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THE FINAL REPORT ON THE SNAP 2 EXPERIMENTAL REACTOR (SER) OPERATION AND TEST PROGRAM

AEC Research and Development Report

RELEVANT FOR ANNOUNCEMENT
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THE FINAL REPORT ON THE
SNAP 2 EXPERIMENTAL REACTOR (SER)
OPERATION AND TEST PROGRAM

By
STAFF OF
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EVALUATION GROUP

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ABSTRACT

This report presents the operating history of the SNAP 2 Experimental Reactor (SER) from initial fuel loading until final disassembly and removal of fuel. The results of the testing program are also presented. The total energy generated during the life of the reactor was 224,650 kwh. This is equivalent to approximately 6 months of full-power operation. The methods used to obtain the reactor parameters are also described. The experimental data obtained were generally in excellent agreement with calculated values. The principal comparisons are tabulated (see Page vii).

PRINCIPAL COMPARISONS¹

<u>Parameter</u>	<u>Calculated</u>	<u>Measured</u>
Critical mass (kg)	2.9	3.0
Excess reactivity (\$)	7.77	9.78
Temperature defect (\$) between 75 °F and 1100 °F core average temperature	3.10	3.42
Prompt temperature coefficient (¢/F °)	-0.2	~-0.05
Overall temperature coefficient at 1100 °F core average temperature (¢/F °)	-0.27	-0.35
Xenon defect (\$) (equilibrium at 50 kw)	0.36	0.39
Reactivity loss rate (¢/day) (at 50 kw and 1100 °F core average temperature)	0.41	0.79
Fuel element worth in outer ring (\$)	1.31	1.40
Maximum fuel temperature (°F) (at 50 kw)	1250	1270
Control drum worth (\$) (at 0 to 180 ° rotation, with one drum)	2.50	3.92

I. INTRODUCTION

Atomics International is actively engaged in the design and development of compact, lightweight, nuclear, power systems to provide auxiliary power requirements for earth satellites and space applications. The SNAP 2 Experimental Reactor (SER) was the first of a series of nuclear reactors in this program. Its primary purpose was to test the reactor core under design power and temperature conditions.

This reactor concept combines the advantages of the homogeneous thermal reactor with those of a heterogeneous reactor; i. e., the clad fuel-moderator rods prevent fission products from contaminating the coolant, and the homogeneous fuel-moderator tends to increase the thermal utilization factor, f , and thus increase the value of k , the neutron multiplication factor. The fuel-moderator was composed of a hydride uranium-zirconium alloy enriched in U^{235} , which was formed into rods and canned in thin-walled, stainless steel tubes. Before final assembly, the inside surfaces of the tubing were coated with a vitreous enamel to reduce hydrogen diffusion. Heat was removed from the core by a eutectic NaK (78 wt % potassium) coolant which flowed upward between the fuel elements. The resultant core was compact, lightweight, and capable of producing the high-temperature conditions required to operate power-conversion units in space. The feasibility of this concept was indicated by results of experiments with the SNAP 2 Critical Assembly. The SER has proved the practicality of the design under full-power operating conditions of 50 kw and at the designed coolant-outlet temperature of 1200°F. This reactor was located in the Burro Flats area of the Santa Susana Field Test Site of Atomics International, approximately 30 miles northwest of Los Angeles.

Many of the components and systems used in the SER contained unique features or were operated under unusual environmental conditions; therefore, a comprehensive testing program was developed to precede actual operation. This testing program included:

- 1) A critical assembly to check the methods of calculation used for nuclear design and to obtain experimentally the control and safety element worths, excess reactivity, neutron lifetime, and other nuclear and physical parameters of the SER configuration.

- 2) Use of the coolant loop mockup to check the SER heat-removal system.
- 3) Performance tests of the safety and control elements with their related instrumentation under expected operating conditions.
- 4) A program to investigate the radiation stability of the fuel-moderator material.
- 5) A program to investigate the physical properties of the fuel-moderator and cladding materials.

Efforts in the areas covered by Items 4 and 5 were continued in parallel with the SER operation.

Table I presents a chronological record of the SER from initial construction to disassembly. Section II of this report provides a detailed description of the design features and performance characteristics of the SER reactor system. Section III is a discussion of the instrumentation and control systems. The operational history of the SER is covered in Section IV. A detailed description and analysis of reactor tests is covered in Section V. Section VI contains the conclusions and appraisal of the SER from the standpoint of operating experience.

TABLE I
SER CHRONOLOGY

Item	Date
Construction started	June 1958
Facility completed	March 1959
Dry critical	September 19, 1959
Wet critical	October 20, 1959
First power operation	November 5, 1959
First operation at design power and temperature	November 9, 1959
Reactor completed 1000 hr continuously at design temperature and power	April 23, 1960
Final shutdown	November 19, 1960
Disassembly initiated	May 19, 1961
Disassembly completed	June 13, 1961

II. SER PHYSICAL DESCRIPTION²

The heat-transfer system consisted of the following: a radioactive primary loop, a nonradioactive secondary loop, an intermediate heat exchanger, and an airblast heat exchanger. Heat was removed from the core by primary NaK coolant flow, transferred to the secondary NaK coolant by means of the intermediate heat exchanger, and finally dissipated to the atmosphere through the airblast heat exchanger.

The reactor core was located below ground level and within a containment vessel. Biological shielding at the top of the containment vessel protected personnel during reactor operation.

The radioactive primary system and the intermediate heat exchanger were located in a shielded vault with a nitrogen atmosphere. The secondary NaK system and airblast heat exchanger were exposed to air atmosphere.

A. FUEL-MODERATOR ELEMENTS

The fuel-moderator rods were made from a zirconium hydride uranium alloy — 93.12 wt % $\text{ZrH}_{1.70}$ and 6.88 wt % uranium enriched to 93%. This material contained a minimum N_{H} of 6.7×10^{22} atoms/cc (compared to cold water with an N_{H} of 6.0×10^{22} atoms/cc). Analysis of the pressure, temperature, and composition characteristics of this alloy indicated that it could be considered as a mixture of zirconium hydride plus uranium metal with a density of 5.89 g/cc.

In considering the safety and hazards aspect of the fuel-moderator, the most significant property of this material is the dissociation pressure of the zirconium hydride as a function of temperature. Although zirconium hydride is fairly stable at moderate temperatures, it dissociates to hydrogen and zirconium at higher temperatures. The pressure-composition-temperature (P-C-T) studies on 93 wt % Zr, 7 wt % U alloy hydride; 90 wt % Zr, 10 wt % U alloy hydrides; and unalloyed zirconium hydride have revealed that the uranium content of the alloy does not influence the dissociation pressures. The uranium apparently precipitates from solution on the addition of hydrogen to the alloy and remains as an inert phase, insofar as the P-C-T properties are concerned. The dissociation pressures of all of the materials can be expressed, therefore,

as a function of the H/Zr atomic ratio.³ Because the experimental data from these tests are still classified, plots of log pressure vs H/Zr will not be shown.

From the data, it can be determined that the dissociation pressure at the peak core-operating temperature of approximately 1300°F will be about 320 mm of Hg or about 6 psia. One important result of this material instability is loss of hydrogen from the core by diffusion through the vitreous-coated cladding and the loss from the bare material in the event of cladding rupture. The former is an operational disadvantage, because it requires excess reactivity to compensate for the hydrogen loss during the lifetime of the reactor. The latter provides an ultimate reactor shutdown mechanism. Figure 1 presents the hydrogen escape rate and the reactivity loss rate from unclad zirconium hydride as a function of temperature. The loss in reactivity attributed to the loss of hydrogen by diffusion through the vitreous-coated cladding will be discussed further in Section V.

Each fuel-moderator element consisted of a solid rod, 0.975 in. in diameter by 10 in. long, with an 0.975 in. diameter by 1-1/2-in.-long beryllium slug at each end, canned in stainless steel tube 1 in. in OD with walls 10 mils thick. The tube was sealed at each end by welding in 1/2-in. stainless steel end caps. The inside surface of the stainless steel tubing was coated with a thin layer of boron-free vitreous enamel. The assembled fuel element is shown in Figure 2.

Sixty-one fuel elements were arranged on 1.015-in. centers in a triangular lattice to form a hexagonal core approximately 8 in. across the flats and 9 in. across the corners, as shown in Figure 3.

B. CORE REFLECTOR AND CONTROL ELEMENTS

The void between the hexagonal fuel lattice and the 9-in. ID core vessel was filled with six Be shims, as shown in Figure 4. The core vessel was surrounded by Be reflector sections approximately 3 in. thick and 13 in. long.

The beryllium reflector external to the core was composed of three sections:

- 1) Plates, 1/4 in. thick, stacked along three flat surfaces. These made up the removable sections for the three safety elements. The excess reactivity of the core was adjusted by varying the number of 1/4-in. plates.

Figure 1. Hydrogen Escape and Reactivity Loss Rates from Unclad Zirconium Hydride for the SER

Figure 2. Fuel-Moderator Element Assembly

Figure 3. Fuel Element Lattice

- 2) Partial circular cross-sections which could be rotated away from the core. These three pieces were the control drums.
- 3) A specially shaped piece that filled the interstices between the above sections and the reactor core vessel.

Figure 4. Radial Reflector Cross-Section

End reflection was achieved with the Be end plugs and the stainless-steel weld plugs together with the NaK coolant plenums and the grid plates. The reactor was controlled by rotating external portions of the radial reflector away from the core, thus changing the effectiveness of the reflector for maintaining neutron economy.

Each control drum was driven with a direct motor-drive geared to provide a maximum insertion rate of 2.5¢/sec. At the beginning of operation, each control drum had a total worth of about \$3.82 over the range from 0° to 180°.

A selsyn readout system provided visual observation of the position of each control drum. The control drum drive mechanisms and position indicators were located behind shielding, to protect them from the high neutron and gamma radiation. The control drums did not scram.

Each safety element was pivoted about a hinge below the core, and safety shutdown of the reactor was achieved by allowing the safety elements to fall by gravity away from the core. Each of the three independent safety elements was worth about \$5.40 in reactivity. The safety element drive mechanism and control drum drive mechanism are shown in Figure 5. Each safety element was moved into position by means of a rack-and-pinion system attached to the pull-up rod by a magnet. The safety elements were scrammed by interrupting power to the magnets. The reactivity insertion rate was limited to about 6¢/sec. The drive mechanism was equipped with appropriate limit switches for interlock and control usage. The safety elements fell from the "full-up" to the "scrammed" position in 220 msec after interrupting the magnet current. There was an additional time delay of approximately 20 msec in breaking the magnet current after a scram signal was received by the nuclear instrumentation. This increased the total scram time to 240 msec.

A Pu-Be neutron source was mounted on a bracket at the base of one of the control drums. The strength of the source was approximately 1.68×10^6 n/sec.

The fuel loading of the core was 3.0 kg of U^{235} . During the critical experiments, the excess reactivity of the cold reactor was adjusted to \$4.77 by the action of a microswitch backed up by a mechanical stop, which limited the inward rotation of the control drums. The location of this limiting arrangement was inaccessible during reactor operation. The excess reactivity requirements for the SER are summarized in Table II.

C. REACTOR CORE VESSEL

The core vessel was a cylinder 9-1/2 in. in diameter with a wall thickness of 3/32 in. The overall length was 16-3/8 in. The core vessel was penetrated at the bottom and the top by 1-1/4-in. coolant lines (the inlet line on the bottom and the outlet line on the side above the top grid plate).

Figure 5. Control Drum and Safety Element Drive Mechanism

TABLE II¹
EXCESS REACTIVITY REQUIREMENTS, \$

	Calculated*	Measured*
Xenon buildup	0.36	0.39
Temperature override (200 to 1100°F)	3.10	3.42
Fuel depletion, fission products and samarium buildup	0.60	0.96
Hydrogen loss	1.79†	2.81
Operating excess	0.18†	0.40
	<u>6.03</u>	<u>7.98</u>

*Calculated values are for 1-yr operation. Measured values are over life of core.

†Estimated.

The core vessel contained the upper and lower grid plates for positioning and holding the fuel-moderator assemblies and a "coolie" hat above the upper grid plate to control flow through the core. The grid plates also distributed the NaK coolant through the fuel rod interstices and around the reactor core. Figure 5 shows the grid plates in relation to the reactor core vessel.

D. REACTOR CONTAINMENT VESSEL

A carbon-steel (1/4-in. boiler plate) containment vessel (Figures 6 and 7) containing the reactor components and the necessary shielding was located below ground level. The vessel was 15 ft, 5-5/8 in. deep and reduced in three steps from 48 in. ID at the top to 38 in. ID at the bottom. Water circulated through cooling coils around the outside of the containment vessel removed the excess heat and prevented overheating of the surrounding concrete. Nitrogen at slightly above atmospheric pressure was used as a cover gas to exclude air from the vessel and thus minimize the possibility of NaK fires and hydrogen-oxygen reaction. In the event of a rupture of the core tank, the fission products would have been contained by the containment vessel.

The containment vessel was designed for a pressure of 305 psig and was hydrostatically tested at 456 psig in accordance with the ASME Boiler and Pressure Vessel Code for unfired pressure vessels containing lethal substances.

Figure 6. Reactor Installation

Figure 7. Containment Vessel in Place at Site

A calibrated rupture diaphragm (set to rupture at 150 psi) was connected to the primary coolant system and was located within the containment vessel (Figure 6). The diaphragm was installed to ensure that no rupture of the system would occur outside this vessel.

E. REACTOR SHIELDING

The reactor shielding consisted of three parts: the shutdown shield above the core, the biological shield which sealed off the containment vessel, and the concrete and earth around the containment vessel.

The shielding was designed for the radiation dose levels resulting from operation at 100 kw (twice the designed power level), as follows:

Accessible area above the heat exchanger	3.6 mr/hr
Accessible area above the core	0.8 mr/hr

This compares with the measured values at 50 kw of 1.3 mr/hr and 0.2 mr/hr, respectively.

The shutdown shield was in two parts — the ring shield body and a stepped inner plug which could be removed separately from the body to enable core removal. The maximum reading measured during reactor operation was 20 mr/hr, caused by streaming from the instrument thimbles.

F. PRIMARY NaK COOLANT SYSTEM

The bulk of the primary NaK coolant system was located within a shielded, sealed-off vault. The internal dimensions of the vault were 9-1/2 ft long, 8 ft wide, and 7 ft high. The location of the primary loop vault with respect to the reactor core, the equipment pit, and the secondary NaK coolant system is shown in Figures 8 and 9. A simplified flow diagram of the primary, secondary, and auxiliary NaK systems is shown in Figure 10.

1. Fill-and-Drain Tank

The primary fill-and-drain tank had a 24-gal capacity and was located in the equipment pit. Loading of the entire primary system with NaK was accomplished by pressurizing the tank with helium. A diffusion cold trap attached to the bottom of the tank was cooled by a refrigerated coil. The fill line from the tank contained a sintered metal filter (2 ft² filter area) with a filter opening size of 5 μ .

Figure 8. SER Building, Plan View

Figure 9. SER Building, Elevation View

Figure 10. NaK Coolant Flow Diagram

2. Primary Surge Tank

This tank was located at the high point in the primary system. A cover gas of helium was admitted through a line at the top of the tank. Samples of the cover gas could be withdrawn from the surge tank and analyzed for contaminants. Coolant level instrumentation was also provided within this tank.

The NaK coolant from the reactor entered at the side of the primary surge tank below the liquid-metal-free surface. The outlet for the NaK coolant was at the bottom of the tank.

3. Intermediate Heat Exchanger

The intermediate heat exchanger was a counterflow type rated at 100 kw. The internal construction is shown in Figure 11. Heat was exchanged from the primary radioactive NaK to the secondary nonradioactive NaK. Design inlet and outlet temperatures were:

	<u>Primary</u>	<u>Secondary</u>
Inlet temperature	1200°F	950°F
Outlet temperature	1000°F	1150°F

The pressure of the secondary system was approximately 1 psi greater than the pressure in the primary system. If, therefore, a leak had occurred in the intermediate heat exchanger, the secondary system would not have become contaminated with the radioactive primary NaK.

4. Electromagnetic Pump

The primary NaK outlet coolant flow from the intermediate heat exchanger entered the suction header of a polyphase, linear-induction, electromagnetic pump designed and constructed by Atomics International to supply 23 gpm, 1000°F NaK with a head of 20 psi. The flattened throat was positioned midway between the edges of a stack of laminations. There were no mechanical connections between the pump throat and the rest of the pump assembly. The pump throat external dimensions were approximately 24 in. by 4-1/2 in. by 0.325 in., with a clearance between the pump throat and the edges of the laminations of 0.037 in. The wall thickness of the throat was 1/16 in. The suction and discharge lines were nominal 1-1/4 in., schedule-40 pipe. The pump coils were capable of withstanding an ambient temperature of 400°F.

Figure 11. Intermediate Heat Exchanger

Pressure transmitters were installed in the suction and discharge lines of the electromagnetic pump to monitor its performance during reactor operation.

5. Primary Flow Meter

The discharge NaK flow from the electromagnetic pump passed through the primary flow meter. This was a permanent-magnet-type flow meter.

6. Plugging Indicator

A plugging indicator was installed in the primary system gallery to determine the relative oxygen content and/or presence of precipitated hydrides in the primary NaK coolant. The plugging indicator was installed across the primary electromagnetic pump. Details of construction of the plugging indicator are shown in Figure 12.

7. Valves and Valve Operators

All valves in the primary NaK coolant system which were located within the vault were of the bellows-seal type and were rated at 300 psig at ambient room temperature. Valve sizes of 1-1/4 in. and 1/2 in. were used in this area. Valve operators extended through sleeves placed in the wall between the vault and the equipment pit and permitted remote operation of all but one of the valves in the gallery. The function of the one valve which was not operated remotely was to drain residual NaK which could not be transferred to the secondary fill-and-drain tank, from the secondary system.

8. Piping and Thermal Insulation

All NaK piping in the primary loop was 1-1/4 in. schedule 40, seamless, type 304 stainless steel. Nominal 1/2-in. piping was used for the plugging indicator circuits and for the fill-and-drain lines. The NaK piping and components within the primary loop vault were thermally insulated, except for the electromagnetic pump, the primary flow meter, and the plugging indicator flow meter. The temperature limitations associated with these components made it imperative that these be at a lower temperature, i. e., 400°F; they were, therefore, not thermally insulated.

Figure 12. Plugging Indicator Assembly

G. SECONDARY NaK COOLANT SYSTEM

The function of the secondary NaK coolant system was to transfer the heat from the primary radioactive NaK to the secondary nonradioactive NaK and to dump this heat to the atmosphere by means of an airblast heat exchanger. The heat exchange from primary to secondary NaK was accomplished in the intermediate heat exchanger located within the shielded vault. The secondary inlet and outlet NaK lines of the intermediate heat exchanger passed through sleeves in the vault wall bordering the equipment pit. These lines led into and out of the pit to join that portion of the secondary coolant system located at the ground floor level.

1. Secondary Fill-and-Drain Tank

The secondary fill-and-drain tank was located in the equipment pit. The diffusion cold trap and sintered-metal filter used with this tank were of the same type as used with the primary fill-and-drain tank previously described. Its capacity was 35 gal. Transfer of the NaK from the tank to the secondary system was accomplished by argon pressure.

2. Secondary Surge Tank

This tank was installed in the secondary loop area of the reactor building at an elevation of approximately 7 ft above floor level. Instrumentation was provided within the tank for continuously monitoring the level of the NaK-free surface. Argon cover gas for the secondary coolant system was applied through an opening in the top of the surge tank. The secondary outlet flow from the intermediate heat exchanger was piped to the secondary surge tank and entered this tank through an inlet on the side below the NaK-free-surface level within the tank. NaK excited from the surge tank through an opening at the bottom of the tank.

3. Airblast Heat Exchanger

The NaK outlet coolant flow from the secondary surge tank entered the inlet header of the airblast heat exchanger. This heat exchanger was designed and built by Atomics International. The position of both the inlet and outlet dampers could be adjusted. A 20-in.-diameter fan powered by a 1/2-hp, variable-speed drive motor was mounted upstream of the inlet

damper section and received its air supply from inside the building. A 20-in.-diameter duct was used downstream of the outlet damper section for exhausting the air to the outside. During reactor operation, the louvers and fan speed were adjusted to give the heat dump desired. At the instant of a reactor scram, power to the fan drive was interrupted and both dampers closed automatically by spring actuators. Typical operating conditions for the airblast heat exchanger at 50-kw operation and 1200°F outlet temperature were:

Secondary NaK flow	11 gpm
Inlet NaK temperature	1155°F
Outlet NaK temperature	1105°F
Air flow	Zero
Fan speed	Zero
Dampers	Closed

4. Electromagnetic Pump

The NaK outlet flow from the airblast heat exchanger entered the suction header of the secondary electromagnetic pump. This pump, which was an exact duplicate of the primary electromagnetic pump described previously in Section E-4, was located on the floor of the equipment pit. Pressure transmitters to measure the pressure rise across the secondary pump during reactor operation were installed.

5. Secondary Flow Meter

The discharge flow from the secondary pump entered the secondary flow meter, a duplicate of the primary flow meter described previously.

6. Electrical Heater

NaK flow from the flow meter was piped to the equipment pit through the vault wall to an electrical heater unit capable of supplying an input of 50 kw to the secondary NaK flow. The electrical heater was originally installed in the primary loop at the exit from the intermediate heat exchanger. Owing to failure of some of the calrods in the heater bundle, however, this unit was removed and a new one installed in the secondary loop just upstream from the intermediate heat exchanger. The original unit contained 12 immersion heaters,

each rated at 4200 watts and 230 volts. The function of the electrical heater was to elevate the system to the operating temperature and to supply makeup heat to the system during reactor operation.

7. Valves

All valves in the secondary cooling system were bellows-seal valves rated at 300 psig at ambient room temperature. No remote valve operators were required, as all secondary system valves necessary for reactor operation were readily accessible. A 1/2-in. drain valve located within the primary loop vault was used to drain the residual secondary NaK which could not be transferred to the secondary fill-and-drain tank.

8. Piping and Insulation

All piping was schedule 40, seamless, type 304 stainless steel. Piping for the main flow circuit of the secondary coolant system was nominal 1-1/4 in. Fill-and-drain lines were nominal 1/2-in. pipe. Headers for the airblast heat exchanger were nominal 2-in. pipe. Finned tubing in the airblast heat exchanger was 1 in. in OD by 0.083 in. in wall thickness. The fill-and-drain piping was not insulated. All other NaK piping was covered with 3-in.-thick pipe insulation. An insulation thickness of 4 in. was applied to the surge tank. The airblast heat exchanger was wrapped with a 3-in.-thick rock-wool insulation blanket.

H. AUXILIARY SYSTEMS

1. Auxiliary NaK Coolant System

An auxiliary NaK coolant system, Figure 10, was utilized to provide a coolant for the plugging indicator used to monitor the oxygen and/or hydrogen content of the primary NaK coolant. The pump, surge tank, and flow meter for this auxiliary loop were located in the equipment pit. NaK lines which passed through sleeves located in the primary vault wall circulated auxiliary NaK through the shell side of the plugging indicator cooler. A fill-and-drain tank was not required for this system. Filling and draining was through a 1/4-in. bellows-seal valve attached to the bottom of the surge tank. A linear induction electromagnetic pump was used for circulation of the auxiliary NaK coolant. This pump was similar to the primary and secondary NaK pumps,

though much smaller. The design point of operation for this pump was 0.5 gpm, 70°F NaK at a head of 1 psi. The piping of the auxiliary system was 1/2-in., schedule-40, seamless, type 304 stainless steel. The piping and components for this loop were not thermally insulated.

2. Helium Gas System

Helium was used as a cover gas in the surge tank and the fill-and-drain tank of the primary system. Argon was originally used in the primary system; because, however, the argon became radioactive in passing through the reactor (either from entrained gas bubbles or because argon becomes slightly soluble in high temperature NaK), it was replaced with helium.

Transfer of NaK from the primary fill-and-drain tank was accomplished by pressurizing with helium; draining the NaK was accomplished similarly. All helium vented from the primary system was collected in a storage tank designed for 300 psi operation. The gas was monitored for activity prior to release through the common vent which exhausted several feet above the reactor building.

3. Argon Gas System

Argon was used as a cover gas in the surge tanks of the secondary and auxiliary NaK systems, as well as in the secondary fill-and-drain tank. In addition, it was used to purge the liquid metal systems prior to filling these systems with NaK. Transfer of NaK from the secondary fill-and-drain tank to the secondary system was accomplished by pressurizing with argon; draining the NaK from the system was accomplished similarly.

4. Nitrogen Gas System

Nitrogen gas was used for maintaining an inert atmosphere within the containment vessel and within the primary loop vault. Inlet lines were located near the top of the containment vessel and over the vault; outlet lines were provided near the bottom of each to permit adequate purging. A rubber gasket was used in the containment vessel top flange assembly to retain nitrogen in the vessel. A similar gasket between the lower and middle shield plugs of the primary vault sealed the nitrogen gas within the vault.

During reactor operation, a positive nitrogen pressure of 1/2-in. water was maintained in the containment vessel and the primary vault. Venting of

the nitrogen system after holdup, to allow the gas activity to decay, was through the vent used for the argon supply. Venting was required to control the moisture and oxygen content in either the containment vessel or the primary vault and to regulate the pressure as the system underwent temperature change. Purging was only required when either the oxygen content exceed 2% by volume, or the moisture content was such that the dewpoint was above 50°F.

5. Water Cooling System

The function of the water cooling system was to maintain low temperatures in the concrete surrounding the reactor containment vessel and the concrete walls of the primary loop vault. Two parallel water-cooling coils were attached to the outer surface of the containment vessel. These coils had a total combined length of 150 ft and were formed from 1/2-in., schedule-40, mild-steel pipe. Four parallel pipes were attached to the outside of the vault liner. These coils had a total combined length of 840 ft and were formed from 3/4-in., schedule-40, mild-steel pipe.

A closed-cycle, distilled-water system with hydrozene addition was used to control the buildup of scale deposits within the cooling coils. Water was circulated by a centrifugal pump rated at 10 gpm water at a head of 50 psi. A spare pump with associated valving was provided. The design water flow in each of the four parallel coils around the vault was 1 gpm. Typical flow during full-power operating conditions was 2.6 gpm for the containment vessel and 6 gpm for the vault. Cooling of the circulating distilled water was by a spray-type, forced-draft, evaporative cooler located on the equipment pad outside of the reactor building. The rated heat removal load of the cooler was 125,000 Btu/hr.

6. Air Cooling System

An air cooling system was used to cool the shutdown shield within the containment vessel. The shutdown shield assembly consisted of two parts — a ring shield body and a stepped shield plug located immediately above the reactor core (Figure 9). Each part had its own cooling system. The borated normal density concrete of the center shield plug contained a single coil of copper tubing for passage of air. This coil, however, developed leaks at an early stage in the operation of the SER and was sealed off. The ring shield body

contained nine separate cooling coils — one for cooling the main body of the shield, one for cooling each of the two pipe sleeves used to bring the primary NaK inlet and outlet lines through the shield, and one for each of the three control and safety sleeves. Typical air flow for the entire system during power operation was 110 scfm.

A positive-displacement, rotary-type air compressor driven by a 25-hp motor was located on the equipment pad outside the reactor building. A drier installed at the compressor discharge ensured the delivery of dry air to the various cooling coils. Exhaust from the cooling system was vented several feet above roof height along with the argon, helium, and nitrogen. All vent gas was monitored periodically to ensure that only nonradioactive gas was released to the surrounding terrain.

7. SER Parameters

The parameters of the SER are defined in Table III.

TABLE III^{1,2,4}
SER PARAMETERS

<u>Core Composition</u>	
Fuel loading	3.0 kg U ²³⁵
Number of fuel-moderator elements	61
Composition of fuel-moderator elements	6.88 wt % 93% U ²³⁵ 93.12 wt % ZrH
Hydrogen density, average	6.06 x 10 ²² atoms/cc
Hydrogen contained in core	378 moles
Diameter of fuel-moderator elements	0.975 in.
Length of fuel-moderator region	10 in.
Total length of fuel elements (1.5 in. long Be plug + 1/2 in. stainless steel on each end of fuel)	14 in.
Fuel cladding	0.010 in. 347 stainless steel
Fuel rod spacing (triangular center to center)	1.015 in.
Diameter of core vessel	9.5 in.

TABLE III (Continued)

Wall thickness of core vessel	0.094 in.
Core volume	0.35 ft ³
<u>Unit Cell Volume Fractions</u>	
Fuel-moderator material	0.845
Cladding material	0.035
NaK coolant	0.120
<u>Reactor Control</u>	
Reflector	3-in. Be
Number of control drums	3
Control drum worths, average (0 to 180°)	\$3.92 beginning of life \$3.36 end of life
Number of safety elements	3
Safety element worth	\$5.40
Control drum insertion rate, maximum	2.5¢/sec
Safety element insertion rate, maximum	6¢/sec
Scram time of safety elements	240 msec
Pu-Be startup source (15.78 gm)	1.68 x 10 ⁶ n/sec
<u>Reactor Physics</u>	
Prompt neutron lifetime	10 μ sec
β_{eff}	0.0084
Cold, clean excess reactivity	\$9.78
Limited motion of control drums (70 to 135°, 3)	\$6.74 start of life \$5.68 end of life
Total control drum worth (3)	\$11.76 start of life \$10.08 end of life
Total safety element worth (3)	~\$13.57
Prompt temperature coefficient (200 to 1000°F)	~-0.04¢/F° at 1000°F
Overall isothermal temperature coefficient	~-0.35¢/F° at 1000°F
Temperature defect (75 to 1100°F)	\$3.42
Equilibrium xenon defect (at 50 kw)	\$0.39
Power defect (0 to 50 kw)	\$0.23
Ratio peak-to-average radial flux	1.26

TABLE III (Continued)

Ratio peak-to-average axial flux	1.22
Ratio peak-to-average total flux	1.54
Average power density in core, 50 kw	5.45×10^5 Btu/hr/ft ³
Average heat flux in core, 50 kw	1.24×10^4 Btu/hr/ft ²
Maximum fuel temperature at 50 kw and 1200°F outlet NaK temperature	1270°F
<u>Reactor Cooling System - Design Conditions</u>	
Coolant, eutectic NaK	78 wt % K
Specific heat of coolant	0.21
Heat capacity of core, fuel and coolant in fuel section	12.64 kw-sec/F°
Total heat capacity - fuel, coolant, reflector, structural material	85.7 kw-sec/F°
Reactor outlet temperature, NaK	1200°F
Reactor inlet temperature, NaK	1000°F
Coolant inlet temperature to heat exchanger, primary	1200°F
Coolant outlet temperature from heat exchanger, primary	1000°F
Coolant inlet temperature to heat exchanger, secondary	950°F
Coolant outlet temperature from heat exchanger, secondary	1150°F
Maximum coolant pumping rate (1000°F, 20 psi head)	23 gpm
Type of primary pump	Electromagnetic
Diameter of coolant lines in main flow circuits	1-1/4 in.
Capacity of electrical heaters at 230 v	50.4 kw
Maximum auxiliary coolant pump rate (70°F, 1 psi head)	0.5 gpm
Diameter of coolant flow lines in auxiliary flow circuit	1/2 in.
Coolant pressure drop across core	<0.5 in. H ₂ O
Cover gas on coolant system	Helium, primary Argon, secondary

TABLE III (Continued)

Primary fill-and-drain tank capacity	24 gal
Primary surge tank operating pressure	1 psig
Secondary fill-and-drain tank capacity	35 gal
Secondary surge tank operating pressure	2 psig
Type of secondary pump	Electromagnetic
Cover gas on containment vessel and pipe vaults (slightly above atmosphere)	Nitrogen
Coolant system design pressure	300 psig
Overall heat transfer coefficient of heat exchanger	818 Btu/hr/ft ² /F°
Type of intermediate heat exchanger	Counterflow
Coolant system design temperature	1200°F
Maximum airblast heat exchanger air flow	2500 scfm
Airblast heat exchanger inlet air temperature	100°F
Airblast heat exchanger outlet air temperature	230°F
Airblast heat exchanger damper closing time	6 sec
Maximum airblast heat exchanger fan speed	1725 rpm
Heat load of H ₂ O cooling system	125,000 Btu/hr
Containment vessel design pressure (tested at 456 psig)	305 psig

III. INSTRUMENTATION AND CONTROL SYSTEM²

A. NUCLEAR INSTRUMENTATION

A block diagram of the nuclear instrumentation is shown in Figure 13. This includes appropriate instrumentation for startup, intermediate, and power ranges of operation. The range covered by each neutron chamber used is shown by Figure 14.

Two fission chambers were used to cover the startup range (5×10^{-4} watts to 5 watts). One channel was used in conjunction with a linear count rate meter and a scaler, and the other with a logarithmic count rate and period meter. The period meter was used for protection of startup but could be bypassed when the intermediate-range instruments were on scale. The logarithmic count rate meter was used as a source interlock to ensure a reasonable count rate prior to moving the control drums or the safety elements from their shutdown positions.

Two compensated ion chambers were used to cover the intermediate range (5×10^{-1} watts to 5×10^5 watts). The outputs of the two ion chambers were fed to two logarithmic current amplifiers and two period amplifiers. The outputs of the logarithmic amplifiers were recorded, while the outputs from the period amplifiers were fed to the safety amplifier used on the reactor scram system. The period scram was bypassed at powers greater than 10 kw, to reduce spurious scrams during operation at full power.

Three uncompensated ion chambers were used with three safety amplifiers to furnish independent high-speed flux level scrams in the power range (5×10^2 watts to 5×10^5 watts). One channel was used with a recorder controller to supply flux level information to the automatic power level controller.

An additional channel consisting of an uncompensated ion chamber and a micro-microammeter was used to furnish the operator with flux level information.

All neutron detectors were supplied from an electronically regulated power source except Channel No. 5, the Keithley Channel, which was used for operation information only. The neutron chambers were located in 5-in. instrument thimbles outside and adjacent to the containment vessel.

* THESE INTERLOCKS ARE BY PASSED WHEN THE SAFETY RODS ARE IN THE COCKED POSITION

Figure 13. Nuclear Instrumentation

Figure 14. Range of Operation of Neutron Detectors

B. PROCESS INSTRUMENTATION

The reactor and auxiliary coolant loops were monitored with temperature, pressure, liquid level, and flow instrumentation. All instruments installed in the containment vessel and primary loop system were capable of withstanding 300 psi. All instrument and control wiring that penetrated the containment vessel was brought out through a conduit filled with a resin compound to make a seal capable of withstanding 300 psi.

The instruments in the liquid-metal-coolant loops were selected to give the utmost reliability while maintaining the integrity of the system. The pressure at the primary and secondary pumps was transmitted through a high-temperature Inconel diaphragm in a stainless steel housing, which, in turn, transmitted the pressure through a NaK-filled capillary tube to an electronic pressure indicator. The diaphragm housing was welded into the system and was capable of withstanding pressures to 300 psig. The liquid level in the surge tanks was measured by a device which changed resistance with NaK level. It consisted of an Inconel tube in the shape of a "J" welded into the surge tank. The "J" tube was sealed and the resistance between two points on the tube measured. Since the resistance is a function of NaK level, a curve of NaK level vs resistance could be obtained.

C. CONTROL CONSOLE

All major functions of the reactor were controlled from the control console and instrument racks. The instruments were arranged so that the operator could quickly and easily detect faulty operation and apply corrective action.

The nuclear and process instrumentation and recorders were mounted in racks located in front of the control console and were visible to the operator. The controls of immediate interest to the operator were located at the console. These controls included:

- 1) Control drum drives
- 2) Safety element drives
- 3) Primary and secondary NaK coolant pump power
- 4) Manual-to-automatic reactor operation transfer switch

5) Airblast heat exchanger fan

6) Emergency scram

The following meters and indicators were also located at the console:

1) Reactor period

2) Reactor power level

3) Control drum position indicator

4) Safety element position indicator

5) Primary and secondary NaK flow rate

6) Airblast heat exchanger fan speed

IV. SER OPERATION¹

A. HIGHLIGHTS OF SER OPERATION

By the middle of September 1959, the SER core was ready to receive the first fuel element. The entire SER complex had completed all functional tests satisfactorily, so that the approach to dry critical tests could begin. The first 14 fuel elements were loaded on September 17, 1959. Dry criticality was attained on September 19 with a core loading of 55 fuel elements. After criticality was achieved, the initial control and safety element calibrations were obtained. The full loading of 61 fuel elements was reached on October 19. On October 20 the reactor was filled with NaK, followed by wet critical on the same date. After a series of tests to determine the operating characteristics of the system, the reactor was gradually raised to design power and temperature. The reactor was first operated at design conditions on November 9, 1959. Subsequent to the attainment of design operating conditions, an intensive testing program was undertaken to determine the nuclear parameters of the reactor. The results of these tests are presented in Section V.

The first extended power run began February 22, 1960, and was interrupted on March 12 by an instrument malfunction which resulted in a scram. The second extended power run was begun on the same day and continued until April 23 when the reactor was manually shut down after attaining 1000 hr of continuous operation at design power and temperature.

From this time until November 1960, the reactor was operated in accordance with the requirements of the various tests that were being conducted. As of November 19 1960, 224,650 kwh of thermal energy had been generated by the SER. A total of 1877 hr was at design conditions of 50 kw and 1200°F outlet temperature. Figure 15 shows the relationship of energy generated to average core temperature.

Figure 16 is a bar graph showing the salient points of reactor operation until final shutdown on November 19, 1960.

A summary of the SER operation is given in Table IV.

Figure 15. SER Operation at Various Core Coolant Temperatures

TABLE IV
SUMMARY OF SER OPERATION

Initial fuel loading	September 17, 1959
Final shutdown	November 19, 1960
Elapsed time during testing	10,306 hr, 440 days
Reactor operating time	6,035 hr, 58.5% of total time
Operation at 50 kw and 1200°F core outlet temperatures	1,877 hr, 31.1% of total operating time
Operation at 50 kw and less than 1200°F core outlet temperature	2,290 hr, 38.0% of total operating time
Operation less than 50 kw and less than 1200°F core outlet temperature	1,868 hr, 30.9% of total operating time
Reactor down time	4,271 hr, 41.5% of total time
Holidays and weekends	1,288 hr, 30.1% of total down time
Routine maintenance and experimental preparation	1.245 hr, 29.2% of total down time
Heater bundle failure	680 hr, 15.9% of total down time
Other component failure	1,058 hr, 24.8% of total down time
Total energy generated	224.6 megawatt hr
Equivalent time at 50 kw	4,493 hr $\approx$ 187 days

Figure 16. SER History

Dismantling of the reactor began on May 12, 1961. No difficulty was encountered in removing the fuel-moderator elements. The last fuel-moderator element was removed on May 31, 1961. The operation went smoothly, but slowly, as only six fuel-moderator elements could be removed at one time. This number was dictated by hazard and safety considerations.

All elements had the same physical appearance as the day they were loaded; i. e., there were no visible signs of deterioration due to high temperatures and radiation. Six of the fuel-moderator elements will be de-canned and the fuel-moderator thoroughly examined in the Component Development Hot Cell located at Atomics International's Field Laboratory in the Santa Susana mountains northwest of Los Angeles.

B. SCRAMS

A number of scrams occurred during SER operation. The number of occurrences and probable cause are listed in Table V in their order of frequency.

TABLE V
SCRAM

Number of Occurrences	Probable Cause
17	Abnormal signal in log power channels 3 and 4 and power level channels 6 through 8
15	Individual safety dropped out
13	Low "J" tube NaK level alarm
12	Circuit noise and accidental
6	False signal, source unknown
3	Disturbing of instrument cable in trench
3	Abnormal signal in startup channel (channel 3) period instrumentation
1	Instrument trouble
1	Heater bundle failure
1	Power failure
0	Initiated internal to core
72	Total through August 31, 1960

The factor that caused the most scrams was "abnormal" signal in log power channels 3 and 4 and power level channels 6 through 8. The majority of these scrams were caused by transient signals introduced into channels 3 and 4 when the period protection was switched into the safety circuitry. The period protection was normally removed from the safety circuitry with an on-off switch when the reactor was at full power and had to be switched in when the reactor power was dropped below 10 kw. This switching nearly always caused a scram.

Malfunction of period channel 3 caused several scrams at periods as long as 30 sec. Normally, a scram should not be initiated by the safety instrumentation until a 3-sec period was measured. Channel 3 instrumentation was subsequently repaired. A scram was initiated from power channel 7 because the scram point for this channel was set too low. The normal scram point was 75 kw.

Fifteen scrams occurred because an individual safety dropped out. This usually occurred during a rise in temperature of the reactor. This would indicate that the expansion of the different components of the safeties and the core vessel may have forced the safety to pull away from the magnet. The tension of the springs on the safety drives was adjusted several times in an attempt to remedy this situation.

Scrams caused by the "J" tube NaK level alarm were usually due to spurious signals in the "J" tube circuitry. The instrument measured a voltage of only about 10 mv a-c, and noise pickup presented a difficult problem. Modification and maintenance were only partially successful in eliminating this noise. Several "J" tube scrams were initiated by surges in the power line voltage.

Circuit noise in the low-level-period meter also contributed to the scrams. No scrams occurred because of a short reactor period. All period scrams were caused by either instrument malfunction or by noise in the circuitry; i. e., no scrams were due to abnormal conditions within the core.

Accidental scrams were due to someone accidentally bumping an instrument such as the "J" tube meter.

One scram was attributed to instrument trouble. This meant that a component in two of the three power safety channels was malfunctioning simultaneously with the other.

Heater bundle failure caused one scram. The magnitude of the resulting NaK leak was not sufficient to cause a scram from low NaK-level indications, but rather was caused by electrical shorts in the primary vault components due to the presence of NaK vapor.

C. MAINTENANCE PROBLEMS

1. Process

Considerable operation experience with liquid-metal cooling loops had been gained from the design, construction, and operation of liquid-metal-cooled reactors such as the Sodium Reactor Experiment (SRE). Therefore, no major problems were encountered in the process system of the SER. Most of the problems were of a minor nature and did not result in any major modification to the system.

The electric heater used to heat the NaK coolant was originally installed in the primary loop and was an immersion type. However, a heater bundle failure which occurred in August 1959, prior to reactor operation, pointed out the possibility of leaking radioactive NaK into the vault should the heater bundle fail during reactor operation. Primarily for this reason, the heater bundle was removed from the primary system and was installed in the secondary loop. One of the reasons for failure of the primary-loop heater bundle was the lack of adequate interlocks that would automatically turn off the heater on loss of coolant flow. Consequently, during pre-startup testing of the primary NaK loop, the primary pump was stopped without cutting off the power to the heater. This caused the NaK to boil and overstressed the calrod heaters, resulting in a NaK leak.

Three different heater bundles were installed in the secondary NaK loop. The first, which was similar in design to the original primary heater, failed after only a few months of operation and was replaced in late November 1959, by one of external design; i. e., the calrod heaters were external to the piping. Trouble-free operation was obtained with this heater until March 1960, when the first calrod failed. Continued difficulties were encountered until this heater was replaced in June. The third heater bundle for the secondary loop was of the immersion type, employing a different design from that of the original heater. No difficulty with this heater bundle was experienced until October 14, 1960,

when a calrod failure caused a leak in the secondary loop, necessitating a shutdown until the leak could be repaired.

Other minor problems that affected reactor operations were: (1) the leaks in the air-cooling system of the shutdown shield which caused a pressure rise in the reactor containment vessel; and (2) a failure of the airblast heat exchanger fan.

Although these problems resulted in an interruption in reactor operation, no difficulty was experienced in accomplishing the necessary repairs.

2. Instrumentation and Control Problems

Component failure in the nuclear and process instrumentation has been quite small. During the 1000-hr operation at design power and temperature, the only failure in the instrumentation was in one of the intermediate power level channels used for period protection at power levels below 10 kw. Consequently, during power operation this did not necessitate a shutdown for repair.

During reactor startup some instrumentation problems were encountered. The most troublesome was noise in the startup pulse channels. Most of the difficulty was the result of placing the low-level pulse wiring and the control wiring in a common trench. By placing the low-level pulse wiring in a shielded, flexible tubing, this condition was alleviated but not entirely solved.

Numerous scrams were caused by spurious NaK level signals resulting from the close proximity of telemetering lines to the control power wires. The voltage level coming into the control room was on the order of 15 mv. Any spurious noise from switching and relay contacts was easily picked up by the NaK-level circuit. Also, fluctuations in power line voltage affected the circuit. This problem was never entirely solved.

Thermocouple failures were numerous, pointing out the need for better materials and improved installation methods. The resistance thermometers on the inlet and outlet NaK piping below the shutdown shield failed soon after reactor startup. Because they were located in an inaccessible spot, it was necessary to rely on thermocouples external to the shutdown shield for inlet and outlet NaK temperature data. The in-fuel thermocouples exhibited a wide variation in output, indicating differences of as much as 200 F° in some instances.

This required that they be calibrated periodically to ensure that no further drift had taken place.

The high ambient temperature of the primary vault damaged the insulation on the heater wiring for the plugging meter, resulting in numerous failures. The installation of high-temperature heater wire resolved this problem.

The automatic control system was designed as a temperature controller; however, early failure of the differential thermocouples on the core in-and-out piping resulted in the modification of the system so that the power level could be controlled automatically by the neutron flux.

The flowmeter for the primary NaK loop was installed in such a way that the entire NaK flow passed through it. However, a small amount of flow bypassed the reactor core to the plugging meter. Consequently, the flowmeter indicated a slightly high flow rate for the core. It was not practical to change the location of the primary flowmeter on the SER, but future installations should allow for this fact.

The nuclear and process instrumentation as a whole has functioned very well with little modification to the original equipment.

V. SER TEST PROGRAM AND RESULTS¹

The objectives of the SER tests were to investigate: (1) the ability of the reactor to override the reactivity losses associated with obtaining operating conditions and extended operation at these conditions, and (2) the stability and safety of the reactor.

A. REACTIVITY OVERRIDE TESTS

To determine the ability of the reactor to override reactivity losses associated with operation, it was necessary to measure the following parameters:

- 1) The cold, clean, excess reactivity
- 2) The worth of the control elements
- 3) The temperature defect, cold-to-hot
- 4) The power defect, zero power to 50 kw
- 5) The defect associated with equilibrium xenon
- 6) The long-term reactivity losses

B. STABILITY AND SAFETY TESTS

To determine the stability and safety of the reactor, it was necessary to measure these additional reactor parameters:

- 1) The worth of the safety elements
- 2) The time associated with the safety element action for reactivity removal
- 3) The temperature coefficients of the reactor

The measurement of these parameters is detailed and summarized.

C. THE COLD, CLEAN, EXCESS REACTIVITY

The determination of the cold, clean, excess reactivity was made during the initial loading of the reactor with fuel. Dry criticality was achieved with a loading of 55 fuel elements. Determinations of zero-power control-element- and safety-element worth took place concurrently during the loading of the core,

so that the values of the fuel-moderator element worth could be found. Figure 17 presents a plot of the excess reactivity vs number of fuel elements loaded. The shape of the curve indicates that the reactivity worth of each fuel element increases per fuel element added. Note that the fuel loaded into the flats of the hexagonally shaped core has a greater worth than fuel loaded into the corners. This result shows that the closer radius of a "flat" element has considerable effect on the element worth. Figure 17 also shows the increased reactivity resulting from the insertion of the center plug into the shutdown shield.

Figure 17. Excess Reactivities for Various Fuel Loadings

The removal of a fuel element next to control drum No. 2 during the 60-element loading caused a pronounced reduction (about 14%) in the worth of the control drum. Similar effects were noted in other cases where fuel elements were missing near a control drum or safety element. During the final fuel loading, a stainless-steel dummy fuel element was inserted into the outer corner of the core lattice. This resulted in an increase in reactivity of approximately 17¢. This increase in reactivity is attributed to the reflective property of the stainless-steel dummy.

Before the addition of the 61st element, the inward rotation of the control drums was limited to 135° to comply with the SER Hazards Report specification of a 4% excess.² Thus, the excess reactivity available to the reactor operator for the 61-element loading was \$5.41, whereas, \$9.78 would have been available were the control drums allowed a full 180° rotation.

After this determination of the cold, clean, excess for a NaK-free core, the change in reactivity associated with the addition of the coolant was measured. The results from this test and from subsequent tests made throughout the core life indicated that this addition caused reactivity losses of the order of 5¢. This small net loss in reactivity is probably due to the positive reflector effect in the outer regions of the core cancelling the poisoning effect of adding NaK to the central region of the core.

The change in reactivity due to changes in fuel content and hydrogen content has been theoretically determined but has not been checked experimentally. These ratios are:

$$(\Delta K/K)/(\Delta N_u/N_u) \cong 0.064$$

and

$$(\Delta K/K)/(\Delta N_H/N_H) \cong 0.46$$

D. WORTH OF THE CONTROL ELEMENTS

During the initial loading of the reactor and at intervals during the operation of the reactor, measurements were made of the worths of the control drums. All of these measurements used the technique of making small motions of the control elements to place the reactor on a measurable, supercritical period. This period was interpreted through the in-hour relationship (using a $\beta = 0.0084$) to give the change in reactivity associated with the control-element motion. The total worth of a control element was taken to be the summation of such determinations over the total span of the drum rotation. This assumption seems to be valid due to the fact that there has been no evidence of interaction between the control elements.

Table VI gives the total worth of control elements measured during the initial loading buildup. The reported worths for those loadings less than

61 elements are for the full 70 to 180° rotation of the element. The region of the drums from 0 to 70° was not used due to the low worth, approximately 17¢.

TABLE VI
TOTAL CONTROL ELEMENT WORTHS, DRY CORE

Fuel Loading (Number of Elements)	Control Worths, Cents (at 70 to 180° Rotation)		
	1	2	3
54	-	-	173
55	-	263	257
57	-	-	336
59	-	-	353
60 + 1 Dummy	340	297	349
	Control motion limited to 135°		
61	-	-	230

A determination of the worth of control element No. 3 was made immediately after the addition of coolant to the reactor and before any power production by the reactor. The total worth (over 70 to 135° rotation) of this control element was \$2.25. Total worth (70° to 180°) is estimated to be \$3.65.

The drums were again recalibrated after 5932 kwh release. For a period of 12 days prior to the calibrations, the reactor had generated no power. The worth of the control elements is given in Table VII for various core average temperatures.

TABLE VII
CONTROL ELEMENT WORTHS, WET CORE
(5932-kwh Energy Release)

Control Element Number	Control Element Worth (cents) (At 70 to 135° Rotation)		% Difference
	At 268°F	At 584°F	
1	216	218	+0.92
2	230	228	-0.87
3	230	228	-0.87

The drums were again recalibrated after 147,358 kwh of energy had been released by the core. For a period of 35 days prior to the calibrations, the reactor had been at zero power. The worth curves established at this time for two different temperatures are presented in Figures 18, 19, and 20; the total worths are presented in Table VIII.

TABLE VIII
CONTROL ELEMENT WORTHS, WET CORE
(147,358 kwh Energy Release)

Control Element Number	Control Element Worth (cents) (At 70 to 135° Rotation)		% Difference
	At 290°F	At 800°F	
1	211	209	-0.95
2	221	217	-1.81
3	217	215	-0.92

The control drums were calibrated in December 1960, after a total energy release of 224,600 kwh of energy. This calibration indicated that no appreciable change in control drum worth occurred between the calibrations of July 1960 and December 1960.

A comparison of the tabulated values of the control drum worths indicates that there is no drastic temperature dependence, and that the addition of coolant to the reactor does not affect the worths. There is an unexplained difference between those calibrations conducted during initial operations and those conducted after large amounts of power generation. There is no reason to assume that either of these calibrations is more valid than the other without knowing the cause for the difference in calibration. This difference has been attributed to various causes, including the uncertainty in the proper in-hour relation due to the effect of gamma-induced photoneutrons and change of geometry due to thermal stress relief of reactor components. The explanations are, at best, speculative. It is hoped that examination of the reactor will permit a more meaningful explanation of this variation. If, however, it is assumed that there is no change in drum calibration with temperature, the percent difference listed in Tables VII and VIII is an indication of the accuracy of the measurements.

Figure 18. Control Drum No. 1, Comparison of Drum Worths After 141,100 kwh of Operation

Figure 19. Control Drum No. 2, Comparison of Drum Worths After 141,100 kwh of Operation

Figure 20. Control Drum No. 3, Comparison of Drum Worths After 141,100 kwh of Operation

For the purposes of this report, all measurements depending upon control drum calibration will refer to those sets used in the interpretation as either the earlier or later set; thus, differentiation is made between those calibrations done prior to the release of large amounts of energy and those done after the release.

E. THE TEMPERATURE DEFECT

The temperature defect of the SER was determined by heating the reactor with external heat to approximately 600°F, then bringing the reactor to 50 kwt power and heating to the design average core temperature of 1100°F. The positions of the control drums were recorded throughout this process and reactivity changes associated with changes in temperature by the use of the earlier control drum calibrations. The plot of change in reactivity with temperature

is presented in Figure 21. A normalization has been performed to remove the effects of xenon buildup and the power defect from that portion of the curve above 600°F. Over the linear portion of this curve, $d\rho/dT = 0.4\phi/F^\circ$.

Figure 21. SER Data, Temperature Defect from Data of Experiments for December 19, 1959

F. THE POWER DEFECT

The change in reactivity with power in the SER is ascribed to the establishment of temperature gradients throughout the core. A more detailed discussion of the effect of these gradients will be presented in the discussion of temperature coefficients. The measured reactivity loss associated with going from zero power to 50 kw was measured to be 23¢ by the interpretation of the drum motion associated with this change and by use of the earlier control drum calibrations.

G. EQUILIBRIUM XENON DEFECT

The observation of control drum motion during the establishment of equilibrium xenon indicates an equilibrium xenon defect of 39¢. There is no peaking

of the xenon concentration after reactor shutdown due to the low flux density of the reactor. A comparison of the calculated xenon transient and the measured xenon transient is presented in Figure 22. It is seen that these two curves are in close agreement.

Figure 22. SER Xenon Transient Experiment

H. LONG-TERM LOSS RATES

The long-term losses associated with the operation of the SER are both power and temperature dependent. The power-dependent losses are attributed to fuel burnup, samarium buildup, and fission-product poisoning other than samarium or xenon. Of these, the buildup of samarium is calculated to be the largest factor. The temperature-dependent losses are attributed to the loss of hydrogen from the fuel-moderator rod. This phenomenon would take place due to dissociation of hydrogen from the matrix material and its subsequent diffusion through the element cladding. Long-term runs at 50 kw were accomplished at core outlet temperature of 1200, 1000, and 700°F and the reactivity loss rates determined from the motion of the control drums during these runs. The drum calibrations used to determine these loss rates were the later calibrations. The loss rates measured during these runs are presented in Table IX and Figure 23.

The assumption is made that the temperatures involved in the 700°F outlet case are low enough for the loss of hydrogen to be negligible. This 0.26¢/day loss rate is interpreted as due to samarium and long-life fission product poisoning plus fuel depletion from operation at 50 kw. The difference between this value and the loss rates established at the more elevated temperatures may be associated with the loss of hydrogen at those temperatures.

TABLE IX
LOSS RATES

Core Average Temperature (°F)	Reactivity Loss Rate (¢/day)
1100	0.79 ± 0.03
900	0.39 ± 0.02
600	0.26 ± 0.04

Figure 23. Measured Loss Rate vs
Average Core Temperature

I. SUMMARY OF PARAMETERS PERTINENT TO THE OVERRIDE OF
REACTIVITY LOSSES

Cold, clean, excess reactivity

Drum rotation to 180° \$9.78

Drum rotation to 135° \$5.41

Control worth, total (old calibration)

Drum rotation to 135° \$6.77

Temperature defect

75 to 1100°F core average temperature \$3.42

Power defect	
Zero to 50 kw at 600°F	\$0.23
Xenon defect	
Equilibrium at 50 kw	\$0.39
Reactivity loss rates	
50 kw, 1200°F core outlet	0.79¢/day
50 kw, 1000°F core outlet	0.39¢/day
50 kw, 700°F core outlet	0.26¢/day

J. SAFETY ELEMENT WORTHS

The worths of the SER safety elements were determined during the initial critical measurements. The technique used was to compensate for an outward motion of a safety element by the inward rotation of a control drum and to rely on the calibration of the control drum to associate a reactivity change with the motion of the safety element. This technique required that some use be made of control drums adjacent to the displaced safety element. Interaction between adjacent control and safety elements would be expected. Therefore, the values quoted for the worth of the safeties can only be used as being indicative of their worth and not as an absolute determination. The values thus obtained for the total reactivity worth of each safety element are:

<u>Safety Element</u>	<u>Total Worth (\$)</u>
1	\$4.60
2	\$4.60
3	\$4.40

A typical safety element worth curve is shown in Figure 24.

K. SAFETY ELEMENT DROP TIMES

The drop time of the safety elements has been measured at intervals during the operation of the SER. The technique used to make these measurements consisted of photographing the cathode ray tube display of the time variation of the voltage from a potentiometer driven by the safety element. The drop times have been consistent throughout the history of the reactor. These times are approximately 240 msec from initiation of the drop signal until contact is made with the shock absorber. Recoil of the safety elements from the shock absorbers is slight.

Figure 24. Typical Safety
Element Worth Curve

L. TEMPERATURE COEFFICIENTS

The stability and safety of operation of the SER are determined mainly by the sign and magnitude of its reactivity responses to temperature changes. Because of the importance of these phenomena, a large portion of the testing on the SER was devoted to a study of temperature effects. Studies were conducted to determine the temperature response not only at the design conditions, but over a spectrum of temperatures, so that information could be obtained to check calculational techniques. The overall temperature response may be considered to be the sum of several effects.

One portion of the temperature response should be that associated with changes in temperature of the fuel-moderator materials. Owing to the intimate contact of the fuel and moderator, a prompt, negative coefficient is associated with these elements. The only time delay from changes in power to action of this coefficient is associated with the thermal capacity of the elements themselves. This thermal capacity has been reported to vary from 14 kw-sec/F° at 200°F to 20 kw-sec/F° at 1000°F.

Another portion of the temperature response of the reactor is referred to as the grid plate temperature coefficients. Changes in the temperature of the plates in which the ends of the fuel are constrained would change the geometry of the reactor, thereby varying the reactivity. The reactivity response through this coefficient to changes in reactor power would have a longer time constant than that of the fuel coefficient due to: (1) the delay time associated with the passage of heat through the fuel element cladding to the coolant to the grid plates, and (2) the thermal capacities of the plates themselves.

Additional effects on reactivity by temperature may be associated with those slower-acting changes related to the heating of the core vessel and safety elements, which change the reflector properties.

By use of these discussions as the definition of three temperature coefficients (a_f , the coefficient associated with the change in temperature of the fuel-moderator elements; a_g , the coefficient associated with the change in core geometry due to grid plate motion; and a_r , the coefficient associated with change in reflector properties) one may write the following equation:

$$\frac{d\rho}{dT} = a_f \frac{dT_f}{dT} + a_{g1} \frac{dT_{g1}}{dT} + a_{g2} \frac{dT_{g2}}{dT} + a_r \frac{dT_r}{dT} ,$$

where T is any reference temperature usually taken to be the average core temperature; and T_f , T_{g1} , T_{g2} , and T_r are the temperatures of the fuel elements, lower grid plate, upper grid plate, and reflector, respectively. This analysis assumes constant temperature coefficients of reactivity. Though it is known that these coefficients vary slowly with temperature, it is assumed that they are constant over the range of small temperature changes.

To investigate these temperature responses, a series of tests was conducted on the SER. These are detailed below.

1. Isothermal Temperature Coefficient

The most straightforward determination of the temperature coefficients associated with the SER is the measurement of the coefficient under isothermal conditions. In this case the statement

$$\frac{d\rho}{dT} = a_f \frac{dT_f}{dT} + a_{g1} \frac{dT_{g1}}{dT} + a_{g2} \frac{dT_{g2}}{dT} + a_r \frac{dT_r}{dT}$$

would reduce to

$$\frac{d\rho}{dT} = a_f + a_{g1} + a_{g2} + a_r .$$

The closest approach to isothermal conditions possible on the SER is that condition under which no power is produced by the core, and any temperature change is produced by heating the coolant external to the reactor. Measurements under these conditions were accomplished in the temperature range from 150 to 925°F. The limitations on the capacity of the external heater precluded higher temperature determinations.

To minimize the errors normally associated with differential measurements, extreme care was used in this test. Whenever possible, control drum motion was limited to one drum, to minimize positioning uncertainty. The reactor was stabilized at a temperature, and several reactor periods were measured with the positions of the controlling drum noted for each period measured. The temperature of the reactor was then changed and stabilized at a new temperature.

The controlling drum was then returned to the positions at which periods were determined at the previous temperature and periods for these positions determined. In this manner, difference in reactivity could be implied from the direct comparison of two periods measured with all reactor parameters the same, except the reactor temperature. Thus, the temperature coefficient $\Delta\rho/\Delta T$ was determined for temperature steps of approximately 40 F°, and no reliance had to be placed on previously measured control drum calibrations. The results of this series of measurements are presented in Figure 25.

Figure 25. Temperature Coefficient ($\Delta\rho/\Delta T$) vs Average Core Coolant Temperature

From this measurement, the overall temperature coefficient seems to be smoothly varying — almost a linear function of temperature, and a value of approximately $-0.35\$/F^\circ$, may be extrapolated for a core average temperature of 1100°F.

2. Power Coefficients

One method which may be used to attempt to separate the measured overall temperature coefficient into its various components is to measure the variation in reactivity with reactor power under carefully controlled conditions. The measured "power coefficient" may be expressed:

$$\frac{d\rho}{dP} = \frac{\partial\rho}{\partial T_f} \frac{dT_f}{dP} + \frac{\partial\rho}{\partial T_{g1}} \frac{dT_{g1}}{dP} + \frac{\partial\rho}{\partial T_{g2}} \frac{dT_{g2}}{dP} + \frac{\partial\rho}{\partial T_r} \frac{dT_r}{dP} .$$

It can be seen that, if the power coefficient is measured in such a way that the gross core temperatures are unaffected, the terms

$$\frac{dT_{g1}}{dP} , \quad \frac{dT_{g2}}{dP} , \quad \text{and} \quad \frac{dT_r}{dP}$$

would go to zero, and

$$\frac{d\rho}{dP} (\text{constant } T_{in}, T_{out}) = \frac{\partial\rho}{\partial T_f} \frac{dT_f}{dP} .$$

A series of power coefficient studies was made under conditions of constant coolant outlet and inlet temperature. This was accomplished by varying the flow. One of the underlying assumptions of this study is that the flow distribution, and thus the temperature profile, are unaffected by such flow changes.

The results of these studies for varying core average temperatures are given in Figures 26 and 27. Studies have shown that the quantity dT_f/dP is dependent upon temperature due to two effects. As the temperature increases, the size of the annular region between the fuel-moderator material increases, effecting an increase in dT_f/dP . At temperatures above 600°F, the evolution of hydrogen from the fuel-moderator material raises the partial pressure of hydrogen in the annulus, causing dT_f/dP to decrease. The net result of these two effects is to cause a decrease in the dT_f/dP term at temperatures above 600°F.

Thus, the decrease in the constant inlet, constant outlet power coefficient can be explained by a variation in the heat transfer coefficient of the fuel element with temperature. Because the absolute calculation of the heat transfer coefficient depends upon a detailed knowledge of the as-built clad clearances, and residual gas content of the annulus, it is impossible to assign a meaningful value to this parameter. The only conclusion that can be drawn from the power coefficient at constant core inlet and core outlet temperatures is that the results do not contradict values for the fuel element coefficient determined by other methods.

Figure 26. SER Power Coefficient

Figure 27. SER Power Coefficient
vs Average Core Temperature

3. Ramp Reactivity Insertions

A set of studies was performed to attempt to separate the various components of the overall temperature coefficient of the SER. These studies consisted of placing the reactor on a positive period by the insertion of known amounts of reactivity by drum rotation at low power levels. The reactor power was allowed to increase until the reactivity insertion was compensated for by the increasing temperature. These studies were conducted with a variety of initial average core temperatures and flow conditions, including conditions of no flow. Typical curves resulting from these ramp insertions are shown in Figures 28 and 29.

Figure 28. Ramp Insertion Experiment, kw vs Time

Figure 29. Ramp Insertion Experiment, Reactor Temperatures vs Time

Reactivity insertions were performed with no coolant flow and with the coolant loop valve closed to limit convection of the coolant. It was hoped in this manner to limit the transfer of heat from the fuel to the other reactor components. The analysis of the transients was performed under the assumption that the only temperature coefficient acting during these transients was that associated with the fuel-moderator elements.

Using values of core heat capacity ranging from 14 kw-sec/F° at 200°F to 20 kw-sec/F° at 1000°F, values of a_f were obtained for each no-flow transient performed. The results indicated a prompt temperature coefficient of approximately $-0.04\text{¢}/\text{F}^\circ$ over the temperature range of from 200 to 1000°F.

Another series of ramp insertions of reactivity was accomplished under varying flow conditions. The purpose of these experiments was to couple the effect of the grid plates into the reactivity compensation. These transients were studied by simulation of the reactivity input on an analog computer. The feedback terms of the analog were varied until the analogous power and temperature traces best matched those recorded from the reactor. A no-flow transient was also analyzed in this manner.

Results obtained from analog studies for no-flow conditions indicated a temperature coefficient of $-0.065\text{¢}/\text{F}^\circ$, which is attributable to the fuel-moderator elements. The coefficient implied from the studies made with flow indicates a coefficient of $-0.15\text{¢}/\text{F}^\circ$. This is attributed to both the grid plate and the fuel-moderator elements.

The values of the various coefficients contributing to the overall temperature coefficient are:

	Value ($\text{¢}/\text{F}^\circ$)
Isothermal temperature coefficient	-0.35
Fuel-moderator temperature coefficient	~ -0.05
Grid-plate temperature coefficient (per plate)	~ -0.10
Not accounted for by measurements (thought to be associated with external reflector)	-0.1

M. ANALOG STUDIES

To determine the reactor performance under transient conditions, a Donner analog computer was set up in the SER control room. A mathematical model of the reactor core dynamic behavior was programmed into the computer. An input driving signal of core inlet temperature, NaK flow rate, and control drum reactivity was obtained from the reactor instrumentation and fed into the analog computer. The response of the reactor and the analog computer to changes in coolant flow and reactivity were simultaneously recorded on a Sanborn multichannel recorder. The parameters of the analog computer model were varied until the output of the computer closely matched the output from the reactor.

1. Reactivity Oscillation Experiments

Control drum reactivity oscillation experiments were performed to obtain data concerning the prompt power coefficient and the reactor time constant. These experiments were also found useful for calibrating control drums at operating conditions.

a. Control Drum Calibration

Control drum oscillations of the sawtooth type shown in Figure 30 were used to calibrate the control drum.

Figure 30. Control Rod Calibrations

Neutron kinetic equations — which are believed to closely represent the kinetic behaviour of the SER with no reactivity feedback — were programmed into the analog computer. An electrical signal representing the control drum position was used as the input reactivity to the analog computer, and its value recorded. The control drum was then manually moved in an in-and-out manner so that the reactor power deviated ± 2 kw from its steady-state level. The frequency of the reactivity oscillation (ca 0.14 cps) was about a factor of 10 greater than the highest frequency at which reactivity feedback effects are significant; the response of the reactor, therefore, was governed by neutron kinetics. The amplitudes of the analog computer oscillations were compared to the reactor power oscillations, and the reactivity change corresponding to a given drum-position change was adjusted on the analog computer until the oscillations matched. When this match was achieved, the analog computer gave a direct indication of the reactivity change corresponding to a change in drum position.

The above method yielded control drum worths differing about 10% from the previously determined values and is recommended as a reasonably accurate and simple means of calibrating control drums at power where conventional methods are not applicable.

b. Prompt Power Coefficient of Reactivity

Control drum oscillations of the trapezoidal type shown in Figure 31 were used to estimate the reactivity feedback parameters of the SER. The frequency of the reactivity oscillations was such that the power behaviour is dependent upon reactivity feedback parameters, such as thermal time constants and power coefficients of reactivity. The reactivity feedback model used consisted of a steady-state, prompt power coefficient delayed by a first order lag. When the following values were used, the behaviours of the computer and the reactor were essentially indistinguishable.

Prompt power coefficient of reactivity	-0.5¢/kw (950°F average core temperature)
Fuel time constant	10 sec
Reactor time constant	43 sec

The location of the thermocouple which measured the core outlet temperature (see Section IV-B-2) made it impossible to obtain a good transient measurement of the core outlet temperature and prevented a comparison of this variable to further verify or disprove the established model.

Figure 31. Reactivity Trapezoidal Oscillations

2. NaK Flow Rate and Inlet Temperature Variation Experiments

The intent of these experiments was to obtain information concerning the components of the prompt power coefficient, i. e., the fuel and grid-plate

temperature coefficients. However, primarily due to a lack of suitable NaK core inlet temperature indication, the experiments were not successful in achieving the desired objectives. They did, however, reveal some interesting phenomena concerning the reactor and thermocouple behaviour.

The only available measurements of the inlet and outlet NaK-coolant temperatures were from thermocouples located on the outside of the piping 8 to 10 ft from the core. It was observed that the thermocouple on the exit NaK pipe responded almost immediately to a flow change. For a flow decrease, the thermocouple indicated a temperature decrease. This type of response can be expected when the pipeline acts as a heat exchanger. This also explains the unexpected behaviour of the reactor power to flow changes. As the flow decreased, the initial response of the reactor was a rise in power rather than a decrease, as would normally be expected. A decrease in the NaK inlet temperature due to the flow change would tend to produce this behaviour. Because the inlet temperature thermocouple was located too distant from the reactor and its response was sluggish, the analog computer did not have a suitable input temperature signal and could not properly simulate the reactor response.

The results of this experiment dramatically point out the need for the placement of thermocouples as close as possible to the point where the temperature measurement is required.

VI. CONCLUSIONS

The Systems for Nuclear Auxiliary Power program was undertaken to fill a real need for small, compact, power sources for use in space vehicles. Present power sources severely limit the range and size of present space capsules. The successful operation of the SNAP 2 Experimental Reactor under design conditions for the equivalent of a six-month operating period is assurance that the development of small, compact, nuclear reactors for use in space vehicles is not only feasible but practical.

The system has proven to be inherently stable, having a negative power coefficient at all power levels, which minimizes the possibility of power excursions. It also facilitates reactor control under orbital conditions and ensures stable reactor operation over an extended period.

During a 1000-hr endurance run, all process and nuclear instrumentation, as well as the reactor and heat transfer system, functioned satisfactorily. This is an indication of the endurance capabilities of the SNAP 2 system.

Although several scrams occurred during reactor operation, none were due to abnormal conditions within the core or to short reactor periods. There were no scrams during the 1000-hr endurance run.

The NaK loop electrical heater presented the only real maintenance problem during the operation of the SER system. It failed on four different occasions. Three of these failures occurred during reactor operation and necessitated extended shutdowns. This heater will not be a part of the SNAP flight system, however, so this problem is of no real concern.

The successful operation of the SER verified the design concept and established the reactor parameters needed in the design of second-generation reactors. Sufficient data have been accumulated from the operation of the SER to enable the second phase of the SNAP program to proceed with confidence.

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