The inside face of the concrete shield is expected to have a lower radiation level. It would be cut by flame cutting and/or core drilling and sawing into pieces for shipment and burial as required. After removal of radioactive material, the reactor cavity and other pits would be filled with sand and gravel (or compacted fill) and surfaced with concrete to provide a surface contiguous with the surrounding area.

The costs of the physical aspects of this plan were greater than for in-place burial approaches; however, there were no significant unresolved regulatory restrictions and, upon completion of the efforts, there would be no continuing problems or costs. Also, the activity would provide a valuable contribution to reactor deactivation technology, which is needed for the long range program of nuclear power development.

C. PLAN 2, COMPLETE DISASSEMBLY, CONFINED R/A SOURCE

Radioactive and contaminated equipment and materials would be removed from the site or placed in the sealed reactor cavity or other designated shielded cells. Sodium systems would be removed, useful components cleaned, packaged as required, and salvaged, and other items scrapped. The reactor vessel, as well as other internal and surrounding radioactive components, would be left in place and sealed, as in the Hallam Nuclear Power Facility (HNPF) retirement program.[8] Prior to sealing, as much sodium as possible would be removed from the reactor vessel and that remaining would be reacted with steam by the methods employed at HNPF.[9] Excavations which would have no potential use in the remaining facility would be back-filled and surfaced with concrete to eliminate the hazard of open pits. The site would be available for other use. This plan is essentially the same as Plan 1, except that radioactive components, including the core vessel and shields, would be confined in place rather than being removed. Studies would be necessary to establish that the specific hazards were eliminated and that residual radioactivity was inaccessible via any reasonable means. Continuing surveillance requirements would be minimized. Approval of the State of California would be sought to classify the site as an unlicensed installation.

The approach avoids the costs of removing the reactor vessel and other large radioactive components. However, the costs of engineering studies, safety reviews, negotiations, etc., would be significant, as evidenced by the HNPF program effort. A clear-cut resolution of the case with the State of California could be expected to take several years and some continued surveillance of the site would probably be necessary.

D. PLAN 3, PARTIAL DISASSEMBLY, CONFINED R/A SOURCE

Funding requirements could have been reduced from Plan 2 by completing only those activities which were required to make components available and to confine the radioactive materials so that the main buildings would be available for other purposes. Radiological hazards would be neutralized and access to the plant would be uncontrolled.

Sufficient analysis would be done to determine disposal requirements for the primary sodium systems, the storage cells, and the waste systems, and to demonstrate that no radiological hazard exists. Residual sodium in the reactor could be drained and/or reacted.[9] Primary sodium piping would be scrapped. Sodium valves would be salvaged. Other sodium components would remain in place with internal inert gas pressure maintained until they are claimed by another program. As in Plan 2, access to the reactor, storage cells, wash cells, and other spaces containing grossly contaminated components would be impossible without power tools or cutting torches. Component handling machines, instrumentation, and electrical equipment would be salvaged, and the SIR steam generator would be scrapped.

Funding would be required in following years for equipment surveillance and for salvaging all unclaimed equipment. No attempt would be made to work out arrangements for the future with the State of California though such arrangements would be eventually required.

E. PLAN 4, MINIMAL DISASSEMBLY

This plan reduced costs for the initial and following fiscal years. When adequate funding became available, disposal of the plant could be completed and radiological and other industrial hazards neutralized. Until then, controlled access would be required.

Unirradiated fuel would be stored elsewhere at the Nuclear Field Laboratory, and the Radiation Alarm System at SRE would be removed. Nonfuel core components which could not be stored in the reactor or the storage cells because of space limitations would be scrapped. Primary and secondary systems would remain in place. Components would be removed from these systems when required for other programs. The SIR steam generator would remain in place. Portable instrumentation, material, and spare parts would be salvaged.

This plan required radiological and equipment surveillance on a continuing basis until funding became available to complete disposal of the plant.

F. PLAN 5, STORED-IN-PLACE

The objectives of this plan were to put the plant in a standby condition, to minimize costs, and to store systems and components under a condition in which they would not deteriorate. It would be expected that some components would become available to other AEC programs or subjected to destructive tests in support of these programs. The steps to achieve this plan included:

1) Drain the primary and secondary sodium system to the fill tanks, and the excess secondary system sodium into drums.

2) De-energize all heaters, allowing sodium to freeze in the fill tanks.

3) Drain all kerosene and water systems.

4) Convert helium system to N_2.

5) Mothball all auxiliary systems and equipment.

6) Separate new fuel from the hanger hardware and store above ground.

7) Provide periodic maintenance and safety inspection including maintenance of N_2 cover gas pressure and pumping out of pits during the rainy season.

G. PLAN 6, STANDBY

This plan was a slight variation of the Deactivated Plan (Plan 5) described above. The objective of the plan was to maintain complete "re-start" capability. Consequently no equipment would be removed from the systems.

H. PLAN 7, RETIREMENT

This plan was approved by the AEC and was used for the retirement of the SRE. Consequently, it will be discussed in detail in the next section (Section IV) of this report.

IV. RETIREMENT PLAN

The objective of the Retirement Plan is to prepare the SRE plant (at a minimum of cost) for an indefinite period of storage prior to disassembly. The activities to meet these objectives are similar to those required for the Stored-in-Place plan (Plan 6 of III-G above), except that equipment which is not built into the systems may be removed. In addition some equipment would not be maintained, i.e., (1) SIR steam generator and (2) uncontaminated support facilities. Also fuel storage would be accommodated at an off-site location in the Radiation Materials Disposal Facility (RMDF) and New Fuels Storage Building.

The basic requirements of the plan include:

1) Components must be protected from corrosion and other damage until those to be utilized in other programs have been identified, removed and transferred to respective programs.

2) Radioactive materials must be removed and/or stored in an acceptable manner.

3) Portable equipment and materials which are difficult to secure and/or which have general utility in other programs are to be transferred to and stored by the AI Equipment Utilization Unit.

4) Surveillance requirements must be defined and arrangements made to secure the facility and to carry out inspection and servicing easily and safely.

5) Assistance must be provided for Southern California Edison Company when it disconnects and removes its equipment from the site.

6) Surveillance must be provided following completion of deactivation activities.

The following outline identifies the activities included in this plan (the specifications and procedures referenced are discussed in greater detail in Table 3):

I. Planning

   A. Electrical Subsystem

      1. Revise specification SS599N70001

      2. Prepare procedure PR-599-70-001

B. Fuel

1. Revise specification SS599N28001
2. Prepare procedure PR-599-28-001

II. Plant Deactivation

A. Sodium System

1. Transfer secondary sodium to drums
2. Cool all components to ambient temperature
3. Supply with nitrogen at approximately 1 psig
4. Cover exterior valve operator boxes
5. Provide dust and/or weather covers for pump motors
6. Provide temporary gallery access ladders and covers

B. Inert Gas

1. Deactivate helium system except for helium makeup lines
2. Tie in bottled nitrogen to sodium system

C. Radioactive Systems

1. Drain and flush liquid waste system
2. Purge gaseous waste system
3. Decontaminate sump pit and wash cell valve pit as required
4. Replace stack filter
5. Shut down vent compressor and shut off cooling water
6. Install sump pit blocks
7. Transport all remaining waste to RMDF
   a. Tanks 1, 2, and 3
   b. Sanitary liquid waste tank
8. Open main circuit breakers to
   a. Liquid waste system
   b. Gaseous waste system
   c. Sanitary waste system
   d. Wash cells
9. Install shield blocks for gaseous waste vaults
10. Drain wash cell steam and water systems

D. Fuel and Moderator Handling Machines

1. Decontaminate external surfaces
2. Park fuel handling machine
3. Install dust covers on Mark I and II fuel handling machines and moderator handling machine
4. Open circuit breakers for cranes

E. Control and Instrumentation

1. Install dust cover over control rod drive storage rack
2. Open circuit breaker to eddy current brakes
3. Open all control power circuit breakers including radiation monitors
4. Shut down instrument air system

F. Hot Cell

1. Decontaminate main and portable hot cells
2. Survey exhaust filters
3. Shut down ventilation system

G. Electrical

1. Open all process power circuit breakers
2. Deactivate batteries, MG sets and diesel generator
3. Provide power for emergency paging system and perimeter lights, etc.

H. Auxiliary Supporting Facilities

1. Area housekeeping, inventory, and R/A surveys
   a. Building 041
   b. CERF
   c. Sodium cleaning facility
2. Shut down heating, ventilating, and plant air systems
3. Survey high-bay exhaust filters

I. Core III Fuel

1. Remove fuel bundles from hanger rods
2. Return hanger rods to storage cells
3. Encapsulate fuel and transfer to New Fuels Storage Building
4. Remove both RAS units

III. Continuing Maintenance and Surveillance

A. Procedures

1. Maintain nitrogen pressure on systems
2. Remove water as necessary
    a. Pits, vaults, and galleries
    b. Dehumidification tanks
    c. High-bay trenches
3. Fuel surveillance
4. Semi-annual facility inspection

B. Review and Approval

IV. Documentation

A. Log Book

1. Console
2. Shift supervisors'
3. Recovery program
4. PEP construction

B. Maintenance Manual

C. Complete Drawing File

D. Test Procedures

E. Modifications Procedures

F. Progress Reports

G. Area Inventories

H. Topical Reports

1. SRE-PEP pump performance
2. Carbon removal from SRE sodium
3. Non-nuclear performance of SRE-PEP systems

I. Photograph and Slide File

J. Technical File

K. Plant Log Sheets

V. Component and Equipment Dispersal

1. Transfer all portable components and equipment to Equipment Utilization for disposal as scheduled in TI-599-19-102.[10]

2. Transfer material to other units.

3. Post equipment location as necessary on master copy of inventory

VI. Final Report

This requirement is satisfied by the subject report.

To expeditiously and safely carry out this effort, specifications and procedures were prepared to establish the standards, quality assurance and follow-on surveillance requirements for the work as well as the procedures for accomplishment of the work under applicable AI and SRE Standard Operating Procedures (SOP). Important among these standards was the establishment of radiation levels and contamination allowed for the equipment and areas during the retirement period. In addition, particular attention was directed to assuring that the systems containing residual radioactivity would retain their integrity, within the framework of the Surveillance program (see Section VIII), to detect deterioriation that might result in the release of radioactivity or to detect inleakage of water that might result in local radioactive contamination.

These documents were prepared by the SRE operating staff and were reviewed and approved by the directors of the Nuclear Operation, Health Safety and Radiation Services, and Facilities and Industrial Engineering Departments, or their delegated representatives, to make certain that all procedures could be performed in a safe and expeditious manner. They were also reviewed and/or approved by the Program Office Project Manager and were then submitted to the AEC for review and/or approval. The approval by the Facilities and Industrial Engineering Department was required because of its responsibility for the follow-on surveillance of the plant. In some cases additional approvals were required; these exceptions are noted on the listing of Specifications and Procedures in Table 3. Also included in this table is a short summary of the area, equipment, or systems that are involved within the scope of work for each specification and/or procedure. The schedule of these activities, as they occurred, is diagramed in Figure 16 (see Section VII). This schedule established a sequence of events which were logical, which assured that systems were deactivated only when their further use was not required, and which were safe.

As the work was completed, leaving the equipment and system prepared for continuing surveillance, the specification and/or procedures were signed off to indicate satisfactory completion of the work and acceptance by the Facilities and Industrial Engineering Department of the on-going surveillance and maintenance requirements established by the specification and/or procedure.

During the Retirement effort, weekly highlights and monthly progress reports have been submitted by the SRE supervision to the program and functional organizations. In addition, several topical reports have been issued during this period. (These reports are included in the listing in Appendix A, Bibliography of SRE reports.)

TABLE 3
(Sheet 1 of 4)

SPECIFICATIONS AND PROCEDURES*

Area, System, Parts, or Equipment Involved	Specification Number	Procedure Number	Special Approvals†
Deactivation Requirements for the SRE System - General Specifications	SS599N19001		Industrial Security
Documentation Requirements for the SRE Deactivation	SS599N19002		
Storage Requirements for SRE Fuel Parts Fuel Clusters (to be stored in Fuels Storage Vault, Bldg. 064) Shield Plugs Hanger Rod	SS599N28001	PR-599-28-001	Reactor Fuels Committee of Reactor Hazards Review Panel
Destruction Requirements for Core Components Parts Control Elements Moderator Elements Core Heaters Neutron Source Dummy Fuel Elements Miscellaneous Parts Equipment Pile Oscillator Miscellaneous Equipment Areas High Bay Storage Bldg. 041 Core Vessel	SS599N28002	PR-599-28-002	
Deactivation Requirements for Obsolete Core Components Parts Lazy Susans Na Temperature Measuring Probe Dummy Fuel Elements Moderator Can Mockup Core II Moderator Cans Channel Thimbles	SS599N28003	PR-599-28-003	
Deactivation Requirements for Hot Cell Areas Hot Change Room Sump Pit Hot Cell Internal Working and Service Gallery Wash Cell Valve Pit Equipment Exhaust Filters Demountable Maintenance Shielding Assembly	SS599N29001	PR-599-29-001	

TABLE 3
(Sheet 2 of 4)
SPECIFICATIONS AND PROCEDURES*

Area, System, Parts, or Equipment Involved	Specification Number	Procedure Number	Special Approvals†
Deactivation Requirements for Sodium Subsystem Systems Main Secondary Auxiliary Secondary Primary Sodium Service Primary Sodium Equipment Associated With Above Systems Areas Primary Fill Tank Vaults Main Gallery Sodium Service Vaults Reactor Core	SS599N31001	PR-599-31-001	
Deactivation Requirements for the Radioactive Liquid Waste Subsystem Equipment Storage Tank - 5,000 gal Change Room Holdup Tank Storage Tanks, T-1, 2, 3 Areas Wash Cells R/A Liquid Waste Sump Pit	SS599N63001	PR-599-63-001	
Deactivation Requirements for Radioactive Gaseous Waste Subsystem System Vent System Equipment Vent Compresser Area Vent Compressor Vault	SS599N63002	PR-599-63-002	
Storage and Disposal of Hazardous Waste Parts and Equipment Irradiated SRE Neutron Sources Drums of Na Drums of Waste Cannisters of Waste Dummy Element Flow Orifices Hot Cell Exhaust Filters Radioactive Na Piping Area Moderator Cell CERF (Bldg. 163)	SS599N63003	PR-599-63-003	

TABLE 3
(Sheet 3 of 4)

SPECIFICATIONS AND PROCEDURES*

Area, System, Parts, or Equipment Involved	Specification Number	Procedure Number	Special Approvals†
Deactivation Requirements for the Kerosene Cooling Subsystem System Kerosene Equipment Cooling Towers Emergency Engine Kerosene Pumps	SS599N64001	PR-599-64-001	
Modification and Deactivation Requirements for the Inert Gas Subsystem Systems Cooling and Dehumidification Instrument Air Nitrogen Helium Equipment Cooling Towers Gallery and Vault Cooling Tanks Dehumidifier Freon Units Gallery Cooling Tower Cooling Waste Tower Evaporator Cooler Evaporator Condenser	SS599N69001	PR-599-69-001	Systems Development and Test Group
Deactivation Requirements for Electrical Subsystem Systems Emergency Power Normal Power Distribution Equipment Motor Generator Set Battery Bank Diesel Generator Emergency Power Supply	SS599N70001	PR-599-70-001	
Deactivation Requirements for Control and Instrumentation Subsystem Systems Reactor Control Process Instrumentation	SS599N76001	PR-599-76-001	

TABLE 3
(Sheet 4 of 4)
SPECIFICATIONS AND PROCEDURES*

Area, System, Parts, or Equipment Involved	Specification Number	Procedure Number	Special Approvals†
Deactivation Requirements for Component Handling Machines Equipment Mark I Fuel Handling Machine Mark II Fuel Handling Machine Moderator Handling Machine Long Gas Lock Pump and Plug Cask Transport Casks Miscellaneous Moderator Handling	SS599N81001	PR-599-81-001	
Requirements for Environmental Surveillance Areas Secondary Pad Installation Liquid Waste, Outside Area Transformer Substation Dormant Storage, Outside Area Tetralin Heat Exchanger Primary Fill Tank Vault Sodium Cleaning Facility and Pad North Ditch Pond	SS599N90001		
Deactivation Requirements for Uncontaminated Support Facilities Systems Electrical Power Emergency System (Fire and Radiation Alarm) Ventilation Domestic Water Areas Office and Control Rooms, Above Related Equipment Rooms	SS599N93001		
Deactivation Requirements for Contaminated Support Facilities Areas Bldg. 143, High Bay Bldg. 041, North End Bldg. 163 (CERF)	SS599N93002	PR-599-93-002	

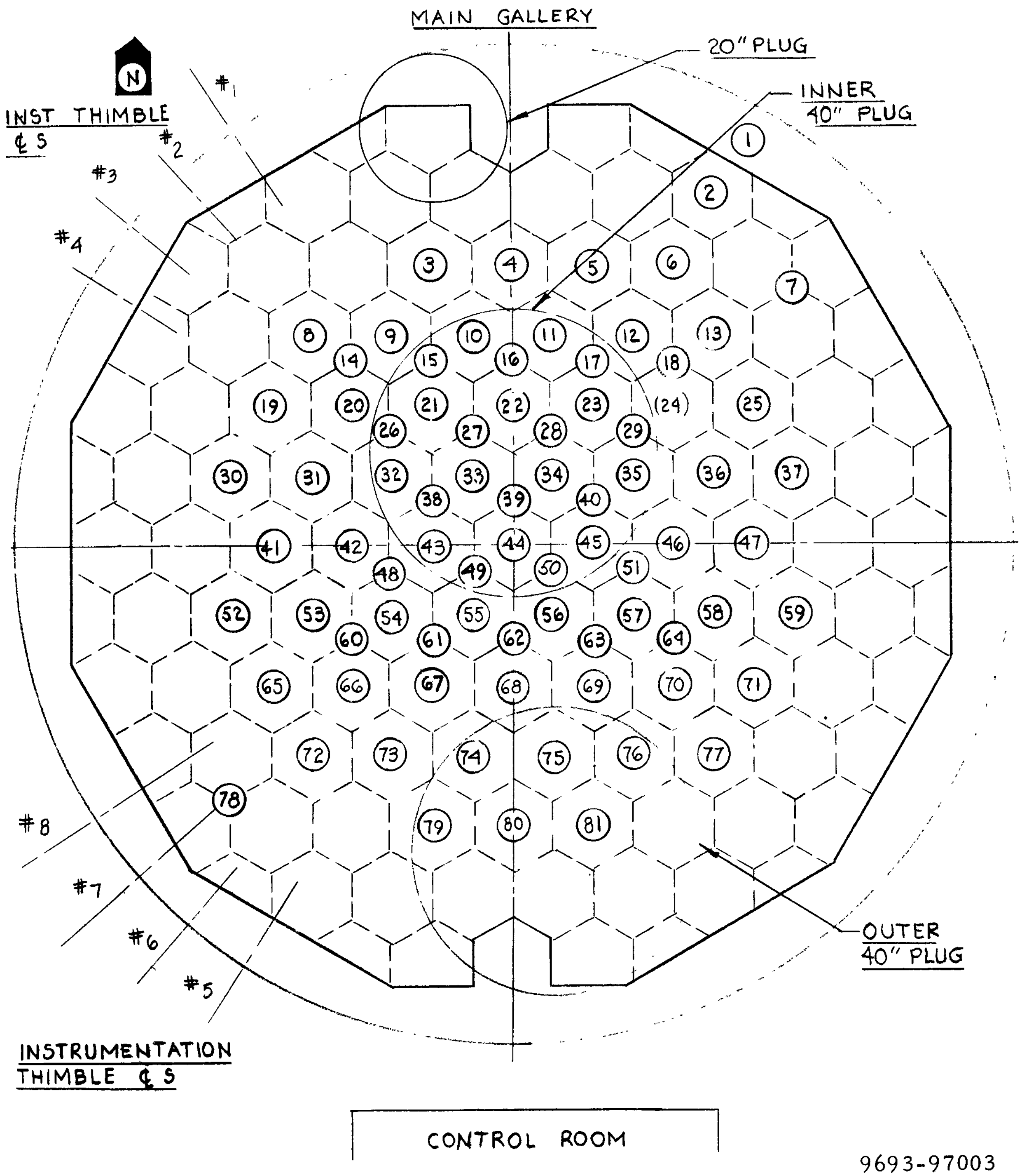
*These specifications and procedures are on file at Atomics International-Engineering Data.
†See text for normal approval requirements.

V. EXCESS PROPERTY

A complete inventory was prepared early in the Retirement Program to allow for AEC dispersal of equipment to other AEC funded programs on a priority basis. An updated inventory(10) of SRE equipment, prepared after the decision to retire the plant, was circulated to other AEC programs.

Removal and shipment of equipment from the SRE to AEC-designated recipients was completed by August 1968. First priority was given to AEC-RDT programs including FFTF, LMEC, EBR-II, and ZPPR. Equipment also was supplied to other secondary priority AEC programs. After filling these requests, the remaining easily removable items were excessed via normal procedures for surplus AEC property. If in the future, requirements arise for remaining items, their value will be weighed against cost of removal and the effects on the remaining system.

Appendix C is a listing of the equipment and components transferred offsite to designated recipients. This listing is limited to only those items which have been removed or approved for removal as of June 1, 1968.



9693-97003

Figure 5. SRE Reactor Loading Face

VI. FACILITY AND SYSTEMS PHYSICAL AND RADIOLOGICAL STATUS

The purpose of this section is to describe the physical and radiological status of the plant as it will exist on June 1, 1968. The maintenance and monitoring necessary to assure that the facility remains in this status will result from the follow-on surveillance program by the Facilities and Industrial Engineering Department. The surveillance program is described in Section VIII of this report.

A. FUEL ASSEMBLIES

The shield plug/hanger assemblies have been disassembled from the unirradiated Core III fuel. The shield plug/hanger assemblies are stored in the storage cells in the high-bay floor of Building 143. All fissile material has been moved from the SRE site for storage in the Fuel Storage Vault. Storage there will be under the control of the Nuclear Materials and Waste Management organization. Appendix D includes the inventory of Core III and experimental fuel elements. The surface contamination levels are also included.

B. CORE COMPONENTS

The reference specification and procedure for the retirement requirements for the core components are SS599N28002 and PR-599-28-002 respectively (see Table 3 also). The storage of the core components is as noted in the following subsections.

1. Core Storage

The following items are stored in the noted core positions of the reactor as indicated on the core loading face diagram in Figure 5.

Item	Location
1) Control Rods (4)	R-21, -23, -67, -69
2) Safety Rods (4)	R-32, -35, -54, -57
3) Core Heaters (10)	R-4, -7, -14, -16, -25, -41, -49, -50, -62, -78

4) Dummy Fuel Elements (6)	R-42, -43, -44, -45, -46, -47
5) Sodium Level Probes (4)	R-60, -61, -63, -64
6) Pile Oscillator (Inner Assembly) and spare Safety Rod Thimble	R-68
7) Core Exposure Facility	R-2
8) Moderator Temperature Probes (2)	R-18, -39
9) Fission Product Monitor Plug	R-17
10) Core II Shield Plug and Hanger Assembly (25)	R-3, -5, -6, -9, -10, -11, -12, -13, -19, -30, -31, -36, -53, -59, -65, -66, -71, -72, -73, -74, -75, -76, -77, -80, -81
11) Spare Safety Rod Boron Assembly	R-52
12) Experimental Thimbles (3)	R-8, -52, -79
13) Neutron Source	R-37

2. SRE Storage (Building 143)

The following items are stored in the Safety and Control Rod Storage Racks in the SRE high-bay:

1) Safety Rod Tower and Drive Assembly (5)

2) Control Rod Tower and Drive Assembly (4 plus 2 spare drive units)

The moderator top latch grapple is stored in the Moderator Handling Machine located in the SRE high-bay.

The long shield plugs are stored in the storage cells (numbers are marked on the floor cover plates) in the high-bay floor area.

3.0 in. diameter	4, 9, 11, 16, 17, 23, 24, 36, 40, 41, 46, 60, 61, 66, 73, 77, 84
3.5 in. diameter	5, 26, 27, 32, 33

3. Off-Site Burial

Some core components — for example, grapples, core lights, Cerrobend heaters, and 3- and 3.5-in. loading face shield plugs — were packaged and shipped off-site for burial.

C. HOT CELLS

The hot cell area in the SRE (Bldg. 143) has been deactivated according to the specification and procedure SS599N29001 and PR-599-29-001 respectively. The present status is outlined below.

1) The internal A and B hot cells are below a contamination level of 500 dpm/100 cm^2 except for the two fuel storage thimbles. These two thimbles are below a contamination level of 2500 dpm/100 cm^2. They have been sealed and Health Physics has tagged them to the actual contamination level.

2) The service and operating galleries of the hot cells are below a contamination level of 50 dpm/100 cm^2.

3) Access blocks have been placed to close the internal areas of the hot cells, and signs of radiation and contamination levels have been posted.

4) The water valves for sinks and showers are closed and tagged off.

5) Electrical power is on.

6) The ventilation system is operating.

7) The CO_2 or N_2 gas system is not operative.

8) The internal surface of the Demountable Maintenance Shielding Assembly (DMSA) contamination is below 5000 dpm/100 cm^2. The access block is installed and the personnel access door is closed.

9) An automatic sump pump has been placed in operation in the sump pit to remove any water that may accumulate.

10) Contamination smear surveys on interior areas of the ventilation ducts adjacent to each filter show that the following β,γ contamination is present (as of January 11, 1968). (Area smeared was 100 cm^2 in each case.):

Sample No.	Description and Location	Results (d/m)
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 floor (under filter)	1,479
9	West plenum floor (under filter)	2,118

D. SODIUM SYSTEMS

Sodium systems include the Main Primary Sodium System, Main Secondary Sodium System, Auxiliary Primary Sodium System, Auxiliary Primary Sodium System, Auxiliary Secondary System, and Sodium Service System. The reactor core is considered common to the Main and Auxiliary Sodium Systems. The system cooling rates used were established as a result of the structural analysis of the stresses in the reactor vessel (particularly, the stresses in the bellows) that could be expected from cooldown.(11) A drain of these systems has been effected as described below. There remains, however, residual sodium throughout the system which is now at ambient temperatures. All circuit breakers to all line heaters throughout the sodium system have been opened. A nitrogen atmosphere at ~1/2 psig is maintained throughout the system by the Modified Helium (low nitrogen) Inert Gas System (see Section VI.I).

Plastic covers are installed over the main primary, auxiliary primary, main secondary and auxiliary secondary pump motors, the magnetic clutches, and the oil seals. Rust inhibitor has been applied to unprotected surfaces. The side louvers of the main air blast heat exchanger are open. The exterior remote valve operator housings have been covered to prevent water inleakage. All seal boxes leading into the main, main auxiliary, primary fill tank, and sodium service vaults have been bolted down to seal against water inleakage.

Approximately 8,000 gallons of primary sodium have been drained to the primary fill tank and the sodium is at ambient temperature. The primary sodium tank storage vault blocks have been sealed to prevent inleakage of water. The radiation map of the tank is shown in Figure 6a. The contamination level on the vault surfaces is less than 50 dpm/100 cm^2.

The secondary sodium was drained to sixty 55-gal drums under a positive argon pressure. These drums are stored in an open-sided storage building (Building 007) at the Santa Susana site. A heel of 5 to 15 ft^3 of solid sodium remains in the Secondary Storage Tank.

Figures 6b and 6c are the radiation maps for the main and auxiliary primary sodium systems. Figure 6d is the radiation map for the sodium services piping in the sodium service vault.

The access plugs have been removed from the following areas: primary fill tank vault, main gallery, sodium service vault, and primary drain pump vault. These areas have been provided with safety barrier chains with radiation warning signs conspicuously placed.

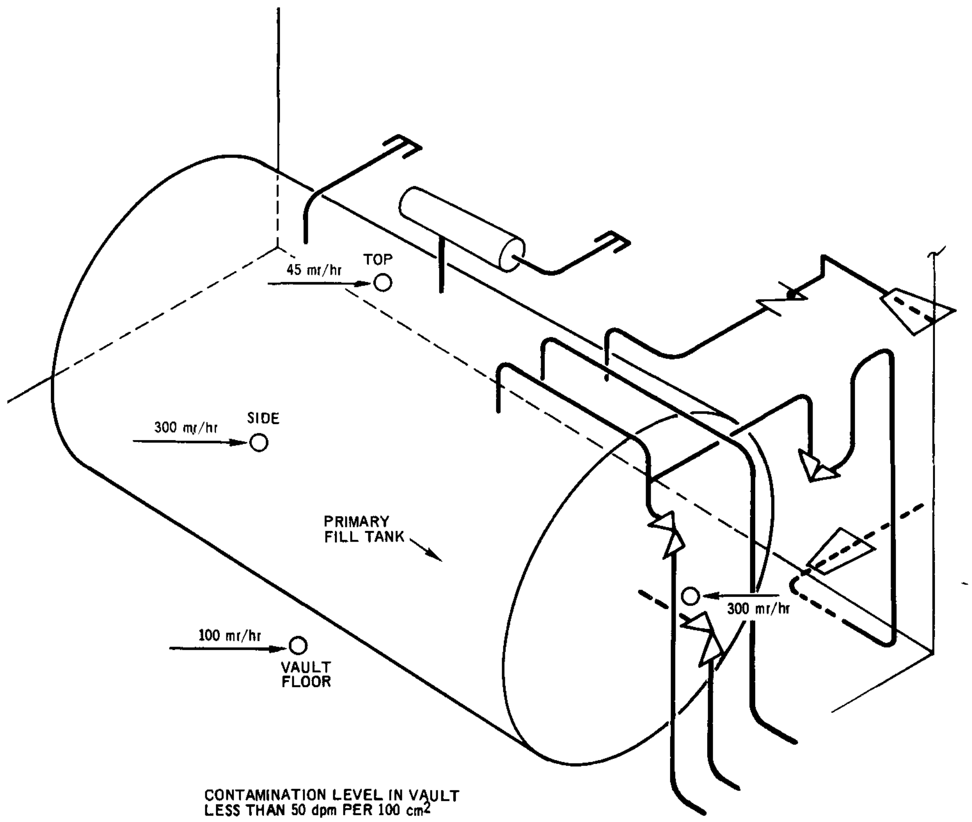
Figure 7 is a cross-section of the outer tank, reactor vessel and core tank liner, with the structural materials indicated. A photograph of the outer tank, taken during construction, is shown in Figure 8. The outer tank, of low alloy steel, may be subject to deterioration from the moisture present over a long period of time.

E. SIR STEAM GENERATOR

The SIR Steam Generator was a prototype unit for the USS Sea Wolf which was made available for further testing at the SRE. At the SRE it operated as a secondary heat dump for a short period of time.[12] It has been stored in place for some time, and is no longer considered safe to operate; consequently, it should be scrapped. The thermal insulation has been removed and all electrical heaters and thermocouple leads have been cut off.

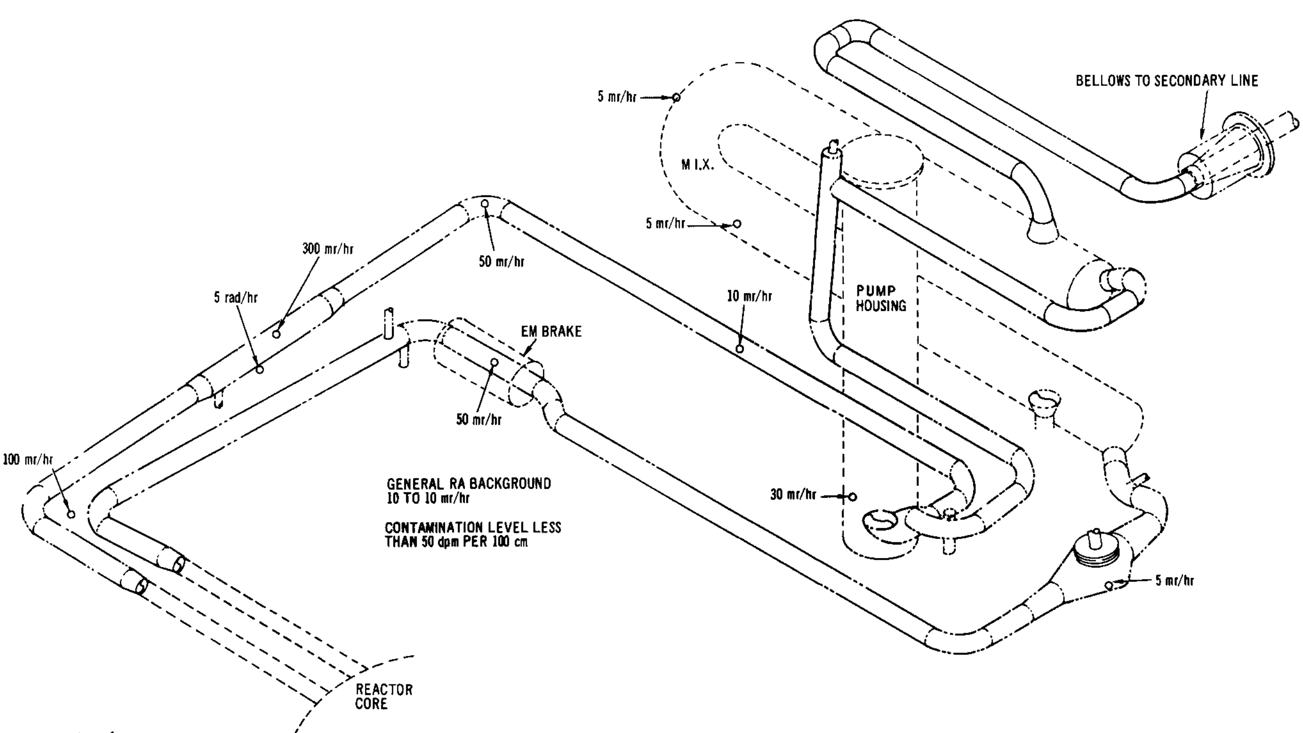
F. R/A LIQUID WASTE SYSTEM

The contamination and radiation levels present at several locations within the liquid waste system, and in areas associated with this system, are indicated in Figure 9. All liquid waste (except inaccessible heels) has been sent off-site.



5-14-68 7704-5705

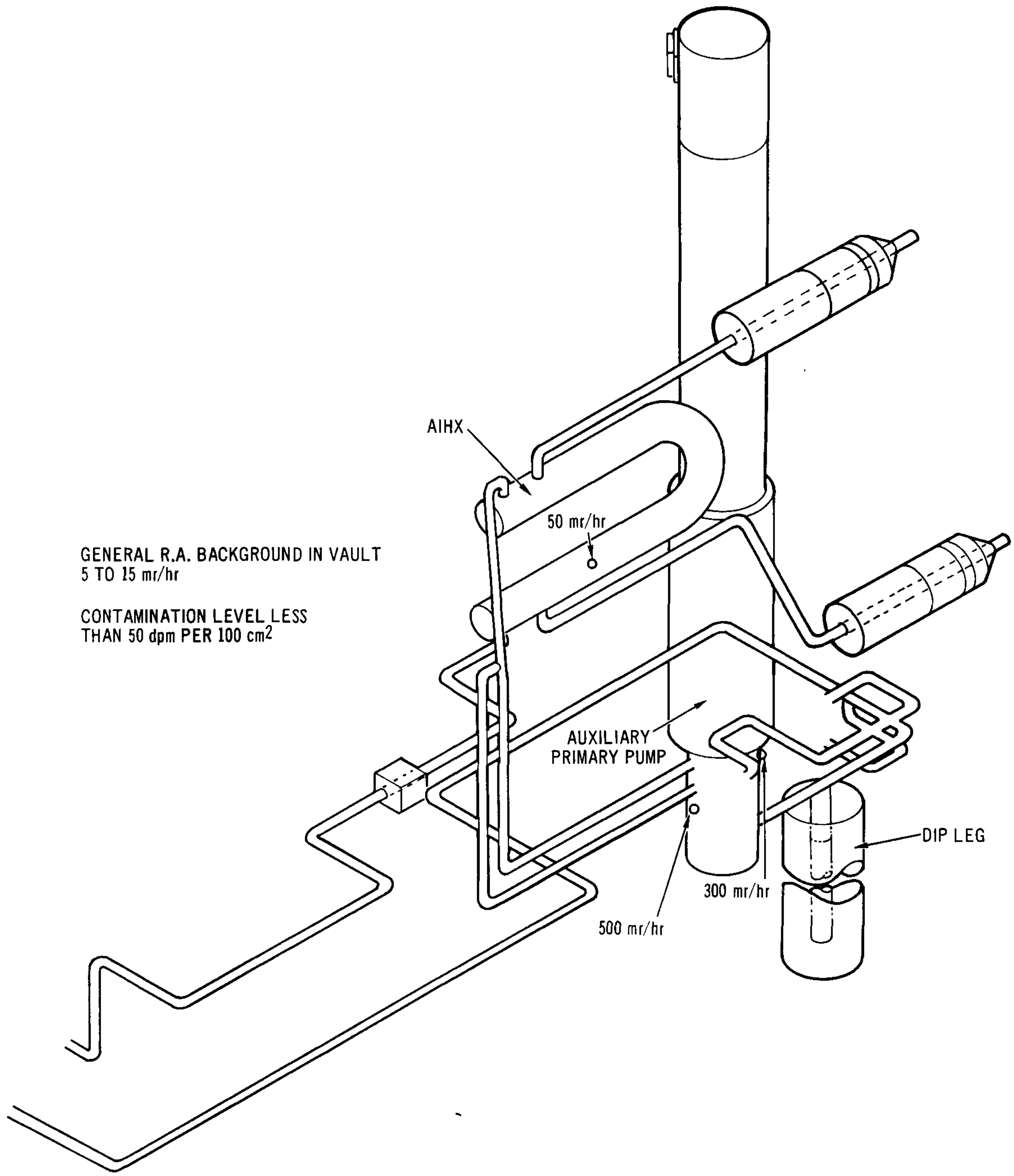
6a. Primary Fill Tank



5-14-68 7704-5703

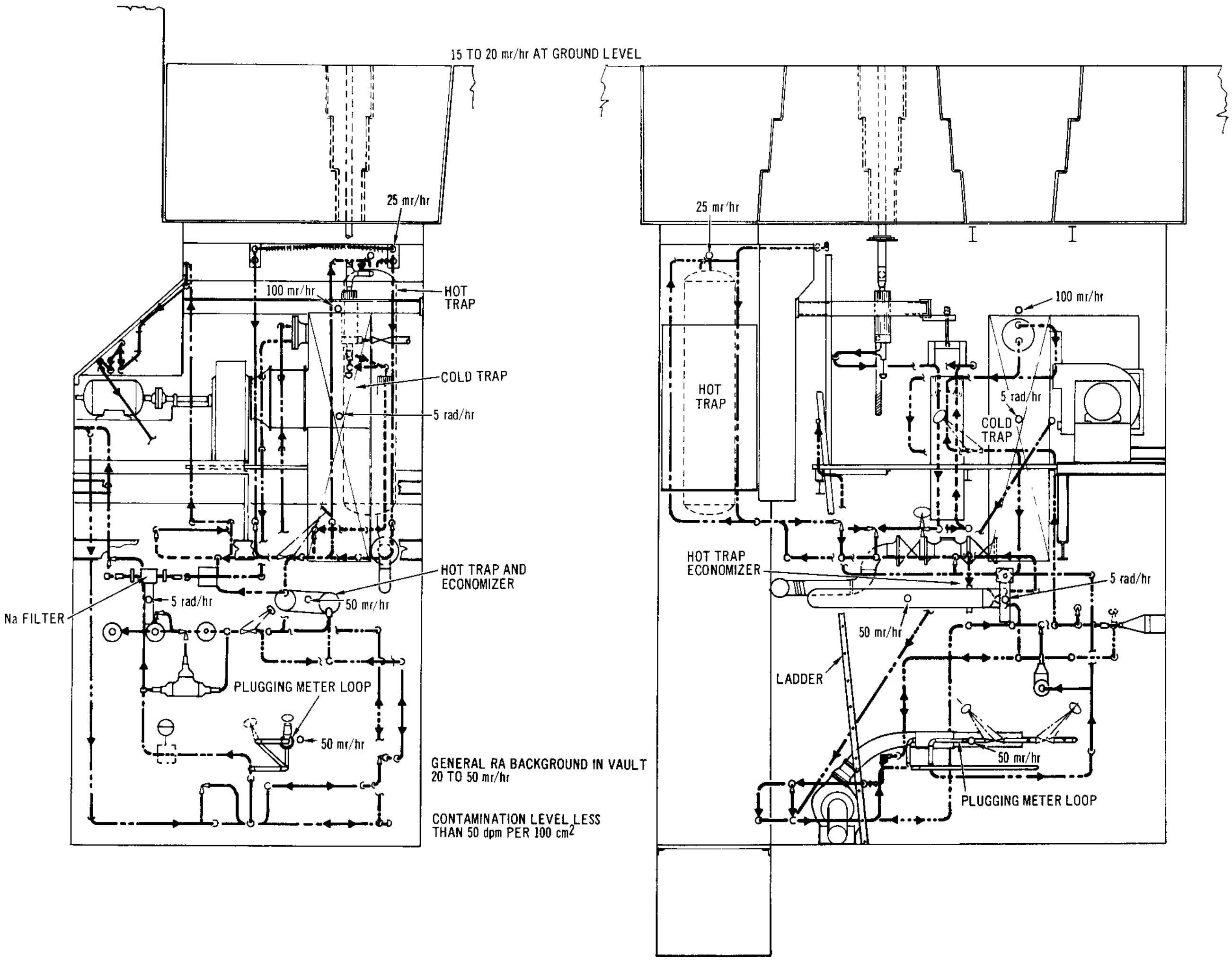
6b. Main Primary Area

Figure 6. Radiation Map (Sheet 1 of 3)



6c. Auxiliary Primary Area

Figure 6. Radiation Map (Sheet 2 of 3)



5-14-68 7704-5701

6d. Primary Sodium Service Vault

Figure 6. Radiation Map (Sheet 3 of 3)

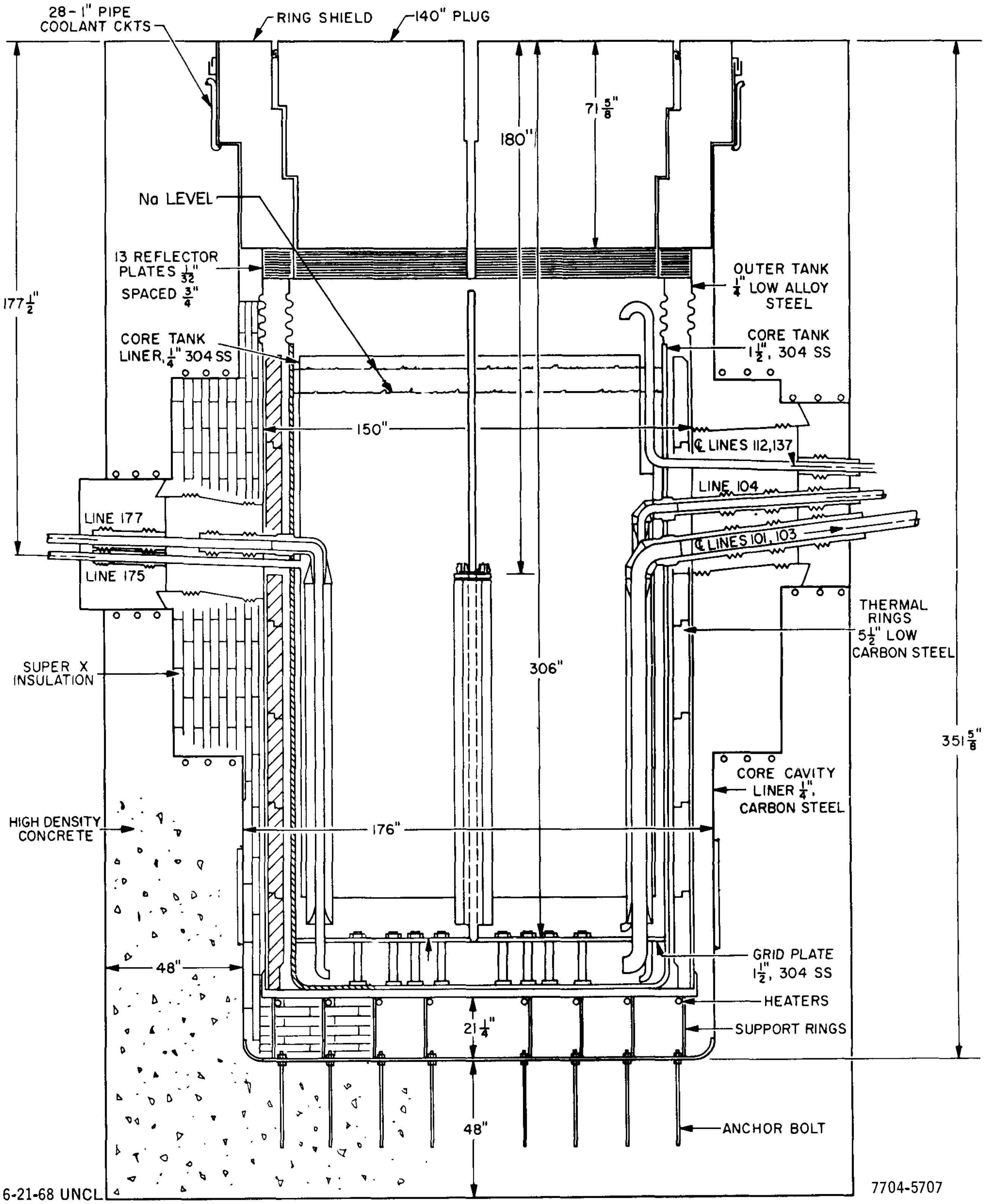


Figure 7. SRE Core Vessel Vertical Section

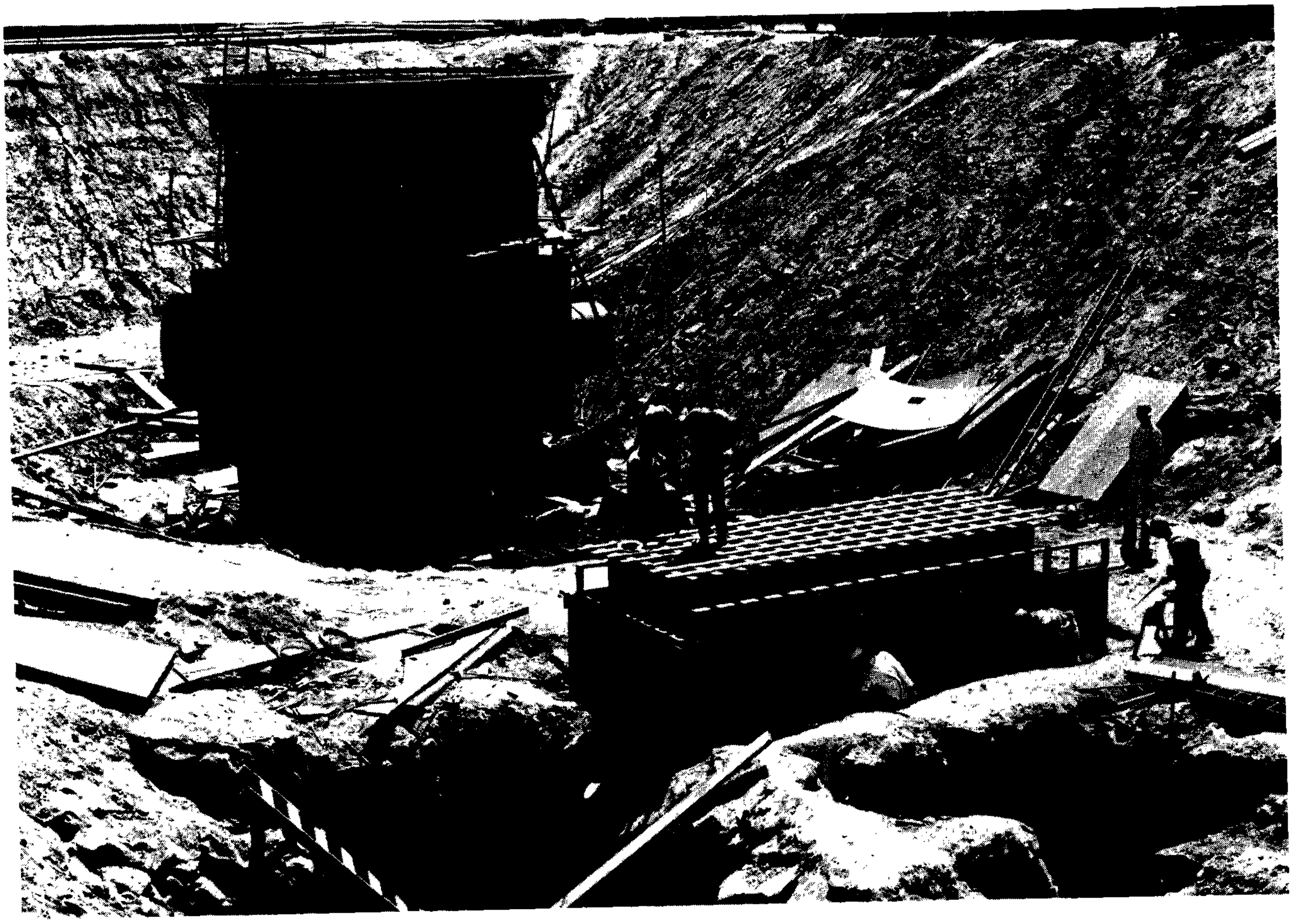


Figure 8. Construction of Outer Tank and Fuel Storage Cells

The wash cells are deactivated. The water has been drained from the compression water tank and the vacuum tank. The walls have been washed with demineralized water and the water drained. The hot cell drain valve has been closed and tagged out.

The two 5,000-gal storage tanks in the obsolete liquid waste tanks have been drained and decontaminated to $<5{,}000\ \mathrm{d/m\text{-}100\ cm^2}$; the sludge has been removed and the tanks have been cleaned. The tanks are black iron and show considerable internal pitting, as shown in Figure 10. Figure 11 shows these tanks and the R/A gaseous waste tanks during the construction stages. Figure 12 shows the "black iron" pipe runs to these tanks (both liquid and gaseous waste). Refer to Figure 9 for representative contamination levels in the liquid waste lines.

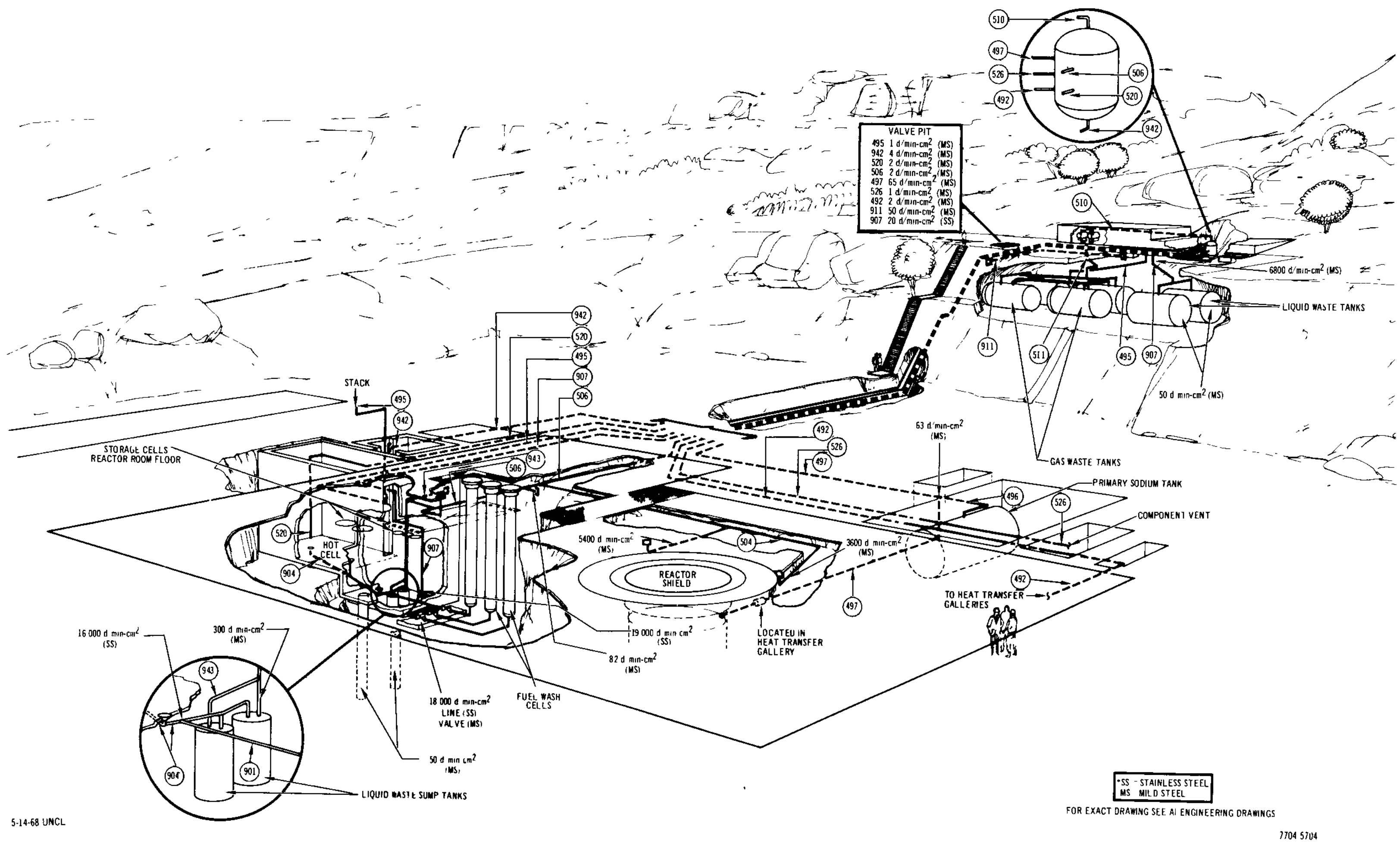


Figure 9. R/A Liquid Waste and Vent Systems Contamination Levels

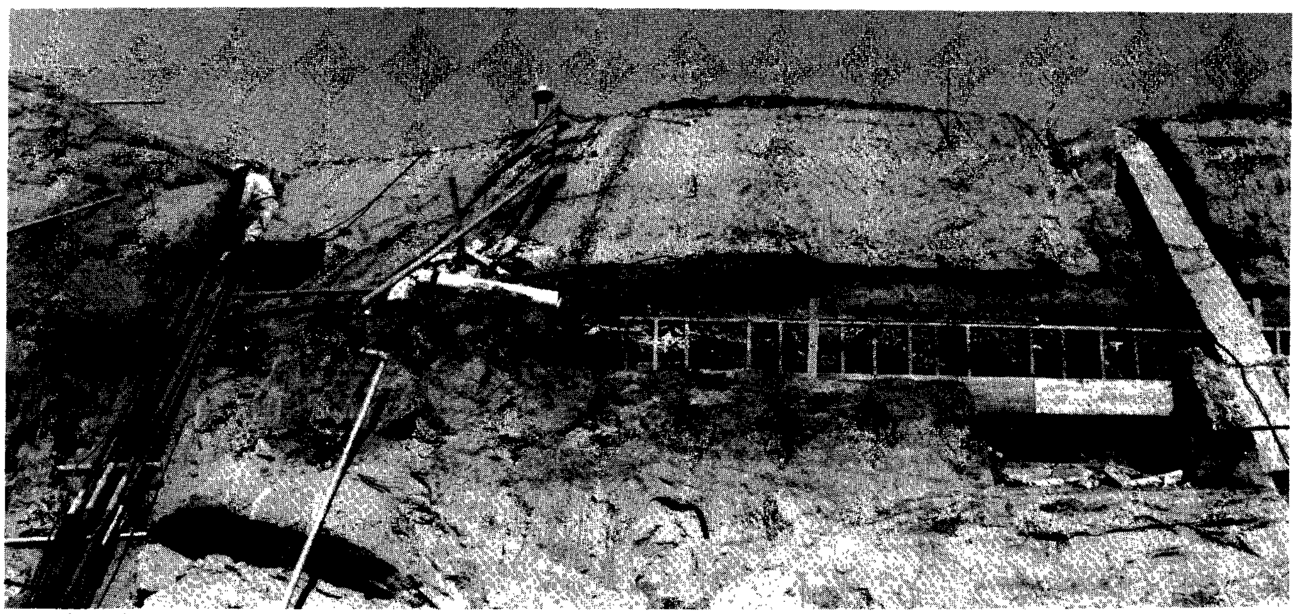
7704-1021CN

12-4-67

Figure 10. Internal Surface 5,000-gal Storage Tank – Obsolete Liquid Waste System

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)

Storage Tanks 1, 2, and 3 have been cleaned. There is a heel of water in each tank which has been made basic by the addition of NaOH and about 400 ppm of sodium dichromate for corrosion control. The R/A liquid waste sump pit has been pumped out. One shield block is in place over the sump pit, the other is set aside to allow access for inspection; operation type guard rails and chain were installed to block off the open pit. The shield block hoist cable and unpainted surfaces are protected with protective grease coating.

Approximately 100-gal of sludge and 900 gal of liquid were pumped from the change room holdup tank. The tank cover is in place and locked. The contamination level is 500 dpm/100 cm^2; the activity inside the tank is 3 mr/hr maximum. A warning radiation sign is in place.

G. R/A GASEOUS WASTE SYSTEM

This system has been purged with gaseous nitrogen to reduce the contamination levels. The levels of activity present are indicated in Figure 9.

The vent compressor has been deactivated by opening the power breakers in the electrical equipment room, by removing the vee belts and storing them (properly identified) in the storage area (Building 041), and by closing the inlet and outlet valves and capping these lines. Cooling water was drained and the system left open to air, after which rust preventative was applied to the unpainted surfaces.

The compressor vault is dry and radiologically clean, and ladder access is provided. The vault is covered, but not weather-proofed.

The four gaseous waste storage tanks (shown in Figure 11 during construction) are leak-tight and sealed off under a nitrogen pressure of <1/4 psig. The pipe runs to these tanks are shown in Figure 12.

H. KEROSENE SYSTEM

The kerosene system has been retired according to deactivation specification SS599N64001, and procedure PR-599-64-001. The present status is:

1) The kerosene has been drained from both the main and limited volume kerosene systems and removed from the site.

2) Kerosene pumps and motors have been removed and the lines blanked or capped off. These pumps and motors are identified and stored in Building 003.

3) All system valves are open, except the inlet and outlet valves for the main primary cooling circuit.

4) Gaseous nitrogen at 0.5 psig is being maintained (as measured on a pressure gauge at the respective surge tanks) in both kerosene systems.

5) The leak rate from the kerosene system is less than 0.2 scfm.

6) Water cooling towers have been drained of water and drain valves are open. The supply lines are blanked off, and inlet valves are tagged out.

7) The gas tank of the gasoline engine (emergency kerosene cooling flow) is drained; no storage battery is present; rust inhibitor is present in the cooling water system per manufacturer instructions. The drive and fan belt are tagged and stored in the storage area (Building 041). The battery charger switch is open.

8) Protective grease has been used to coat unpainted surfaces of the kerosene pumps cooling tower fan and gasoline engine components.

I. INERT GAS SYSTEMS

The systems included are the Gallery Cooling and Dehumidification System, Vault Dehumidification System, Gaseous Helium and Nitrogen System, and the Instrument and Plant Air Systems. The alteration required to convert the gaseous helium to nitrogen will be described. These systems have been deactivated according to the specification SS599N69001, and procedure PR-599-69-001.

1. Gallery Cooling and Dehumidification System

The deactivation of the gallery cooling and dehumidification system is described below.

1) Treated cooling water has replaced the water in the cooling lines and the pumps and at the base of the cooling tower, to prevent corrosion damage. The water makeup valve (V-445) has been disconnected and the line capped off. The gallery blower units drive belts are removed and attached to the unit.

2) The freon compressor has been removed.

3) By opening the main breakers in Panel WW in the Electrical Equipment Room, power has been disconnected from the cooling water circulating pump (P-60 A and B) gallery blow units (W-60 A and B), cooling tower fan and the dehumidifier compressor (K-60).

4) The unpainted surfaces of the cooling tower fan, drive motor, and gallery blower have been coated with a protective grease.

2. Vault Dehumidification System

The following breakers in the vault dehumidifier system are open:

Breaker Number	Function
1	Nitrogen circulating fan
2	Water spray pump (S-1)
3	Pre-cooler condensor fan
4	Pre-cooler condensor water pump, south
5	Refrigeration condensor fan
6	Water spray pumps (N-1)
7	Compressor motor
8	Main transformer switch

Treated water has replaced normal cooling water in the evaporator cooling system. The drive belts have been removed and the uncontaminated belts are identified and stored in the storage area (Building 041). The unpainted surfaces of the blowers have been coated with a protective grease.

3. Gaseous Helium System

The helium system has been modified to operate with gaseous nitrogen, as shown in Figure 13. A NaK bubbler had previously been added to the system to exclude oxygen and water vapor from the sensitized surfaces of the core and piping in both the primary and secondary main auxiliary systems. The bubbler was removed where "re-start" was no longer a requirement. The lines to the R/A vent system and pressure regulators have been valved to minimize leakage paths. The leak rate at 1.0 psig is 0.1 scfm. The modified helium system is supplied with four nitrogen 6-packs of bottled nitrogen with a total volume of nitrogen gas of 5100 scf.

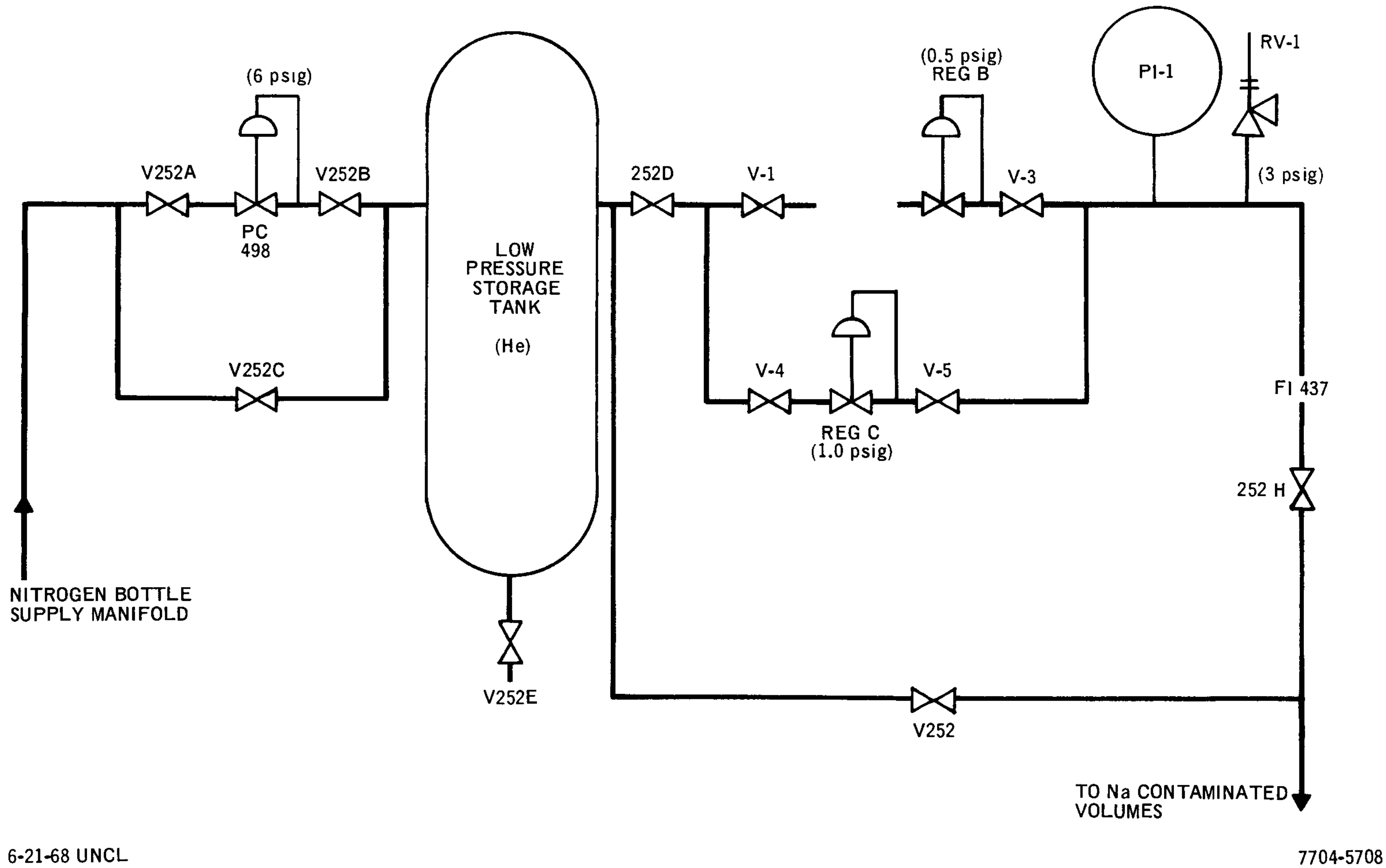


Figure 13. Modified He System

4. Gaseous Nitrogen System

Figure 14 shows the nitrogen system in its present configuration. The system assures a nitrogen pressure in all components previously supplied with nitrogen except in the vaults and galleries. The nitrogen system is open to the kerosene system and reactor insulation cavity. The liquid nitrogen storage tank is stored in place.

5. Instrument Air System

The instrument air system remains in normal operation to supply air for the building and for other uses as required. Breaker L-30 is clearly identified for quick opening in case of fire or other emergencies.

6. Plant Air System

This system remains in normal operation to supply air needs for the plant. The breaker is clearly identified for quick opening in case of fire or other emergencies.

J. ELECTRICAL SUBSYSTEM

The electrical subsystem has been partially deactivated; functions still in service are:

1) General lighting for facilities and areas
2) Public address system
3) Fire alarms and smoke detectors
4) Alarm for inert cover gas pressures.

The emergency power for the public address system will be furnished from another facility (off SRE site). The fire alarm system uses batteries supplied for this purpose. These Exide emergency lighting units are still in operation. All circuits remaining in service are tagged in the electrical equipment room. In addition, there is a posted list of these circuits and of the locations of the circuit breakers, and a line diagram showing the energized portion of the electrical system for use in the event of an emergency.

The motor generator set and diesel generator will be deactivated and all power breakers to them opened when another source of emergency power is provided.

All the wet cell batteries in the emergency battery bank will be removed.

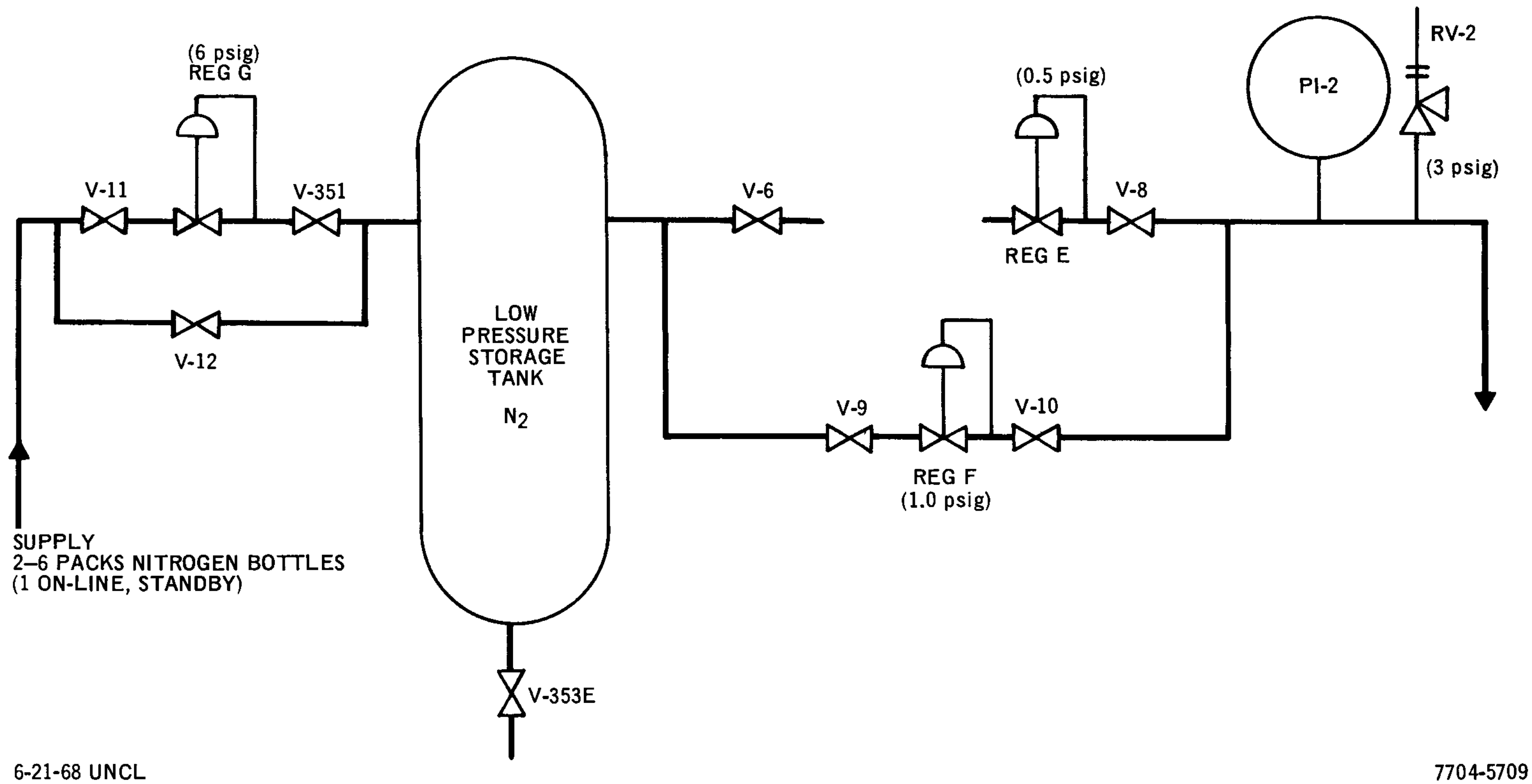


Figure 14. Modified Nitrogen System

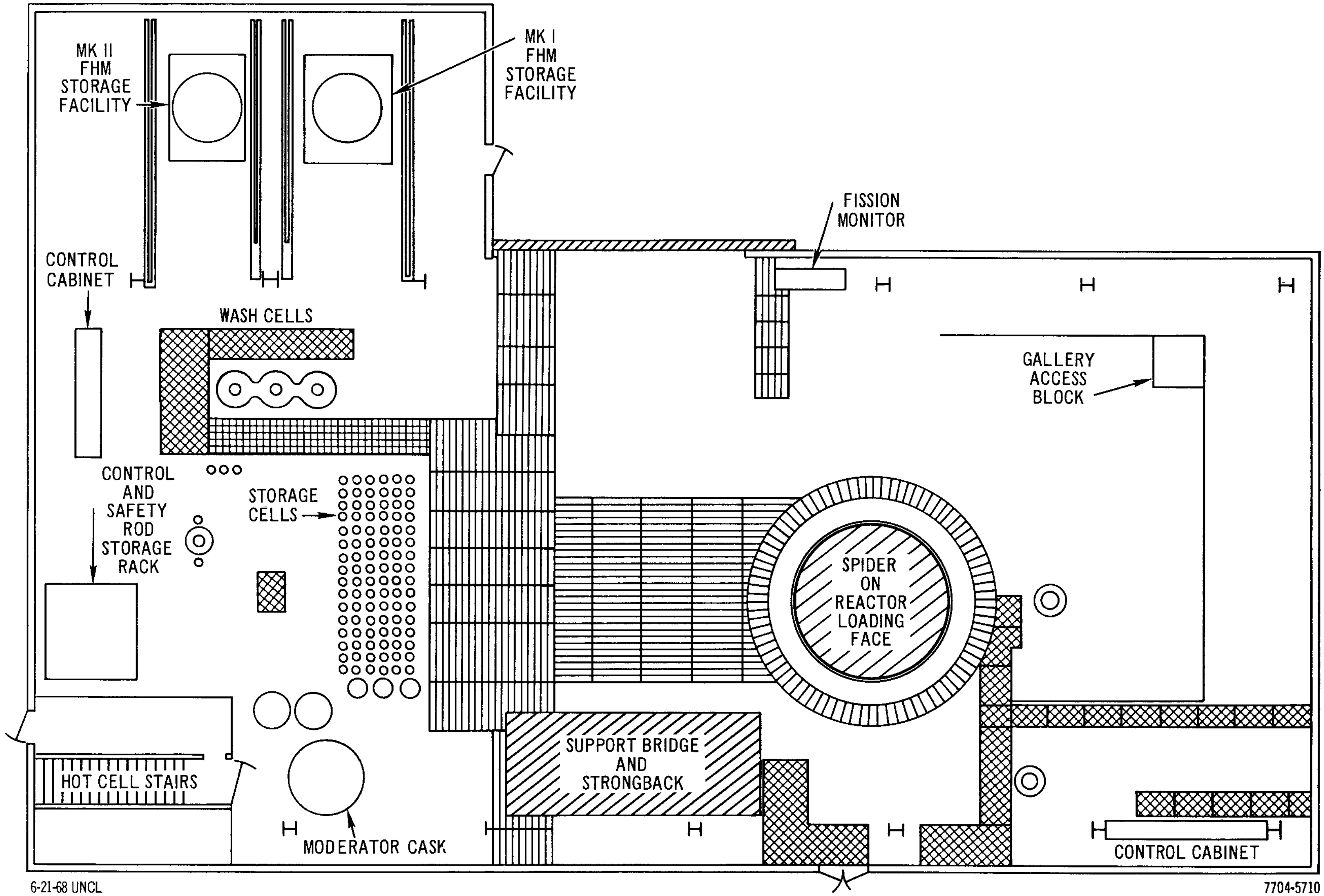


Figure 15. High Bay Storage Areas

K. CONTROL AND INSTRUMENTATION SUBSYSTEM

With the single exception of the inert gas supply alarm, this subsystem has been completely deactivated by removal of all power by opening all control room circuit breakers except Nos. 7 and 17 in the control room power breaker panel, and the visitors' gallery breaker in the electrical equipment room. Instruments used in other programs have been removed and turned over to Property Management (Section V). Other removable equipment, spare parts, and materials have been removed to the storage area for proper inventory and control (Section V). The remaining equipment is stored in place.

L. COMPONENT HANDLING MACHINES

The high-bay plan shown in Figure 15 indicates the storage areas for the component handling machines as well as for other components of the SRE. The β-γ contamination present on this equipment at the exposed surfaces was measured on February 22, 1968 with the results shown below. In each case, the area smeared was 100 cm^2

Sample No.	Description and Location	Results (d/m)
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

(continued next page)

Sample No.	Description and Location	Results (d/m)
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

The component handling machines are described below.

1. Mark I Fuel Handling Machine – Moderator Handling Machine

These machines are stored in the high-bay of the SRE (Figure 15). The bottom end of each cask is bagged to seal off the internal contaminated surfaces, and the bottom closure valves are closed.

2. Mark II Fuel Handling Machine

The bottom portion of the machine is covered with plastic to seal off the more highly contaminated areas of the cask. The nitrogen gas bottle has been removed and the vent hose and helium supply lines are removed and stored in a plastic bag with the machine. The sliding bottom closure valve is closed. The storage location of the machine and the rigging equipment (cables coated with protective grease) are indicated in Figure 15.

3. Pump and Plug Cask

The pump and plug cask are stored in the high bay area (Building 143). Internal surfaces are sealed off with plastic.

4. Moderator and Fuel Transport Casks

These casks have been shipped to the RMDF.

M. UNCONTAMINATED AREAS

The uncontaminated areas are listed below (Figures 1 and 2).

1) Building 003, ETB High Bay, Office Area, and Furnace Pit
2) Building 041, SRE Storage
3) Building 143, SRE, all areas except those noted in Subsection VI. N as contaminated
4) Building 153, Sodium Service
5) Building 185, SIR Control Room
6) Building 687, Helium Dock.

The following systems and/or services are being maintained in these areas.

1) Fire alarm system
2) Emergency paging system
3) Electrical power for lighting and Exide units for emergency lighting
4) Domestic water supply.

Other uncontaminated areas within the SRE site are listed below. (See Figure 2 for location.)

1) Secondary Pad Installation (413)
2) Liquid Waste Outside Area (653)
3) Transformer Substation (683)
4) Dormant Storage, Outside Area (686)
5) Tetralin Heat Exchanger (743)
6) Sodium Cleaning Pad (723)
7) North Ditch
8) Pond

These areas have been radiation-monitored and all were below background.

N. CONTAMINATED AREAS

The contaminated areas are listed below (Figures 1 and 2). The contamination levels for these areas are shown in Table 4.

1) Building 143, portions of SRE High Bay including fuel storage cells, wash cells, and hot cells. (See Section VI. F and G.)

2) Part of Building 163, Sodium Cleaning Facility

3) Building 724, SRE Oil Cleaning Facility

The entrances to these areas are posted with radiation warning signs giving the radiation levels to be expected therein. Other contaminated areas are the Hot Cells (see Section VI-C), Wash Cells (see Section VI. F and G), and Primary Sodium Vault (see Section VI. D).

The requirements for control of the contaminated areas at the levels designated are:

$>5,000$ dpm/100 cm^2	Area sealed
$>1,000$ dpm/100 cm^2	Locked door access
$<1,000$ dpm/100 cm^2	Tagged with a barrier
<50 dpm/100 cm^2	Clean area.

O. EDISON EQUIPMENT

The Edison substation at the SRE site, previously located as designated in Figure 2, will be removed and the grade restored.

TABLE 4

CONTAMINATION LEVELS
(Sheet 1 of 3)

Building 143 (Main Building)

Storage Cells

Cell No.	dpm/100 cm^2 β,γ	Cell No.	dpm/100 cm^2 β,γ
-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	dpm/100 cm^2 β,γ
Cell - A	1,980
- B	1,440
- C	900

TABLE 4

CONTAMINATION LEVELS
(Sheet 2 of 3)

Building 143 (Main Building) (Continued)

Pump Storage Cells	dpm/100 cm^2 β,γ
- East	780
- West	500

High Bay Floor

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

Building 163 (CERF)

	dpm/100 cm^2 β,γ		dpm/100 cm^2 β,γ
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		

TABLE 4

CONTAMINATION LEVELS
(Sheet 3 of 3)

Building 724 (SRE Oil Cleaning Facility)	
	dpm/100 cm^2 β,γ
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

VII. RETIREMENT SCHEDULE AND COSTS

The directive for deactivation of the SRE was received in January 1967 after an extended period of review and evaluation by AEC-DRL and AEC-RDT preparatory to operation of the plant. During this period answers were prepared to questions from both AEC organizations while preoperational tests were carried out to the extent possible. The program costs were averaging about $120,000 per month mostly to support an operating staff of about 35. Both the primary and secondary main sodium systems, as well as most of the auxiliary and supporting systems, were in operation on a continuous basis.

The directive received on January 4, 1967 requested immediate close-out of all activities that had been directed toward readying the facility for operation. It also requested preparation by January 25, 1967 of a plan for "mothballing" the plant and development of an inventory of SRE components and equipment which might be of use to other LMFBR program activities. These documents were prepared and submitted as requested. On March 31 an AEC-RDT memorandum was issued which approved "stored-in-place" deactivation and requested particular attention and/or additional effort in several areas including the following:

1) A site environmental survey program to assure continuing radiologically safe conditions.

2) Confirmation, by stress analyses, that cooling the reactor vessel would not cause damage, especially near the top of the bellows.

3) A complete inventory of AEC equipment and materials for circulation to other AEC programs to be designated, the inventory to include condition, cost for removal and date of availability of each item.

4) A detailed plan to show a safe sequence of system shutdown and deactivation to ensure that systems and instruments required for safe completion of each activity would be kept in the plant as long as they would be needed.

5) A technical report at the completion of the deactivation effort, providing minimum descriptions of the courses of action considered, the nuclear safety considerations, procedures employed, unexpected

problems and their solutions, adequacy of procedures from after-the-fact experience, and recommendations for future undertakings of a similar nature.

In part, the requirements outlined above were planned and scheduled in a revised plan, TI-599-19-001, Revision A[5] which is fully described in Section IV. The PERT schedule for this plan is a part of Figure 16. The retirement report and the detailed specifications and procedures prepared to perform the work complete these requirements. In July, after AEC and AI review of the documents relating to reactor vessel cooling and of the procedures for sodium system deactivation, consideration was given to taking pre-cooldown photographs of the inside of the reactor vessel and to conducting tests and inspections after cooldown. After investigation it was decided not to carry out these activities because of the marginal value of the added information relative to the costs involved. The principal factor here was the extension of continued shift operation in order to remove the oxygen and other impurities from the primary system sodium which could have been introduced in opening the reactor vessel to take photographs. It was concluded that pressure tests to assure that gas leaks were not introduced during cooldown would be adequate. Also, concern about oxygen and water impurities from the nitrogen gas supply to the sodium systems introduced a requirement for addition of a NaK bubbler to the inert gas supply system. These evaluations and changes in plans delayed by several weeks the draining of the primary sodium systems and extended the requirement for continued shift operations (in order to maintain sodium circulation). The primary and secondary sodium systems were drained, and shift operation was terminated by September 15, 1967.

During this period, the planned activities for storing the unirradiated Core III fuel were suspended because of problems in obtaining approval to construct fuel storage racks. Nor was approval given for proposed construction of fixed guard railings around open pits and ladders, to facilitate inspection of vaults. The basic problem was an objection to making capital improvements to facilities being deactivated. The delay in fuel handling also affected the electrical system requirements. Ultimately, arrangements were made to move the fuel to storage in Building 064, eliminating the need for maintenance of a radiation alarm system (RAS) at the SRE and clearing the fuel from the floor

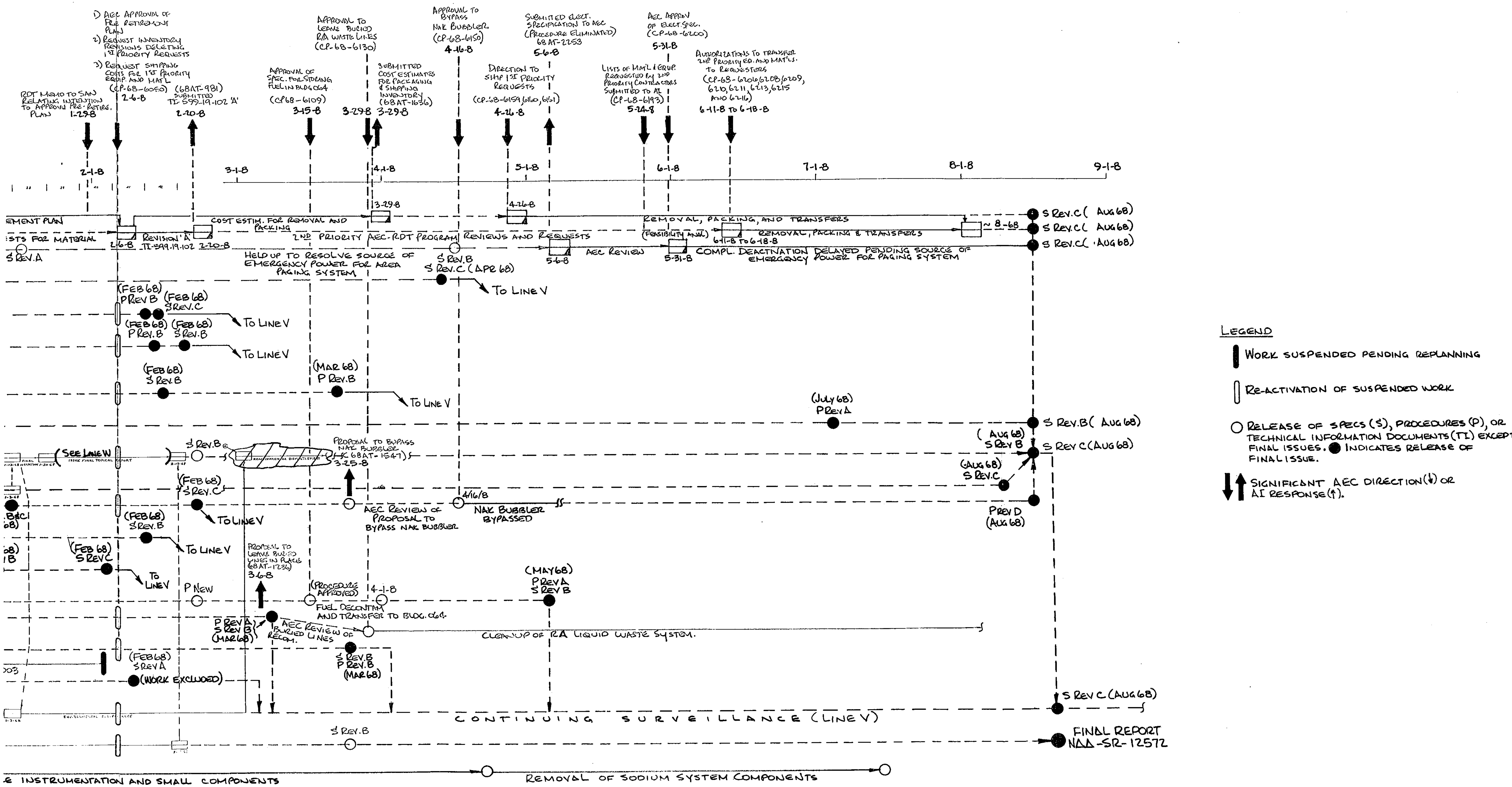
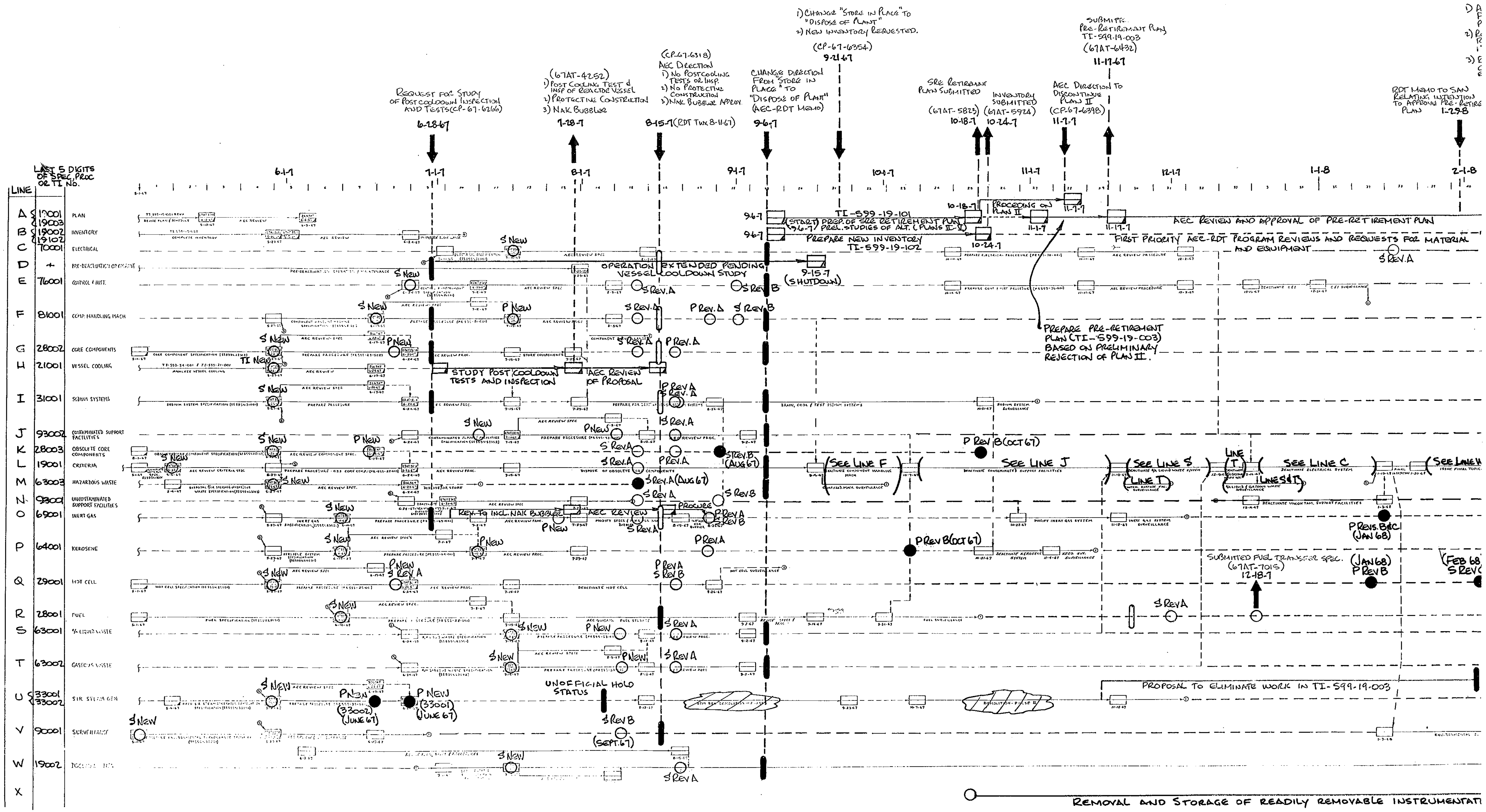


Figure 16. SRE Deactivation Schedule

Right hand side



Left-hand side

storage cells in the SRE building. The fuel removal was desirable to prevent the possibility of fuel corrosion and contamination of ground water.

On September 6, 1967 a new AEC-RDT memorandum was received redirecting the objective of SRE deactivation from "stored-in-place" to disposal of the plant, rendering the site safe from radioactive and other hazards, protecting usable plant components and equipment until such items could be transferred to other projects, and disposal of unsalvable and/or contaminated scrap and waste. The redirection also requested revision of the equipment and material inventory and circulation of the inventory list to designated AEC-RDT programs. Upon receipt of this memorandum, those activities having only a "stored-in-place" objective were suspended and revision of the inventory and study of retirement alternates was initiated. The resulting documents, TI-599-19-102[10] and TI-599-19-101,[6] were issued early in October.

After determining that a less extensive retirement effort than any of the four cases studied in TI-599-19-101[6] was desired by the AEC, a minimal "pre-retirement" plan, TI-599-19-003,[7] was prepared and submitted in November 1967. This provided for shutting down all systems and leaving everything in place with minimal continuing surveillance. This plan, plus the removal of equipment requested by other programs, became the retirement plan which was executed.

The overall schedule of activities is shown in Figure 16. A broader grouping with associated costs is shown by Figure 17. Schedules were planned to utilize a declining level of manpower because this was the most effective way to accomplish the work and facilitate an efficient reassignment of the personnel.

The inventory was transmitted to the AEC, who circulated it to the FFTF, EBR-II, ZPPR, AARR and LMEC programs. Some 200 items — mostly instrumentation but also including secondary sodium system pumps, valves and auxiliary airblast heat exchanger, as well as manipulators and dehumidification units — were subsequently requested and provided as discussed in another section. These items were later deleted from the inventory which was then circulated, as Revision A to TI-599-19-102,[10] to other AEC-RDT programs. An additional 200 items, mostly classified as instrumentation, were requested. Other accessible and available components were then excessed in accordance with usual AEC practices.

* NOT INCLUDED IN RETIREMENT COST TOTALS

△1 DIRECTIVE TO DEACTIVATE

△2 APPROVAL OF "STORED-IN-PLACE" DEACTIVATION PLAN

△3 DIRECTIVE TO DISPOSE OF PLANT

△4 APPROVAL OF 'RETIREMENT PLAN"

COSTS ($000/MONTH)

ACTIVITY	GFY 1967											
	J	A	S	O	N	D	J	F	M	A	M	J
PROJECT MANAGEMENT AND REPORTS							6	4	4	4	4	3
							△1				△2	
RETIREMENT STUDIES AND PLANNING							11	4	3	0	12	5
PRE-OPERATIONAL TESTING AND PREPARATIONS FOR OPERATION						*						
OPERATIONS AND MAINTENANCE PENDING DEACTIVATION							61	56	56	46	35	31
DEACTIVATION											22	20
WASTE DISPOSAL							4	1	3			
ASSISTANCE TO SOUTHERN CALIFORNIA EDISON												
SURVEILLANCE												
EQUIPMENT REMOVAL												
TOTAL							82	65	66	50	73	59

ACTIVITY	GFY 1968											
	J	A	S	O	N	D	J	F	M	A	M	J
PROJECT MANAGEMENT AND REPORTS	6	6	4	4	4	4	2	3	4	5	6	4
			△3				△4					
RETIREMENT STUDIES AND PLANNING	2	3	7	15	8	2	0	1	0			
PRE-OPERATIONAL TESTING AND PREPARATIONS FOR OPERATION												
OPERATIONS AND MAINTENANCE PENDING DEACTIVATION	18	17	20									
DEACTIVATION	18	27	30	28	3C	22	24	22	13	8	8	5
WASTE DISPOSAL				4	1	1	2	4	3	3	2	2
ASSISTANCE TO SOUTHERN CALIFORNIA EDISON												
SURVEILLANCE											0.8*	0.7*
EQUIPMENT REMOVAL									4	5	10	7
TOTAL	44	53	61	51	43	29	28	30	24	21	26	18

ACTIVITY	GFY 1969												TOTAL COST ($000)
	J	A	S	O	N	D	J	F	M	A	M	J	
PROJECT MANAGEMENT AND REPORTS	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.1	0.1	0.1	0.1	0.1	82
RETIREMENT STUDIES AND PLANNING													73
PRE-OPERATIONAL TESTING AND PREPARATIONS FOR OPERATION													NA*
OPERATIONS AND MAINTENANCE PENDING DEACTIVATION													340
DEACTIVATION													277
WASTE DISPOSAL													30
ASSISTANCE TO SOUTHERN CALIFORNIA EDISON				2	2	2	2						8
SURVEILLANCE	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	2.5*	NA*
EQUIPMENT REMOVAL	4.0	2.0	1.0										33
TOTAL	5	3	2	2	3	2	3						843

Figure 17. Schedule and Manpower Loadings

VIII. SURVEILLANCE REQUIREMENTS

The objective of the surveillance program is to assure that the retired SRE complex remains safe and secure so that there will be no opportunity for the SRE to become a source of radiation to the environment and so that the usable SRE plant equipment and systems are protected from deterioration.

The SRE surveillance effort is summarized in Table 5. This table lists the requirements and the action necessary to meet the objectives of the surveillance program. The principal responsibility for the surveillance and required maintenance resulting from this surveillance is with the Facilities and Industrial Engineering Department, with the Industrial Security Department responsible for the daily surveillance of the uncontaminated area within the buildings and the areas around these buildings on the SRE site. Significant support will be required from the Health and Safety Group.

Certain modifications to the plant, consisting of the installation of ladders and protective railings around open pits, were necessary to facilitate the surveillance. The ladders give free access to the sodium system, R/A liquid waste, R/A gaseous waste vaults and other areas as listed in Table 5.

Time Interval	Systems or Areas	Unit	Specification Procedure No. Reference	Requirements and Action
Daily	Uncontaminated areas	Bldg 003 ETB high bay Office area Furnace pit	SS 599 N 93001	Inspected for evidence of damage, unauthorized entry, and other abnormal conditions.
		Bldg 041 SRE storage.		
		Bldg 143 SRE, all areas except hot cell and high bay		
		Bldg 153 Sodium service		
		Bldg 185 SIR control room		
		Bldg 687 Helium dock		
		Other outside areas (uncontaminated) Secondary pad installation (413) Liquid waste (outside area) (653) Transformer substation (683) Dormant storage, outside (686) Tetralin heat exchanger (743) Sodium cleaning pad (723) North ditch Pond	SS 599 N 90001	
Bi-weekly	Inert gas	Gaseous nitrogen (modified helium)	SS 599 N 69001 PR-599-69-001	Record following pressures: 1. Gaseous nitrogen supply manifold pressure 2. Gaseous nitrogen tanks, low pressure 3. Pressure indication, PI-1 and -2
				Valve in standby bank and replace "empty" bottles if Item 1 < 300 psi.
				Adjust pressure regulator if Item 2 is outside specification of 6 ± 0.5 psig.
				Adjust pressure regulator if Item 3 is outside specification of 0.5 + 0.25 psig.
Monthly	Inert gas>	Gaseous nitrogen (modified helium)	SS 599 N 69001 PR-599-69-001	Record system pressures throughout system (see monthly check sheet for listing).
				Check valve positions and check for leaks if any pressures fall out of specifications (0.5 ± 0.25 psig except manifold supply and tank supply low pressure) if unable to correct notify Industrial Engineering.
				Check out level in sodium pumps (4) and in pump leak-off chambers. Add oil as required.
	Contaminated areas	Bldg 143 SRE high bay storage cell sump	SS 599 N 93002 PR-599-93-002	Inspect for physical condition, deterioration, water damage, and presence of vermin.
		Bldg 163 CERF		Inspect for water in storage cell sump by operating sump until red light goes off and pump stops*
		Bldg 724, SRE Oil Cleaning Facility		
	R/A Gaseous waste	Tank vaults – compressor and compressor suction	SS 599 N 63002 PR-599-63-002	Inspect for standing water, pump out as necessary*
	Core components	Storage boxes –Bldg 041 storage	SS 599 N 28002 PR-599-28-002	Inspect boxes for deterioration (report abnormal conditions to Industrial Engineering)
		Reactor loading face		Check loading face for evidence of moisture (report abnormal conditions to Industrial Engineering)
		Plastic cover		Check that plastic cover is intact (report abnormal conditions to Industrial Engineering)
	Hot cell	Sump pit Valve pit	SS 599 N 29001 PR-599-29-001	Inspect for standing water, request Health and Safety check for contamination in water; if uncontaminated remove with portable pump, if contaminated, pump into barrels for disposal at RMDU (sump pit water does not require Health and Safety survey)*

Table 5. SRE Surveillance
(Sheet 1 of 2)

Time Interval	Systems or Areas	Unit	Specification Procedure No. Reference	Requirements and Action
	Kerosene cooling	Rust prevention coatings on kerosene pumps, cooling tower fans, gasoline engine components	SS 599 N 64001 PR-599-64-001	Inspect for condition of coating; reapply as required.
		Gasoline engine radiator		Check water level, add water and rust inhibitor as required.
		System components		Inspect physical condition. Report abnormal conditions to Industrial Engineering.
	Electrical	Exide emergency lights	SS 599 N 70001 PR-599-70-001	Inspect operation of lights, wet cell water levels; repair and add water as necessary.
Quarterly	R/A gaseous waste	Vent compressor – rust preventative coating	SS 599 N 63002 PR-599-63-002	Inspect for integrity of coating; reapply as required
	R/A liquid waste	Sump pit	SS 599 N 63001 PR-599-63-001	Inspect for standing water; if present, request Health and Safety analysis to determine disposition.*
		Shield block hoist – rust preventative coating		Inspect for integrity of coating. Reapply as required.
	Sodium	Reactor vent valve vault Main and auxiliary vent vault Sump pump vault Solenoid vent valve vault Stand pipe over primary drain pump vault Primary drain pump vault Main gallery Auxiliary gallery Primary fill tank vault Sodium purification vault	SS 599 N 31001 PR-599-31-001	Inspect for standing water, rusting, deterioration, and presence of vermin. Pump out standing water, report all other abnormal conditions to Industrial Engineering. (Exception – stand pipe over primary drain pump vault need only be pumped if level is over 2 ft.)*
Semi-Annual	Inert gas	Gaseous nitrogen (modified helium) gas supply	SS 599 N 69001 PR-599-69-001	Analyze sample for oxygen content; also analyze bottled gas for oxygen and replace as necessary. Locate source of contamination.
Annual	Sodium	All exterior sodium piping and heat exchangers	SS 599 N 31001 PR-599-31-001	Inspect for deterioration, loose wiring, corrosion and rusting. Report presence of conditions to Industrial Engineering
	Component handling machine	Mark I and II fuel handling machine Moderator handling machine Long gas lock Pump and plug cask Transport cask Miscellaneous moderator handling equipment	SS 599 N 81001 PR-599-81-001	Inspect for physical condition, integrity of containers or covers and condition of rust preventative coating. Report abnormal conditions to Industrial Engineering and replace rust preventative coatings as required.
	Inert gas	Components specified in paragraphs 5.1, 5.2, 5.3, 5.4, and 5.5 of PR 599-69-001	SS 599 N 69001 PR-599-69-001	Inspect for deterioration, damage, rusting and presence of vermin. Report abnormal conditions to Industrial Engineering
		Pressure indicator – see monthly check sheet for listing		Calibrate with dead weight tester as per procedure (PR-599-69-001)

*Requirements and actions so marked will be performed as necessary during "rainy" periods.

Table 5. SRE Surveillance (Sheet 2 of 2)

IX. SUMMARY AND RECOMMENDATIONS

The objectives of the SRE retirement program have been achieved. The SRE is presently in a condition that does not pose a safety or radiological problem. All nuclear fuel has been removed and all sodium-contaminated systems are being protected by an inert cover gas. The remaining equipment is protected from the environs so that its future utilization is assured if the need should arise. The inventory and storage of spare parts and loose equipment assures ready access to any equipment if required. The surveillance program has been initiated to assure that the objectives of the SRE retirement plan and the present SRE status are not compromised in the future. SRE records, including the operation log book, print files, and photographs, have been removed to the Engineering Data Files for control and safe keeping.

Several comments, including recommendations for the future, are listed below:

1) Considerable effort could have been avoided during the initial phases of the work by clearer definition of the requirements for the deactivation or retirement of the SRE. This would have resulted in a more organized approach to the final disposal of the plant than the plan that resulted from the uncertain and shifting requirements that occurred.

2) During the time that the equipment was being made available to other AEC contractors it became clear that a more realistic assessment of needs and requirements would be fostered if the requesting contractors were required to pay the expenses of removal, packaging and shipment of the equipment they requested.

3) New plants should provide for inspection capability of all contamination-bearing lines and vessels.

4) Contamination-bearing system should be palletized in design wherever possible for easy removal and shipment for burial

5) After one year, the results and costs of the surveillance effort should be evaluated. To assure justification of continued costs, the evaluation should consider the future requirements and utilization of parts and supplies being protected.

APPENDIX A
SRE BIBLIOGRAPHY

INDEX OF TOPICAL REPORTS

NAA-SR Number	Title
48	Numerical Computation of Neutron Distribution and Critical Size
79	Some Mechanical Properties of Graphite
702	Neutron Leakage from the 30 Megawatt SGR-P 4 Reactor
748	Startup Incident in 30 Mw SGR
857	The Thermal Insulation of the SRE Core
878	Sodium Graphite Reactor Quarterly Progress Report - June, Aug.
898	Summary of Safety Aspects
902	Preliminary Safety Evaluation
919	Startup Incident in SRE
931	Survey of Fission Gas Problem in the SRE Fuel Elements After Burnups
964	Mech. System Description Reactor Coolant System Flow - AEI-73001
965	Mech. System Description Reactor Coolant System Flow - AEI-73006
988	SRE Fuel Rod Drop Tests
1021	SRE Criticality Analysis
1049	Sodium Graphite Reactor Quarterly Progress Report -Mar-June
1069	Multigroup Theory for Non-Multiplying Regions and Application to the SRE Shield Calculations
1082	SRE Criticality Calculations
1108	Design of the SRE
1110	SGX Lattice Constants and Reactivity
1116	Use of SRE Fuel in an Exponential Experiment
1131	Containment of Radioactivity in SRE
1154	Flow and Thermal Analysis of SRE Moderator Coolant
1165	SRE Fuel Element Manufacturing Procedure
1347	Sodium Graphite Reactor Quarterly Progress Report

NAA-SR Number	Title
1388	Nuclear Calculations on SRE
1392	Plug Requirements for SRE
1447	Preliminary Hazard Study of SRE Steam-Electric Plant
1472	Analog Study of Main ABHE
1506	Sodium Instruments and Components
1517	Two-Group Calculations of the Critical Core Size of the SRE Reactor
1520	Effect of Reactor Irradiation at Temp. 400° & 700°C on Thermal Conducting of Graphite
1552	Sodium Vapor Detection Test
1557	Flushing SRE Sodium Systems and Cleaning Primary Sodium Pump
1559	Core Tank Galling Tests
1565	Summary Report "Project Freeze Seal"
1567	Control Rod Evaluation
1578	SRE Steam Electric Generating Station Startup & Oper. Prob.
1584	Calibration of SRE Type Orifice
1586	Preliminary Estimate of Pressures Expected in a Closed Moderator Can
1604	Shield Requirements for a Portable Shielded Compartment to be Used in Removal of Moderator Cans
1612	Interaction of Nuclear and Power Plant Controls
1623	Effects of Liquid Sodium Upon Zirconium Metal
1627	X-Ray Measurements of Irradiated Graphite Annealed at Elevated Temperatures
1633	SRE Design Changes
1636	Evaluation of Predicted SRE Hot Trap Performance
1639	SRE Instrumentation & Control
1647	Trip Report-Re-Liquid Sodium Bellows Seal Valve for SRE
1656	(Rev.) SRE Project Data Sheet
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1681	Pre-Operational Acceptance Test Procedures for the SRE Prepared by Sodium Graphite Reactors Group
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1684	Additional Analysis of SRE Safety
1691	Outline of Reactor Physics Startup Experiments on SRE

NAA-SR Number	Title
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1699	Process Heat from Nuclear Energy Use in Pulp and Paper Industry
1725	Pump Wash Experiment
1728	Fuel Handling Experiment
1738	Manual of SOP's for the SRE
1745	Heat Transfer Analysis of 19-Rod Fuel Element in Handling Cask
1755	Sodium Graphite Reactor Alternate Core Study - Circular Stainless Steel Canned Moderator Elements
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1843	Reactor Coolant Pumps
1865	SRE Room Temperature Sub-Critical Test
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1973	Liquid Metal Heat Transfer Components
1991	Stainless Steel - Sodium Corrosion Problem
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2006	Zirconium Moderator Reflector Can Development for the SRE
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2008	Development of SRE Mark I Safety Rod

NAA-SR Number	Title
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3481	High Strength Zirconium Alloys
3482	Development of SRE Mark II Safety Rod
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3515	Two-Group SRE Fluxes in Two Dimensions
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3775	Thermal Performance of SRE MIHX
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9003	SRE Systems and Components Experience Core II
9430	Nuclear Parameters of the SRE Cor II
9469	Project Proposals FY 65 and 66
9510	Core II Physics Tests on the SRE
9511	On-Line Shim Rod Calibration
9516	SRE PEP Reactor Safety Analysis Report
9759	Uranium Carbide Fuel for Sodium Cooled Reactors
9999	Annual Technical Progress Report FY 64
10010	SRE Reactivity History Program
10029	Example of the Application of a Digital Dynamics Simulator to Reactor System Accidents
10061	Two-Region SRE Fast Flux Irradiation Facility
10151	Temperature Oscillations in the SRE
10153	Temperature Distributions in the SRE Core II Fuel Elements
10721	Drilling Graphite Samples from Irradiated SRE Moderator Elements
10817	SRE Mark II Fuel Handling Machine
11020	Project Proposals Fiscal Years 1966 and 1967
11160	SRE-PEP Moderator Elements Final Report
11309	The Effect of Neutron Irradiation on the Liquid Sodium Dilation of Graphite
11359	TR-SRE Fast Flux Irradiation Facility
11396	Effects of Long-Term Operation on SRE Sodium System Components
11450	Annual Technical Progress Report FY 1965
12408	Non-Nuclear Performance of SRE-PEP Systems
12409	SRE-PEP Pump Performance
12410	Carbon Removal From SRE Sodium Following PEP Modifications

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Number	Title
AI-MEMO-1671	SRE Steam-Electric Generating Station Operating Notes
AI-MEMO-1684	Additional Analysis of SRE Safety
AI-MEMO-1748	AI Reactor Designs
AI-MEMO-1766	Radiological Safety During AI Division's Relocation
AI-1612	A Report on the Interaction of the Nuclear and Power Plant Controls for the SRE Steam Electric Generating Station
AI-8212	Two-Region SRE (TR-SRE) A Coupled Fast and Thermal Reactor
AI-64-96	ESADA SGR Development Program Quarterly Progress Report January-March 1964
AI-64-169	ESADA SGR Development Program Quarterly Progress Report April-June 1964
AI-64-231	ESADA SGR Development Program Quarterly Progress Report July-September 1964
AI-65-5	ESADA SGR Development Program Quarterly Progress Report October-December 1964
AI-65-64	ESADA SGR Development Program Quarterly Progress Report January-March 1965
AI-65-133	ESADA SGR Development Program Quarterly Progress Report April-June 1965
AI-65-219	ESADA SGR Development Program Quarterly Progress Report July-September 1965
AI-66-53	Effects of Long-Term Sodium Exposure on Materials in SRE

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NAA-SR-TDR Number	Title
A-2	SRE Auxiliary Secondary Motor and Pump, Structural Design
8	SRE - Instrument Thimble Temperature
16	SRE - Main Airblast Heat Exchanger
17	Main and Auxiliary Secondary Sodium Pressure Indicators and Transmitters
27	Stress Analysis of SRE Fuel Rod Jackets Under Operating Conditions
28	Flow Distribution, Friction Factor, and Entrance and Exit Coefficients for the SRE Hollow Fuel Element
44	Moderator Can Concept for SRE Alternate Core
52	Pressure Drop and Orificing for the SRE 7 Rod and Hollow Fuel Elements
59	Fuel Handling Coffin (9693-78201)
61	Stress Analysis of SRE Fuel Element Jacket Using Fuel Growth Sufficient to Fill Annulus at Center of Element
63	Sodium Removal From SRE "Calandria" Type Core
74	Calculated Thermal Performance of SRE Reactor Loaded with 37 2.78 w/o Enriched Uranium 7-Rod Elements
76	Enrichment of Th-U Fuel for SRE
76	Enrichment of Th-U Fuel for SRE Rev. I
78	Negative Pressure in Galleries
79	Power Level Determination of SRE
83	Interim Report on Nuclear Aspects of Th-U Loading for SRE
83	Interim Report on Nuclear Aspects of Th-U Fuel Loading for SRE Rev. I
89	Steady State Temperature Coefficient of the SRE
90	SRE Control Rod Worth
94	Neutron Flux Measurements in the SRE Biological Shield
100	SRE Moderator Can Handling Procedure

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102	Upper Limits of Reactivity Changes in SRE Due to Long-Term Effects
110	Initial SRE Power Operation
114	Max. Stress in SRE Type Fuel Element Jacket Due to NaK Superheat
131	Eddy-Current Brake for Throttling Natural Convection Flow
135	Estimated Maximum Reactivity Changes Due to Long-Term Effects in SRE with Th-U Alloy Fuel
	2000-2199
2070	Accuracy of Control Rod Position Indication
2071	Investigation of Feasibility of Replacing Helium with Nitrogen as Cover Gas in Certain Specified Areas as Economy Measure
2074	Activation of Stainless Steel
2077	Strain Gage Installation on MS Sodium System of SRE
2079	Zircaloy-2, Mark II Control and Safety Rod Thimbles
2081	Radiation Survey of Reactor Complex & Neutron Flux Measurements in Main Gallery, Flux Measurement and Instrument Thimbles
2082	Shielding Survey of the SRE Fuel Coffin
2086	NaK Requirements for SRE Large Hollow Fuel Element
2098	Zircaloy - Steel Control and Safety Rod Thimbles, SRE
2099	Thermal Shock in Fuel Rod Jackets During Scram
2101	Calculate Thermal Performance of SRE Reactor With Th-U Fuel Loading
2104	Predicted Transient Temps and Stresses in Core Tank and MP and MS Loops During a Full Power Scram
2111	Plot of Radiation Leakage Through Gallery Shield Blocks
2112	Installation of Electromagnetic Pump for Moderator Cooling
2116	Preliminary Investigation for Control of SRE Convection Flow Problem
2117	Calculation of Fuel Element Orifice Plate Sizes for Forty-two Fuel Element Loading and 120% Flow
2170	Preliminary Analysis on Maintenance of SRE Core Components
2174	Orifice Sizes for SRE Loaded with 42, 7-Rod Uranium Elements
2175	Transient Study of Thermal Effects of Loading Cold Core Elements into Heated Sodium Filled Core

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2178	SRE - Tetralin System
2187	SRE Operating Time Versus Number of Standard Uranium Fuel Clusters
2188	SRE Control Rod Shielding Requirements
2189	Re-evaluation of SRE Moderator Cans Considering Observed Data From Initial Power Runs
2198	Modification of Dummy Fuel Element Orifice

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2201	Maintenance Procedure for SRE MP Sodium Pump
2202	Experimental Evaluation of Neutron Streaming in SRE Galleries as Measured During First Two Power Runs of SRE
2203	Fuel Temperatures During SRE Power Operations, July 1957
2204	Time Constant of SRE Eddy-Current Brake
2205	Fabrication of Th-U Fuel Rods for SRE
2207	Electronic Circuit Design for Providing an Indication When Mark II Safety Rods are Released from Holding Magnet
2211	Tests for Gas in MS System
2215	Fuel Coffin Transfer Links (172-75-40)
2241	Tests of Moderator Pedestal and Grid Plate Seal
2242	SRE Safety System
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2248	Shielding Analysis of SRE Fuel Handling Coffin and Comparison with Radiation Survey Data
2250	Emergency Power System
2261	Grid Plate Leakage Experiment
2266	New Requirements for Electromagnetic Pump for Moderator Coolant
2267	Excess Moderator Coolant Flow to be Removed by Electromagnetic Pump in Moderator Coolant Line
2284	Removal of Damaged Element From SRE Fuel Channel R-57
2285	Strain Test on SRE Mark II Safety Rod
2325	Temp of Irradiated SRE Fuel Element in Proposed Moderator Can Coffin Including Effects of Diameter & Temp of Coffin Wall
2326	Effect of Fuel Rotation on Channel Output Temperature
2336	SRE Control Rod Calibration and Reactivity Excess at Present 43 Element Loading
2353	Neutron Flux Traverse in Reactor Hole No. 2 Thru 140" Plug
2357	SRE Radioactive Liquid Waste Storage System P&I Diagram 9693-978827

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2358	SRE Radioactive Vent System P&I Diagram 9693-978102
2367	Facilities for SRE Experimental & Analytical Programs
2372	Control System for Electromagnetic Eddy-Current Brakes & Throttle Valve
2406	Problems Associated With the MIHE in the SRE
2407	Method of Program Controlling Reactor Flux or Temperature with Present Servo Amplifier
2408	Auxiliary Secondary Sodium Loop
2422	Vertical Temp Traverse in Reactor Hole Number 1
2429	Description of Proposed Poison Ring Test in SRE
2434	Origin of Water Vapor in the Galleries
2446	Investigation of Fuel Handling Coffin
2448	Control System for Moderator Flow Electromagnetic Pump
	2500-2699
2504	Hazards Analysis for Th-U Loading of the SRE
2506	Comparison of the Calculated and Measured Isothermal Temperature Coefficient of Reactivity in the SRE
2507	Hollow Fuel Rod Heat Transfer Analysis
2524	Helium and Nitrogen Systems 9693-978101 & 9693-978104
2525	Fuel Element Hanger Rod Disconnect
2530	MS Sodium Pump and System Characteristics
2532	Evaluation of After-Shutdown Radiation Levels in SRE Sodium Galleries Due to Neutron-Inducted Activation of Gallery Components and Determination of Specific Activity of the Secondary Sodium
2576	Effect of Carbide Precipitation and Carburization on the Mechanical Properties of Type 304 Stainless Steel
2578	SRE Experimental Scram Data - November 20, 1957
2579	SRE Gallery Dehydration
2588	Ruptured Bellows Seal Valves
2593	Shielding Analysis of the SRE Moderator Handling System
2637	Ball Latch for Control-Safety Rod Drive
2639	MP and AP Sodium Systems
2660	Fuel Element Decay Heat Determination
2662	Natural Convection Flow in Primary Sodium System After a Reactor Scram

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2666	Conditions for Safe Thermal Storage of SRE Fuel and Moderator Cans
2683	Experimental Evaluation of the Neutron Streaming in the SRE Galleries as Determined During the November Power Run of the SRE
2745	Process Review of Tetralin Cooling System
2757	Pipe Nozzle Loading on the Main SRE IHX
2782	SRE Control Rod Calibrations
2805	Reactivity in SRE of the Bettis Alloys
2817	Afterglow Heating in SRE Hot Cell
2826	Reactivity of Thorium Uranium Fuel
2827	Fuel Rod Meltdown Conditions
2830	Sintered Metal Filters for Sodium Coolant Systems
2834	Moisture in the SRE Insulation Cavity
2835	Rates of Alloying of SRE Metal Fuels with Their Jackets Above 1600°F
2836	Stress Analysis of Large Hollow Fuel Elements
2844	Redistribution of SRE Fuel
2887	Evaluation of SRE Vapor Traps, Na Samples, Na Filters, NaK Bubblers and Freeze Traps for Suitability for HNPF
2912	Review of Safety Rod Snubber
2918	Instrumentation for the SRE Gallery Dehumidification Equipment
2920	Study of Various Types of Freeze Type Liquid Metal Valves
2923	SRE Hazards Analysis
2933	Measurement of Neutron Flux and Gamma-Ray Dose Rate During Full Power SRE Operation and Subsequent to Scheduled Full "Δ T" "Scram"
2999	Test Results - SRE Inpile Poison Ring Test No. 1

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3001	Predicted Maximum Surface Temps of UO_2 Experimental Fuel Element After Irradiation in the SRE
3002	Cooling Times After One Week - 20 KW Intervals
3003	Graphite Temperatures in the Central Core Element
3006	Auxiliary Sodium Loops
3011	Evaluation of Standard SRE Fuel Element Exposed to Approximately 0.01 a/o Burnup
3105	SRE Nitrogen Gallery Seal
√ 3115	SRE History of Reactor Scrams

NAA-SR-TDR Number	Title
3131	Neutron and Gamma Flux Survey of SRE Instrument Thimbles
3132	Automatic De-canner - SRE Fuel Elements
3134	Design and Fabrication of SRE Core Heater Protective Tubes
3135	Seven Rod Fuel Cluster Surface Temperature in SRE Storage Cell with Simulated Decay Heat
3142	Reactivity of MC-2-2 in the SRE
3145	Bearing Removal From Modified 6" SRE Sodium Pump
3193	Rating Calculations on Six SRE IHX Bids
3198	Preparation of U^{235} Capsules
3238	Re-evaluation of the Nuclear Properties of the Th-U Loaded SRE Calandria Core
3244	Storage of SRE Moderator and Reflector Cans
3287	Thermal Efficiencies
3291	Liquid Hydrocarbon as SRE Core Tank Protection During Core Exchange
3293	Investigation of the Failure of SRE Fuel Element R-36
3295	Design and Performance of Induction Pump for SRE Moderator System
3298	Pressure Drop - Flow Tests of Two Types of SRE Fuel Elements

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3303	SRE Calandria-Core Process-Tube Bellows
3306	Gamma Traverse of Mark II Safety Rod
3315	Irradiation of 19-Rod Uranium Oxide Fuel Element in SRE
3316	Performance of the 6" Free Surface Pump
3323	Air Cooled Plugging Meter Tests
3324	Release of Fission Products into NaK From an SRE Fuel Element
3328	Driver and Detector Circuit for Precision Sodium Level Probe
3331	SRE Project Technical Specifications
3344	Storage of Fuel and Core Components
3347	Calculated Thermal Performance of Calandria Core Loaded With 37 Th-U 7-Rod Elements
3358	Removal of Moderator Cans From SRE Core
3360	Pressure Rise in Reactor Galleries, Due to a Major Sodium Leak or Breakdown of Cooling Systems

NAA-SR-TDR Number	Title
3361	Moderator Coolant Pressure Drop Across Lower Plenum
3413	SRE Side Shield Heat Generation Rates - Comparison of Nuclear and Heat Transfer Calculations
3431	Nuclear Parameter Study of the SRE Uranium-Carbide Core
3437	Heat Transfer Analysis of Hollow UC Fuel Element
3522	Expanded Scale Temperature Recorder
3523	Reactivity of Zircaloy II Thimble in the SRE
3532	Current Limiting Control for the Diverter Pole System While in Use as a Battery Charger
3536	Experimental Results of Radiation Measurements Performed About SRE Rotatable Top Shield Area
3542	MIHE Study
3559	Preliminary Report on the Nuclear Aspects of Uranium-Carbide Fuel - Seven Rod Cluster Variation

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NAA-SR-TDR Number	Title
3641	Strength Tests on Low-Melting-Point Alloy Seal
3648	Recalibration of SRE Shim Rod No. 3
3681	Experimental Evaluation of the Neutron Streaming During the May-Sept, 1958 Power Runs After Installation of Proposed Neutron Shielding in SRE AP Gallery
3682	Steady State and Transient Heat Loss From SRE Main Airblast Heat Exchanger
3685	Worth of HNPF Inpile Poison Ring II Assembly
3760	SRE-Edison Plant Control System
3830	Sodium-24 Specific Activity in HNPF U-Mo and UC Reactor Systems
3844	Television Monitor - SRE Fuel Coffin
3854	SRE Criticality for Uranium - Carbide Fuel
3857	Compilation of Experimental Determinations of the Average Thermal Neutron Flux in 7-Rod Fuel Clusters
3920	Fabrication and Inspection Techniques for the SRE Alternate Calandria Core
3924	Comparison of Calculated and Measured Gamma-Ray Dose Rates at SRE Instrument Thimbles
3933	Worth of Enriched Uranium Fuel After Irradiation in SRE
3934	Radial Statistical Weight for the SRE

NAA-SR-TDR Number	Title
3935	Sensitivity Gains Effected by Surrounding Neutron Ion Chambers with Hydrogeneous Materials
3944	Investigation of SRE Fuel Element Failures
3984	NaK Free Convection Cooled Shaft Freeze Seal for SRE Pumps
4032	Recalibration of SRE Shim Three
4144	MRII Driver and Detector Circuit for Precision Sodium Level Probe
4148	Neutron Flux and Gamma Dose Rate in SRE Instrument Thimbles 7 and 8
4166	Calculation of SRE Core Tank Wall Thermal Stresses
4379	Reactivity Changes Associated with a Failure of an SRE Fuel Element
4408	Calculated Temperature Effect if Orifice Plates are Located Above the Fuel Clusters in the 37 Th-U SRE Core
4479	Observations of Carbon in SRE Sodium
4550	Progress Report on Wash Cell Testing
4550	Wash Cell Testing, Final Report Add. I
4676	SRE Neutron Flux/Power Deviation Limit Circuit
4678	Variable Orifice for SRE - Hydraulic Calculations
4872	Testing a Sodium Jet Pump
4887	Carburization of Type 304 Stainless Steel in Sodium-Carbon Systems
4906	Centrifugal Pump Characteristics in the Near-Boiling Range
4952	Study of Fuel Temperature and Flow Effects of Plugging in SRE Fuel Channels
4977	Preliminary Evaluation of U-C Fuels
	5000-5699
5124	Hydraulic Tests of the 5-Rod and Dummy SRE Fuel Elements
5162	Analysis of the SRE Power Excursion of July 15, 1959
5165	Post Irradiation Examination of SRE Dummy Element No. 71626-007 - Part I: Disassembly and Initial Measurements
5227	Criticality Hazard Evaluation for SRE Fuel Storage and Fuel Transfers
5248	Preliminary Nitrogen Cover Gas Studies
5252	Elevated Temperature Wear Characteristics of Zircaloy-2 & Type 304 Stainless Steel in Contact with Hastelloy-X

NAA-SR-TDR Number	Title
5301	Calculation of Uranium and Plutonium Inventories in the SRE First Core Fuel Loading
5379	Zircaloy-2 for SRE Replacement Moderator Cans
5431	Heat Transfer for 43 and 55 Th-U Fuel Element Cores in SRE
5436	Irradiation and Temperature History of Moderator Can Which Contained Channel R-24 During SRE Power Runs 1-14
5437	Irradiation and Temperature History of SRE Dummy Element Cans
5550	A Note on the Shape of the Power Transient Following a Step Input of Reactivity
5588	A compilation of Room Temp and 1200°F Properties of Metallic Materials with Specific Reference to Their Use as Fuel Cladding in Sodium Cooled Thermal Reactors
5603	UC Test Element in SRE Th-U Core
5649	Irradiation of 7-Rod U-10 Wt. % Mo Fuel Element, SU-9-3 in SRE
5659	Handling and Processing of UC

5700-6399

5775	SRE Operational Development
5769	Some Aspects of the Carbon-Oxygen-Sodium System
✓ 5798	Investigation of Sodium Leak Plugging
5817	Radiological Implications Concerning the Use of Argon as Core Cover Gas in SGR's
5842	Physical Condition of SRE Zirconium
5853	SRE Temperature Coefficient
5916	Effect of Hydrogen Content on the Tensile Properties of Zirconium
5920	Hydraulic Test Procedures
5989	Analysis of Transient Thermal Behavior of UO_2 Experimental Fuel Elements in SRE After Loss of Pump Power
6064	Calcium Nitride in Sodium
6130	Process Design Basis for SRE
6190	UC Compatibility with 304 Stainless Steel
6221	Results of Thermal Cycling on Proposed T/C Splice for Experiment SU-20
6287	Designs of Helical-Rotor Electromagnetic Pumps for SRE-PEP Program

NAA-SR-TDR Number	Title
6356	Reactivity Effect in the SGR Critical Assembly of SRE Fuel-Pin Bowing
	6400-9999
6539	Design of Cylindrical Eddy-Current Brake
6688	Indentification of Real and Spurious Scrams Initiated by the SRE Log N - Period Circuit
6753	Cylindrical Eddy-Current Brake Designs
7101	HNPF Control Rod Drive-Speed Determination and Evaluation
7134	Measurements to Determine Thermal Utilization in Hallam Exponential Lattices Containing UC Fuel Elements
7212	Fuel Handling Machine Shieldisng Requirements for SRE Core III
7336	SRE Dispersion Fuel Feasibility Study
7348	Tube Side Flow Distribution Effects on Heat Exchanger Performance
7360	Irradiation History of AI-3-1
7477	Distributions of Temperature and Thermal Stresses in UC Fuel
7497	Carbon Transport From Ruptured Moderator Can into Sodium
7936	SRE Reactivity - Calculation Technique
8041	Concerning the Hazard of Freon Vapor in the SRE
8336	Coefficient of Friction Between Zirconium Alloy II and Graphite
8400	Modification of SRE Auxiliary Heat Transfer System
8406	Hydraulic Studies of a Proposed O-Rod SRE 3rd Core Fuel Element
8438	Design and Analysis of a 5-Rod Uranium-Carbide Fuel Element
8586	Sodium Level Indicator Tests
8983	Inherent Core Stability
9183	Moderator Heat Transfer Analysis for SRE-PEP Third Core
9207	Criticality Study SRE Core III Fuel Element Fabrication
9264	Temperature Coefficients Calculated for SRE-PEP Core III
9406	SRE-PEP Title I Nuclear Analysis
9523	Period Spikes Related to SRE Operating History
9739	Heat Transfer Analysis of Irradiation Bowed Moderator Element for SRE-PEP
9741	SRE-PEP Transient Excursion Analysis

NAA-SR-TDR Number	Title
9763	Graphite Shrinkage and Predicted Bowing of the SRE-PEP Moderator Elements
9789	The Creep Properties of Zircaloy-2 at 1200° and 1300°F
9850	Feasibility Report - SRE Core III
	10000-
10149	SRE Hot Trapping Experience During Operation with Core II
10207	Postirradiation Examination of the SU-9-3 Fuel Element
10266	Metallurgical Evaluation of Zircaloy-2 Skin From SRE Moderator Can G-12
10593	A Summary of SRE Sodium Systems Exposure History
10631	Nuclear Analysis for Initial Loading of SRE-PEP Core III
10745	Estimated Helium Solubility in Sodium
10758	Hot Cell Examination of SRE Control Rod No. 3
11044	Experimental Investigation to Evaluate Creep-Fitting the Sheath on Moderator Elements for SRE-PEP Core III
11158	Stress Analysis of the SRE-PEP Moderator Elements
11163	Hydraulic Studies of the SRE-PEP Core III Process Channel
11319	Isothermal Fatigue Tests of Typical Zr-II Corner Specimens of an SRE-PEP Moderator Can
11370	Compatibility of Dsyprosia with Hastelloy X at 1850°F
11556	Reliability Test of an SRE-PEP Moderator Element
11828	SRE Multi Element Shipping Cask
11986	Evaluation of Ferritic Steel in the Secondary Sodium System Rev.
12221	Examination of the Use of Plutonium Fuel in SRE
12223	SRE Testing System - Component - Irradiation for the Liquid Metal Fast Breeder Reactor Program
12253	Comparison of Fast Flux in SRE-PEP Core III and the SEFOR Mockup
12261	Further Examination of the Use of Plutonium Fuel in SRE
AI-67-TDR-15	Failure Rate Estimates of Some Nuclear and Conventional Power Plant Components

APPENDIX B
RETIREMENT PROCEDURES

Experience using the prepared procedures was quite satisfactory in performing the work necessary to retire the facility. Where difficulty was experienced in following the prepared procedures, necessary field changes were approved by the shift or facilities supervisor. These difficulties arose principally because of the lack of complete accurate or up-to-date construction drawings and documentation during preparation of the procedures. For example, some procedures were inadequate because the condition of the equipment was not as described in the procedure. In other cases, previously decommissioned equipment had not been fully removed, thus interfering with the procedural sequence. Another problem during the retirement activities was the necessity to identify many valves by tag because the tags had presumably been destroyed during the plant operation. Evidently the usage of the valves by the experienced operators had made it unnecessary to retag them during operation; however, this would not hold true in the future when the experienced personnel will no longer be available.

The staff performing the retirement procedures found that the most satisfactory procedures were those that were prepared in a manner which would allow personnel unfamiliar with the facility to do the work. There were no problems which resulted in hazardous conditions. This was due in part to the extensive review that the procedures received by management and by specialists trained to evaluate the procedures for hazard potential.

Several procedures used during the retirement efforts may be of particular interest to others who may be faced with similar problems in the future.

1. Decontamination of the Obsolete R/A Liquid Waste Storage Tanks

These two 5000-gal tanks are constructed of black iron and have been buried in dirt fill since the construction of the SRE (Figure 11) in 1955-1956. Initial peak radiation levels in the tanks from the liquid waste and sludge present were 1.8 r/hr beta and 0.2 r/hr gamma. Air samples indicated the presence of Sr-90 and Cs-137. The decontamination effort consisted of the following steps:

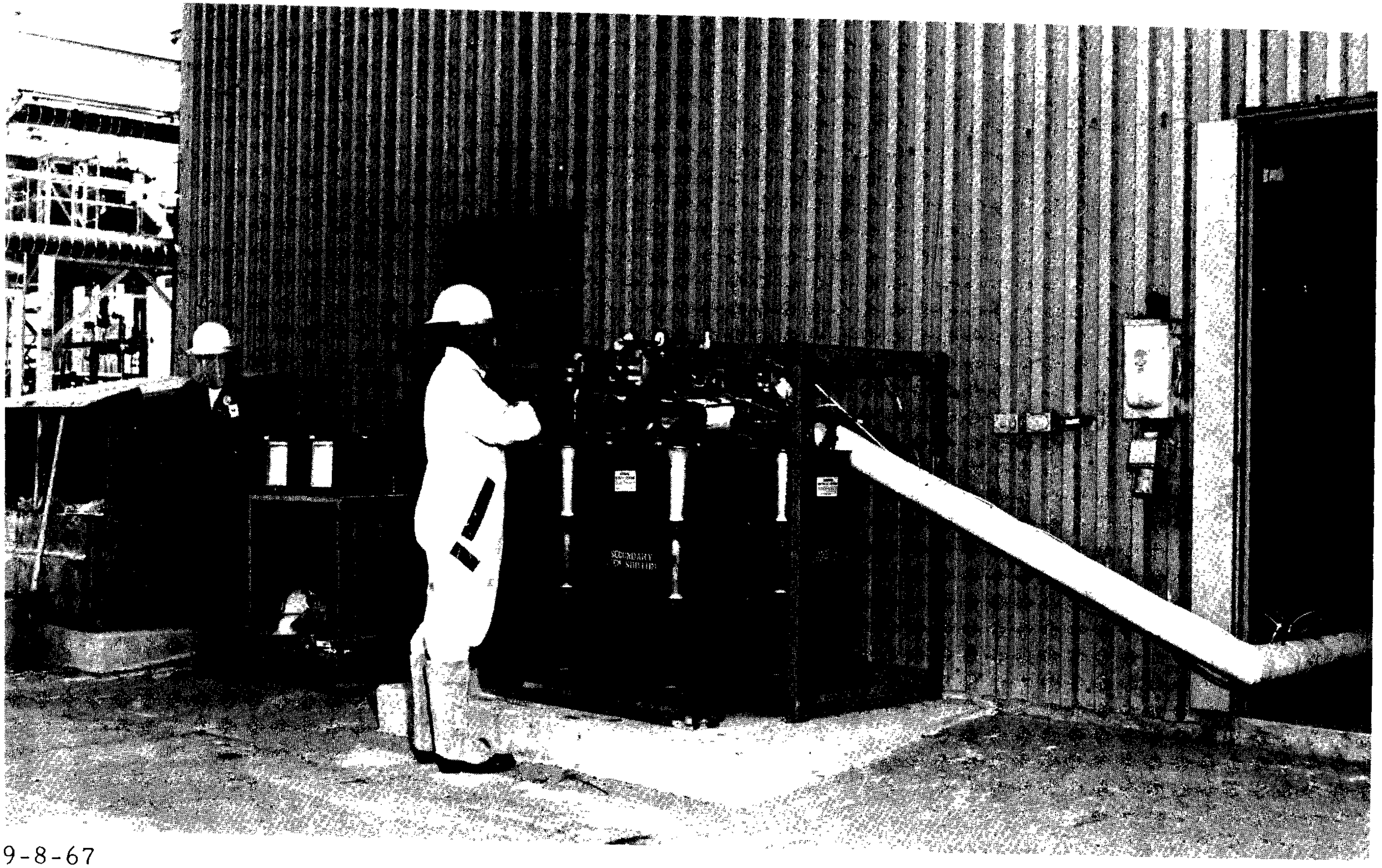

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Figure B-1. Secondary Sodium Drainage Operation

1) Remove the liquid and sludge, utilizing an industrial vacuum cleaner to the extent possible.

2) Spread a quantity of liquid-absorbing agent "Dri-zit" to absorb the remaining liquid and to dry out the sludge.

3) Vacuum out this mixture. (It was possible to remove all material from the tank by this treatment.) After this the measurable beta gamma contamination ranged up to 15 mr/hr/100 cm^2.

4) The tanks were then flushed with "TURCO" (commercial decontamination agent) and the material was vacuumed out of the tank after each flushing.

5) The tanks were then entered by personnel, properly suited and wearing full-air breathing face masks, to wipe down the surfaces with rags using water and detergent "Alconox." Areas were marked off inside the tanks in order to more easily monitor the progress of the decontamination.

6) Scrub brushes were used on the more persistent areas of contamination

7) During the cleanup process it became evident that a downcomer (pump-out) line would have to be removed to meet the level of decontamination required. These lines were removed and the opening sealed with an in-place application of styrofoam.

8) This process was repeated until the average contamination levels were less than 5000 dpm/100 cm^2.

The effectiveness of this cleaning method is best demonstrated by Figure 11 which shows extensive pitting into the internal surfaces of walls.

2. Secondary System Sodium Removal

Sodium was removed from the Secondary System Drain tank through an existing stub-out at the bottom of the tank. A fully insulated line with installed electrical heaters was run from this stub to outside the building where four 55-gal drums at a time were set on a wooden pallet to be filled from a common manifold. Thermocouples were placed along the line to supply signals to control equipment to maintain the drain lines at the required temperature of ~250°F.

The liquid level within the drums was detected during the filling operation with a simple, electrically insulated "spark plug" probe that completed an electrical circuit on contact with the shorting liquid metal. The argon gas system was used to flush the empty drums prior to filling and as an inert gas for storage. There was no difficulty with the wooden pallet during the transfer of the hot sodium into the drums; drum transfer and final storage were facilitated by use of the same pallet, eliminating any handling of individual drums. Figure B-1 is a photograph of the draining operation into the storage drums.

3. Wash Cell Vent System

The wash cell vent system as described in the SRE-PEP Reactor Safety Analysis Report[1] was constructed on a 12 x 2-1/2-ft pallet. The primary purpose of this equipment was to oxidize the effluent, which contained hydrogen as a result of the sodium-state reaction in the wash cell, to separate the effluent into liquid and gaseous products and to transfer them to the respective hold-up tanks. Kerosene coolant and water supply connections were also necessary for this function. On retirement, the radiation activities throughout this system varied from 40 to 100 mr/hr at the surface. Decontamination of this system would have been expensive and difficult; however, the pallet mounting allowed the unit to be boxed in a 12 x 2-1/2 x 2-1/2-ft crate and shipped to the Beatty, Nevada site for burial "as-is." This method of disposal was possible as a result of prior engineering consideration which had been given to the problem, resulting in a final considerable cost saving in disposal without jeopardizing the functional requirements during operation.

4. Contamination Surveys in Pipe Runs and Vessels

A procedure was developed to determine the level of contamination remaining in closed lines of the R/A waste lines. The procedure consisted of drilling holes into the line, taking the required smears through these lines, then tapping these holes and inserting a pipe plug into the tapped holes.

APPENDIX C

DISPOSITION OF EXCESS PROPERTY

The disposition of the items from the SRE have been made to programs and/ or laboratories as directed by the AEC. The items are listed below. More complete description of each item can be obtained from the referenced reports as indicated:

Transferred to the Liquid Metal Engineering Center (LMEC)

(Reference AI Document TI-599-19-002, Rev. A)

Page	Line	Item
8	27	Rain Gauge
8	28	Hygro-Thermograph
8	29	Wind Speed/Direction Recorder
9	7	Hand Tachometer
9	11	Transistor Tester
9	12	Battery Tester
9	13	A. C. Ammeter
9	14	D. C. VTVM
9	15	Millivolt Source
9	17	Humidigraph
9	24	Calibrator
9	25	VTVM, Mod. 311
9	37	Megger
9	38	Hand Pyrometer
9	39	Megger
9	41	Tube Tester
9	42	Oscilloscope
10	1	Oscilloscope Plug-In Unit
10	3	Manometer, Mercury
10	4	Manometer, Water
10	5	Gauge Tester, Dead Weight
10	8	Amplifiers, A00-7
10	9	Amplifiers, A00-6
10	15	A. C. Voltmeter
10	16	D. C. Voltmeter
10	24	VTVM, Mod. 209A
10	34	Digital Voltmeter
10	36	VTVM, Mod. 400D
10	41	Function Generator
10	42	Function Generator
11	1	Function Generator
11	2	Function Generator
11	4	A. C. Clamp-On Ammeter
11	14	Voltage Recorder
11	26	A. C. - D. C. Preamplifier
11	27	A. C. - D. C. Preamplifier
11	34	A. C. - D. C. Preamplifier
11	35	A. C. - D. C. Preamplifier
12	3	Power Supply
12	4	Power Supply
12	5	Power Supply
12	6	Power Supply
12	12	Analog Computer
12	13	Analog Computer
12	14	Analog Computer
12	15	Function Generator
12	16	Function Generator
12	17	Problem Board
12	29	Oscilloscope, Dual Beam
12	30	Plug-In Unit
12	31	Plug-In Unit
12	32	Scope-Mobile
12	33	Power Supply
13	15	Amplifier, Mod. A00-6
13	19	Problem Board
13	20	Function Generator
13	21	Function Generator
13	22	Time Base Generator
13	23	Time Base Generator
13	24	Quarter Square Multiplier
13	25	Quarter Square Multiplier
13	26	4 Channel Recorder
13	27	Time Interval Meter
13	28	Time Interval Meter
17	19	Recorder, Variplotter
17	20	Recorder, Variplotter
18	37	Analyzer, Ph
19	4	Temperature Sensing Device
19	5	Reference Junction, T/C
19	6	Reference Junction, T/C
24	3	M. G. Set
25	16	Generator, Techometer
25	18	Generator, Techometer
27	5	Pump w/Motor
29	14	Tank, S. S. Water
31	11	Hydro Lift
31	12	Chain Hoist, Electric
31	15	Chain Hoist
31	16	Hydro Set
31	17	Hydro Set
31	18	Hydro Set
31	19	Hydro Set
32	2	Fork Lift, Electric
32	3	Fork Lift, Gasoline
32	4	Lift, Personnel
32	6	Crane, Mobile Boom
32	8	Sandblaster
32	9	Drill Press
32	14	Hacksaw, Power
32	16	Sander, Power
32	17	Grinder, Power
32	19	Vulcanizer, Rubber "0" Ring
32	22	Engraver
32	23	Ramset Stud Driver
33	2	Caliper
33	3	Caliper
33	4	Transit, Surveyors
33	9	Beam Balance
33	10	Chart Viewer
33	17	Refrigerator

(Reference AI Document TI-599-19-102, dated October 24, 1967)

Transferred to LMEC (Cont'd)

Page No.	Line No.	Description
5	11	Auxiliary Air Blast Heat Exchanger
9	12	Transient Voltage Indicator
9	13	AC-DC Decade Resistor
9	14	Decade Voltage Divider
9	15	Resistance Tester
9	16	Impedance Bridge
9	20	Differential Voltmeter
9	25	Millivolt Potentiometer
9	29	Hand Tachometer, Model 2302M
9	33	Resistance Bridge, Model 5305
10	1	Potentiometer, Model 9B
10	3	Industrial Analyzer, Model 639
10	6	Potentiometer, Secondary Standard
10	11	Differential Voltmeter, DC, Model 801
10	23	Thermocouple Reference Junction
10	28	Expanded Scale Voltmeter
10	37	Low Level Preamplifier, Model 150-1500
11	10	Voltohmeter Model 630A
11	34	Universal Potentiometer, Type K3
12	3	Audio Signal Generator, 205 AG
12	4	Audio Signal Generator, 205 AG
13	6	Recorder, Temperature
13	15	Recorder, Dual Pen
13	16	Recorder, Temperature
13	30	Recorder, Temperature
13	35	Recorder, Temperature
14	3	Recorder, Millivolt
14	8	Recorder, Millivolt
14	16	Recorder, Millivolt
14	36	Indicator Controller, Temp.
14	37	Indicator Controller, Temp.
14	38	Indicator Controller, Temp.
14	39	Indicator Controller, Temp.
14	40	Indicator Controller, Temp.
14	41	Indicator Controller, Temp.
14	42	Indicator Controller, Temp.
14	43	Indicator Controller, Temp.
15	1	Indicator Controller, Temp.
15	2	Indicator Controller, Temp.
15	5	Indicator Controller, Temp.
15	10	Indicator Controller, Temp.
15	11	Indicator Controller, Temp.
15	12	Indicator Controller, Temp.
15	13	Indicator Controller, Temp.
15	14	Indicator Controller, Temp.
15	15	Indicator Controller, Temp.
15	16	Indicator Controller, Temp.
15	21	Indicator Controller, Temp.
17	24	Alarm, Windows

Transferred to LMEC (Cont'd)

Page No.	Line No.	Description
17	35	Vacuum Tube Voltmeter
17	36	Vacuum Tube Voltmeter
22	2	Motor, Electric
25	14	Tank

(Reference AI Document TI-599-19-102, Revision A, dated February 12, 1968)

Page No.	Line No.	Description
10	32	Low Level Preamplifier Model 150-1500
10	33	Low Level Preamplifier Model 150-1500
10	34	Low Level Preamplifier Model 150-1500
10	35	Low Level Preamplifier Model 150-1500
10	36	Low Level Preamplifier Model 150-1500
15	6	Indicator Controller, Temp.
24	8	Pump with Motor
24	9	Pump with Motor

Transferred to FFTF Program

(Reference AI Document TI-599-19-102, Dated October 24, 1967)

Page No.	Line No.	Description
4	4	Plugging Meter (1 unit only)
4	5	EM Pump (1 unit only)
4	7	Secondary System Valves (See Note 1)
4	8	Dehumidification System (See Note 2)
5	1	Vault Dehumidification System (See Note 2)
5	3	Auxiliary Secondary Pump
5	5	Main Secondary Pump
7	41	AC Operated High Volume Air Sampler
7	42	AC Operated High Volume Air Sampler
9	8	Variac, 115 V in, 130 V Out
9	22	Millivolt Potentiometer
9	24	Temperature Potentiometer
9	31	Amp-probe, Model 10 (1 unit only)
9	32	Variac, 115 V in, 135 V Out
9	34	Micro-microammeter, Model 410
9	39	Phase-sequence Meter
9	43	Motor Rotation Tester
10	15	DC Regulated Power Supply
10	22	Variac, 115 V in, 135 V Out
10	25	Dymec, Cabinet with 4 Amplifiers
10	38	Low Level Preamplifier, Model 150-1500
10	39	Low Level Preamplifier, Model 150-1500
10	41	Carrier Preamplifier, Model 150-1100
11	2	AC-DC Preamplifier, Model 150-1000

Transferred to FFTF (Cont'd)

Page No.	Line No.	Description
11	3	DC Coupling Preamplifier, Model 150-1300 Z
11	5	Power Supplies, Model 150-400
11	6	Power Supplies, Model 150-400
11	7	Power Supplies, Model 150-400
11	8	Power Supplies, Model 150-400
11	11	Volt/ohm Meter, Model 260
11	14	Power Supplies, Model 150-400
11	15	Power Supplies, Model 150-400
11	16	Power Supplies, Model 150-400
11	19	2-Channel Recorder
11	20	4-Channel Recorder
12	6	O_2 Detector
12	7	O_2 Detector
13	2	Recorder, Dual Pen
13	3	Recorder, Dual Pen
13	5	Recorder, Temperature
13	7	Recorder, Temperature
13	8	Recorder, Temperature
13	9	Recorder, Temperature
13	11	Recorder, Temperature
13	12	Recorder, Temperature
13	14	Recorder, Dual Pen
13	17	Recorder, Temperature
13	24	Recorder, Dual Pen
13	26	Recorder, Temperature
13	39	Recorder, Millivolt, Dual Pen
14	2	Recorder, Millivolt
14	5	Recorder, Millivolt, Dual Pen
14	12	Recorder, Millivolt
14	13	Recorder, Millivolt
15	3	Indicator Controller, Temp.
15	4	Indicator Controller, Temp.
15	7	Indicator Controller, Temp.
15	9	Indicator Controller, Temp.
15	17	Indicator Controller, Temp.
15	18	Indicator Controller, Temp.
15	19	Indicator Controller, Temp.
15	20	Indicator Controller, Temp.
16	13	Controller, Series 60
16	14	Controller, Series 60
16	20	Recorder
16	21	Recorder
16	22	Recorder
16	23	Recorder
16	31	Recorder, pneumatic
16	32	Recorder, pneumatic
16	33	Recorder, pneumatic
16	34	Recorder, pneumatic
16	40	Recorder, pneumatic
17	42	Analyzer, Oxygen
17	43	Analyzer, Oxygen (See Note 3)

Transferred to FFTF (Cont'd)

Page No.	Line No.	Description
18	1	Analyzer, Combustible
18	5	Ref. Junction, T/C
18	6	Ref. Junction T/C
18	8	Intercom System
18	11	Tachometer Head
20	8	Transmitter, Electric Flow
20	11	Transmitter, Electric Flow
20	13	Transmitter, Electric Flow (2 units)
20	14	Transmitter, Electric Flow (1 unit only)
20	16	Transmitter, Electric Flow
20	18	Transmitter, Electric Moisture
21	14	Element, Level
23	12	Blower With Motor
23	13	Blower With Motor
25	3	Pump, Canned Rotor (2 units)
25	5	Pump, Vacuum With Motor
25	8	Pump, Vacuum With Motor
25	10	Pump, Vacuum
25	11	Pump, Gas Sample (3 units)
26	11	Air Drier
26	13	Heat Exchanger, U-Tube (1 unit)
26	14	Heat Exchanger, U-Tube (1 unit)
27	4	Steam Generator
28	9	Dynamometer
28	19	Autotransformer, Variable (6 units)
29	5	Autotransformer, Variable (1 unit only)

NOTES:
1. Transfer all valves feasible to remove from the system, with the exception of the two one-inch valves authorized for the LMEC.
2. Refrigeration unit only

Transferred to ANL — Idaho

(Reference AI Document TI-599-19-102, dated October 24, 1967)

EBR-II

Page No.	Line No.	Description
4	5	EM Pumps (1 unit only)
8	7	Count Rate Meter
8	8	Count Rate Meter
8	9	Hand and Foot Counter
8	13	Scintillation Detector
8	14	Scintillation Detector
8	15	Radiation Alarm
8	17	Count Rate Meter
8	20	Count Rate Meter
8	23	Count Rate Meter

Transferred to EBR II (Cont'd)

Page No.	Line No.	Description
8	24	Count Rate Meter
10	8	Potentiometer, Model 9B
10	9	Potentiometer, Model 9B
13	40	Recorder, Millivolt, Dual Pen
14	1	Recorder, Millivolt, Dual Pen
15	33	Amplifier
15	34	Amplifier
17	28	Xenon Computer
18	2	Count Ratemeter
18	3	Scanner Recorder
18	4	Lead Pig & Photomultiplier Tube
20	14	Transmitter, Electric Flow (1 unit only)
22	5	Motor, Electric
23	16	Blower with Motor
28	8	Detector, Helium
28	10	Manipulator, Master Slave (4 units)
29	4	Autotransformer, Variable
29	5	Autotransformer, Variable (1 unit only)

ZPPR

Page No.	Line No.	Description
8	10	Vertical Lead Shield
9	11	Capacitance Test Bridge
9	23	Millivolt Potentiometer
9	27	Megohm Bridge
10	14	Potentiometer, Secondary Standard
11	26	Vibrating Reed Electrometer, Model 31-1252
11	27	Voltage to Frequency Converter, Model 2211BR
17	33	Count Ratemeter
18	30	Cooling Fan Units
18	31	Cooling Fan Units
18	32	Cooling Fan Units
18	33	Cooling Fan Units
23	15	Blower with Motor

Transferred to Task 25, AT (04-3)-701

(Reference AI Document TI-599-19-102, Revision A, dated February 12, 1968)

Page No.	Line No.	Description
15	25	Amplifier, DC
15	26	Amplifier, DC
15	27	Amplifier, DC
15	28	Amplifier, DC
15	29	Amplifier, DC
15	30	Amplifier, DC
15	31	Amplifier, DC
15	32	Amplifier, DC

Transferred to LASL

(Reference AI Document TI-599-19-102, Rev. A. Dated Feb. 12,1968)

Page No.	Line No.	Description
10	12	Amplifier
10	26	Powerstat
10	27	Powerstat
11	39	Transpac, Power supply
11	40	Transpac, Power Supply
11	41	Micro-microammeter
13	4	Recorder, Temperature
13	27	Recorder, Temperature
13	33	Recorder, Temperature
14	10	Recorder, Millivolt
14	11	Recorder, Millivolt
14	22	Indicators, Temperature
14	33	Indicators, Temperature
25	13	Tank
27	12	Hydraulic Pump
28	20	Autotransformer, Variable
29	1	Autotransformer, Variable (1 only)
29	2	Autotransformer, Variable (1 only)
29	3	Autotransformer, Variable

Transferred to NRTS

(Reference AI Document TI-599-19-102, Rev. A, dated Feb. 12,1968)

Page No.	Line No.	Description
13	20	Recorder, Temperature
13	21	Recorder, Temperature
14	9	Recorder, Millivolt
17	21	Alarm, Windows
17	22	Alarm, Windows
17	23	Alarm, Windows
17	25	Alarm, Windows
27	15	Drill Press with Stand
28	11	Manipulator, Master Slave

Transferred to PNL

(Reference AI Document TI-599-19-102, Rev. A, dated Feb. 12, 1968)

Page No.	Line No.	Description
9	42	Amperes, DC Meter
11	32	Linear Amplifier
11	44	Preamplifier
19	23	Transmitter, Pneumatic (1 only)
19	24	Transmitter, Pneumatic (1 only)
20	19	Indicator, Pressure
20	20	Indicator, Pressure
21	1	Indicator, Flow
21	10	Element, Flow
25	6	Pump, Vacuum with Motor
29	1	Autotransformer, Variable (1 only)

Transferred to ANL

(Reference AI Document TI-599-19-102, Rev. A Dated Feb. 12, 1968)

Page No.	Line No.	Description
8	12	Vacuum Pump
9	1	Potentiometer
9	45	Resistance Bridge
11	29	Electrometer Amplifier
11	35	Helipot, Lab
11	36	Helipot, Lab
13	18	Recorder, Temperature
13	19	Recorder, Temperature
13	22	Recorder, Temperature
13	23	Recorder, Temperature
13	25	Recorder, Temperature
13	31	Recorder, Temperature
13	32	Recorder, Temperature
13	34	Recorder, Temperature
13	36	Recorder, Dual Pen
14	14	Recorder, Millivolt
14	20	Recorder, Millivolt
15	35	Amplifier
15	36	Amplifier
15	37	Amplifier
15	38	Amplifier
15	39	Amplifier
16	1	Amplifier, Speedomax
16	2	Amplifier, Speedomax
16	3	Amplifier, Speedomax
16	4	Amplifier, Speedomax
16	5	Amplifier, Speedomax
16	24	Recorder, Voltage
16	25	Recorder, Voltage
18	13	Electro-pneumatic Converter
18	14	Power Range Monitor
18	15	Power Range Monitor
18	16	Power Range Monitor
18	17	Inter. Range Monitor

Transferred to ANL (Continued)

Page No.	Line No.	Description
18	18	Inter. Range Monitor
18	19	Source Range Monitor
18	20	Ann. Control Drawer
18	21	Safety Monitor Drawer
18	22	Safety Monitor Drawer
18	23	Safety Monitor Drawer
18	24	Aux. Trip Drawer
19	3	Dual Trip Units
19	4	High Voltage Power Supply
21	17	Positioner, Air Cylinder (8 units)
29	2	Autotransformer, Variable (1 only)
29	6	Autotransformer, Variable
29	7	Autotransformer, Variable (2 units)
29	14	Capacitor (3 units)
29	15	Capacitor (3 units)
29	16	Capacitor (8 units)
29	17	Capacitor (3 units)

Transferred to ORNL

(Reference AI Document TI-599-19-102, Rev. A, Dated Feb. 12, 1968)

Page No.	Line No.	Description
10	43	Servo Monitor Preamplifier
10	44	Servo Monitor Preamplifier
11	1	DC Coupling Amplifier
11	37	Servo Monitor Phase Shifter
13	1	Recorder, Temperature
14	24	Indicators, Temperature
14	25	Indicators, Temperature
14	27	Indicators, Temperature
14	28	Indicators, Temperature
14	29	Indicators, Temperature
14	30	Indicators, Temperature
14	31	Indicators, Temperature
14	32	Indicators, Temperature
14	35	Indicators, Temperature
16	19	Recorder
16	27	Recorder

Transferred to KAPL

(Reference AI Document TI-599-19-102, Rev. A, Dated Feb. 12, 1968)

Page No.	Line No.	Description
8	2	Scaler
8	3	Scaler
10	18	Decimal Counting Unit
10	19	Decimal Counting Unit
10	24	Precision Pulser
15	40	Amplifier, Speedomax
15	41	Amplifier, Speedomax
16	17	Controller, Series 60
28	18	Scale

APPENDIX D

CORE III FUEL ELEMENT INVENTORY

Element Number	Surface Contamination (d/m/100 cm^2 $\beta\gamma$)	Element Number	Surface Contamination (d/m/100 cm^2 $\beta\gamma$)
F-029	400	F-010	200
-024	150	-033	100
-011	120	-019	200
-032	600	-005 Special Test	100
-020	60	-003 Special Test	<30
-028	120	-009	150
-001	<30	-003	200
-018	100	-004	100
-030	<30	-008	200
-027	<30	-007	100
-002	4000	-022	200
-004	120	-026	120
-012	1500	-002	500
-013	<30	-015	100
-021	100	-023	100
-025	100	-014	100
-006	200	-017	100
-005	200	-016	100

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2. SRE (Nucleonics Reactor File ... No. 3) Nucleonics, December 1957

3. D. A. Mannas, "Effects of Long-Term Sodium Exposure on Materials in the Sodium Reactor Experiment," AI-66-53 (June 15, 1966)

4. A. I. Hansen, "The Effects of Long-Term Operation on SRE Sodium System Components," NAA-SR-11396 (August 31, 1965)

5. "SRE Deactivation Plan," TI-599-19-001 Rev. A (May 10, 1967)

6. "Sodium Reactor Experiment Retirement Plan (Preliminary)," TI-599-19-101 (October 12, 1967)

7. "Sodium Reactor Experiment Pre-Retirement Plan," TI-599-19-003 (November 15, 1967)

8. "Hallam Nuclear Power Facility Retirement Plan," NAA-SR-MEMO-12340 (February 24, 1967)

9. T. J. Boardman, report to be published

10. "Equipment and Materials Inventory — Sodium Reactor Experiment," TI-599-19-102 (October 24, 1967)

11. "Structural Effects of Cooling the SRE Reactor Vessel," TI-599-21-001 (May 26, 1967)

12. R. D. Welsh, "Preliminary Test of Natural-Circulation Double-Tube Steam Generator," NAA-SR-3969 (December 1, 1959)