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**SYMPOSIUM
ON
SODIUM
REACTORS
TECHNOLOGY**

held in
Lincoln, Nebraska

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Preface

The Sodium Reactors Technology Symposium was held May 24 and 25, 1961, in Lincoln, Nebraska. The meeting was cosponsored by the U. S. Atomic Energy Commission and the Consumers Public Power District of Nebraska as part of the education and promotional function of the AEC Division of Reactor Development.

The symposium provided the occasion for various organizations and industry now working with or closely associated with reactor projects in which sodium is used as a heat-transfer medium to exchange current information on common problems. In addition to the status type of reports, the panel session in particular projected future thinking of the represented organizations. The differences of opinion expressed during the panel session in regard to the most promising fuel for sodium-cooled reactors is well worth reviewing and analyzing.

Full international participation was handicapped by the relatively short time allowed for announcements and invitations, as indicated by the list of attendees.

This document comprises the conference proceedings and includes the papers in the same order as listed on the agenda. The postmeeting answers to the questions submitted by the attendees and the list of attendees are located in the appendix.

In most cases the authors' prepared papers were used for this document. Those papers taken from the oral presentation by the court reporter, Albert Morwitz, were submitted to the authors for editing and approval. The program chairman has attempted to eliminate mistakes and misstatements in the process of editing, with particular attention to the transcribed discussion portion of the panel session.

Grateful acknowledgment is made of the services of the session chairmen, G. W. Wensch and C. A. Pursel; of the dinner chairman, T. A. Nemzek; and of the program committee, D. P. Rudolph and L. A. Redecke. Special recognition is given to C. D. Sayre for the arrangement of the conference and liaison with various organizations in Lincoln, Nebraska.

THOMAS P. HECKMAN*
Program Chairman

*U. S. Atomic Energy Commission, Chicago Operations Office.

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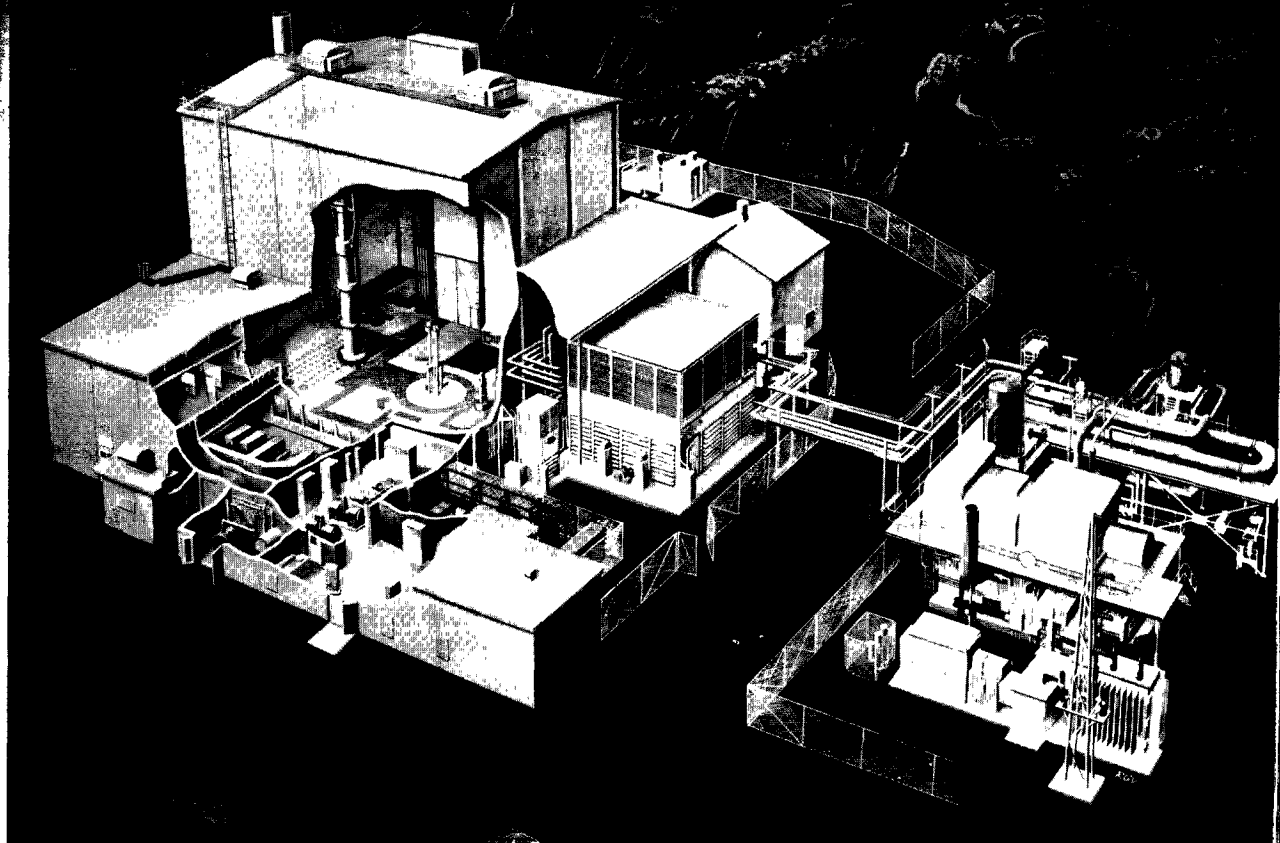
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Sodium Reactor Experiment

L. E. GLASGOW

Atomics International

The Sodium Reactor Experiment (SRE) was built and is being operated for the Atomic Energy Commission by Atomics International, a Division of North American Aviation, Inc. The design of the plant was begun in July 1954 and the construction was completed in March 1957.

This plant is a 20-Mw(t) sodium-cooled graphite-moderated low-enrichment thermal reactor. It is of the tank type with a graphite moderator contained in individual hexagonally shaped 0.035-in.-thick zirconium moderator cans. The cans are 11 in. across the flats and 10 ft long. The 0.17-in. space between the cans is filled with sodium. Forty-three of these assemblies contain 2.8-in.-diameter axial tubes to accommodate the fuel clusters. The active core is a 6- by 6-ft right-circular cylinder. The core tank is 19 ft high and 11 ft in diameter. The two primary (radioactive) and two secondary (nonradioactive) mechanically pumped cooling circuits are external to the core vessel. The primary and secondary loops are thermally coupled by means of sodium-to-sodium shell-and-tube heat exchangers. The design temperatures for the primary loops are 500°F for the reactor inlet and 950°F for the reactor outlet. The secondary-loops design temperatures are 460 and 900°F. Flow in the main loops is 1100 gal/min.

The principal advantages of sodium-cooled systems are (1) a safe, simple, easily contained reactor system operating at pressures less than 1 psig, (2) thermodynamic cycle efficiencies in excess of 42 per cent, (3) ease of operation and control, (4) simple, direct-contact maintenance, and (5) complete chemical compatibility of the reactor coolant and the materials of construction.

The main purpose of the SRE is to test the technical feasibility of a large sodium-cooled reactor system operating at high temperature. The technical evaluation is based on a broad range of operating experience, both scheduled and unscheduled. This experience is supple-

mented with specific engineering tests aimed at defining any weakness or uncertainty in the plant indicated by analysis or plant performance characteristics.

The over-all technical feasibility of the concept has been established by more than three years of operations and tests. During this time this relatively small [6.0 Mw(e)] experimental plant has, in conjunction with the Southern California Edison Experimental Station, produced 15,331,050 kw-hr of electrical energy for the Southern California grid. During fiscal year 1959, when the experimental program was primarily limited to biweekly fuel inspection, the plant achieved a 75 per cent availability factor.

The plant has operated for extended periods with an outlet sodium temperature of 1000°F; and, during one test run, it operated at 1065°F in order to test the characteristics of the steam generator at a steam temperature of 1000°F.

The experimental high-temperature-reactor fuel program requires that the reactor be frequently shut down for fuel-element removal and examination. This activity, combined with experimental and normal scrams, has resulted in the equivalent of 20 years of normal start-up experience. As a result of this experience, normal reactor start-up time following a scram has been reduced from 2 hr to 20 min. Stresses in the core and core tank due to thermal transients are avoided by controlling the flow rate and maintaining the critical system temperatures constant. The flow is controlled by means of eddy-current brakes in both the primary and secondary systems.

Tests to determine the ability of the reactor to follow load show that the reactor can easily follow load swings at the rate of 20 per cent per minute on manual control. The limit on the rate of power change is determined by the steam turbine rather than by the reactor.

The steady-state operating stability of the SRE was demonstrated by noting during a 144-hr period that integrating timers on the control rod showed only 3.5 min of control-rod movement. The dynamic stability of the reactor at all frequencies up to 20 cycles/sec has been shown by pile oscillator tests made at power. The power coefficient for Core I is

$$- \frac{3.5}{N_0} \times 10^{-4} \left(\frac{\Delta K}{K} \right) / \text{Mw}$$

at a constant temperature of 335°F across the reactor, where N_0 is the power level at which the coefficient is measured.

Measurements of the radial power distribution in the core indicate a ratio of outer channels to inner channels of 0.70 as compared to a predicted value of 0.50. This provides for better utilization of the fuel and will in future reactors, where the fuel temperatures are expected to be limiting, permit additional total power from a given core.

Our experience with specific components has been as follows.

Fuel Elements

The SRE first-core fuel elements have been structurally reliable. Irradiation of about 1000 Mwd/ton produced swelling that increased the diameter of the rods about 3 to 4 mils. This simple alpha-rolled beta-heat-treated uranium fuel would appear to have a life no better than about 2000 Mwd/ton. The second-core fuel, which is in the reactor now, is a thorium - 7.6 wt. % uranium alloy; and it is expected to last to approximately 5000 Mwd/ton. Advanced fuels of high thermal conductivity and high temperature stability such as uranium carbide are indicated for advanced sodium reactors. These fuels should be good to at least 25,000 Mwd/ton. Central temperature of the slugs has proved to be ~100°F lower than expected. At 1000°F reactor outlet temperature and 20 Mw(t), maximum slug temperature is about 1100°F.

Fuel-handling Experience

Our fuel-handling experience is equivalent to approximately 20 years of normal operation and has led to the design and construction of an improved fuel-handling cask. The original fuel-

handling cask, although adequate for the purpose, required excessive maintenance and was a single-purpose device. A new cask which eliminates these difficulties and provides greater flexibility for a variety of shielding tasks has been built and operated. The principal design changes include complete streamlining of the cask interior to prevent fuel hang-up, the elimination of sliding O-ring seals where the surfaces can become contaminated with sodium, and provisions for easily disassembling the cask in sections for maintenance and decontamination.

The original fuel-washing system, which utilized a direct water wash, worked perfectly the first 1100 times, but during the 1101 wash a pressure excursion blew the shielding plug from the wash cell. Subsequent investigations disclosed that the 2.5-in.-diameter 6-ft-long cylindrical hold-down tube, a part of the fuel assembly, failed to drain clear of sodium. This failure introduced about ten times more sodium than the cell was designed to handle. The wash cells have been redesigned to utilize steam as the washing medium.

Main Intermediate Heat Exchanger

An unexpected result from the first power run was the 50 per cent excess log mean temperature difference observed on the main intermediate heat exchanger. Between power runs 60 thermocouples were distributed over the length and around the girth of the heat-exchanger shell. The temperature curve rose uniformly at the straight portions of the exchanger but was horizontal around the bend, indicating that no heat transfer is occurring in that section. Examination of the construction photographs disclosed a wide gap between the tube bundle and the shell; and, since this section is not baffled, the fluid simply by-passed the tubes. The loss of this amount of heat-transfer area accounts for the high log mean temperature difference. Transient data taken on this equipment disclosed that rather complete stratification occurs on the shell side of the exchanger. A replacement for the present heat exchanger, embodying improvements based on these studies, is on hand. In the interim the present exchanger will be operated without modification.

Steam Generator

During the initial operation of the once-through sodium steam generator, it was noted that the upper tubes were running approximately 200°F hotter than the lower tubes. This temperature difference is initiated by a small amount of temperature stratification in the shell-side sodium, which causes the upper tubes to run slightly hotter than the lower tubes. The effect is amplified because the effective superheater length of the upper tubes is increased, which in turn increases the tube pressure drop and decreases the water flow. The process continues until the flow to the upper tubes is virtually cut off. The opposite sequence of events occurs in the lower tubes. This difficulty was resolved by the boiler manufacturer and the Southern California Edison Company with the addition of orifices at the water-tube sheet. The orifices provide enough self-regulation to limit the top-to-bottom temperature difference to about 50°F. The steam generator has performed perfectly since the modification.

Control and Safety Rods

The control- and safety-rod performance has been excellent. The maximum dimensional change observed on the control rods was 0.006-in. diametrical change after 2140 Mwd (7.5 per cent B¹⁰ burn-up). Maintenance on the rod drives has been extremely low, requiring a brake replacement on one of the four rod drives. There have been no difficulties with the four safety rods.

No difficulty has been experienced in maintaining the O₂ concentration below 10 ppm by means of the cold traps. As the cold traps become filled they are simply replaced. This has been done about eight times and without difficulty.

Sodium Valves

The SRE is equipped with two kinds of sodium valves. One type uses a frozen-sodium shaft seal that is similar to the pump shaft seal. This type has operated with 100 per cent reliability. The other type is a bellows-sealed valve. Eight of these in the service lines have had to be replaced owing to damage to the bellows resulting from solid-sodium extrusion through the pipe during preheating operations. We have never had a failure of a bellows-seal valve on lines that remain heated. All the bellows-seal valves that were in the main primary loops have been eliminated.

Sodium Pumps

The SRE sodium pumps are conventional centrifugal pumps with modified shaft seals. The shafts are sealed against sodium leakage by freezing off a 20-mil annulus of sodium around the shaft; as the shaft turns it continuously shears the solid sodium in the annulus. The total integrated time logged on all four pumps amounts to 75,000 hr with an over-all reliability of 95 per cent. Although all four pumps are identical, our experience with them has been quite varied. Any given pump will run completely trouble-free for long periods (~10,000 hr) while one or more of the remaining pumps is experiencing shaft-seal difficulties.

The several trouble areas with the shaft seal have been shaft binding, solid-sodium extrusion, and gas in-leakage into the sodium system. The fact that the pumps can perform perfectly on occasion suggests that there are some not yet understood details that need to be remedied. The 95 per cent pump reliability is not adequate for power-plant operation, especially in the primary system where there is an approximately 4- to 10-day delay to gain access to the pumps for maintenance. Recently designed shielded handling equipment can reduce this waiting period to approximately one day, which is still excessive. For future reactors the reliability of the freeze-seal type pump must be greatly improved or a different pump type, such as a free-surface pump or an electromagnetic pump, is strongly indicated.

It was the failure of the shaft seal on the main primary pump which introduced tetralin into the main primary sodium.

Core Maintenance Experience

The tetralin leak that developed in the main primary pump added approximately 4 gal of tetralin to the main primary sodium. The tetralin decomposed into hydrogen and carbon. The carbonaceous material lodged in the fuel channels, partially obstructed the flow, and provided local coolant by-passing of the fuel. This resulted in overheating and damaging the fuel elements to the extent that they swelled and stuck in the channels. Of the 43 fuel elements in the reactor, 13 were damaged. The upper halves of 10 of the broken elements were removed by a modified standard procedure utilizing the new fuel-handling machine. A special cylindrical probing tool was devised to explore the possibility of grappling and removing the remaining fuel from the 10 channels involved. These probings revealed that the fuel had distorted sufficiently to completely take up the channel clearance (nominally 29 mils), and they could not be removed without damaging the moderator can. Therefore it was decided to remove the lower part of the fuel element and moderator assembly as a single unit. The hanger rods and shield plugs of the two stuck elements (R-24, R-76) were separated from their respective fuel clusters by rotating the shield plugs. This action sheared the hanger rod at the shear pin that is located near the top of the moderator cans. Following the removal of the 30 sound fuel elements, one complete but defective element, and the upper halves of the 10 parted elements, the sodium was drained from the core, and electrical heaters were added to maintain the core at 350°F. The helium cover gas was replaced with argon.

At this time inspection of the core was begun. The radiation level at the bottom of the loading face shield was measured and found to be 43 r/hr. One of the 3-in. plugs in the top shield was removed by using the fuel-handling cask, and a glass window $\frac{1}{2}$ in. thick was in-

stalled. The radiation level above the glass was about 50 mr/hr. The contamination in the cover gas was determined to be about $4 \times 10^{-3} \mu\text{c}/\text{cm}^3$ of Kr^{85} in helium; replacement of argon reduced activity to about $7 \times 10^{-4} \mu\text{c}/\text{cm}^3$. The condition of the top of the core was examined with the aid of special viewing devices constructed for this purpose. This examination revealed a layer of black, flocculent, carbonaceous-appearing material on the tops of the cans. On top of this layer were about 82 separate fuel slugs scattered at random, some bits of fuel cladding, and several pieces of wire wrap. This material had fallen on top of the core during the removal of damaged fuel clusters. Location of this debris was mapped, and preparations were made for its removal.

The fuel slugs and cladding fragments were picked up by articulated grapples fitted through the 3-in. fuel-plug openings in the top shield and placed in a 2.5-in.-diameter 3-in.-long bucket suspended from a shield plug in an adjacent corner channel. The buckets were removed by means of the fuel-handling machine, canned, and stored with the other damaged fuel. Viewing was accomplished by means of a "corescope," which was simply a 3-in.-diameter 10-ft-long tube with a lens sealed in the bottom end and a telescope mounted in the upper end. Lighting was provided by means of mercury-vapor lamps suspended from shield plugs. Removal of most of the flocculent carbonaceous material was accomplished by vacuuming the tops of the cans. The vacuum cleaner consisted of a commercially available blower, a nozzle, glass-fiber flexible hose, filter and settling tank, and shielding. The filtered gas was discharged back to the reactor. One filter change was needed to keep the radiation level below 90 r/hr at the surface of the filter. Roughly 3 lb of carbon, estimated from the associated radioactivity, were removed from the system by this means.

Preparations for the moderator can removal through the 40-in. plugs in the loading face shield went on concurrently with the core and system cleanup. Although the moderator replacement equipment had been largely designed and most of the parts manufactured, it was necessary to complete the assemblies and thoroughly check them out. In addition, means were developed to detect sodium in-leakage to individual moderator cans to determine which cans had failed. The two most useful techniques were (1) use of an induction probe lowered through a thimble placed in the moderator-can fuel channel to detect the electrical resistance change expected in sodium-saturated graphite and (2) a probe of the can heights to detect the 1 to 1.5 per cent dilation of graphite accompanying sodium absorption. Both methods indicated leaks in cans R-10, which had lengthened $\frac{7}{8}$ in., and R-42, which had elongated $1\frac{1}{2}$ in.

The moderator-can encapsulating station was equipped for viewing the cans and obtaining zirconium samples. After the cans were inspected, they were sealed in argon-filled metal storage capsules. The capsules were then transported to the solid-waste storage building.

All the equipment used was gas tight; seals between the various assemblies were checked for leak tightness at each step. All seals were double; spaces between the seals were purged to the reactor radioactive vent system. These precautions prevented escape of radioactivity from the core into the work area.

During the moderator-can removal operation, all the cans were inspected for carbon deposit. None was observed on the sides of the removed cans or on the six adjacent can panels. A light carbonaceous deposit was noted on the bottom of some cans. Observations of the top of the gird plate through the void created by the removed can disclosed a very thin and somewhat mottled dark coating. The amount observed did not warrant the additional manipulative and exposure hazard that would have been necessary for its removal.

By means of a special draining device, the sodium level in the lower plenum was reduced to approximately $\frac{1}{4}$ in. in the deepest area. This left several mirror-surfaced sodium puddles and exposed about one-half of the core tank bottom which was perfectly clean and bright. Two fuel slugs were observed on the floor of the tank; these were removed.

After it was established that the system was reasonably clean, it was refilled with sodium, heated to 600°F, and sodium circulated for seven days to check for any additional leaky cans. Three were found: R-32, R-44, and R-45. These were replaced. It was subsequently established that two of the three cans (R-32 and R-44) failed because of mechanical damage to the short unprotected zirconium pump-out tubes during fuel-slug grappling operations. The third can (R-45) was not examined in sufficient detail to establish the point of sodium entry. This made a total

of 16 cans replaced. A second sodium circulation test and height probe showed that all the cans were sound. The critical loading was conducted, the cans were again tested and found to be intact, thus validating critical-loading measurements. Any future can failures can be detected by changes in reactivity.

It is interesting to note that it took approximately one week to replace the first moderator can whereas the last can was replaced in a little less than 2 hr.

It was observed that carbonaceous material taken from the core had a specific activity about 10^5 times as great as that of the sodium. This phenomenon provides a ready means of locating carbon deposits in the piping and equipment. Measurements made in a 10-in.-diameter thimble that penetrates the main gallery showed that the gamma field at the main intermediate heat exchanger decreased from 2.5 to 1.8 r/hr while the cold trap increased from 68 to 80 r/hr over the same time interval (August 18 to 25, 1959), indicating that the activity was being transferred from the piping into the cold trap. After the gallery shield blocks were removed, a detailed survey was made of the piping. Over most of the pipe runs, the activity level was in the range of 0.1 to 0.2 r/hr. At several specific locations the radiation levels were somewhat higher. A sodium circulation test was performed to reduce the radiation levels. The resulting reduction in radiation levels at the control points were as follows: hot-trap economizer, 7.0 to 0.95 r/hr; main primary pump discharge, 18 to 7.0 r/hr; and bottom of the pump case, 0.4 to 0.16 r/hr. At the same time the radiation level of the replaced cold trap increased from 2 to 11 r/hr, again indicating the transport of carbon from the piping to the cold trap.

Piping System

Maintenance or modification of the sodium piping has proved to be remarkably straightforward, simple, and safe. On the primary loop the sodium activity is allowed to decay for about 10 days. The system is then cooled permitting the sodium to solidify in the pipe. A conventional pipe cutter is used to cut through the stainless-steel piping. The sodium is then removed from the piping, and the sodium in the pipe is dug back about 6 in. with the use of a spatula. The new part is then welded in the line, and the weld is checked for leaks with helium and x-rayed. There has been no piping maintenance on the primary sodium system, only modifications. On the secondary system four valves have been replaced and the system has been modified from time to time for experimental purposes.

Current Activities

We are at present studying the dynamic characteristics of the second SRE core at low power. At the completion of the studies, the reactor temperature and power will be raised, and hot trapping of the sodium to remove dissolved carbon will begin. Full power will not be achieved until after the carbon content is reduced below 18 ppm.

Conclusions

The following conclusions have been reached:

1. The technical feasibility of the sodium-graphite reactor concept has been amply demonstrated up to 1000°F.
2. Maintenance work, whether it involves replacing a primary system valve or a part of the core, can be accomplished in a simple, direct, and safe manner.
3. The containment features of the SRE primary system were proved by retaining all the fission fragments released from the twelve severely damaged fuel elements.
4. With the exception of the inert gases, all the fission fragments remained with the sodium. Ninety-five per cent of these have been removed from the sodium by cold trapping.
5. The components that were developed for the SRE have proved to be functionally satisfactory. Improvements in the reliability aspects of the SRE pumps and valves should provide a plant availability factor in excess of 90 per cent.

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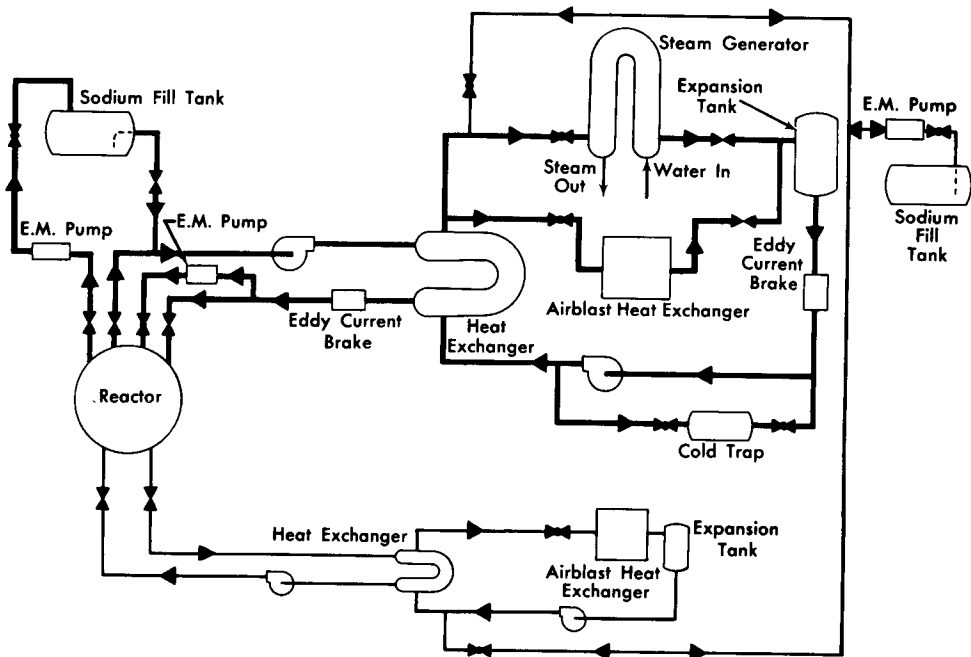


Figure 1

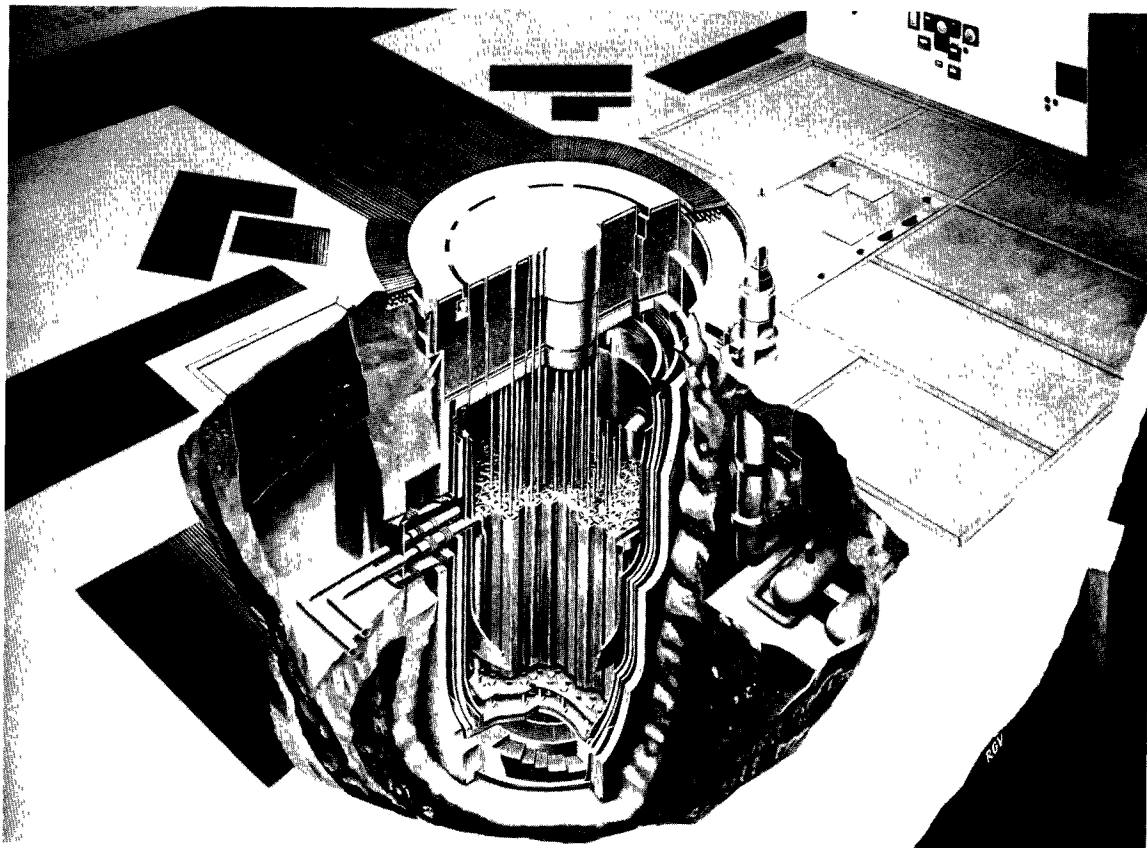


Figure 2

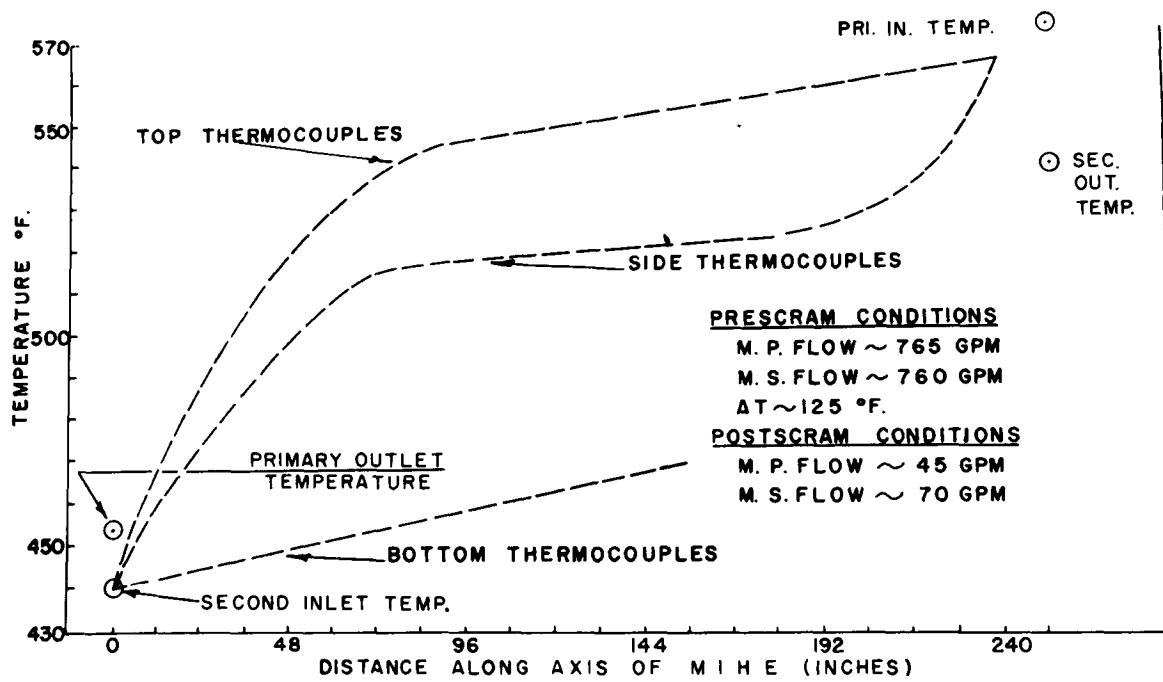


Fig. 3—MIHE shell temperatures 18 min after scram.

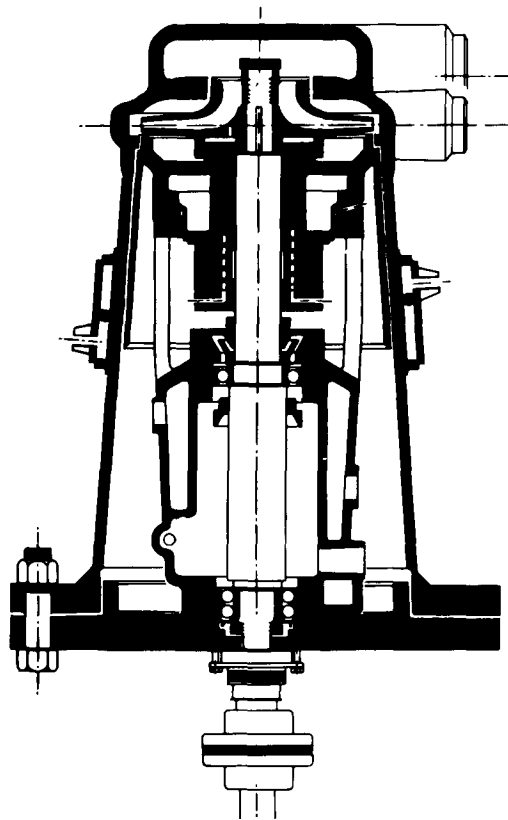


Fig. 4—Frozen seal. Liquid sodium at 1200°F. 30 hp, 1200 rpm.

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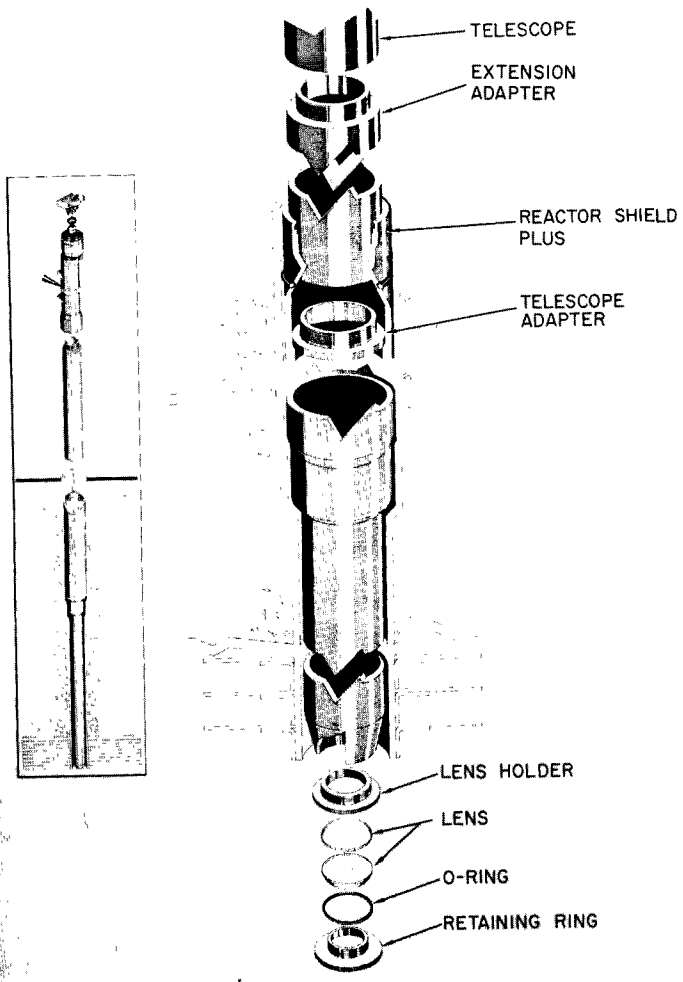


Fig. 5—Mark II corescope.

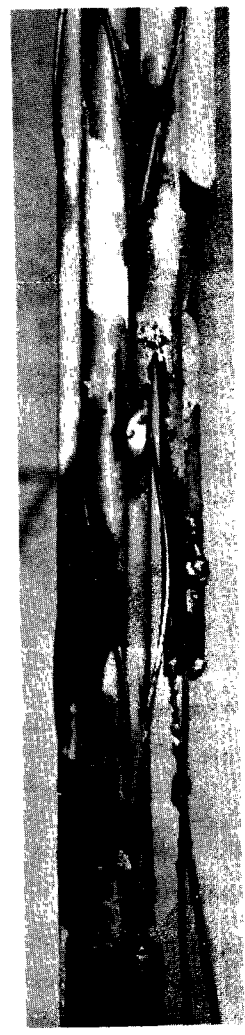


Fig. 6—Bottom of R-55 fuel element.

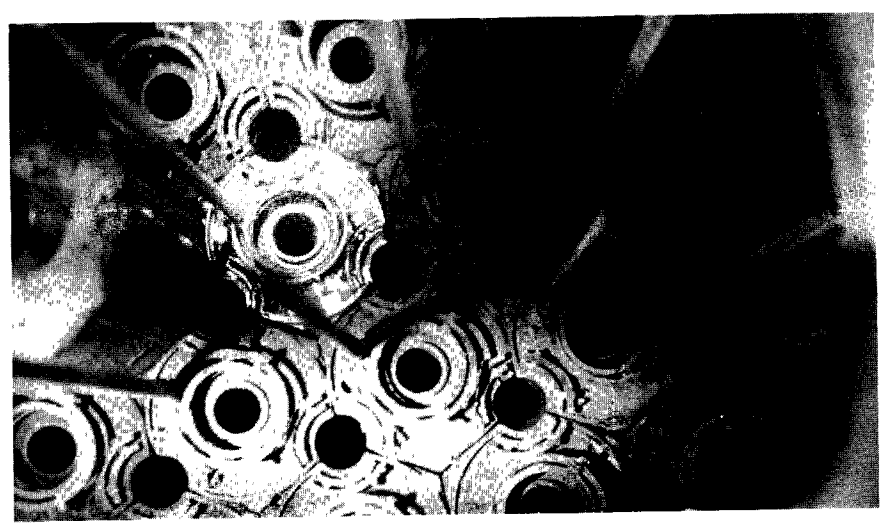


Figure 7

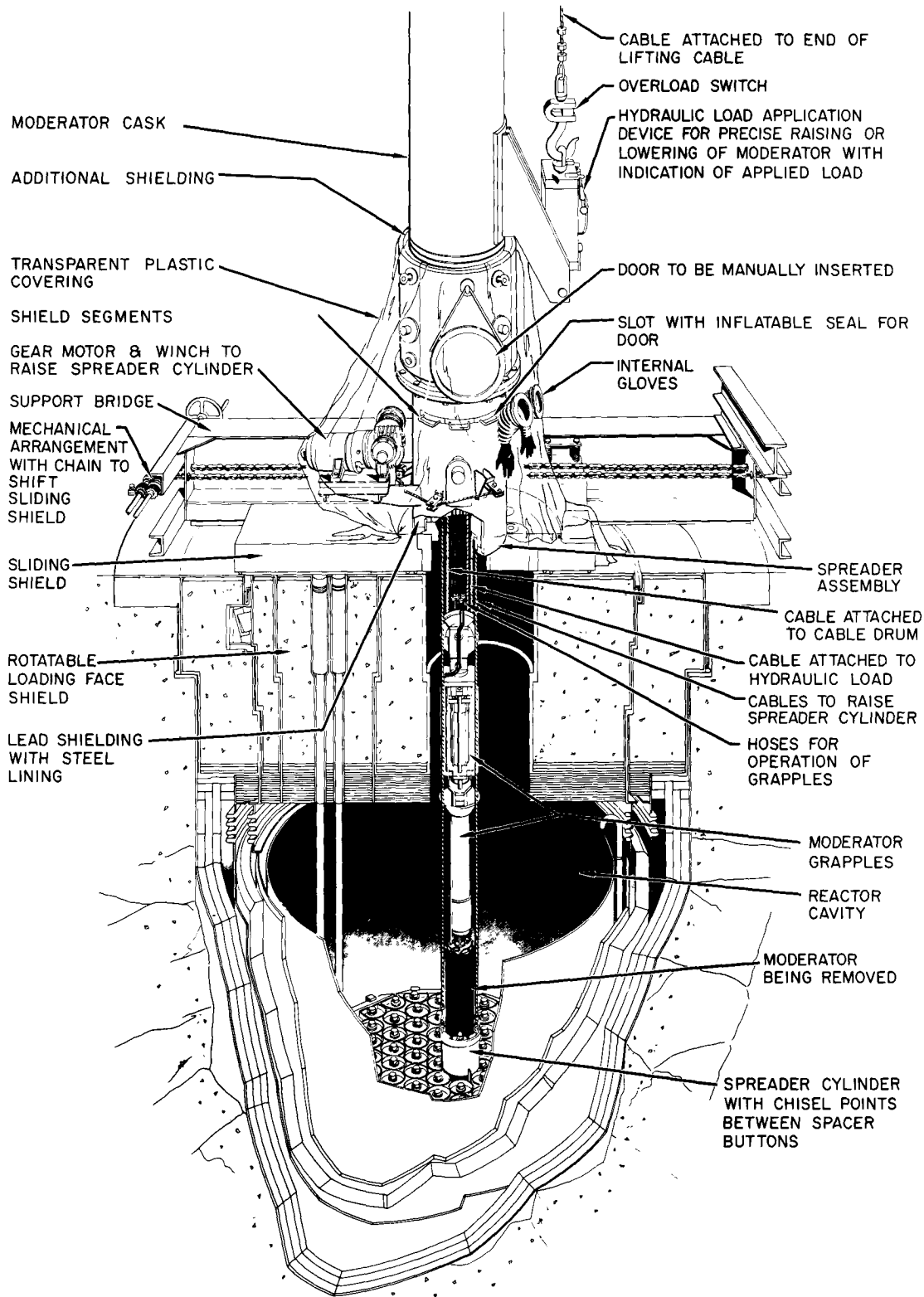


Fig. 8—Moderator removal equipment.