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SAFETY EVALUATION OF SODIUM GRAPHITE REACTORS

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NAA-SR-1626
REACTORS-POWER
M-3679 (17th Edition)

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SAFETY EVALUATION OF SODIUM GRAPHITE REACTORS

WRITTEN BY:

R. C. GERBER

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
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I. INTRODUCTION

This paper discusses the safety evaluation of the type of sodium-graphite reactors in the design and construction stages at Atomics International. Some of the statements apply generally to graphite-moderated, sodium-cooled reactors and others just to the concept described in the paper. Specific reactors discussed are the Sodium Reactor Experiment (SRE) now under construction at a site near Los Angeles, and a similar 75-Mw (electrical) reactor in design. The design concept is considered first, then the safety and hazards features of sodium-graphite reactors and, finally, the safety evaluation.

II. DESIGN CONCEPT

The reactors are composed of a graphite moderator and reflector, sodium coolant and low-enrichment uranium-metal fuel. The moderator region is approximately 6 feet in diameter by 6 feet high for the SRE and 12 feet in diameter by 10 feet high for the larger reactor. The reflector region is 2 feet thick for both reactors. Hexagonal prisms of graphite contained in thin-walled zirconium cans constitute the moderator and reflector. A central zirconium tube welded into each end of the moderator can provides a coolant channel and fuel element location. These cans are supported inside a large tank by a grid plate - attached near the bottom of the tank. Coolant is pumped into the plenum chamber formed by the tank and the grid plate and through the coolant channels to a free-surface pool. An inert atmosphere of helium is used to "blanket" the sodium. Fuel assemblies are suspended in the coolant channels by hanger tubes that extend from shield plugs, to the top of a cluster of fuel rods. Individual rods are made up of stainless steel tubes containing cylindrical slugs of enriched uranium. A NaK bond is provided inside the tube to insure good thermal contact between the uranium and the steel tube.

Control and safety units are located at corners of the hexagonal moderator cans in the SRE and in the fuel channels for the large power reactor. The "poison" portion of the units are segmented rods consisting of short, hollow cylinders of nickel-boron alloy supported on central steel rods. Stainless steel thimbles, supported by the top shield, separate the rods from the surrounding sodium.

Control rods are driven vertically by a ball-nut mechanism located in the shield section of the thimble. Safety rods are released for a gravity drop by a latch at the bottom of the rod and picked up by a ball-nut mechanism.

Heat is removed from the reactor in radioactive primary sodium loops, transferred to nonradioactive secondary sodium loops and then to steam generators or to air-blast heat exchangers. Motor-driven centrifugal pumps are used to circulate the sodium in all loops.

The reactors are constructed so that the top of the reactor shield is at floor level in the reactor building. Foundation concrete and the earth act as the side and bottom shields for the reactor. The radioactive primary sodium loops are located in concrete-covered cells adjacent to the reactor. Secondary sodium lines are brought outside the reactor building to steam generators located above ground in a separate building.

Service systems required for operation of the reactors are: an organic-fluid shield cooling system; a sodium service system for filling and draining the sodium loops; helium and nitrogen gas systems; an electrical distribution system, including an emergency power supply; waste-disposal facilities for radioactive-liquid and -gaseous wastes; and equipment for handling, storing, cleaning, examining, and shipping the reactor fuel.

III. SAFETY AND HAZARDS FEATURES

An important consideration in the choice of the sodium-graphite concept is safety. Graphite is used to thermalize neutrons - providing relatively long time responses for control. Graphite also provides a large heat capacity to minimize temperature transients. Sodium is used to remove the heat energy at high temperatures with a low-pressure reactor system. The particular concept described in this paper incorporates a large volume of sodium over the reactor which is also useful in minimizing extreme temperature transients. The thermal capacity of a heat machine compared to the power output gives an indication of the time available for control systems to correct difficulties. The sodium-graphite systems have relatively large thermal capacities compared to most other reactor systems and are comparable to conventional fuel-fired boilers. Table I shows SRE values of heat capacity of fuel, sodium and graphite and corresponding average times at full power absorption of these materials to reach critical temperatures. Discussions of incidents in the last section of this paper emphasis the desirability of relatively low heat capacity in the fuel compared to the sodium and graphite.

TABLE I

SRE AVAILABLE HEAT CAPACITIES
SODIUM GRAPHITE REACTOR (68 x 10^6 Btu/hr)

	Heat Capacity (10^6 Btu)	Temperature (° F)	Time (at Full Power) (Minutes)
Fuel Elements			
Operating Temperature to Melting Point of Uranium	0.3	2070	0.3
Sodium			
Operating Temperature to Boiling Point of Sodium	12	1620	11
Graphite			
Operating Temperature to 1620° F	23	1620	20

The sodium-graphite reactors are operated at near atmospheric pressure, eliminating the hazard of potential energy of gases and liquids under high pressure. In this reactor concept, a free surface of sodium is maintained over the reactor and in inert gas is used to blanket the sodium. The SRE design gas pressure over the sodium surface is maintained at 3 psig to insure outward sodium leakage in the event of piping failure. Other operating pressures are about 1 psig at the pump suction, 17 psig at the pump discharge and 14 psig in the plenum under the reactor. Over-temperature excursions in the reactor will not change those pressures appreciably. The vapor pressure of sodium at 960° F, the pool temperature during reactor operation, is about one hundredth of an atmosphere. At 1200° F the vapor pressure is only seven hundredths of an atmosphere; at 1620° F the pressure is one atmosphere.

Sodium-graphite reactors are inherently safe from the release of chemical energy as a result of the compatibility of the materials of construction. There are no reactions between uranium, sodium, NaK, steel, zirconium and graphite or between these materials and the blanketing gases or the organic shield coolant. In fact, the lack of the coolant-uranium reaction makes detection of cladding ruptures difficult.

Another "safe" feature of the sodium-graphite reactors is the negative temperature coefficient of reactivity. In normal operation this will stabilize the power level and, in the case of failures that lead to temperature increases, will be instrumental in checking the power rise. Examples of negative temperature coefficient of reactivity will be discussed later in this paper.

Some of the characteristics of the sodium-graphite systems are hazardous compared to other reactor systems. Sodium decays with a 14.7-hour half life, emitting a 2.8-Mev gamma and a 1.4-Mev gamma. The neutron capture rate in sodium is relatively high, resulting in a high level of coolant activity. In the case of the SRE, the equilibrium level is expected to be 0.3 curies/cm^3, or a total of 8×10^6 curies in the primary sodium coolant loop. This level will not be much higher in the larger reactors since sodium increase in volume outside the reactor tends to be proportional to power increase. The high-level activity leads to the use of thick shields and a requirement of a high degree of containment integrity throughout the primary system.

Some energy is stored with graphite as a result of irradiation. Operating temperatures of the graphite will range from 600° to 1000° F - a range that will promote annealing. The total stored energy is small under these proposed operating conditions, and certainly insufficient to sustain its autocatalytic release.

The most hazardous feature of sodium-graphite reactors is the chemical reaction between the sodium coolant and air, steam and water. The design philosophy is to isolate the coolant from these fluids by a series of barriers. Active coolant is separated from air by the steel structure, an inert-gas blanket and by a thick layer of dense concrete maintained at room temperature. Water and steam used in the turbine-generator part of the plant, and used for cleaning of fuel elements, are separated from the active sodium by large distances in addition to the containment features of the sodium system. The secondary cooling system of nonradioactive sodium is handled by precautionary measures similar to those used in chemical plants handling fluids subject to burning. Fire-fighting equipment is installed, and power plant operators are specially trained to fight sodium fires. The present practice is to use double-tube steam generators with a monitor fluid between the tubes. Tests conducted over the past few years* show that with proper pressure relief in the design, mixing of these fluids can be safely accomplished; future steam generators may not require these precautions.

IV. SAFETY EVALUATION

The safety evaluation included in this portion of the paper is a summary of information collected to judge the safety features of sodium-graphite systems. Most of the specific information applies to the SRE, but conclusions should also be valid for other reactors of similar construction. This presentation consists of site considerations, design features, operation characteristics, containment features and a study of reactor incidents.

A. SITE CONSIDERATIONS

Site descriptions for sodium-graphite reactors are probably very similar to descriptions prepared for other types of reactors. The site chosen for the SRE is located in the Simi Hills, 30 miles northwest of the center of Los Angeles. There are no residences within the 1.4-mile exclusion radius. A large volume of information was collected on the climatology, hydrology, and seismology of the site. Balloon tracking was used to determine the nature of local winds. Radioactive aerosol diffusion studies gave more quantitative answers to contaminant transport. This aerosol study combined the effect of both temperature inversions and winds. The highest counts were obtained when the inversions were at or near the elevation of the site, found in and near the exclusion area; the lowest counts were found in the populated valley areas below the site.

B. DESIGN FEATURES

The general design features of the reactor are described in detail in a safety evaluation. These features of the SRE and the larger SGR have been describe in the introductory portion of this paper and will not be repeated here except as they are related to operation and containment.

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C. OPERATION CHARACTERISTICS

A normal startup of the SRE reactor begins with the reactor at an isothermal temperature over 350° F, heated either by electric heaters or by residual heat from fuel elements. First the sodium flow is established at full flow. Then the reactor is brought up to critical, checks are made, and the reactor power level is increased to about 15 per cent of full reactor power. At this point, the coolant flow is gradually reduced to establish the operating temperature gradient. The reactor is then manually brought to full power by ramp changes in heat removal, with reactor power changes to maintain the operating temperature gradient. After the desired flux level has been attained under manual control, a servo system may be brought into use to drive one of two regulating control rods to hold the neutron flux constant. A manual control system is used to adjust pump speed and maintain constant temperatures in the main coolant loops. When heat is rejected through the air-blast heat exchangers, the temperature of the sodium leaving the unit is held constant by fan speed control. The power generation unit that can be connected to the SRE is experimental in nature and its operation is based on heat removal from the reactor rather than the conventional power demand systems used in the larger reactors designed primarily as power plants. Constant sodium temperature return to the intermediate heat exchangers is maintained by control of the feedwater to the once-through steam generator. Control is based on sodium temperature and sodium flow signals at the steam generator.

The auxiliary heat removal system will operate continuously to reject about 1 Mw of thermal heat through a radioactive primary sodium loop, a nonradioactive secondary sodium loop and an air-blast heat exchanger.

Normal shutdown and scram shutdown of the reactor are now being studied by mocking up the many possible combinations of control rod, safety rod and pump speed on analog computers. The maximum rate of reactivity decrease is attained by dropping the four safety rods, which takes approximately 1/2 sec., and by driving in the four control rods at 0.3 ft/min. Each rod represents about 2-1/2 dollars in reactivity. The analog studies have shown that temperature transients are severe if full sodium flow is maintained. The probable scram shutdown of the reactor will be a release of the four safety rods and immediate interruption of power to the main pumps. A single rod drop, a delay of a few seconds and a drop of the other three rods is also being considered for some of

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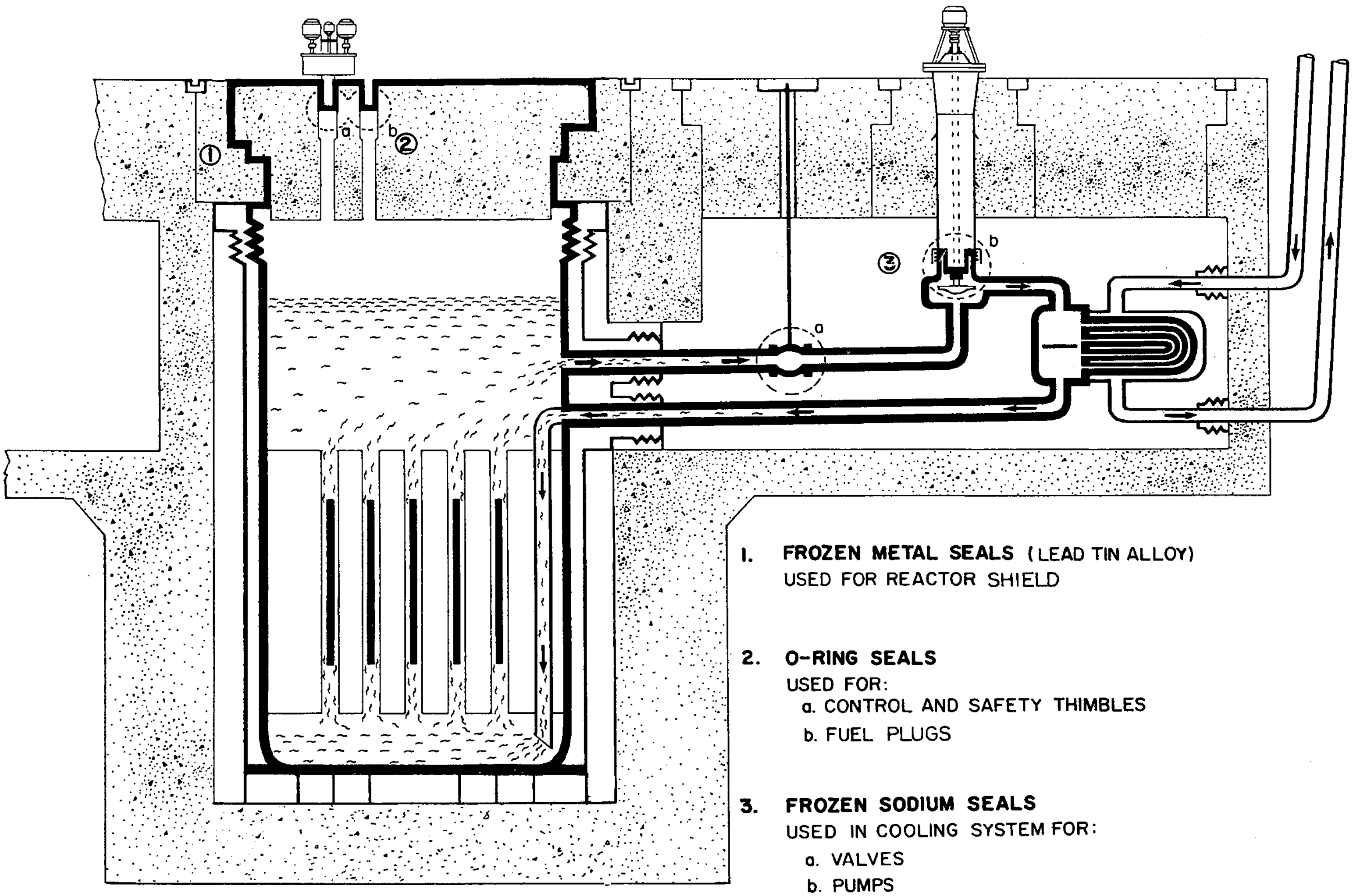

Fig. 1. Reactor Core and Primary Coolant Loop
(The Heavy Line Defines the Gas-Tight Barrier)

S25-94A

the conditions now on scram circuits. Safety circuits are being provided to cause reactor scram due to: a high flux level; a short period of neutron flux; loss of flow in the main sodium loops; a high temperature at the fuel element outlets; failure of electrical power; earthquake; loss of feedwater when operating with steam plant; loss of air-blast fans when operating with air-blast heat exchangers.

Normal shutdown will involve manual step-changes in reactor power and flow, maintaining the operating gradient until a 15 to 20 per cent power range is reached. Then the reactor is made subcritical and the temperature gradient is reduced to the shutdown condition based on afterglow heat loads. Safety rods are finally dropped, probably testing one of the scram circuits.

D. CONTAINMENT FEATURES

The containment features of sodium-graphite reactors required to maintain an extremely low oxide content in the sodium, and the shielding required for the normal sodium activity, provide a series of barriers that adequately contain the radioactivity in all but the compounding of many improbable accidents. The fission-product activity will be distributed rather uniformly throughout the body of the metallic uranium slugs. These slugs are contained in stainless steel tubes, welded closed and tested for leak-tightness. The uranium slugs are thermally bonded to the jacket with NaK alloy. The NaK becomes radioactive in the neutron flux and will contain some fission-product activity as the result of fission fragments formed near the surface of the uranium. The activity of the long-lived gaseous fission products in the NaK is estimated at 100 curies per element for the SRE fuel element, which is an extremely small fraction of the total activity. Rupture of the steel jacket can permit this small fraction of the fission products to enter the primary coolant system where they are retained by the barriers provided for the coolant. It is important to note that there is no accelerated corrosion of the uranium fuel and hence no resultant increase in the amount of fission-product activity which might be admitted to the primary coolant.

The primary sodium is contained in the stainless steel envelope formed by the reactor tank, heat exchangers, pumps, valves and piping, except for the free surface of sodium in the reactor tank. This arrangement is shown in Fig. 1, the heavy outline showing the gas-tight barrier. In the reactor region, this barrier is provided below the free surface by the tank and piping, and above the free surface by the steel-lined top shield and various access plugs. O-ring seals and gaskets near the top of the access plugs are exceptions to the stainless steel envelope. The steel barrier in the heat exchanger cell, shown on the right side of the Fig. 1, is formed by the heat exchangers, pumps, valves and piping. Frozen-sodium seals are used in the pumps and valves.

The biological shield structure forms a secondary barrier to the escape of radioactivity from the primary coolant system. The shield temperature is maintained below 150° F, so the shields will act as effective vapor traps for sodium vapor in the event of a rupture in the steel envelope. Passages between the shield blocks are sealed with neoprene gaskets to prevent gas escape.

Design features minimize escape of radioactivity into the surrounding earth. An outer tank is provided around the reactor tank to contain sodium in the event of a rupture in the main tank. The cavity in the reactor region is steel-lined as a part of the shield cooling system and as a foundation form. The heat exchange cells are also steel-lined to a height that could conceivably contain liquid sodium in the event of a pipe failure. The concrete foundations and bitumal paint outside the concrete form a final barrier to escape of radioactivity.

Figure 2 shows the frozen-sodium pump seal. Two seals are made by cooling the sodium below its freezing point; they are around the impeller and between the impeller unit and pump housing. The organic shield cooling system supplies the coolant for both seals. Possible failure of this portion of the shield cooling system is covered by connection to the emergency electrical system and by parallel-operated pumps. A helium atmosphere is maintained in the housing above both seals, allowing limited pump operation in the event of freeze-seal failure. The gas is sealed above the shielding by a conventional shaft seal, a static O-ring and a gasket. The seal between the housing and impeller unit provides a simple means of impeller removal for repair or replacement.

The valve seals, as shown in Fig. 3, are simplified versions of the pump seals. The freeze section of the seal is backed-up by a helium section with a conventional seal and then is open to the inert gas in the heat exchange cell.

The seals for the access plugs in the top shield are shown in some detail in Fig. 4. A sodium vapor seal is provided by cooling the lower extremity of the top shield to approximately 150° F. The cooling is provided by organic shield coolant pipes laid in a lead layer in the thermal-shield region between the steel

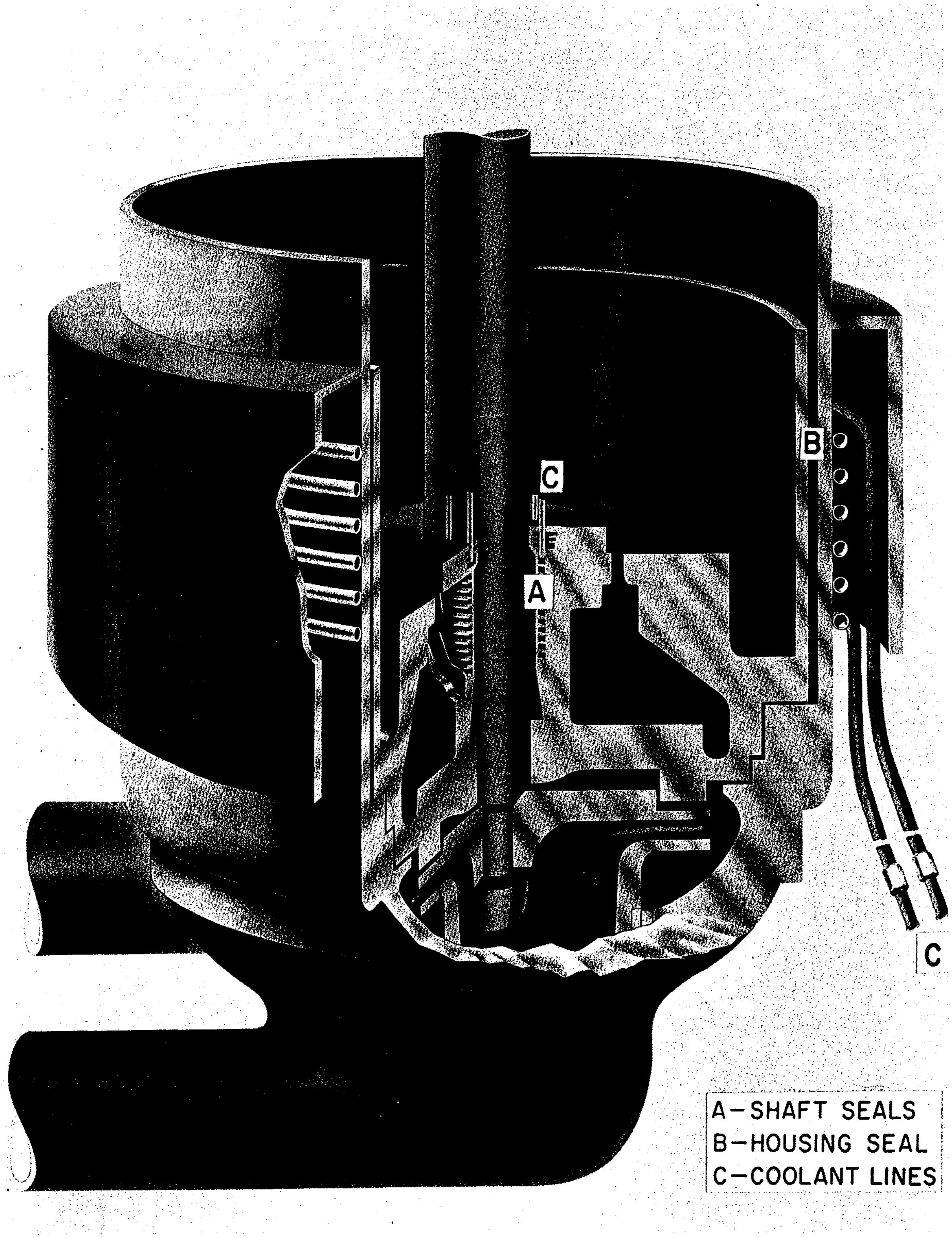

Fig. 2. Frozen-Sodium Pump Seal

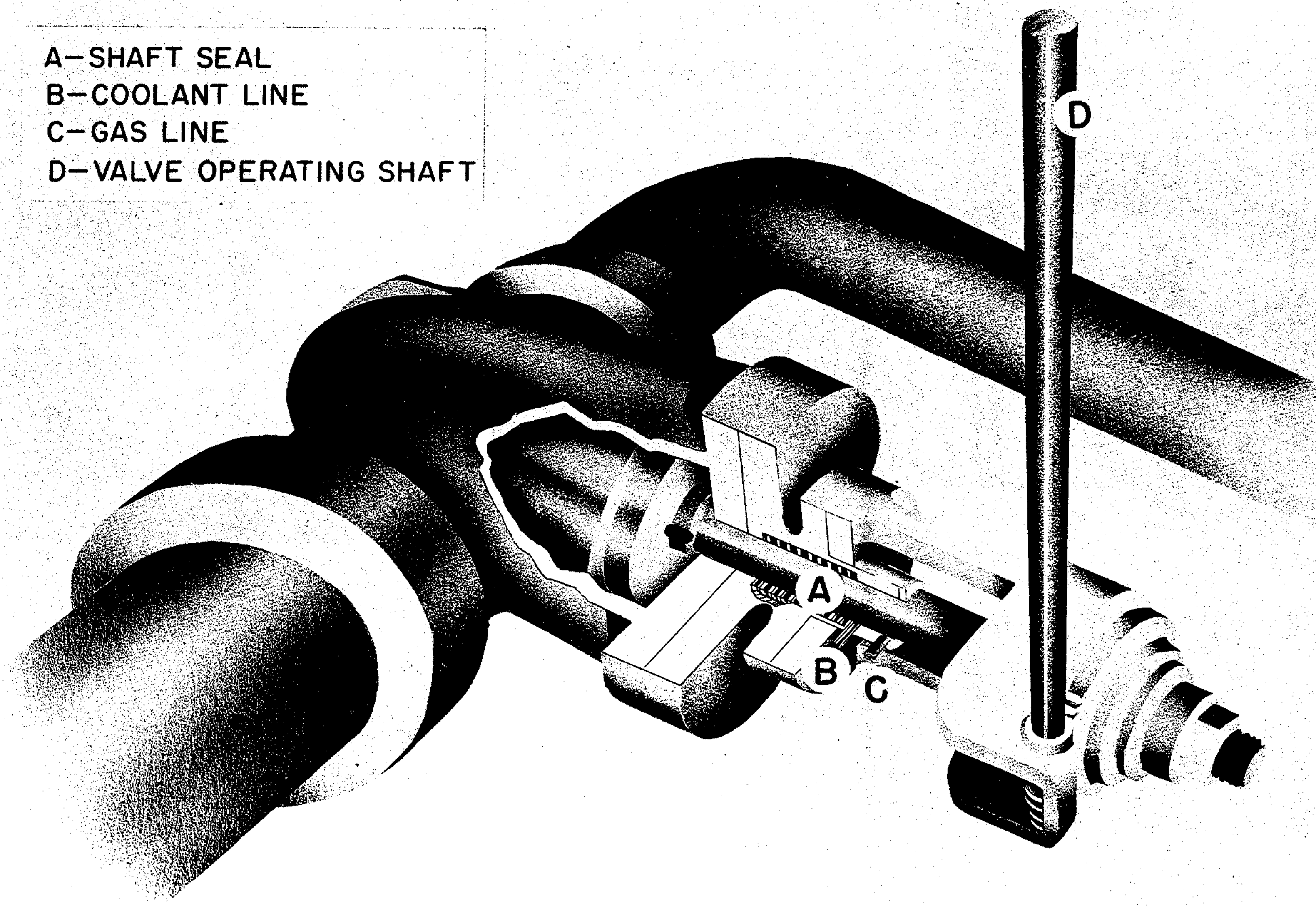

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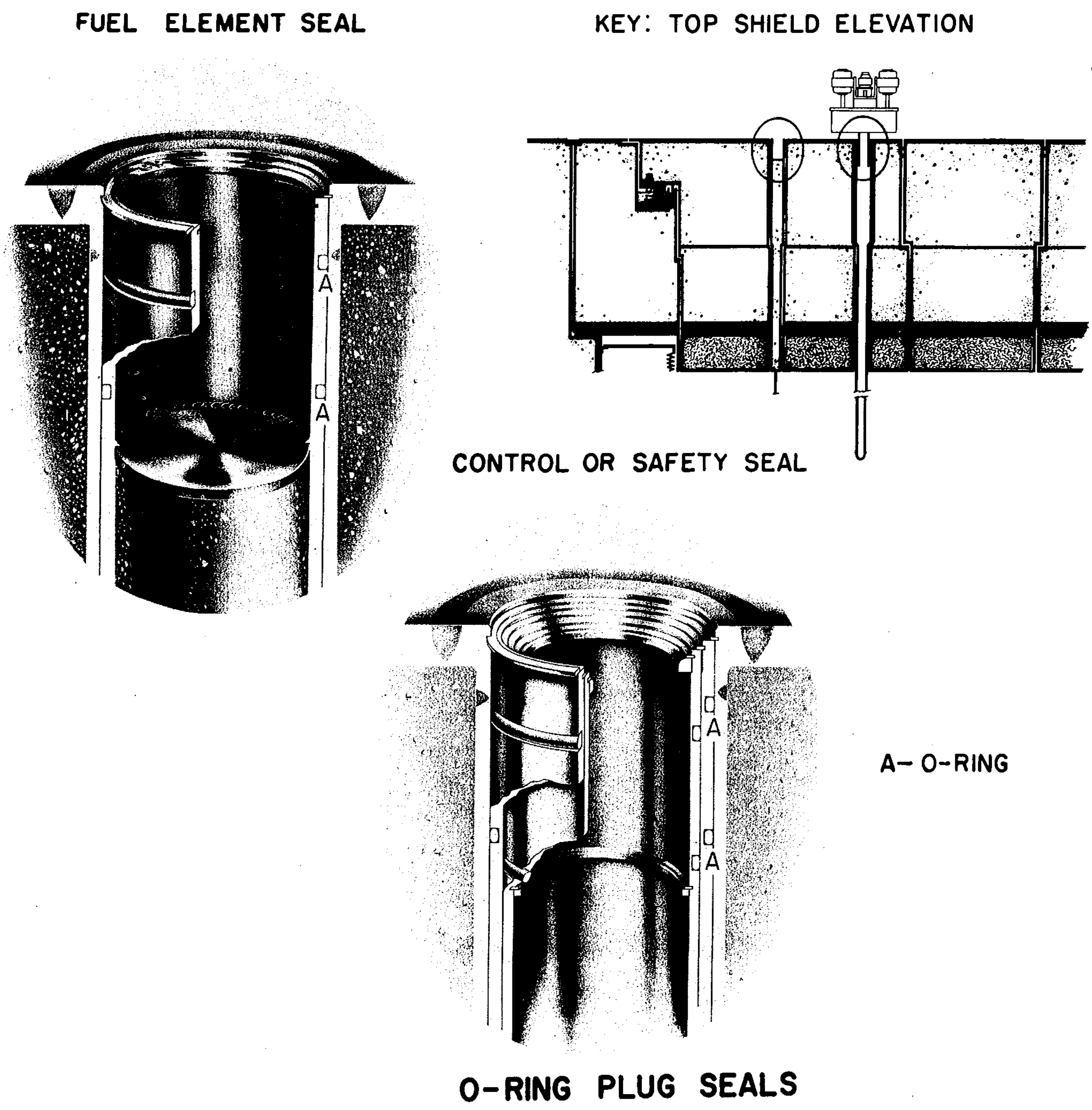
Fig. 3. Frozen Metal Valve Seal



S25-91

Fig. 4. Access-Plug Seals in Top Shield

thermal shield and the bottom plate of the shield structure. Seals for the helium gas are O-rings located at the top of the shield as shown for the fuel element at the left of Fig. 4, and for the control and safety units shown in the center. The inner set of O-rings, shown on the control and safety units, seals a helium atmosphere inside a steel thimble that houses the drive mechanism.

The gas seal, required around the outside of the top shield structure, is provided by freezing an alloy that melts at a low temperature in a special trough between the shield and the foundation. A lip, welded to the shield, fits into the trough. The alloy used expands during solidification to maintain the seal. Certain situations requiring repair in the reactor require rotation of the shield. For this purpose, heater units are built into the trough to melt the alloy. The trough-lip arrangement can provide a low-pressure seal during this repair period.

E. STUDY OF REACTOR INCIDENTS

Containment of radioactivity under normal conditions has been discussed in Section IV-D. The behavior of the reactor plant and possible dispersal of radioactivity during reactor incidents provides a final test in the evaluation of the reactor system.

There are a few incidents that can be considered where reflection on the design of the sodium system leads to the conclusion that design practice required for normal operation satisfactorily handles the incident. In other cases, considerable computation is required to judge the reactor behavior under postulated incidents.

Burning of primary sodium potentially releases radioactivity. Normally, this sodium is contained in a sealed envelope, surrounded by an inert atmosphere of nitrogen and protected by a five-foot thick layer of high-density concrete maintained near room temperature. In event of a rupture, the products of burning caused by the small amount of oxygen in the nitrogen are easily contained by the concrete. The reactor would be shut down, the ruptured loop isolated and as much sodium drained to the service system as possible; the remaining sodium would be frozen by heat removal through the shield cooling system. Repair work could not be started until the activity level is low, so the possibility of burning during repair does not involve radioactivity dispersal.

Other chemical reactions, with accompanying release of appreciable activity, are not possible. All materials used in the reactor and primary sodium systems are chemically compatible.

Three incidents are to be discussed that fall into the "compute" category. These are: (a) the interruption of core cooling, (b) the sudden reduction of average coolant temperature and (c) the unrestrained withdrawal of control rods. Detailed information applies to the SRE.

a. The interruption of core cooling can be caused by operator mistake, equipment failure, sabotage or "acts of God." Power, pump or pipe failure could result from these causes. The reactor will be shut down immediately following any indication of failure. Power and pump failure are the least serious since emergency provisions are made that result in immediate decrease in system operating temperatures. The auxiliary pumps are supplied with emergency power from batteries connected through a reversible motor-generator. A diesel-driven motor-generator is provided for long power outages. In event of pump failure in one loop, the parallel coolant loop will remove the shutdown, or "afterglow" power. Thermal convection in the primary and secondary loops provides additional cooling in the event of power or pump failure. Recent results from analog studies have shown that thermal convection in the main coolant loop results in an immediate decrease in coolant outlet temperatures.

A major break in the coolant lines of both loops would be an incident involving core cooling resulting in temperature increases in excess of operating conditions. It is assumed that the reactor is shut down and that the only reactor cooling is by the organic shield cooling system. The shield cooling system has not been designed to remove the afterglow heat from the reactor; consequently, the temperature of the reactor structure will be raised. The decay of afterglow power with time and the heat removal capacity of the shield cooling system are plotted in Fig. 5. These heat rates cross at 52 hours after shutdown; at this time, a maximum reactor temperature will have been reached. An energy balance of this situation is shown in Fig. 6. For a short time, most of the afterglow heat will go into raising the temperature of the reactor. At the 52 hour point, the reactor temperature is estimated to be at 1260° F, or 360° below the boiling temperature of sodium. About 168 hours after shutdown, the reactor has been returned to the average operating temperature. Detailed calculations have shown

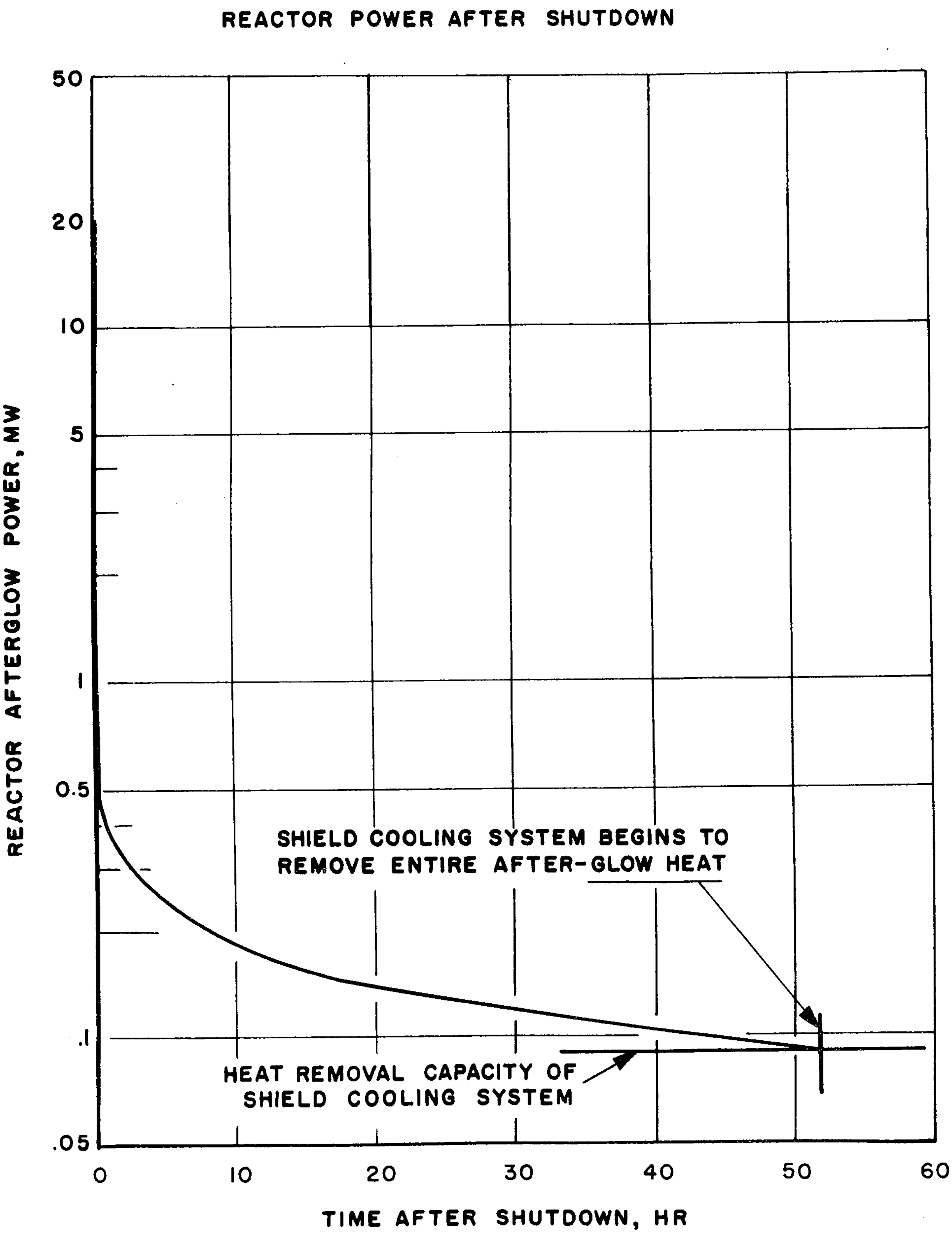


Fig. 5. Reactor Power After Shutdown

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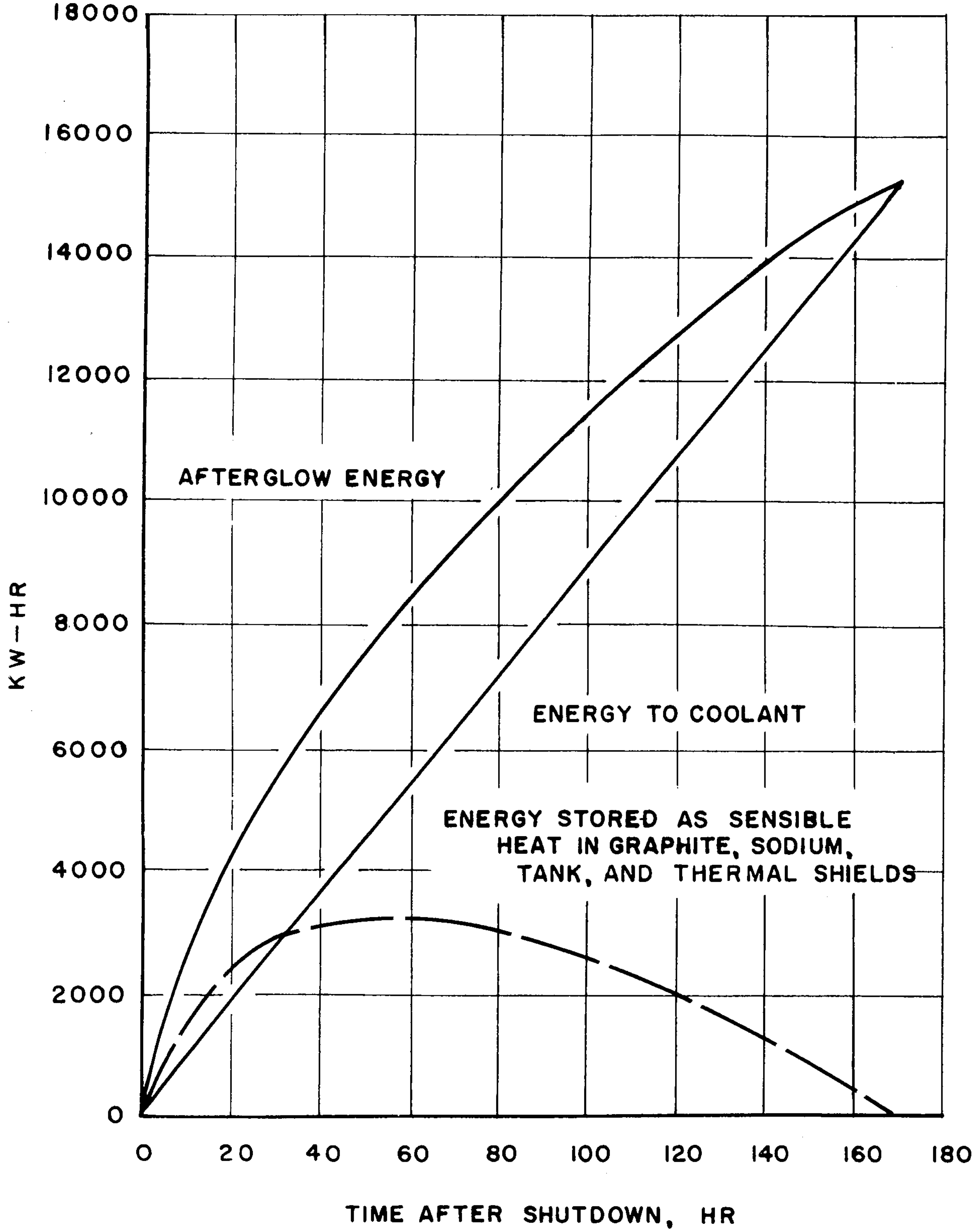

S25-100

Fig. 6. Afterglow Heat Dissipation as a Function of Time After Shutdown

that the whole structure of the reactor acts as an effective sink for the afterglow heat due to internal thermal convection circuits. Temperature differences within the core tank will be less than 50° F after the first few minutes of the incident under discussion.

b. The sudden reduction in the average coolant and fuel temperatures results in a sudden increase in reactivity, due to the negative coefficient of reactivity. Two causes of sudden reduction in temperature have been considered in the SRE studies. They are the unscheduled speed-up of the main sodium pumps and the sudden raising of the cooling rates in either the air-blast heat exchanger or the steam generator. These situations result in the most drastic reactivity changes at low power operation. The pump speed-up case is described here since it is considered to be both more likely and more severe. It is assumed that the reactor is operating at 5 per cent of full power and at the sodium steady-state temperature condition of 500° F inlet and 960° F outlet temperature (although present startup plans propose 10 to 15 per cent power before these temperature conditions are reached). The coolant is, therefore, flowing at 5 per cent of its full-power value. A cold sodium transient is introduced by assuming that the main pumps are suddenly turned to full capacity, causing the average sodium temperature to drop from 730° F to just a little over 500° F, or by nearly 230°. In a short time, the temperature in the fuel drops by about the same amount, resulting in an estimated reactivity increase of 0.173 dollar* and initiating a power surge in the reactor. Figure 7 shows the response of the SRE to this incident. The initial step at time zero is caused by prompt neutrons. Then the reactor goes on a slow period corresponding to the reactivity change. As the power increases, the fuel temperature increases and the negative temperature coefficient immediately begins to lower the reactivity. Since fuel and sodium temperatures lag behind power by only a few seconds, the original change in reactivity will be counteracted before full power is reached and the reactor will stabilize at a level somewhat below full power.

c. The most severe form of a nuclear runaway is considered to be caused by unrestrained control rod withdrawal during startup. In the study, it was assumed that withdrawal of all the control rods at the maximum speed (equivalent to 0.01 dollar per second) is maintained throughout the incident. Failure of all

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*One dollar = 0.75% Δk

instruments and safety devices was postulated, leaving the negative temperature coefficient of reactivity of the fuel rods, or melting of the rods, as the only effective means of stopping an otherwise unrestrained power rise. The object of study was to determine whether a runaway would lead to release of radioactive material from the reactor. The containment features described earlier in this paper are adequate if the incident does not lead to excessive pressures. Cladding disintegration and fuel melting will not cause chemical reactions with sodium and hence cannot result in a pressure increase. In fact, the early occurrence of either of these events is desirable because they cause dropping of fuel rods and reactor shutdown.

The following conditions were assumed for the study:

1. Reactor critical at zero power and full coolant flow at the start of the incident.
2. Full-speed withdrawal of all shim rods throughout the incident.
3. Reactor and coolant at the uniform startup temperature of 350° F.
4. Temperature of coolant entering reactor 350° F throughout the incident.

The results of the study are summarized graphically in Fig. 8, 9 and 10. The abscissas for these curves are "Time After Reactor Attains 1/10 Full Power." Evaluation of temperatures was not started until this time because temperature changes are negligible at lower power levels. The time to attain this power level is 99 seconds.

Figure 8 illustrates the temperature behavior of coolant and fuel rods during the runaway. Curves 1 and 2 show the fuel temperature at the center and at the surface of the rod at the point of maximum power in the central plane of the reactor. Curves 3 and 4 show temperatures at corresponding points at the top of the same central fuel rod. Curve 5 shows the temperature of the sodium leaving the core. The temperature at the center of the rod is rising much faster than the coolant temperature or the temperature at the rod surface. This shows that the temperatures and power are far from equilibrium. A slight extrapolation of the curves shows that rod temperature will reach the uranium melting point of 2070° F at about 9.5 seconds with a corresponding sodium temperature of about 1000° F. Two factors cause the marked temperature lag. One is the fact that high heat-transfer rates into the coolant must wait until a sufficient temperature gradient

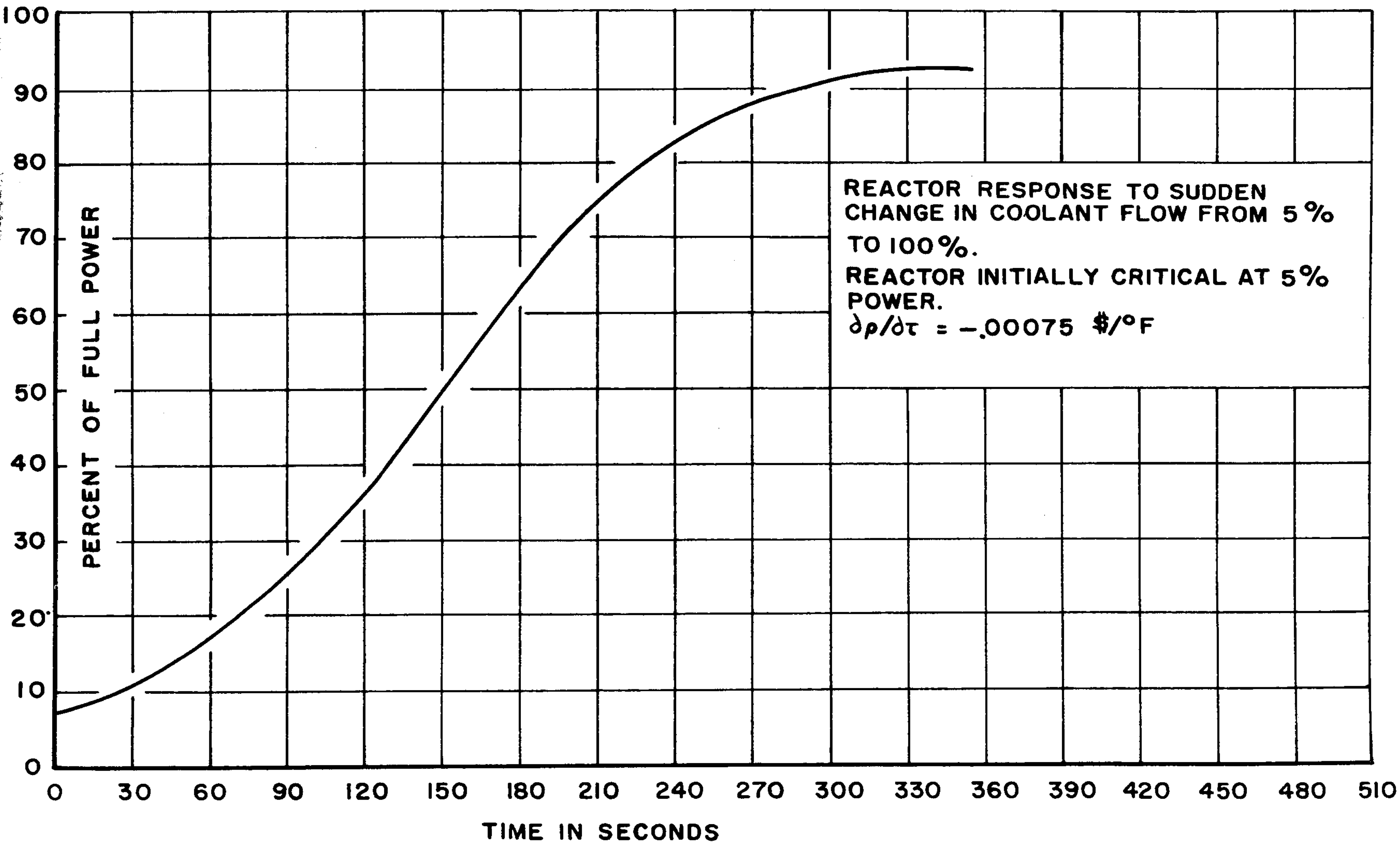

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Fig. 7. Reactor Response to Sudden Reduction in Average Temperature

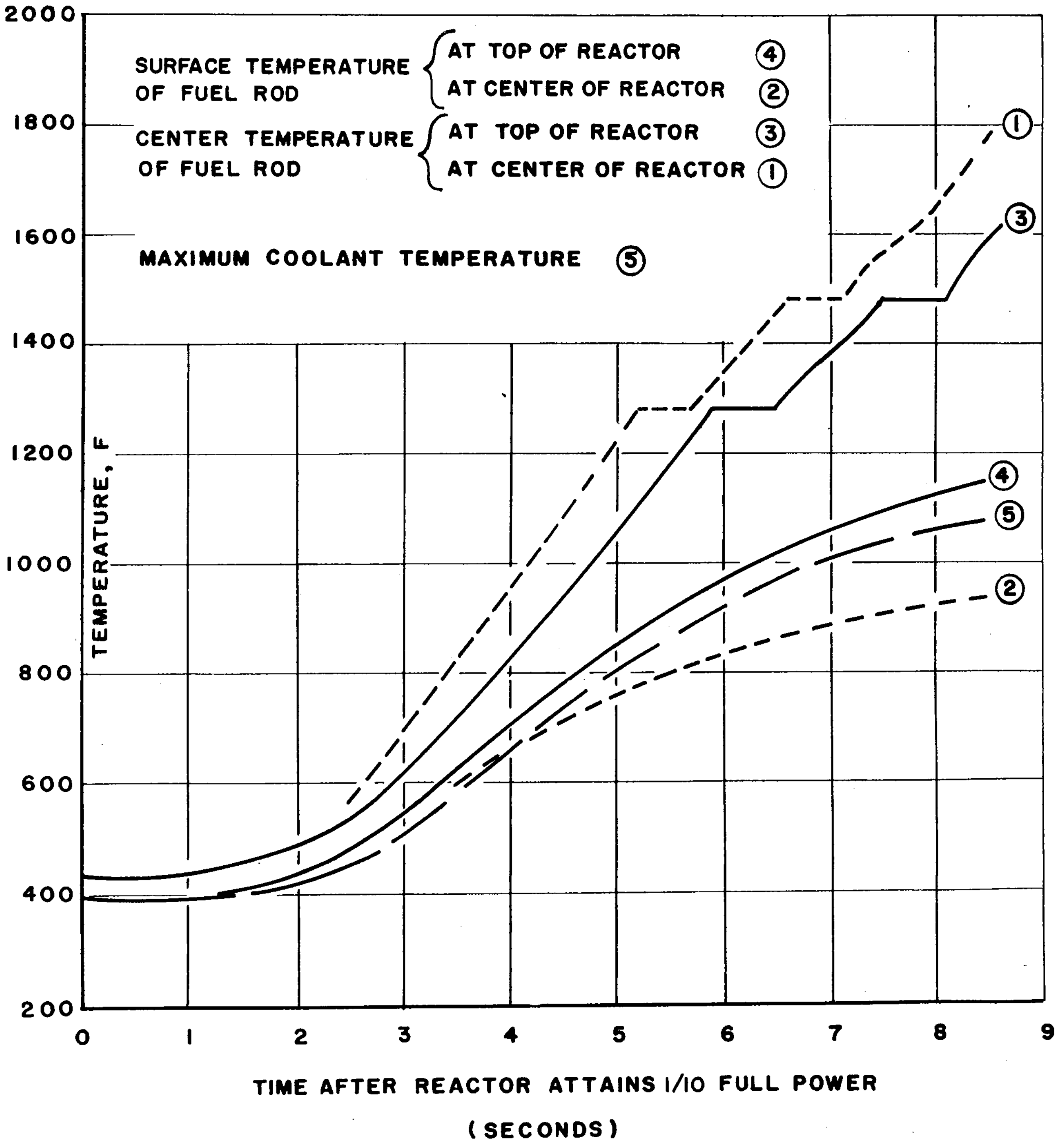


Fig. 8. Temperature Distribution in the Central Fuel Channel

has been built-up in the rods. For rods of this size, it takes several seconds for a heat wave to travel from the interior to the surface of the rod. The second factor is the loss of heat by the coolant to the moderator which is a relatively large mass with a high thermal conductivity. The plateaus in Curves 1 and 3 represent endothermic reactions at the α-β and β-γ phase changes. If this heat had been absorbed as sensible heat, it would have increased the rod temperature by approximately 350°. Since this is equivalent to a decrease in reactivity of 0.262 dollar, it is apparent that the phase transformation can accelerate the runaway.

Figure 9 shows the effective reactivity during the runaway. The contribution of shim rod reactivity is plotted at the top, showing a straight line that has been increasing at 0.01 dollar per second since the start of the incident. The dashed curve at the bottom of the figure shows the effect of increase in fuel temperature; the curve in the center is the sum of these other two curves. The leveling in reactivity at about 6 seconds coincides with the time the rods begin passing through the uranium phase changes.

Figure 10 shows the effect of the runaway on reactor power. 3.5 seconds after the reactor has reached 1/10 full design power, the negative temperature coefficient tends to level the power curve. After approximately 6 seconds, the phase changes result in a decrease in temperature rise and a corresponding increase in reactor power. From this point on, the power will rise until some fraction of the fuel rods drop and shut down the reactor. The two most likely mechanisms that will cause the fuel rods to drop are (a) melting of the fuel slugs and the formation of a uranium-iron eutectic which will weaken the stainless steel containing tube, or (b) the melting of the fuel slugs and a resulting build-up of pressure in the NaK bonding fluid which will cause a steel tube rupture. These curves indicate fuel temperatures that will cause the rod drop before sodium reaches the boiling point. However, if some local boiling occurs first, the sodium vapor will condense in the 6-foot deep pool of subcooled sodium over the reactor and no appreciable pressure build-up will occur.

The incidents described lead to a moderate temperature rise if core cooling is completely interrupted, a power surge from 1/20 power to approximately full power if pumps are suddenly speeded up during startup, and fuel-rod melting if control rods are withdrawn continuously at maximum rate during startup. None of these incidents, or others discussed, cause dispersal of radioactivity from the underground cells provided for the reactor and the primary cooling system.

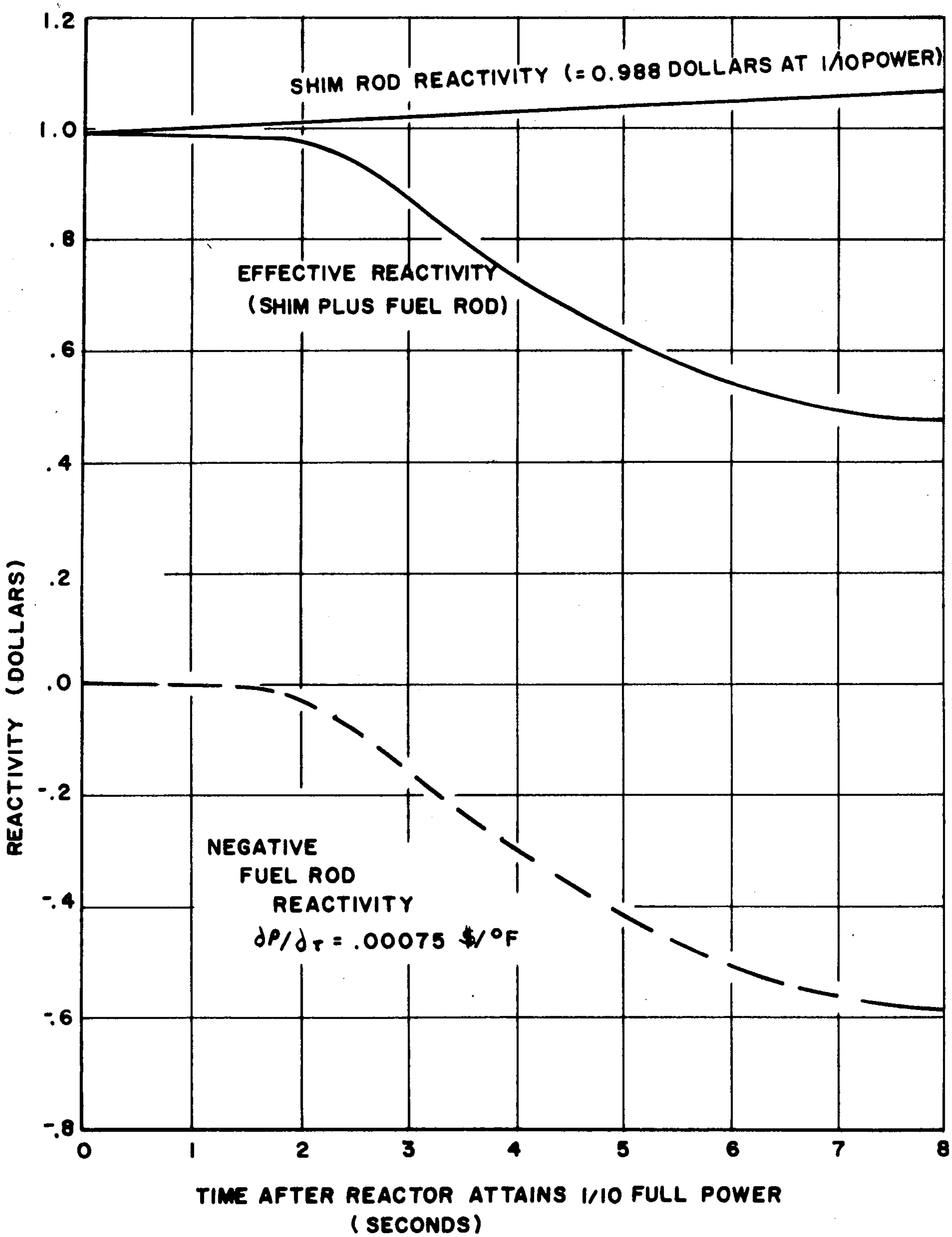
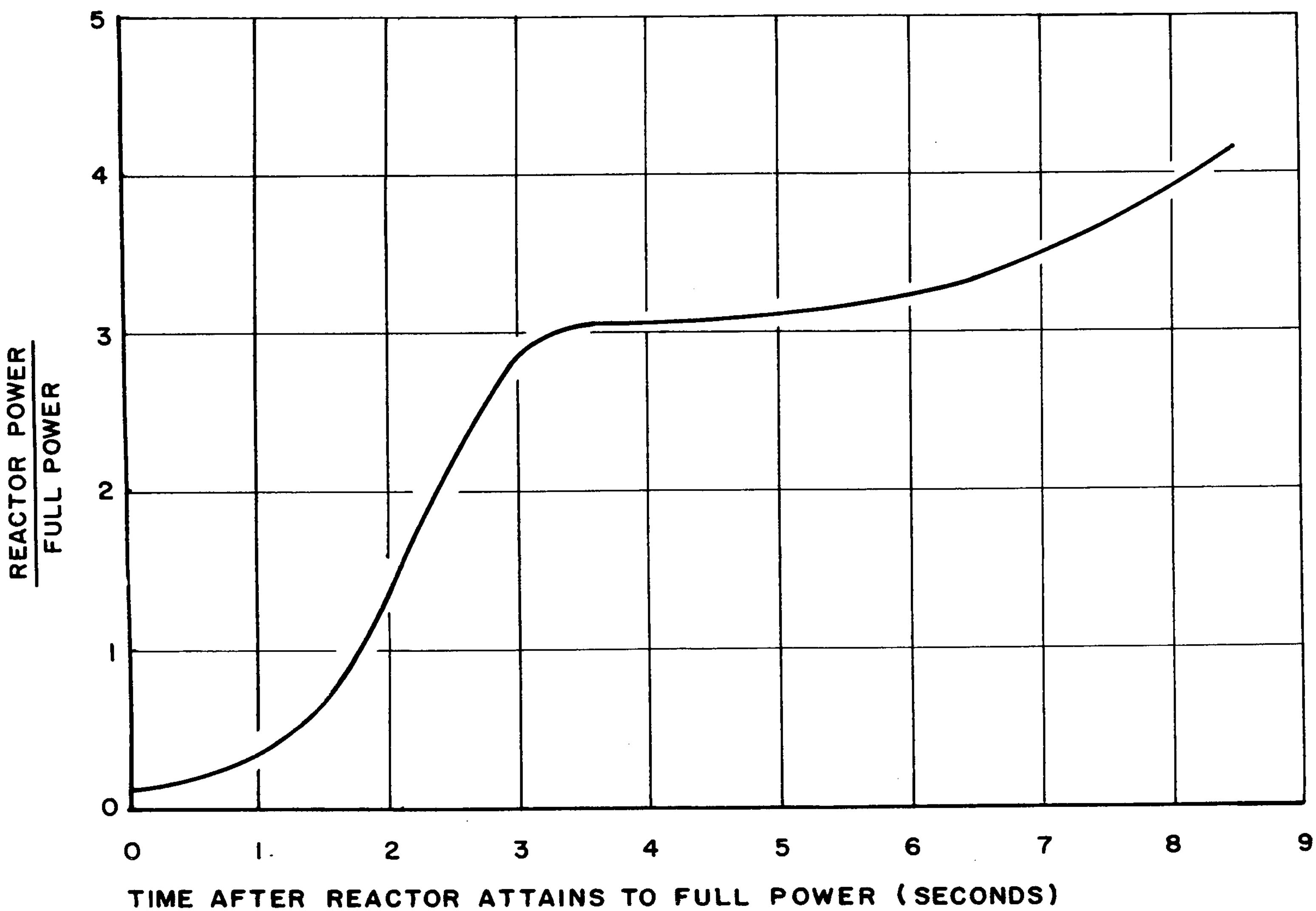


Fig. 9. Variation of Reactivity During Reactor Incident S25-103



S25-97

Fig. 10. Increase in Reactor Power During Incident