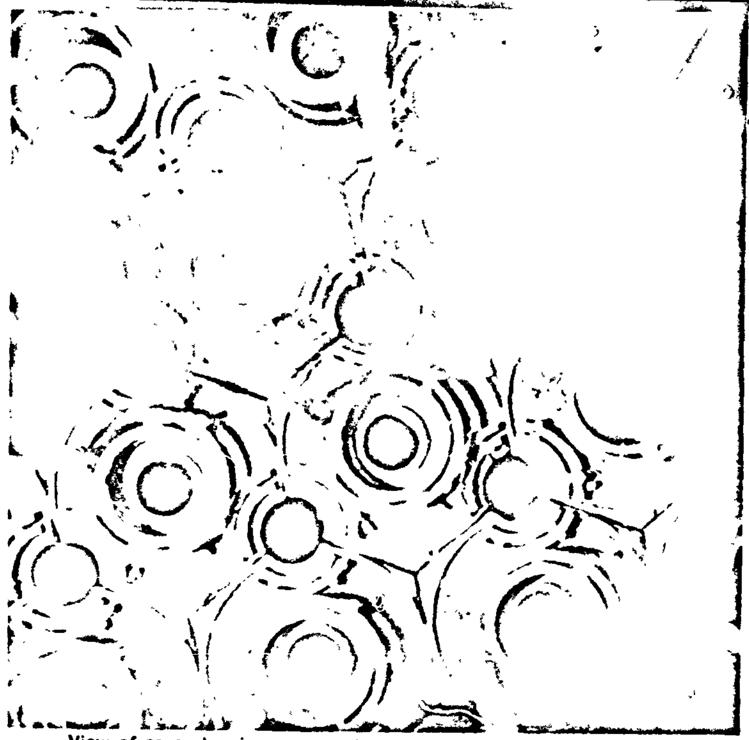

View of core showing new moderator cans and one partly removed can.

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Sodium reactor operating experience

Operation and power generation in the SRE have been successful. Sodium-graphite reactors being designed offer promise of competing with conventional power plants.

THE SODIUM REACTOR EXPERIMENT WAS BUILT and is being operated by Atomics International, a division of North American Aviation, for the Atomic Energy Commission. The object of operating the reactor is to obtain information to make it possible to build full-scale sodium-graphite reactor plants which will be competitive with conventional power plants.

Reactor plant description

A brief description of the reactor and process systems is necessary to permit discussion of the operating experience which the SRE has provided. The SRE is a type known as a sodium-graphite reactor. Liquid sodium is the coolant and graphite is the moderator. The boiling point of sodium (~1600°F) permits the reactor to operate at temperatures in the neighborhood of 1000°F without the necessity for pressurizing the reactor vessel. Sodium is also an excellent heat transfer agent, as are liquid metals in general, and is well adapted to heat removal applications where the heat flux is high.

Graphite has been used as a moderator since the construction of the first nuclear reactor. It is readily obtainable in the high purities required and can be machined with ease. Graphite retains its strength at high temperatures and has good thermal conductivity. As a moderator, graphite is poorer than heavy water, but better than ordinary water.

The sodium-graphite concept is directed primarily toward large central-station power plants. In principle and in fact, superheated steam at temperatures commonly used in modern steam plants can be generated with relative ease. This feature is in itself an attractive incentive for pursuing the development of this type of reactor.

The cutaway view of the reactor, Figure 1, shows that the stainless steel core vessel is doubly contained. The outer tank, which surrounds the core tank, is intended to contain sodium in case of a leak in the core tank. The space between the outer tank and the core tank is filled with steel rings, which form a 5.5-in. thick neutron shield. Because this space could conceivably contain sodium, a helium atmosphere is maintained within it at all times. The outer tank is surrounded by approximately one foot of thermal insulation. Another tank, called the cavity liner tank, serves as a form for the concrete foundation. Cooling for the concrete is provided by steel pipe welded to the cavity liner. Kerosene is circulated to remove heat generated by gamma radiation.

Within the core tank, the grid plate acts as the supporting member for the core assembly. Graphite moderator cans, hexagonal in cross section, rest on the grid plate. Each can is 10-ft. high and approximately 11 in. across the flats. Zirconium sheet, 35 mils thick, protects the graphite from contact with sodium. Since each can merely rests on the grid plate, the cans are individually replaceable. The cans which make up the center region of the reactor have a central zirconium process tube in which the fuel elements are suspended.

Sodium enters and leaves the core tank through penetrations in the core tank wall which are located above the top level of the moderator cans. In this way, a leak in the sodium heat transfer system cannot result in draining the reactor completely.

The inlet sodium lines extend vertically downward and discharge into a plenum below the grid plate. The sodium then flows upward through the center process tubes, cooling the fuel elements. A pool of sodium about 6 ft. in depth is maintained above the

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core. Thus, the core tank serves as an expansion tank for the primary sodium system.

The moderator is also cooled by sodium, which enters a plenum just above the grid plate and below the moderator cans. Approximately 10% of the heat produced in the reactor is generated in the moderator. Sodium flows slowly up the spaces between the cans and then discharges into the upper pool.

The fuel elements shown in Figure 2 are a simple five-rod design. The active portion of each element is 6-ft. in length. Thorium-uranium alloy, enriched with uranium to 7.6 wt. %, is fabricated in the form of cylindrical slugs of ½-in. diameter and 6-in. length. The slugs are contained in a stainless steel jacket, and the resulting assembly makes up a rod. Each five-rod element hangs from a small shield plug which penetrates the concrete top shield of the reactor.

Reactor heat transfer system

A simplified process flow diagram for the sodium heat transfer systems is shown in Figure 3. The main heat transfer system is sized to remove 20 mw of heat, and is divided into two separate systems, the main primary system and the main secondary system. As the primary sodium passes through the reactor, the isotope Na-24 is generated by neutron absorption. With the reactor running at full power, the activity of the primary sodium reaches about 0.3 curies/gram. Therefore, the primary sodium piping is completely shielded by concrete. A sodium-to-sodium heat exchanger of U-tube design transfers heat to the main secondary sodium system, which is similar to the main primary sodium system, but does not require shielding. The main secondary sodium passes through a U-tube steam generator, which generates steam at 900°F and 600 lb./sq. in. gauge.

As a backup, an auxiliary heat transfer system of 1 mw capacity is installed to remove reactor afterglow heat in case the main heat transfer system becomes non-functional. This system is similar to the main heat transfer system except for its size and the fact that heat is dissipated by means of an air cooled heat exchanger.

Pumps used in all heat transfer systems are centrifugal, with a frozen sodium shaft seal. A sleeve around the shaft is cooled by NaK freezing the sodium and making an effective seal. A similar method is used to prevent sodium from entering the pump case and is used on some of the larger valve stems. The frozen seal has proven to be successful, but of course its success is highly dependent on the reliability of the cooling system.

Typical operating conditions of the plant are shown in Table 1. Approximate cost of operation and maintenance of the plant is estimated to be $422,000 for fiscal year 1961. Operation cost is $322,000 and maintenance cost is $110,000.

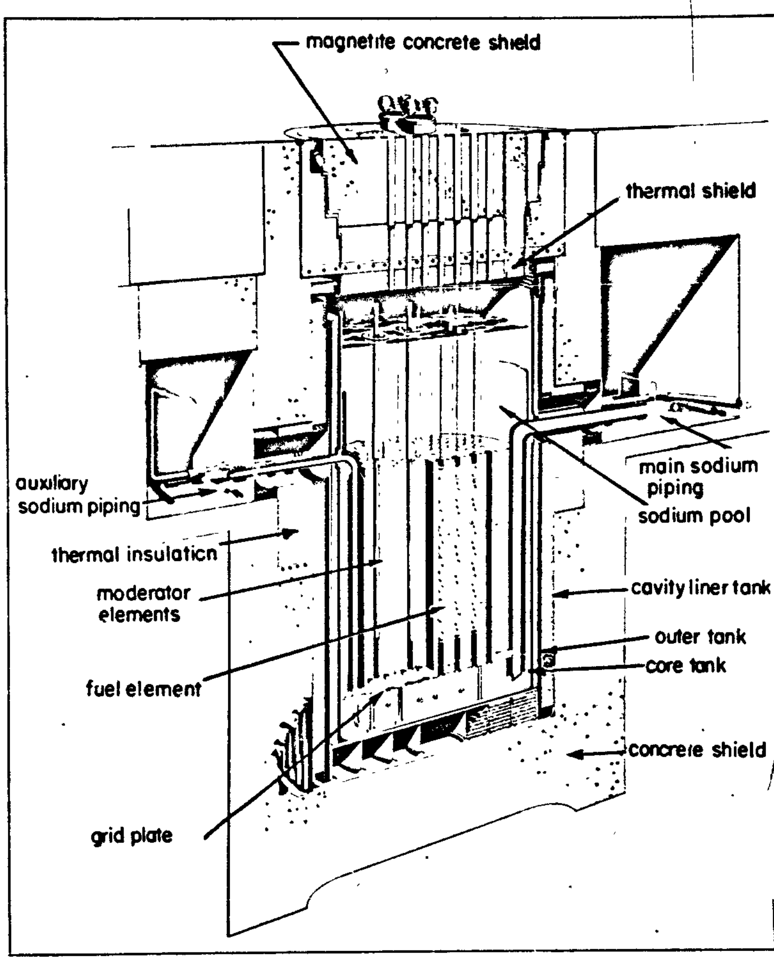

Figure 1. Cutaway diagram of the experimental sodium-type reactor.

Table 1. Typical plant conditions.

	Actual	Design
Reactor pressure, lb./sq. in. gauge	3	3
Sodium flow rate, gal./min.	1100	1100
Reactor inlet temperature, °F	540	500
Steam temperature, °F	900	825
Steam pressure, lb./sq. in gauge	600	600
Reactor power, mw	20	20
Electrical output, mw	5.8	5.8

Start-up and operation

The reactor was started up for the first time on April 25, 1957. Figure 4, showing accumulated power as a function of time, makes it evident that the reactor has been shut down frequently. Most of the reactor shutdowns have been made to permit modifications in the plant design and to examine fuel elements in connection with the fuel development program. Operation at low power levels in 1957 indicated that two important plant modifications were required. One modification was to reduce sodium convection flow after a scram (rapid shutdown) and the other was to permit wider range of control of moderator temperature.

It had been predicted during the

original design of the reactor that sodium flow, owing to convection following a reactor scram, would cause the reactor to cool down at a more rapid rate than desired. It was decided to operate the reactor to determine the exact magnitude of the convection flow before taking corrective action. Too rapid cooling (greater than 1°F/min.) results in high stresses in the reactor coolant outlet nozzle. To control the convective flow after a scram, a device called an eddy current brake was designed and fabricated. The brake retards the flow of sodium to a degree depending upon the current flowing through the coils. Its action is similar to that of an electromagnetic pump for pumping liquid metals.

Figure 5 compares the rate of reactor cooling before and after the brake installation. After a scram, the brake automatically comes on. Brake current is automatically adjusted by a control system to maintain sodium flow between 0 and 250 gal./min. The desired sodium flow rate may be selected by the console operator.

The reactor moderator is cooled by sodium which flows in the spaces between the moderator cans. It was found early in the operation that excessive cooling was present. The excess cooling was attributed to leakage of sodium from the lower sodium plenum, through the grid plate, and into the moderator plenum (Figure 1). Excess cooling of the moderator could have been detrimental in the following ways:

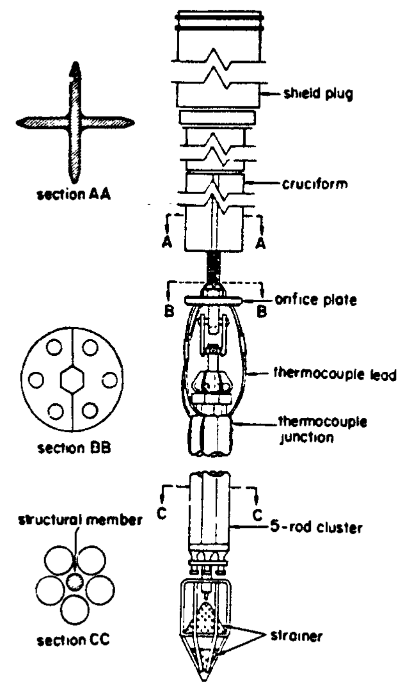

Figure 2. SRE fuel element design.

(1) The outside skin of the moderator cans could be overcooled, resulting in thermal stresses which might eventually rupture the can.

(2) High temperature gradients would be experienced on the core wall near the reactor outlet nozzle.

This problem was solved by installing a linear induction electro-magnetic pump in the sodium line which supplies sodium to the moderator coolant plenum. The pump was reversible and hence could be used to pump sodium out of the plenum, thus compensating for the leakage. During 1960, the induction pump was replaced by an eductor, or jet pump, which performs the same function. The eductor is installed in the 6-in. inlet line to the reactor.

Typical plant problems

Operation of the reactor disclosed unexpected problems in the main sodium-to-sodium heat exchanger. At steady-state power operation, thermocouples installed on the shell of the heat exchanger showed an apparent lack of heat transfer around the U-bend section. It was estimated that about 50% of the flow by-passed the tube bundle in this region. The result of this by-passing was to increase the terminal temperature difference across the tube sheets.

Transient performance of the heat exchanger showed that shell-side internal convection forces opposed the sodium flow in the upper region of the shell. This, coupled with internal circulation through the exchanger tubes, resulted in temperature stratification in both the shell and tubes of the heat exchanger.

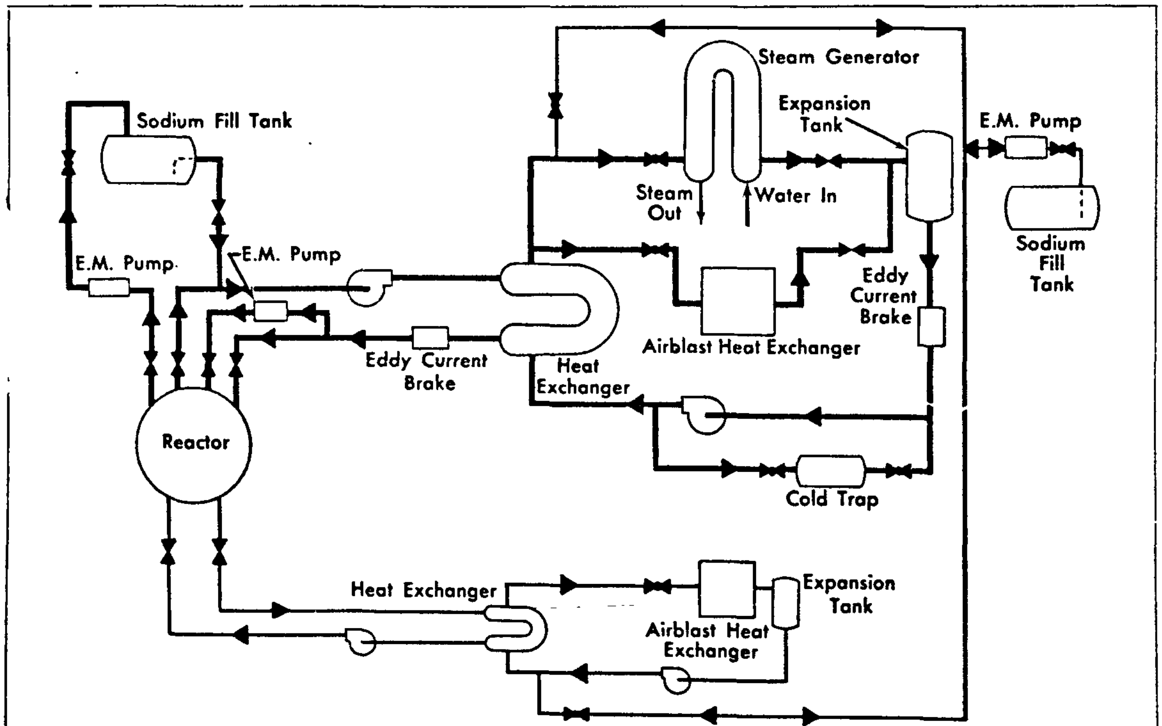

Figure 3. Simplified process flow sheet for sodium heat transfer system.

The heat exchanger was operated under the above conditions while a new heat exchanger was being designed and fabricated. Although the new heat exchanger is also of a U-tube design, the problems encountered are overcome by improved baffling and a change in orientation of the exchanger from horizontal to "on edge."

Moisture caused another problem when it was found during operation to pass through the concrete walls into the sodium-pipe galleries which are kept filled with dry inert gas (N_2). Since it is possible that a sodium leak might occur, the presence of water in the galleries must be minimized.

To control the moisture content a drying system was installed. This system utilizes a refrigerating unit to condense water on cooling coils. A fan circulates the N_2 through the galleries.

An auxiliary use for the drying system is to indicate leakage of radioactive sodium in the gallery. Radiation monitoring instruments permanently installed in the cooling ducts cause an alarm when the activity level exceeds a pre-set value. Valves in the cooling ducts automatically close to isolate the gallery in which activity is occurring.

Plant operation and testing

After plant modifications, the reactor was operated at a power level of 20 mw and at or above design temperature of 960°F. On May 22, 1959, during a short demonstration run, reactor temperature was raised to 1065°F to permit generation of steam at 1000°F.

During these power runs, the stability of the reactor was found to be excellent. During a 144-hr. period of steady-state, full-power operation, integrating timers on the regulating rods recorded a total of only 3.5 min. of rod movement.

The reactor also demonstrated its ability to change power at 20%/min. on manual control. An automatic plant control system, now being installed, will permit more rapid changes in reactor power. "Xenon tilt," a phenomenon caused by uneven burnup of Xenon in a reactor, is not a problem at the SRE, primarily because the dimensions of the reactor are small and the neutron diffusion length is large.

On June 3, 1959, the reactor was shut down because it was found that the auxiliary coolant, tetralin, was leaking into the system through one of the cooling lines on the main primary pump. The leak was repaired,

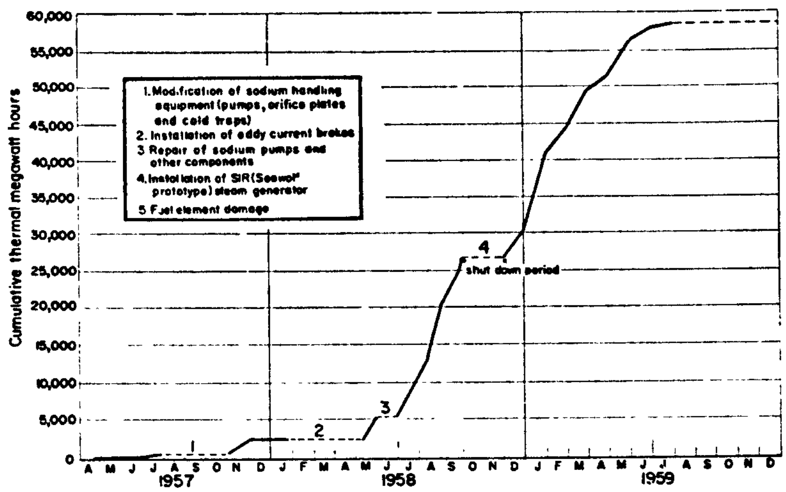

Figure 4. Accumulated power as a function of time since reactor was started.

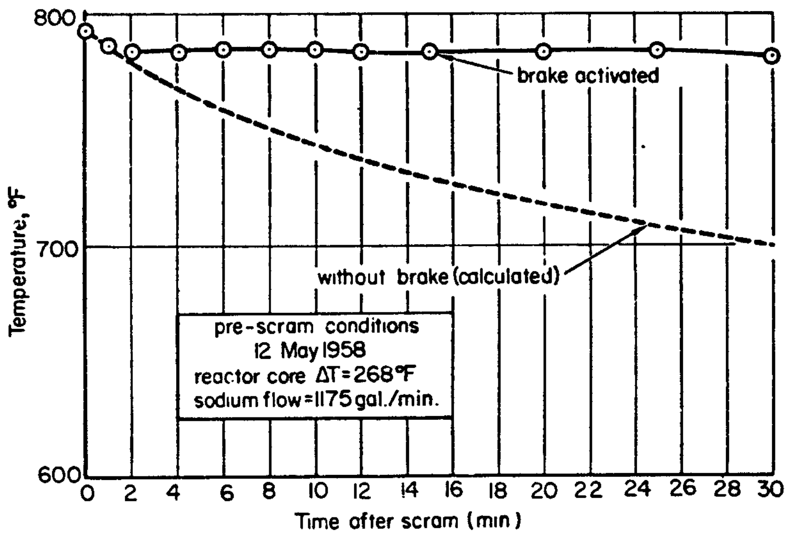

Figure 5. Rate of reactor cooling before and after the brake installation.

but during the following power run (July 12-26, 1959) the decomposition products of the tetralin caused fouling of 13 of the 43 fuel process channels. High temperatures generated within the fuel caused formation of an iron-uranium eutectic, which destroyed the fuel jackets.

The reactor was shut down for about a year to permit development, fabrication, and use of equipment to remove some of the moderator cans from the core and exchange them for new cans. Simultaneously, modifications were made in the plant design to prevent hydrocarbons from leaking into the sodium; and a steam fuel washing facility was installed to permit safe and effective removal of sodium residue from SRE core elements. Instrumentation was added to monitor additional reactor parameters and to provide alarms for off-normal conditions. Finally, the number of rods in the SRE fuel clusters was reduced from seven to five to provide additional clearance in the fuel channels.

Maintenance of sodium system

The SRE offers a variety of maintenance problems. The usual maintenance situations arise in connection with instrumentation, purchased equipment, and auxiliary systems for liquid and inert gas handling. These areas are not unique to a developmental sodium-cooled reactor. There are, however, two maintenance problems peculiar to this type of reactor.

The first concerns maintenance on sodium system components. Because most of the sodium systems are entirely welded, maintenance usually requires that the piping be cut and rewelded.

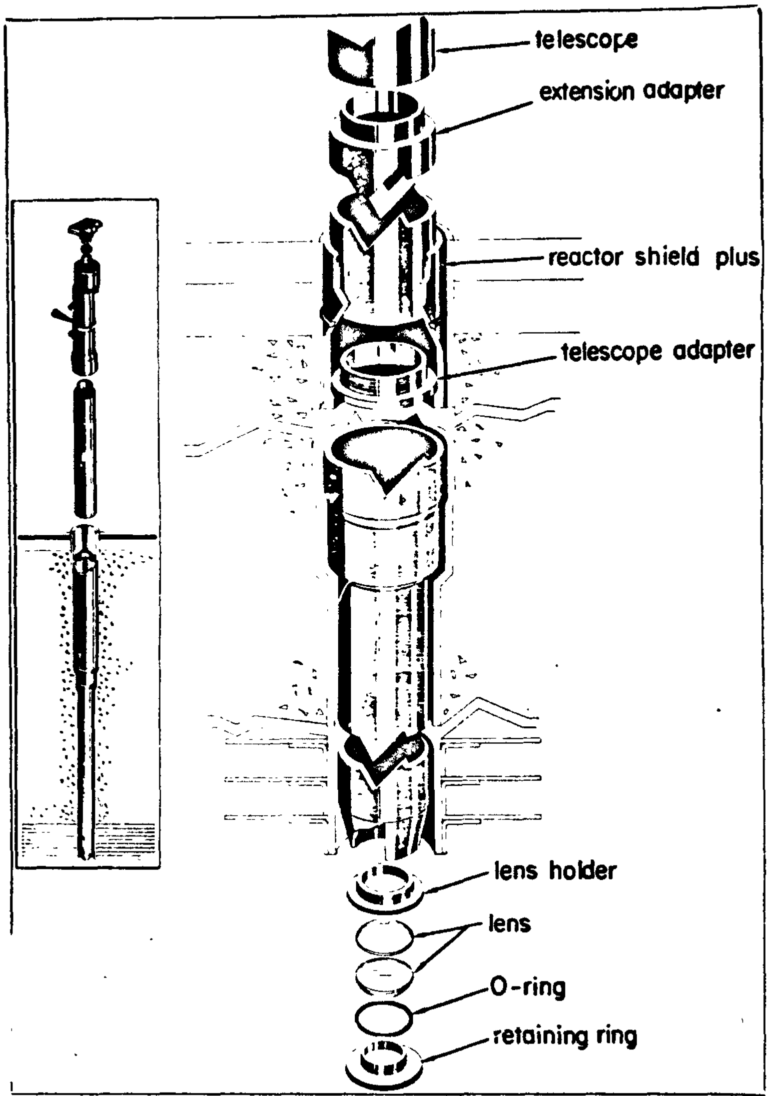

Figure 6. Viewing device, Mark II Corescope, used to observe core interior.

Figure 7. Grappling equipment for removing foreign objects from the core.

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An example of this is replacement of a bellows seal valve used in many portions of the sodium systems. They are initially leak tight and are used in applications where leakage around a valve stem is undesirable.

Experience has shown that if sodium is allowed to freeze and is then reheated and melted in the vicinity of a bellows-seal valve, expansion of the sodium frequently collapses the bellows, finally rupturing it. The valve must then be replaced in the following manner. The stainless steel pipe is allowed to cool to room temperature, and the insulation, thermocouples, and heaters are removed. The pipe is then cut with a pipe cutter or hacksaw, depending upon accessibility.

As soon as an opening is made in the pipe, the surface of the sodium quickly oxidizes. The layer of oxide prevents further reaction with the air. If moisture content of the air is high, some slight bubbling of the sodium surface will occur. A face shield, flame proof overalls, and gloves are used as a safety precaution.

To prepare the pipe for rewelding, the sodium is removed about six inches in each direction from the place to be welded. Since sodium is soft, this can be done manually with a spatula. The surfaces are then cleaned with butyl alcohol to remove any remaining sodium. A purge hole is drilled in the pipe, and inert gas (argon or helium) is admitted during the welding. As the weld is completed, it is important that the inert gas pressure be maintained as low as possible to prevent blowing through the weld. The completed weld is x-rayed and helium-leak checked. The purge hole is then welded shut.

This technique has been extremely effective in the repair of sodium systems. The fact that sodium is solid at room temperature is an asset rather than a liability, since it permits sealing off the piping at any point in a convenient manner.

Decomposition product problems

The decomposition products of tetralin entering the reactor caused damage to a number of the fuel elements. These elements were found to be swollen and impossible to remove from the process channels. This circumstance made it necessary to perform maintenance on the reactor core itself, an area of activity where there has been little experience in comparison with the maintenance experience on other components of reactor systems.

This particular maintenance problem was resolved into several phases. The first phase involved retrieval and removal of the broken fragments of the fuel elements. The second phase was removal and replacement of the moderator cans containing the swollen fuel elements. The third phase consisted of visual and photographic observation of the core interior, to determine if abnormalities existed.

Before any maintenance could be started, equipment had to be developed for illuminating and seeing the core interior. Handling equipment had to be designed and fabricated to permit removal of various foreign objects from the core, and to permit removal and replacement of moderator cans. In addition, methods for detection of faulty moderator cans had to be developed.

Lighting was provided by means of incandescent lamps. These lamps were fastened on the end of plugs installed in the reactor top shield. Hard solder was substituted for soft solder because of the temperature. Simple viewing devices consisted of an objective lens and a commercial telescope, Figure 6.

Grappling equipment for removal of foreign objects from the core was made as simple and rugged as possible. Each tool was approximately 200-in. long. Figure 7 shows the grappling end of one of the special tools used for retrieval of foreign objects in the reactor. Using this and similar tools, over 80 fuel slugs and pieces of cladding were successfully extracted from the reactor core.

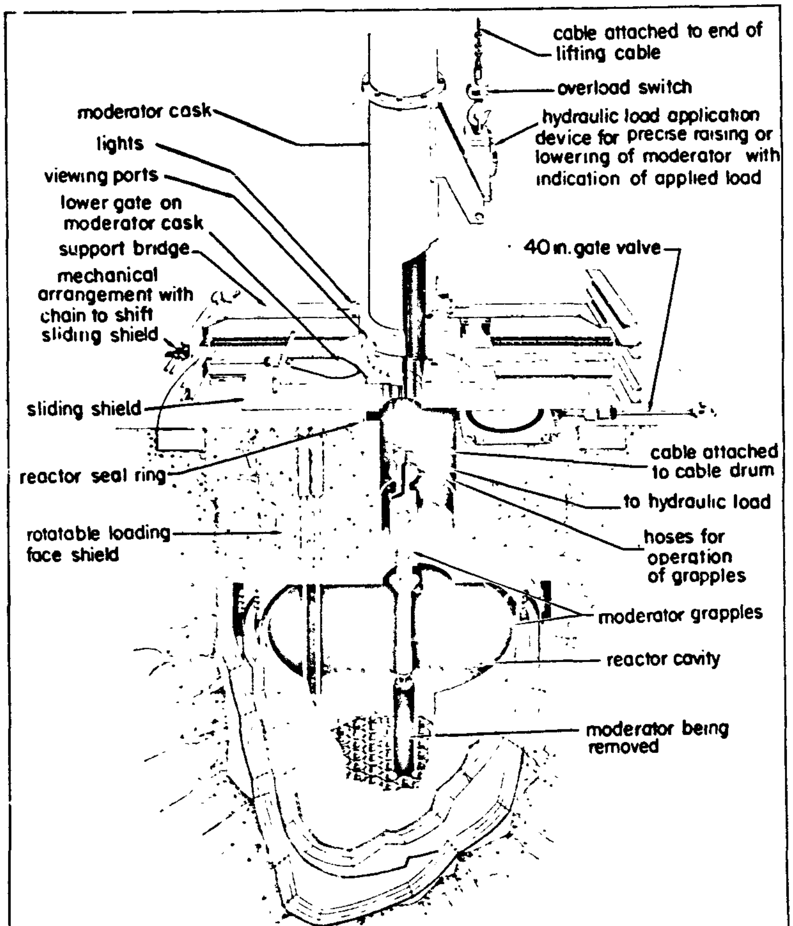

Figure 8. Moderator removal equipment with one moderator can being removed.

Equipment for handling moderator cans was somewhat more elaborate. To remove a moderator can and replace it with a spare, the spare can was first installed in a moderator can storage hole. A 40-in. gate valve, installed on the reactor loading face after removal of one of the 40-in. access plugs, permits removal and replacement of core components without release of radioactive inert gas.

The moderator handling cask, Figure 8, is sealed to the 40-in. gate valve, an inert atmosphere established, and the grapple lowered to remove the can. After the can is drawn into the cask, the valve is shut. The can is then deposited in a storage hole, the replacement can picked up with the grapple, transferred to the reactor and lowered into place. Sixteen of the cans in the reactor were replaced in this way during the summer of 1960. The introductory photograph shows the core with a moderator can partly removed. Several new cans can be seen.

Detection of faulty moderator cans can be accomplished in two ways. First, when sodium is absorbed by graphite, the graphite experiences an expansion of about 1%. Thus, a moderator can 10-ft. long will expand longitudinally about 1.2 in. This expansion can be readily measured by mechanical probing.

Second, specially designed induction and pickup coils can be lowered into the central process tube of the moderator can. If sodium is present in the moderator can, the output of the pickup coil is higher than if sodium is not present.

Conclusion

The SRE has demonstrated the feasibility of generating power by means of a sodium-cooled, graphite-moderated reactor. High quality superheated steam is a natural end result of the process used.

Operating experience with the reactor since 1957 has given information equivalent to that which would be gained over a period of approximately 20 yr. with a production-type power reactor. Although ⅓ of the fuel elements in the reactor failed on one occasion, this difficult situation was dealt with.

Methods for core maintenance on sodium graphite reactors are now on a firm basis. As a result of operating experience, sodium-graphite reactors are now being designed which will compare favorably with other reactor types and which offer promise of competing successfully with conventional power plants. ♦