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Section VI

CURRENT EVENTS

SRE Operating Experience

Editors' Note: This article was prepared by Robert W. Dickinson of Atomics International at the request of the editors of* Nuclear Safety. *It therefore represents the views of Atomics In-*

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During World War II he served as a naval officer aboard U. S. submarines, chiefly in an engineering capacity. He participated in seven combat war patrols, and was awarded the Silver Star medal. After obtaining the M.S. degree from the U. S. Naval Postgraduate School in 1948, Dickinson was a submarine squadron engineer with the U. S. Navy in San Diego, Calif., for two years. From 1950 to 1952, he was an assistant planning and estimating superintendent at the Mare Island Naval Shipyard in California.

Following study at Massachusetts Institute of Technology, Dickinson was connected concurrently with the Navy's Bureau of Ships as an engineer concerned with the application of nuclear propulsion for naval use and with the AEC as head of the Nuclear Components Section of the Naval Reactors Branch. He resigned from the Navy in 1956 with the rank of commander and joined Atomics International the same year.

ternational and is consequently not a critical review in the same sense as most of the material presented in this journal.

The Sodium Reactor Experiment (SRE) is a sodium-cooled graphite-moderated reactor system that operates with slightly enriched fuel at a thermal power level of 20 Mw. It was originally designed solely as a reactor experiment, but a small turbine-generator set and a steam generator were added by the Southern California Edison Company (SCEC) to dissipate the heat produced. The plant for generating electricity, including the steam generator, is owned and operated by SCEC; the reactor is owned by the AEC and is operated by Atomics International. Criticality was achieved in April 1957, and electric power was first generated on July 12, 1957. During the life of the first fuel loading, slightly more than 15×10^6 kw-hr of electricity was generated. Operation of this first core was terminated by metallurgical damage to the unalloyed uranium fuel as a result of fission-gas swelling. Coincidentally, an auxiliary coolant system failure, which will be discussed in detail below, caused restrictions of coolant flow to the fuel elements and required suspension of reactor operation on July 27, 1959. It was found that 13 of the 43 fuel assemblies had been damaged.[1,2] A new fuel loading consisting of 7.6 wt.% thorium in a uranium alloy was inserted, and criticality was achieved with the new core loading on Sept. 5, 1960.

The SRE was the first high-temperature reactor of any significant power rating to be operated in this country and, as such, has provided much information relative to sodium technology and to high-temperature reactor operation in general. A "once-through" steam-generating system is used which provides steam at 600 psi and 825°F. These conditions were set, not by the ability of the liquid-metal-cooled reactor system to generate steam at these conditions, but rather by the ability of the small 7.5-Mw(e) turbine-generator set to employ them. Summaries of SRE operating experience have been provided from time to time in periodic progress reports and, more recently, by a comprehensive report dealing with the damage to the fuel assemblies in the 1959 incident.[3] An over-all summary of the operating history to date is being prepared, and detailed descriptions of the reactor and the early operating experience have been issued.[4]

System Operating Information

The design and construction philosophy of the system emphasized the use of conventional, commercially obtainable components wherever possible. To this end, the coolant circulating pumps are simple adaptations of hot oil pumps. The stuffing box was replaced by a cooled annulus around the shaft to freeze sodium and thus seal liquid sodium in the pump casing from the supporting bearings on the shaft and the drive motors.

The primary and the secondary systems each have a 6-in. pump and a 2-in. pump, and three of these pumps have had erratic operating histories. The 2-in. pump in the primary system has bound on numerous occasions. It is normally operated at minimum speed, and minor fluctuations in the temperature of the coolant to the freeze seal result in fairly large variations in the torque required to shear the frozen-sodium shaft seal. These fluctuations have been reduced as experience has been accumulated. Closer control over coolant temperatures is now maintained, and the pump is operated routinely at 300 rpm, a speed which experience shows provides sufficient torque to keep the pump running but still does not circulate an excessive amount of sodium through the primary system. Conversely, the 6-in. 1500-gpm-capacity pump in the secondary system has never given difficulty and has operated uninterruptedly for 1000-hr periods that were terminated by shutdown of the system rather than by difficulties with the pump.

The 6-in. 1500-gpm pump in the primary system, on the other hand, was the main offender in the fuel-element damage incident that occurred in July 1959. Difficulties had been experienced from time to time with binding similar to that of the 2-in. pump, and, on two occasions, the auxiliary coolant (Tetralin) in the freeze seal had leaked into the main sodium coolant stream. This was detected by identifying hydrocarbon vapor in the atmosphere above the reactor pool. The first of these occurrences, in April 1958, resulted in an insignificant amount of leakage before the freeze seal was repaired. This leak was caused by a pinhole in the freeze-seal casting. The second such leak occurred in May 1959 and resulted in somewhere between 2 and 10 gal of Tetralin being admitted to the primary sodium stream.[5] This leak was traced to failure of a thermocouple well. The Tetralin-cooled shaft seal was then replaced with a NaK-cooled seal arranged in such a way that two independent barriers would have to fail before mixing of NaK and the primary sodium could occur. As will be mentioned later, however, sufficient Tetralin had already been admitted to the system to create the condition that damaged the fuel assemblies.

The heat-transfer performance of components outside the core proved to be both enlightening and somewhat disappointing.[6] The main intermediate heat exchanger is a shell and U-tube unit of the simplest configuration deemed feasible in 1954. The straight sections in the shell are baffled, but the expansion space in the U end is unbaffled. Measurements have indicated two primary faults with this design. First, the unbaffled end section is substantially ineffective in heat transfer, and, second, the baffles that direct flow back and forth across the tube bundle horizontally are relatively ineffective in preventing stratification of sodium on the shell side. The high temperature drop taken across this heat exchanger from inlet to outlet, 460°F or greater, causes significant density changes in the sodium, and the cooled sodium sinks to the bottom of the exchanger. This, of course, upsets the temperature distribution, affects the efficiency of heat-transfer surfaces, and also introduces unforeseen thermal-stress patterns. Results of these phenomena are that (1) the heat exchanger is only about two-thirds as efficient as designed, thus requiring a log mean temperature difference across the heat exchanger (temperature drive) of 90°F rather than 60°F, and (2) thermal-stress patterns are set up which are expected to limit useful life in this piece of equipment. In fact, in the vicinity of the tube sheet, "wrinkles" 0.0625 in. in height have already been observed in the shell of the heat exchanger, which is 0.14 in. thick. A new heat exchanger designed to alleviate the problems with the original equipment has been obtained for installation when convenient. Although the new heat exchanger is, in general, similar to the original equipment, it contains additional heat-transfer surface and is baffled to inhibit stratification of cooled sodium on the shell side.

The once-through steam generator procured by the SCEC has, in general, been satisfactory.[7] It, too, is a horizontal U-shaped, shell-and-tube unit, with sodium in the shell and feed water (and steam) in the tubes. Each tube is made up of two concentric tubes, with mercury in the annulus. The mercury serves as a leak detector in case of failure of either of the tubes. Heat-transfer performance has been satisfactory in all but one respect; the stratification noted in the intermediate heat exchanger also exists in the steam generator. The shell side, which contains secondary-system sodium, is essentially unbaffled except for tube supports, and temperature differences of up to 150°F have been measured vertically across the outlet (steam) tube sheet. Apparently saturated steam is being delivered from at least some of the bottom tubes in the area, and highly superheated steam is produced by the top tubes. As these streams mix, a net superheat is achieved, but undesirable stress patterns are set up across the tube sheet. It has been noted that the shell of this unit, which has a developed length of 80 ft, has become bowed to the extent that the outlet end of the heat exchanger has raised 4 in. from its original position, both measurements having been taken with this end unrestrained. On the other hand, no evidence of deposition of foreign materials on the water side has been found, and there have been no indications of chloride stress corrosion, despite construction of the unit from type 304 stainless steel. A rigid program of water purification has been developed and adhered to by SCEC. Total dissolved solids are maintained at less than 0.5 ppm, and oxygen is maintained at an undetectable level through hydrazine injection. Internal inspection of the steam generator during a recent shutdown period indicated that these measures have been highly successful in preventing the development of conditions that caused difficulty in previous sodium-heated steam-generating systems.

Bellows-sealed valves in sizes of 2 in. or less were used throughout the auxiliary systems. These valves have been a source of some difficulty, particularly as a result of bellows failure. Collection of sodium oxide in the bellows area and occasional freezing of sodium in the volume provided by the bellows have interfered with subsequent operation of the valve and have been the prime causes of system failures, as established by post-mortem examination. Nine valves of this type have failed in the primary and secondary systems. None of these failures have occasioned more than inconvenience, since leakage sodium usually freezes, oxidizes, and prevents further leakage from the failed valve; however, the valve is rendered inoperative. These valves are used only in auxiliary systems, and therefore it has not been necessary to shut down the plant immediately to repair a valve that failed. Such repairs have, however, inevitably extended maintenance shutdown periods.

Conversely, the primary sodium system stop valves, which are sealed by frozen sodium at the stem, have been completely satisfactory. Therefore, as bellows-sealed valves are removed from the auxiliary systems, they are replaced with freeze-sealed valves cooled by ambient inert gas at the gallery temperature (140° F or less). Such valves have been completely adequate in experimental loops containing sodium at temperatures up to 1200° F.

Reactor Structure

The reactor structure is unique in that the required containment is immediately adjacent to the core, and no external sphere or pressure-containment building is provided.[8] This design is based primarily on the following three considerations:

1. The coolant is essentially at atmospheric pressure, and thus it represents negligible stored mechanical energy.
2. The fuel, moderator, and coolant are chemically compatible, and thus there is no potential for release of chemical energy.
3. The system has a large thermal capacity that is available for absorbing any conceivable unplanned release of nuclear energy.

The stainless-steel core tank, which contains the moderator and fuel elements, is surrounded by a second containment vessel, which has dimensions such that, in the event of a leak in the reactor vessel, the sodium coolant will be retained and the level of the sodium in the reactor pool will not drop below the level of the outlet nozzles. An inert-gas atmosphere is maintained in the second containment vessel, and therefore sodium circulation can be continued even in the event of failure of the core tank. Should both these vessels fail, tertiary containment is provided by a carbon steel "cavity liner," which serves to support thermal insulation on its inner surface. The space between the secondary container and the cavity liner is such that a rupture of both the two inner vessels will not drop the sodium pool level below the top of the core. Thus adequate decay heat cooling of the fuel elements would be available in the unlikely event that both inner containers ruptured. Leak detectors are provided to indicate failures of the two inner tanks.

The core tank and the inner containment vessel are cylindrical and are sealed at the top by a rotatable steel plug filled with magnetite ore and concrete grout, with a total weight of 75 tons. This top plug is pierced for accommodation of 81 smaller shield plugs from which fuel elements are suspended, control and safety rods are inserted, and core instrumentation is manipulated. These smaller plugs are sealed with rubber O rings that operate at ambient temperature. The entire shield plug is sealed to the surrounding biological shield by a low-melting metal alloy (Cerrobend) which solidifies in a tongue-and-groove seal. Double containment of the reactor nozzle extends 6 ft from the reactor pressure vessel to a point immediately adjacent to the system stop valves and terminates in a bellows seal that accommodates differential expansion. An inert-gas atmosphere is maintained within this containment space, and leak detectors are provided to indicate a pipe or nozzle failure. Sodium is conveyed in single-walled piping outside these bellows, with secondary containment provided by concrete shielding blocks over the galleries. The shielding blocks are sealed with neoprene gaskets, and an oxygen-deficient atmosphere (less than 1 per cent O_2) is maintained inside the galleries during reactor operation.

No failures have been detected in piping or containers by leak detectors or other indications, such as loss of pressure or sodium smoke. The neutral atmosphere maintained in the galleries proved effective in preventing sodium combustion in the one instance of a leak in a 2-in. bellows valve in the secondary system. The leak was first indicated by a wisp of sodium oxide "smoke" from reaction with air in the valve operator. Only static sodium was involved, and there was no radioactivity associated with the smoke. In this case the sodium was frozen in the line by turning off resistance heaters, and operation proceeded until the next scheduled shutdown period, at which time the leaking valve was cut out and replaced. Examination of the valve indicated that less than 1 lb of sodium had leaked out and that this sodium, after solidification and oxidation, had effectively sealed the leak. The valve was, of course, rendered inoperative. Other valves in the secondary (nonradioactive) system had leaked from time to time, with generally similar results (i.e., sodium fires did not occur because of solidification and the formation of a protective crust over the free sodium).

Difficulty was experienced early in the reactor outlet nozzle area.[7] This nozzle conducts sodium at the outlet temperature of 960°F to the main circulating piping system, and it was not provided with a thermal baffle. The tank wall to which the nozzle was attached was baffled to maintain approximately 1 in. of stagnant sodium against the tank wall in order to alleviate temperature gradients. When power operation was initiated, it was found that the circulation of sodium by natural convection after a scram was approximately 30 per cent greater than originally calculated. If the circulation were not controlled, the fuel elements would be cooled more rapidly than decay heat was being generated, and the temperature at the top of the sodium pool would be rapidly lowered. This created thermal stresses in the nozzle, since the nozzle assumed the new sodium temperature; whereas the wall, which was thermally baffled, retained its original temperature for a significant period of time. This did not create a problem during startup, steady-state operation, or normal shutdown operation, but calculations showed that it would cause excessive stresses during a scram from full power. Consequently, the reactor power level and the temperature were limited to values that could not induce stresses past the yield point of type 304 stainless steel until an eddy-current brake could be installed in the primary system to control convective sodium flow after a scram. Temperature measurements taken after installation of the eddy-current brake have indicated that the brake is completely successful in matching flow and heat-generation characteristics following a scram at full power.

Radiation Levels

The primary cooling system of the SRE is, of course, contained in biological shielding of concrete aggregate to control radiation from the Na^{24} generated in passage of the coolant through the reactor. The buildup of radiation on system piping through the various agencies was, however, of great interest in terms of accessibility after periods of reactor operation. It was considered that the temperature conditions of the primary sodium system (i.e., a maximum temperature of ~1000°F and a temperature difference across the system of 500°F) were potentially conducive to accelerated mass transfer of iron, cobalt, nickel, and other constituents of the stainless-steel and zirconium components of the circulating system. In addition, the possibility of fission-product leakage is always present.

No activity transport could be measured during the first two years of reactor operation, and during that time sodium was circulated in the system for approximately 16,000 hr, 4500 hr of which was at a high temperature and a large temperature differential. During this two-year period, two fuel elements were found to have pinhole leaks, and there was some release of gaseous fission products. These were detected by the presence of Xe^{133} in the reactor helium atmosphere, as indicated by a pulse-height analyzer. The leaking fuel elements were identified by withdrawing the elements into the fuel-handling cask during a shutdown period and noting any increase in gas activity within the cask. The two fuel elements which were defective revealed their presence quite clearly by immediately producing an increase in the count rate measured by the cask-gas sampling apparatus from a background value of 200 dis/min to count rates of 3000 and 16,000 dis/min, respectively. The remaining 41 elements caused no increase in count rate upon sampling of the cask gas. Visual examination of the suspect fuel elements in the SRE hot cell indicated no defects visible to binocular-aided vision, but these elements were removed from the reactor and stored for eventual reprocessing.

A massive release of fission products throughout the primary system occurred in the July 1959 incident, as described previously.[2,3] Thirteen of the 43 fuel assemblies sustained damage sufficient to cause failure of the cladding on all seven rods of the assembly and to form at least some iron-uranium eutectic that was molten for a short period of time in the reactor. Measurements of cover-gas activity extrapolated to the time of the fuel-element damage indicate that approximately 10,000 curies of mixed fission-product activity was released to the coolant stream, although this massive release of activity remained undetected for some days because of the complete containment afforded by the SRE shielding system. This occurrence, which caused some fuel-element temperatures to rise abnormally high, at least to 1430°F, came about during a period of reactor operation in the 2- to 3-Mw(t) range. This power level was being maintained in order to alleviate a temperature maldistribution throughout the reactor, which was considered, on the basis of previous similar experience, to be caused by impurity collection on the orifice plates. Temperature-induced reactivity fluctuations amounting to 0.3 per cent $\Delta k/k$ were found upon examination of records after the fuel-element damage was discovered. The operators noted subjectively that the reactor was somewhat "touchy" to control, and, when the reactor went on a 7-sec positive period, it was immediately shut down. A cautious restart did not reproduce any of the symptoms present prior to the fast period, and low-power operation was continued for several days. These events were described previously[2] and have been analyzed by Fillmore.[9]

The primary piping system was opened three months after the release of radioactivity to the system. At that time, the radiation level throughout the primary gallery averaged 500 mr/hr, but isolated spots in the system gave readings as high as 15 r/hr at the piping surface. These spots were almost invariably associated with irregularities in the system, such as thermocouple wells, heat-exchanger end bells, and horizontal piping runs containing sodium that was normally circulated at a low velocity. Cold trapping through a 30-gpm sidestream loop was undertaken, and it was apparent that fission products were being removed by the cold trap, since the radiation level rose from substantially zero to 190 r/hr at a specific point on the surface of the trap in a period of six days. At this time, the cold trap was removed because the radiation level was increasing beyond that which could be handled by the available shielding. A new cold trap was installed and monitored, and a similar increase in activity from zero to about 90 r/hr occurred in five days. While the cold trapping was being performed, sodium was being circulated at the maximum flow rate of 1500 gpm through the primary system. After the cold-trapping period, the radiation levels in the pipe gallery had dropped to an average of 150 mr/hr. By applying lead-sheet shielding locally in the region of the cold leg of the intermediate heat exchanger and at some piping depressions, the radiation level was reduced in working areas to the 8-hr tolerance.

Upon draining the sodium from the reactor, it was discovered that a black sponge-like material, which had apparently been floating on top of the sodium pool, had been deposited on the tops of the moderator cans. Later analysis proved this substance to be amorphous carbon with about 25 per cent sodium and sodium oxide in the vacant holes. Visually, this mass had the appearance of black foam with small spherical bubbles that ranged from 0.1 to 1.0 mm in diameter. Radiation analysis of this deposit, which was removed with a vacuum device operated through hermetically sealed shield plugs in the loading face shield, indicated that much of the activity was concentrated in the carbon. The activity of sodium samples taken six months after the fuel-element damage was 0.3 $\mu c/cm^3$, but the carbon activity was 0.03 $curie/cm^3$, a factor of 10^5 greater. The specific carbon activity was difficult to determine because of the spongy nature of the material, but the orders of magnitude are considered to be correct. Vacuum cleaning of the tops of 100 of the 119 moderator cans removed approximately 10 lb of carbonaceous, sodium-bearing material in which the bulk of the fission-product activity was apparently concentrated. The remaining carbonaceous material is currently being removed by filtration in the cold trap and by hot trapping the dissolved carbon.

The specific activity of the sodium is declining in accordance with the decay curve for mixed fission products. The sodium has been reintroduced into the reactor, but the maximum temperature has been limited to 800°F because of the dissolved carbon. Concern is felt that, at temperatures greater than 800°F, the dissolved carbon might cause carburization of the 0.010-in.-thick type 304 stainless-steel fuel cladding.[10] Higher temperatures will not be generated until the cold and hot traps have reduced the dissolved carbon content of the sodium to 20 ppm or less.

Analysis of the helium cover gas shortly after discovery of the activity release showed that only the noble gases were released. Xenon and krypton were identified as the sole contributors to cover-gas activity. No I^{131} has been detected except that combined with sodium. Removal of the irradiated fuel from the reactor reduced the cover-gas radioactivity to near normal background, despite the presence of fission products in the sodium. Disposal of piping and components replaced during the recovery period has not proved to be a severe problem. The normal methods used for cleaning components containing uncontaminated sodium were applied, with the additional precaution of retaining the sodium-steam-reaction products which apparently contained all the fission products not deposited on pipe walls. Fission products remaining on pipe walls created radiation levels of only a few milliroentgens per hour, and these components could be disposed of as relatively low-level wastes by conventional methods. The reaction products were disposed of as low-level liquid waste by diluting and mixing with concrete.

Repair Under Radioactive Conditions

For the first nine months of the repair period, exposed, irradiated fuel elements were present within the reactor structure. This necessitated performing all operations on the core structure through hermetic seals, since the core cover-gas activity ranged from 0.3×10^{-2} to 1.0×10^{-2} $\mu c/cm^3$. Inasmuch as the activity was well above the atmospheric release tolerance, complete containment was required. Inflatable seals were found to be adequate and were widely employed in the equipment designed and constructed for removal of damaged fuel elements, moderator cans, fuel slugs, and other debris that had been deposited in the core. In addition, it was necessary to prevent the spread of activity over the reactor floor by repair equipment which had been mated to the reactor face and through which irradiated material had passed. Vinyl plastic sheet, laid over the entire floor to prevent contamination of the concrete, performed this function effectively. Regular replacement was required because of wear and tear, and the sheet was necessarily disposed of as contaminated material. Cloth and plastic booties, used in conjunction with coveralls, were employed by the operating crews involved in reactor floor operations. Continuous monitoring of personnel did not reveal any instance of contamination. Work in the primary-system pipe galleries, which involved cutting and welding, was performed by Atomics International craftsmen. Components that required replacement were cut from the system by men using respirators; the area adjacent to the cut was thoroughly cleaned of sodium by using butyl alcohol prior to conducting rewelding operations. These operations were carried out under the supervision of health physicists; again, no activity release above tolerance was noted, and no person received more than 300 mr/week, despite the presence of contaminated equipment and a rather high background activity level from piping and system components.

Operations within the reactor core were particularly time consuming because tools and viewing apparatus had to be improvised for each situation. The operations were further complicated by the necessity for keeping the entire reactor structure at a temperature above 250°F in order to prevent solidification of any residual sodium that might be adhering to moderator cans or might be trapped by carbonaceous deposits. Mercury-vapor lamps, built into shield plugs 3 in. in diameter, were inserted into the top shield for illumination, and various remotely controlled grappling tools were employed to remove 82 fuel slugs 6 in. long and $^3/_4$ in. in diameter from the top of the core where they had fallen upon removal of the partially severed elements. A particularly difficult situation was presented when inspection of the bottom plenum (33 ft from the operating face of the reactor), with the use of locally built periscopes and illumination, revealed two fuel slugs. It was necessary to devise a tool capable of penetrating 33 ft from the reactor face through a minimum opening of 2 in. and then reaching 18 in. to the position of the slugs. A "clamshell" device was designed to recover these slugs so that, in the event they were dropped, there would be no possibility of their shattering in the bottom plenum. Shattering was considered a good possibility if the slugs were dropped because uranium-iron eutectic is quite brittle, and there was no assurance that it was not present to some degree on these two slugs. Indeed, slugs removed from the top of the core were found upon hot-cell examination to be extremely brittle. The entire recovery operation was quite tedious, but a thorough inspection of all available parts of the reactor structure indicated that all fuel material was removed.

System Kinetics

The period of operation prior to fuel damage demonstrated the stability of the reactor, which has a prompt negative power coefficient of 2.2×10^{-5} $(\Delta k/k)$/Mw(t), and a positive moderator temperature coefficient of approximately 1×10^{-5} $(\Delta k/k)$, with a time constant of 10 min at normal operating temperatures.[11] The over-all power coefficient at 20 Mw(t) is -7×10^{-5} $\Delta k/k$ per thermal megawatt. The reactor is stable under all normal operating conditions, and it follows power swings of 4 Mw(t)/min (20 per cent per minute) induced by the SCEC steam plant. An automatic control system has been installed so that reactor operation can be completely controlled by the turbine operator; this system will be programmed from a remote station when electric power generation is resumed in 1961.[12] Careful measurements have indicated no power shift resulting from xenon oscillations, and application of existing theory demonstrates that such power shifts should not be expected in sodium-graphite reactors of any conceivable size. The fuel irradiation lifetime of the first core loading averaged only 1100 Mwd/ton, and therefore changes in the fuel temperature coefficient with irradiation were not measurable. Reactivity losses from burnup and fission-product production were consistent with calculations,[13] although they were of minor absolute magnitude, since burnup proceeded to only a maximum of 0.15 at.%.

The radial power distribution across the SRE core would cause significant temperature differences between outlet fuel channels if orificing were not employed. Orificing is, in fact, employed to maintain the outlet temperatures from each channel within 50°F of the desired mixed-mean outlet temperature. The orifice plates are readily changeable in a hot cell. Prediction of orifice sizes has become quite accurate with experience, and, as the power profile changes with control-rod position and total power generation, orificing may now be predicted for full-power operation with confidence.

Pile oscillator techniques have been extensively used in determining stability.[14] Separation of fuel, coolant, and moderator power and temperature coefficients has been demonstrated, as well as over-all stability of the reactor under any conceivable operating condition.

Even under extremely adverse conditions, such as those present immediately prior to the fuel damage, the reactor was completely controllable. Although the moderator temperature apparently rose significantly, no runaway tendencies were observed. Completely stable operation of the reactor was observed at all times, except during the few minutes in which the actual fuel damage occurred. This damage apparently resulted in clearing blocked channels to some extent and permitted stable temperatures to be maintained in the core structure despite the existence of severely damaged cladding. A complete discussion of the reactivity fluctuations has been prepared.[9]

Conclusions

Operation of the SRE for three and one-half years, including a period of recovery from extensive fuel damage, has demonstrated the following:

1. The reactor system is both stable and responsive to power changes.
2. Maintenance and repair of sodium systems uncontaminated by fission products is a routine procedure.
3. Release of fission products to the primary system of a sodium-graphite reactor does not result in hazards to operating personnel or to the general public when the SRE containment philosophy is employed; the sodium apparently traps the bulk of fission products released from the fuel.
4. Detection of fission-product release by monitoring the cover gas is effective.
5. Repair of contaminated sodium reactor systems is feasible.
6. Prediction of sodium-graphite reactor behavior, both in normal and abnormal situations, has been confirmed.
7. Special containment measures, other than those currently employed, are unnecessary.
8. The sodium-graphite reactor concept has been demonstrated, by operation of the SRE, to be technically feasible.
9. The SRE is conservatively designed with respect to power capability.