SRE FUEL ELEMENT DAMAGE

FINAL REPORT

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SRE FUEL ELEMENT DAMAGE

FINAL REPORT

OF
THE ATOMICS INTERNATIONAL AD HOC COMMITTEE

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ABSTRACT

Following power run 14 on the Sodium Reactor Experiment (SRE), 13 of the 43 fuel elements were found to be damaged. An Ad Hoc Committee was established to assist in determining the source of the difficulty, to review and advise on steps taken to return the reactor to operation, and to recommend any necessary changes in the reactor system or in operating procedures to avoid a recurrence. An Iterim Report was issued, "SRE Fuel Element Damage," NAA-SR-4488, which presented the evidence available at that time as to the cause of the damage.

This Supplement is the concluding report of the Ad Hoc Committee. The conclusion as to cause of the fuel element failures has been modified to some extent on the basis of data developed since the first report was written. Several modifications have been made to the reactor system to avoid a recurrence.

I. INTRODUCTION AND CONCLUSIONS

During the course of power run 14 on the Sodium Reactor Experiment (SRE) at low power, temperature differences among various fuel channels were found to be undesirably high. Normal operating practices did not succeed in reducing this temperature difference to acceptable values, and on July 26, 1959, the run was terminated. A series of fuel element inspections was begun to ascertain the cause of these circumstances, and 13 of the 43 fuel elements were discovered to have suffered substantial damage.

On July 29, 1959, an Ad Hoc Committee was appointed to:

a) Assist in analysis of the existing situation in the reactor and determination of its origin;

b) Review and advise on steps taken to remedy the situation and bring the reactor back into operation;

c) Recommend any necessary changes in operating procedures or the reactor system to prevent occurrence of a similar situation.

On November 15, 1959, the Committee issued an interim report, "SRE Fuel Element Damage," NAA-SR-4488,[1] which reported on the origin, the nature and consequences of the damage to the SRE fuel based on activities, data gathered, and evaluations performed to October 19, 1959. The Ad Hoc Committee has now completed its investigations. This final Committee report revises and supplements the earlier interim report.

Since the publication of the interim report, sodium was drained from the reactor system and the top of the reactor remotely inspected and cleaned. Sixteen moderator cans were removed and replaced. Sodium coolant was returned to the reactor system for continuous circulation with cold trapping for impurity removal. Information on recovery equipment and techniques used will be published.[2]

As a result of the evaluation of the data accumulated during the first core operation of the SRE, a number of system modifications have been made.[3] These changes include elimination of tetralin as a service coolant, modification of the fuel element design, installation of a continuous cover gas monitor, and installation of a number of instruments designed to provide more detailed data on reactor

operation. As a result of these changes, an addendum to the hazards report has been prepared.[4]

Metallurgical examinations of the fuel, moderator cans and other components of the reactor have been made and results reported.[5,6] Filtering and cold trapping of the sodium coolant have resulted in removal of the bulk of insoluble contaminants and fission products from the coolant. Strontium-90 has deposited to the extent of about 0.8 $\mu c/cm^2$ in the primary system. A report discussing the movement of fission products in the primary sodium system will be published.[7] Reactivity changes have been investigated more completely, and mechanisms for these changes have been postulated. A report[8] has been written summarizing these conclusions.

Conclusions developed from the Ad Hoc Committee investigation into causes of fuel element damage are as follows:

a) The fuel elements failed as a result of thermal cycling through the uranium α-β transformation temperature and by formation and melting of Fe-U alloys.

b) Both effects were caused by partial blockage of coolant passages by tetralin decomposition products.

c) The high temperature runs (12 and 13) on SRE did not contribute to the fuel cladding failures of run 14.

d) The reactor excursion during run 14 is explained by the expulsion of sodium from several of the partially blocked fuel channels.

e) The 1.2% loss in reactivity incurred during run 14 resulted from penetration of sodium into moderator assemblies R-10 and R-42.

f) Carburization and nitriding of the stainless steel in the primary system were negligible. Some hydriding of the zirconium moderator cans occurred. The probability of can failures may have increased as a result of the hydriding.

g) In spite of the cladding failure to 13 fuel elements and the release to the primary coolant of several thousands of curies of fission product activity, no radiological hazard was presented to the reactor environs. Recovery operations were conducted by SRE operating crews, working within standard AEC regulations on radiation exposure.

h) Modifications to the system have reduced the probability of a similar major fuel element failure to a negligible value.

After a final committee review of causes and effects of fuel element damage, modifications to the reactor system, changes in the operating procedures, and revisions of organization and lines of authority at the SRE, the Committee recommended approval for operation. Approval for operation was received from the AEC. On September 7, 1960, the SRE was made critical with a Th-U loading.

II. CHRONOLOGY OF EVENTS SINCE ISSUANCE OF INTERIM REPORT

In the first report a chronology was presented of pertinent events in the reactor history previous to the fuel cladding failures and of the initial steps taken in restoring the reactor to operating condition. The chronology is continued in this report. Only those operations at the reactor site are discussed in this section; supplementary examination, analyses, and experiments in the hot cell and other laboratories are discussed in Section IV. Major steps in the recovery operations are shown in Figures II-1 and II-2. Throughout the entire operation, strict Health Physics surveillance and procedures were followed. Only very light contaimination of personnel occurred. The few cases which did occur were quickly detected and remedied. No one received a radiation dosage greater than standard AEC tolerance.

A. FUEL ELEMENT REMOVAL

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. It was decided, therefore, 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 which 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.

B. SURVEY AND CLEANUP OF THE REACTOR

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 using the fuel handling cask and a

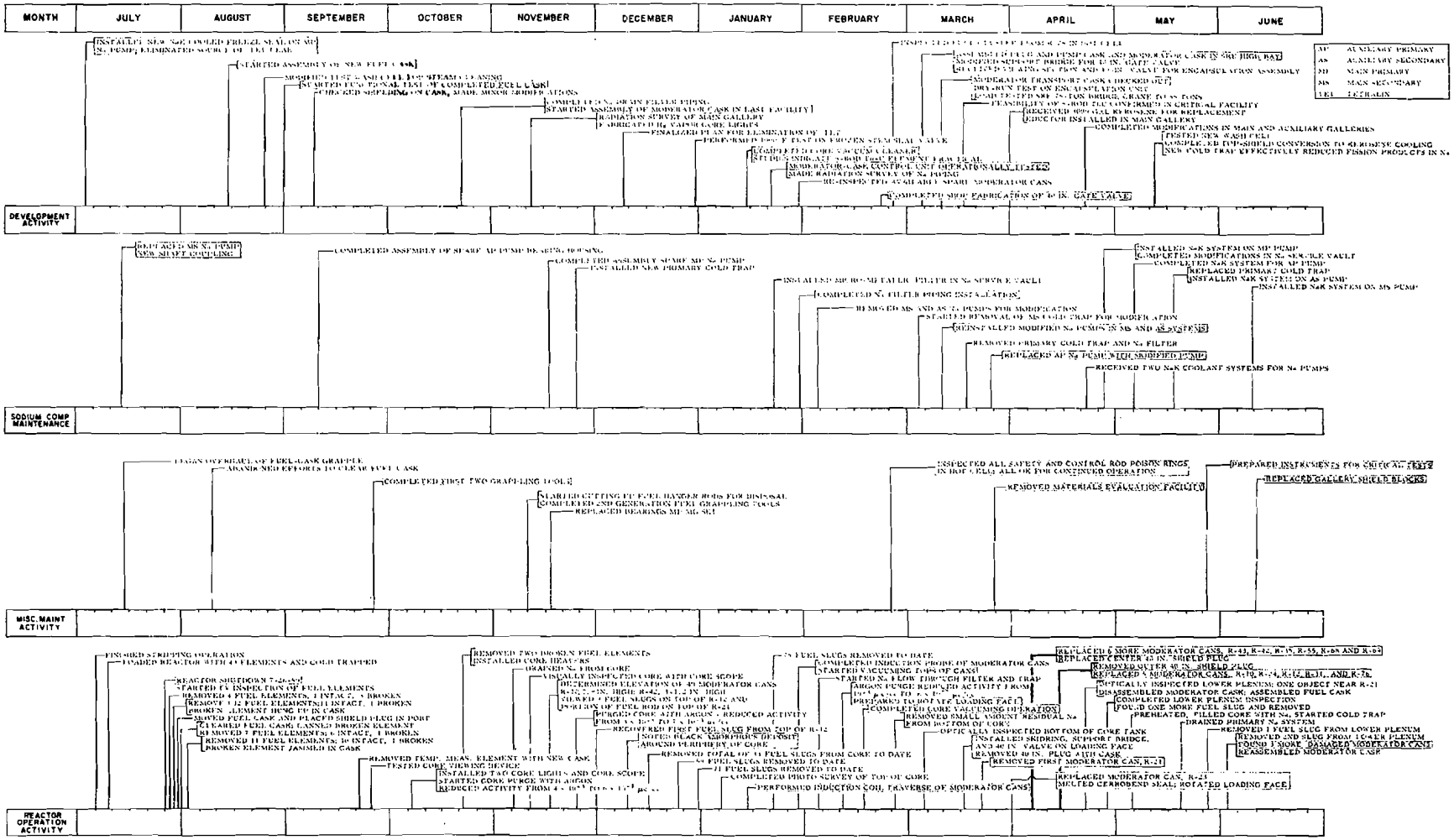


Figure II-1. Chronology of SRE Recovery Operations (FY-1960)

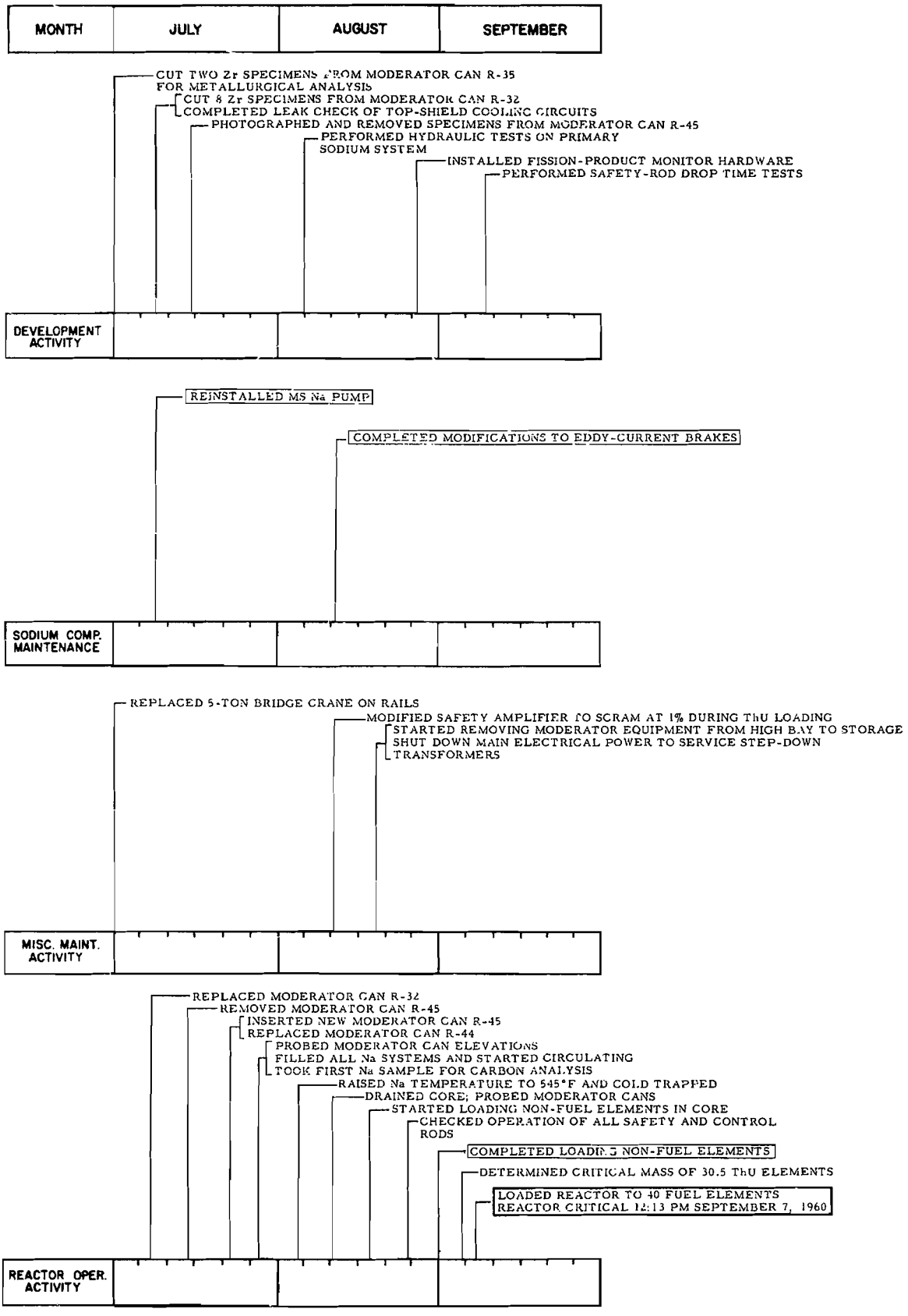


Figure II-2. Chronology of SRE Recovery Operations (FY-1961)

glass window 1/2 in. thick was installed. The radiation level above the glass was about 50 mr/hr. The contamination in the cover gas was determined to be about 4×10^{-3} $\mu c/cm^3$ of Kr^{85} in helium; replacement by argon reduced activity to about 7×10^{-4} $\mu c/cm^3$. 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, and scattered at random, were about 82 separate fuel slugs, some bits of fuel cladding, and several pieces of wire wrap. The 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, 30-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., 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 2 lb of carbon, estimated from the associated radioactivity, were removed from the system by this means.

C. REMOVAL AND REPLACEMENT OF DAMAGED MODERATOR ASSEMBLIES

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% dilation of graphite accompanying sodium absorption. Both methods indicated leaks in cans R-10, which had lengthened 7/8 in., and R-42, which elongated 1-1/2 in.

Equipment for removal of the 40-in. plugs and the moderator elements is shown in Figures II-3 and II-4. Use of this equipment is described in Reference 2.

The moderator-can encapsulating station was equipped for viewing the cans and obtaining zirconium samples.[6] 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 of 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 of 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 grid 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 which 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 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 establishing 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 pumpout tubes during fuel slug grappling operations. The third can (R-45) was

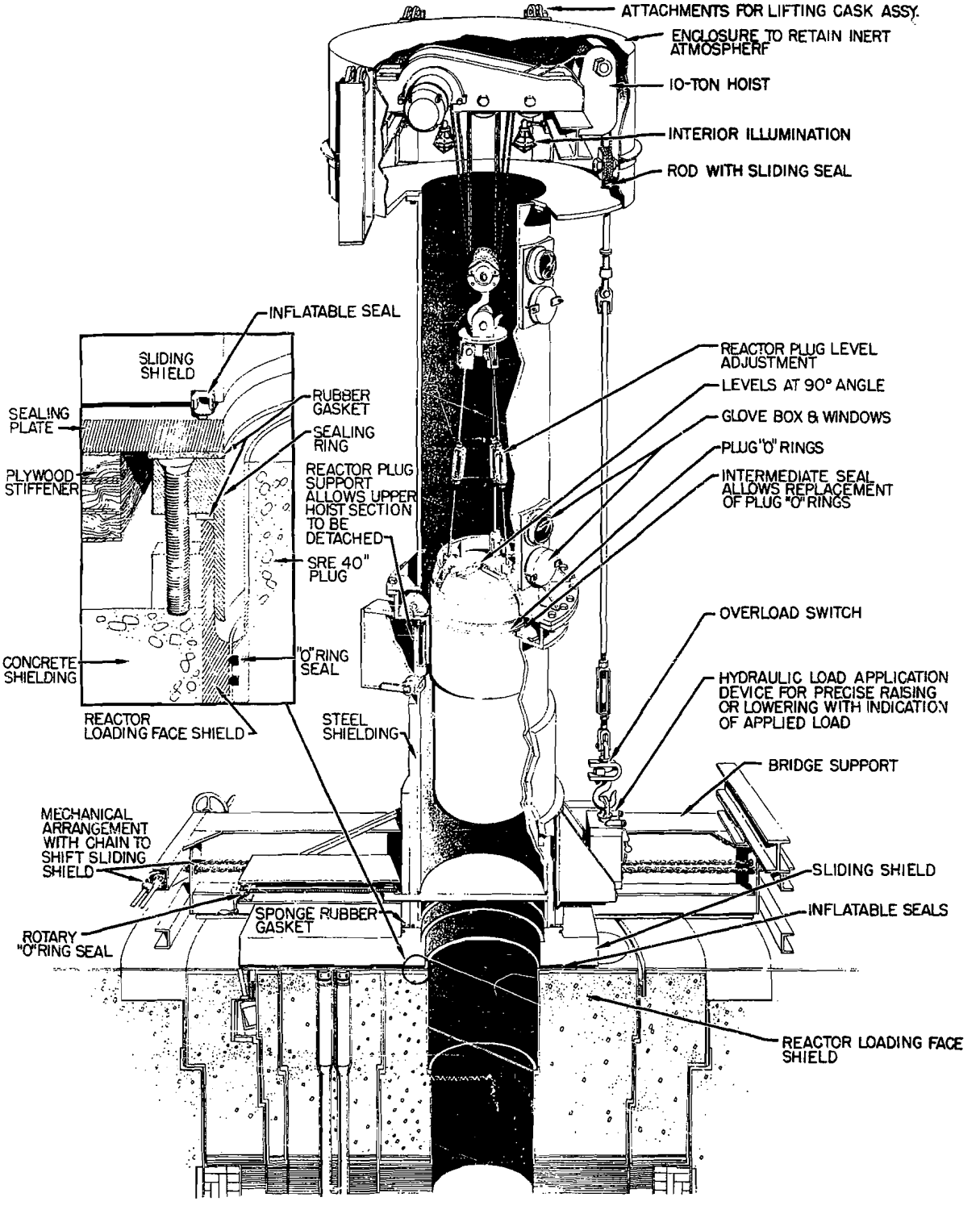


Figure II-3. Equipment for Removing Moderator Cans

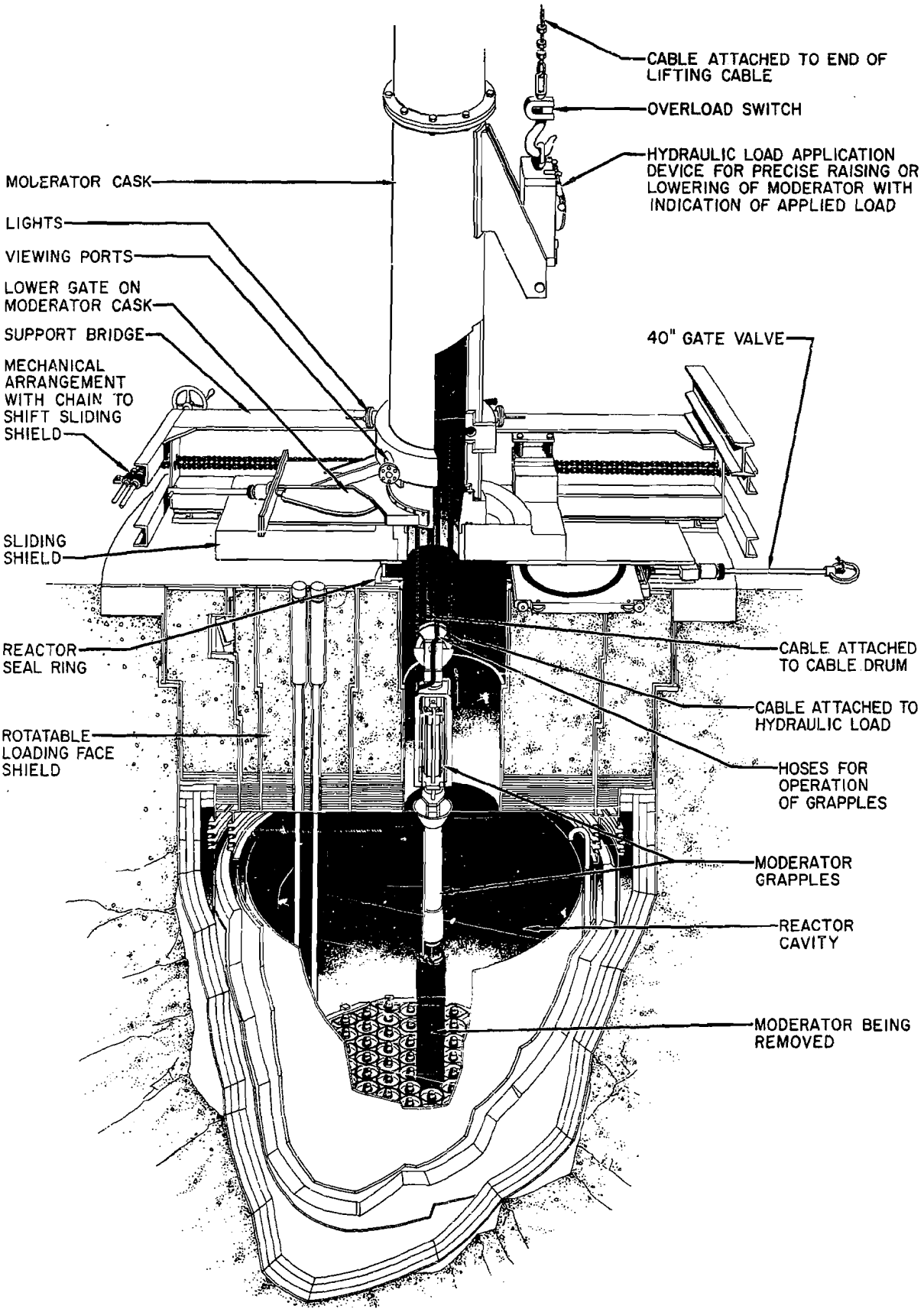


Figure II-4. Removing a Moderator Can

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.

D. CLEANUP OF PIPING AND EQUIPMENT

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 provided a ready means of locating carbon deposits in the piping and equipment. Measurements (made in a 10-in.-diameter thimble which penetrates the main gallery) showed that the gamma field at the main intermediate heat exchanger decreased from 2.5 r/hr to 1.8 r/hr while the cold trap increased from 68 r/hr 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: hot trap economizer 7.0 r/hr to 0.95 r/hr; main primary pump discharge 18 r/hr to 7.0 r/hr; and, bottom of the pump case 0.4 r/hr to 0.16 r/hr. At the same time, the radiation level of the replaced cold trap increased from 2 r/hr to 11 r/hr, again indicating the transport of carbon from the piping to the cold trap.

III. DATA AND EVALUATION

A. FUEL DAMAGE INFORMATION

Since the Interim Report was written, more information has been obtained on the mode of failure of the fuel elements. A tentative conclusion of the Interim Report was that the elements had failed through diffusion of uranium into the cladding to form low melting alloys. Such alloys had been identified through chemical analyses of pieces of cladding and wire wrap taken from the area of failure. It was stated that the high temperatures required for the diffusion process had been produced by local plugging of the coolant channels by tetralin decomposition products, and photographs of such plugs were shown. Much of the new information was obtained from the examination of a complete element in the Component Development Hot Cell (CDHC). The additional data demonstrate that temperature cycling was a major contributor to the failures.

1. Examination of Element From Channel R-24

When the parent report was written, two complete fuel elements remained stuck in their coolant channels. These were the elements in channels R-24 and R-76. They were removed from the core along with the associated moderator assemblies.

The assembly containing channel R-24 was moved to the CDHC where it was opened in an inert atmosphere. The moderator assembly was sectioned axially and the outer can and graphite removed. The zirconium process tube was then cut axially and the tube opened, exposing the fuel element.

The results are shown in Figure III-1. Several features should be noted: (a) the solid plug of material about 1-ft above the bottom of the fuel rods, (b) the Fe-U alloy melt-through area about one-third of the way up the element and extending for several inches, (c) the region above the melt-through where the cladding had been distended until it burst, and (d) the black spongy plug in the channel below the element.

The solid plug was determined metallographically to consist of Fe-U alloy of near-eutectic composition. Its source was the melt-through area above, from which Fe-U alloy, formed near the surface of the slugs, penetrated and destroyed the cladding.

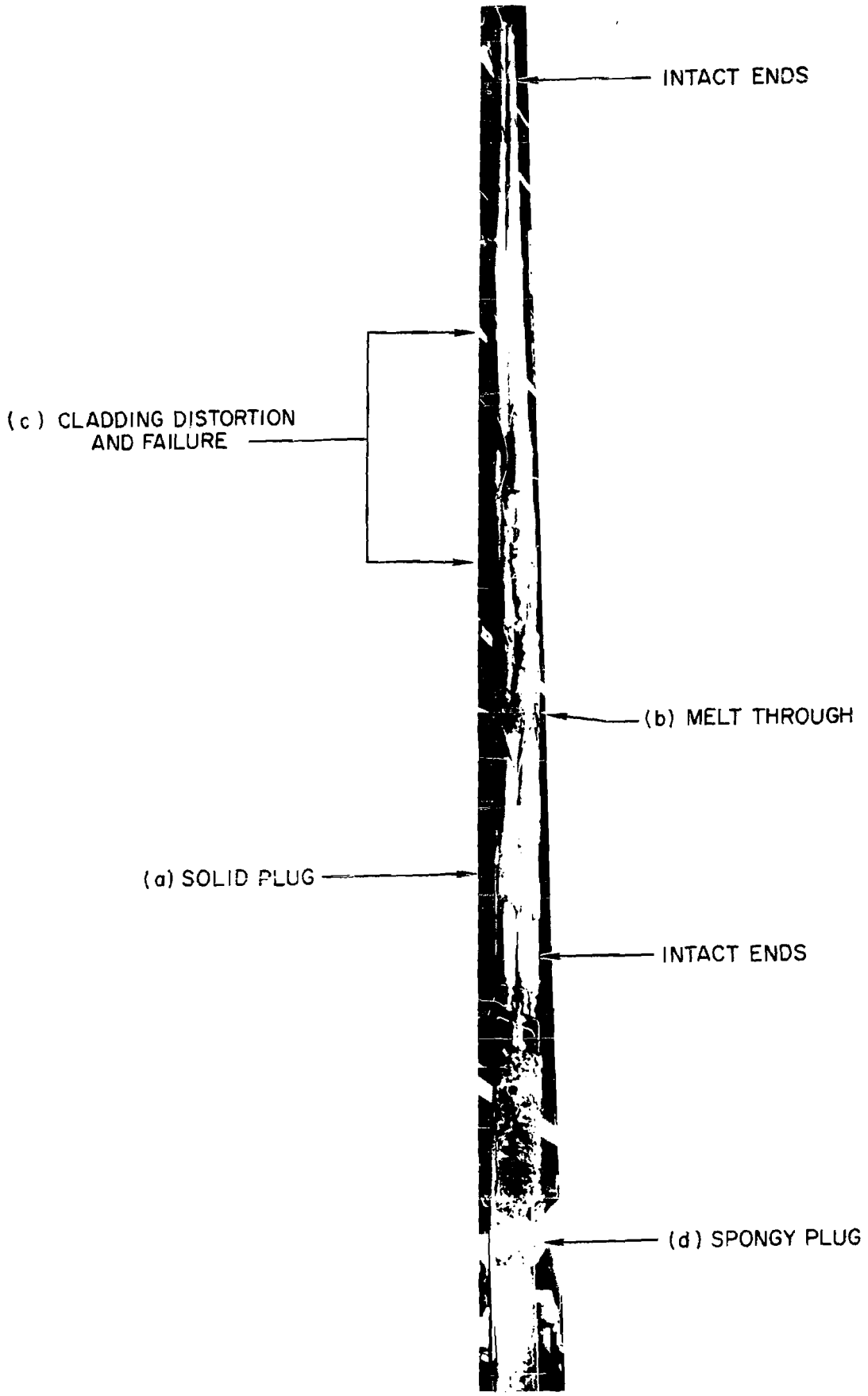


Figure III-1. Fuel Element From Channel R-24

The cladding breaks above the melt-through area were typical of a ductile material, and cladding pieces from this area were readily bent with the hot cell manipulators. There was no evidence of Fe-U diffusion in this area; the failure is attributed to swelling of the fuel from thermal cycling.

Part of the spongy plug was immersed in alcohol in which it partially dissolved, indicating a substantial sodium content. The insoluble residue appeared to be primarily carbon. The porosity of the plugging material to sodium flow is clearly demonstrated by the fact that the channel drained when sodium was removed from the reactor.

It cannot be assumed that either the location or physical character of the plugging material at the time of fuel element failure was necessarily the same as that observed in the hot cell.

2. Supplementary Experimental Results

In addition to the hot cell examination of element R-24, metallographic and chemical analyses of small sections of fuel element were made for evidence of carburization, nitriding and Fe-U diffusion. Experiments on Fe-U diffusion couples were also conducted with unirradiated material, and the microstructure was found to be similar to that of specimens taken from the broken elements.

A thermal cycling test was also conducted on unirradiated uranium. Two capsules, similar in geometry to an SRE fuel rod, were thermally cycled between 900 and 1300°F, crossing the α-β phase transformation temperature of 1220°F, with a frequency of about 4 cph. The cladding burst within 24 hr.

Measurements on individual slugs removed from the reactor showed substantial increases in diameter. One such slug is indicated in Figure III-2. The surface is severely cracked, giving it a bark-like appearance. Diameter measurements on this slug ranged up to 0.95 in.; the original value was 0.75 in. The distorted slugs were severely out of round; in one case the wire wrap was deeply embedded in the clad slug. The highly localized nature of the high temperature regions in the fuel channels is demonstrated by the slug shown in Figure III-3, one end of which is in good condition.

Bend tests on sections of cladding taken from regions distant from eutectic failure areas showed good ductility. Specimens found to contain uranium were brittle, particularly those of near-eutectic composition. The eutectic alloy has

Figure III-2. Fuel Slug Showing Effects of Thermal Cycling

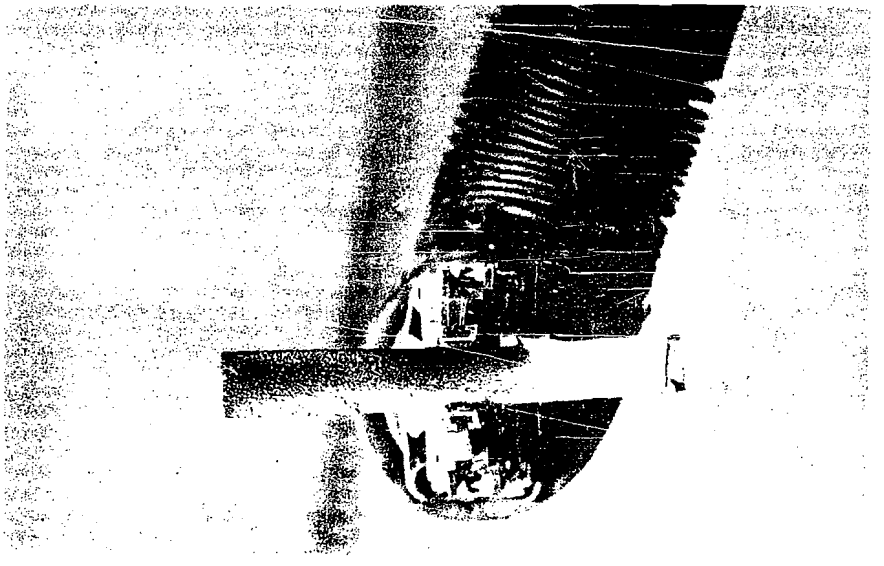


Figure III-3. Fuel Slug From Element From Channel R-21

virtually no ductility. Specimens taken from the burst-cladding area of the element from R-24 showed good ductility, indicating no alloying had occurred.

Several specimens of cladding were examined chemically and metallographically for evidence of carburization and nitriding. The results were negative for all specimens except those taken from an area of failure; such specimens were found to contain as much as 0.3% carbon, about four times the carbon content of the original material.

Four undamaged elements, removed from the reactor after run 14, were examined in detail. Cladding and fuel slugs from these elements were completely normal.

3. Effects of High Temperature Runs

In runs 12 and 13 the reactor outlet temperature was increased from the nominal operating value of 960°F to about 1000°F. For a period of 55 min during run 12 the outlet temperature was raised to 1065°F. There is good evidence that these runs did not contribute in any way to the damage incurred during run 14.

It was known that unalloyed uranium would swell in the SRE irradiation environment, and several selected elements were periodically checked for swelling by measuring the outside diameter of the fuel rods with a micrometer. Several of these were measured before the start of run 12 and showed a diameter increase of a few mils. They were measured again at the end of run 12, and no further diameter increases were found. It was on this basis that the higher reactor outlet temperature was again approved for run 13.

Following run 13 preparations were made to measure the elements again, but the first element washed (from R-56) was severely damaged in the wash cell. Recent metallographic examinations of cladding specimens from this element, taken from an area where the cladding had ruptured in the wash cell, showed no evidence of uranium reaction with the cladding. Following run 14 the rods of four undamaged elements were measured, and no dimensional changes were noted. These tests demonstrate quite conclusively that the high-temperature runs did not contribute to the fuel cladding failures.

4. Mode of Element Failure

The evidence clearly points to two modes of cladding failure, (a) thermal cycling through the uranium α-β transformation temperature which caused the

fuel to expand until the cladding burst, and (b) diffusion of uranium into the 304 stainless steel cladding to form low-melting alloys. Both types of failure were caused by the presence of plugging material in the coolant passages of the fuel elements.

The outlet channel temperature data, analyzed in detail after publication of the first report, reveal two significant facts: (a) the flow of coolant in the channels which contained broken elements was at least 30% of nominal full flow, and (b) the outlet temperatures of some of the partially plugged channels cycled through a range of about 50°F.

The fuel channel outlet thermocouple readings can be used to determine sodium flow rate in each channel. Partial plugging of the channel is reflected in a high outlet temperature. Using the thermocouple information in this manner, it was established that the minimum flow (in the most severely plugged channel) was never less than about 30% of nominal full flow. The validity of the thermocouple data was then checked by comparing the sum of flows through all channels, as calculated from thermocouple readings, with the total flow through the reactor. Three such comparisons, for three different occasions in run 14, agreed within 3% in each case. The possibility that the flow rate in several plugged channels was less than about one-third of nominal full flow is thus very small. With one-third of nominal flow through the channel at the power levels of run 14 (less than 2 Mw for most of the run), the design temperature limits for the fuel element could not be exceeded unless the plugs interfered locally with heat transfer. To reach the fuel temperatures observed, it was necessary for a plug to have occupied the region of failure at the time of failure.

Physical characteristics of the plugs in the fuel channels during reactor operation could be determined only in a very general way. The basic material of which the plugs were formed was probably particulate carbon; in addition long-chain hydrocarbons were almost certainly present at the start of run 14 and some probably persisted in this form throughout the run. The material was somewhat permeable to sodium flow as indicated by the observations of the blockage in channel R-24.

The plugs must have been large in order to reduce flow in a given channel to one-third of nominal flow. The relationship of flow to plug geometry in a channel is indicated in Figure III-4. As shown in the figure, a very short plug (thin

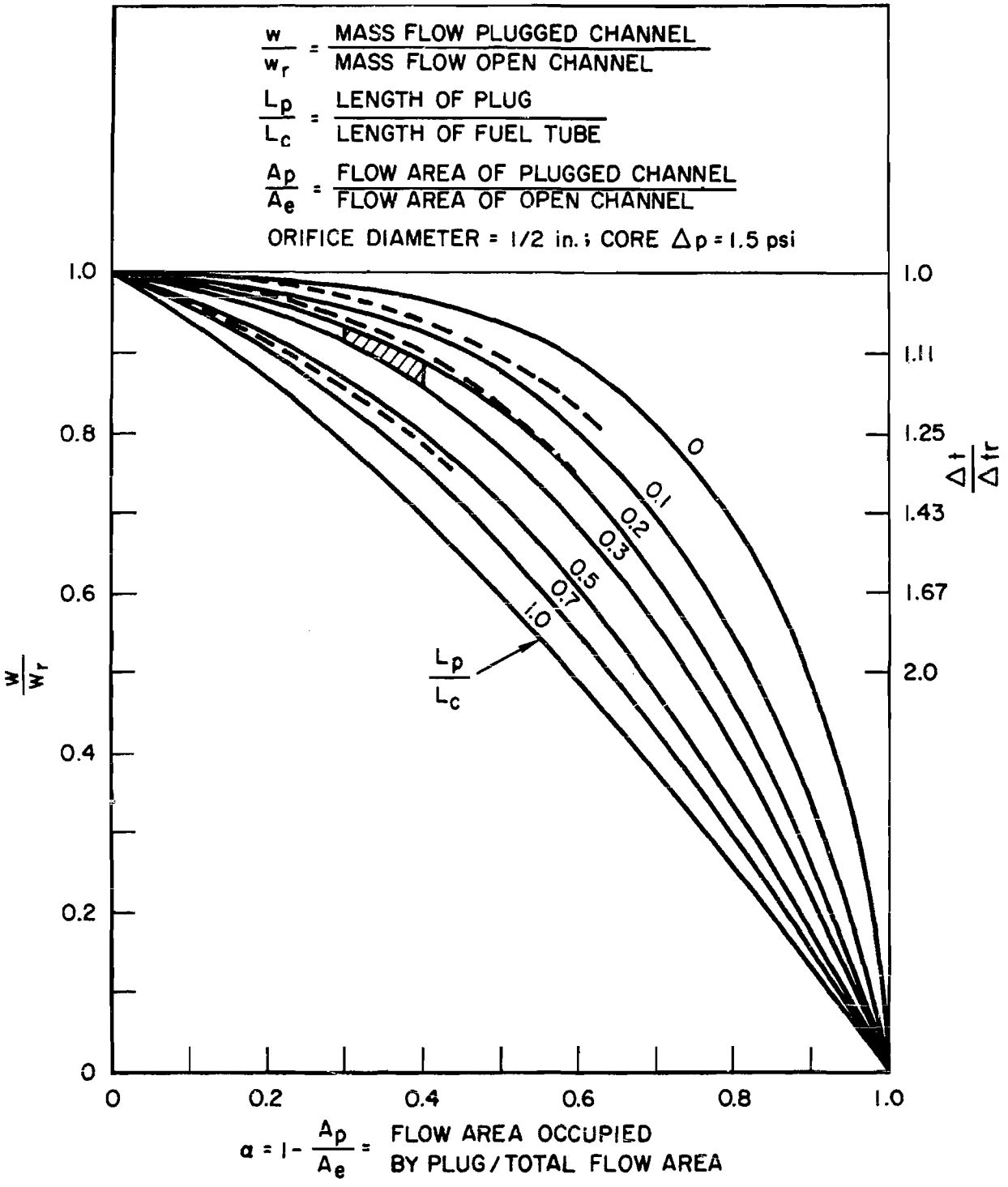


Figure III-4. Mass Flow Reduction in SRE Fuel Channel Due to Formation of Plugs in Fuel-Tube Region

plate) would have to occupy about 95% of the flow area around the fuel rods to reduce flow to one-third; if the plug were as long as the channel, it would have to occupy about 75% of the flow area to reduce flow to one-third. The calculations leading to Figure III-4 assumed no plug porosity. With some flow through the plug, the plug would have to occupy a still larger fraction of the channel to restrict flow a given amount. The few observations of the spongy plugging material, in channel R-24 and on top of the core, together with the data of Figure III-4, would indicate that the plugs were probably long (a few feet) and occupied most of the flow cross sectional area.

Temperatures reached in the blocked region of some of the elements are indicated by data from the element in R-55. On July 22, 1959 with the reactor power level at about 4 Mw, the outlet temperature for this channel was about 850°F, indicating about 50% of nominal full flow. In-fuel thermocouple No. 10, located 15 in. from the top of the element, was reading 1200°F at this time. Thermocouple No. 9, 27 in. from the top of the element, was reading 1465°F. The upper end of the region of cladding failure was about 1 ft below thermocouple No. 9 and in a region of higher neutron flux. A linear extrapolation of the temperature data would yield a value of 1730°F at this point. Thermocouples 9 and 10 were located on the slug centerline, and surface temperatures were somewhat lower; however, even at the reactor midplane in the central channel, the temperature drop from center to surface of the fuel slug is less than 100°F at 4 Mw. The length of the region of failure for this element is not known, but if similar to the element from R-24, it extended for a length of about 2 ft. At any rate, it is clear that still higher temperatures were reached in the region of failure, and it is likely that sodium boiling occurred in that region.

The concept of sodium boiling in channel R-55 while flow was 50% of nominal full flow throws some light on the nature of the plugs in the channels. Figure III-4 would indicate that, with 50% flow, from 60 to 90% of the sodium flow area was blocked. If some of the flow was through the plug, which Figure III-4 neglects, the plug occupied a still larger fraction of the flow area. To reach temperatures above the 1465°F indicated by thermocouple No. 9, a substantial degree of insulation of fuel rods from the flowing sodium was required. Such insulation could not be obtained with a plug containing substantial amounts of liquid sodium, and it is difficult to escape the conclusion that the plug contained gas bubbles. The gas could have been helium or hydrogen but sodium

vapor seems a likely contributor. Vapor, once formed in a small section of the plug, would tend to propagate throughout the plug volume.

With the limited evidence at hand, the conclusion is reached that the plugs (a) occupied most of the cross sectional flow area, (b) were relativcly long, and (c) contained a substantial volume fraction of gas.

A careful review of the chart from run 14 showed that all of the channels in which fuel failures occurred had shown fluctuations in outlet channel temperatures. The record of temperatures from channel R-24 for a typical 24-hr period is shown in Figure III-5. Fluctuation in outlet channel temperatures also occurred for a few hours in some channels in which the fuel elements did not fail. The element from R-25, in which temperature fluctuations occurred for only part of the run, suffered the least damage of the thirteen damaged elements. The temperature history for R-25 over a 24-hr period is shown in Figure III-6 Some indication of the relation between fuel temperatures at some regions of the element and the outlet channel temperatures can be gained from the data for R-55

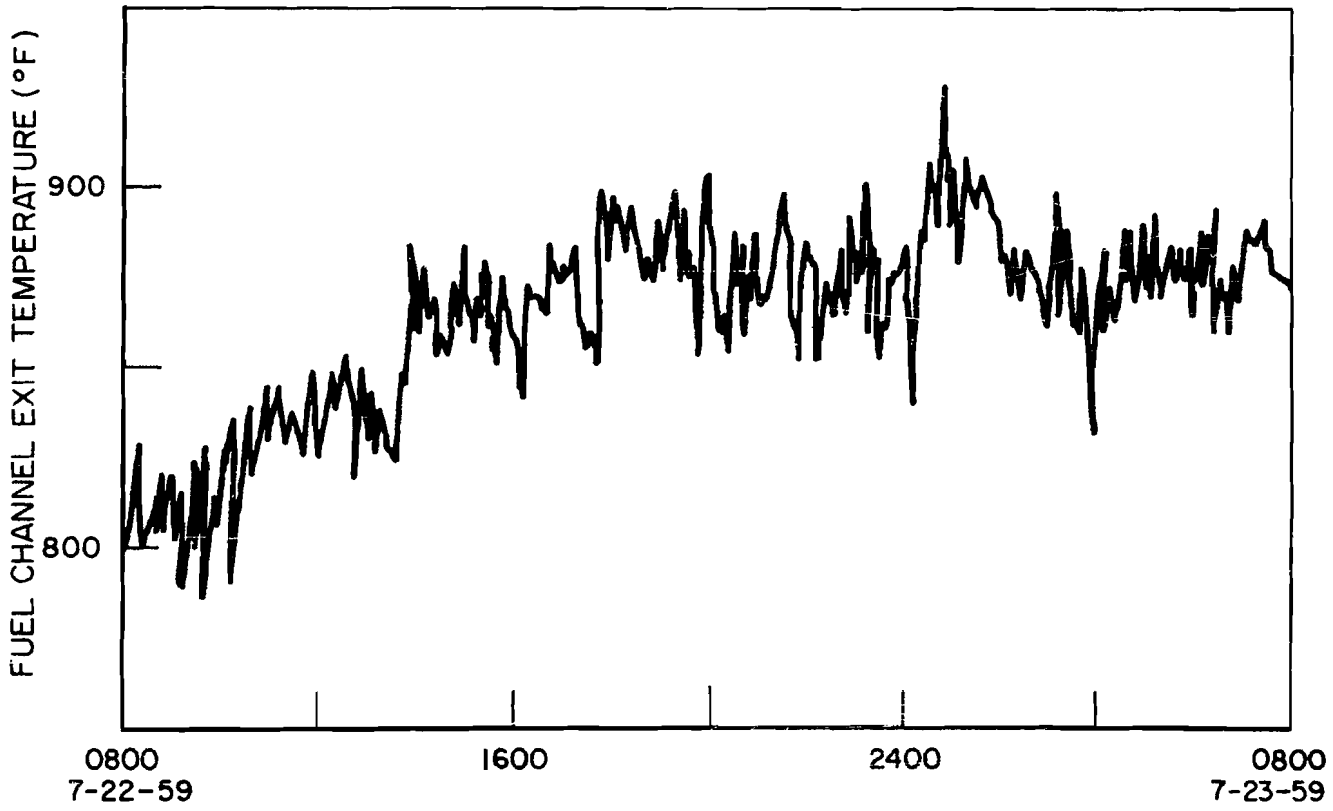


Figure III-5. Outlet Temperature for Channel R-24

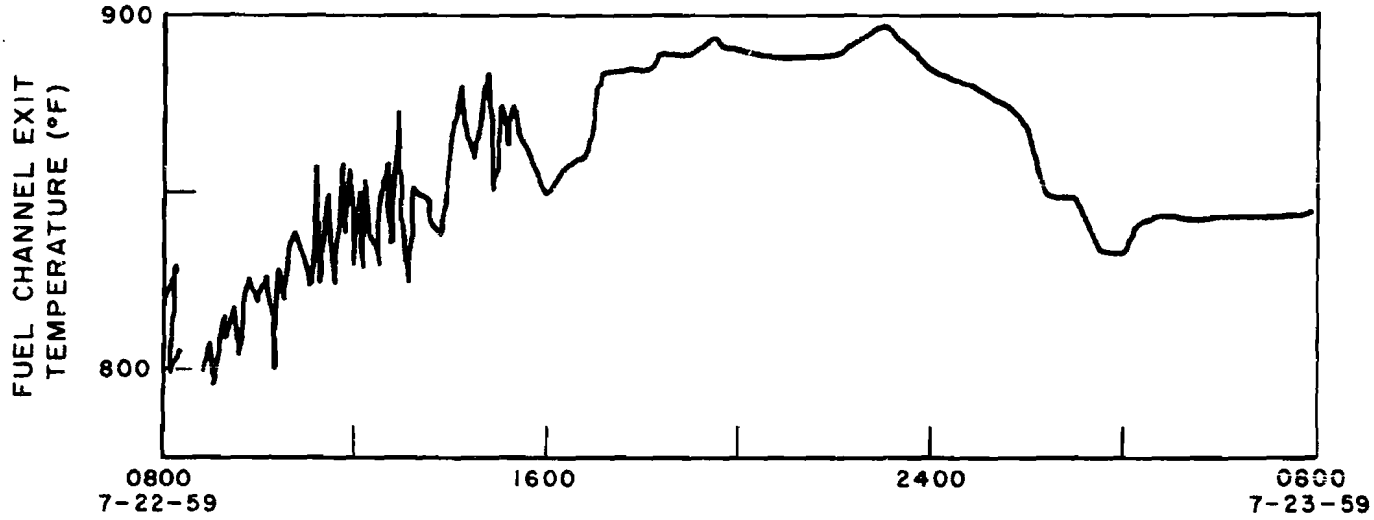


Figure III-6. Outlet Temperature for Channel R-25

shown in Figures III-7 and III-8. As indicated by the figures, the fuel temperature was cycling over a range of about 150°F while outlet channel temperature variation was about 50°F.

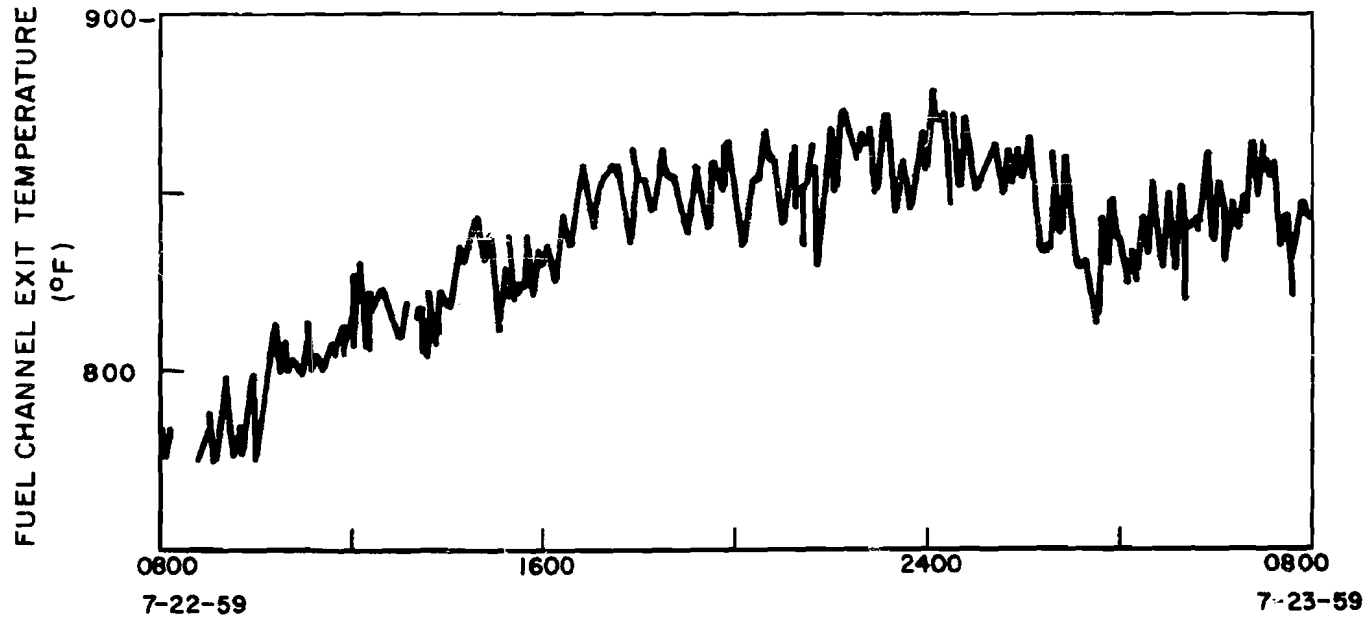


Figure III-7. Outlet Temperature for Channel R-55

Examination of the temperature charts from previous runs showed only one other period in which outlet channel temperatures fluctuated. This was during the early part of run 8, which was preceded by a tetralin leak.

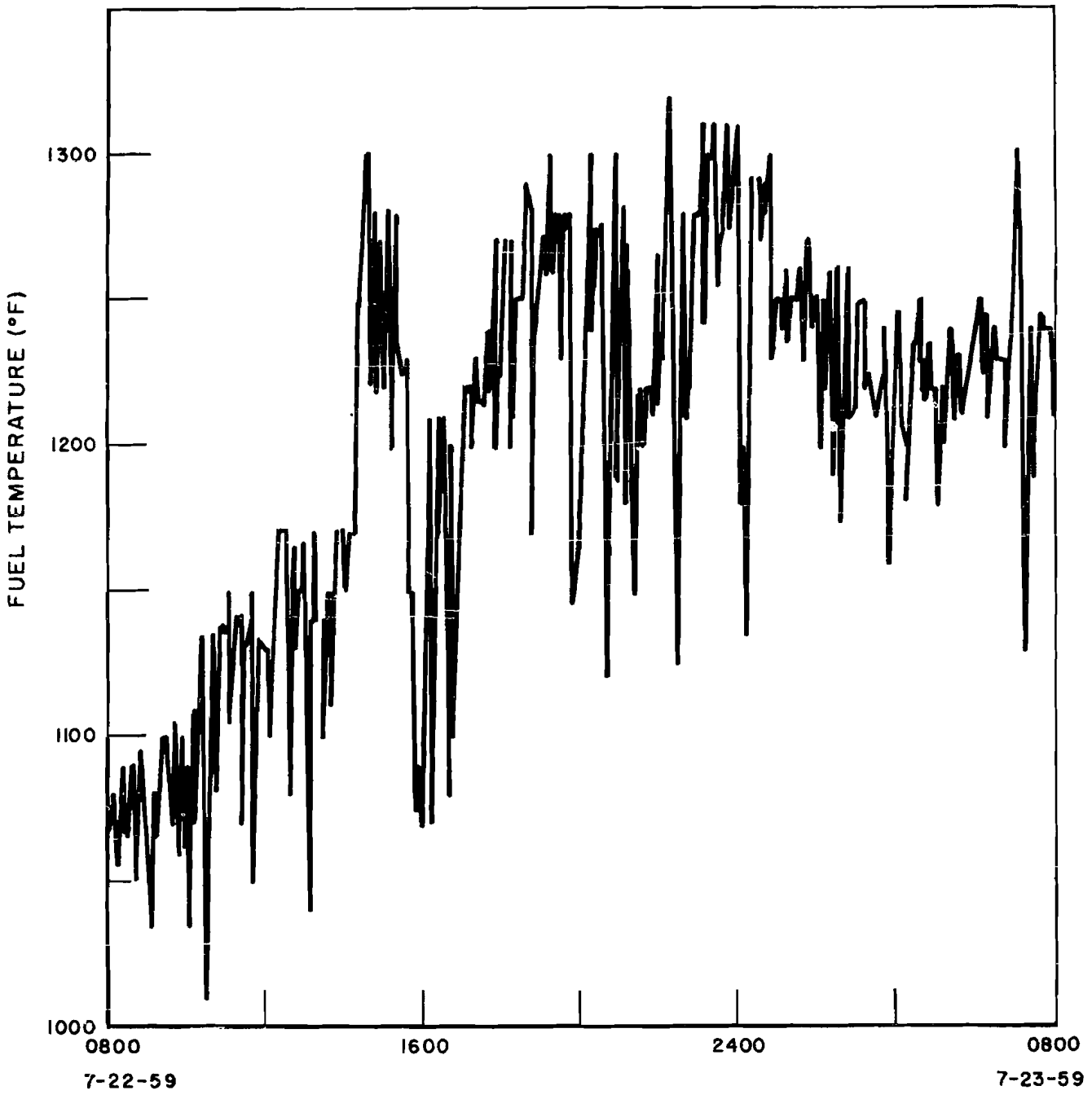


Figure III-8. In-fuel Temperature for Element in R-55

Cycling of the channel outlet-temperatures demonstrates a corresponding variation in flow. With the type of plug described in previous paragraphs, the variation in flow can readily be ascribed to boiling of sodium within the plug. Similar oscillations have been observed in an experimental sodium boiling-condensing apparatus at Atomics International and with a similar period.

The local boiling postulate is supported by evidence that boiling temperatures could have been reached and by the observance of outlet temperature oscillation. The boiling model also provides a logical explanation of reactivity changes which occurred during run 14, as discussed in Section IV-D. Whatever the mechanism, it is clear that thermal cycling occurred and with sufficient severity to cause cladding failure by expansion of the fuel.

The sequence of events leading to fuel element failures is thus believed to be as follows:

a) Tetralin slowly leaked into the primary sodium via a freeze seal on the main primary pump. The tetralin leaked into 1000°F sodium for a period of about 5 days. The reactor was shut down to repair the pump, the tetralin continuing to leak into 350°F sodium for an additional 9 days.

b) Nitrogen was admitted to the system to purge the sodium of tetralin and some of its decomposition products. Seventeen fuel elements were removed from the core for this operation. These were viewed in the handling machine and appeared to be in good condition.

c) The pump and fuel elements were reinstalled and run 14 started at low power. Carbon and other tetralin decomposition products, as well as quantities of oxide, were still present in the core, though undetected. Partial plugging of several fuel channels is evident from the spread in fuel channel outlet temperatures.

d) The partial plugs in the channels were located randomly in the core and in the channels. It is possible that some of the elements were plugged before the run started. A few of those which had been removed and re-inserted between runs 13 and 14 plugged shortly after re-insertion.

e) In addition to excluding coolant locally, the plugs resulted in fluctuations in coolant flow. The fluctuation in flow produced severe thermal cycling of the fuel. The thermal cycling caused the fuel slugs to expand to such an extent that the cladding tubes were ruptured. The severity of the fuel distortion forces the conclusion that the fuel was cycled through the uranium α-β phase transformation temperature (1220°F). In at least one area (the "burst" area of the element

from R-24) the temperatures were not sufficiently high to promote accelerated Fe-U diffusion.

f) The plugging interfered locally with heat transfer to such an extent that very high (above 1400°F) temperatures were reached in several places even at low power, and Fe-U alloys were formed and melted.

The Fe-U alloying and thermal cycling are probably inter-related. It is quite possible that the cycling and distortion occurred first, further restricting coolant flow, increasing local temperatures, and improving contact between fuel slugs and cladding. Much of the background for the above general picture was obtained from examination of the element from R-24. It is considered that adequate information is in hand for understanding what happened and for preventing a recurrence.

B. REACTOR COOLANT CONTAMINATION

1. Contaminants in SRE Sodium

Impurities potentially present in SRE sodium, which could contribute to metallurgical changes in structural members are carbon, hydrogen, nitrogen, and oxygen. Carbon is present primarily as a result of thermal decomposition of tetralin and possibly from dissolution of graphite from ruptured moderator cans. Hydrogen is also a decomposition product of tetralin. Nitrogen was introduced during the attempt to remove the tetralin by stripping. Some oxygen was introduced along with this nitrogen, but in-leakage of air during shutdown operations was also a major contributor of both elements. Fission products (in the system since the fuel elements were damaged) are not present in amounts sufficient to cause any significant metallurgical changes in the structural materials in the system.

It is estimated from plugging temperature data that the primary sodium was saturated with oxygen upon circulation of sodium during the critical experiments. The cold trap was operated until the system oxygen content was reduced to an acceptable operating level of approximately 20 ppm, in February 1961.

a. Effect of Carbon on 304 Stainless Steel

Two types of carburization, grain boundary and volume, have been observed in 304 stainless steel exposed to carbon-saturated sodium at elevated temperature.[9] It has been found that grain boundary carburization has negligible

effects on mechanical properties. The brittle case formed by volume carburization can produce two characteristic effects on mechanical properties, depending on the ratio of case depth to section thickness.

Relatively thin carburized layers cause weakening of the section when plastically strained. This appears to result from cracking of the case and consequent reduction in cross-sectional area. Under fatigue and stress-rupture testing of smooth and notched specimens, however, the cracks seem to have no tendency to propagate in the ductile base material.

Carburized specimens with relatively high ratios of case depth to section thickness exhibit higher strength and lower ductility than uncarburized material. The ratio of case depth to section thickness which appears to be a conservative limit for ductile behavior under plastic strain is approximately 10%. On this basis the 0.010-in.-thick fuel jackets could tolerate 0.001-in. depth of carburization.

Studies of the carburization of 304 stainless steel in sodium[9] have provided information necessary to calculate the surface carbon content of a carburized specimen, which can then be related to the carbon content of the sodium. Each of the following combinations of data can be used for the calculation of surface carbon content:

1) Mean carbon content of the full specimen thickness, time, and temperature of exposure.

2) Depth of volume carburization metallographically detectable, time and temperature of exposure.

3) Surface hardness of carburized specimen.

4) Hardness vs distance from surface, time, and temperature of exposure.

Each of these combinations was used to obtain the surface carbon content of a sample of the SRE hot-trap inlet pipe, removed from the system after run 14. This component was operated at 1200°F for 450 hr. From these results an estimate of the effective carbon content of the sodium was made. The four methods yielded results in the range 20 to 25 ppm for the effective carbon concentration in the sodium.

A sample of the hanger rod from the fuel element in R-24, cut from a section which was submerged in the sodium, was also subjected to hardness measurement traverse. From these data an estimate of 20 to 25 ppm was made for the carbon content of the sodium. This element was in the reactor during the period covering runs 9 through 14. The system is known to have contained particulate carbon throughout this period.

The following table shows time at temperature which produces 0.001 in. of volume carburization of 304 stainless steel for various concentrations of carbon in sodium. These data are based on capsule tests in which no thermal gradient existed.

TABLE III-1

HOURS TO CARBURIZE TO 0.001 IN.*

Temperature (°F)	Carbon in Sodium		
	22 ppm	40 ppm	Saturated
1200	30	8	6
1150	50	13	11
1100	90	24	20
1000	295	78	69
900†	1180	311	290
800†	5680	1490	1390

*To convert time for any case depth, multiply by the square of ratio of desired case depth to 0.001 in.
†Extrapolation of diffusion data.

The carburizing potential of the SRE sodium remains in doubt as visual evidence of solid carbonaceous material in the sodium indicates the possibility of achieving saturation. To avoid possible extensive carburization of 304 stainless steel fuel cladding during operation with sodium at high carbon concentration, maximum operating temperature will initially be limited to 800°F at the sodium-cladding interface. This limitation will be modified after metallurgical examination of thin tabs exposed in the upper sodium pool. With the restrictions designed to protect the fuel cladding, there is no problem with excessive carburization of the intermediate heat exchangers or other parts of the primary system.

b. Effect of Nitrogen on Stainless Steel

Addition of nitrogen to austenitic stainless steel promotes susceptibility to cracking when exposed to relatively weak corrosive environments. No effect would be expected during operation of the SRE since a noncorrosive environment exists. During or following steam cleaning of a stainless steel component, such as a fuel element, corrosion would probably occur providing, of course, that nitrogen had diffused into the material.

Although traces of nitrogen may still remain in the sodium in the SRE primary system, no evidence of nitriding was obtained from metallurgical examination of a sample from the hanger rod of the fuel element in position R-31. Analysis of tabs of stainless steel suspended in the sodium pool over the core provides a means of detecting any nitriding that may occur during future operation of the reactor.

c. Effect of Hydrogen on Zirconium

The effect of increasing hydrogen content on the mechanical properties of zirconium is to decrease the ductility and fatigue life at room temperature. Above 250°F, the tensile properties are unaffected[6,10] and the ductility and fatigue life[11] are altered but little. Examination of the zirconium sheath of moderator assemblies and dummy fuel elements removed from the SRE was performed[6] to determine the extent of hydrogen absorption and the resultant effects on the mechanical properties of the material.

The general condition of the zirconium, as determined by visual observation of the moderator-can skins, appears to have been unchanged by 4500 hr of SRE operation. Breaks in the skin (see Section II) in the vicinity of the head welds on failed cans, resulted from tensile forces produced by swelling of the graphite. The swelling was due to interaction between the graphite and sodium. The cans stretched about 1 in. before failing.

Bend tests of zirconium cladding from two dummy fuel elements gave the first indication that the SRE zirconium was brittle at room temperature. The better ductility of the zirconium from the bottoms of the cans, which contained considerably less hydrogen than zirconium from other regions of the cans, indicates that the major part of the embrittlement was due to the pickup of hydrogen during SRE exposure.

Hydrogen is one of the gases which is absorbed in large quantities during manufacture of graphite. When the temperature of the graphite is increased, these gases are slowly released. Approximately 80% of the total gas absorbed initially in the SRE graphite was hydrogen, which is rapidly gettered by the zirconium. If all the hydrogen absorbed in the graphite in a moderator can were released and gettered by the zirconium, the hydrogen content of the zirconium would be approximately 400 ppm, assuming uniform distribution along the length of the can. A relatively uniform distribution of hydrogen, except at the very bottom, was found[6] in the zirconium of the dummy fuel element cans. Since not all of the gases contained in the graphite would be released at reactor temperatures, the hydrogen content of the SRE zirconium would be expected to be less than 400 ppm. The measured hydrogen content, approaching 1000 ppm at the tops of the moderator cans, indicates that another source of hydrogen was present in the SRE core. The zirconium from the dummy element can which was in the core during power run 14 contained about three times more hydrogen than the dummy element can removed from the core after power run 11.

Tensile testing of unirradiated zirconium containing 1000 ppm hydrogen showed that between room temperature and 250°F, the ductility of the hydrided zirconium was less than that of "as-received" material containing about 40 ppm. Above 250°F, the ductility of the zirconium containing 1000 ppm hydrogen was similar to that of 40-ppm material. Thus, the tensile properties of the SRE zirconium remaining in the core should be similar to "as-received" zirconium at reactor operating temperatures.

Bend testing of the SRE zirconium was the only simple means of determining its ductility outside of a hot cell. Because of the high radiation level of the material, it was possible to work only with very small specimens. Room-temperature bend tests of the parent zirconium, containing about 1000 ppm of hydrogen, showed almost a complete loss of ductility. The transition from brittle to ductile behavior for SRE zirconium was in the 175° to 200°F temperature range, and is most likely due to the hydride going into solution in the zirconium as a result of increasing temperature. In the reactor the zirconium temperature will always be above 250°F.

SRE moderator-can welds behave in a brittle manner below 350°F in a bend test. Above this temperature, the ductility improves but, even at 500°F,

some of the welds were found to be more brittle than unexposed zirconium welds at room temperature. No well defined brittle-ductile transition temperature was found for the weld material. This is undoubtedly the result of variations in weld thickness, imperfections within the weld, variations in grain size, and differences in hydrogen content. The better ductility of the weld material near the bottom of the cans is due to the low hydrogen content found in this region.

One possible source of future difficulties with moderator cans is a difference in volume change during hydriding between weld and parent materials. The weld areas appeared, in one test, to hydride less and, as a result, to expand less than the parent metal. The cans remaining in the core may therefore be buckled slightly or overstressed in the vicinity of welded areas. Creep and diffusion of hydrogen would relieve such stresses; higher operating temperatures would promote such relief, but must be limited until the carbon level in the coolant is reduced.

d. Zirconium Grain Growth

No general grain growth was observed in zirconium which was exposed to 1000°F sodium during reactor operation. Six specimens, two each from the top, middle, and bottom of can R-24, showed a normal zirconium structure with grain sizes of 0.01 to 0.02 mm in diameter. The same result was found for material from the scallop area. Some grain growth of the zirconium was observed in the vicinity of the damaged region of the fuel element in moderator can R-24, which shows that its temperature was substantially above normal for some part of its exposure in the SRE.

e. Effect of Nitrogen on Zirconium

The room temperature mechanical properties of zirconium are affected by nitriding, but at elevated temperatures the effects decrease.[10] Any effects of small additions of nitrogen are probably completely masked by the effects of hydrogen. A Kjeldahl analysis of a sample of moderator sheath material indicated little, if any, increase in nitrogen content over the original material. Results of vacuum fusion analyses were inconclusive as nitrogen content is obtained by difference, an inherently imprecise method.

The results of the zirconium examinations are summarized as follows:

1) General appearance of the zirconium surfaces exposed to the sodium was unchanged during reactor exposure.

2) Bend testing of the SRE irradiated zirconium determined that ductility, at normal reactor operating temperatures, is not adversely affected by high hydrogen content.

3) At normal SRE operating temperatures, no change occurred in the grain size of the moderator-can zirconium.

4) Moderator-can sheet ruptures resulted from a swelling of the graphite upon penetration by sodium.

5) High operating temperatures (above 900°F) would probably be beneficial to the zirconium.

Based on the results of this investigation, the SRE zirconium remaining in the core after approximately 4500 hr of reactor operation is deemed satisfactory for continued operation under reactor design conditions.

2. Estimate of Tetralin Leakage

It was estimated that from 1 to 10 gal of tetralin had leaked into the primary sodium system prior to run 14. An additional estimate can now be made with the analyses for hydrogen absorbed in the moderator can zirconium. Assuming complete decomposition of all the tetralin to carbon and hydrogen, absorption by zirconium of all the hydrogen released, and assuming that all the hydrogen came from the tetralin and none from the graphite, a figure of approximately 4 gal of tetralin was obtained. This value is in agreement with the previous estimate, but is quite uncertain because there were several other sources of hydrogen, particularly outgassing of the graphite logs, which could contribute more than half of the observed hydride. It is also possible that sodium hydride was removed in the cold trap.

C. RADIOLOGICAL CONSIDERATIONS

In the parent report, the section dealing with radiological aspects of SRE operation was concerned with sources of activity, correlation of release of activity with events occurring in the reactor, and the effects of these releases on the environment. This report discusses the distribution and management of the fission products during the recovery operations. During the recovery effort the objectives were:

1) To limit personnel exposure to an average dosage rate of 1.25 rem/quarter (5 rem/yr).

2) To limit the stack gas release to less than MPC.

3) To confine unpackaged radioactive contamination to the SRE reactor room.

4) To assess effectiveness of the sodium in retaining the fission fragments.

5) To assess latent problems which might develop as a consequence of the contamination of sodium by fission products and uranium.

Throughout the recovery effort the radiation exposure to each individual was limited to less than 5 rem/yr. It was occasionally necessary to permit the weekly exposure for some key individuals to reach 600 mrem per week, in which case the individual was not exposed to radiation during the following week. Such exposures required a special permit, and only 30 such permits were issued. For the 150 persons directly involved in the work, the average exposure was 2 rem/yr.

Constant monitoring for airborne contamination in the reactor room was continued during the recovery effort. The highest level recorded was $3 \times 10^{-8} \mu c/cm^3$, which occurred on 5/21/60 at 2400 during core inspection activities. Operations were suspended for one hour, by which time the level had returned to $10^{-9} \mu c/cm^3$. At no other time did the level exceed the tolerance level of $10^{-9} \mu c/cm^3$. No release above MPC was detected by the stack monitor during 1960.

The general contamination levels on the walls, floors, tools, etc. in the reactor room ranged from 100 to 1500 d/m/100 cm^2 of beta and gamma activity. Occasional high counts of 150,000 d/m/100 cm^2 were experienced and immediately cleaned up. There were several instances where minor contamination of the asphalt blacktop occurred just outside of the SRE access door. These were promptly cleaned up and no further spread occurred.

With the exception of the inert gases Xe^{133} and Kr^{85}, all of the fission fragments remained in the sodium, or were absorbed by the carbon or by the sodium-wetted metal surfaces. Fission fragment data from the sodium samples show considerable scatter due to trace quantities of particulate carbon. It has been shown that at least the gamma emitters are being effectively removed from the sodium by the cold traps (Section III). Typical values showing the decrease of gross fission-product activity in sodium with time are: 8/6/60, 0.2 $\mu c/g$ and 10/23/60, 0.04 $\mu c/g$.

Samples of SRE sodium were sent to Argonne National Laboratory for fluoroscopic analysis; these analyses showed a uranium concentration of 0.2 ± 0.1 ppm. This concentration is lower than that specified for the EBR-II sodium specification and will cause no future operational difficulties.

Examination of sample sections of SRE primary piping has shown some absorption and possibly adsorption of certain fission products. The principal contributors and their typical concentrations (normalized to 1/1/61) are: (1) Sr^{90}, 0.78 $\mu c/cm^2$; (2) Ce^{144}, 0.6 $\mu c/cm^2$; and (3) Cs^{137}, 0.02 $\mu c/cm^2$. It has been shown by etching techniques that 99% of these isotopes are contained in the first 0.1 mil of depth. It is not known at the present time whether they have diffused into the stainless steel or are retained in surface micro-fissures. It is also not yet known how effective the cold trapping and hot trapping of the sodium will be in removing this surface contamination. Because of the contamination, welders working on the primary system are required to wear self-contained breathing apparatus.

Continued routine monitoring of soil, vegetation, water, and air revealed no increase in background radiation levels.

D. REACTOR PHYSICS

1. Introduction

In the parent report, several possible sources of reactivity change were discussed as to magnitudes and possible rates of insertion. Some of the quantities reported were in error and were corrected in an errata sheet (see Appendix). Those points which were left unanswered in the interim report are considered and their present status set forth. In particular, a considerable amount of effort has gone into the analysis of the power excursion which resulted in a 7.5-sec reactor period.

2. Correction of Errors

One of the most significant errors in the interim report was the statement that the power at the peak of the excursion reached 24 Mw. A re-evaluation of reactor instrument records showed that the peak power reached was about 14 Mw. The 24-Mw value was obtained by a linear extrapolation of the log N recorder chart from power levels of about 2 Mw, and linear extrapolation is not valid. All other values of reactor power given in the parent report are correct to within about 10%.

A second correction is to the value of the reactivity that is introduced when a fuel channel is voided of sodium. A recalculation of this quantity gave $\Delta\rho$ = +0.03% for a typical fuel channel (Appendix).

3. Reactivity Gain During Run 13

During run 13 a slow increase in reactivity of about 0.3% took place over a period of about 6 hr. Two possible causes of the increase are postulated. The first is an increase in average moderator temperature, the second is collection of gas between moderator cans.

During run 13 the entire moderator coolant supply was bypass flow around the moderator pedestals at the grid plate. These small passages were partially blocked by contaminants in the coolant, as was indicated by an increase in moderator coolant outlet temperature. An average increase of about 180°F in moderator temperature would produce the 0.3% change in reactivity. The presence of a negative pressure coefficient during run 14 indicates the possible presence of bubbles in the core. Displacement of the sodium from around the central moderator assembly would produce a reactivity increase of about 0.2%. Either or both of these mechanisms may have contributed to the observed reactivity change.

A careful review of reactor operating history revealed a probable correlation between tetralin leaks and increased reactivity. Leaks occurred before runs 4 and 8 and during run 13. In each case a few tenths of a percent increase in reactivity appeared immediately after the leak. The increase slowly disappeared, over a period of several months. The anomaly in reactivity during run 13 thus appears to be due to the tetralin leak, but the mechanism is uncertain.

4. Reactivity Loss During Run 14

During the first four days of run 14 there was a gradual loss in reactivity of about 1.2%. It was stated that the only reasonable explanation of this loss was the flooding of moderator cans with sodium. During the recovery operations it was learned that the moderator cans associated with core channels R-10 and R-42 had in fact failed and that the graphite contained in them was saturated with sodium. It is believed that this is a satisfactory explanation for the above loss in reactivity.

An alternate explanation for a part of this reactivity loss was that the damaged fuel elements had parted and the bottom parts fallen out of the core. No evidence has been found that the bottom parts of broken fuel elements fell. In fact,

the nature of the plugging that was observed in R-24 suggests that most of the damaged fuel was securely wedged in the process channels.

5. Explanation of Reactivity Changes Prior to and During the Power Excursion

A study was made of the reactor behavior over a period of 1 hr prior to the excursion which occurred on July 13, 1959. The details of this study are given in Reference 8. The results and conclusions are summarized in the following paragraphs:

The solution of the reactor kinetics equation shows that all but three of the changes in power level that occurred during the time interval considered are explainable in terms of reactivities introduced by control rod motion and by fuel and moderator temperature changes. The three exceptions are:

a) The negative excursion that occurred at 1807 hours.

b) The gradual increase in power from 3.0 to 4.5 Mw between approximately 1821 and 1824 hours.

c) The positive excursion with 7.5-sec period that occurred at about 1825 hours.

The reactor kinetics calculations showed that the unexplained reactivity changes involved in these three exceptional instances had approximately the following characteristics:

a) Negative excursion: = -0.06% step followed by a recovery of most of this reactivity loss within the next few minutes.

b) Slow rise: = +0.04% in a 3-min ramp.

c) Positive excursion: = +0.3% in a 5 to 10-sec ramp.

Using these reactivity changes in the calculations, in addition to those changes being caused by control rod motion and reactor temperature changes, produced a reasonably good approximation to the observed changes in power level and reactor period.

Reasonable explanations for these reactivity changes follow:

a. Negative Excursion

In Section IV-A, a mechanism for formation of vapor within partially plugged channels was discussed. Shortly before the negative excursion, the

reactor power level was being increased. The evolution of vapor as the power rose introduced additional void into the core by the displacement of sodium from the plug region. This resulted in the introduction of a small amount of positive reactivity. Fuel temperature in the immediate vicinity of the plug rose as the formation of vapor took place. Control rods were inserted in order to restrain the power increase, and this insertion halted and started to reverse the rise in power. As power started to drop, the volume of vapor in fuel channels decreased, thereby introducing negative reactivity and speeding the drop in power. The collapsing bubbles allowed more coolant to enter the plugged regions, thereby rapidly cooling the overheated fuel contained therein. The reduction of fuel temperature introduced positive reactivity due to the Doppler effect, thereby reducing the rate at which power was dropping. This recovery of reactivity allowed the reactor to be made critical again with a relatively small withdrawal of control rods.

This model fits all phases of the observed reactor behavior during this period.

b. Slow Rise

A moderator temperature rise of about 25°F above that used in the kinetic studies would account for the reactivity change of 0.04%. The temperature data used in the calculations were obtained from a record of moderator coolant temperature which, with the reactor condition during run 14, was an unreliable indication of moderator temperature. Most of the power changes were small during run 14, and the assumed moderator temperature changes were adequate in other cases to match calculations to observed reactor behavior; however, with a change in power level from 3.0 to 4.5 Mw, the assumed temperature change may not have been adequate.

An additional possible source of the 0.04% increase was an increase in sodium vapor volume in the core as power level was increased. If the picture of sodium vapor in the channels is correct, some expansion of vapor volume would accompany the 1.5-Mw increase in power level.

c. Positive Excursion

In order to produce a 7.5-sec reactor period, it is necessary to introduce about 0.3% reactivity in a time interval of not more than about 10 sec. A reasonable mechanism for accomplishing this is the creation of void in about 10 fuel channels.

The discovery of thermal cycling effects on the fuel in some channels produced a picture of cyclical pulsing of sodium vapor within channels. Several channels were involved, each with a cycle period of about 2 min. The cycling in each affected channel introduced reactivity changes which, at about 4 Mw of power, were not large enough to influence the cycling in the other channels. However, shortly before the excursion the reactor power level was about 5 Mw. It appears plausible that at this power level, the vapor volume was larger, and the effect on reactivity was of sufficient size to produce a significant increase in reactor power level, so that all cycling channels were affected. This action produced a combining of the reactivity contributions from several channels, resulting in an accelerating reactivity increase.

d. Gas Bubbles

An alternate mechanism that has been suggested to account for voids in the core is the presence of gas other than sodium vapor. The movement and growth of gas bubbles could easily account for all three of the reactivity changes noted. Thus, the introduction of a bubble into the core would introduce positive reactivity. If such a bubble were then to move out of the core, it could introduce the negative reactivity required to cause the negative excursion. A re-accumulation of the gas in the core would then add reactivity so that the reactor could again be made critical with a relatively small withdrawal of control rod. The positive excursion could have been caused by the movement of large bubbles up through the reactor core. The transit time for sodium to flow through the fuel channels is on the order of 2 sec. This time is consistent with the required rate of insertion of the 0.3% reactivity needed to cause the 7.5-sec reactor period.

It is not obvious how bubbles of gas other than sodium vapor could have produced the cyclical variation in the outlet temperature of the partially plugged fuel channels. The explanation of observed anomalies on the basis of local boiling is therefore deemed more plausible.

IV. MODIFICATIONS TO REACTOR SYSTEM

Several modifications were made to the SRE during recovery operations.[3] Some will prevent a recurrence of the difficulties which caused the fuel cladding failures during run 14. Others have the purpose of correcting functional difficulties, improving reliability, or supplying more operating information. The important changes are summarized in the following pages.

A. ELIMINATION OF TETRALIN

Tetralin (tetrahydronaphthalene) has been eliminated as a service coolant at the SRE in favor of NaK, nitrogen gas, and kerosene. Experience has shown that tetralin decomposes in sodium to form materials which can block reactor coolant channels; tetralin also has been discovered to have other undesirable features as a service coolant.

Tetralin forms unstable peroxides upon exposure to air. At the SRE a nitrogen cover gas in the tetralin expansion tank and use of an amine as an anti-oxidan resulted in tetralin peroxide concentrations of less than 0.05% in the tetralin. However, high peroxide concentrations could conceivably appear in portions of the SRE system during an extended shutdown or after components have been removed from the system. An additional undesirable feature of tetralin is that naphthalene, one of its decomposition products, reacts with sodium to form a very reactive, pyrophoric naphthyl-sodium compound.

Considerations in the choice of a service coolant include heat transfer characteristics, available temperature difference, and coolant flow area. A gas is adequate in some areas where heat fluxes are low and temperature differences high. Nitrogen, the inert atmosphere of the primary galleries, is now used to cool the primary cold trap and plugging meter and as a backup coolant on the freeze-stem valves. These components in the secondary system are cooled with air.

Freeze-seals on sodium pumps are cooled with NaK. Its use was recommended by consideration of coolant flow area, available temperature difference, and need for a coolant that does not react with sodium in case of a freeze-seal leak. Pump replacement with a type not requiring a freeze seal was deemed impractical from system considerations, which precluded use of a free-surface pump without very extensive structural modifications.

An organic coolant is used in some areas, with a double barrier between the organic and sodium. The organic deemed most appropriate for service coolant is a high grade of kerosene. This material does not form compounds with sodium and does not form peroxides in the presence of air. A major consideration in this choice is the extensive industrial history of kerosene uses. Kerosene is used to cool the top shield, core cavity liner instrument thimbles, and freeze-stem valves. The top shield cooling system is a separate, limited-volume system with positive means of detecting leakage.

B. SODIUM SYSTEM CHANGES

Several modifications were made to the sodium heat transfer systems to improve performance or reliability.

1. Moderator Coolant Eductor

Sodium coolant for the moderator cans flows through a line which enters the moderator coolant plenum between the reactor grid plate and the bottoms of moderator cans. An EM pump which was installed to provide flow developed a leak in one of the throat sections. It has been replaced by a jet pump, or eductor, which uses the main sodium flow for actuation.

The jet pump is shown in Figure IV-1. Its location in the primary system is shown in the simplified flow diagram of Figure IV-2. By manipulation of the two valves shown, it is possible to pump sodium into or out of the moderator coolant plenum. In the past there has been enough coolant leakage past the grid plate at the moderator pedestals to provide more than the required cooling to the moderator cans, and it is desirable to be able to pump in either direction.

2. Main Secondary Bypass Line

A 2-in. line has been installed which connects the steam generator inlet and outlet lines. It was added to permit isothermal circulation at higher temperatures than is feasible through the steam generator, thereby increasing the temperature range (to 700°F) in which isothermal reactor physics tests can be made. A valve in the line prevents flow during normal plant operation.

3. Materials Evaluation Facility

This facility was provided during SRE construction to obtain sodium samples and to expose metal specimens to the hot and cold legs of the primary sodium

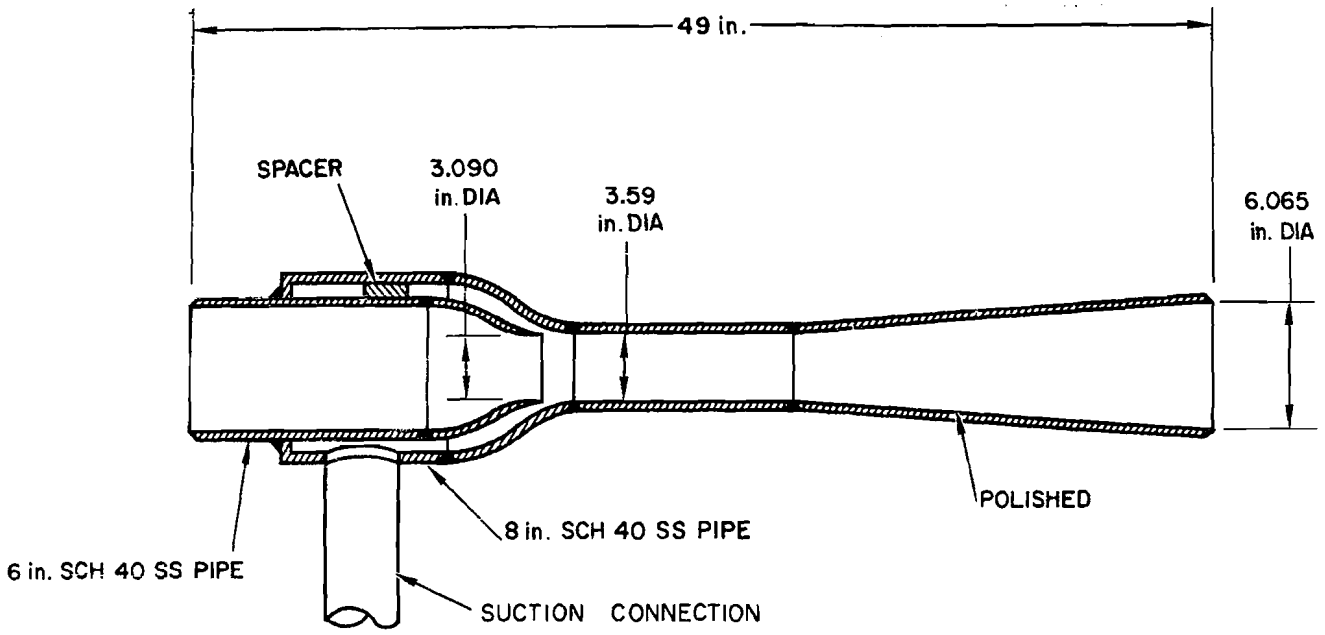


Figure IV-1. Moderator Coolant Jet Loop

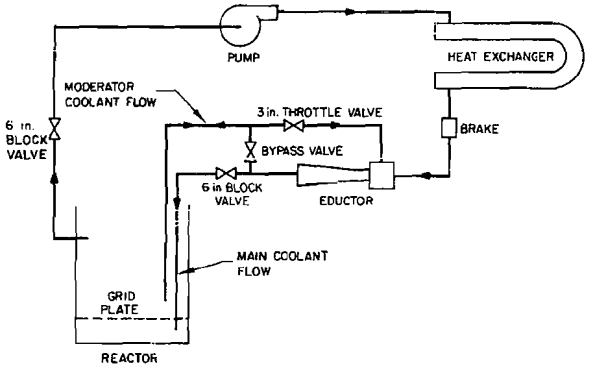


Figure IV-2. Simplified Flow Diagram, Main Primary Sodium System

system. The facility incorporated a tetralin-cooled freeze seal which would have required modification; it was also a source of gas on the suction side of the primary pump, which produced flow variations. The facility was therefore removed. A sodium sampler was installed in a special top shield plug which takes samples directly from the pool over the reactor. Steel and zirconium specimens will be exposed to pool sodium in an exposure facility incorporated into a top shield plug.

C. FUEL WASHING CELL

The original wash cells at the SRE used water for the removal of sodium. While the procedure was successfully used more than 1000 times, several fuel elements were damaged by pressure surges which partially collapsed the fuel cladding at the top of the fuel rods. On June 4 (immediately following run 13) a pressure surge occurred which severely damaged the fuel element and ejected the shield plug from the wash cell.

A test apparatus was fabricated and an extensive series of tests conducted to determine the cause of the pressure excursions and to develop an improved washing procedure. The tests indicated that the pressure surges were caused by submerging a substantial quantity of sodium under water. Apparently the drain holes in the cylindrical holddown tube were plugged sufficiently, with decomposition products from the tetralin leaks which had occurred, to prevent complete draining of the sodium. (The holddown tube has been modified; see Section IV-E).

Following the water tests a series of washings was conducted with steam. These tests were highly successful in removing substantial quantities of sodium without pressure buildup. The steam cleaning left NaOH residue, however, which can be removed with water. The new wash cells therefore incorporate steam cleaning and a water rinse. A hydrogen recombiner has been added to the system because the gases from the washing operation are vented into the radioactive vent system, into which air leakage is unavoidable.

D. INSTRUMENTATION MODIFICATIONS

1. Fission Product Monitor

The analyses of core cover gas (helium) samples after run 14 indicated a large increase in Xe^{133} activity. This suggested that continuous monitoring of the Xe^{133} activity would provide a reliable means of detecting fuel element cladding failures. Techniques were developed for gas sampling during reactor operation. The detector is a scintillation spectrometer used in the system shown schematically in Figure IV-3. Gas flow through the spectrometer is at a rate of about 0.3 cfm. The differential analyzer is adjusted to discriminate in favor of the 0.081-Mev gamma from Xe^{133}. If the level exceeds a predetermined amount, a signal is provided to an annunciator alarm in the SRE control room. Purge lines are provided to permit calibration with a sample of Xe^{133}.

2. Reactor Transient Monitor

The transient monitor is a fast data-recording system to record selected parameters during off-normal conditions. It was installed to provide detailed data for diagnosis of abnormal occurrences.

In operation seven selected signals are recorded continuously at a fast tape speed on a continuous-loop magnetic tape. Three heads are provided for recording, reproducing, and erasing. The recording and erasing heads are in

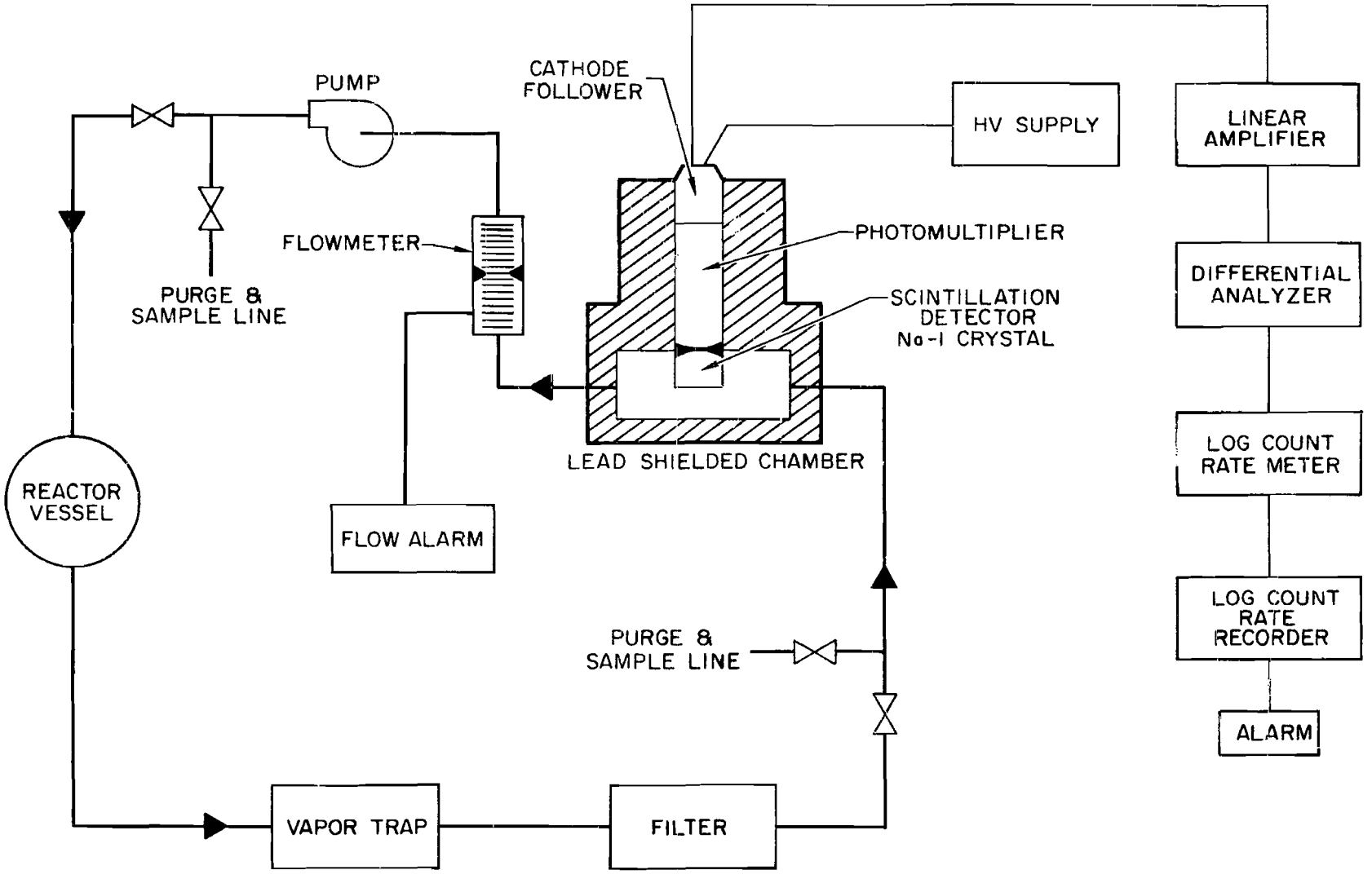


Figure IV-3. Fission-Gas Detection System

continuous operation. Upon receipt of an off-normal signal, such as a reactor scram, the reproducing head is activated, and the information is transferred from the continuous tape to a second tape recorder. Because the continuous tape has a circuit time of 8 min, the system has an 8-min "memory"; i.e., information is recorded on the second tape which covers the operating period beginning 8 min before the activating signal is received. Up to 4 hr of recorded data can be stored on the second tape after receipt of the activating signal.

Typical reactor parameters which will be monitored by the transient monitor are: (a) neutron flux level, (b) reactor period, (c) a selected fuel channel outlet temperature, (d) a selected in-fuel temperature, (e) moderator coolant temperature, (f) primary coolant flow, and (g) shim rod position.

3. Reactor Temperature Measurements

Two recorders have been added to record data from thermocouples installed in some of the fuel slugs and in special tubing which serves as wire wrap to separate the fuel rods (see Section IV-E). Chart speed is 12 in./hr and each point is recorded at 50-sec intervals. The recorders are faster than previous equipment by a factor of six and will provide more detail in fuel element temperature histories.

There are eight special fuel elements with in-fuel and wire-wrap thermocouples. Their locations in the core are indicated in Figure IV-4. The thermocouple locations are shown in Figure IV-5.

4. Operation Events Recorder

Three 20-pen recorders have been installed to indicate when any of 60 functions is being performed. The recorders are of the off-on type and therefore measure only the time duration of events being monitored. The following will be monitored:

10 reactor scram circuits
9 reactor scram bypasses
3 reactor setback circuits
3 reactor setback bypasses
9 plant protective system alarms
4 control circuit interlock bypasses
4 safety-rod-in circuits
4 safety-rod-out circuits
4 shim-rod-in slow drive circuits
4 shim-rod-out slow drive circuits
2 shim-rod-in fast drive circuits
2 shim-rod-out fast drive circuits
2 types of automatic control (temperature or flux).

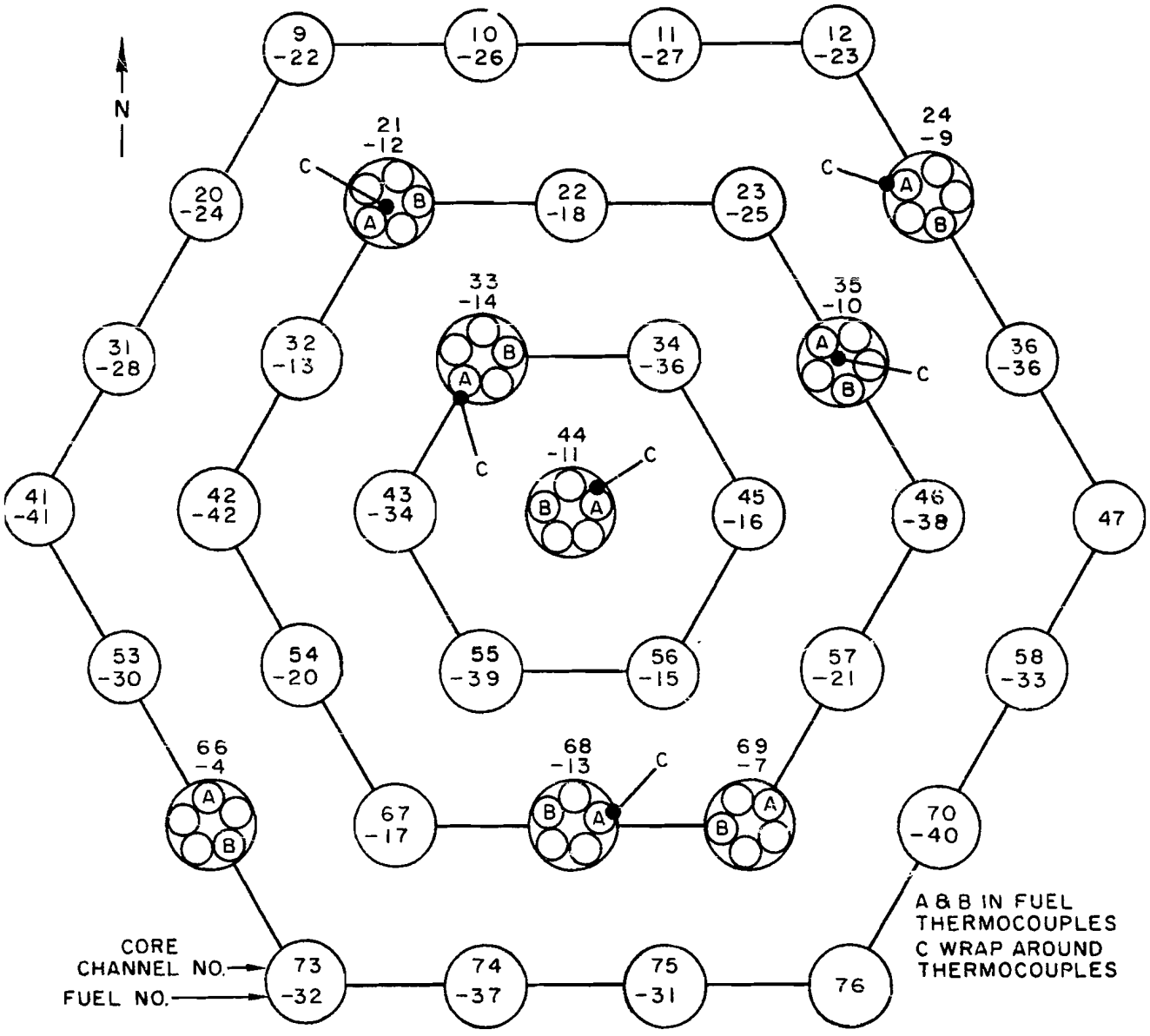


Figure IV-4. Core Location of Instrumented Fuel

The recorders have synchronous chart drives, and the charts can be simultaneously time-marked by operation of a switch. They therefore provide an accurate chronology of operating events and aid in separating cause and effect in analysis of reactor operation.

5. Flux/Power Deviation Circuit

As a supplement to the standard plant protective system, a flux/power deviation circuit has been added. The function of the circuit is to prevent large mismatches between heat generation and heat removal rates. Reactor power as determined from neutron flux level is compared to power as determined from

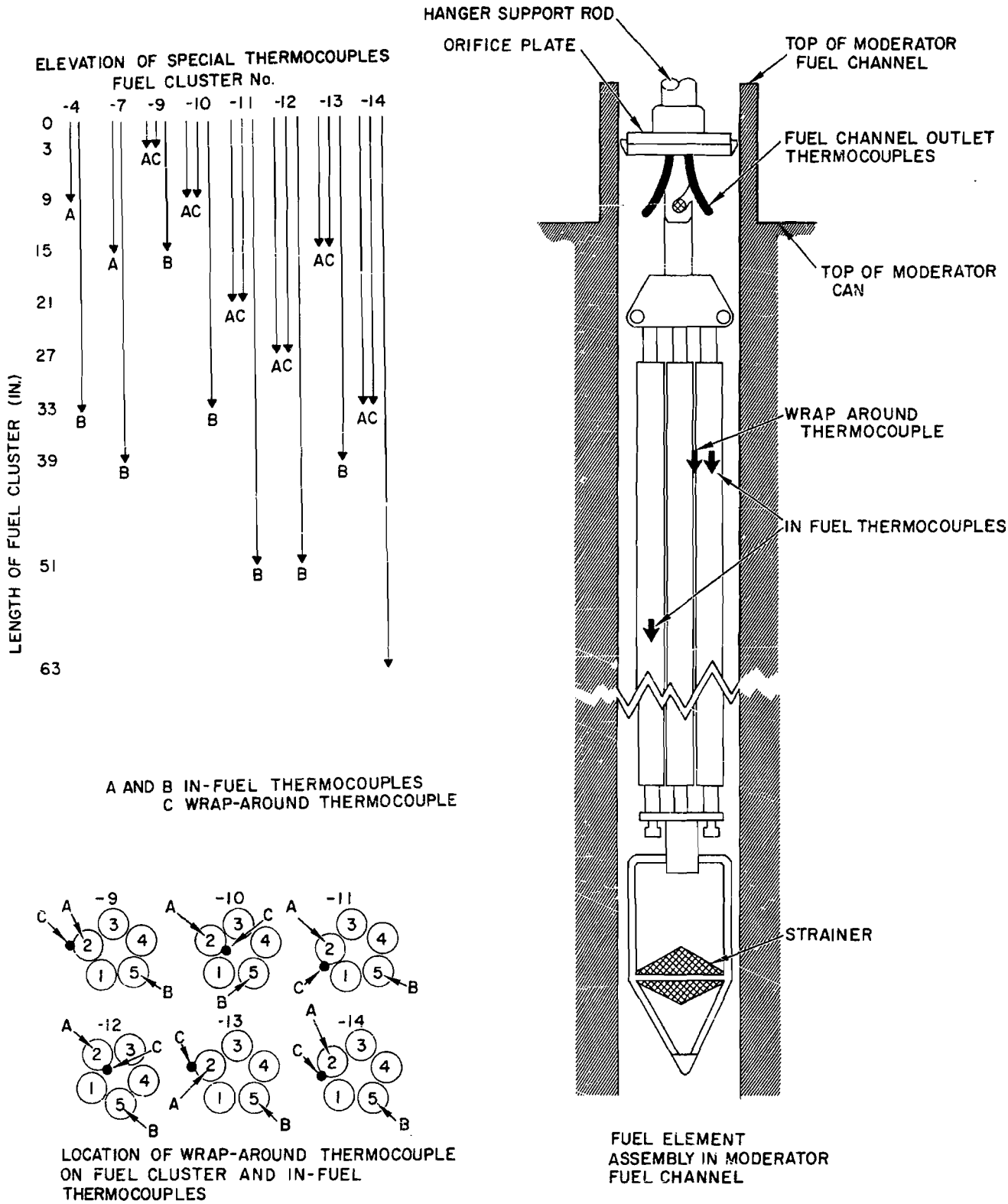


Figure IV-5. Thermocouple Location of Instrumented Fuel

coolant flow and temperature rise across the core. A deviation between the two values, because of variations in the input data, represents an off-normal condition, and a correction signal is provided to the plant protective system. The circuit is activated at power levels above about 20% of full power. Deviation limits and corrective actions are indicated in Figure IV-6.

E. FUEL ELEMENT CHANGES

To reduce the probability of fuel element failures, several changes were made in the fuel elements for the second core loading. Clearance between the element and fuel channel has been substantially increased. The orifice plate was moved to a position which should provide for more accurate prediction of flow through each channel. A structural member was added to the cluster of rods, and a strainer has been attached to the bottom of the element.

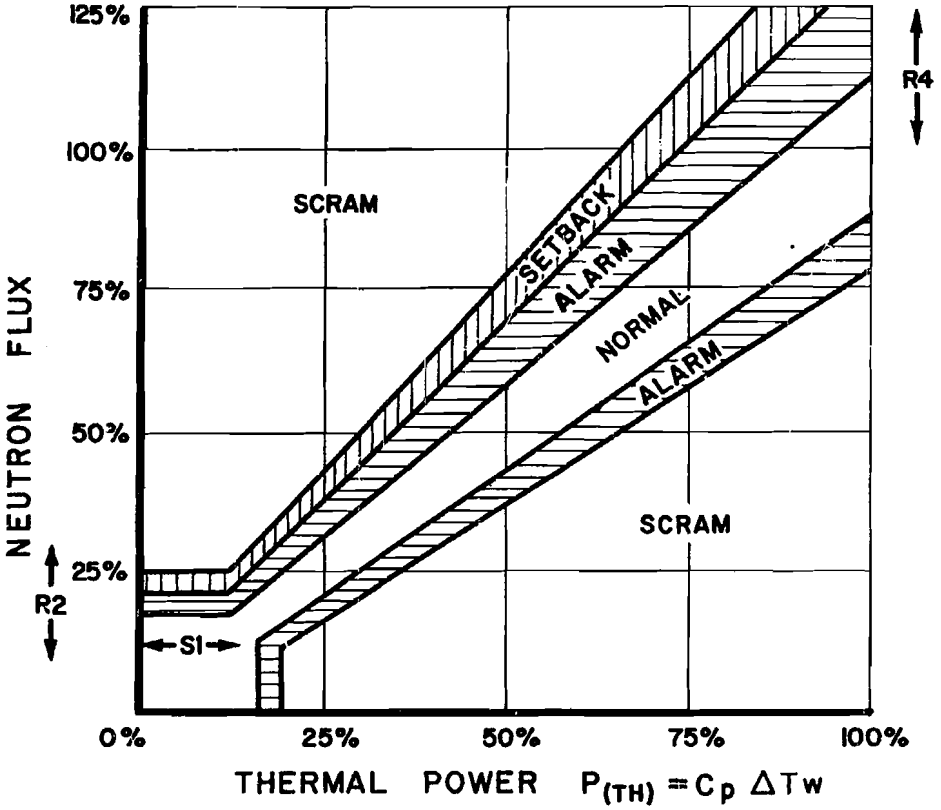
1. Fuel Element Geometry

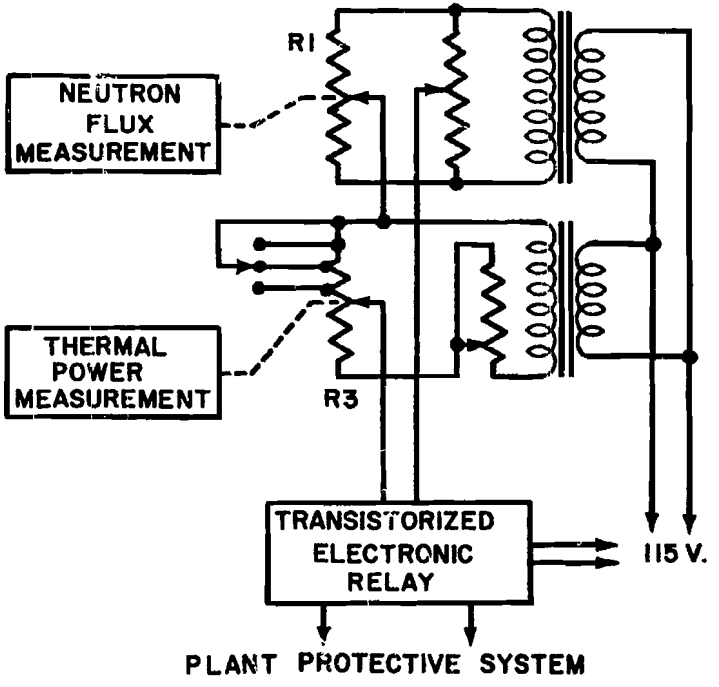
Changes in fuel element geometry, indicated in Figures IV-7 and IV-8, are as follows:

The 7-rod cluster was changed to a 5-rod cluster arranged in a ring around a 3/8-in. stainless steel support rod. Nominal radial clearance between the element and process tube was increased from 0.029 in. to 0.162 in.

The bottom guide (birdcage) was modified to receive two conical strainer sections. They are designed to strain particles larger than about 0.05 in. from the coolant stream. The purpose of the strainer is to accumulate particulate materials in an area where high temperatures cannot result. If the strainer accumulates material and starts to plug, the outlet channel thermocouple will indicate the reduction in flow before excessive temperatures are reached in the fuel rods. Local plugs on the fuel rods would lead to very high local temperatures before a definite indication is discernible from the coolant channel outlet temperature. The purpose of the lower filter screen is to catch material which may be washed from the top screen during fuel removal.

The orifice plate was moved to the top of the element and is attached to the hanger rod. The orifice plate is now located in a machined hole in the top casting where bypass flow around the plate can be more accurately predicted. This change should decrease the variation (channel to channel) in outlet temperatures.





DEVIATION CIRCUIT

ABNORMAL CONDITION	ΔT	FLOW	FLUX	POWER	PROTECTION
LOSS OF FLOW	CONSTANT	DECREASE	CONSTANT	DECREASE	
INCREASED FLOW	CONSTANT	INCREASE	CONSTANT	INCREASE	1. ALARM
HIGH COLD LEG TEMP.	DECREASE	CONSTANT	CONSTANT	DECREASE	2. SETBACK
HIGH FLUX TRANSIENT	CONSTANT	CONSTANT	INCREASE	CONSTANT	3. SCRAM
LOW FLUX TRANSIENT	CONSTANT	CONSTANT	DECREASE	CONSTANT	

Figure IV-6. Operating Limits on Flux Power Deviation Circuit

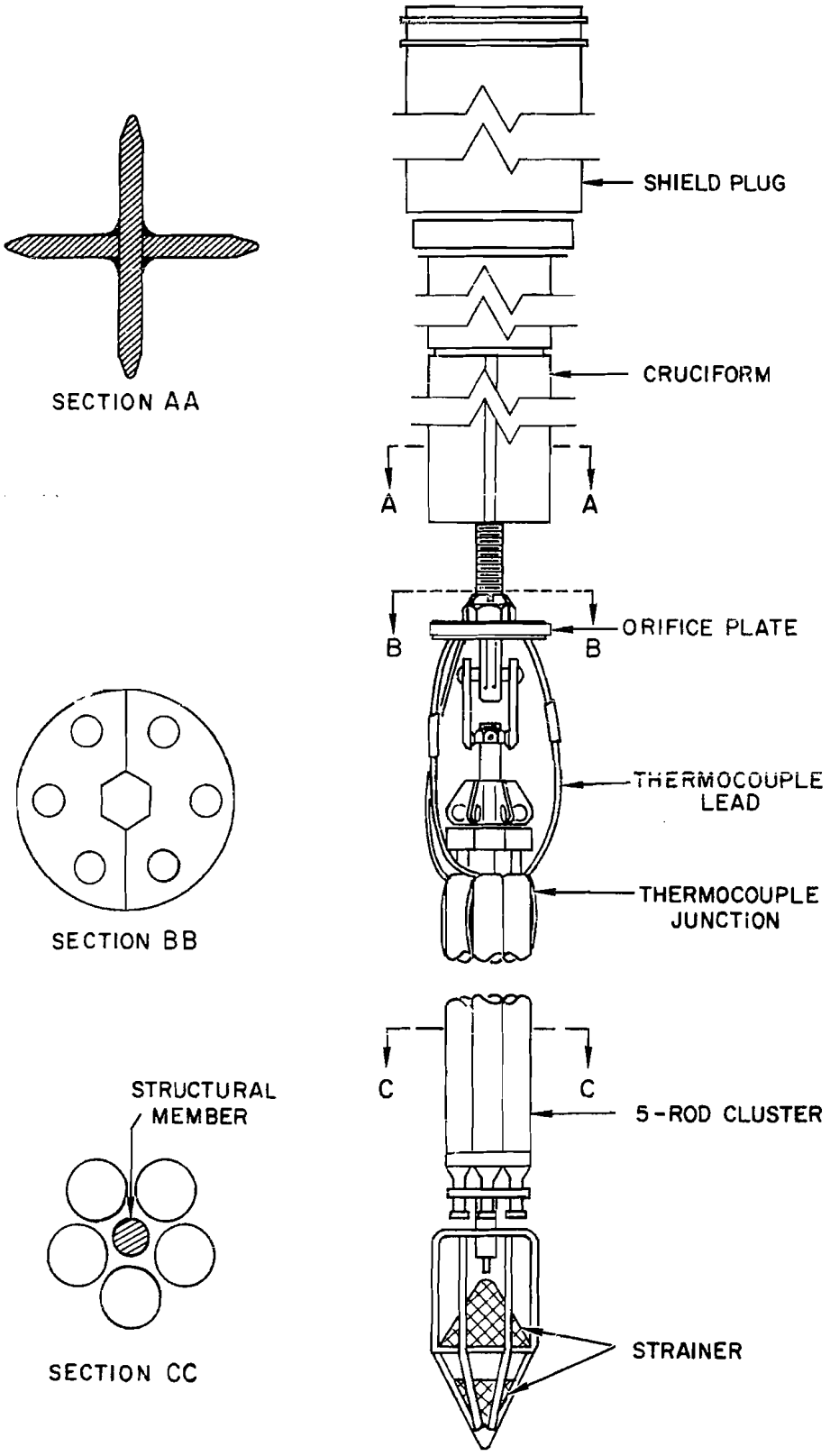


Figure IV-7. Thorium-Uranium Five-Rod and Seven-Rod Clusters

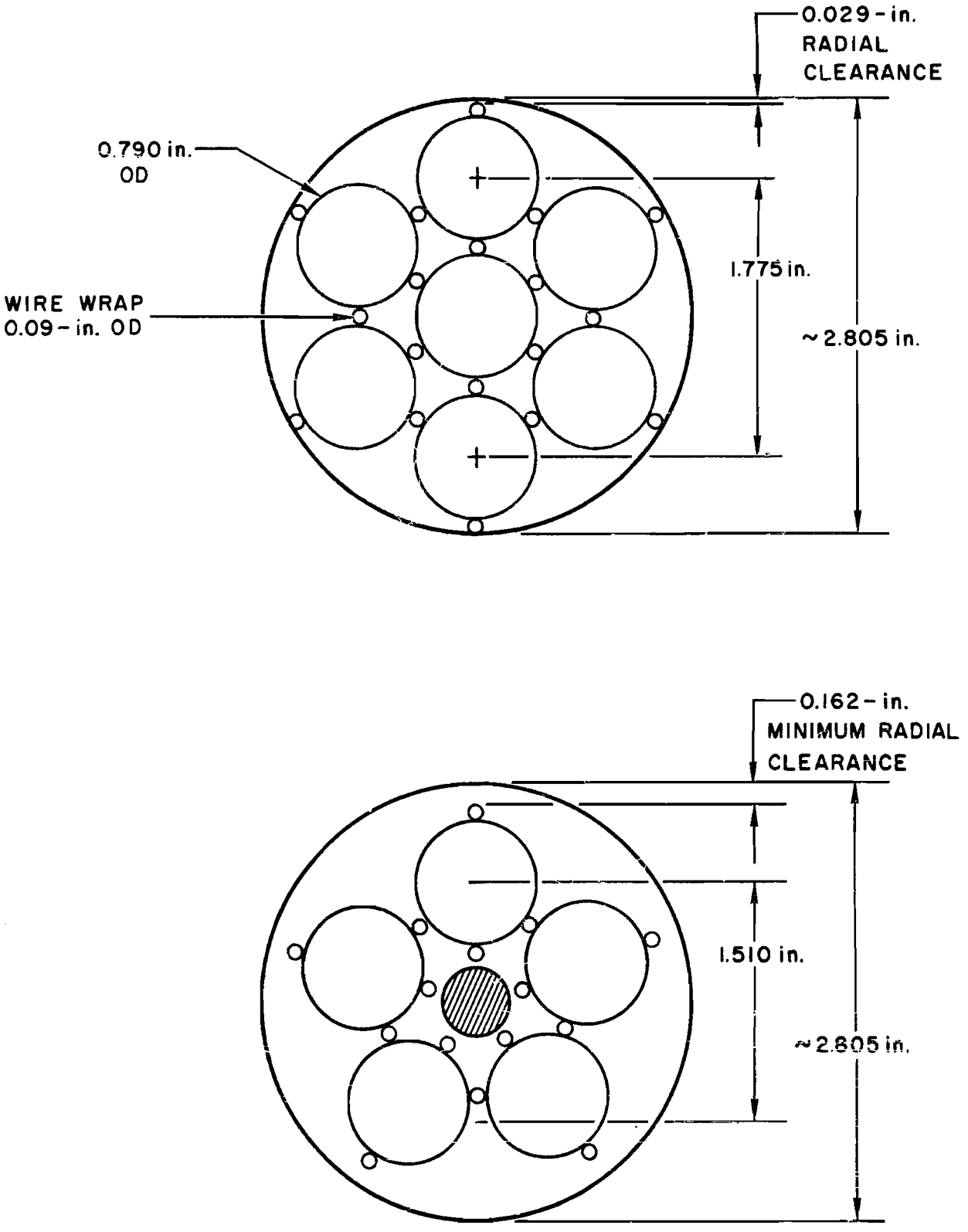


Figure IV-8. Thorium-Uranium Fuel Element

The hanger rod and holddown tube of the first core loading has been replaced by a cruciform section, as shown in Figure IV-9, to reduce potential hold-up of sodium and subsequent difficulties in the wash cell.

2. Fuel Element Instrumentation

Fuel element instrumentation has been improved. The two outlet channel thermocouples (one spare) on each fuel element are farther into the cooling channel than was previously the case. The thermocouple junctions have been placed directly in the coolant stream (see Figure IV-9) to improve response time.

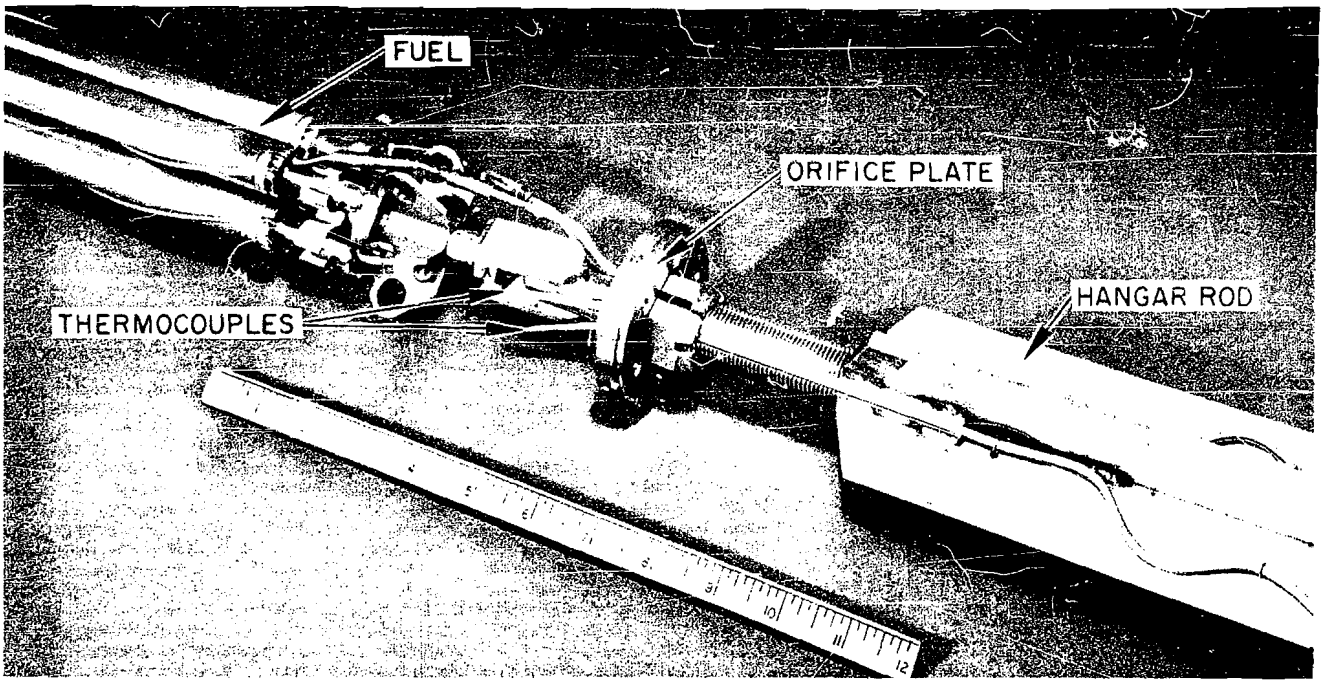


Figure IV-9. Cruciform Holddown on Fuel Element

Eight fuel elements have special thermocouples. Each of the eight elements has two thermocouples in the fuel slugs. Elevation of the thermocouples in the fuel rods is shown in Figure IV-5. Six of the eight elements have an additional thermocouple in special tubing, which serves as a wire wrap for fuel rod spacing; these thermocouples measure local sodium temperatures. Each of the wire-wrap thermocouples is opposite an in-fuel thermocouple to facilitate heat flux determinations.

F. EFFECTS OF MODIFICATIONS ON REACTOR OPERATIONS

The modifications essentially eliminate the possibility of major contamination of the primary coolant. The new fuel element has adequate clearance in the

fuel channel to permit routine removal even in the event of fuel distortion or a major cladding failure. The possibility of multiple fuel element failures has been markedly reduced through use of the fission-gas monitor on the reactor, which should indicate faults in the cladding. Better control of outlet channel temperatures through relocation of the orifice plates should reduce thermal stresses in moderator cans. The new transient monitor and the operation events recorder provide detailed data during periods of off-normal conditions. which should aid greatly in understanding the reactor system.

V. SUMMARY

Investigation into the SRE fuel element damage has been completed. A general picture of both causes and effects has been reached, and resumption of operation of the SRE has been recommended.

A. CAUSE OF FUEL ELEMENT DAMAGE

Failure of the fuel cladding can be attributed to abnormal temperatures caused by interference with heat removal capability by decomposition products of tetralin. Local blockage of coolant passages prevented an adequate supply of coolant from reaching parts of the fuel elements.

The fuel cladding failure was caused by (1) thermal cycling and expansion of the fuel slugs until the cladding burst, and (2) formation of low melting alloys of uranium and the iron, nickel, and chromium constituents of 304 stainless steel used for cladding.

1) Failure by thermal cycling and expansion was indicated by examination of the top sections of the element from R-24. The cladding was split although there was no appreciable change in its ductility, and no evidence of Fe-U diffusion. The thermal cycling was probably caused by pulsing of sodium-sodium vapor in the partially obstructed process tubes, which in turn produced corresponding variations in coolant flow and fuel temperatures. Careful examination of the records of channel outlet temperatures showed indication of cyclical variations in several channels with a period of approximately 2 min.

2) Chemical analyses definitely established the presence of iron-uranium alloys, both in and near regions of cladding failure. Examination of the element from R-24, made by sectioning the moderator element in which the element was stuck, revealed about 800 g of Fe-U alloy near the eutectic composition.

B. CAUSES OF REACTIVITY CHANGES

The major reactivity changes requiring explanation are three: (1) the slow increase of about 0.3% during run 13, (2) the loss of about 1.2% during the first part of run 14, and (3) the rapid insertion of about 0.3% during run 14 which caused

a power transient. All causes of the reactivity changes are directly related to the presence of tetralin decomposition products in the reactor, and should not present difficulties in future operation.

1) The most likely causes of the 0.3% reactivity increase which occurred in a 6-hr period on May 30, 1959, are the displacement of some of the core sodium by bubbles and an increase in moderator temperature. This reactivity change, determined after the termination of run 14 from records of control rod position, is in fact the best indicator of the start of the tetralin leak.

2) The reactivity loss of about 1.2% during the first three or four days of run 14 is due to entry of sodium into the moderator cans in positions R-10 and R-42. The failures were probably caused by abnormal temperature conditions in the core during run 14.

3) The most likely cause of the 0.3% reactivity increase, which put the reactor on a 7.5-sec period and produced an excursion to about 70% of full power, is the expulsion of sodium from several fuel channels in quick succession. It is also possible that the cause of the reactivity insertion was the upward movement of a large bubble between moderator cans toward the center of the core.

C. CURRENT CONDITION OF REACTOR SYSTEM

In order to determine operability of the SRE after the fuel element damage, it was necessary, as part of the investigation, to determine its physical condition. In general it was concluded that there has been little change and the system condition is satisfactory; however, the probability of moderator-can failure may have increased.

The zirconium cans are known to be hydrided. Specimens taken from several of the cans removed from the reactor indicated hydriding over a range of about 100 to 1000 ppm of hydrogen. Ductility of the weld material in the moderator cans has been reduced by the hydriding.

With the addition to the primary system of substantial quantities of carbon, carburization of the stainless steel, with resultant loss in ductility, is a possibility. Tests of stainless steel specimens show that no appreciable amount of carburization has occurred except in those sections of the fuel cladding near failures.

Examination of specimens of fuel cladding (0.010-in. wall) taken from points distant from failures, sections of the upper element hardware, and a small section of piping, prove that no general carburization occurred.

The nitrogen purge operation before run 14 was suspected of providing conditions for nitriding of the stainless steel; however, no evidence of nitriding was found.

It is known that fission products have deposited on the walls of all primary system components. In particular, Sr^{90} is present to the extent of about 3/4 $\mu c/cm^2$. While the presence of this isotope is of little concern in maintenance operations around primary system components (it is a pure beta emitter), it is of importance in any operation in which the piping is welded.

D. MODIFICATIONS TO AVOID RECURRENCE

To minimize the possibility of a second major fuel element failure at SRE, modifications have been made to the system. Tetralin is no longer used as a service coolant. The freeze-seals on the sodium pumps are now cooled with NaK. Kerosene is used for auxiliary cooling, but only in locations where at least two barriers exist between the organic material and sodium.

A fission-gas monitor has been added to the reactor system to provide a continuous check on radioactivity of the cover gas. This instrument will inform the operators of the beginning of a cladding failure, and the reactor can be shut down before extensive damage occurs.

E. REACTOR OPERABILITY

The general conclusion reached regarding operability of the SRE is that there has been no important change in its physical condition as a result of the fuel element damage to the first core. The one reservation in this regard is that the probability of moderator can failure may have increased. The sodium may be saturated with carbon, and temperature limits have been imposed on operation until it has been demonstrated that the carbon content has been reduced to a satisfactorily low level.

APPENDIX

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IV-D-15	26	0.05%	0.02%
IV-D-10	Table IV-D-1,		
	entry 1	-0.03 to -0.05	-0.02 to -0.034
	entry 7	0 to 0.7	-0.05 to -0.7
	entry 12	0.01 to 0.02	0.02 to 0.04

REFERENCES

1. A. A. Jarrett et al., "SRE Fuel Element Damage, an Interim Report," NAA-SR-4488 (November 15, 1959)

2. W. J. Freede, "SRE Recovery Operations," (to be published)

3. G. Deegan et al., "Design Modifications to the SRE During FY 1960," NAA-SR-5348 (Rev.) (February 15, 1961)

4. Atomics International Staff, "Supplement to Hazards Summary for Thorium-Uranium Fuel in the Sodium Reactor Experiment," NAA-SR-3175 (Rev.) (to be published) (April 8, 1960)

5. J. L. Ballif, " Metallurgical Aspects of the SRE Fuel Damage Episode," NAA-SR-4515 (to be published)

6. J. J. Gill, "Zirconium Moderator Cladding Irradiated in the Sodium Reactor Experiment," NAA-SR-5970 (December 15, 1960)

7. R. S. Hart, "Investigation of Radioactivity in Coolant of SRE," (to be published)

8. F. Fillmore, "Reactivity Effects in SRE," NAA-SR-5898 (to be published)

9. W. J. Anderson and G. V. Sneesby, "Carburization of Austenitic Stainless Steel in Liquid Sodium," NAA-SR-5282 (September 1, 1960)

10. B. Lustman and F. Kerze, The Metallurgy of Zirconium, (1st ed., New York, McGraw-Hill Book Co., Inc., 1955) p 509

11. J. C. Bokras, "Effect of Sodium on Mechanical Properties of Zirconium," NAA-SR-1867 (June 15, 1957)