NAA-SR-4488
COPY

SRE FUEL ELEMENT DAMAGE

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

Price $3.00
Available from the Office of Technical Services
Department of Commerce
Washington 25, D. C.

NAA-SR-4488
REACTORS-POWER
181 PAGES

SRE FUEL ELEMENT DAMAGE

AN INTERIM REPORT

OF

THE ATOMICS INTERNATIONAL AD HOC COMMITTEE

R. L. ASHLEY
R. J. BEELEY
F. L. FILLMORE
W. J. HALLETT
B. R. HAYWARD, JR.
T. L. GERSHUN
~~A. A. JARRETT~~
J. G. LUNDHOLM, JR., CHAIRMAN

GENERAL EDITOR
A. A. JARRETT

(Nov 15, 1959)

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(11-1)-GEN-8
ISSUED:

DISTRIBUTION

This report has been distributed according to the category "Reactors-Power" as given in "Standard Distribution Lists for Unclassified Scientific and Technical Reports" TID-4500 (15th Ed.), August 1, 1959. Additional special distribution has been made. A total of 700 copies was printed.

CONTENTS

CONTENTS

CONTENTS

CONTENTS

CONTENTS

TABLES

FIGURES

FIGURES

ABSTRACT

During the course of power run 14 on the Sodium Reactor Experiment (SRE) at low power, the temperature difference among various fuel channels was found to be undesirably high. Normal operating practices did not succeed in reducing this temperature difference to acceptable values and on July 26, 1959, the run was terminated. A series of fuel element inspections was begun to ascertain the cause of these circumstances, and several fuel elements were discovered to have suffered substantial damage. On July 29, 1959, an Ad Hoc Committee was appointed by Atomics International to assist in the analysis of the existing situation in the reactor and the determination of its origin.

During the three-month period since the termination of power run 14, there has been a very active program of investigation. The data accumulated during the operation of the SRE have been re-examined and evaluated. Metallurgical examination has been made of a few samples of the fuel and other components of the reactor where possible. Some chemical analysis has been made of the coolant and its contaminants. Radiochemical analyses have been made of the coolant and gaseous activity. Reactivity effects have been investigated. Some experimental programs have been initiated to examine mechanisms of damage and potential deleterious effects on the reactor system.

Tentative conclusions, based on data obtained to date by the current investigation into the causes of the fuel element damage, maybe summarized as follows: (1) The fuel cladding failed as a result of the formation of low-melting iron-uranium alloy which was produced because of partial blockage of some of the coolant passages and local overheating of the fuel elements. Coolant channel blockage was initiated by accumulation of the decomposition products of tetralin which had leaked into the primary system. Sodium oxide and sodium hydride may have contributed to this situation. (2) The high temperature runs on SRE bear no relation to the cladding failure of run 14. (3) While several possible explanations have been suggested for the reactivity changes incurred during run 14, no definitive conclusions are as yet available. (4) The extent of possible surface damage to the components of the primary system by carburization, nitriding or hydriding is not yet known. It is anticipated, however, that damage has been sustained by fuel cladding (known), moderator cans (probable), and the intermediate heat exchanger (just possible). (5) In spite of the cladding failure to at least 11 of the fuel elements, no radiological hazard was present to the reactor environs.

I. INTRODUCTION AND SUMMARY

During the course of power run 14 on the Sodium Reactor Experiment (SRE) at low power, the temperature difference among various fuel channels was found to be undesirably high. Normal operating practices did not succeed in reducing this temperature difference to acceptable values and on July 26, 1959, the run was terminated. A series of fuel element inspections was begun to ascertain the cause of these circumstances, and several fuel elements were discovered to have suffered substantial damage.

On July 29, 1959, an Ad Hoc Committee was appointed to:

1) Assist in the analysis of the existing situation in the reactor and the determination of its origin;

2) Review and advise on steps taken to remedy the situation and bring the reactor back into operation;

3) Recommend any necessary changes in operating procedures or the reactor system to prevent the occurrence of a similar situation.

This document is an interim Committee report on the origin, the nature, and the consequences of the damage to the SRE fuel, based on activities, data gathered, and evaluations performed to October 19, 1959.

For the reader who is not familiar with the SRE, the report begins with a brief description and operating history of the reactor.

The chronology of events begins in December 1958, because certain happenings at that time have a direct relation to similar events which occurred at the end of run 13 and during run 14. Most of the types of events which occurred during power run 14 had been observed, evaluated, and corrected by the SRE Operations Group during the previous two years of successful operation of the SRE. The Ad Hoc Committee has selected only those items for the chronology of events from the operating history of the SRE which are pertinent to an understanding of the fuel element damage which was observed at the termination of power run 14.

During the three-month period since the termination of power run 14, there has been a very active program of investigation. The data accumulated during the operation of the SRE have been re-examined and evaluated. Metallurgical examination has been made of a few samples of the fuel and other components of the reactor where possible. Some chemical analysis has been made of the coolant and its contaminants. Radiochemical analyses have been made of the coolant and gaseous activity. Reactivity effects have been investigated. Some experimental programs have been initiated to examine mechanisms of damage and potential deleterious effects on the reactor system. It would be impractical to include in this report all of the data and information gathered and analyzed. Furthermore, removal of components from the reactor, data gathering and analysis, and experiments to evaluate effects and consequences are still in progress. Therefore, the Committee has selected only a portion of the data accumulated to this time for this report.

Tentative conclusions, based on data obtained to date by the current investigation into the causes of the fuel element damage, may be summarized as follows:

1) The fuel cladding failed as a result of the formation of low-melting iron-uranium alloy which was produced because of partial blockage of some of the coolant passages and local overheating of the fuel elements. Coolant channel blockage was initiated by accumulation of the decomposition products of tetralin which had leaked into the primary system. Sodium oxide and sodium hydride may have contributed to this situation.

2) The high temperature runs on SRE bear no relation to the cladding failure of run 14.

3) While several possible explanations have been suggested for the reactivity changes incurred during run 14, no definitive conclusions are as yet available.

4) The extent of possible surface damage to the components of the primary system by carburization, nitriding or hydriding is not yet known. It is anticipated, however, that damage has been sustained by fuel cladding (known), moderator cans (probable), and the intermediate heat exchanger (just possible).

5) In spite of the cladding failure to at least 11 of the fuel elements, no radiological hazard was present to the reactor environs.

The investigation into the causes and effects of the fuel element damage in the SRE is still underway. At the present time all of the fuel has been removed from the core with the exception of two elements which are stuck in their process channels* and the lower sections of ten parted elements. Examinations are continuing into the condition of the primary system, the sodium coolant, and the metallurgical state of the parted fuel elements, moderator cans, and other components of the reactor system.

---

*Unless otherwise noted, "channel" refers to reactor core channel. Storage channels are always noted as such in this report. When space demands in tables and figures, core channels are identified only by "R-"; thus, R-57 would be core channel 57.

II. THE SODIUM REACTOR EXPERIMENT

A. REACTOR SYSTEM DESIGN

The following brief description of the SRE is intended only to provide the necessary background for this report. Additional details may be found in reference 1.

The SRE was designed and constructed by Atomics International, a division of North American Aviation, Inc., as part of a joint program with the Atomic Energy Commission to develop a sodium-cooled, graphite moderated, thermal power reactor for civilian application. The Southern California Edison Company installed and is operating the steam electric power generating plant utilizing heat from the SRE.

The SRE is a thermal reactor designed as a flexible developmental facility. The reactor was built as a development tool with emphasis on investigation of fuel materials. The sodium systems incorporate the use of conventional equipment to the maximum; e.g. the sodium pumps are modified hot oil pumps.

The reactor, Figure II-A-1, is cooled by liquid sodium in the primary loop. Induced Na^{24} activity in the primary loops introduces the need for an intermediate heat exchanger in which reactor heat is transferred to the secondary loop containing nonradioactive sodium. The secondary system can dissipate reactor heat either in an airblast heat exchanger or in the steam generator of the Southern California Edison Company's installation. The steam generator is normally used.

1. Coolant

Reactor coolant flow is single pass, with sodium flowing up through the core and collecting above it in a top pool. Over the top pool helium, as a blanket gas is maintained at approximately 3 psig. This pressure prevents cavitation in the suction of circulating pumps which draw heated sodium from the reactor top pool and force it through the heat transfer apparatus. The system pressure drop is about 15 psi at rated flow of 1080 gpm.

High-temperature characteristics of sodium are exploited by utilizing a ΔT of about 450°F across the reactor and heat exchangers. The design outlet temperature is 960°F.

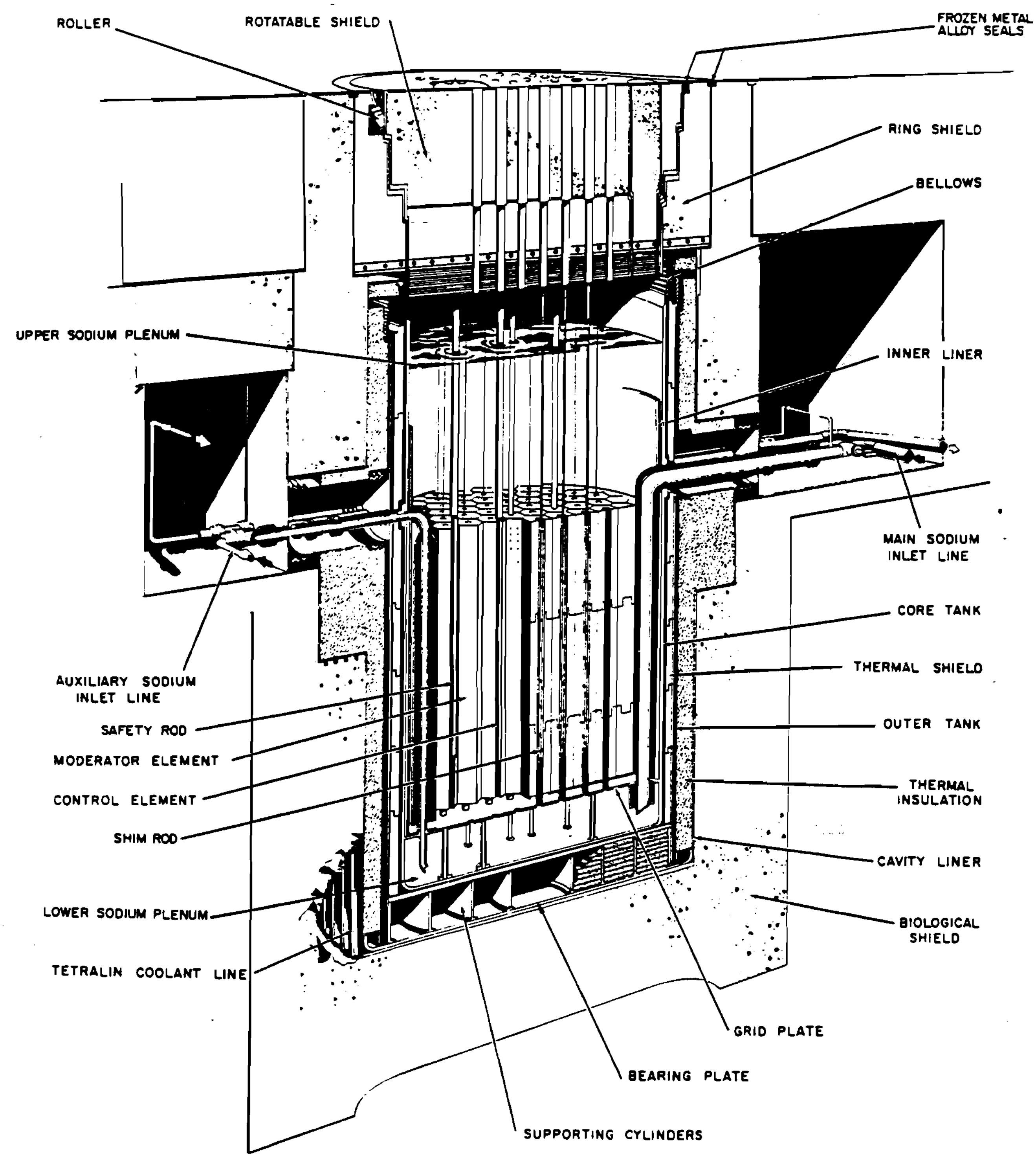

Figure II-A-1. Cutaway View of SRE Reactor

2. Moderator and Reflector Assemblies

The moderator is National Carbon Co. grade TSP graphite in hexagonal prisms measuring about 11 in. across flats. Each prism, 10 ft high, is made up of three logs stacked vertically, machined as a unit, and keyed together with cylindrical graphite plugs, as shown in Figure II-A-2.

Preventing contact with the sodium coolant is the principal problem associated with the use of graphite. Free contact would result in the absorption of sodium into void spaces in the graphite, which constitute approximately 27% of its volume. This volume of sodium in the moderator would constitute an important neutron poison. The graphite cladding is formed of zirconium sheet fabricated into individual can assemblies, as shown in Figure II-A-2. A 0.035-in.-thick sheet is used on the side panels of each hexagonal graphite column, and 0.10-in.-thick zirconium stock is used for the bottom and top can heads. The distance across the flats of each can assembly is slightly less than the 11-in. center-to-center spacing of the triangular fuel lattice. This reduction is sufficient to provide an average gap between cans of approximately 0.170 in. during normal operation. The gap forms a thin flat channel through which sodium may flow to remove heat generated within the graphite. The can is dimpled to maintain clearance.

Each moderator/reflector assembly is bolted by zirconium studs to a supporting pedestal at the base of the can, and to a spacer plate at the top. Pedestals and spacer plates are fabricated from type 405 stainless steel to minimize thermal expansion problems. The pedestal supports the can and locates it in the lattice by fitting into an accurately located hole provided in the grid plate. The bottom of the pedestal is formed as a section of a sphere which mates with a conical taper in the grid plate hole to form a seal. This seal was designed to prevent excessive leakage into the plenum formed between the top of the grid plate and the bottom of the moderator cans.

Each moderator can pedestal has a circular channel along its axis which permits flow of sodium from the main lower plenum up through the coolant tube in the moderator can assembly. The top spacer plate serves as a lifting fixture for the entire assembly and as a means of lateral support for the cans. Spacer plates on adjacent cans nest together and are maintained in place by a crimping

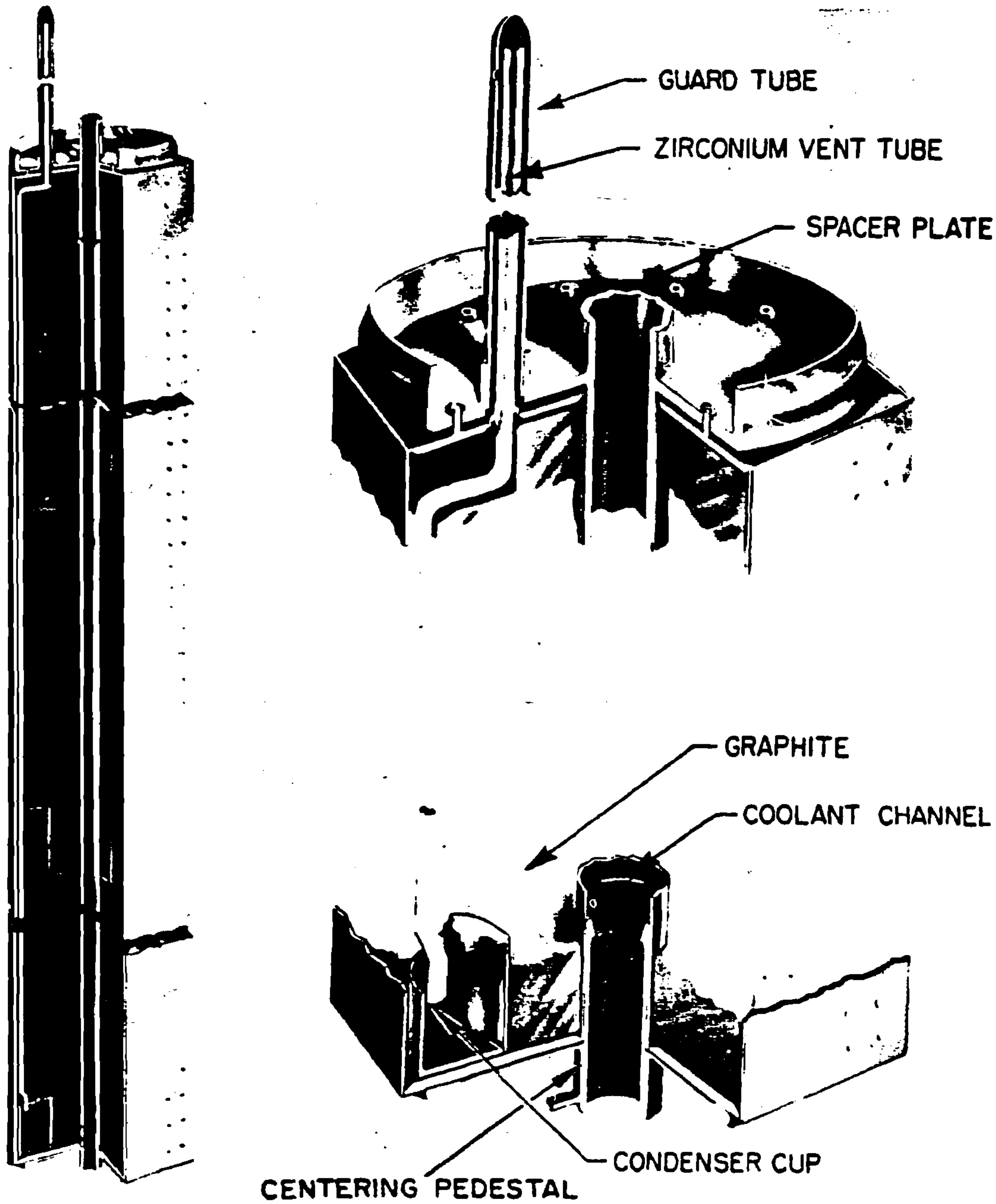

Figure II-A-2. Moderator Assembly

band around the periphery of the outer ring of cans. Each moderator can assembly in the core region is penetrated along its vertical axis by a zirconium tube of 2.80-in. ID and 0.035-in. wall thickness, welded to the bottom and top heads. Each tube normally contains a fuel element.

Some of the moderator cans were modified from the hexagonal cross section to provide a groove at one, two, or three corners for the full length of the column. These grooves, when the cans are installed, form 3-1/2-in.-diam channels at the adjacent corners of three cans and provide space for control rods, safety rods, and special reactor accessory elements, such as the source element and a liquid-level measuring device.

Reflector assemblies are generally similar to moderator assemblies except that no axial channels have been provided, and the outermost rows of reflector assemblies are canned in type 304 stainless steel sheet rather than zirconium. The types of cans are indicated in Figure II-A-3. There are a total of 119 cans in all.

The extent of outgassing from the graphite under radiation and temperature conditions expected in the reactor core was unknown when moderator can design was commenced. Consequently, it was considered necessary that provisions be made for venting the moderator/reflector assembly in order that gas

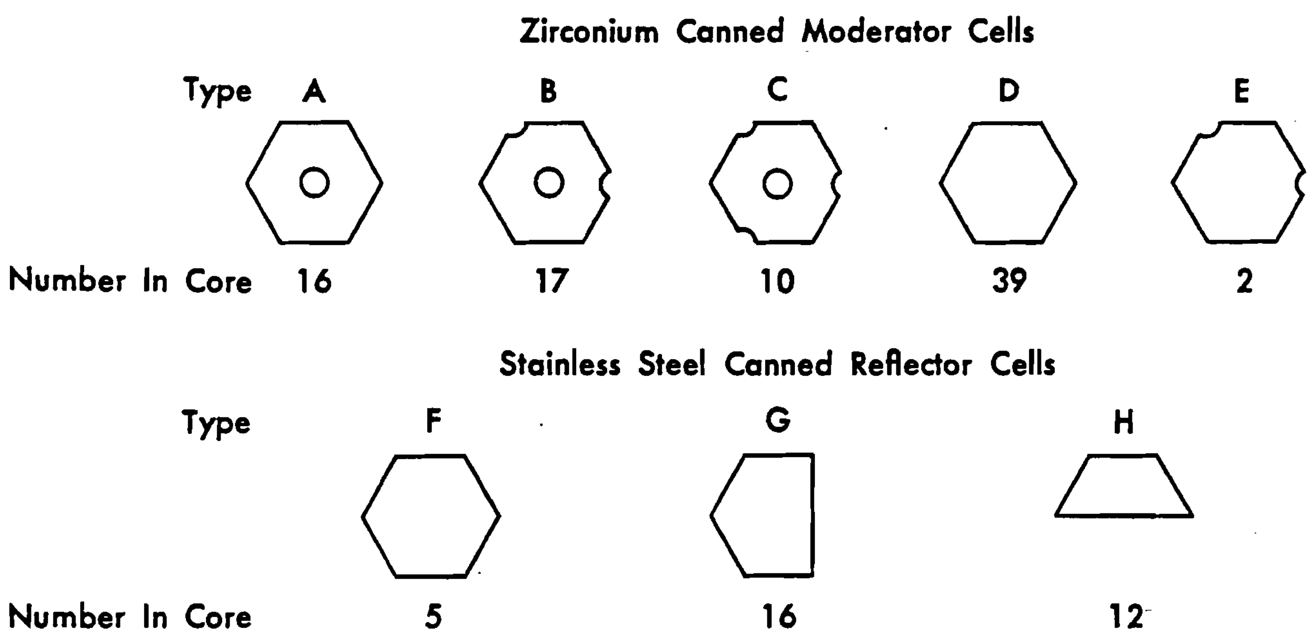

Figure II-A-3. Moderator and Reflector Can Types buildup would not bulge the cladding and distort the coolant channels. A "snorkel" tube was attached to each can, running internally to the bottom of the moderator/ reflector assembly and terminating just above a 2-1/2-in.-deep by 2-7/8-in.-diam stainless steel cup. The purpose of the cup is to accumulate any condensed sodium vapor which may pass down the snorkel tube. This 1/4-in.-diam zirconium snorkel tube projects above the can about 7 ft, so that it terminates in the gas atmosphere above the top pool. That portion of the tube which projects above the moderator can is protected by a 3/4-in. stainless steel guard tube.

3. Fuel Elements

The fuel elements, Figure II-A-4 are fabricated in clusters of 7 rods, each consisting of a 6-ft-high column of uranium slugs in a thin-walled (0.010-in.) stainless steel jacket tube. The 12 slugs are 0.75 in. in diameter and 6 in. long. They are thermally bonded to the jacket by a 0.010-in. NaK annulus. Above the column of slugs is a space containing helium. This gas-filled space allows

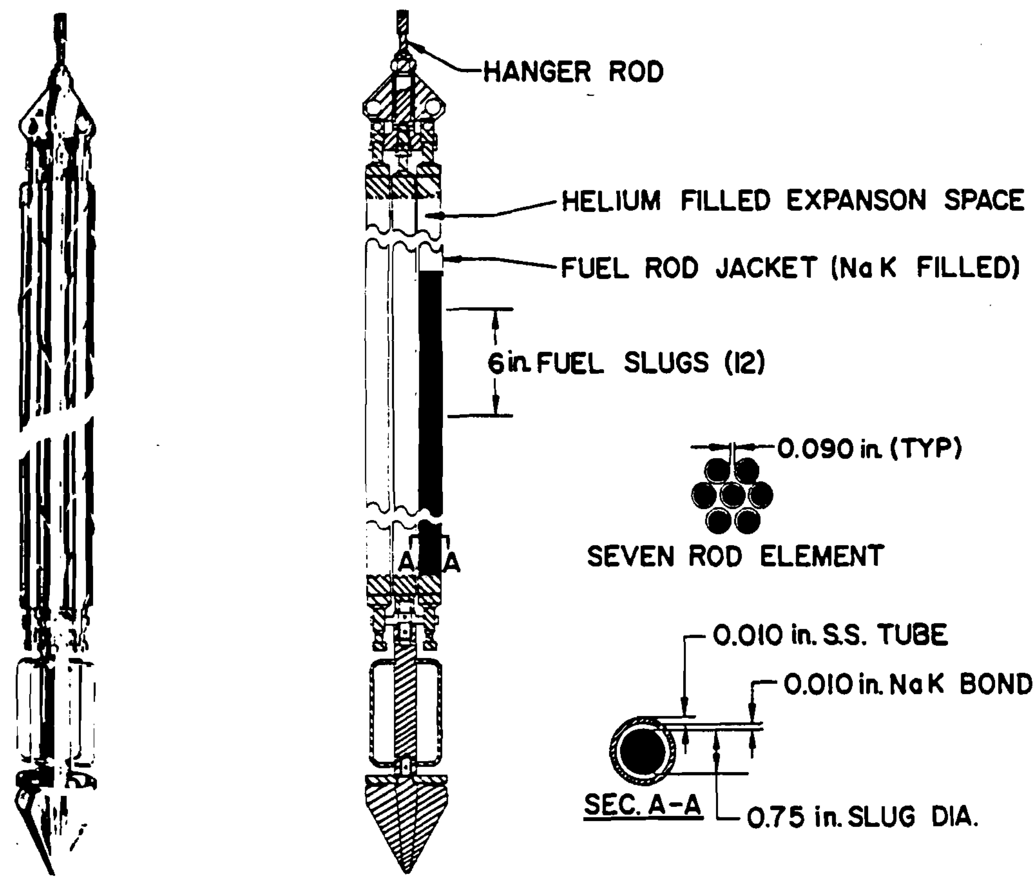

Figure II-A-4 Typical Fuel Element

expansion of the bonding NaK and serves as a reservoir for any fission gases not retained by the fuel slugs. The six outside rods of the cluster are spirally wrapped with satinless steel wire; this prevents the fuel rods from touching each other or the process channel within the moderator can. A location guide and replaceable orifice plate for controlling sodium flow are fastened to the bottom of the cluster assembly.

The cluster is supported by a hanger tube attached to a stepped shield plug. The shield plug rests in the rotatable top shield when the fuel element is inserted in its process channel. The hanger tube is designed to serve as a holddown for the moderator cans if they should, for any reason, tend to rise in the core. The tube has twelve 3/4-in.-diam drain holes cut in the wall and six 5/8 in.-diam holes in the end plate to insure sodium drainage when the element is lifted out. A thermocouple indicates the temperature of the sodium at the outlet from the fuel channel, the lead wires passing up the center of the hanger tube. The locations of fuel elements in the core are shown in Figure II-A-5.

During normal operation, there is a pressure drop of about 2.5 psi

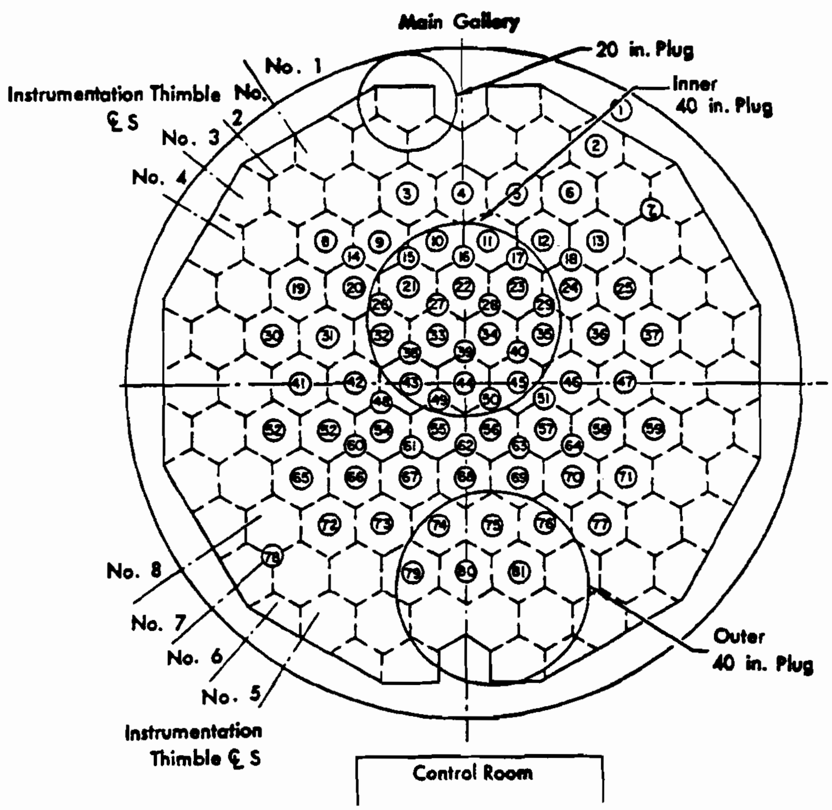

Figure II-A-5. Locations of Fuel Elements in the Core across the central fuel element and 1.5 psi across its orifice plate. The design flow past the central fuel element is 17,500 lb/hr of sodium, at a velocity of 5 ft/sec. Outer fuel clusters are orificed to pass lesser amounts of sodium, in accordance with radial power generation distribution within the core. This maintains approximately equal temperatures of the sodium discharge from each process channel into the top sodium pool.

4. Dummy Elements

Spare channels are occupied by graphite-filled zirconium thimbles called "dummy" elements. These elements increase the density of the moderator and serve to displace sodium from unused channels. The graphite is the same as that used in the moderator elements. The zirconium thimble has a 0.035-in. wall and loosely fits the diameter of the channel. The lower end is weighted with a stainless steel plug which forms a ball and socket seal in the grid plate opening. The upper end has a hanger rod similar to that used for the fuel elements except that it includes a slip joint to permit thermal expansion. A 5/16-in. tube, sealed at its extreme end, extends up from the top of the element to act as a gas reservoir.

5. Control Elements

a. Shim-Regulating Elements

There are four rods or elements, capable of regulating a total of 7.0% reactivity. Each element is contained in a stainless steel thimble assembly extending from the top surface of the rotatable top shield plug to a point just below the active section of the core. This assembly prevents contact between flowing sodium and the poison element, which is operated within the thimble. The poison column of the element is made up of a series of 18 cylinders of boron-nickel alloy suspended on a pull tube. The boron concentration in the alloy is approximately 2% by weight. This poison column is essentially black to thermal neutrons.

The rods, partially inserted in the core much of the time the reactor is critical, are cooled by conduction through an atmosphere of helium at 16 psig introduced into the top of the thimbles.

Rod motion is obtained by a ball-nut screw arrangement in which the pull tube is attached to the nut, and a drive mechanism above the top shield turns the screw. The nut is prevented from rotating by guides moving in flutes machined on the inside of the heavy wall portion of the thimble above the core. Rods have dual speed drives that produce 0.29-ft/min motion for shim action and 3.75-ft/min motion for regulating action. To avoid removal of a large amount of negative reactivity at any single time, the high speed drives are mechanically limited to a travel range without resetting of ± 7 in. The motion-limiting stops can be reset if travel through a different range of insertion is required. These rods cannot be disengaged or dropped for scram purposes.

b. Safety Elements

Four safety rods or elements may also be inserted into the core. These elements can control approximately 5.1% reactivity. They are contained in thimbles similar to those of the shim elements. However, safety elements are operated only by a high-speed drive of 3.75 ft/min for withdrawal. They have a hold magnet which can release the rod at any time during withdrawal from the core or when in the fully withdrawn position. When the magnet is disengaged by a scram signal, the poison rod assembly falls within the thimble by gravity. The safety rods, not normally inserted in the core during operation, are maintained in a helium atmosphere.

6. Experimental Facilities

Six corner channels in the reactor core are available for experimental purposes. Of small diameter, (3-1/2 in.) they extend from the top of the core to the lower sodium plenum and can be used for the full 6-ft height of the active core. Removable plugs in the top shield provide access.

Three central-channel locations in reflector assemblies are also available, at successively greater radii from the core center. Center channel openings are similar to those of the corner channels.

7. Reactor Vessel

The reactor core, Figure II-A-1, consisting of the moderator assembly, fuel rods, and control and safety rods discussed above, is situated below floor level within a stainless steel core tank, 19 ft deep and 11 ft in diameter. A stainless steel liner, supported near midpoint by brackets on the core tank, provides a 2-1/2-in. sodium annulus at the inner surface of the tank. The liner, open at both ends and pierced by the coolant circulating pipes, projects above the upper sodium pool. This liner minimizes transient thermal stresses in the core tank wall by providing a stagnant sodium layer which reduces the effect of mismatches in the temperature of the flowing sodium.

Surrounding the core tank is a thermal shield composed of a vertical stack of seven low-carbon steel rings with interlocking joints. Immediately outside the thermal shield rings is the outer tank, also fabricated of low carbon steel. This tank provides containment of sodium if a leak develops in the core tank. Both the core tank and the outer tank have bellows attached between the tops of the tanks and the shield structure directly above them. These bellows allow vertical thermal expansion of the tanks and serve as seals for the inert gas atmosphere maintained between the tanks.

A cavity liner of low carbon steel surrounds the outer tank. The annulus between the cavity liner and the outer tank is filled with a 1-ft layer of calcined diatomaceous silica and asbestos thermal insulation under an inert atmosphere. This cavity liner, which prevents ground water from permeating the thermal insulation, is cooled by circulating tetralin through steel pipes located outside the liner. The concrete shield surrounds the cavity liner.

Four concentric cylinders of low carbon steel support the flat bottom of the outer tank. These cylinders rest on circular bearing plates attached to the bottom of the cavity liner by anchor bolts which extend into the concrete foundation.

8. Instrumentation

The SRE nuclear control and safety instrumentation consists of eight channels, two for startup, two period circuits, two power channels, and two safety channels. The reactor thermal power is determined by signals from coolant delta T and coolant flow instrumentation.

The main shutdown interlock circuit system receives shutdown signals from the following instrumentation: neutron flux level, pile period, high channel temperature, moderator delta T, secondary cold leg temperature, loss of coolant flow, power failure, earthquake, and manual functions. Upon activation of the shutdown system, the four safety rods are released from holding magnets and drop by gravity into the reactor core. Setback is obtained by driving the shim-regulating rods into the core.

Each fuel element is equipped with a thermocouple located in the process channel near the bottom of the hanger rod. These thermocouples measure the outlet sodium temperatures of each fuel channel directly. The temperatures are continuously scanned and recorded on multipoint recorders in the control room. In addition to the above, six standard fuel elements and four which contain experimental fuel materials, have thermocouples in the center of the fuel slugs at various elevations in the active core and measure directly the fuel temperature. Other thermocouples located in dummy elements measure moderator coolant temperature. There are themocouples to measure the core inlet and outlet sodium temperature and at various locations in piping and apparatus throughout the plant.

9. Shielding

A 4-ft-thick, reinforced-concrete pad poured on a limestone base supports the cavity liner. An annular cylinder of reinforced concrete about 3 ft thick surrounds the cavity liner.

The top biological shield which terminates at floor level, is made of magnetite iron ore aggregate set in concrete. It is a ring-shaped shield supported on a ledge at the top of the cavity liner. The 70-ton ring shield has three steps to prevent radiation streaming. A circular rotatable shield is supported on these steps and a gas seal between the rotatable shield and the ring shield is made by a lip and trough arrangement filled with a low-melting alloy. Both the ring shield and the rotatable shield are of dense concrete (3.7 gm/cm^3), 5-1/2 ft thick.

The rotatable top shield is a stainless steel shell filled with magnetite iron ore and dense concrete grout. It weighs approximately 82 tons when all internal plugs are in place. Eighty-one small plugs, two 40-in.-diam plugs, and one 20-in.-diam plug extend through the shield. The large plugs are located so that removal of any graphite assembly from the core tank may be achieved when the shield is rotated to a proper position. The small plugs provide access to the core for fuel rods, control and safety rod thimbles, neutron sources, and experimental assemblies. Attached to the underside of the concrete in the top shield is a plate of low carbon steel. The lower surface of this plate is in contact with a layer of lead in which coolant tubes are embedded. Immediately below the lead is a stainless steel seal plate. The periphery of this seal plate is welded to the steel side shell of the rotatable shield. Suspended horizontally from the bottom seal plate are a stack of thin, stainless steel plates, separated from each other by about 1/2 in., which serve as a thermal radiation shield. Each plug in the top shield is stepped to prevent radiation streaming, and is filled with concrete and aggregate or lead to provide the same degree of shielding as the rotatable shield.

Eight 1-3/4 in. ID radial tubes embedded in the concrete outside the reactor extend from floor level downward ending just within the cavity liner and below the core center. These tubes contain the neutron detectors.

10. Cooling System

The primary and secondary systems each have two separate circulating loops, a main loop designed for transferring 20,000 kw of heat and an auxiliary loop for transferring 1000 kw of heat, Figure II-A-6. The auxiliary loops are for removal of afterglow heat in the event of outage of the main loops.

Sodium flows through 6-in. stainless steel pipe in the main primary and secondary circuits at a velocity of 13 ft/sec. Pressure drop is 15 psi in the primary circuit and 42 psi in the secondary circuit. The auxiliary sodium velocity is 30 gpm, 3 ft/sec through 2-in. stainless steel piping with a total pressure drop of 1 psi in the primary loop and 2 psi in the secondary loop. An airblast heat exchanger is the only means for dissipating heat in the auxiliary secondary loop, while the main secondary loop is provided with both an airblast heat exchanger and a steam generator.

Inlet lines from the main and auxiliary primary coolant loops enter the core tank above the graphite assemblies and extend vertically downward in double-walled, stainless steel pipes into a plenum between the bottom of the core tank and the grid plate. An inert atmosphere is maintained between the walls of the double-walled pipe to prevent excessive heat transfer.

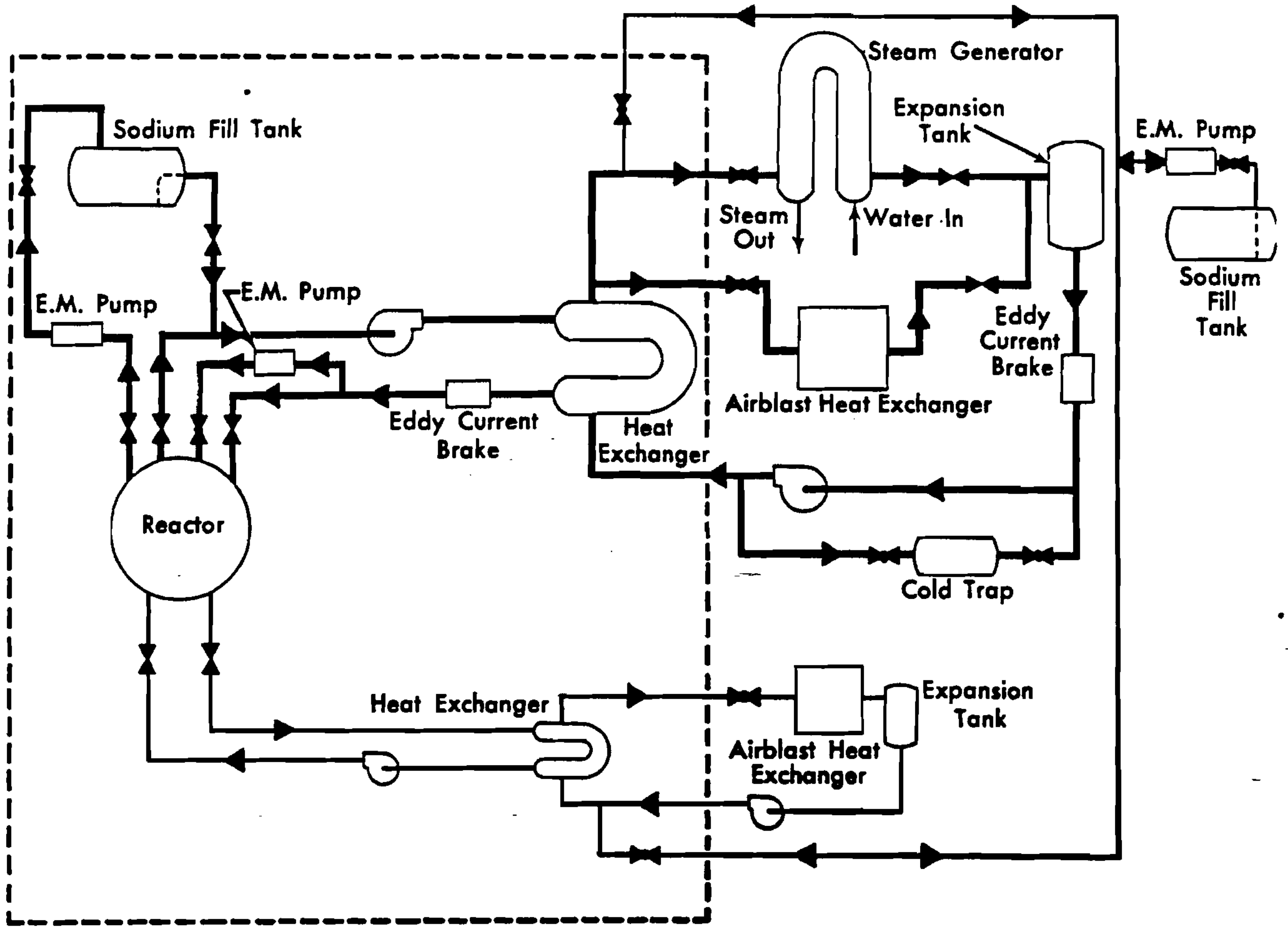

Figure II-A-6. Cooling System

At full power, the sodium at a temperature of 500°F passes from the lower plenum up through the fuel channels, absorbing heat from the fuel elements, and discharges into the upper pool about 6 ft deep at a mixed mean temperature of 950°F. Separate outlet pipes for the main and auxiliary primary loops are located in the core tank above the graphite assemblies. Moderator coolant flow is controlled by a reversable EM pump. The main source of the moderator coolant flow is leakage through the lower grid. If the leakage is not of the required rate, flow can be added to or subtracted from the grid plate leakage by the moderator coolant pump. After several power runs, the leakage was found to be about right for moderator cooling. During power run 14 the EM pump was out of the system and the moderator coolant line was capped-off.

The coolant pipes extending from the core tank are double-walled for a distance of about 6 ft, with an inert atmosphere in the annulus between pipes. this prevents gross leakage of sodium if the inner pipe containing sodium should fail. The inlet and outlet pipes are located to prevent uncovering of the fuel elements in the event of a major break in the piping. Bellows are provided around the pipe nozzles at the interface between the cavity liner and the piping gallery to permit thermal expansion of the piping while maintaining a seal for the inert atmosphere within the tank annulus.

a. Intermediate Heat Exchanger

The main and auxiliary intermediate heat exchangers are counterflow, sodium-to-sodium, shell-and-tube heat exchangers. The main intermediate heat exchanger is a U-shaped shell and tube design mounted horizontally, with a slight pitch for gravity draining. It contains 316 single-wall seamless (type 304, 3/4-in. OD by 0.058-in. wall) tubes with a total surface area of 1155 ft^2. The tubes terminate in a tube sheet at each end. Primary sodium flows in the tubes. The secondary sodium is in the shell. The unit was designed to exchange 20 Mw of heat with a log mean temperature difference of 60°F at a flow of 485,000 lb/hr.

b. Sodium Pumps

Sodium pumps are modified hot oil process pumps similar to those used in refinery services. Principal modifications are vertical mounting and the addition of frozen sodium seals at the pump casing and the pump shaft, Figure II-A-7. The conventional stuffing box around the shaft is replaced with a narrow, cooled annulus that provides a continuously shearing film of frozen sodium to seal the impeller space within the pump, Figure II-A-8. This "freeze seal" is cooled with tetralin. Main primary and secondary pumps are driven by electric motors. The main primary pump can deliver 1480 gpm against a 60-ft sodium head The main secondary pump can deliver 1240 gpm against a 140-ft sodium head. Low-power electric motors serve the two auxiliary pumps. All pumps are controlled in the reactor control room.

c. Auxiliaries

Piping and vessels containing sodium are heated by rod and strip heaters. These heaters provide system preheating and keep sodium molten

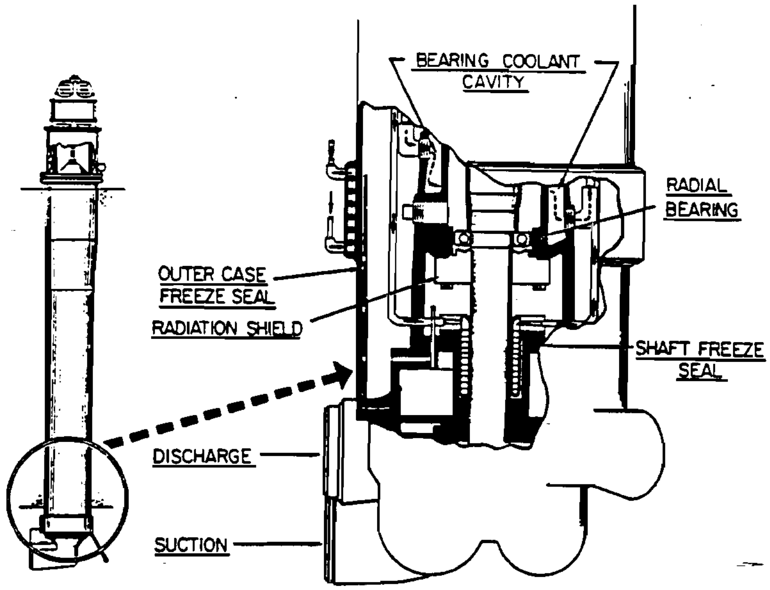

Figure II-A-7. Main Primary Sodium Pump

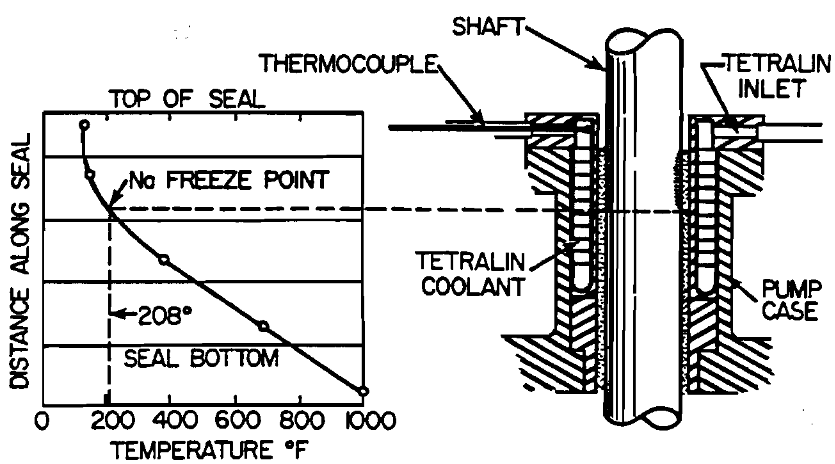

Figure II-A-8. Pump Shaft Freeze Seal

while filling and when reactor heat is not being generated. Thermocouples are attached to piping and vessels to indicate temperatures. Control of heaters, to maintain uniform temperatures, is effected by swich gear located in the reactor building. Temperature readout instruments and automatic controls are located as close as feasible to the associated switch gear. Sodium leak-detecting cable is attached to the underside of piping and vessels and leads to instruments located in the control room. Thermal insulation is strapped to all sodium containing surfaces. Electrical terminals projecting through the insulation are protected by metal tubes plugged with thermal insulation.

Hot and cold traps maintain sodium oxide content at specified low levels in the sodium cooling systems. The main primary loop has two circulating, disposable filter hot traps which can be used alternately and one circulating cold trap. The hot traps are maintained at about 1200°F by electric heaters and employ the principle of gettering on a large area of zirconium foil. Regenerative heat exchangers attached to the hot traps reduce the heat loss in this system. The main secondary coolant loop also contains a circulating cold trap. The auxiliary secondary loop is fitted with a tetralin-cooled diffusion cold trap.

One-in. pipes, welded into the inlet and discharge lines of the main intermediate heat exchanger, carry sodium to two vertical stand pipes. This permits insertion of material samples from the reactor room floor into the stand pipes, one normally containing flowing sodium at 500°F and the other containing flowing sodium at 950°F. The outlets from these two stand pipes are connected to the flowing sodium streams through stress-relieving nozzles. Samples of construction materials are inserted into this facility (Materials Evaluation Facility) to study corrosion and mass transfer effects in radioactive flowing sodium under reactor conditions.

Component cooling in the SRE is accomplished by a circulating liquid-tetralin system. The tetralin is circulated from a reservoir by two parallel electrically driven process pumps. The heated tetralin is cooled in two evaporative cooler units operated in parallel. Total cooling load can be maintained by either evaporative cooler. In the event of complete electrical failure, coolant may be circulated by a remote-starting gasoline engine attached to one of the tetralin pumps.

11. Fuel Handling

To remove any element from the reactor core it is necessary to use the shielded cask designed for this purpose, Figure II-A-9 . It is carried on the 75-ton crane bridge and may be moved as required within the reactor room. To remove an element from the SRE, the fuel-handling cask is located over the fuel element shield plug after electrical and thermocouple connectors and the retaining ring with its gasket have been removed from the plug. A pneumatic mechanism within the cask (see Figure II-A-10) forces a cylinder vertically downward to make an O-ring seal at the top of the plug casing. Following this, a large lead shield shirt is pneumatically lowered to the surface of the shield. A gas lock at the lower end of the cask, where the seal has been made to the casing, is then evacuated and helium is admitted to match the pressure existing in the core tank. A latch mechanism is lowered until it engages the top of the plug of the element to be removed. The direction of the motion is then reversed to raise the plug and attached element into the cask. A mechanism then rotates the entire lifting assembly to bring a new element into position. The procedure is then reversed to lower the new element into place, disengage the latch, retract the lifting mechanism into the cask, close the opening between the lock and cask body, flush and admit air into the lock, raise the shielding skirt, and break the seal made by the lock at the casing. The cask is then free to transport the irradiated element to the cleaning or storage facilities or the hot cell at the other end of the reactor room.

A second cask, designed to remove moderator cans, is also available (Figure II-A-11). This cask can also be modified to remove fuel elements from the core or to transport core components to facilities outside the SRE.

12. Fuel Storage Cells

There are 99 fuel storage cells located in the west end of the reactor building. The cells are steel tubes, 4 in. in diam and 21 ft 6 in. in length, recessed below floor level with access at floor level. The acess is designed to form a seal with the O-ring on the fuel plugs or control rod thimbles. Three of these cells are isolated from the others, and are designated for new fuel storage. The remaining 96 form a rectangular lattice 6 cells wide and 16 cells long. Eighty of these are cooled by tetralin circulating through a pipe welded to the

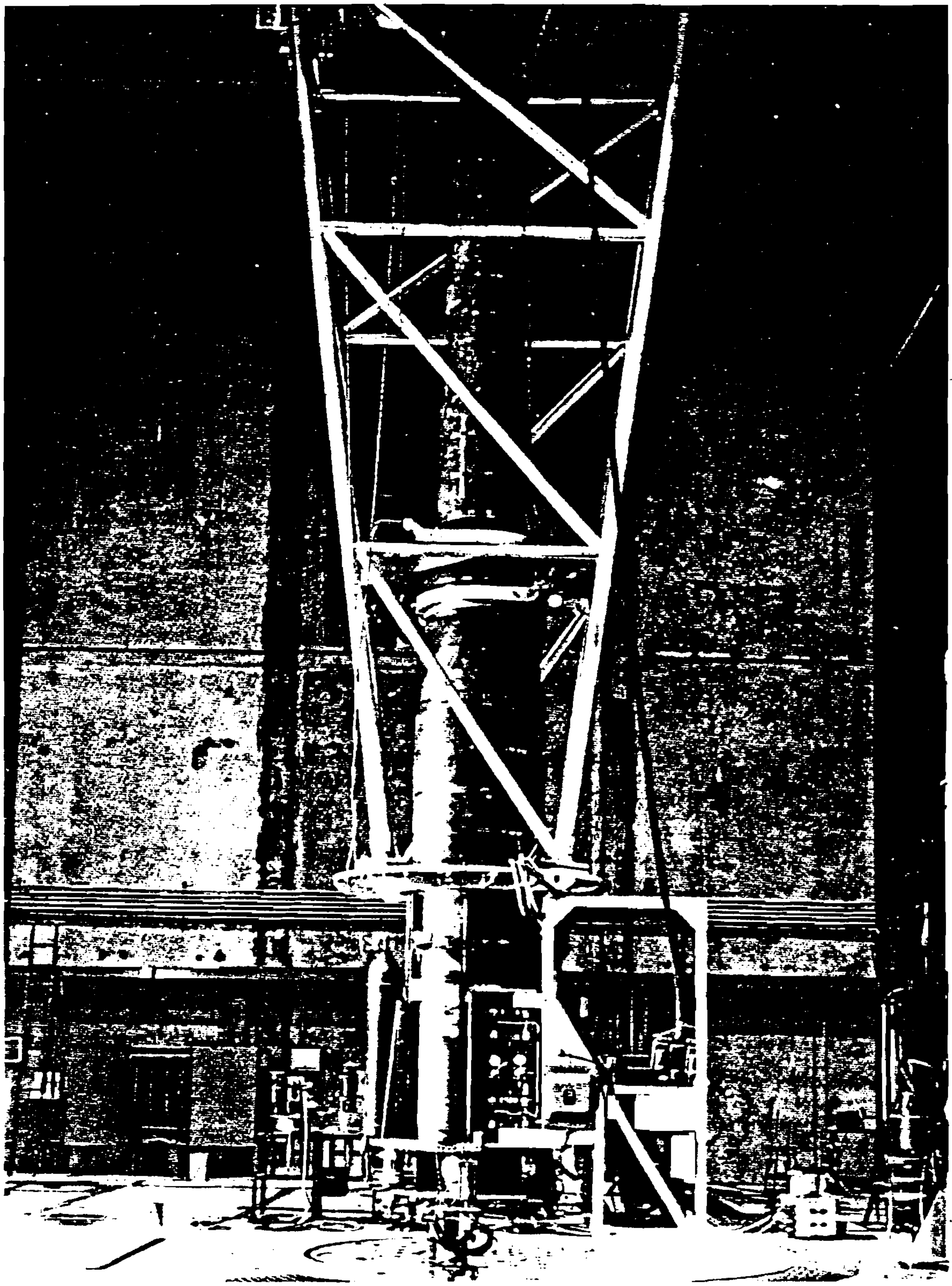

Figure II-A-9. Fuel Handling Cask

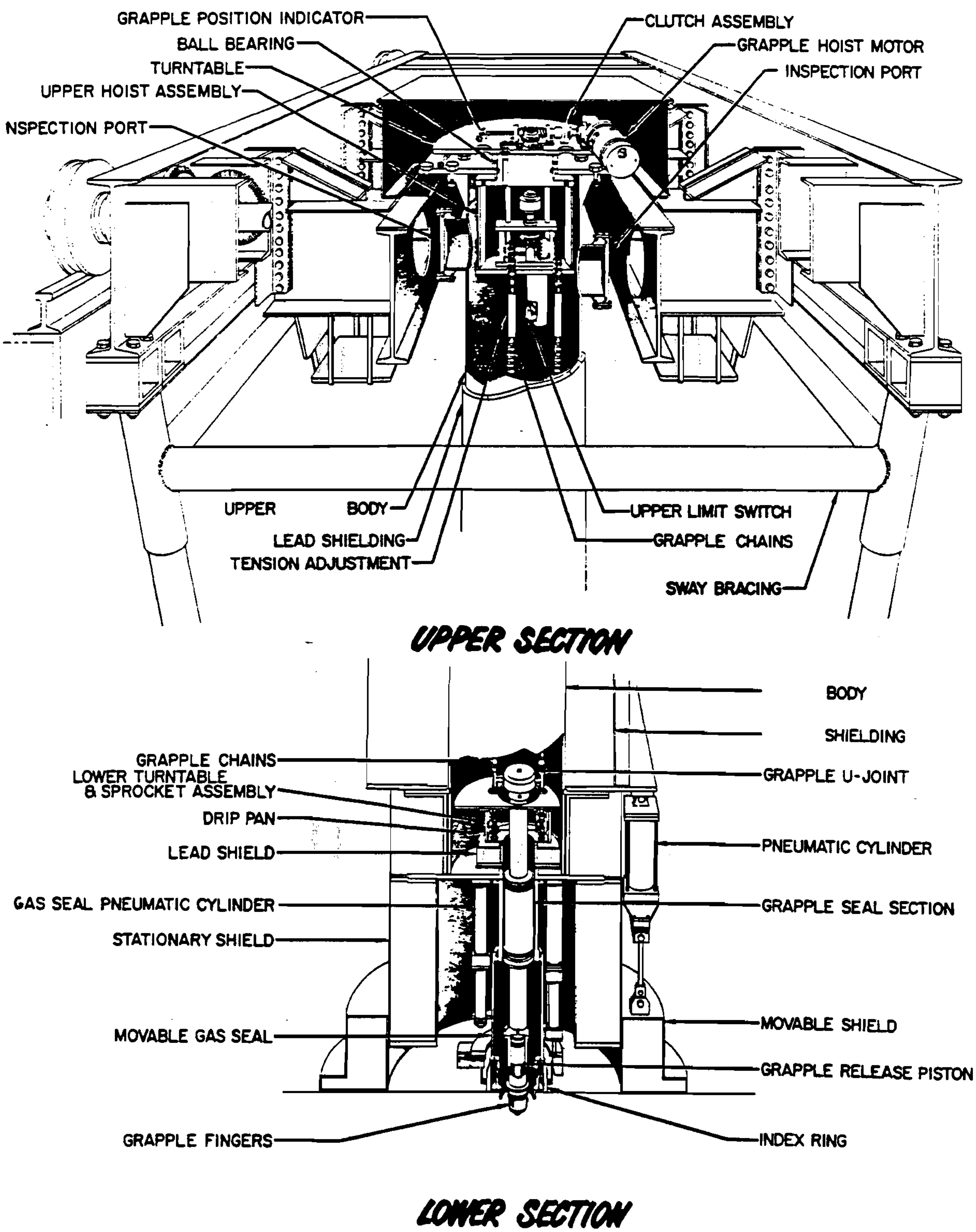

Figure II-A-10. Cutaway View of Fuel Handling Cask

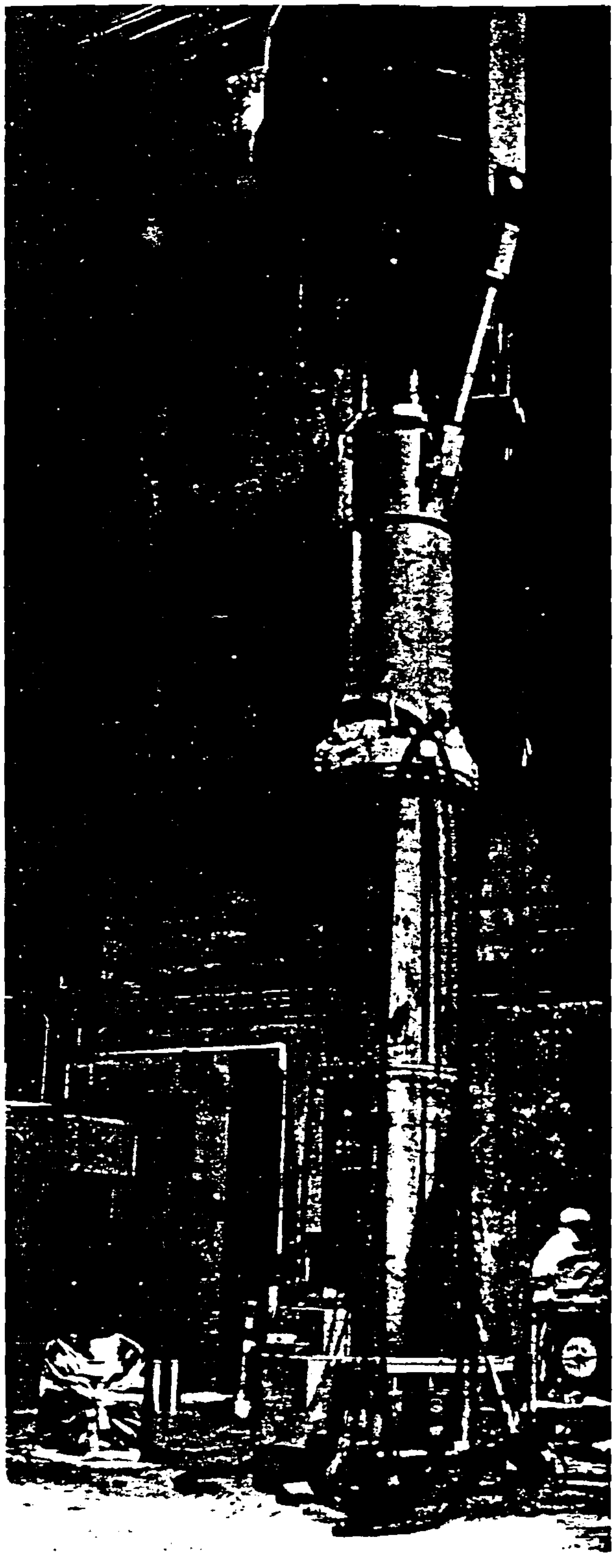

Figure II-A-11. Moderator Handling Cask

outside of the tube. An inert atmosphere is established in the cells by an auxiliary cask designed for this purpose.

13. Fuel Cleaning Facilities

Three cleaning cells are located adjacent to the fuel storage cells. They are designed to wash sodium from the fuel elements, control rod thimbles, etc., with water under an inert atmosphere. Gas liberated from the sodium-water reaction is passed to the radioactive-gas vent system. Wash water from the cells is passed to the liquid-waste system.

14. SRE Hot Cells

The SRE hot cells are located in a basement area of the reactor building. The cells are designed to afford limited examination facilities for the evaluation of fuel elements. The physical location of the cells allows direct transfer of fuel from the reactor to the cells via the fuel handling cask. This feature dictated that the cells below the level of the reactor building floor, with the transfer point within reach of the main crane.

15. Inert Gas System

An inert gas system, which supplied both helium and nitrogen to the SRE installation, furnishes nonreactive gas blankets for contact with sodium. Helium is used as the reactor sodium-pool cover gas, and nitrogen is supplies to regions (not normally containing sodium) into which radioactive sodium might leak. Individual regulators control helium and nitrogen pressures to appropriate values. These systems are protected by relief valves and flow is monitored by in-line flow meters. Helium is used also in the fuel handling cask, cleaning and storage facilities, seals for sodium pumps and valves, core tank, and all system components with a free sodium surface. Nitrogen is used to maintain a low-oxygen atmosphere in the primary piping galleries and in the cavity containing reactor thermal insulation.

16. Waste Disposal System

Operation of the SRE may generate both gaseous and liquid radioactive wastes. Vent lines are connected to the systems containing potentially radioactive gases. These gases are discharged from the building vent line to a stack on the roof of the reactor building. Gases are continuously monitored, and if activity rises above a preset level, the gases are automatically diverted to one of four shielded storage tanks. Compressors and controls are located in a shielded vault some distance from the reactor building. Radioactivity in gases normally stems from sodium vapor and impurities in cover gases exposed to a neutron field.

Liquid wastes, generated primarily from washing reactor components that have been in contact with primary coolant, are normally directed to a sump from which they are pumped into one of a series of holdup tanks for sampling. Highly radioactive waste are transferred from the holdup tank into shielded storage tanks. Low activity wastes go to portable, disposable, containers. Controls and valves for the liquid-waste system are manipulated entirely outside of the shielding barriers.

17. Emergency Electrical System

There are several sources of emergency power for the SRE if normal electric power fails. The emergency load, which approaches 50 kw, is required to maintain operation of the instrumentation and heat removal equipment.

Two sets of battery-motor alternators are provided to supply continuous emergency power and are powered by lead-acid storage batteries rated for one hour at full discharge. In addition, there is a 100-kw diesel-driven alternator that starts automatically on failure of the main electric power.

Operation of the tetralin cooling system is vital to the plant because pump and valve freeze seals depend on this coolant system for their integrity. Therefore, a remote-starting gasoline engine is connected to one of the tetralin pumps by belting, ensuring coolant circulation if all of the previously noted standby systems fail to operate.

18. Reactor Building

The reactor building is not designed as a containment pressure vessel, since the maximum credible accident would not release enough gas volume to require pressure containment. It is designed, however, to retain gases at about atmospheric pressure, and to reduce diffusion leakage of potentially contaminated gas. Aside from this, the building is designed simply to provide reactor shelter, office space, and support for a 75-ton bridge crane.

The building is of preformed tiltup concrete slabs, and the exterior shell is fireproof throughout. Nonradioactive sodium systems as well as shielded concrete vaults for primary fill tanks and service systems that are not in constant use are outside the reactor building.

Electricity is supplied by a substation capable of furnishing 1000 kw of electric power. Water, for other than drinking purposes, is supplied from local walls.

19. Steam-Electric Facilities

The steam plant is a conventional steam electric installation. A 7500-kw turbine-generator is supplied with steam from a once-through, liquid-metal to water steam-generator, providing a plant capacity in excess of rated reactor output. With the exception of the steam generator, the high-purity water conditioning system necessary for the steam generator and the interconnections between the reactor and the steam plant, all components are conventional.

B. BRIEF OPERATING HISTORY

This section of the report outlines the operating history of the SRE (see Figures II-B-1 through II-B-7). Details of the difficulties which developed during power run 14 are presented in section III.

1. Operating Data

Construction of the SRE was initiated in April 1955, and the first dry critical experiment performed in March 1957.

The reactor was first critical with sodium in the core on April 25, 1957. The turbine equipment of the Southern California Edison Co. was first operated, at low power, on July 12, 1957. During the following 10 months, a series of corrections and modifications were made to the system. The most important of these was the installation of eddy-current brakes on the main primary and secondary coolant loops to reduce thermal stress following a reactor scram.

During FY 1958 (July 1957 through June 1958) the reactor was used to generate electricity during July 12 to 14, July 25, 26, November 9 to 20, 1957, and May 21 to 28, 1958. Power levels were up to 21 Mwt and 5.8 Mwe. The total thermal energy generated in FY 1958 was 218 Mwd; the total electricity developed was 569,910 kwh. A large part of the reactor down time was consumed in reactor physics experiments.

During FY 1959 (July 1958 through June 1959) the reactor logged a total of 2191 Mwd of power operation, bringing the total to 2409 Mwd and the total electricity developed to over 15×10^6 kwh. The reactor was critical for 179 days of the year, of which 144 were at significant power levels; the remainder were at low power for experimental purposes. The last 260 Mwd of operation was accumulated at a reactor outlet temperature of 1000°F or higher, generating steam at 900°F and 600 psig. For a short period of time the reactor was operated with a peak outlet temperature of 1065°F generating steam at 1000°F. This performance exceeded the design specification (960°F outlet sodium) and demonstrated the capability of sodium-graphite reactor systems to achieve modern steam temperatures.

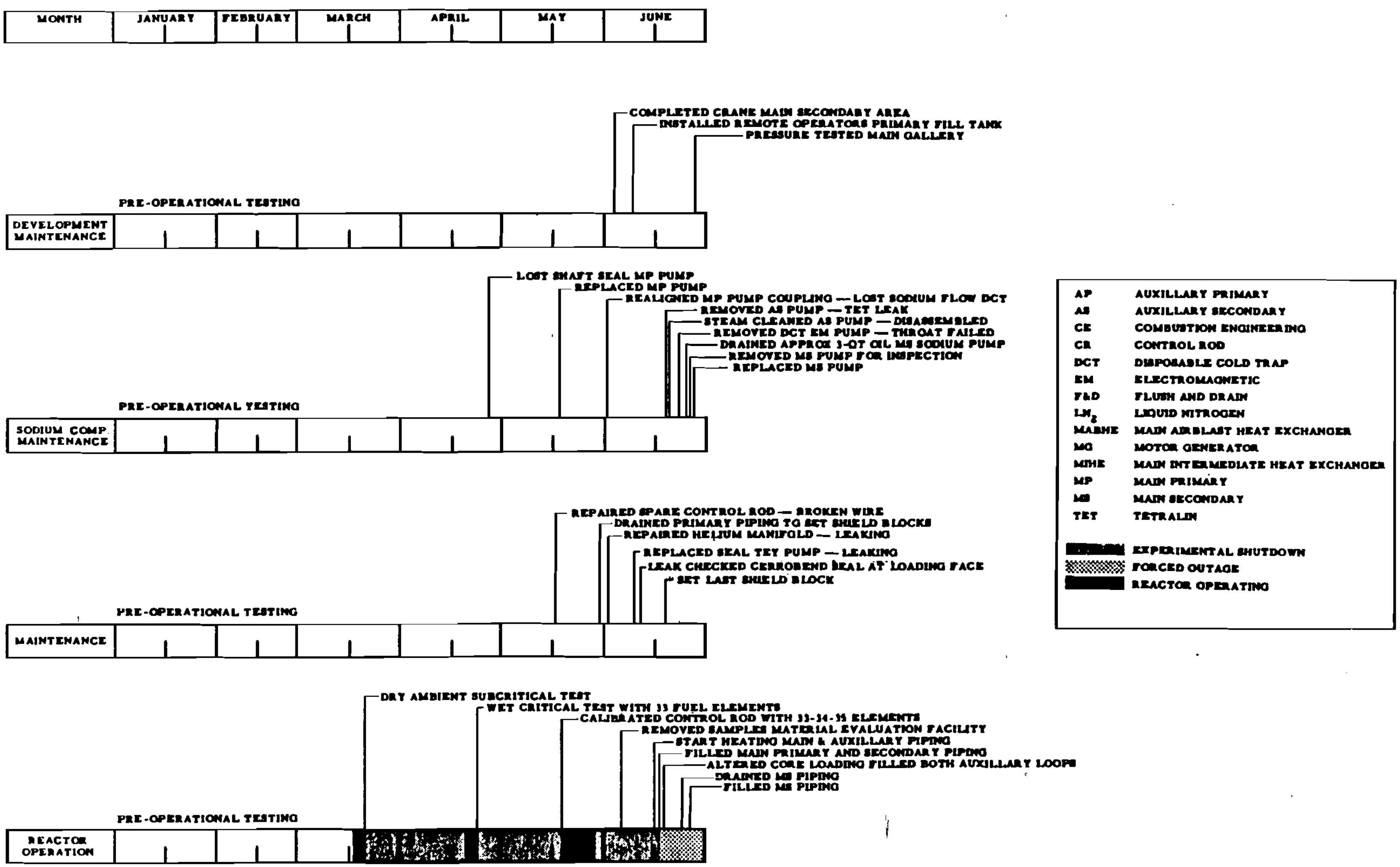

Figure II-B-1. Operating History of the SRE (FY 1957)

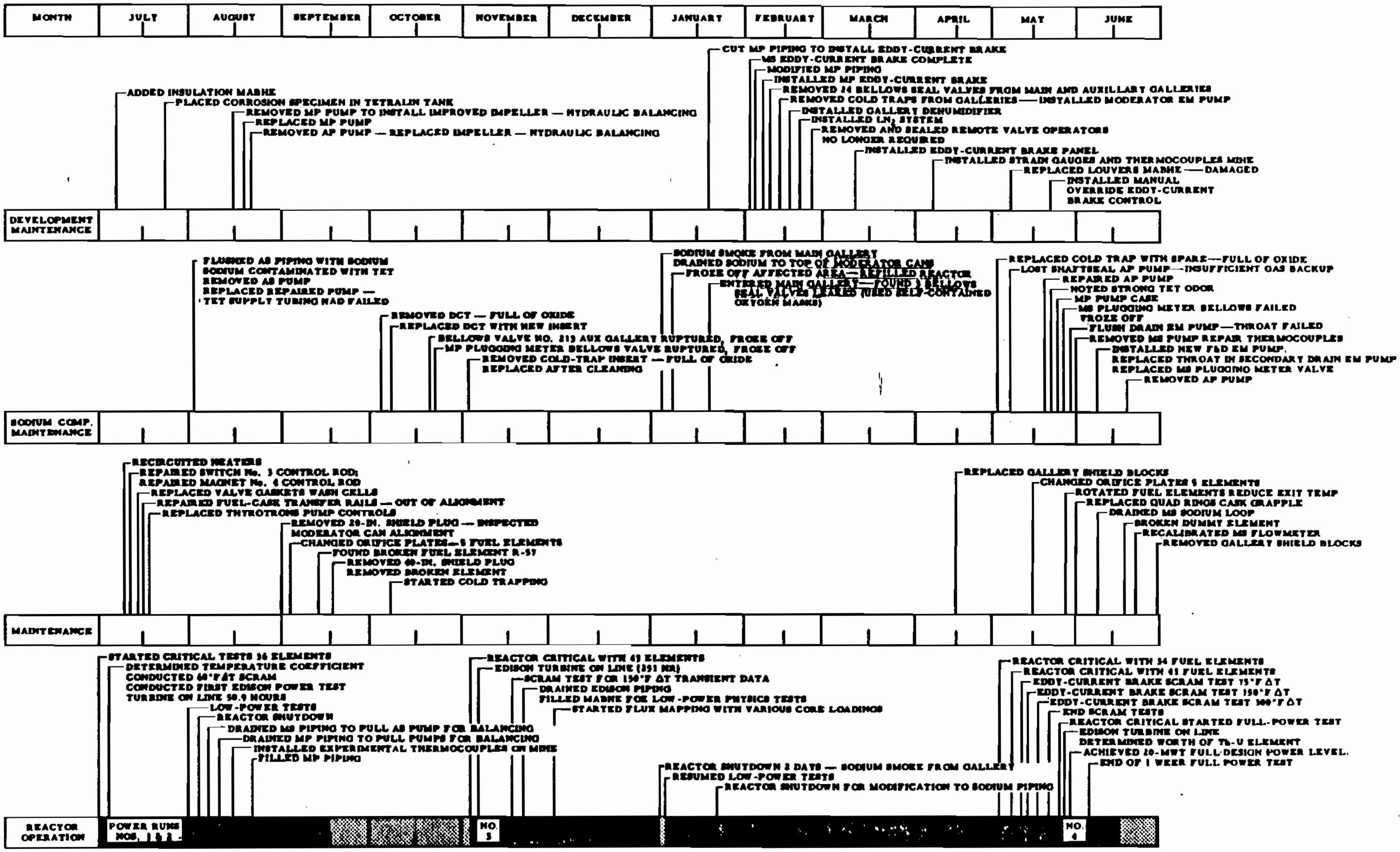

Figure II-B-2. Operating History of the SRE (FY 1958)

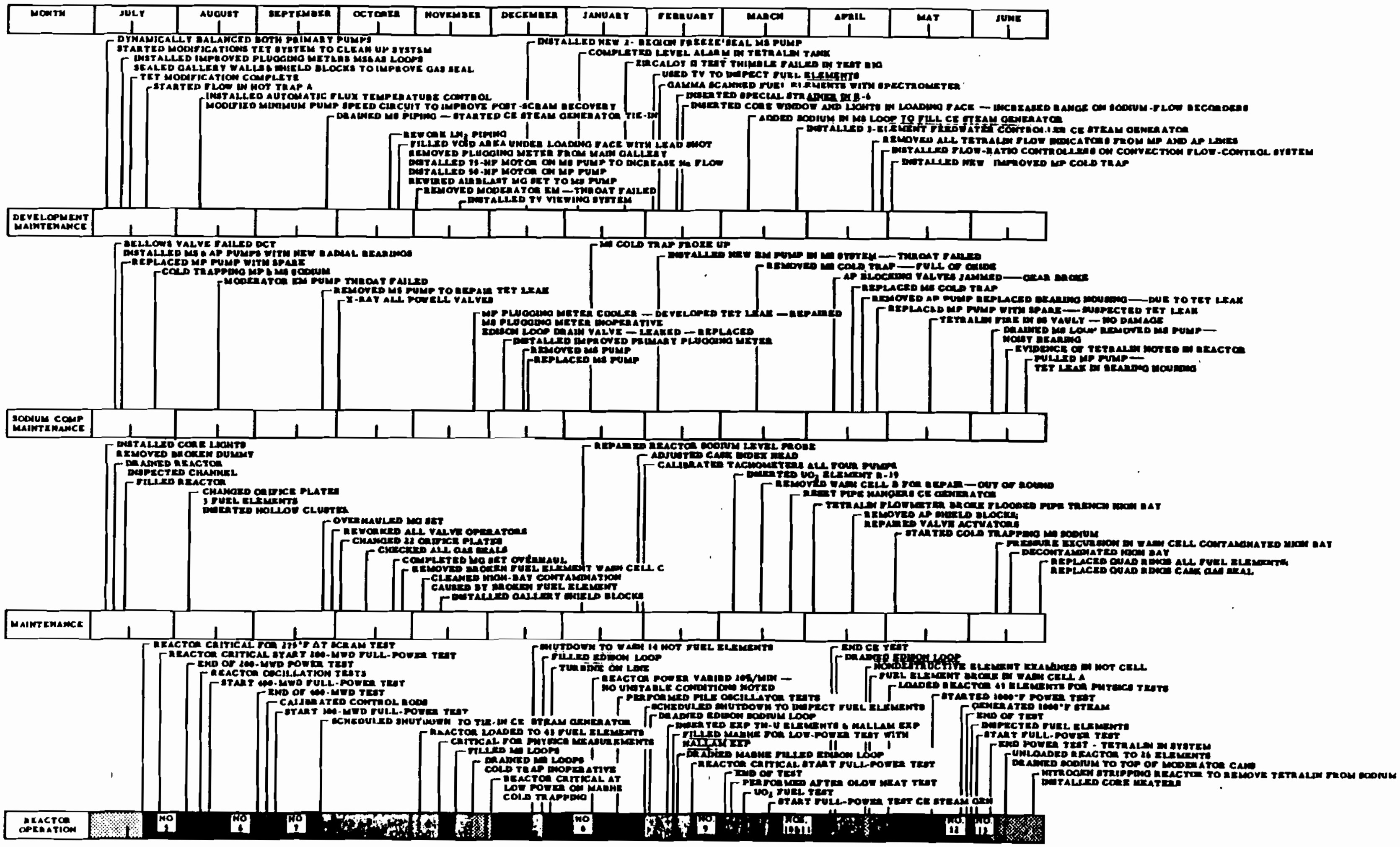

Figure II-B-3. Operating History of the SRE (FY 1959)

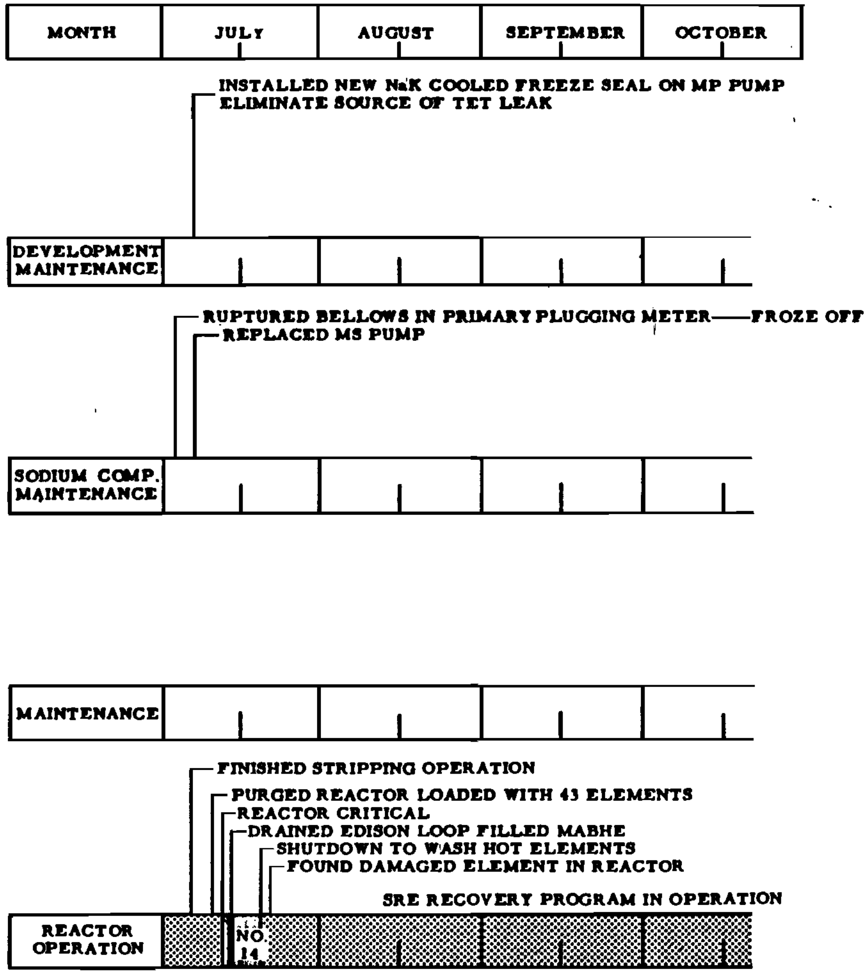

Figure II-B-4. Operating History of the SRE (FY 1960)

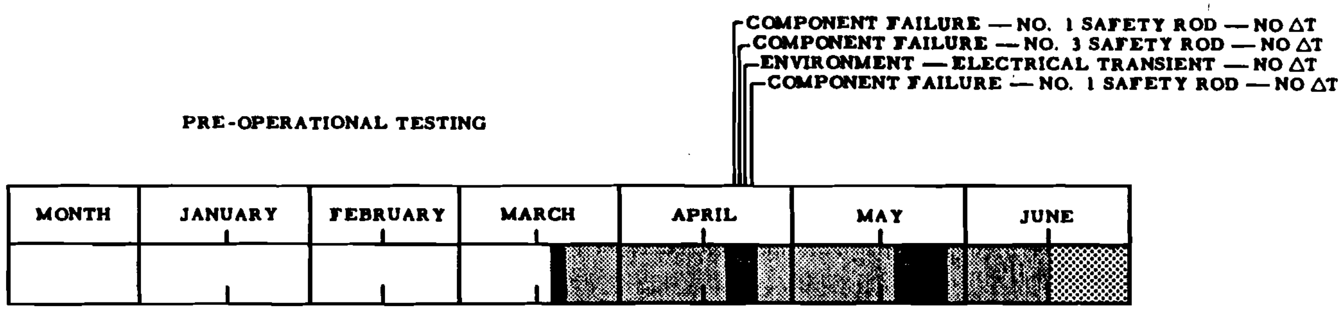

Figure II-B-5. Scram History of the SRE (FY 1957)

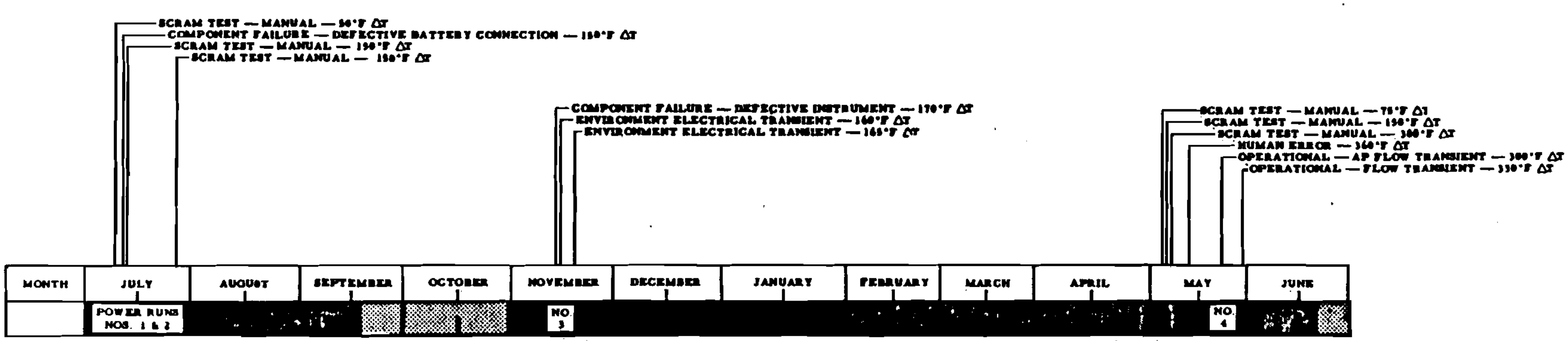

Figure II-B-6. Scram History of the SRE (FY 1958)

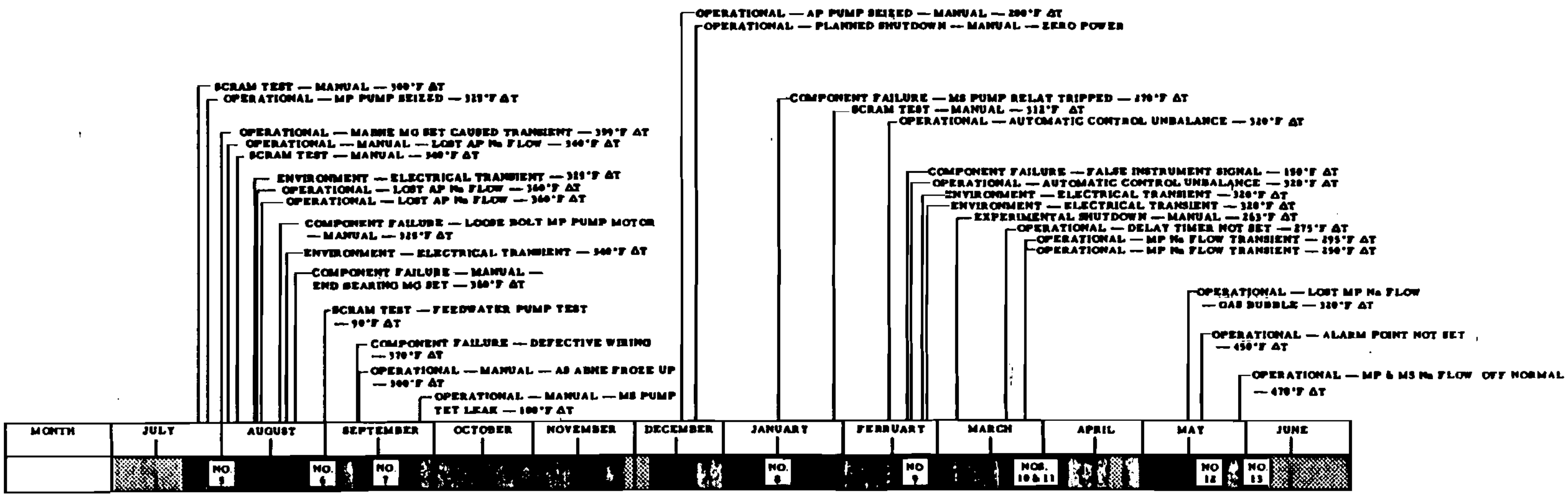

Figure II-B-7. Scram History of the SRE (FY 1959)

2. Plant Performance

a. Fuel Element Data

Operation of the SRE demonstrated that unalloyed uranium metal is an unsatisfactory fuel material for a high-temperature reactor because of a tendency to swell. Fuel rods irradiated to about 1000/Mwd/T and examined in the SRE hot cell showed an increase in diameter of 3 to 4 mils. The fuel rod was manufactured with a NaK annulus (between fuel slug and stainless-steel canning jacket) of 0.010 in. (0.020 in. on the diameter). This annulus had been occupied by the swelling uranium and the can (0.010-in. wall thickness) slightly distended.

The design maximum fuel temperature of 1220°F was never reached before run 14. Thermocouples in the uranium slugs indicated a maximum temperature of about 1100°F with a sodium outlet temperature of 1000°F. Not all of the slugs (3612 in reactor) were temperature monitored. It has been estimated that flux variations might have produced a maximum hot spot temperature of 1150°F at some point where there was no thermocouple.

The generally unsatisfactory performance of this experimental fuel material led to fabrication of a Th-U alloy loading, which is now available for use. Run 14, in which the fuel element damage occurred, was the last scheduled run for the first core loading. Use of the Th-U loading was awaiting only approval of the supplementary Hazards Report, which has now been received.[2]

b. Sodium Components

The sodium systems have generally performed well, demonstrating the feasibility of quite conventional components and fabrication techniques for use with this coolant. Two of the components, the sodium pumps and the intermediate (sodium-to-sodium) heat exchanger, have, however, caused difficulty in the SRE operations. The major difficulty with the pumps has been with the tetralin-cooled freeze seals which have leaked tetralin on three occasions. The intermediate heat exchanger has been found to be poorly baffled and stratification of the sodium on the shell side has resulted in a larger log mean temperature difference across the exchanger than was anticipated (90°F rather than the design value of 60°F at full power).

Thermal convection flow of sodium proved to be greater than desirable as indicated by analog computer studies and in-plant tests. Eddy current brakes were installed to avoid a sudden temperature decrease at the top (outlet) of the reactor following a reactor scram. This temperature transient would have induced undesirably large stresses in the outlet nozzle during a scram from full power; full power operation was therefore deferred until flow control was installed.

Several of the small, bellows-seal valves in the sodium service system failed and had to be replaced. Failures occurred at the bellows and apparently were due to extrusion of frozen sodium from adjacent piping during melting operations. The freeze-stem valves in the system have operated with 100% reliability.

Modification and maintenance of the sodium system have been accomplished with ease and safety. Piping can be cut and welded by first freezing the contained sodium. There have been no sodium fires during any operation involving cutting or welding piping containing frozen sodium. The cold traps and hot traps have performed well in removing sodium oxide; no difficulty was experienced in maintaining the oxide concentration below 10 ppm. The "once-through" steam generator has been highly reliable and has demonstrated its ability to operate in excess of design performance (900°F sodium inlet temperature, 825°F steam.) Steam at 1000°F was produced for about 1 hr at reduced power during run 12.

The reactor and sodium systems have demonstrated excellent flexibility. The ability of the reactor to change power levels quickly was demonstrated by changing from one-half to full power in 2-1/2 min. Sodium and steam temperatures changed less than 20°F in this test.

c. Reactor Control and Instrumentation

Performance of the control and safety rods in the SRE has been excellent. Examination of one control rod after an exposure of 2140 Mwd showed variations from nominal dimensions of less than 6 mils on the diameter. The major difficulty with the instrumentation has been a series of spurious scrams caused by fluctuations in voltage from the power supply.

One instrument failure of importance occurred. This was during run 14 and is described in section III. The cause of failure was readily corrected.

d. Reactor Physics Experiments

The most important physics experiments on the SRE have been in the area of kinetics. The SRE has proved to be an unusually stable reactor. During a 144-hr period of steady-state operation in full power, integrating timers on the automatic control system recorded only 3.5 min of control-rod movement. Extensive experiments with the pile oscillator have demonstrated the reactor stability at all power levels and at all frequencies. The ratio of the prompt neutron lifetime to the effective delayed neutron fraction in the SRE was measured to be 75 $\pm$7.5 msec, from which a prompt neutron lifetime of 0.525 $\pm$0.035 msec was obtained. A considerable amount of reactor time has been consumed in additional experiments such as control and safety rod calibrations, measurement of the radial power distribution, determination of temperature coefficients and danger coefficient measurements on experimental fuel elements.

e. Reactor Auxiliaries

Containment has been readily accomplished even during the cladding failure incurred during run 14. The washing of fuel elements to remove radioactive sodium has generally been successful, although some difficulties have been encountered as discussed in section III. A total of 387 fuel washings was successfully accomplished in the course of fuel element examinations. The fuel handling cask was used to effect fuel element transfers from the core on approximately 2000 occasions. It was not properly designed, however, to handle the broken elements found after run 14, and the moderator cask modified for handling fuel is currently in use.

In summary, the SRE has proved to be easy to operate once the minor difficulties with the new sodium systems were overcome. The reactor has exceeded its design performance and has successfully demonstrated the capabilities of the SGR concept.

III. CHRONOLOGY OF EVENTS

This section presents the background information necessary to an understanding of the present condition of the reactor and how it came about. It was decided to start this account at the beginning of run 8 because conditions existed at that time which were similar to those which existed during run 14 when the damage to the reactor fuel occurred. A discussion of the activities following the end of run 14 is given so as to bring the chronological story up to date. An evaluation of the events that occurred and their possible relation to the fuel damage will be given in section IV.

A. RUN 8 (November 29, 1958 to January 29, 1959)

Prior to run 8, there was an extended shutdown lasting about two months. During this time considerable repairs and modifications were made to the primary sodium system. While this work went on, the primary sodium was pumped back and forth several times between the primary loop and the primary fill tank which was known to contain a large amount of sodium oxide. This resulted in the introduction of a substantial amount of oxide into the primary sodium. Eighteen fuel elements were examined in the SRE hot cell and reinstalled in the reactor.

At the beginning of run 8 a large spread in the fuel-channel exit temperatures existed, and the plugging temperature was high. Extreme reactor operating conditions at that time are shown in Table III-1.

TABLE III-1

REACTOR OPERATING CONDITIONS, RUN 8
(0800 November 29, 1958)

Power (Mw)	3.6
Inlet Temperature (°F)	458
Outlet Temperature (°F)	565
Flow (gpm)	900
Fuel Channel Exit Temperature (°F)	
Minimum	415
Maximum	800

With the reactor power at about 3.6 Mw, there was a spread of 385°F between the highest and lowest fuel-channel outlet temperature in the reactor; under normal conditions at 20-Mw power, the maximum spread is usually less than 100°F. This unusually high temperature spread was attributed to oxide plugging in the process tubes since the oxide content of the sodium, determined from the sodium plugging temperature, was high at the start of run 8.

The reactor was shut down to reduce the oxide content of the sodium by cold trapping. On December 12, fuel elements from channels 9 and 10 which had been running excessively hot were washed. Both had black material on them before washing. Washing the fuel elements was found to be the most successful method for reducing the temperature spread. Reactor operation continued intermittently at low powers of 1 to 2 Mw until December 18. Some improvement in the fuel-channel exit-temperature spread occurred merely by running the reactor at elevated temperatures. This increases the rate at which oxides pass into solution. It was observed that moving a fuel element up and down about 1 in. or less in its process tube (jiggling) frequently had a beneficial effect on sodium outlet temperature for that channel. Movement of the element apparently caused foreign matter to dislodge from the element, particularly from the orifice plate.

On December 18, reactor power was increased to 12 Mw. In order to raise the power to this level without introducing an excessive spread in the fuel-channel exit temperatures, it was necessary to increase the primary sodium flowrate to 1400 gpm. Normally the flowrate is about 1100 gpm with the reactor operating at 20 Mw. On the next day the power was increased to 14 Mw, and operation continued at that level for several days. At 1800 on December 23, the reactor was shut down to inspect fuel elements.

Before resuming operation on December 27, 15 fuel elements were washed, and more cold trapping was done. A power level of 20 Mw was achieved on December 28 with the sodium flowrate at 1460 gpm. The practice of jiggling hot elements up and down to dislodge foreign matter was continued. The spread in the fuel-channel exit temperatures continued to improve as operation continued at various power levels until the end of run 8 on January 29, 1959.

On January 7, a sample of the core cover gas was bubbled through cyclohexane and the solution analyzed. Napthalene was identified, indicating that tetralin had entered the primary sodium at some earlier time. Prior to this analysis the presence of tetralin in the primary system was not suspected. The only known tetralin leak prior to this occurred in June 1958 when a crack was found in the bearing housing casting of the main primary pump. It is not known if any tetralin had entered the primary sodium at that time.

Run 8 was terminated because the desired exposure of 600 Mwd was attained. No reactivity anomalies were observed during this run. During the ensuing shutdown, more fuel elements were washed and more cold trapping was done. The sodium was cold trapped down to less than 5-ppm oxygen content.

B. RUN 9 (February 14, 1959 to February 26, 1959)

On February 16, reactor power of 20 Mw was achieved. However, some difficulties were still being experienced with the fuel-channel exit temperatures. Therefore, the reactor was shut down on February 18 to wash some fuel elements that had been running hot and to further cold trap the primary sodium. The fuel from reactor channels 10, 20, 36, and 42 was washed and replaced in the reactor.

Reactor operations were resumed on February 20. An examination of the records of the shim-rod positions (made after run 14) indicated that an increase in reactivity of 1/2% had occurred. Such an increase is expected because of the xenon decay during a shutdown of this length. However, a preliminary calculation of this effect indicated an expected increase in the reactivity of about 1%. It is believed that this discrepancy is due to approximations made in calculations of the xenon correction. Similar discrepancies are noted in later runs. The record shows that this discrepancy had disappeared by February 22 and that the reactivity remained normal during the remainder of this run.

After operation was resumed on February 20, the reactor power was maintained between 20 and 21 Mw. The condition of the fuel-channel exit temperatures continued to improve. There were two reactor scrams caused by excessive temperature drop across a moderator can (moderator delta T) and several scrams caused by power line transients. The reactor has a long history of scrams due to the latter effect.

Run 9 was terminated after the desired exposure of 125 Mwd was achieved. During the ensuing shutdown, the fuel element in reactor channel 56 was examined in the SRE hot cell. This examination was made as a part of the reactor fuels program and included micrometer measurements of fuel rod diameters. The orifice plate had a thin black deposit. The fuel element was washed and replaced in the reactor.

C. RUN 10 (March 6, 1959 to March 7, 1959)

This run was made for the purpose of making a temperature test on a UO_2 fuel element. No unusual circumstances were noted, and the fuel-channel exit temperature spread continued to improve. After completion of the test, the run was terminated after an exposure of 3 Mwd. During the ensuing shutdown a thimble was replaced in the reactor. An examination of the records of shim rod position (made after run 14) shows that at the start of run 11 a loss in reactivity of 1/4% had occurred. This loss may have been due to the thimble change.

D. RUN 11 (March 16, 1959 to April 6, 1959)

The reactor was operated at reduced power of 2 to 4 Mw until March 20. Some difficulties with the fuel-channel exit-temperatures spread were still being experienced. From March 20 until March 23 the reactor power was slowly increased to 20 Mw. Operation was continued at this level until March 27 when several reactor scrams occurred. These were caused by fluctuations in the main primary sodium flow, which occurred because helium had gotten into the primary sodium and helium bubbles caused cavitation of the pump. The helium is believed to have come from a leak in the Materials Evaluation Facility. The reactor was returned to 18-Mw power on March 28.

Examination of records of the shim-rod positions (made after run 14) indicated that a gain in reactivity of about 1% had occurred. This gain is due to the xenon effect but is about 1/3% greater than the gain indicated by preliminary xenon calculations. By March 31, the reactivity had returned to normal and remained so until the end of this run. This occurrence gives further evidence that approximate calculations of the xenon correction are inadequate and that reactivity changes of this nature are normal.

The reactor continued to run routinely at a power level of about 19 to 20 Mw until the end of this run. Further reduction in the fuel-channel exit-temperature spread was noted. The run was terminated after an exposure of 295 Mwd had been reached.

During the ensuing shutdown, 21 fuel elements were examined visually by means of a television camera in the fuel handling cask. They were all found to be in good condition and were reinstalled in the reactor.

About 10 days after the end of run 11 it was found that the radiation level in the main sodium gallery was somewhat higher than one would expect, although not sufficiently high to prevent maintenance work in the main sodium gallery. This observation was not surprising because it was realized that there was some fission-product contamination in the primary sodium.

A filter was installed in a fuel channel after the end of run 11, and sodium was circulated through the primary system. Further details and results of analysis of the material collected on the filter are given in section IV-B.

E. RUN 12 (May 14, 1959 to May 24, 1959)

The reactor performed normally during all of run 12. On May 15, the reactor outlet temperature of 1000°F was reached. The primary sodium flowrate was 1040 gpm. On May 22, the reactor outlet temperature was raised to 1065°F for about 1 hr with the reactor power at 6 Mw and the primary sodium flowrate of 500 gpm. During this period steam was produced at a temperature of 1000°F. No unusual reactor conditions were observed during this run which was terminated after an exposure of 154 Mwd had been achieved.

During the 4-day shutdown which followed, the fuel element from core channel 56 was examined in the SRE hot cell and found to be within satisfactory limits with regard to rod diameters. There was no measurable change from measurements made following run 9. This element was then replaced in the reactor.

On May 26, the core gas radioactivity was found to be 1.7 x 10^{-3} $\mu c/cm^3$ activity was assumed to be xenon-133 and was considered to be normal. Xenon activity had been observed after reactor operation for many months and was attributed to small pin-hole leaks in the cladding of a few elements or to uranium contamination from the outside of new fuel elements.

F. RUN 13 (May 27, 1959 to June 3, 1959)

This run was planned to be a repeat of run 12 with the reactor outlet temperature at 1000°F. An exposure of 150 Mwd was desired.

1. Initial Operation

The run proceeded smoothly at a reactor power of 20 Mw until 1124 on May 29 at which time a reactor scram occurred. This scram was caused by an abnormal sodium flow rate. Recovery was immediate and the reactor was returned to normal operating conditions and remained there until 0900 on May 30. At this time, several unusual phenomena started to occur. They were:

a) The reactor inlet temperature started a slow rise from 545°F to 580°F. The rise was very slow, exending over a period of about 3 days.

b) The log mean temperature difference across the intermediate heat exchanger started to increase which indicated an impairing of its heat transfer characteristics. A rather sharp increase in this quantity occurred on June 1.

c) A thermocouple located in a fuel slug in the element in core channel 67 showed an increase from 860°F to 945°F. This change started at 0840 and ended at 0900 on May 30. A similar thermocouple in the fuel element in core channel 36 did not show a corresponding increase.

d) Some of the fuel-channel exit temperatures showed a slight temperature increase of about 10°F.

e) The moderator delta T chart shows an abrupt jump of about 30°F at 2230 on May 30. The chart shows fluctuations of about 18°F for the 4 hr immediately preceding. Prior to this it has been quite stable.

f) The temperature indicated by a thermocouple in a probe located in corner channel 16 showed fluctuations of about 30°F. A few hours later this temperature settled down to a steady value.

g) Although it was not noted at the time because the reactor was on automatic control, an examination of the record of shim-rod position (made after run 14) showed that a shim-rod motion corresponding to a reactivity increase of about 0.3% had occurred. This change in reactivity was gradual and extended over a period of about 6 hours. Following this the reactivity showed a steady increase of about 0.1% over the next three days of operation.

2. Tetralin Leak

By June 2, it was obvious that something had occurred to impair the heat transfer characteristics of the system. It was believed that this could not be due to oxides in the sodium because the plugging temperature had been low at the start of this run. It was decided that a tetralin leak had occurred again and was the cause of the trouble. The fairly quick recognition of the symptoms of tetralin leaking into the system was the result of the experience during run 8. The odor of tetralin was detected in the pump casing of the main primary pump. When the pump was removed after the end of this run, a leak was found in the thermocouple well in the freeze seal of the pump. (Figure II-A-8)

This run was terminated on June 3 after an exposure of only 114 Mwd. The purpose of the shutdown was to repair the main primary pump and to examine fuel. Seventeen fuel elements were visually examined by means of the television camera and found to be slightly dirty but in good condition.

3. Wash Cell Incident

On June 4, an attempt was made to wash the fuel element from core channel 56. During the washing operation a pressure excursion occurred of sufficient magnitude to sever the fuel hanger rod and lift the shield plug out of the wash cell. This event may be related to the tetralin leakage that occurred during run 13. It is possible that hydrocarbons could cause a substantial amount of sodium to be trapped in the hold-down tube on the hanger rod by blocking the sodium drain holes. This sodium could then cause the reaction. As a result of this incident, no further washing of elements was done.

The reactor cover-gas radioactivity was measured on June 13, 16, and 20. The activity was found to be about 10^{-4} $\mu c/cm^3$ which is not an unexpectedly high value. More complete data are given in section IV-C.

The main primary pump was pulled on June 12. The delay of about 10 days was required to allow the radioactivity of the sodium to decay.

4. Stripping Operation

In order to remove the tetralin from the primary system, nitrogen gas was bubbled through the sodium. This stripping procedure had been used on October 12, 1958 when the secondary fill tank was purged with nitrogen in order to remove tetralin and any decomposition products.

Seventeen elements were removed from the reactor and the sodium level in the upper pool lowered to about 6 in. above the moderator cans. The nitrogen stripping process was begun on June 17 and continued until July 5 with 26 fuel elements remaining in the reactor. The sodium temperature was 350°F at the start and was raised to 425°F by the end to enhance the removal of tetralin. Nitrogen was admitted to the system through the primary sodium pump casing, passing through the heat exchanger, and then into the bottom of the reactor; 400,000 ft^3 were used.

About 3 pints of tetralin and about 1500 cm^3 of napthalene crystals of unknown density were removed from the system by this process. The stripping process was terminated when no more impurities were being removed. The system then was purged for 10 hr with 4700 ft^3 of helium and argon.

The primary sodium pump was reinstalled with a NaK-cooled freeze seal in place of the tetralin-cooled unit. The 17 fuel elements that had been unloaded were reinstalled in the reactor and run 14 was started.

G. RUN 14 (July 12, 1959 to July 26, 1959)

1. Temperature Spread

In view of the experience on run 8, this run was started with the expectation of encountering difficulties with the reactor fuel-channel outlet temperature. The reactor was brought to criticality at 0650 on July 12. Shim-rod positions at criticality were 46 in. out, which was about the expected position for criticality.

(The differential reactivity worth of the shim rods in this position is approximately 0.1%/in.) At 0835, as the reactor was slowly increasing power to about 0.5 Mw, it was noted that large fluctuations were present on the moderator delta T recorder. The magnitude of these fluctuations was about 10°F; whereas under normal operating conditions at 20 Mw, the fluctuation is usually less than 5°F. Also, the fuel-channel exit temperatures started to diverge, showing a spread of about 200°F. Some temperature spread is not abnormal at low reactor power. Operation continued at low powers less than 1 Mw until 1142, at which time a reactor scram occurred due to loss of auxiliary primary sodium flow. At the time of the scram, reactor outlet temperature was 485°F.

Criticality was re-established at 1215. Rod-position readings at 1300 were still 46 in., indicating no change in reactivity during the scram. Operation continued at slowly increasing power levels and temperatures. Typical reactor operating conditions during this period are shown in Table III-2.

TABLE III-2

REACTOR OPERATING CONDITIONS RUN 14
(1700 July 12, 1959)

Power (Mw)	2.7
Inlet Temperature (°F)	470
Outlet Temperature (°F)	550
Flow (gpm)	800
Fuel-Channel Exit Temperature	
Minimum	510
Maximum	770
Moderator Coolant Temperature (°F)	650

It should be realized that a possible systematic uncertainty of as much as 20% is present in the power levels given for run 14. This is caused by uncertainties in the heat balance data taken at low power levels. However, relative power levels as indicated by nuclear instruments are reliable.

Fluctuations of about 30°F were observed on moderator delta T temperature recorder with reactor power at about 1.5 Mw.

2. High Air Activity

At 1530, both reactor room (high bay area) air monitors showed a sharp increase in activity. In an attempt to reduce the activity level, the reactor pressure was lowered to less than 1 psig from its former pressure of 2 psig. A survey of the loading face shield revealed that an excessive radiation reading existed over the reactor sodium level coil thimble located in core channel 7. The initial reading was 500 mr/hr. A high bay air sample had an activity of 3×10^{-7} $\mu c/cm^3$ after 15 min of decay and 4.5×10^{-8} $\mu c/cm^3$ after 90 min of decay. At 1620, it was noted that the filter from the air sampler showed an activity level of 160,000 c/m. At 1700, a sharp increase in the stack activity to 1.5×10^{-4} $\mu c/cm^3$ was noted. This returned to normal by 2200.

By 1700, the radiation level over core channel 7 had reached 25 r/hr. It was decided to shut the reactor down and replace the thimble with a standard plug. Accordingly, at 1730, reduction of reactor power was begun. At 2057, the reactor was shut down, the drive units removed, and the cask placed in operation. The sodium-level probe was removed from core channel 7, a shield plug installed in its place. A manual sodium-level probe was inserted in a spare thimble located in core channel 50. The level of the sodium was found to be 121 in. below floor level, which is normal. This device was used to check sodium level until the sodium-level indicator was reinstalled in the reactor on July 15. While the reactor was shut down, the temperature of the reactor slowly decayed to 325°F just prior to startup.

The reactor was brought to criticality at 0440, July 13. Rod positions were 49.5 in. out. The scram set-points on fuel-channel exit temperatures were set down to 800°F. Temperatures were gradually increased until at 1300 reactor operating conditions shown in Table III-3 were noted.

No significant activity was noted in the high bay area. At 1330 it was observed that the moderator delta T followed a rise in the sodium outlet temperature. The moderator temperature did not respond properly to an increase in sodium flow. It appeared that very little sodium was leaking across the grid plate for moderator coolant.

TABLE III-3

REACTOR OPERATING CONDITIONS, RUN 14

(1300 JULY 13, 1959)

Power (Mw)	2.4
Inlet Temperature (°F)	473
Outlet Temperature (°F)	542
Flow (gpm)	803
Fuel Channel Exit Temperature (°F)	
Minimum	510
Maximum	750
Moderator Coolant Temperature (°F)	605

3. Reactor Excursion

At 1728, the reactor power was at about 1.6 Mw, and a planned increase was started so as to be able to deliver heat to Southern California Edison electrical substation. After the start, the power level persisted in rising somewhat faster than expected even though control rods were being slowly inserted in an attempt to hold it back. By 1807, the power had increased to about 4.2 Mw. At this time a negative period of about 45 sec was observed and the power fell to about 2.4 Mw in about 3 min. Control rod withdrawal was started and the reactor restored to criticality at 1811. Control rod withdrawal was continued and the power slowly rose to 3.0 Mw by 1821. At this time the power started to increase more rapidly, so control rod insertion was started. In spite of this rod insertion the rate of power rise continued to increase. Three positive period transients indicating about a 50-sec period were observed at about 1824, and at 1825 a 7-1/2 sec period was indicated. At this time the reactor power was rising rapidly; so the reactor was scrammed manually by the operator. The peak power indicated on the recorder was about 24 Mw. An analysis of this sequence of events is given in section IV-D.

A setback was not initiated automatically, as it should have been at a 10-sec period. The setback is actuated mechanically by means of a cam in the period recorder. A notch in the cam trips a switch which sets back the reactor by driving the control rods into the reactor. The setback was set to operate when the period reached 10 sec. During the excursion the period reached 7-1/2 sec without tripping the setback switch, and the power level reached about 24 Mw before the reactor was manually scrammed. Examination of the setback mechanism subsequently showed that it would operate satisfactorily only if the period decreased at a fairly slow rate. If it decreased quickly, the switch would fail to operate. The cam was modified and tested to operate at all rates of change of the reactor period.

During the power excursion, the maximum temperature recorded was the fuel-channel outlet temperature of 755°F on channel 10. Moderator coolant temperature reached a maximum of 620°F as indicated by a corner channel probe thermocouple. The two metal temperatures recorded on an instrument in the control room were less than 755°F.

Recovery from the scram was made cautiously. Criticality was attained at 1955. Approximately 2-1/2 hr after the scram, reactor power reached about 2.0 Mw. Rod positions were about 52 in. out, as compared with about 49.5 in. prior to the scram. This difference was, at the time, attributed to the xenon effect following the scram; particularly in view of the fact that by 0200, July 14, the rods had returned to a position of 50.5 in. at a power level of about 4.0 Mw.

It was decided that the power excursion had not affected the reactor adversely. The spread on the fuel-channel exit temperatures and the moderator delta T had not increased following the scram. As a result, reactor power was slowly increased to a maximum of 4.0 Mw at 0700, July 14. Operation at this level continued until 1300.

At 0900, on July 14, high bay activity increased to 14,000 cpm on the air monitor. By 1100, the source of activity had been localized to channels 29 and 50 in the core loading face. Some activity was also detected at several locations around the Cerrobend seal.

Seal rings were placed on channels 29 and 50, and the holes were taped over. This action reduced the high bay airborne contamination level by a factor of 4. By 1400, high bay activity had decreased sufficiently so that the area was opened for unrestricted access.

At 1300, a scram was caused by a short circuit introduced into the demand circuit for the main primary pump. This occurred while making the electrical connections for a test in which primary flow was to be oscillated. Recovery was made rapidly. The reactor was critical at 1311 at an average rod position of 50.4 in. (The differential reactivity worth of the rods in this position is approximately 0.1%/in.) Rod positions prior to the scram, at a power level of about 3.7 Mw, were at an average of 51.1 in. At 1500, after complete recovery from the scram, rod positions were an average of 51.9 in. out. Reactor power at this time was about 3.5 Mw. Operation continued at a power level of about 3 Mw with a reactor outlet temperature of 600°F. Maximum fuel-channel exit temperature was about 735°F (core channel 10). Sodium flowrate was 900 gpm.

4. Pressure Effect on Reactivity

It was decided to pressurize and vent the reactor atmosphere once in order to reduce the radioactivity level caused by the xenon in the cover gas. At 0550, July 15, the reactor pressure was reduced from 1.8 psig to 0.6 psig, repressurized to 3.0 psig, and then reduced to 1 psig. Upon venting core pressure down, reactivity increased, causing the rod to drive in on automatic control; when core pressure was increased, reactivity decreased; hence rod was pulled automatically. The magnitude of the reactivity change indicated by rod motion was of the order of 0.01%, but the observations are not adequate to permit a reliable evaluation. On August 26, 1958, a similar qualitative pressure effect had been observed, but this occurrence was not recalled at the time. A pressure effect is not normal, and it is believed that the presence of helium gas in the sodium may account for the effect noted in 1958. Operation was continued at a power level of approximately 3 Mw and 600°F reactor outlet temperature.

5. Operation on Airblast Heat Exchanger

During the morning of July 15, a review of fuel-channel exit-temperature spread indicated that it would be pointless to continue trying to get the Edison turbine generator "on the line," since maximum power level attainable would probably be less than 4 Mw. It was decided to shut the reactor down, drain the Edison lines, fill the airblast heat exchanger, and operate at higher reactor inlet temperatures than are possible while circulating through the steam generator. Reactor power was slowly reduced and the reactor was shut down at 1148, July 15. During the shutdown, the following activities were completed:

a) The Edison loop was drained.

b) The main airblast heat exchanger was filled.

c) The reactor sodium level coil was reinstalled in core channel 7. (New quad rings were included.)

d) A complete helium leak check was conducted on the core loading face. As a result, new quad rings were installed on the control-rod thimble in core channel 62.

On July 16, at 0704, the reactor again achieved criticality, this time at an average rod position of 57 in. out, indicating that there had been a substantial loss in reactivity since the beginning of the run. Careful examination of the history of shim-rod positions was made which indicated that a reactivity loss of about 1.2% had occurred gradually during the four days since the start of run 14. Reactor outlet temperature on July 16 was 360°F. Intermittent operation continued at low power (less than 2 Mw) until July 20.

On July 18 at 1100, the main primary flow rate was varied over the range of 400 to 1200 gpm. No effect on reactivity was observed. At 2148 on July 18, the MG set which supplies stabilized power for the vital bus (instrument power) failed. Operation was resumed using the unstabilized Edison power. The MG set was not repaired until July 21.

During the next several days measurements of the primary sodium plugging temperature were made. The results are given in Table III-4.

On July 19 and 20, some additional tests were made in an attempt to evaluate the effect of core cover-gas pressure on reactivity. The results showed an effect, but the data do not permit a reliable evaluation of its magnitude. Varying the sodium level in the reactor may have also produced a very small effect on reactivity.

On July 19, the reactor outlet temperature was gradually raised from 360 to 540°F while the power was kept below 1 Mw. It was desired to operate

TABLE III-4

PRIMARY SODIUM PLUGGING TEMPERATURES, RUN 14

Date	Time	Sodium Temperature (°F)	Plugging Temperature (°F)
July 18	1130	400	350
July 18	1520	400	335
July 19	1000	475	390
July 19	1345	530	395
July 21	1330	575	520
July 25	0035	510	455
July 26	0700	510	350

for a while at increased temperature to see if this would improve the reactor operating conditions. This is related to the experience on run 8. It was not possible to wash fuel elements because the wash cell had not been available since June 4.

On July 20 at 0900, reactor power was increased slightly in order to begin raising loop temperature to 700°F. On July 20 at 2305, reactor cold-leg temperature reached 700°F. Reactor power level at this point was about 2.5 Mw. Sodium flow rate was 850 gpm. Maximum recorded fuel-channel exit temperature was 865°F at a reactor outlet temperature of 740°F.

On July 21 at 0210, a scram was caused by a fast period indication. It should be recalled that at this point the nuclear instrumentation was still running on "normal" Edison power. The reactor was critical at 0225 at an average rod position of 49 in. Operation continued at about 2.5 Mw.

At 0645, radioactivity in the reactor began building up, as indicated on the continuous air monitors. This buildup continued until at 1000 the two air monitors were reading 15,000 and 18,000 c/m. At 1400, the results of a high bay air sample showed that the high bay activity level was 2×10^{-9} $\mu c/cm^3$.

A reactor scram was initiated manually at 0945, July 21 when flow was lost in the main secondary loop. The loss-of-flow scram was caused by a low sodium level in the secondary expansion tank which resulted from a faulty level coil on the main secondary expansion tank. After the scram, fuel-channel exit temperatures dropped. When the main secondary loop was restored to service, temperature swings were noted in the reactor cold leg. At 1130, the reactor was again critical at an average rod position of 54 in. The spread in the fuel-channel exit temperatures was still present, but safety limits were not being exceeded. Prior to restarting the reactor, the vital bus was restored to service.

6. Fuel Temperatures

During the day of July 22, the fuel temperature recorder on the element in core channel 55 showed fluctuating temperatures in the 1100 to 1200°F range. This cluster was composed of various experimental fuels. The temperatures of six fuel slugs in three experimental elements were being recorded on a point recording instrument located in the high bay area. However, fuel temperatures being recorded on instruments located in the control room were very much lower. Due to lack of confidence in the reliability of the recorder on the experimental fuels, no attempt was made to operate in such a manner as to reduce the temperatures recorded by the experimental fuels instrument. After the end of this run, a calibration of this instrument was made and it was found to be in good operating condition. This instrument was being repaired during the period July 12 to July 15, so no record from its exists during the power excursion that occurred on July 13.

During the interval between 1400 on July 20 to 1300 on July 21, the instrument in the high bay area which records the temperatures in the experimental fuel elements was not completely operative. At some time during this period the maximum temperature recorded was 1350°F. An examination of the record on this experimental instrument (Figure IV-A-5), made after the end of the run, shows a maximum temperature of 1465°F recorded at 1300 on July 23.

Operation continued at power levels up to 4.5 Mw, with sodium flow rates up to 1500 gpm, and reactor outlet temperatures up to 790°F.

7. Reactor Shutdown

On July 23, it was decided to shut the reactor down, in view of the previously reported high fuel temperature for the element in channel 55 and since the fuel-channel exit-temperature spread was not improving noticeably. Shutdown was set for 1700, July 24. Until 0900, July 24, reactor outlet temperature was kept between 700 and 800°F. A few fuel-channel exit temperatures reached the 900 to 1000°F range. Most of the fuel-channel exit temperatures were below 900°F. At 0950, July 23, a reactor scram was caused by a fast period indication. It was believed that this indication was due to an electrical transient. The reactor was critical again at 1015.

Between 0000 and 0800, on July 24, while jiggling elements in an attempt to dislodge foreign material and hence lower the fuel-channel outlet temperatures, it was noted that the elements in core channels 10, 12, 35, and 76 were stuck in place. On the evening of July 22 when a similar operation had been performed, the element in core channel 10 was free.

A scram was caused by a fast period indication at 1250 on July 24.* The reactor was critical again at 1314. Accidental loss of auxiliary primary flow caused a reactor scram at 1540. The reactor was critical again at 1556.

The reactor outlet temperature was gradually reduced to 510°F, at which point the primary cold trap was put back in service at 0025 on July 25. During this last portion of the power run, rod positions were an average of 54 in. out. At 0035, the primary plugging temperature was 455°F at a sodium temperature of 510°F. Cold trapping continued, and the primary plugging temperature reached 350°F at 0700 on July 26.

On July 26, it was noted that the elements in channels 12 and 35 were no longer stuck. The element in channel 76 was somewhat freer than before, while the element in channel 10 was still stuck.

---

*It should be noted that the reactor has a history of spurious scrams due to apparent period transients. This is one reason that the period scram is normally bypassed during power operation above 1%. Many so-called period scrams have been traced to voltage and frequency instability in the power supplied to the log N amplifiers. As a result, a new system involving a small motor-generator set, fly-wheel stabilized, has been designed to remedy this difficulty.

On July 26 at 1120, the reactor was shut down after a total exposure of about 16 Mwd on run 14. The shutdown was made to examine each fuel element which had been running hot with the television camera and to try to clear any obstructions in channels. The first damaged fuel element was observed at 1915 on July 26.

H. EVENTS FOLLOWING THE END OF RUN 14

1. Fuel Examination

The first fuel element that was removed from the core was visually inspected and found to be apparently in good condition. The next element to be removed was from core channel 25. This fuel element appeared, upon inspection by means of the television camera, to have a piece of cladding missing. It was stored in storage cell 56. The third fuel element to be removed was from core channel 31. This element, upon visual examination, was found to be parted. It was removed from the reactor and placed in storage cell 35.

Fuel elements which were found by visual inspection to be intact were either reinserted into the reactor or removed to the fuel storage facility. By August 2, 6 parted fuel elements had been removed to the fuel storage facility. On August 2, a parted fuel element from core channel 12 became lodged in the fuel handling cask. Attempts to dislodge it were not successful, so it was necessary to suspend the operations of examining the fuel. This left 19 fuel elements in the reactor core. Of these, 10 had been examined visually by means of the television camera, and 9 had not yet been examined. The 10 which had been examined were apparently intact. The 9 which had not been observed all ran hot during run 14 and hence had a good probability of being damaged.

2. Fuel Handling Cask Operations

As mentioned previously, the fuel element from core channel 12 lodged in the fuel handling cask and all attempts to dislodge it were unsuccessful.

Twice before portions of broken fuel elements had stuck in this fuel handling cask and were dislodged only after considerable effort. A considerable amount of radiation exposure and contamination in the high bay area was resulting from the attempts to dislodge the jammed fuel element. When it became apparent the operation would be quite time consuming, all efforts were directed to modifying the moderator handling cask for fuel removal. The moderator handling cask had been under development for a considerable length of time, and its fabrication and assembly were nearly complete. A careful study of its construction and operation was made in order to ensure that parted fuel clusters could be handled successfully without danger in jamming. Careful consideration was given to the hazards involved in moving broken fuel elements. The completion of the assembly and testing of the new fuel handling cask took several weeks. Very thorough operational tests were performed with dummy fuel clusters in order to ensure proper operation of the device. On September 22, the removal of a corner-channel dummy element from the core was started. This was in preparation of removing the remaining fuel elements from the core. The removal of the last 19 fuel elements from the core was started on October 8. All but 2, which were found to be stuck, were removed by October 19. The present status of the fuel elements which were in the reactor during run 14 is shown in Tables III-5, III-6, and III-7.

3. Fuel Canning

The fuel that had been removed from the reactor prior to August 2 had been transferred to the storage facility in the SRE building. The remaining fuel removed from the reactor was canned prior to storage to minimize contamination of the SRE storage facility and difficulties encountered when a parted fuel cluster was to be placed in a storage cell. Moreover, this canning would have been eventually necessary in any case so that the fuel could be transferred to the Component Development Hot Cell where a more complete examination could be made. The SRE hot cell facilities have an air atmosphere and are not suitable for complete examination of unwashed or parted fuel clusters.

TABLE III-5

DAMAGED FUEL ELEMENTS

R-10	At 2200, July 27, 1959, it was found that the fuel cluster was broken in two with approximately two-thirds of the fuel cluster remaining in the fuel channel. Shield plug and broken section of fuel cluster were stored in storage cell 69.
R-12	At 1745, August 2, 1959, it was found that the fuel cluster was broken in two with approximately two-thirds of the fuel cluster remaining in the fuel channel. Shield plug and broken section of fuel cluster are contained within the fuel transfer cask awaiting transfer to a storage cell.
R-21	At 2245, October 10, 1959, an unsuccessful attempt was made to remove the fuel cluster from R-21. The hoist cable power was tripped out because of overload $\simeq$ 800 lb. The fuel cluster was finally withdrawn from the core at 2330. Observations made in the SRE hot cell indicated that the lower third of the fuel cluster had broken off and had probably remained in core channel R-21. The portion of cluster removed has been canned and is being stored.
R-23	At 1413, October 11, 1959, the fuel element from R-23 was removed from the core. It was noted during removal operations that the element stuck momentarily after 4 ft of upward travel. The element was broken, with the lower third believed to be still in R-23. The portion of cluster removed has been canned and is being stored.
R-24	Attempts to remove the fuel cluster from R-24 had failed. This cluster, with its moderator can, was raised and blocked-up approximately 1 in. in an attempt to free the fuel cluster from the moderator can. No change was noted.
R-25	At 1915, July 26, 1959, the element was being viewed by using the portable television camera when it was noted that the cladding appeared to be split open on one of the fuel rods. The element was lowered back in the core, rotated 180°, and viewed again showing an additional ruptured fuel rod. Element was stored in storage cell 56.
R-31	At 0200, July 27, 1959, it was found that the fuel cluster was broken in two with approximately one-half of the fuel remaining in the fuel channel. Shield plug and broken section of fuel cluster were stored in storage cell 35.
R-35	At 1700, July 27, 1959, it was found that the fuel cluster was broken in two with approximately one-half of the fuel remaining in the fuel channel. Shield plug and broken section of fuel cluster were stored in storage cell 68.
R-43	At 1905, October 15, 1959, the fuel element from R-43 was removed from the core. The fuel cluster was broken in half. The portion of cluster removed has been canned and is being stored.
R-55	At 1900, July 28, 1959, the experimental fuel element was found to be broken in two with approximately one-half of the cluster remaining in the fuel channel. Broken section of fuel cluster was examined and photographed in the hot cell, then the remains of the cluster were removed from the shield plug section and placed in a container and left in the hot cell. The shield plug was placed in storage cell 73.
R-68	At 2300, August 1, 1959, it was found that the fuel cluster was broken in two with approximately two-thirds of the fuel cluster remaining in the fuel channel. Shield plug and broken section of fuel cluster were stored in storage cell 72.
R-69	At 2230, October 14, 1959, the fuel element from R-69 was withdrawn from the reactor. The cluster was broken in half. The portion of cluster removed has been canned and is being stored.
R-76	Attempts to remove the fuel cluster from R-76 have failed. In-core observations of the cluster, during removal attempts, indicate that the moderator can containing the cluster lifts as the element is raised. This cluster, with its moderator can, was raised and blocked-up approximately 1 in. in an attempt to free the fuel cluster from the moderator can. No change was noted.

TABLE III-6

FUEL ELEMENT STATUS

Core Channel	In During Stripping	Ran Hot During Run 14	Damaged	Burnup (Mwd/T)
4	yes	yes	no	581.0
9	no	yes	no	440.6
10	yes	yes	yes	645.2
11	yes	no	no	793.5
12	yes	yes	yes	747.4
19	yes	no	*	462.7
20	yes	no	no	826.1
21	yes	†	yes	852.1
22	yes	no	no	810.3
23	yes	†	yes	801.1
24	yes	†	§	303.5
25	yes	yes	yes	707.9
31	yes	yes	yes	745.0
32	yes	no	no	447.7
33	no	no	no	939.5
34	yes	no	no	1117.0
35	yes	yes	yes	541.9
36	yes	no	no	472.2
41	yes	no	no	733.9
42	yes	no	no	963.9
43	yes	†	yes	1144.4
44	yes	no	no	463.3
45	no	no	no	876.4
46	no	no	no	843.5
47	no	yes	no	271.3
53	no	no	no	516.0
54	no	no	no	759.5
55	yes	yes	yes	732.8
56	no	yes	no	593.2
57	yes	no	no	100.8
58	no	yes	no	867.7
65	no	no	no	620.7
66	no	no	no	790.3
67	no	no	no	511.7
68	no	yes	yes	870.1
69	no	yes	yes	793.5
70	no	no	no	323.7
71	no	no	no	671.1
73	no	no	no	739.5
74	yes	no	no	833.4
75	yes	no	no	812.1
76	yes	yes	§	517.8
80	yes	no	no	635.8

*Questionable
†Fluctuating
§Remains in Core

TABLE III-7

SUMMARY OF FUEL ELEMENT STATUS

Listed below are the number of elements that correspond to specific combinations of: (a) in during stripping; (b) ran hot during run 14; and (c) damaged

Number of Elements	In During Stripping	Ran Hot During Run 14	Damaged
6	yes	yes	yes
1	yes	yes	no
13	yes	no	no
11	no	no	no
2	no	yes	yes
4	no	yes	no
1	yes	no	*
3	yes	†	yes
1	yes	†	§
1	yes	yes	§

* Questionable
† Fluctuating
§ Remains in Core

A considerable amount of study was required to prepare the SRE hot cell for the fuel cluster canning operation. Special equipment was designed and fabricated to provide an inert atmosphere around the element in order to minimize the risk of a fire in the hot cell during the canning operation. The new equipment included a special tube with a transparent section for a viewing window and sleeve which could be raised when it was desired to disconnect the hanger rod from the fuel element. The container for canning a fuel element is placed in this tube and the assembly is then filled with argon gas. The fuel element is then lowered from the fuel handling cask into this tube while the inert atmosphere is maintained. After canning the fuel element, it is transferred to storage. A complete trial canning run was made on September 29, using a corner-channel dummy.

4. The Sodium System

Circulation has been continuously maintained in all four sodium systems since the end of run 14. Temperatures have been held at about 400°F by means of piping heaters. Cold trapping has been continuous on the primary sodium system.

On September 23, the surface of the sodium was viewed through a window which had been installed in the top shield of the reactor. The surface of the sodium appeared to be clean and reflective.

5. Tests and Observations

A considerable number of tests and examinations has been made upon materials removed from the reactor. Fuel and cladding have been examined in the hot cell along with foreign material which has been found adhering to some of the fuel elements. Samples of the primary and secondary sodium have been analyzed, radiation levels measured, and the cover-gas radioactivity evaluated. Details of the results of these observations are presented in section IV.

6. Recovery Equipment

Apparatus needed for the removal of the broken fuel element pieces from the reactor core has been procured. Viewing windows and lights have been installed in the reactor. A borescope which will be used to examine the process tubes and the moderator cans was obtained. Several types of "fishing" tools have been designed and tested in a mockup.

7. Improved Fuel Washing Procedures

A detailed study of the fuel wash cycle had been under way since the incident following run 13. Tests are being performed to study the pressure surges in a wash cell while a fuel element is being washed. A redesign of the washing procedures and equipment is in progress so that fuel elements can be washed without damage.

8. Preparation of Charts and Data

Charts and tables have been prepared which are useful in describing the reactor history and operating conditions. Several of these are presented later in this report. Only those data believed to be pertinent to the fuel damage or needed to describe reactor operating conditions have been included. Charts are presented which show reactor power, temperatures, reactivity, etc., during runs 13 and 14. An analysis has been made of various phenomena that could influence reactivity. Reactor kinetics studies were made on the power excursion that occurred at 1825 on July 13. All available information on fuel and moderator temperatures was carefully examined. Records of radioactivity levels were studied. Information was obtained from the examination of fuel and other reactor components. Plans were formulated for the gathering of additional information and data. The details and results of these activities are given in section IV.

IV. DATA AND EVALUATION

A. FUEL TECHNOLOGY

1. Uranium Metallurgy

A condensed review of pertinent metallurgical aspects of uranium, uranium-rich alloys, and thorium will provide a clearer picture of the SRE fuel element damage. SRE Core I contained 35 standard 7-rod elements of unalloyed uranium, one 19-rod UO_2 element, 2 Th-7.6 wt % U elements, and 5 experimental elements containing uranium alloys, uranium, and Th-5.4 wt % U alloy. The uranium alloys included U-2 wt % Zr, U-1.5 wt % Mo, U-1.2 wt % Mo and contained the same percent enrichment as the standard fuel, 2.8 wt % U^{235}. Table IV-A-I indicates the physical property data of interest on both U and Th.

TABLE IV-A-I

PERTINENT PROPERTIES OF URANIUM AND THORIUM

	Melting Point (°F)	Crystal Structure	Phase Changes	Density (g/cm^3)	Thermal Cond. (cal/cm-sec-°C)
U	2060	α, orthorhombic β, tetragonal γ, cubic	α 75 to 1220°F β 1220 to 1420°F γ > 1420°F	19.0 to 18.3 18.3 to 18.1 18.1 to 18.0	0.063 0.086 0.095
Th	3090	<2550°F fcc >2550°F bcc	minor one at 2550°F	11.9 (20°C)	at 20°C, 0.090 at 1200°F, 0.11

The physical properties of uranium are directly affected by the method of fabrication and the impurities present (particularly carbon). The unalloyed uranium was fabricated by rolling in the high alpha phase followed by beta heat treatment to randomize the grain orientation. The SRE fuel was exceptionally pure material; C content was between 50 and 250 ppm. The average density at room temperature was 19.0 g/cm^3. Uranium has adequate strength properties in the alpha phase but decreases in ductility when both alpha and beta phases are present. The cubic gamma phase is relatively soft. The low alloy additions to uranium will markedly increase the strength in the alpha phase and produce a sluggish alpha-beta phase transformation. However, the effects on melting point, phase change temperatures, density, and thermal conductivity will be relatively minor. In the U-2 wt % Zr alloy, there is a change in phase from a simple system to a complex system at a slightly higher temperature than unalloyed uranium. The effects of irradiation on these properties are not well known under SRE temperature conditions, but are believed to be minor. The principal effect observed to date on the properties of unalloyed uranium is the increase in strength and hardness at the lower alpha phase temperatures.[3]

Since thorium has a cubic structure, it is a relatively simple metal and has normal metallic properties. It has lower strength than uranium at low temperatures but this reverses at higher temperatures. Capsule irradiations have indicated excellent stability at 1% burnup at 1200°F. Relatively little is known of the effects of radiation on the properties of Th-U alloy at SRE conditions.

The radiation distortion in all metal fuels is a major problem. It is of two types: (a) anisotropic growth with no density change; and (b) swelling, accompanied by a density decrease. Uranium, due to its anisotropy, has been shown to grow extensively at low alpha phase temperatures. The usual temperature associated with growth is less than 840°F and swelling generally occurs above 840°F. This transition between the two types of deformation does not occur at a definite temperature and is further complicated by additional factors, such as thermal cycling and burnup rate. Due to the high surface temperature of SRE fuels, swelling appears to be the main problem. The SRE fuel has been destructively examined at various exposure intervals, and some marked dimensional changes have been observed (see Figure IV-A-1).

2. Fuel-Clad Compatibility

Both uranium and thorium form low melting alloys with all major constituents in the stainless-steel clad. Table IV-A-2 shows the eutectic melting points of uranium and thorium alloys with major chemical elements in 304 stainless steel.

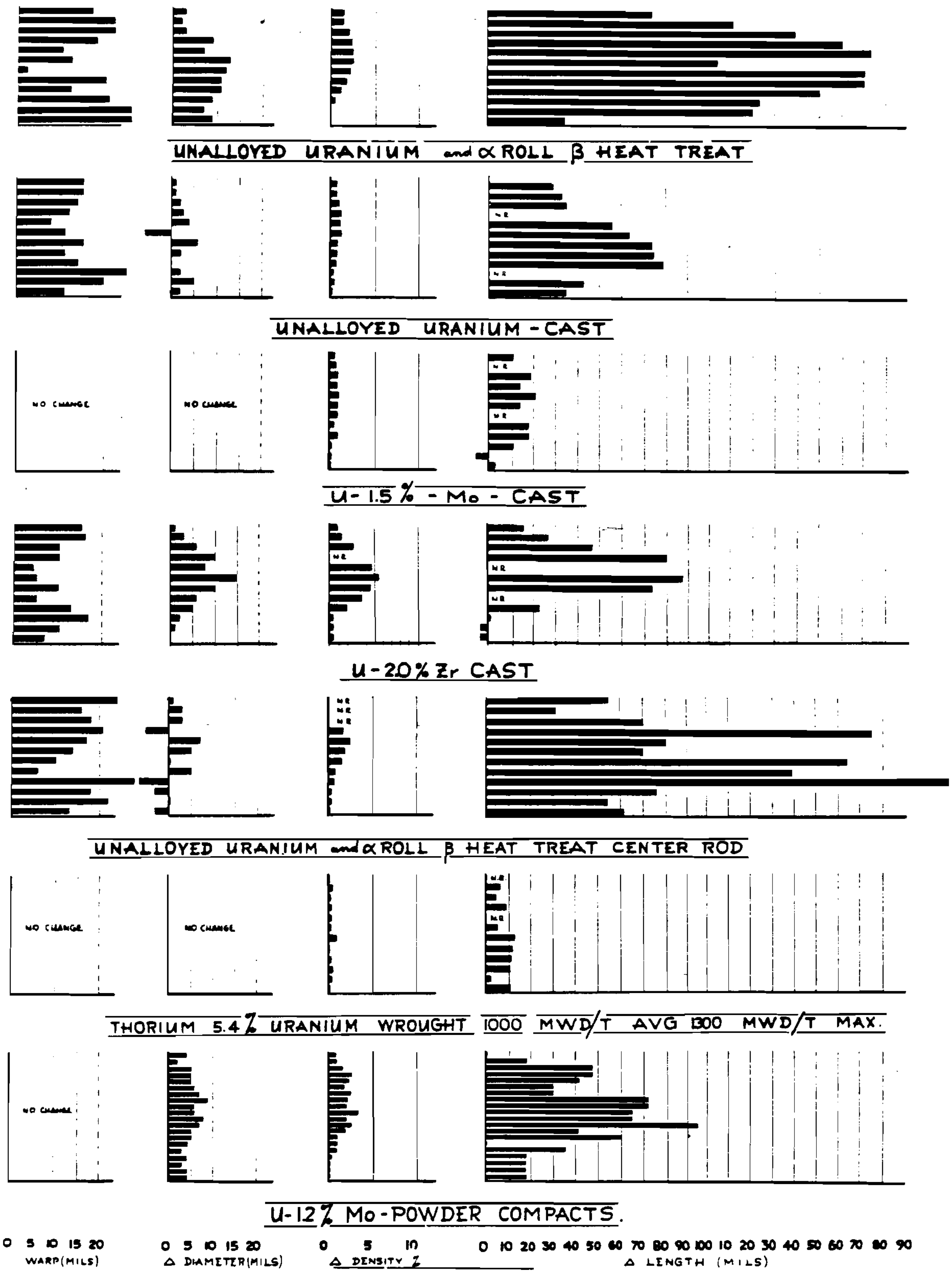

Figure IV-A-1 Radiation Effects on SRE Fuel
(burnup 850 Mwd/T maximum)

TABLE IV-A-2

EUTECTIC COMPOSITIONS OF URANIUM AND THORIUM

U base		Th base	
Composition	Melting Point (°F)	Composition	Melting Point (°F)
U-5 wt % Cr	1578	Th-25 wt % Cr	2265
U-12 wt % Ni	1365	Th-7 wt % Ni	1830
		Th-30 wt % Ni	1875
U-12 wt % Fe	1340	Th-17 wt % Fe	1580

At the present time there are only a few out-of-reactor data on the rate of formation of the most significant eutectic, uranium-iron. During the solid-solid reaction between uranium and iron, all alloy compositions from pure iron to pure uranium will be found. The effect of sodium or radiation on this reaction is not well known. The solubility of both uranium and iron in sodium is low at the eutectic temperature. British results[4] show that a 0.020-in. stainless-steel can is penetrated in one minute at 1560 to 1740°F. More recent tests at Atomics International[5] show that uranium reacts with stainless steel within seconds at 2010°F, within minutes at 1790°F, and within a relatively few hours at 1560°F (nonreproducible results to date). Tests are continuing in the 1400 to 1560°F range for more precise reaction rates.

Important factors which affect the reaction rate are the surface oxide layer on the uranium and the degree of contact between the two materials. All fuel slugs in SRE have been centerless ground to 250-rms surface finish. Furthermore, all fuel slugs were electropolished before assembly and thus contained at most a relatively thin oxide film. The exact oxygen content of the NaK bond is not known but was believed to be low. It has been observed by numerous experimenters that the uranium will getter oxygen from the NaK. It is also known that NaK wets uranium oxide more easily than it does uranium metal. Removal of irradiated slugs from destructive examinations after run 8 has shown that all slugs were relatively clean. Thus, it is reasonable to assume that a thin oxide film was present on the uranium and was not considered undesirable.

The slugs are loose in their jackets (nominal 10-mil annulus) and consequently probably contact the tube at all times at the side or ends. The fuel and clad make additional contact either through fuel diameter increases or fuel warp. Due to the radiation-induced surface wrinkling of the U, the contact between the SRE uranium fuel slugs and the stainless-steel clad would be point-to-point contact. The unalloyed uranium and the U-2 wt % Zr alloy slugs had been observed to grow in diameter sufficiently to stretch the clad slightly in the former case and to at least contact the clad in the latter case. The surface wrinkling of the unalloyed uranium examined at the 850-Mwd/T exposure level was sufficiently severe to estimate only point-to-point contact by this mechanism. The degree of diameter increase, and thus intimacy of contact, is directly proportional to burnup. The measured diameter increases were of sufficient magnitude to contact the clad and occurred at an exposure of 850 to 950 Mwd/T. The fuel slug which had this diameter increase was located near the midplane of the reactor, or position of maximum exposure. The fuel slugs also exhibited sufficient warp at much lower exposures (100 Mwd/T) to cause only point contact between fuel and clad at all slug positions in the 6-ft rod length (see Figure IV-A-1). The warp measurements indicated a saturation of this effect at low exposures and in no case was it sufficient to cause stretching of the clad.

3. Fuel Operating Limits

Design conditions for SRE fuel operation are based on two primary factors: (a) temperature; and (b) burnup. These are not independent variables and are influenced by numerous additional factors which are only now becoming known. It is important to note that the fuel elements in the SRE are operating under untried conditions. There is little data from other reactors that provide sufficient information for even reasonable extrapolation. A few capsules irradiated by AI at MTR and initial data from other sites indicated swelling was a major problem. Knowing this, the operational limits of the fuel were approached with caution. Both destructive and nondestructive examinations were made at intervals during the reactor operating history to permit sound extrapolations for true core life. These radiation stability data are further supported by measured fuel temperatures at various positions in the core. All of the experimental elements and six standard elements were temperature monitored (see Figure IV-A-2). Each of these elements contained two thermocouples. All of the thermocouples on four of the standard elements were inoperative by the start of run 14.

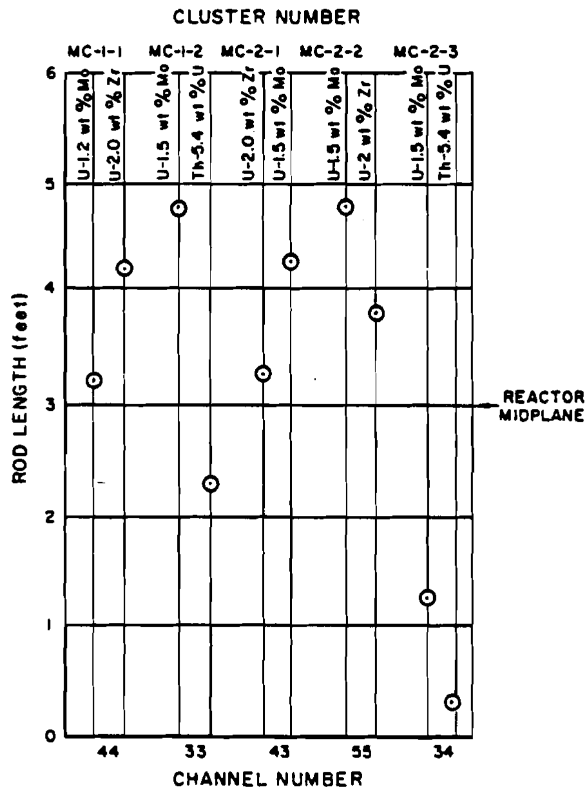

Figure IV-A-2. Thermocouple Distribution in SRE Fuels

The initial design limits were 2500-Mwd/T burnup at 1200°F. This temperature corresponds to the alpha-to-beta phase transformation in uranium. It was conservatively selected due to lack of data and from out-of-reactor results of thermal-cycling tests in this temperature range. The burnup limit was purely an estimate. Hot channel factors were included in the 1200°F limit. Fortunately, under normal operation, the hot channel factors appeared conservative. The operating peak fuel temperature at full power was calculated to be in the range of 1065°F. The peak temperature initially occurs slightly