TID-7553
(NAA-SR-2600)

PROCEEDINGS OF THE SRE-OMRE FORUM

Held at Los Angeles, California
Feb. 12 and 13, 1958

Issuance Date: May 1958

Atomics International Division
North American Aviation, Inc.
Canoga Park, California

UNITED STATES ATOMIC ENERGY COMMISSION
Technical Information Service Extension, Oak Ridge, Tenn.

REACTORS—POWER

Printed in USA. Price $3.00. Available from the Office of Technical Services, Department of Commerce, Washington 25, D. C.

AEC Technical Information Service Extension
Oak Ridge, Tennessee

TID-7553
(NAA-SR-2600)

PROCEEDINGS OF THE SRE-OMRE FORUM

Held at Los Angeles, California

February 12 and 13, 1958

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

PREFACE

In furtherance of the Atomic Energy Commission's policy to assist industrial, municipal, and other groups in keeping currently informed on the status of the Civilian Power Reactor Program, the Division of Reactor Development sponsored an information forum on the Sodium Graphite and Organic Moderated Reactor development programs on February 12 and 13, 1958. This was the second in a series, the first of which was held on November 8 and 9, 1956. The Atomics International Division of North American Aviation, Inc., which has the responsibility for these programs, was the host at this forum, held at the Institute of Aeronautical Sciences Building, 7660 Beverly Boulevard, Los Angeles 36, California. During the first day, the subject of discussion was concerned with the Sodium Reactor Experiment and the associated Edison Steam Electric Plant. This was followed by a description of the full-scale SGR, which is being designed for the Hallam Nuclear Power Facility and will be constructed near Hallam, Nebraska. The morning of the second day was devoted to a discussion of the Organic Moderated Reactor Experiment, followed by a description of the full-scale OMR, which is to be built at Piqua, Ohio.

This report includes the papers presented at this forum, together with reproductions of the slides shown during the two sessions. Each group of slides is numbered separately and is included at the end of the contributing author's paper. Included also is a record of the discussion periods following each of the two sessions.

LM-00182
Whole document

TABLE OF CONTENTS

FOREWORD

U. M. Staebler

Chief, Civilian Power Reactors Branch
Division of Reactor Development
United States Atomic Energy Commission

It was just four years ago that we completed months of intensive planning and review and obtained approval for proceeding with a formal civilian power reactor development program. That program contemplated the building of five experimental power reactors - each of a different type. According to the original plans, four of those reactors should be completed now. These four reactors are complete and have gone critical, although they have not all achieved their full design power level. On the other hand, one reactor already has demonstrated that it can operate at over twice its design power level. I believe that others soon will demonstrate similar capabilities of higher performance. One of those four originally planned reactors which has operated is the Sodium Reactor Experiment, which is a subject for discussion at this information meeting.

This is a very much oversimplified summary of progress in our program. I would not want to imply by it that all has gone exactly as scheduled or that there have not been difficulties and disappointments along the way. You know as well as I do that this would not be a true picture. If that were, in fact, the case, it probably would be a pretty strong indication that we were too conservative and lacked imagination in the planning of the program in the first place. I am sure you appreciate that for a truly experimental program we must be prepared for the unexpected and must not get discouraged if the job gets a little more difficult than some might have expected or hoped.

Just as any development program must be prepared for the unexpected, it cannot be a static program. Experimental results uncover new problems or suggest new directions. New ideas warrant careful consideration and, in some cases at least, they justify major new projects. The civilian power reactor development program has been subject to constant scrutiny, revision, and expansion.

One of the reactor experiments added to the program since its original formulation already has been completed and is operating as planned. That is the Organic Moderated Reactor Experiment, which also will be discussed in detail during this two-day meeting.

Reactors such as the Sodium Reactor Experiment and the Organic Moderated Reactor Experiment are built and operated as part of our efforts to establish a technological basis for a nuclear power industry. These projects - and others like them - constitute the vitally important first steps toward our objective of *economically* justified nuclear power. While they are vitally important first steps, they are only first steps. They must be followed by other projects which extend the technology, improve the engineering, and reduce costs. Ultimately, we are seeking a situation in which management of a utility can decide to build a nuclear power plant with sound economic justification and without further assistance from the government. I do not believe we can afford to wait for that situation to arise mainly through increased costs of other fuels. With this objective in mind, it is clear that our development program must have a broad base both technically and administratively.

The most difficult phase of the development program from the viewpoint of program planning and program direction is what I call the transition phase. That is the phase between the reactor experiment which, in general, will be largely government financed as research and development, and the commercial nuclear power plant. The difficult part of this phase is not in terms of technical requirements but rather in terms of program policy. Some of the difficulty arises in determining just what constitutes research and development.

There is no disagreement – to my knowledge – regarding the need to build reactors as part of this transition phase. There is some disagreement regarding the timing for construction of successive plants. There is much disagreement regarding proper financing arrangements and technical responsibility for successive plants. I do not believe there is any disagreement that the program objective is economically-competitive nuclear power without government assistance, or that we have not yet reached that objective.

We want an aggressive development program. An aggressive program requires that planning for the projects which are to follow as part of the transition phase must proceed on the basis that there will be favorable results from research and development, including the reactor experiments. These plans are, of course, always subject to review and redirection as the experimental results become available. It is to be expected that there will be cases which warrant cancellation or postponement of plans made in advance of having a sound technical basis. There has been frequent criticism of the long-time delays in contract negotiations. Very often, the real difficulty is the absence of a satisfactory technical basis for the proposed arrangement.

In the case of the sodium-graphite and organic reactor projects, plans are well along for new reactor projects which will take us into the transition phase of their development. A contract has been signed by the Commission and Consumers Public Power District of Nebraska, providing for construction of a nuclear power plant with a capacity of 75,000 electrical kilowatts, using a sodium-cooled, graphite-moderated reactor. Contract negotiations are in progress for construction of a nuclear power plant with a capacity of 12,500 electrical kilowatts at Piqua, Ohio, using an organic-moderated and -cooled reactor. You will hear reports relative to the technical basis for these projects, also.

Actually, the subject matter for this meeting represents a good cross section of the civilian power reactor program as a development program. It should provide you with a good sample of progress, problems, and potential. Emphasis in the presentations will be on technology, plant designs, and – where available – operating experience. These are the things which I believe will make the real difference in the nuclear power industry. An industry will grow ultimately only if there is a sound technological foundation on which it can grow. The results we are obtaining now from projects planned four years ago will provide a significant start for that technological foundation. These results will provide a better fix on how difficult the job ahead is going to be. They will give guidance to future development and design effort. We are learning where the real hurdles are and where major improvements can be made.

I believe that continued and accelerated progress will depend more on *ideas* than on any other single factor. Fortunately, no one has a monopoly on having ideas. I'm sure each of you will have many ideas arising from your individual backgrounds stimulated by the reports and discussions you will hear during this meeting. Some of these ideas undoubtedly will be good ones. Good ideas are important to us, but they will be useful only if they are used; they will fade away and be forgotten if they are not acted upon. I know that important ideas will be born here. I hope they will not be allowed to fade away and be lost to the program.

If anyone here still has the impression that a lack of extensive experience in the nuclear field disqualifies him from having ideas important to power reactor development, I urge that he change that notion immediately. Nuclear power plants embrace many different skills and technologies. It often occurs to me that one of the difficulties in the field today may be that we have tended to overemphasize the nuclear aspects of design to the detriment of more conventional engineering aspects. This thought is supported by the fact that essentially all of our important delays and set-backs in nuclear power plant construction and operation are traceable to factors which are not related to the fact that a reactor is involved at all. The difficulties have been caused by such things as inspection practices, chemical corrosion, thermal stresses, and other mechanical design problems. This does not mean that there are not major problems associated with the physics of reactor design, with fuel elements, and with effects of irradiation. Rather, these problems generally are given better recognition and greater attention in the development effort and in project management.

I wish to set as the theme for this conference – emphasis on technology. That is the only sound basis for nuclear power. Success in providing technology will determine ultimate success in meeting program objectives. Success in providing technology depends upon ideas properly applied. You will have ideas. You can contribute to the success of the development effort. That is why we are glad to have you here.

SESSION I

SODIUM REACTOR EXPERIMENT

FEBRUARY 12, 1958

WELCOME

C. Starr

Vice President, North American Aviation, Inc.
and General Manager, Atomics International Division

On behalf of the Atomics International Division of North American Aviation, I would like to welcome all of you to the 1958 SRE-OMRE Forum.

Since the last forum was held a little over a year ago, both the Sodium Reactor Experiment and the Organic Moderated Reactor Experiment have been brought to criticality: the SRE in April, 1957 and the OMRE in September, 1957. During the next two days we plan to bring you up to date on information gained since the last meeting, both in the area of reactor operating experience and in the background technology related to these two reactor concepts.

Basically, I can say that all of our experience since the last forum has resulted in an increased feeling of enthusiasm about the possibilities of these two approaches to atomic power. You will hear from the various project engineers during the forum about the problems that have arisen, how some of these have been solved, and the areas that remained to be investigated. I am sure that these speakers will also reflect our confidence and enthusiasm toward the sodium graphite and the organic reactor concepts.

We will start the program this morning by showing you a film on the SRE which depicts some of the background material related to the general design and construction of the experiment. Although some of this material may be in the nature of a review for a number of you, we feel that it is worthwhile to do this so that we can all start off the meeting with the same general background.

L 1

OPERATING EXPERIENCE WITH THE SRE

AI

L. E. Glasgow *

The results of the dry subcritical experiment (extrapolated critical loading of approximately 22 fuel elements), the wet critical experiment (critical loading of 33 fuel elements at 350° F), and the low-power physics measurements are discussed. Data from engineering tests, including temperature distributions and results of transient tests, are presented, together with information on engineering improvements resulting from the tests. Experience obtained from power operation, with particular emphasis on items of interest for the operation and the maintenance of full-scale Sodium Graphite Reactor Plants, is described. Future plans for the experiment, operation of which has been quite encouraging thus far, are presented.

* Group Leader, Sodium Graphite Reactors

I. INTRODUCTION

The major design features of the SRE are very similar to those of a full-sized power reactor. It was not designed, however, to be a continuously operated nuclear plant, but rather a large-scale experimental installation constructed for the purpose of providing definitive answers to the many technical questions associated with the sodium graphite reactor concept. During the course of the design and construction of the plant, extensive engineering analyses were carried out. Where the results of the calculations clearly showed that improvements in the design could be made, these improvements were incorporated into the system. In other cases, because of the many assumptions involved in the calculations, the recommended changes were deferred pending actual tests in the reactor complex. Consequently, a careful and extensive testing program was planned and is being carried out. This program includes physics tests, engineering tests, and power operation. The results of these tests find application in improving the SRE and provide the physics and engineering information needed for the design of a full-scale plant. It is expected that modifications to the SRE will be made periodically to raise the performance level, increase the reliability, improve the operational characteristics and provide for easier maintenance. The rather extensive test and modification program associated with the SRE would not be required on a full-scale plant; consequently, a power reactor could be started up much more quickly.

II. PHYSICS TESTS

The first data obtained for the physics tests were taken during the course of construction and involved a complete and direct determination of the "as built" weights and dimensions of all of the core components. This was done to permit the best possible comparison between the calculated and measured critical masses.

Immediately following the usual and uneventful functional testing of the reactor equipment, and in preparation for the criticality tests, an antimony-beryllium source was loaded into the center of the reactor. The source strength was 5×10^5 neutrons per second and was easily detectable by the five special fission chambers located in the core.

An ambient dry critical mass of 22.2 fuel elements was determined by loading the sodium-free reactor to 21 fuel elements and extrapolating the normalized inverse count rate *vs* loading curve to the zero intercept (Fig. 1).

After the dry critical test, the reactor and the sodium piping were electrically heated uniformly to 350° F and the system was filled with sodium. Sodium circulation tests confirmed the integrity and the design criteria of the heat transfer loops. The wet critical experiment was then conducted in much the same way as the dry subcritical test, with the exception that the reactor was brought slightly past the critical point as indicated by a slowly increasing counting rate with all the reactor parameters fixed. The loading curve is shown in Fig. 2, and indicates a wet critical mass of 32.6 fuel elements at a temperature of 400° F.

The purpose of determining both the dry and wet critical masses was to permit a direct evaluation of the nuclear worth of the sodium. This quantity cannot be determined from geometrical measurements because the spaces between the moderator cans are inaccessible in the assembled core. An error of 0.020 inch in the gap determination would introduce an error of 10 per cent in the calculated critical mass. The difference between the wet and the dry critical masses establishes the worth of the sodium at approximately eleven fuel elements.

The two-group theoretical calculations proceeded from first principles and were made without the benefit of a critical assembly. The values used for the resonance escape probability and the unit cell flux distribution were determined from an exponential experiment. A comparison of the most recently calculated and measured critical masses normalized to ambient temperature is shown in Table I.

TABLE I

NUMBER OF FUEL ELEMENTS FOR CRITICALITY

	Dry Critical	Wet Critical
Measured	22.2	33.2
Calculated	21.1	28.4

The agreement shown for the dry critical case can be considered fortuitous, as the theoretical accuracy has been estimated to be about 20 per cent. This is only a single data set and one cannot draw a too definite conclusion about the calculational methods from this comparison. Another set of data will be available from the Th-U fuel loading which is expected in July or August.

During the preparation for, and immediately following the reactor power runs, additional physics tests were made. These tests included a measurement of the overall isothermal temperature coefficient of reactivity. This quantity is positive, as predicted, but is a factor of ten smaller than the theoretical value. It varies approximately linearly from about + 6 x 10^{-4} per cent reactivity per °F at 500° F to about + 2 x 10^{-4} per cent per °F at 750° F. The time constant associated with this coefficient is approximately three minutes. The fuel temperature coefficient is negative and amounts to -10 x 10^{-4} per cent reactivity per °F. The time constant associated with this coefficient is approximately two seconds; consequently, the negative coefficient controls the reactivity on short-time temperature fluctuations. The strong, fast-acting negative coefficient, combined with the fact that the fuel temperature exceeds coolant temperature under power operations, explains the observed extreme stability exhibited by the reactor during power runs.

Data on the worth of the individual fuel elements, safety rods, and control rods have been taken. The data have not been completely reduced, but preliminary results indicate that the excess reactivity available with a forty-three element loading is approximately 3.2 per cent and the sum of the four control rods is 4.5 per cent. An additional 1.6 per cent of reactivity is available by replacing the stainless-steel control and safety rod thimbles with Zircaloy thimbles, bringing the total excess to 4.8 per cent which is sufficient for approximately 250 days of full power operation.

The gain in control rod worth resulting from this change is equivalent to one additional control rod. Additional physics tests will be made to confirm these estimates.

III. ENGINEERING TESTS

During the power runs, a number of planned "scram" tests were carried out to observe the rates of temperature change throughout the system and to compare these with the values which were calculated in the system analysis studies. Uncertainties in the calculations derive from having to make assumptions regarding the degree of thermal coupling existing between the graphite and the coolant, the hydraulic impedance to the thermal driving head which regulates the flow rate after the pumps are stopped, the amount of internal thermal convection in the core, and the degree of mixing of the hot and cold sodium in the upper plenum.

The thermal transient in the core tank nozzle agreed very closely with the calculated value. The data indicate that a scram with a full temperature gradient across the core would introduce a stress in the core tank nozzle, which would exceed the yield strength by a factor of approximately 2.5. The way in which this stress develops is shown in Fig. 3. This transient nozzle stress is the only reason that the reactor has been limited to one-third of full power. This problem is being eliminated by the addition of an eddy current brake to control the convective flow in both the primary and secondary main coolant loops. These will be installed as shown in Fig. 4. The device itself is shown in Fig. 5, and is simply a means of applying a 6,000-gauss d-c magnetic field across a flattened section of pipe. The head loss at full flow is less than 1 psig. The forces from the induced eddy currents in the liquid metal coolant act to oppose the fluid flow. Varying the magnetic field varies the flow rate. Programming the magnetic field will be done automatically, using a thermocouple signal from the fuel coolant channel, so that the sodium coolant temperature will remain fixed. The addition of this device will eliminate transient thermal stresses in the core tank nozzle and in the moderator cans. The brakes which are now installed in the SRE were previously tested in a sodium test loop. The results of these tests are shown in Fig. 6.

The time constant of the braking action is approximately 2 seconds, compared to a 3 minute time constant for the coolant channel temperature change. The brake will, therefore, permit close control of the reactor transient temperatures. A second-order benefit inherent in the system is the back-up flow control it will provide in the event the pumps fail to drop out during a "scram." Should the brakes come on inadvertently, the reactor will be scrammed automatically by off normal flow circuit and the coolant channel temperature monitor.

An unexpected result from the first power run was the 50 per cent excess log mean temperature difference observed on the main intermediate heat exchanger. Between power runs, 60 thermocouples were distributed over the length and around the girth of the heat exchanger shell. The steady-state temperature distribution along the shell is shown in Fig. 7. The temperature curve is seen to drop uniformly at the straight portions of the exchanger but remain horizontal around the bend, indicating that no heat transfer is occurring in that section. Examination of the construction photographs disclosed a wide gap between the tube bundle and the shell, and since this section is not baffled, the fluid simply bypasses the tubes. The loss of this amount of heat-transfer area accounts for the high log mean temperature difference. Transient data taken on this equipment disclosed that rather complete stratification occurs on the shell side of the exchanger, which produces a 90° F temperature difference from the bottom to the top of the shell for an inlet to outlet initial temperature difference of 135° F. The resulting stresses have not yet been calculated. It is planned to simulate, by mechanical means, the expected thermal loads in the nozzles and the tube sheet to shell transition in order to provide a check on the calculated stresses at these points, which indicate that the yield point will be exceeded at full-power operation. A replacement for the present heat exchanger, embodying improvements based on these studies, is planned. In the interim, the present exchanger will be operated without modification.

Particular attention was given to the transient phenomena associated with the moderator cans. Early simulator studies pointed to excessive stress at the can head coolant tube juncture. This joint was redesigned to take a step change of 460° F. The reactor transient tests showed that the maximum temperature difference between the side panel and coolant tube will never exceed 100° F under the worst conditions and with no flow programming, so that we have a comfortable margin here of a factor of 4-1/2. The eddy current brake should reduce this temperature difference to nearly zero, which will eliminate transient stress in the cans altogether.

During the first power run using a 36-element core loading, steady-state temperature observations revealed that the moderator coolant flow persisted even though the programming valve was closed and the flowmeter indicated a zero signal. This flow amounted to 50 per cent more than that required to produce a temperature match between the moderator exit sodium and the fuel-channel exit sodium; it is due to leakage past the grid plate and is excessive only for a light core loading. At a 43-element core loading, the leakage flow is just equal to the design flow rate. Control of this flow is not needed for normal power operation but is required for experimental purposes. This control will be achieved by installing a 60-gallon per minute electromagnetic pump in the moderator coolant line to pump the leakage back to the main coolant stream.

IV. OPERATIONAL EXPERIENCE

There have been three power runs to date. The first generation of electrical power by the SRE occurred at 12:47 P.M. on 12 July 1957 and the run continued until 15 July 1957. Although most of the first run was performed at less than one megawatt of electrical power, a peak operation of 1.7 electrical megawatts with a thermal power of 6.5 Mw was achieved. The second power run occurred on 25 and 26 July with an electrical output of 1 Mw, and the third extended from 7 November to 20 November with a maximum electrical power of 2 Mw, corresponding to a thermal power of 7.6 Mw. During the first, second, and third power runs, the electricity generated amounted to 59,800 kwh, 18,400 kwh, and 290,850 kwh, respectively. The temperature of the sodium entering the reactor ranged from 475° F to 505° F during the runs, and during the peak power production of the third run, the temperature of the sodium leaving the core was 675° F (cf. 960° F expected for full-power operation at 20 megawatts). The corresponding steam conditions for the third run were 628° F and 470 psig. To date, we have logged 300 hours of power operation.

Start-up of the plant is quite direct and uncomplicated. A typical start-up proceeds as follows: First, the various reactor systems such as the tetralin system, inert gas system, and the reactor vent system are checked out. At the same time the various safety circuits are checked and the safety rods are given a functional check by dropping them. The steam generator, which has been filled with pressurized water, and its associated piping are heated to 350° F and the system is filled with sodium. Full sodium flow is then established. The source strength is checked just prior to cocking the safety rods. The control rods are withdrawn one at a time until criticality is reached. Reactor power is increased on a 30-second period until reactor heat can replace the electrical heat. The overall temperature is raised isothermally to 500° F, and the feedwater pumps are turned on. All of the steam which is generated at this time is dumped to the condenser except that which is used to preheat the steam turbine. The reactor power is increased until approximately 15 per cent of full power is reached. At this time, the sodium flow rate is reduced to provide superheat and establish the operating temperature gradients. The turbine is brought up past the operating speed to check the overspeed trip. It is then brought back to speed and put on the line. Power is then increased by increasing the flow at a fixed temperature gradient. The reactor is put on automatic control and operates steadily without drift or fluctuation. The run is usually terminated with a planned scram to provide transient data. The start-up time with our present ultra-conservative procedures is approximately eight hours.

The steady-state periods in power runs served to demonstrate the excellent stability of the reactor steam generator complex. During one 17-hour steady-state period, a timer connected to the control rod indicated a mere two minutes of integrated operation, and this included shim adjustments for xenon build-up.

During the power runs, several measurements of shielding effectiveness were made. The shield blocks above the galleries and the loading face shield above the reactor were found to be more than adequate. In fact, some measurements directly over gaps between shielding blocks above the galleries, which purposely had not been filled prior to the runs, indicate that gamma streaming through these gaps is approximately a factor of ten less than the calculated value. That is in the order of 100 mr/hr measured as opposed to values of the order of an r/hr calculated.

Preparations for the power runs and the operations during the physics tests provided approximately 700 fuel transfer manipulations with the fuel handling cask. This equipment appears to be satisfactory. Time for one fuel element change including the washing cycle amounts to two hours.

The hydraulic balancing holes in the sodium pump impellers have been increased in number from three to seven, and the size has been increased to 15/64 inch from the original 9/64 inch. Since this modification was made, the pumps have logged 3,200 hours of trouble-free operation.

The cold traps appear to work well if the initial oxide content in the sodium is less than, say, about 50 ppm. At this concentration, the traps easily bring the oxide concentration down to 10 ppm or below and maintain it there. During a maintenance operation, the oxide concentration rose as high as 130 ppm. Under these conditions, we had difficulties with plugging of the traps. An improved trap designed to operate at the higher concentrations is being fabricated and will be installed in the system.

V. MAINTENANCE EXPERIENCE

Several maintenance activities, including the replacement of a few bellows seal valves in the sodium service systems and the removal and the replacement of the disposable cold trap cartridge, were also carried out. The most important maintenance operation performed during this period was the removal of a damaged fuel element from the core. These operations, which were performed with relative ease even though the sodium has become activated, provided confidence in the maintenance techniques developed for sodium graphite reactors.

The damaged fuel element was discovered when difficulty was encountered in trying to remove it with the fuel handling cask subsequent to the second power run. When the shielding plug was finally freed, it was found that the hanger rod had parted and that the seven-rod cluster had remained in the core. Examination after the sodium level had been lowered showed that the cluster was protruding approximately 18 inches above the top of the moderator can and that the portion protruding was bent at approximately 30° to the vertical. The cluster, which had an activity corresponding to 200 r/hr at one foot, and other broken pieces from the assembly were removed in a single day with grappling tools devised for the purpose.

The investigation of operating records indicated that the element, which was loaded between the wet critical experiment and the first power run, had never occupied its proper position in the core. This discovery illuminated a previously unexplained small decrease of reactivity which had been observed prior to the power runs. The records showed that the particular fuel element was loaded at a time when the sodium level had been lowered to the tops of the moderator cans.

Calculations indicate that under these conditions freezing of sodium can occur when a cool element enters the fuel channel. This freezing of sodium as the element was lowered undoubtedly explains why the seven-rod cluster never completely entered the core. Operating procedures have now been modified so that fuel and other core elements are never loaded without the presence of the sodium pool above the core, and a soaking period in the pool is now specified to ensure that the elements will be hot enough to prevent freezing before they are placed in the coolant channels.

VI. FUTURE OPERATING PLANS

Future operating plans for the SRE include full power operation some time in April. It is also planned to reload the core with a 7.6 wt.-% Th-U fuel. Irradiation of the various experimental fuels will begin with the higher-power operation.

Reactor developmental efforts will be directed toward raising the reactor power level as well as toward studies and reactor tests on alternate reactor components such as improved control and safety rods, alternate pump designs, and improved moderator elements.

Additional sodium steam generators will be installed in parallel with the present equipment to permit testing under service conditions.

Changes in the system will be made to permit the reactor to be "load following" as contrasted to the present load forcing mode of operation.

A number of physics tests will be conducted with the new core loading, and improvements in the testing procedure will be made, utilizing pile oscillator techniques in addition to the classical methods.

VII. CONCLUSION

The main conclusions to be drawn from the operation of the SRE ~~thus far are~~ as follows:

1) Reasonably good agreement has been obtained between the calculated and measured critical masses.

2) The simulator studies have provided a reliable means of anticipating system performance.

3) The plant has performed extremely well – up to 1/3 of full power. The stability of the reactor and reactor steam plant complex is particularly impressive.

4) If the SRE were treated as a power plant, the initial start-up time would amount to two months, including normal start-up testing.

5) Excess thermal stresses are not an inherent characteristic of sodium cooled reactors. With the information gained from the SRE, it should be possible to design a reactor free from these stresses.

6) Maintaining the oxygen concentration at or below 10 ppm can be easily accomplished under normal operating conditions.

7) Maintenance and modifications of the main primary and secondary sodium systems can be directly and easily accomplished.

8) The shielding for the SRE is more than adequate. In particular, the gap tolerances are more stringent than necessary and economies can be realized in future reactor designs by relaxing them.

9) With the rectification of the problems mentioned, full power operation should be possible in April.

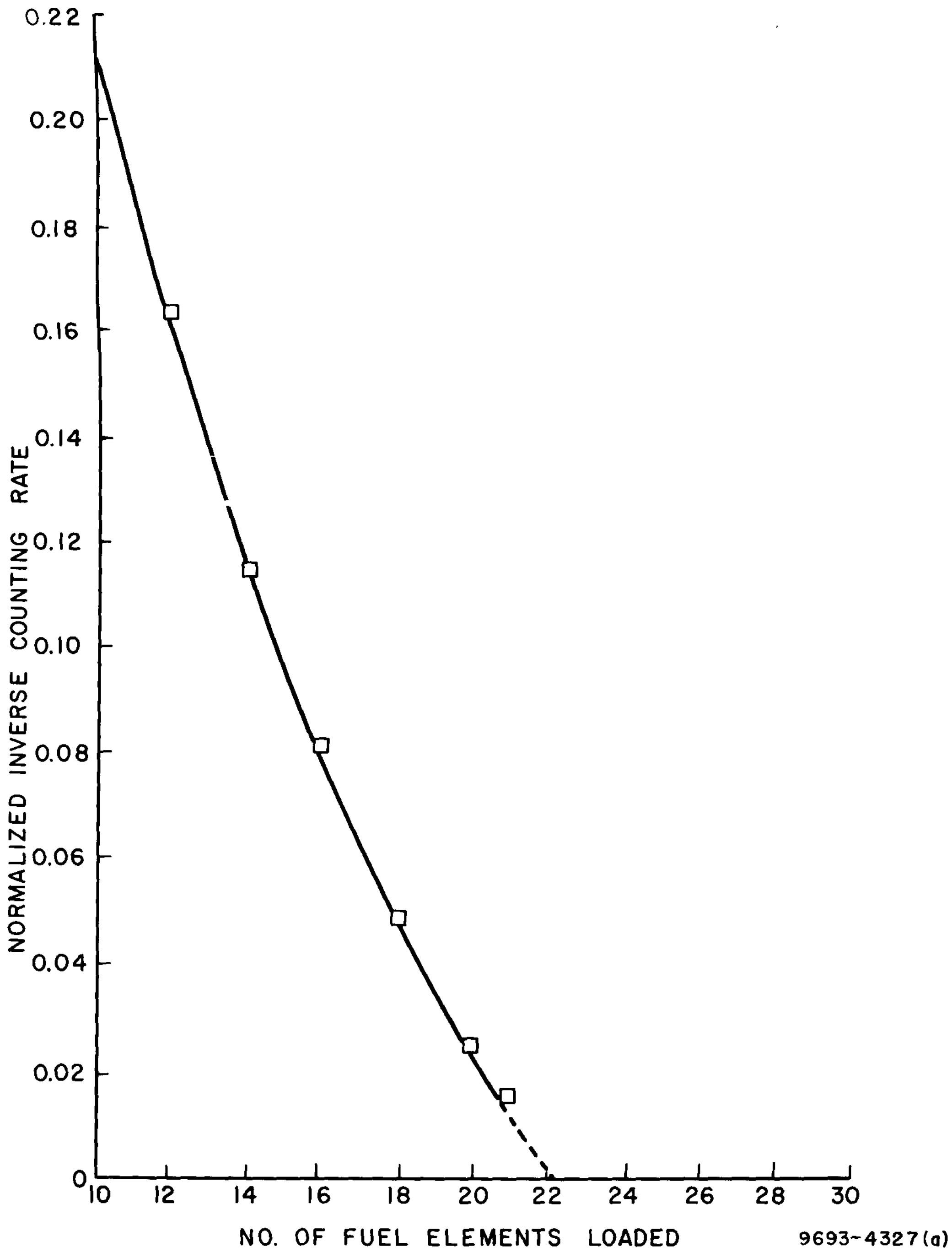


Fig. 1. Approach to Criticality of Dry SRE

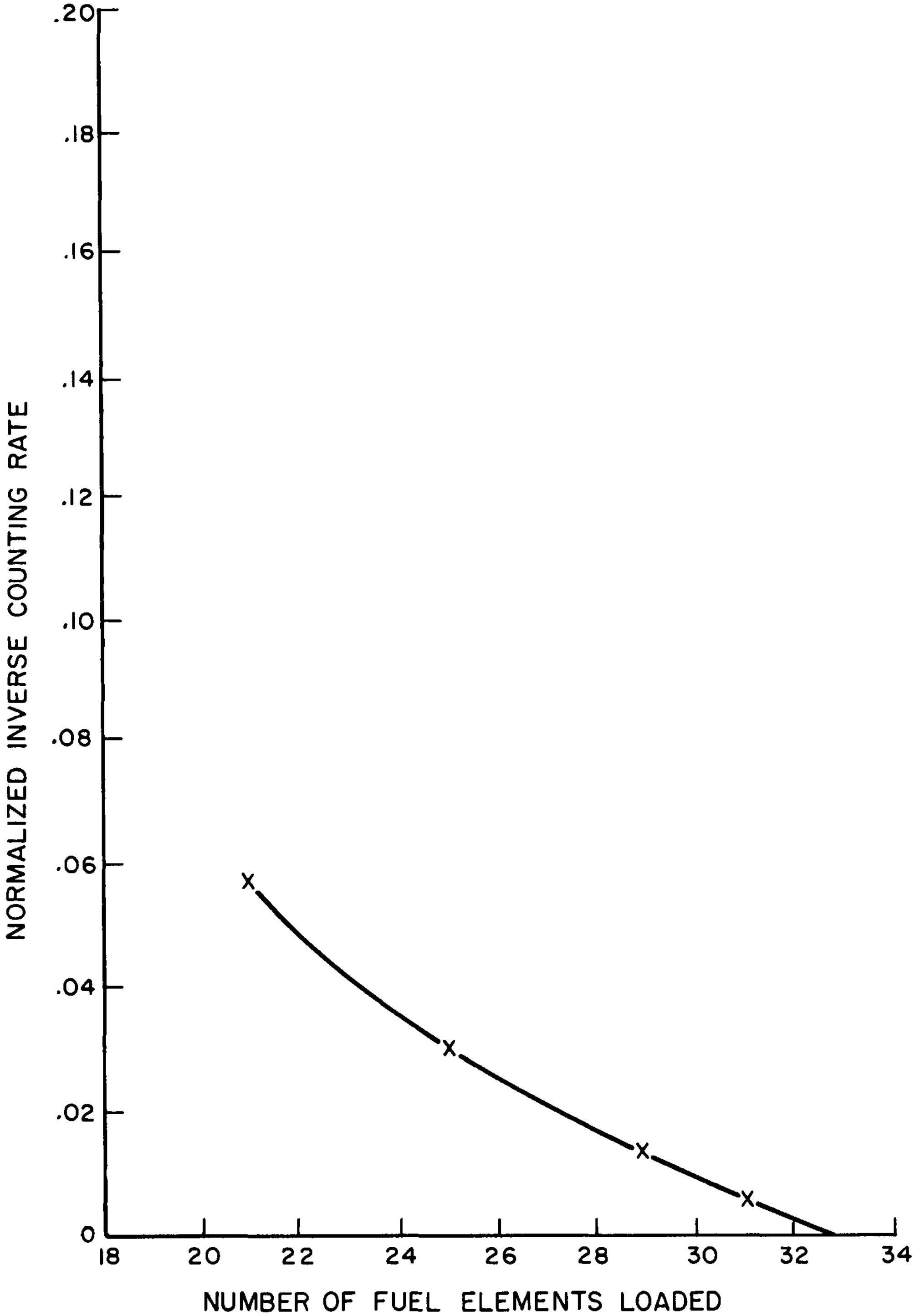


Fig. 2. Approach to Criticality of Wet SRE

9693-4328(a)

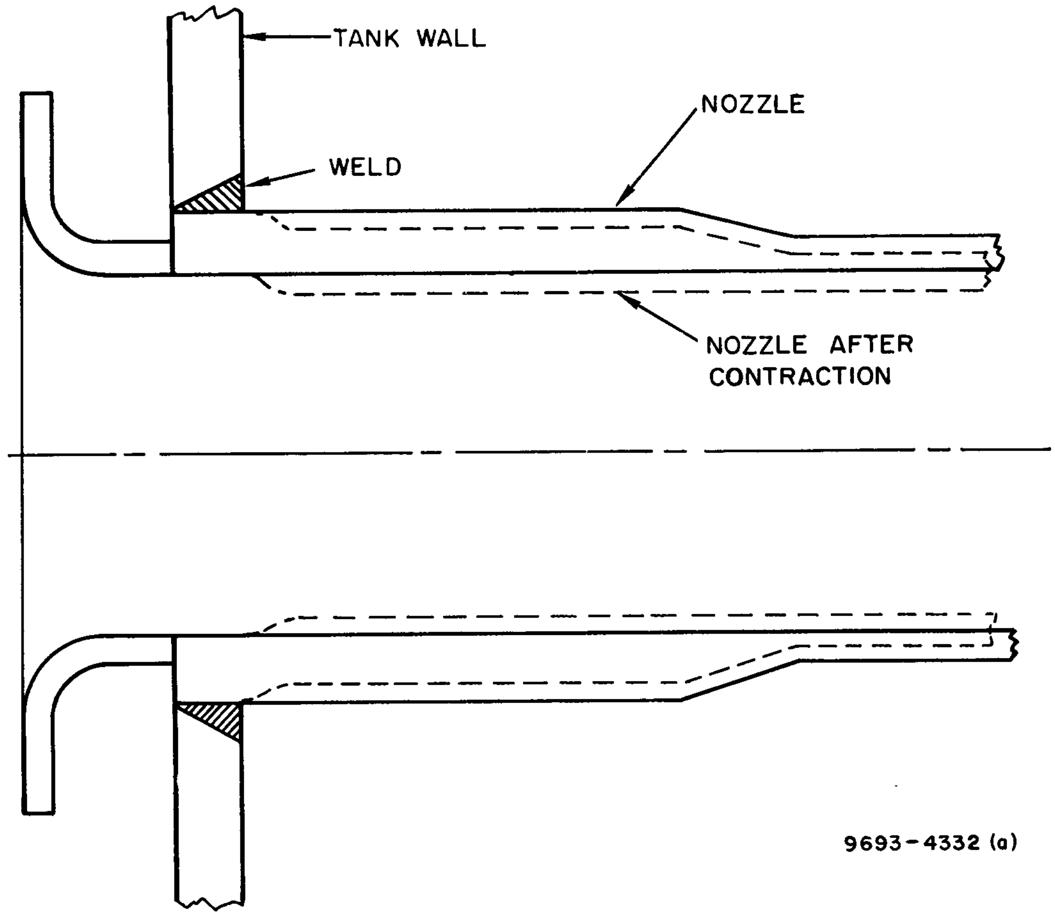


Fig. 3. Core Tank Nozzle

Fig. 3. Core Tank Nozzle

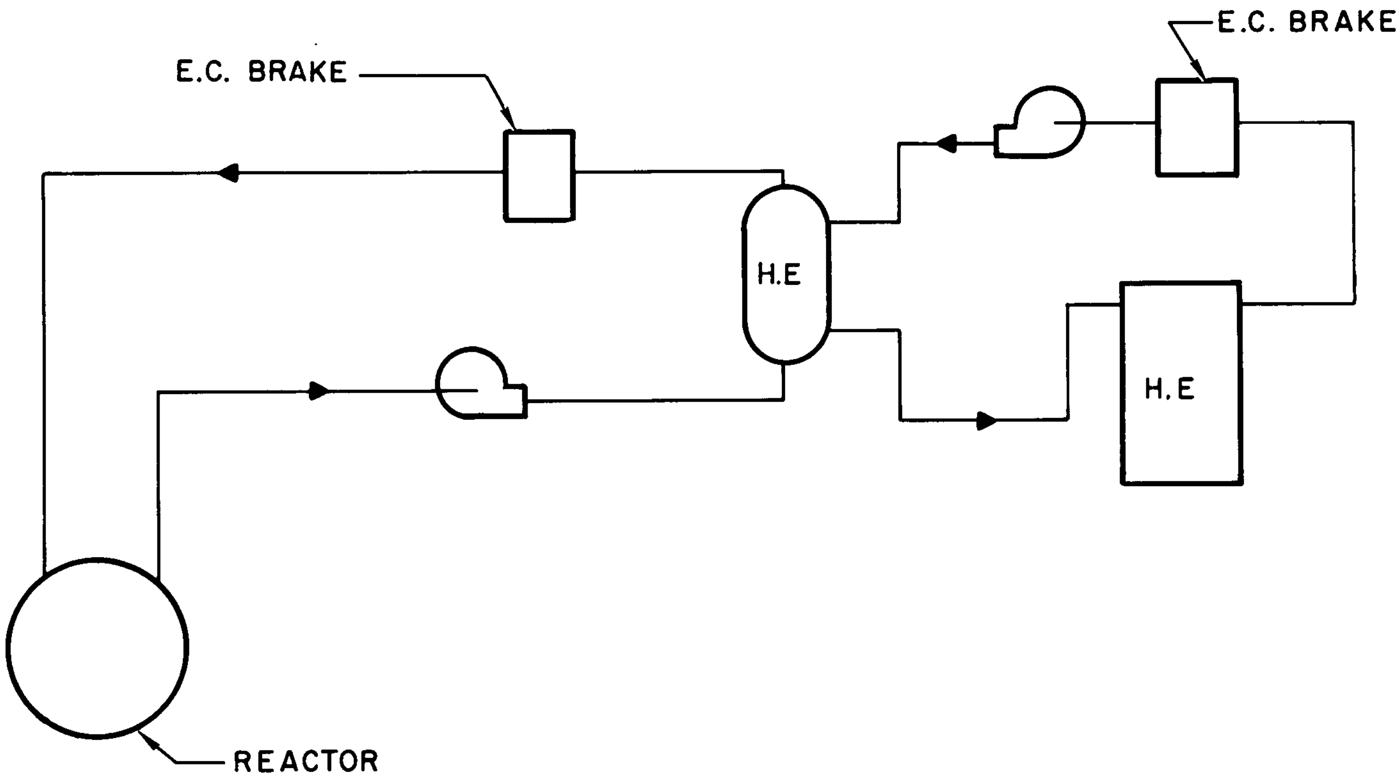


Fig. 4. Simple Heat Transfer Circuit

9693-4334 (a)

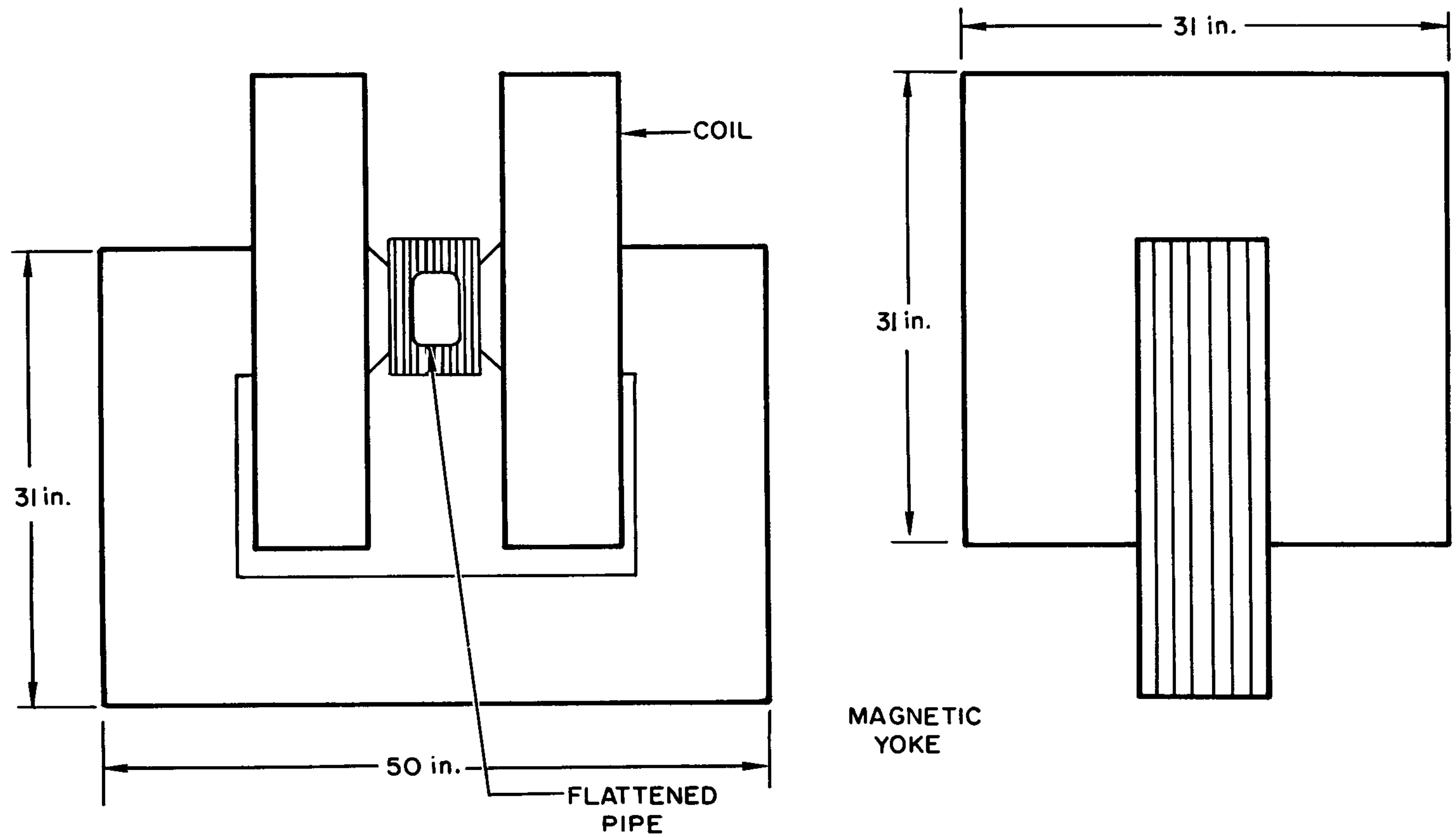


Fig. 5. Eddy Current Brake

9693-4333 (a)

9693-4333 (a)

Fig. 5. Eddy Current Brake

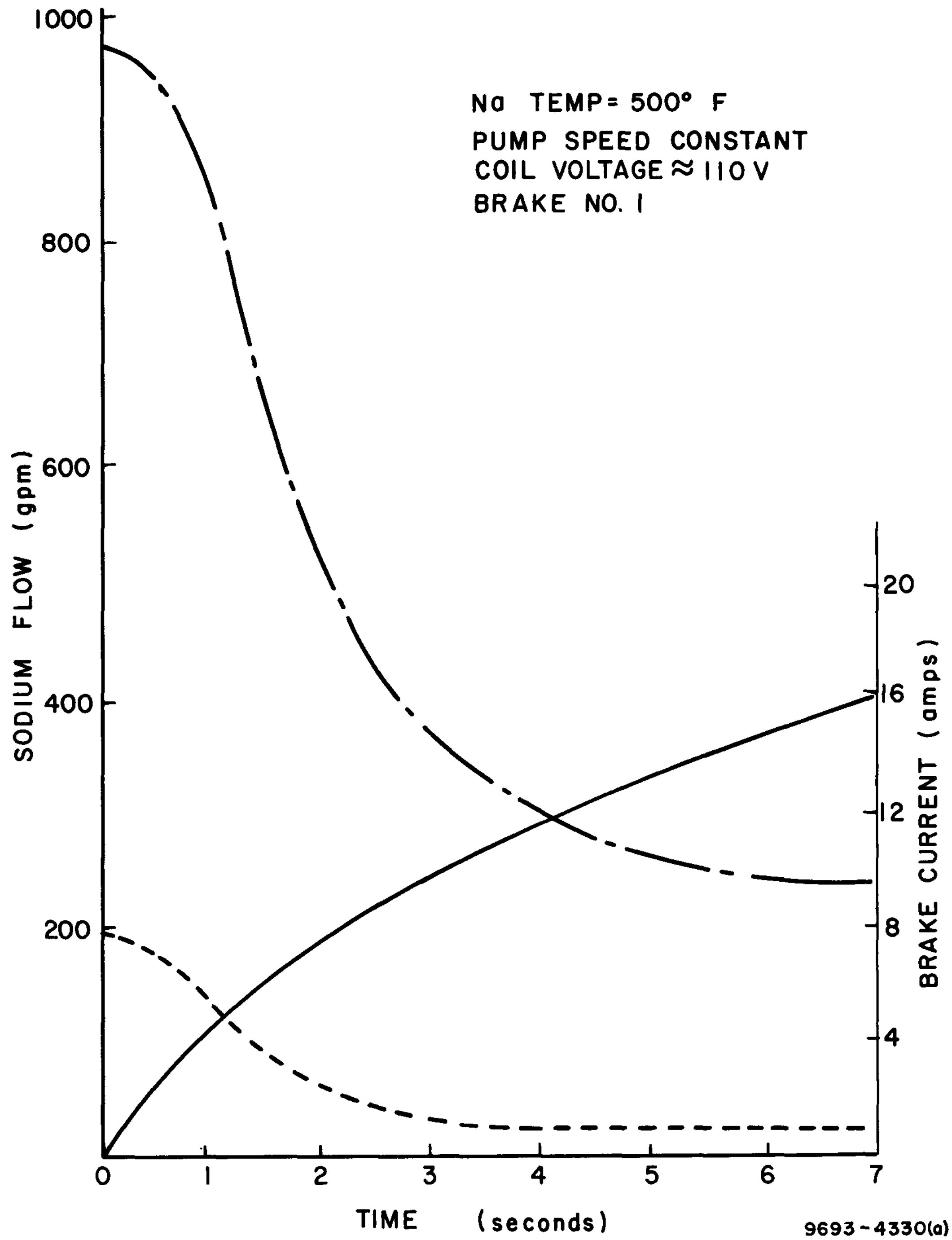


Fig. 6. Eddy Current Brake Performance

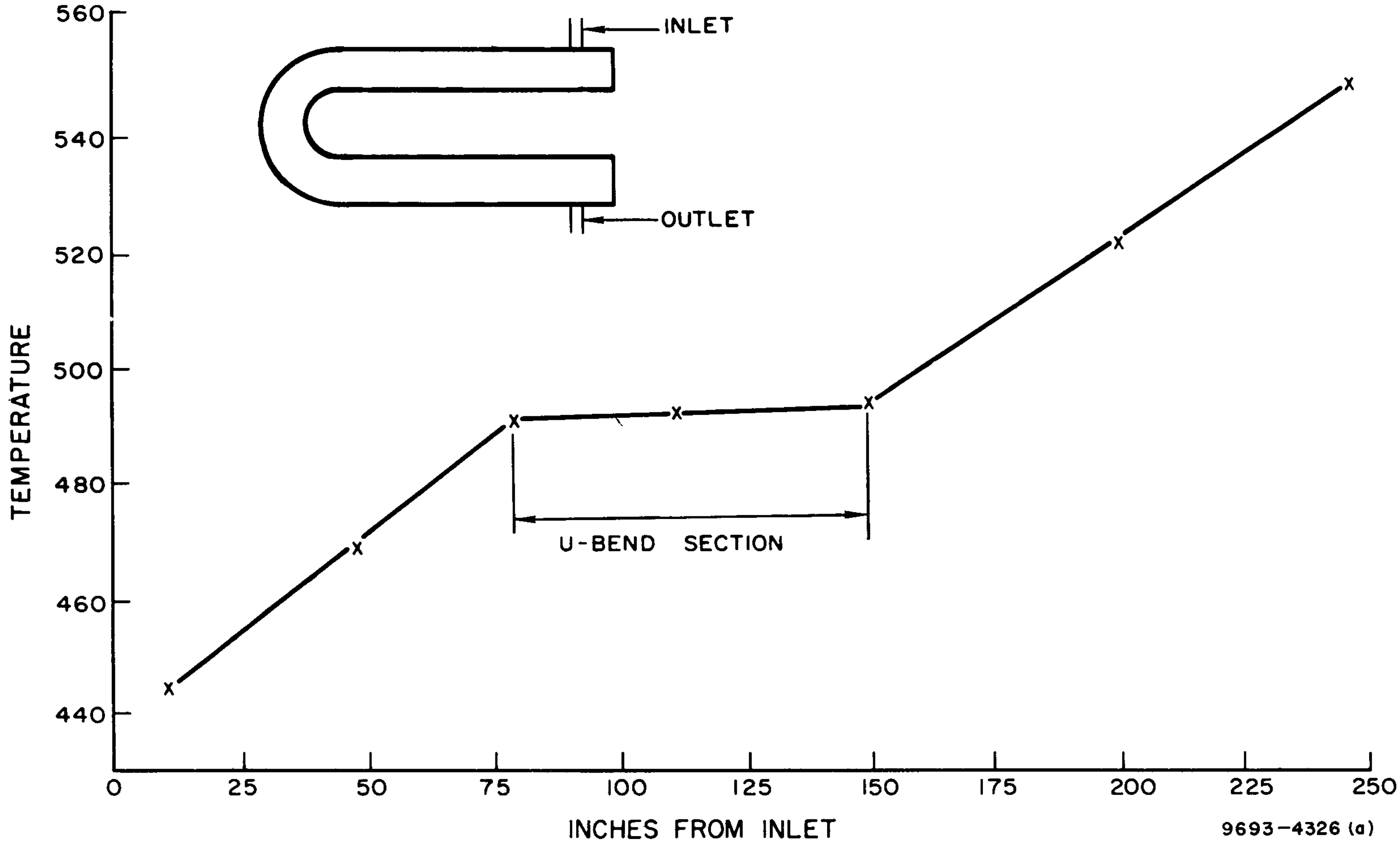


Fig. 7. M.I.H.R. Temperatures During Steady State Operations

SOUTHERN CALIFORNIA EDISON COMPANY
STEAM ELECTRIC PLANT OPERATION

A. C. Werden, Jr.*

The steam electric plant was initially placed in operation on July 12, 1957. It ha~~s~~ ~~since~~ been operated intermittently for a total of over 300 hours during which period it has generated approximately 370,000 kilowatt hours. This operation was preliminary in character and was performed under subnormal design conditions as imposed by the reactor in order to study reactor performance at low temperature and power. The maximum output during these trial runs was 2000 kilowatts. The steam conditions were 480 psig and 630° F as compared to design values of 600 psig and 825° F. Allowing for these non-design conditions, the operation was generally considered to be satisfactory and gave indication of improving as temperatures were increased. Operating start-up procedure as well as operating experience with the steam generator, together with the feed water condition problems, are presented.

* Southern California Edison Company

I. INTRODUCTION

The Southern California Edison Company has contracted to purchase the surplus heat energy from the SRE and to convert it into electrical energy for distribution over the Company's system.

Under this contract with Atomics International, and as approved by the Commission, the Company has installed and is operating a 7500-kilowatt steam electric plant and is doing so at the Company's sole expense.

To date, the total costs to Edison Company are slightly under 1.5 million, which includes design, construction, operation, and the purchase of the heat energy at 45¢ per million Btu.

Before going into the details of operation, I would first like to review briefly the general plant layout and the liquid metal-to-water steam generator, which is an important connecting link in the conversion of reactor heat into electrical energy.

II. DESCRIPTION OF PLANT

A. GENERAL

Figure 1 is a general view of the plant with the reactor building shown in the background.

The Edison control room building is seen in the foreground. Mounted on the platform to the right center is the liquid metal-to-water heat exchanger, or once-through type steam generator, as it is commonly called.

The banded pipe lines to the right of the steam generator conveys the non-radioactive sodium between the steam generator and the reactor's intermediate heat exchanger, located in the reactor building.

To the left, you will note the turbine generator which rests on and is supported by its condenser.

A little further to the left center is the power transformer and metal-clad switch gear. The power transformer raises the generated voltage from 4160 to

69,000 volts for the transmission of the energy into the Edison network. In the left center is the feed water purification equipment and storage tank.

B. PURIFICATION SYSTEM

Figure 2 shows the feed water purification installation in more detail with the 69-kv transmission line in the background. The water available to this plant has a hardness of around 600 ppm and is supplied from local deep wells. It is processed to less than 0.1 ppm total dissolved solids.

This raw water is first passed through a water softener and degassifier, which are the two units seen in the rear center. It is then passed through one of the dual mixed-bed demineralizers shown in the left center. The two storage tanks to the right contain acid and caustic for regeneration of the mixed bed demineralizers.

This regeneration is an automatic process initiated by remote manual control.

C. CONTROL ROOM

The Edison control room is shown in Fig. 3. For the most part, the controls are similar to those of a conventional plant.

Reactor intelligence indicated on this board includes reactor power, sodium inlet and outlet temperatures, sodium flow, and reactor scram and setback alarms.

D. STEAM GENERATOR

Figure 4 shows an artist's sketch and simplified cut-away view of the steam generator. It is an all stainless-steel shell-and-tube type unit, with a developed length of about 80 ft. Sodium enters the left hand leg and flows on the shell side. Water enters the right hand leg and flows through the tubes in a counter flow direction. It is thus evaporated and superheated in one pass through the steam generator.

In the lower view, you will note that the tubes are of double-wall construction with mercury in the annulus. The mercury is pressurized intermediate to the steam and sodium and thus monitors by pressure change for the detection of leaks.

Not shown in this view are about one hundred thermocouples attached to the shell of the heat exchanger and distributed over its entire length. These temperatures are recorded on a multipoint temperature recorder. Of particular importance are those located in the top portion of the steam outlet head adjacent to the upper tubes, and those located in the bottom of the head adjacent to the lower tube rows.

They thus indicate the temperature of the steam in the top and bottom tube rows, and were found to be particularly helpful during start-up operations, as will be discussed later.

III. OPERATION

The basic flow diagram of the steam plant thermal cycle is shown in Fig. 5. The cross-hatched lines entering from the upper left represent the sodium connections to the reactor heat transfer system. Under design conditions, sodium would be received at 900° F and returned at 440° F. Also under design conditions, feed water at full load would enter at 297° F and be converted to superheated steam at 825° F, 620 psia.

The steam leaving the steam generator is first passed through a direct contact type attemperator which utilizes feed water to maintain constant steam temperature when operating under light load and elevated sodium temperature conditions.

The turbine exhausts into a two-pass surface condenser which in turn is cooled by means of an induced-draft cooling tower.

The condensate is collected in hot wells from which point dual condensate pumps circulate it through the steam jet air ejector and de-aerator, to the suction of the boiler feed pumps. The boiler feed pumps then pressurize the feed water and pass it on through a closed feed water heater back to the steam generator.

The controls for the plant are presently connected to facilitate reactor experiments, and hence the turbine generator is caused to follow reactor output. This is accomplished by means of a pressure-sensitive device located on the main steam header, which automatically operates the turbine admission valves to maintain a constant steam pressure by varying the turbo-generator output.

The feed water to the steam generator is regulated to maintain a constant return sodium temperature and is so regulated automatically by a combination of temperature and flow sensitive devices located in the sodium loop.

IV. START-UP PROCEDURE

There are two principal criteria in the operation of the plant that are important to both start-up and normal operation. One is to maintain a constant sodium return temperature, and the second is to avoid steaming in the steam generator with less than 10 per cent of full flow. This is to provide sufficient scouring to lessen the quantity of solids deposited in the tubes.

The first step in start-up is to raise the temperature of the sodium piping and steam generator to 350° F. The pipe throughout its length is heated by calrod electrical heaters. The steam generator is heated by circulating hot water, which is electrically heated and pumped with one of the boiler feed pumps. This closed hot water loop is pressurized at about 175° psig with nitrogen gas, to avoid boiling.

With the sodium system at 350° F, the second step is to start circulating 350° F sodium from the reactor system, and then to slowly raise the system to an isothermal temperature of approximately 440° F.

Coincident with this latter operation, the steam generator is cut over from its auxiliary heating loop and pressurizing system into its normal operating cycle and pressurized with the boiler feed pump.

At this stage, the feed water flow through the system generator is very low. It is adjusted to maintain a constant sodium return temperature by allowing a little steam to flash through the by-pass pressure-reducer directly into the condenser. By this operation the by-pass pressure-reducer can be set to automatically control pressure in the steam generator, so as to prevent boiling at this low flow. Also, a little steam is allowed to flash into the closed feed water heater as necessary to control feed water temperature.

With the sodium system nearly at a 440° F isothermal temperature, sodium inlet temperature is gradually increased until the desired minimum feed water flow has been established. At this point, steaming is permitted in the steam generator by either a further increase in sodium temperature, or by slightly reducing the steam pressure. When a good quality of steam is available, with at least 50° F of superheat, it is transferred from the by-pass system into the turbine. The turbine is then rolled for a warm-up period, and placed into operation in the conventional manner.

V. OPERATING EXPERIENCE

A. GENERAL

The steam electric plant was initially placed in service on July 12, 1957. It was subsequently operated from July 12th to the 15th, again on July 25th and 26th, and lastly from November 9th to the 20th. In all, it has been operated a little over 300 hours and it has generated approximately 370,000 kwh.

This operation was preliminary in character and was performed under subnormal design conditions, as imposed by the reactor, in order to study reactor performance at low power and low temperature. Subsequent operations are expected to be more nearly at design conditions.

During these trial runs, the steam conditions were 480 psig and 630° F, as compared to design values of 600 psig and 825° F. Partially because of these subnormal conditions, most of the operation to date has been on manual control. Steam pressure, for example, was controlled by diverting about 10 per cent flow through the main steam by-pass directly to the condenser. Feed water was controlled entirely by manual remote control.

Even though this operation was below design values, considerable valuable experience and information was obtained that was of interest to us and that will be helpful in future operations.

During the first three days of operation in July, for example, the steam plant caused several reactor scram operations, whereas we have since operated for twelve days with no indications of the former troubles reoccurring.

Figure 6 is a graph of the principal operating characteristics of the plant during the final ten days of operation beginning November 9th.

The uppermost (solid) line represents sodium inlet temperature, which reached a maximum of 635° F. compared to the design valve of 900° F. Next are two lines, almost superimposed, the dotted one representing steam temperature. and the one denoting the upper boundary of the shaded area recording the temperature of a thermocouple located in the upper surface of the steam generator outlet head. As would be expected under stable operating conditions, these three temperatures should, as they do, follow along together quite closely.

The sodium outlet temperature (represented by the short-dashed line) was maintained at about 460° F during this run. Its maximum variation, with hand control operation, was about 10° F. Such a variation was considered to be acceptable for this particular reactor operation.

You will notice also from the graph that sodium flow (long-dashed line) and feed water flow (alternate dashed and dotted line) remained in a fairly close relationship throughout the entire run.

The kilowatt output is represented by the lowermost (dashed and double-dotted) curve. The maximum capacity permitted was 2050 kw.

During this test run, two scram operations were experienced, one on the 16th and a second on the 17th. The first is shown in more detail in Fig. 8. However, it can be observed on this one that the equipment functioned properly and that the operator was able to control operations satisfactorily.

It is of interest to note that from the experience acquired during the first scram, the time required to return to normal operation was reduced from 12 to 8 hours, even though there had been no particular emphasis given to expediting the return of the unit of service.

The shaded area of the graph represents the temperature differential between the top and bottom tubes in the steam end of the steam generator. The upper boundary is a plot of thermocouple temperatures in the top surface of the steam head, while the lower boundary is a plot of thermocouple temperatures in the bottom surface. The significance of this magnitude of temperature differential is that the flow through the steam generator is not uniform under these low temperature and load conditions. Consequently, we are apparently getting a mixture of superheated and saturated steam out in the head. Contributing to this condition is the fact that the sodium ΔT is only 170° F, compared to the design value of 460° F; and that consequently the maximum degrees of superheat in the steam are only 170° F instead of a normal of over 340° F.

On the optimistic side, however, there is some evidence from a closer examination of the performance data, not conclusively evident on this slide, that the unbalanced flow in the tubes as indicated by temperature appears to be improving with higher inlet sodium temperatures.

One such evidence of improved flow distribution can be noted on this chart, despite the many complicating variables and the fact that all instruments and recorders had not been accurately calibrated.

This indication can be observed by comparing the depth of the shaded area with the sodium inlet temperature and flow relationship.

Obviously the worst conditions occurred between November 12th and 14th. During that period, sodium flow was at nearly 100 per cent of full flow, whereas sodium inlet temperature was only 70 per cent of design temperature. Roughly there is a spread of 30 per cent.

In the next period (around November 15th) the flow distribution had slightly improved. Here sodium flow was at 74 per cent and sodium inlet temperature was at 63 per cent, with a spread of 11 per cent.

In the next two periods (around November 17th and 19th) the feed water flow distribution definitely appears to have improved. In these two periods, the percentage spread between sodium inlet temperature and flow had decreased to 6 per cent and 4 per cent, respectively.

This plot was of further particular interest and of operating value to us, because from it and a similar plot of previous runs, we learned that the lower temperature boundary was a sensitive and positive indication of an inevitable drop in steam temperature.

Consequently, since we were operating with such a small margin of superheat, this advance indication provided a better control point than the slower and less sensitive steam temperature gauge. By maintaining this lower temperature above a predetermined minimum of about 450° F, we were better able to judge feed water adjustments and control the stability of the heat balance cycle.

Incidentally, the temperature representing saturated steam is 465° F. This temperature level about coincides with the return sodium temperature.

B. START-UP EXPERIENCE

Figure 7 is a magnified graph of the start-up performance. As in the Fig. 6, the uppermost curve is sodium temperature and just below it is the steam temperature curve.

Again the shaded area represents the maximum temperature difference between top and bottom surfaces of the steam generator outlet head. The cross hatched area represents the sodium ΔT. Feed water and sodium flows are indicated below as before.

Observing the feed water curve, you will note that flow increased until steaming occurred in the steam generator. At this point the feed water flow is immediately reduced, because under steaming conditions its unit heat removal capability is substantially increased.

Steam temperature follows inlet sodium temperature as it should.

This figure illustrates quite well the sensitivity of the bottom temperature of the steam generator head to feed water flow and sodium temperature.

When the feed water flow was increased from 9000 lbs per hour to 14,000 lbs per hour, the bottom temperature dropped approximately 100° F from 550° to 450° F. Fortunately, however, as may be noticed on the previous figure, further increases in feed water flow, as necessitated principally by increased sodium flow, did not continue to lower this bottom temperature in the steam generator.

As can be noted on this graph, the lower steam generator temperature rapidly improves with an increase in sodium inlet temperature when it is not accompanied by an increase in feed water. This is the relative effect that is expected to occur as the design conditions of operation are approached.

The maximum load carried on this particular day was only 800 kw. The maximum sodium ΔT was only 120° F, as compared to a design value of 460° F.

C. SCRAM OPERATION

A magnified graph of the first reactor scram operation is shown in Fig. 8. Upon receiving a scram signal, the unit is immediately removed from service and feed water is reduced to an amount sufficient to maintain a constant sodium temperature. Here it can be seen that the sodium outlet temperature was held very well within acceptable limits.

Steam temperature came down in an orderly fashion with sodium inlet temperatures.

During this off-the-line period, however, unavoidable steaming does occur in the steam generator, below the desired minimum flow rate. The steam is bypassed directly to the condenser until normal reactor operation is restored and suitable quality steam is available to operate the turbo-generator.

D. WATER PURIFICATION

The third phase of operation to be discussed rather briefly consists of the problems associated with maintaining the proper degree of water purity that is required by a once-through type all-stainless-steel steam generator.

Water purity, along with the avoidance of extreme thermal shock and low-flow boiling, are important criteria of operation.

The water specifications limit contaminents to the following values: total solids 0.5 ppm, a chloride content of 0.1 ppm, an iron content of 0.01 ppm, and an oxygen content of 0.005 ppm. The specified pH value is 9.6.

In order to comply with these specifications, we have installed conductivity sensing devices and water-sampling connections at several locations. These include the discharge of the condensate pumps, the effluent of the demineralizers, and the feed water line to the steam generator. This latter point is used for control. It is our hope to correlate sample analysis at the control point with conductivity readings to aid in establishing the need and extent of polishing the condensate through the demineralizers.

The oxygen content and pH value are also measured and sampled at this same feed-water control point.

Supplementing the function of the de-aerator, the control of dissolved oxygen is achieved by adding hydrazine (N_2H_4).

The pH value is controlled by adding morpholine (C_4H_9NO). Both of these chemicals are presently being added to the suction side of the boiler feed pumps by means of a common injector. Experience to date, however, has indicated that separate independent injectors will be necessary for proper and economical control.

The demineralizers are so connected that in addition to treating raw water, they can be valved to treat the entire condensate flow or to polish it continuously on a predetermined percentage basis up to a maximum of 30 per cent of the full-load flow.

Figure 9 is a plot of the various factors pertinent to maintaining the specified feed water conditioning that were experienced during the November test run period. The lowest (solid) curve indicates the oxygen content in ppm.

The curve above (alternate dashed and dotted) is conductivity in micromohs. The dashed curve is the residual hydrazine content in ppm. Above that is a dotted curve representing pH value, and finally in bar form are the periods during which the demineralizers were in service polishing a percentage of the condensate.

Unfortunately, we were plagued with air inleakage during this period of operation. This resulted in an erratic oxygen content variation of from 5 ppb to 20 ppb, which naturally complicated our problems. We learned a great deal from this experience, however, and are confident that we will be better able to control oxygen in the future.

The source of the air inleakage was determined to be in the idle boiler feed pump. This condition has been corrected by pressurizing it from the running pump.

In an effort to control the high (0.2 ppm) oxygen content, we were compelled to maintain a residue hydrazine content of 2 ppm. This compares to a normal of about 1/1000 of that amount, as was established to be necessary on the previous runs.

The excess hydrazine broke down into ammonia, causing the pH to rise erratically and abnormally, and causing the conductivity to rise erratically and abnormally. In addition, ammonia is undesirable in the presence of copper-bearing materials throughout the water-steam loop.

Polishing the condensate through the demineralizer removed the ammonia, but of course it also removed the morpholine. This in turn caused the pH and conductivity to drop rapidly. Polishing also removes the hydrazine, all of which tends to be wasteful of these additive chemicals. For the present, we expect to bring this condition under satisfactory control by eliminating excessive air inleakage.

The problem of correlating conductivity to total solids and polishing requirements remains to be solved.

We have established that the demineralizer will produce water with a total solids content considerably less than 0.5 ppm and will reduce other elements to the desired values.

We have only made one total solid measurement of the condensate. That test, which was made shortly after start-up in July, indicated a total solid content of 2.8 ppm. A part of this was doubtless organic compound, but such a value is nevertheless very good considering the short period of operation.

The relation between conductivity and total solids is greatly complicated by the addition of morpholine. Where a sample of demineralized water will have a conductivity of less than 0.1 ppm, the addition of morpholine as required for proper pH control raised this value to about 20 ppm. The difficulty, then, in attempting to detect an increase in total solids varying from a half ppm to, say, one ppm, which would only affect a conductivity reading of 20 by about one point, is quite obvious.

We expect to materially improve this condition by the installation of cation exchangers in the conductive recorder or the condensate and on the feed water lines. The primary function of the cation exchanger is to remove the hydrazine and morpholine and thus secure a more indicative measurement of total solids.

The feed water record will then include an "as is" conductivity measurement, and a cation conductivity measurement. The latter, being free of chemical additives, can be compared relatively with the conductivity recorded on the demineralizer effluent, serving as a base.

The "as is" record, in addition to being a true conductivity measurement of the feed water, will also serve as a better judge of the addition of morpholine for pH control, because of its higher sensitivity to the additive than the pH recorder.

In order to better establish the total solids content, we are considering the installation of an evaporative-type instrument that will indicate total solids over a long period of time (perhaps weeks), on an integrated basis. At least we will know what the total solid values were, and possibly we can use the information to control them better.

Our overall objective in regard to water purity is twofold. First to maintain the purity specified, and second to have a record of what the prevailing conditions actually were during operation.

VI. CONCLUDING REMARKS

It has been a pleasure to present our operating experience with this plant, and we will be most happy to share this, and any other information we may acquire in the future with you.

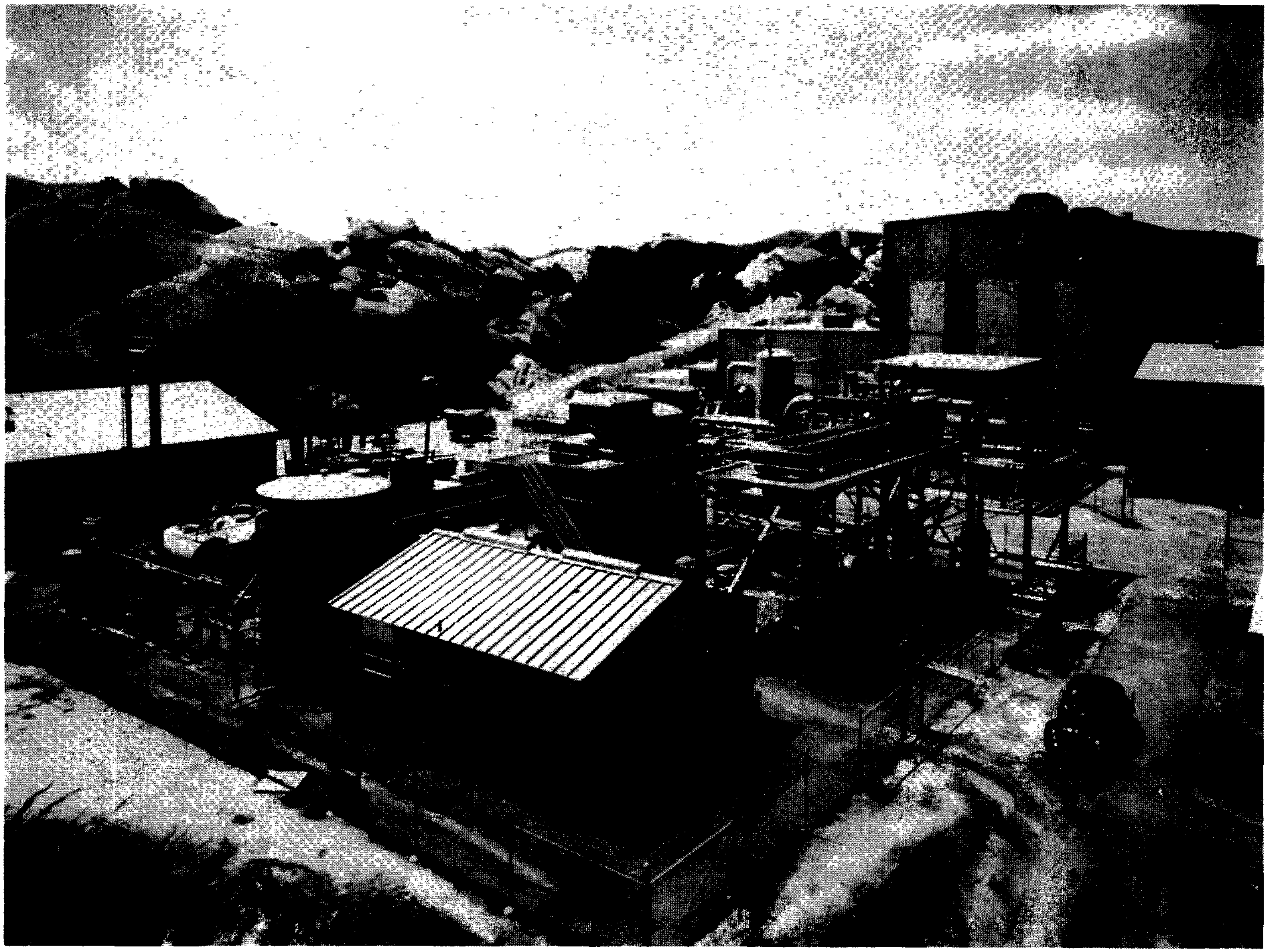


Fig. 1. General View of Edison Steam-Electric Plant

9693-5915 B

Fig. 1. General View of Edison Steam-Electric Plant

9693-5915 B

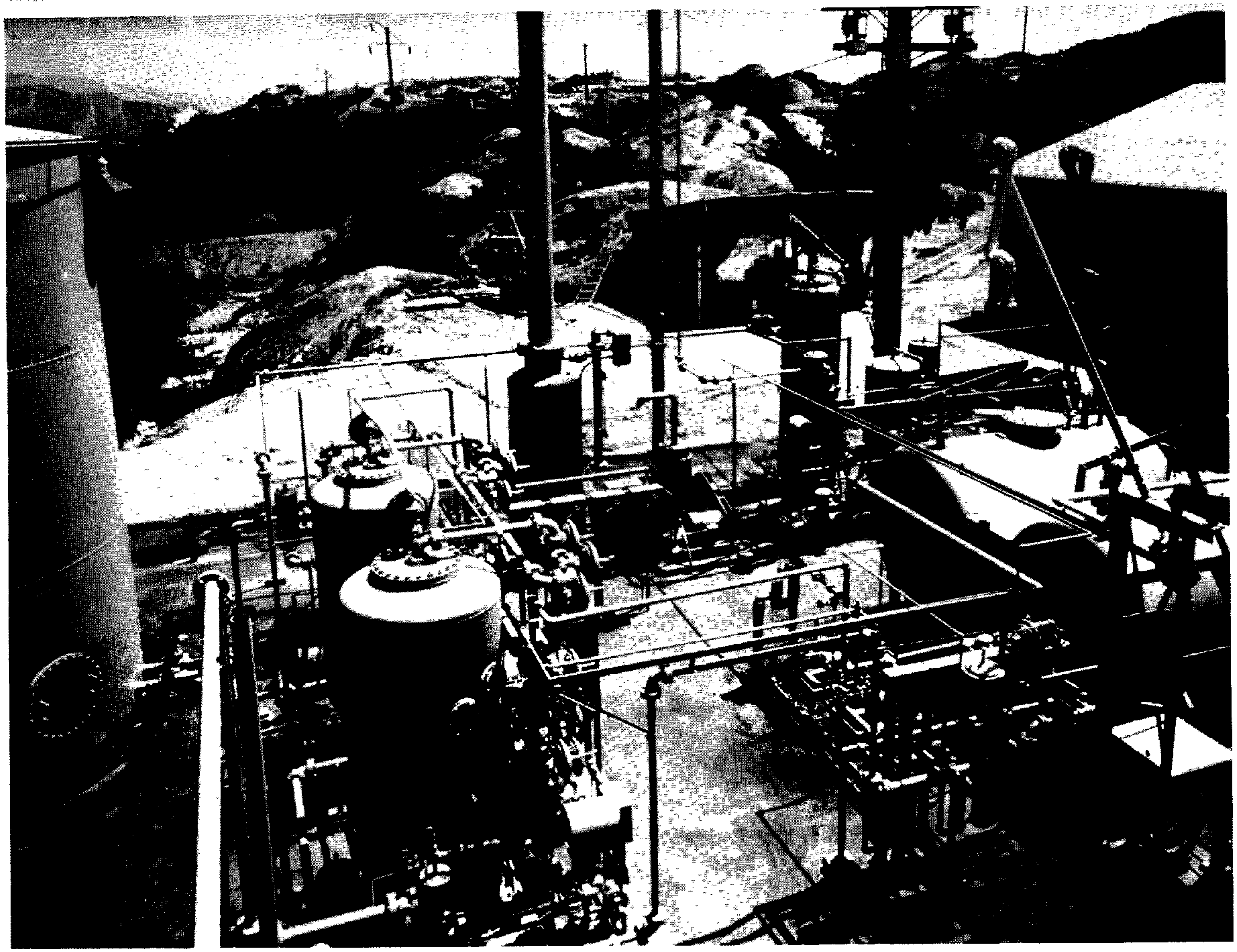


Fig. 2. Feed Water Purification Installation

9693-5915A

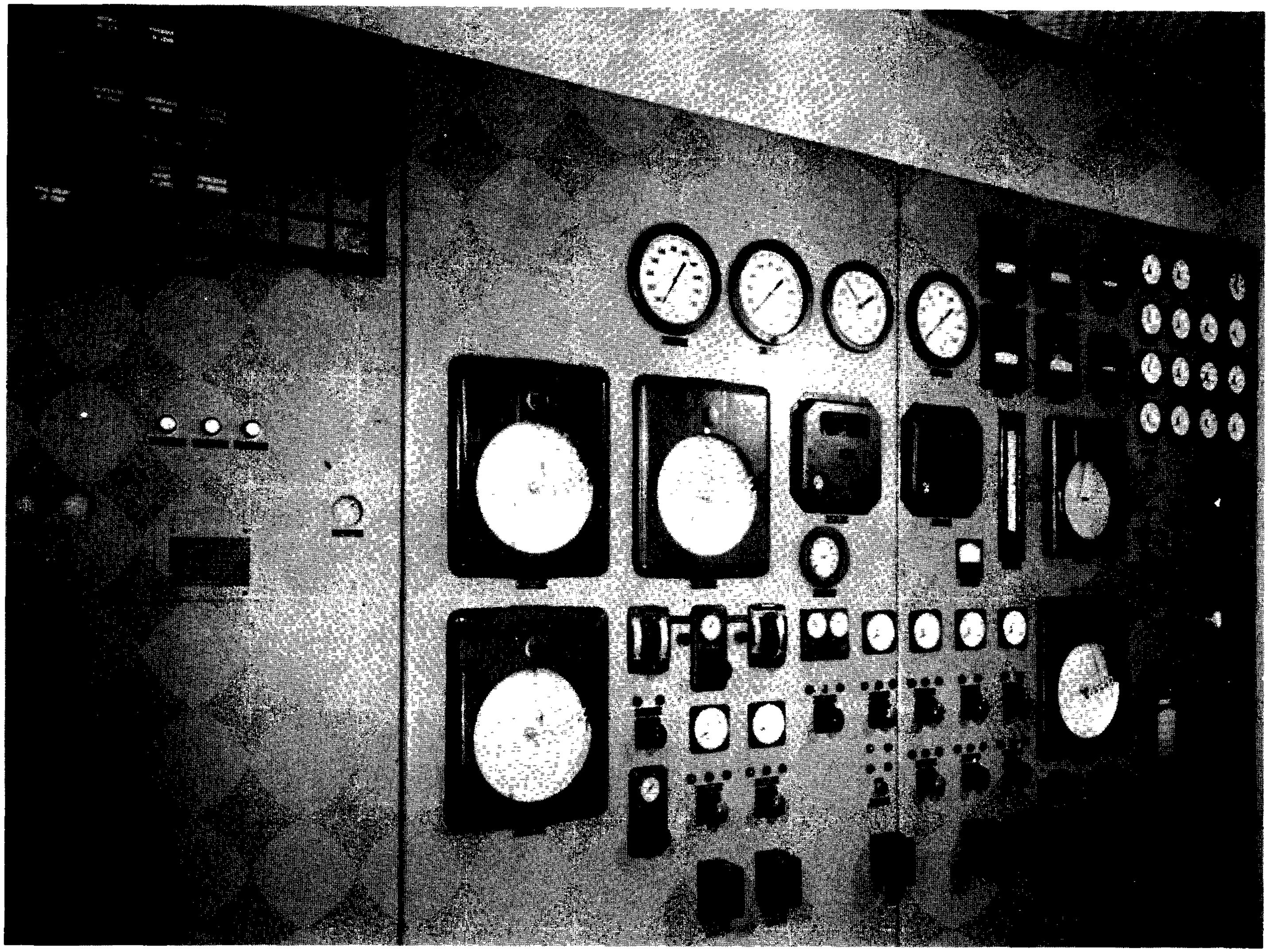


Fig. 3. Edison Control Room

9693-5914

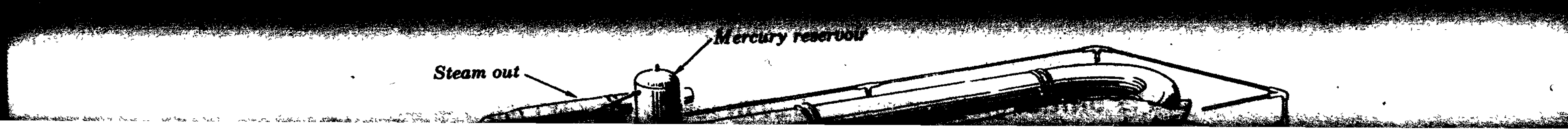


Fig. 3. Edison Control Room

9693-5914

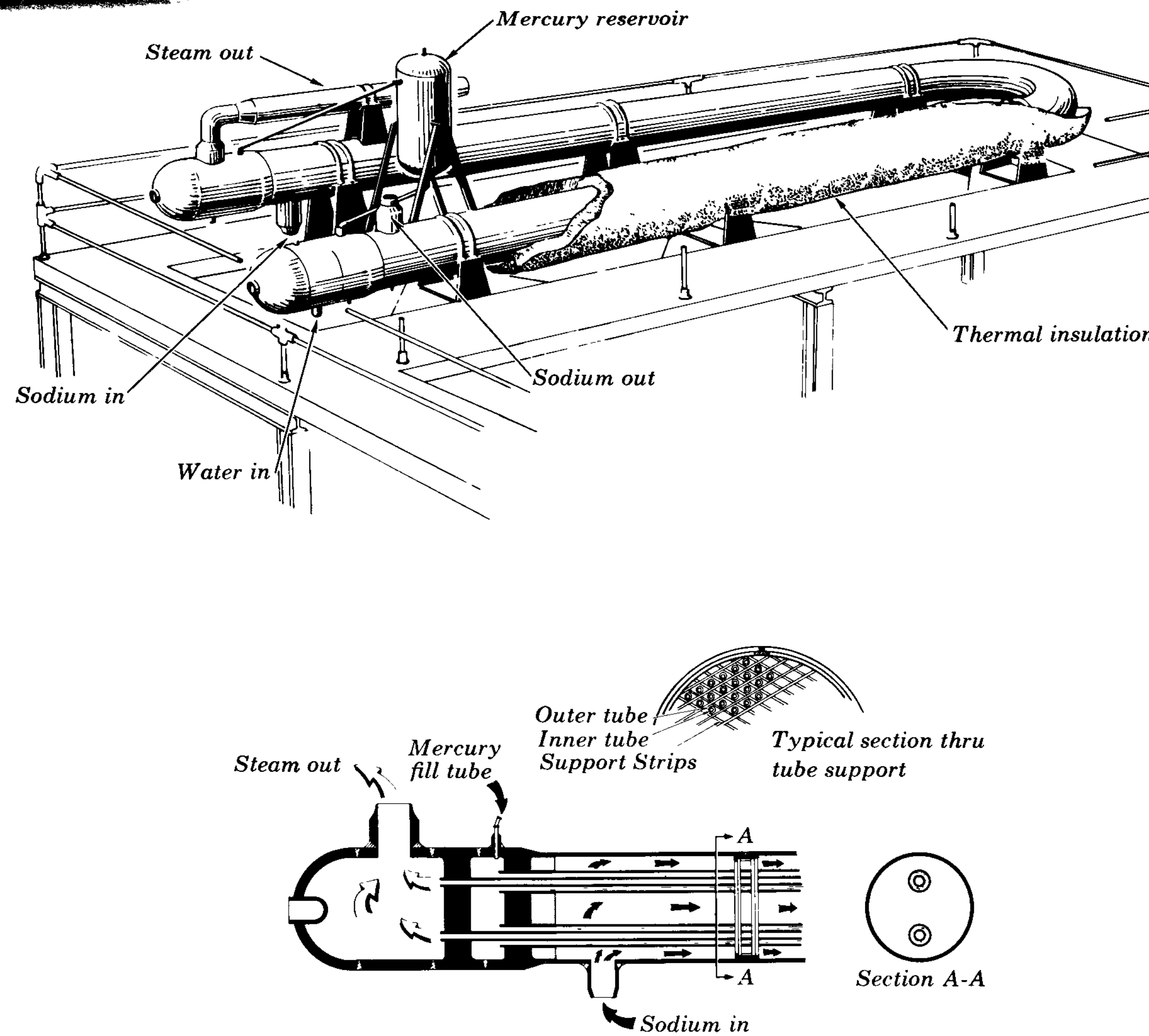


Fig. 4. Cutaway View of Steam Generator

9693-5903 A-B

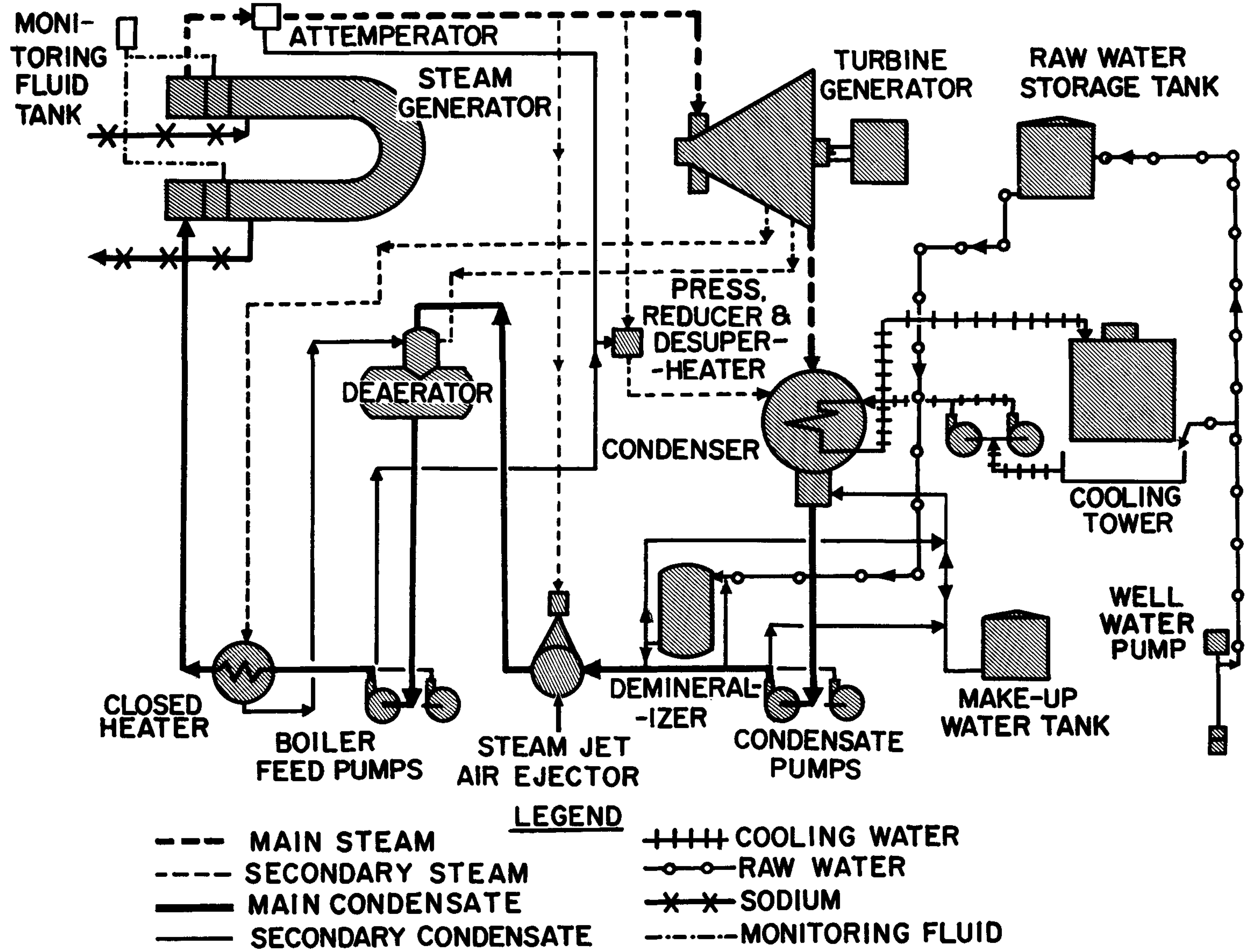


Fig. 5. Basic Flow Diagram of Steam Plant Thermal Cycle

9693-5902

MAIN CONDENSATE
SECONDARY CONDENSATE
SODIUM
MONITORING FLUID

Fig. 5. Basic Flow Diagram of Steam Plant Thermal Cycle

9693-5902

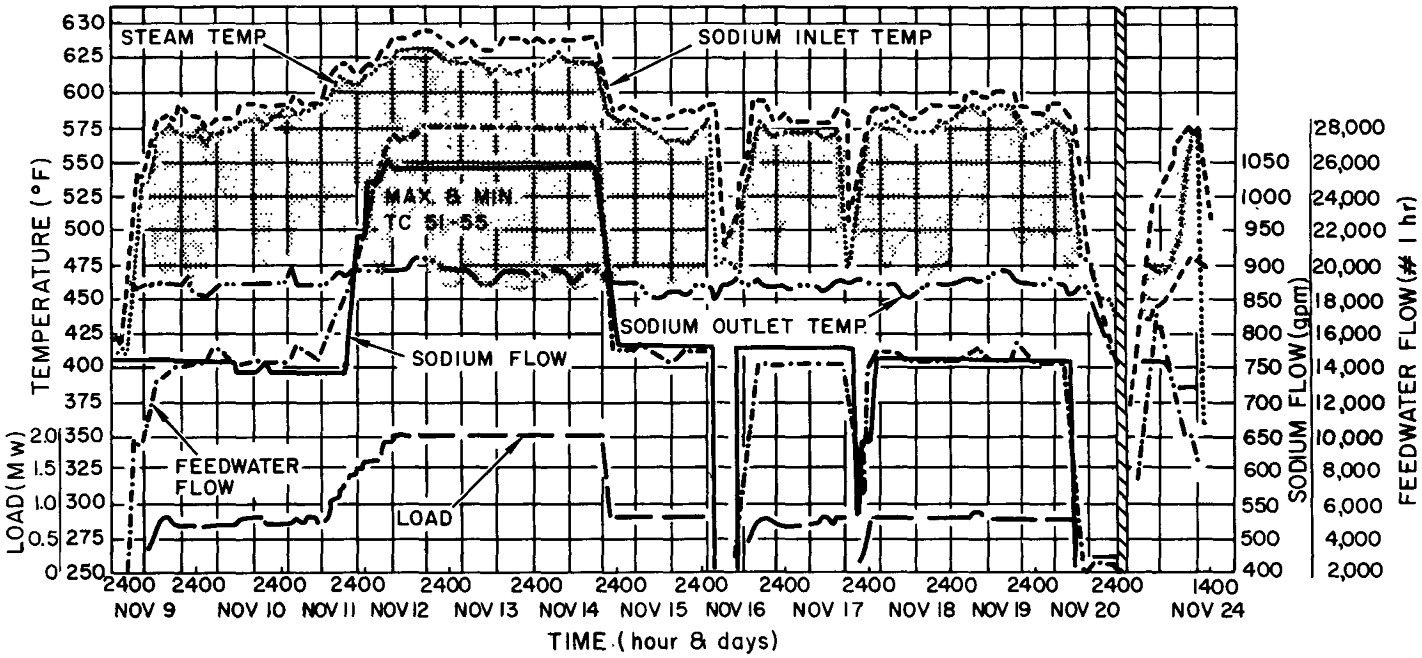


Fig. 6. Operating Record for November 9 through 24

9693-5911 A ⓐ

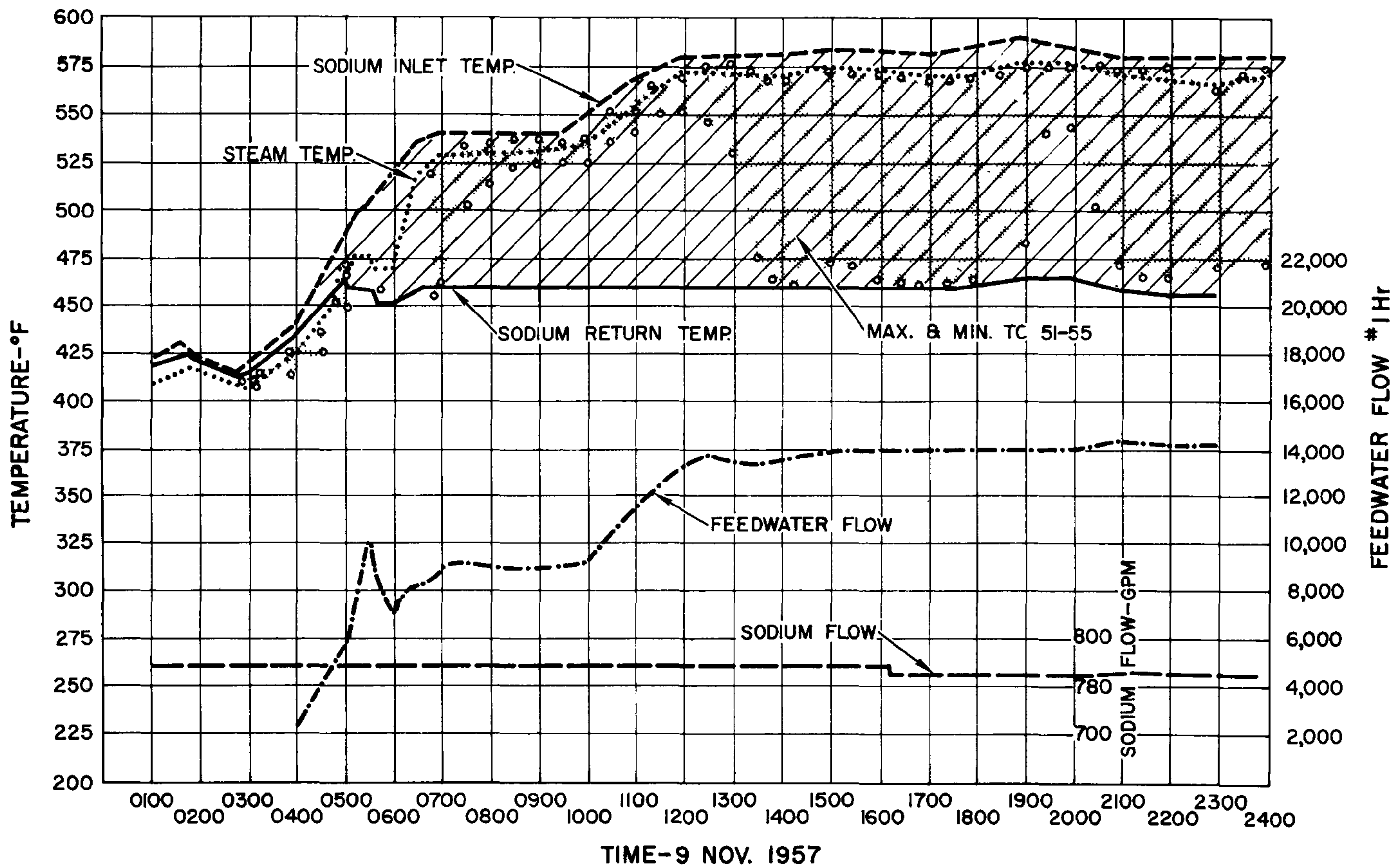


Fig. 7. Start-up Record for November 9

9693-5912 (a)

Fig. 7. Start-up Record for November 9

9693-5912(a)

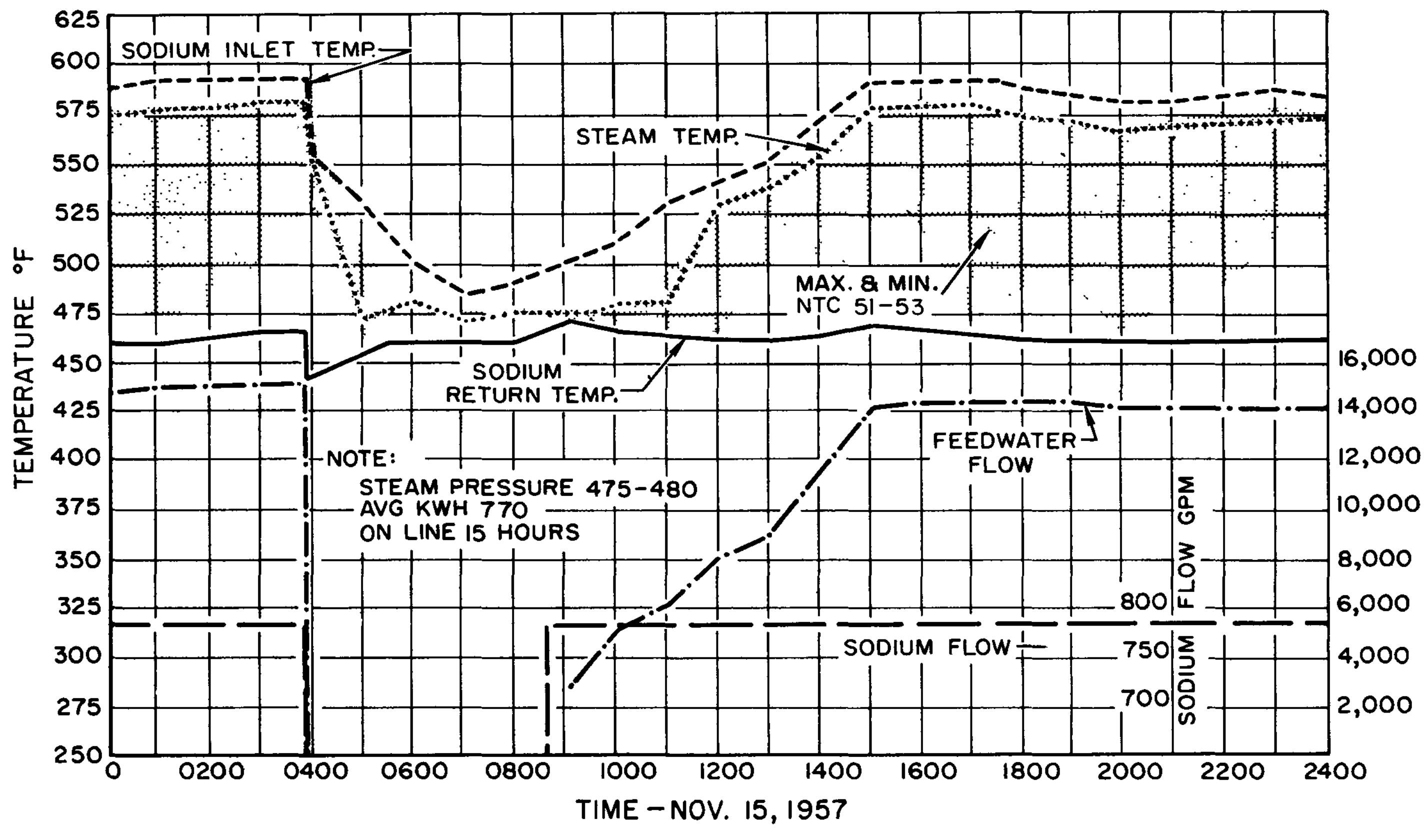


Fig. 8. Scram Record for November 15

9693-5913(a)

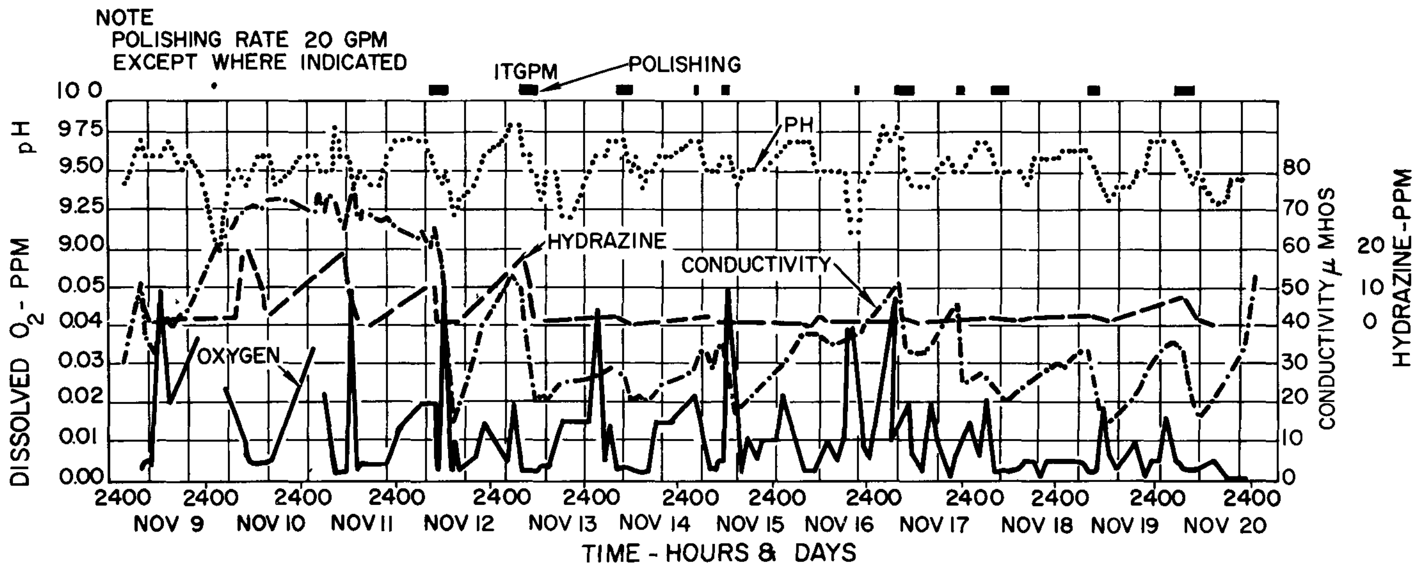


Fig. 9. Feed Water Conditioning Record for November 9 through 20

9693-5911 B@

SGR COMPONENT DEVELOPMENT TECHNOLOGY

R. W. Dickinson*

Specific examples of developments in pumps, valve configurations, and cold traps are noted. Experience with SRE leading to the development of these designs is discussed. Studies of alternate materials of construction for SGR systems are covered, with a brief summary of current results at this and other laboratories; prospects for the use of low-alloy steels are particularly noted. The status of sodium-heated steam generators is reviewed, and preliminary results from the operation of the "once through" steam generator as they affect the heat transfer system are noted. Developments in the method of cleaning sodium-filled cavities are discussed, noting an unusual procedure which has proved satisfactory in preliminary tests. Experiments in progress to determine the necessity for extensive cleaning procedures in sodium system construction are covered. A brief review of experiments under construction or planned for the immediate future is made.

* Chief Project Engineer, Sodium Graphite Reactors

I. INTRODUCTION

Dr. Wienberg, in a recent article* in Nuclear Science and Engineering, compared the philosophies of the homogeneous and heterogeneous reactors. He prefaced his remarks by noting that the homogeneous reactor was really a chemical plant, and therefore the development efforts were aimed at improving the chemical fuel processing. On the other hand, heterogeneous reactors are essentially mechanical engineering devices, and development efforts took the form of improving the heat transfer apparatus. We are in the heterogeneous reactor business, and I am glad to be in such distinguished company regarding our belief that improvement of heat transfer components is essential to sodium graphite reactor development. Sodium system and component development is being undertaken at AI in order to translate our experience in the SRE into lower-cost, more reliable components for advanced SGR systems. I shall touch on those components, exclusive of fuels, which have been recently developed or are now in the process of active test or development. I shall not attempt to describe component developments which are now in the planning stage.

II. DEVELOPMENTAL EFFORT ON COMPONENTS NOW IN USE

A. FREEZE-SEALED PUMPS

During the period of start-up and initial operation of the SRE, component development effort was concentrated on those things which the operating experience demonstrated to be real or potential problems. As an example, the freeze-sealed pumps depend for their operation on an external tetralin cooling system. Although no significant difficulties have been encountered in the operation of these pumps either in the SRE or in prior tests, it appears desirable that a pump configuration which does not depend on operation of a cooling system be obtained if possible. In addition, a pump configuration which would be susceptible to simpler maintenance of bearings, wear rings, and other rotating parts was considered desirable. The first possibility that comes to mind is the "sump type" pump with a liquid-metal-lubricated steady bearing close to the impeller. This type of pump

* A. M. Weinberg, "Molten Fluorides as Power Reactor Fuels," Nuclear Science and Engineering 2 No. 6, 797 (November, 1957)

does, of course, require control of gas pressure over the free surface of the sump, but would be expected to offer potential advantages as far as simplicity of operation is concerned. A large pump of this general concept has been tested at Argonne for the EBR-2. Accordingly, an experimental pump body to this concept was ordered from the Byron Jackson Company. It was desired that this pump body be fitted with a case freeze seal similar to the installed pumps so that it could directly replace the main SRE secondary system pump. The pump impeller will have the same characteristics as the presently installed pumps. The replacement "cartridge" will be tested in a loop prior to installation in the non-radioactive main secondary system. Due to the static height differences in this secondary system, which are not ideal for a sump-type pump with a free surface, close control of gas pressure by an automatic system is necessary.

B. HELIUM GAS SEAL

A difficulty which showed up rather early in SRE operation was the performance of the helium gas seal at reactor floor level. This rotating seal is necessary in order to retain an inert atmosphere over the sodium seal, and would also probably be required in a sump type pump. Excessive use of helium indicated that the seal, regardless of the several modifications made on it, was not adequate for the service to which it was put. This service is retaining helium at pressures of 1 to about 10 psig at essentially ambient temperature, with a shaft rotating at 200-1200 rpm. Investigation of other commercially available shaft seals was undertaken, but none proved completely satisfactory on test. A seal was devised at Atomics International, which used lubricating oil in a "cup and lip seal" with the oil retained by a commercial automotive type oil retainer. The seal is estimated to cost about $5.00, as compared to rather higher prices for other commercially available seals. The seal has successfully operated for several hundred hours with a leakage of about 10cc per hour of helium. Seals of this type are being installed on the SRE.

C. VALVES

Valve problems are well known, and, after investigating many vendors' proposals, a valve which represented a departure from the usual disc and seat arrangement was selected for test. This valve employed, instead of a valve disc, a ball captured in a cage (see Fig. 1). When the cage containing the ball is lowered into the flowing sodium stream, the ball is wedged into a seat by the final lowering of the cage. In the open position, the cage and ball are fully withdrawn from the flowing sodium stream, thus minimizing pressure drop. Stem packing is provided by a frozen sodium shaft seal, externally cooled by flowing tetralin. Two such valves have been ordered for test, in 2- and 12-inch sizes, and will be tested early in 1958.

D. BELLOWS

Bellows seals on small valves have been a problem, as satisfactory bellows are now inordinately expensive; as a consequence, a general investigation of bellows for valves and also other applications was undertaken. It was quickly apparent that the vendors' statements concerning the capabilities of their bellows were, in general, derived from test and service experience extrapolation rather than analytical effort. Therefore, analytical methods were developed here, which indicated that most bellows configurations were operating well past the yield point of the material when subjected to the deflections specified by the manufacturer. Using these analytical methods, various bellows configurations were screened and one configuration has been discovered so far which operates well below the yield point of stainless steel, and holds promise for operating below the yield point even of Zircaloy. The analytical efforts have been supplemented by experimental tests made on aluminum bellows by stress coat and photostress analysis. Aluminum was chosen, as it duplicated to a large degree the properties of Zircaloy at high temperatures. This was considered appropriate, since Zircaloy has rather less strength than stainless steel, and it is desired to procure a Zircaloy bellows suitable for use in improved moderator cans. The experimental and analytical knowledge gained from this type of bellows is, of course, applicable to all bellows operating at high temperature in liquid metal. Zirconium bellows are now being fabricated which will be subjected to life tests under flexure in liquid metal at temperatures of 500° and 1000° F. These temperatures correspond to those expected in the inlet and outlet legs of the SRE. In the improved moderator can application, the bellows is used to absorb differential expansions between process tube and moderator can shell when the reactor core is subjected to fast temperature transients. The bellows is located at the lower end of the central process tube, within the moderator can. These expansions are taken up by flexure of the can head in the installed cans, and it is desirable to reduce this flexure to a minimum.

E. STEAM GENERATOR

The steam generator employed at the Edison Experimental Station has been previously described.* Its performance will be noted by other speakers. The general subject of sodium-heated steam generators and superheaters has been accorded somewhat more than routine attention since the difficulties associated with the Navy Program in this regard became known. The Navy had two alternate steam generator programs in progress or completed, when the decision was made to emphasize water as a coolant for Naval reactors. One of these developments was actually carried through to the construction stage, namely, the pancake-type steam generator and superheaters built by the Combustion Engineering Company. This concept (Fig. 2) is illustrated in the Liquid Metal Handbook,† and is unique in that it dispenses with the heavy tube sheets associated with the conventional shell-and-tube heat exchanger. Connections to the coiled, sodium-containing pancakes are made at the shell of a cylindrical container, and are connected outside this container through stress-relieving piping to cylindrical sodium pipe headers. This promising concept was not tested in the Navy program. Atomics International has obtained a complete set of these boilers which will be delivered in April, and will be mounted in parallel with the existing Edison steam generators at the SRE. Evaluation of this steam generator superheater combination will then proceed. The Navy also had under development a "thimble tube" type steam generator (Fig. 3), proposed and developed by the Griscom Russell Company. This boiler attempts to overcome the thermal stress problem by use of a configuration which requires restraint at one end only. Sodium is passed down the core tube, returns along the outside of the core tube, and steam is generated outside the thimble. Both tubes are free to expand and contract as required. A 3-Mw model of this boiler-superheater combination has been completed,

---

* A. C. Werden, Jr., "The California Edison Company Steam Electric Plant," *Proceedings of the SRE–OMRE Forum* (Los Angeles, November 1956), TID–7525 (NAA–SR–1804), January 15, 1957

† C. B. Jackson, *Liquid Metals Handbook* (Sodium–NaK Supplement), TID–5227, July 1, 1955, p. 283

with AI supervising the later stages of construction. It has been delivered to the MSA Research Corporation for test in a Navy-owned facility there, which has been loaned to the Civilian Reactor Branch for a one-year period, and will be tested under central station type requirements. This boiler, in addition to its unique configuration, is constructed entirely of 2-1/4 chrome, 1 moly steel. Test of this unit will provide information on a fairly large scale of the performance of this structural material in liquid metal service. It should be noted in this connection that both of the above boilers use double-tube construction, the Combustion Engineering unit employing mercury as a leak detector between inner and outer tube walls, and the Griscom Russell unit employing gas such as helium or nitrogen. Preliminary evaluations of both these novel concepts should be forthcoming during 1958.

III. DEVELOPMENTAL WORK ON NEW DESIGNS

With the approval for design and construction of the Consumers Public Power District plant, it was recognized that component technology of the present and immediate future should be utilized in this plant. Accordingly, pump, valve, freeze seal, gas seal, and purification device developments were turned over to the Consumers Project for immediate exploitation. It then became possible for the SGR program to consider systems component developments of longer range which carried the possibility of significant improvements in reliability and cost reduction, but which required more development than that which could be incorporated in the schedule of the Consumers plant. A review was undertaken of the broader problems which had been associated with sodium-cooled reactor plants. It appeared that components and systems have been designed to rather familiar commercial plant concepts, and that the relatively fast temperature transients and excellent heat transfer properties of sodium-cooled reactors had been given later consideration during the design and development, rather than recognizing these qualities as a primary criterion of design and keeping these phenomena in the forefront as concept and design progressed. The temperature gradients which can be induced in massive components when sodium temperature changes abruptly can contribute significantly to performance deterioration unless properly handled during design.

A. SODIUM PUMPS

In system design, sodium valves differ from normal valves only in material of construction, process and workmanship of manufacture, and in stem sealing. As a consequence, we have the usual valve troubles of leaky seats, undesirable pressure drops, and mechanical failures, as well as stem leakage with bellows or freeze-seal failure. The valves are thus both expensive and bulky, as well as not completely reliable. In an effort to avoid this problem, we have established a program for setting out to examine the problems associated with a valveless system, in which isolation of loops when necessary is accomplished by gas pockets or siphon breakers, into which inert gas is deliberately introduced into a vertical gooseneck in the piping. I am certain that this elementary concept contains some "booby traps," and we are mocking up flowing systems to determine how to engineer around these traps. It may be, of course, that the difficulties involved are greater than those encountered in conventional valved systems, but the potential advantages are both obvious and great.

Another area deserving of attention is the accessibility of rotating parts in sodium systems. Removal of a mechanical pump which has been immersed in sodium demands careful attention to maintenance of an inert atmosphere. If the sodium is radioactive, the problem becomes even more complicated. Electromagnetic pumps as currently employed are an expensive and only partial satisfactory solution. A concept developed at AI, which appears to combine some of the advantages of the electromagnetic pump and the mechanical pump, is that of an electromagnetic pump in which the magnetic field is progressed mechanically (see Fig. 4). Sodium is contained in an annulus, and never contacts the rotating equipment, thus permitting all mechanical and electrical parts to be removed for repairs with no attention necessary to either radioactivity or the chemical activity of sodium. A 300-gpm, 10-psi pump of this type is now under construction and should be completed in April. The predicted efficiency is 44 per cent, which, although less than a mechanical pump, is still well within the bounds set by economics.

B. MATERIALS OF CONSTRUCTION

Materials of system construction, cheaper than the 18-8 chrome-nickel now employed for sodium systems, has been actively investigated. Results of our laboratory scale experiments indicate that 2-1/4 chrome, 1 moly steel and 5 chrome-titanium bearing steel are satisfactory from a corrosion standpoint up to at least 1000° F. Both of these materials offer the promise of less expensive systems, both in material cost and in the savings which might be expected due to the lower coefficient of thermal expansion, which results in lower stress and fewer expansion loops.

C. CLEANLINESS AND CORROSION

We are now investigating the necessity for the extreme cleanliness presently required in construction and assembly of sodium reactor systems. Thermal harp circulating loops have been constructed with various degrees of cleanliness ranging from ultra-clean to deliberately dirty, but including the usual boiler shop fabrication methods. These experiments will be run for six months, then opened and examined metallographically. The presence of large foreign particles in the system is, of course, obviously bad, and we are working on filtering schemes concurrently with the foregoing cleanliness experiments. Hopefully, we may be able to relax cleanliness and fabrication requirements to the point where we could build a reactor system in accordance with normal erection practice, flush with sodium to remove large impurities, and then either purify the sodium or dump it and replace with fresh sodium. This scheme would, of course, have to take into account neutron activation of impurities which might be leached out into the sodium. Certainly, there is a great opportunity for cost reduction in this area.

The inert gas used over sodium pools in the SRE is helium. In the interest of reduction in cost of making up the cover gas leakage, and for conserving the relatively scarce helium, nitrogen is being investigated as a cover gas. Experiments to date indicate that nitriding of steel is a problem only at the interface between the sodium and the gas atmosphere, as far as steel is concerned. Work is now under way and will continue on the determination of the severity and consequences of this nitriding. Nitriding of zirconium appears to be quite serious, and the possibility of using nitrogen as a cover gas for systems containing presently-known alloys of zirconium appears rather small.

Along the lines of cleanliness, we have been concerned with removing residual sodium from a drained core tank, from which sodium has been pumped. There will always be a small residue of sodium remaining which the pump cannot pick up. We have completed laboratory and preliminary field tests using a material known as HB-40. This is a hydrogenated terphenyl which has the property of picking sodium up in an emulsion. Sodium thus picked up freezes in small globules and rises to the surface of the mixture. The settled HB-40, now free of sodium, can then be recirculated to pick up more sodium. This method has proved very effective for removing all but the wetted film adhering to metal surfaces. This wetted film is removed with a butyl alcohol flush, and the vessel then cleaned finally with steam.

D. CONTROL MECHANISMS

Control mechanisms which operate within the core are also under active development. Although there is no doubt that the present SRE mechanisms are completely effective, development of other schemes which are potentially cheaper and offer hope of a clear reactor face are now in preliminary test. Development of clear-reactor-face mechanisms would permit refuelling without clearing control drives off the face prior to bringing up the fuel transfer cask, thus saving time in refuelling. A significant advantage would also be realized in providing a method to keep safety rods out of the core but cocked when adding fuel. The reactor could then be loaded steadily, with no danger of achieving excess reactivity. As it is now, criticality must be measured rather frequently during loading, to ensure that excess reactivity will not be accidentally added to the core. Two methods under development are a metal-tape-operated poison column, with positive drive in both "in" and "out" directions, and a pumped liquid poison scheme.

E. HELIUM-3 AS A POISON MATERIAL

In addition, the reactivity effects of He^3 are being measured on water boiler reactor experiments, and its diffusion rate in and out of graphite as a function of temperature and pressure have been measured. These efforts are being made to establish the usefulness of this material as a nuclear poison, prior to developing schemes for its use as a uniformly distributed shim control poison within the graphite moderator.

F. EXPERIMENTAL SODIUM TEST TANK

In order to test out both core and system components in full scale, in flowing sodium and under temperature change conditions expected in a reactor system, a large experimental test setup has been designed and is under construction. The heart of this setup is a tank 35 feet high and 8 feet in diameter which can accommodate full-scale reactor core components, such as thimbles, moderator cans, and fuel elements. This core tank is connected to two flowing loops of 2- and 6-inch capacity, and in addition is connected to two 5000-gallon tanks. Sodium can be heated above or cooled below the temperature existing in the core tank and loops, and can be drained from the top tank through either the core tank or one of the loops, and into the bottom 5000-gallon tank.

G. LEAK DETECTION

The double-wall concept, with a monitoring fluid in the annulus between the walls, has been generally employed for separating liquid metals such as sodium from substances with which they may react chemically, such as water. In the construction of sodium-heated steam generators, this has lead to considerable complication in construction, with attendant difficulties in stress concentrations. These requirements have created a situation in which both component engineering and production are quite costly compared to conventional single-wall heat exchangers. In an effort to arrive at a configuration that would not be as costly as the double-tube and double-tube sheet arrangements now used, but which would still retain the maximum amount of advantages inherent in this system, a novel concept is under investigation. In this configuration, for steam generator applications, a double tube is used for the sodium container, with the water on the outside of the tube. The two tubes are swaged together after grooving and knurling the outside surface of the inner tube. The space thus formed by knurling is filled with a concentrated dye. The tube is then incorporated in an exchanger or boiler in the same manner as a single tube. Inasmuch as almost all failures of sodium-heated steam generators have occurred on the water side, it is expected that pits or cracks originating on this side will result in liberation of the dye to a degree sufficient to color the boiler water which is observable in the boiler gage glass, and thus give indication of an incipient failure before an actual sodium-water contact is made. Experimental work is now proceeding for the purpose of developing dyes which will stand up under the temperature required for long periods of time, and still provide adequate coloring power. Some promising inorganic dyes, suitable for use up to about 850° F, have been discovered to date, and investigation is proceeding in an attempt to find adequate dyes for use at 1000° F. Parenthetically, by use of an inert gas in the grooves, a leak detection method for tubing inside the core might well be devised, with leak indication by mass spectrometer detection of the gases used in the grooves.

IV. CONCLUSION

The component development efforts noted above, combined with new efforts which so far are strictly in the planning stage, are expected to aid significantly in approaching the goal of less-expensive, reliable electric power from sodium graphite reactors.

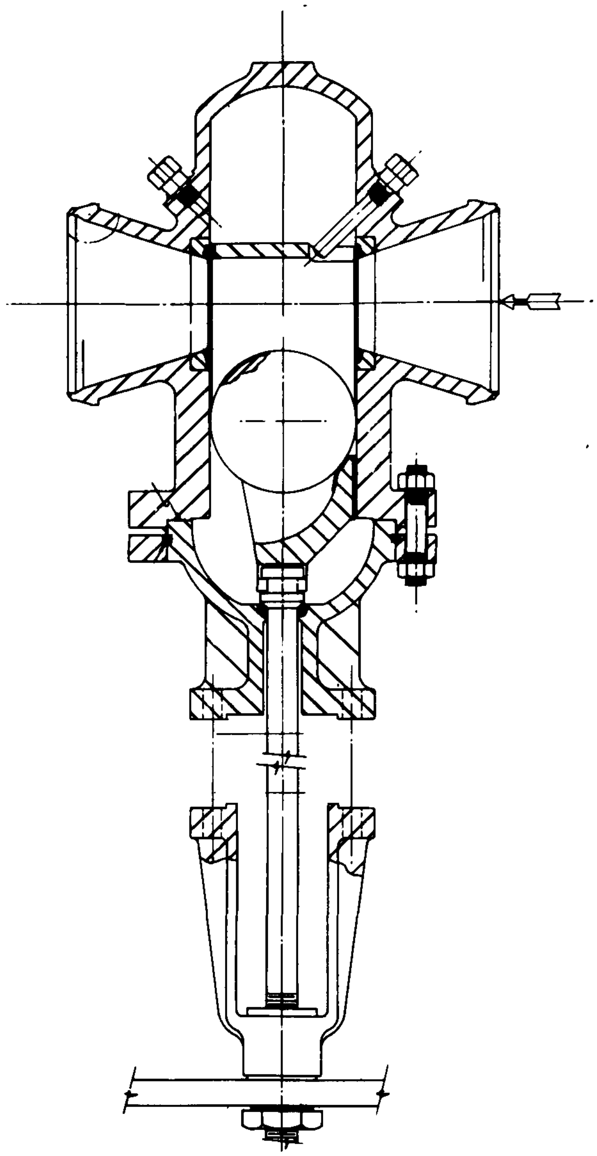
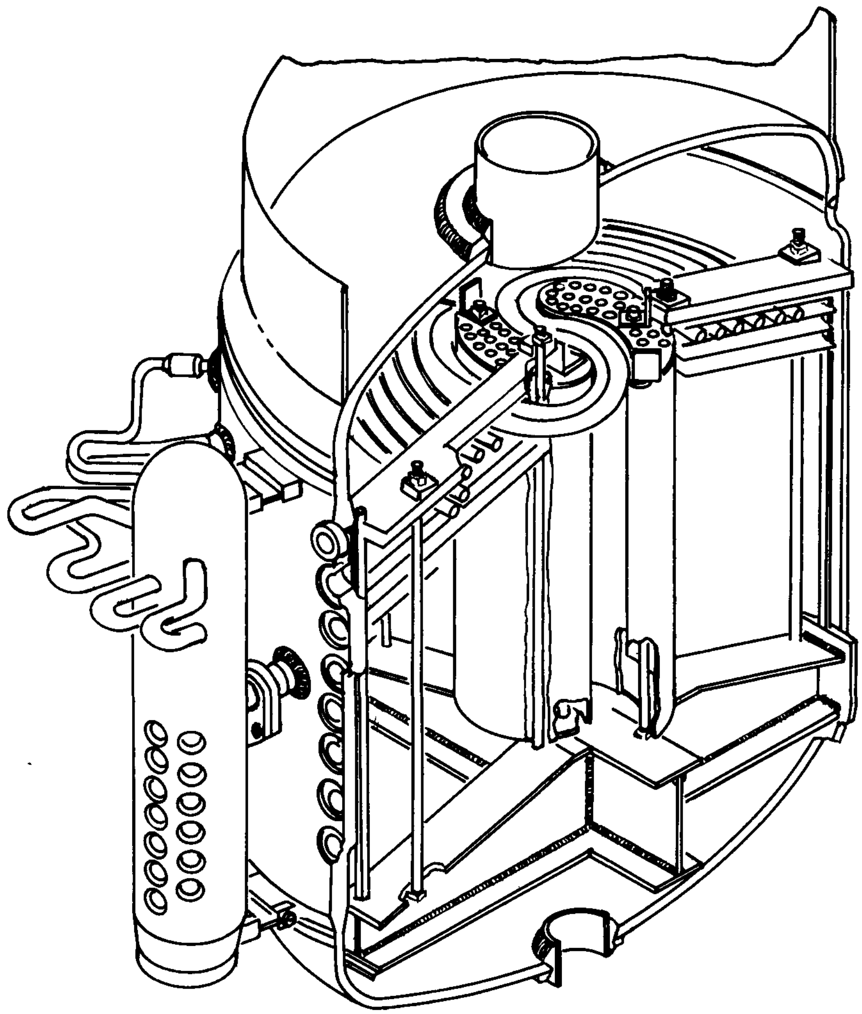


Fig. 1. Ball Type Stop Valve



7504-5406(a)

Fig. 2. Combustion Engineering Steam Generator

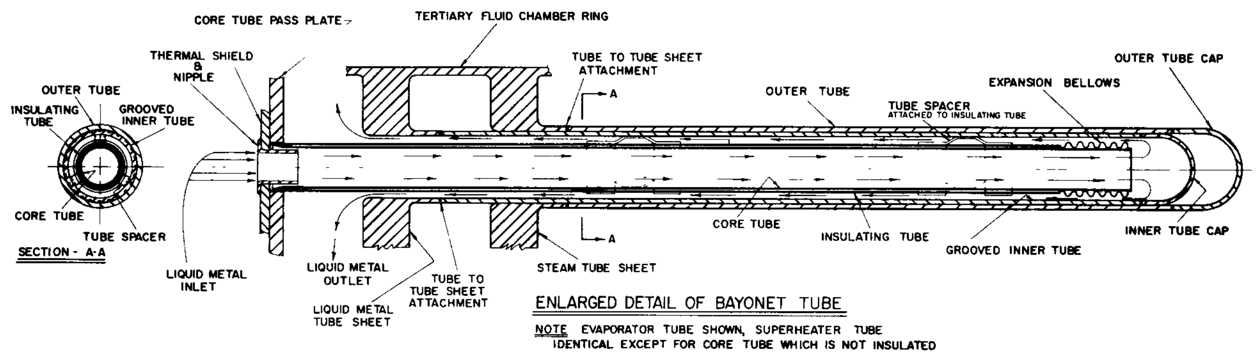
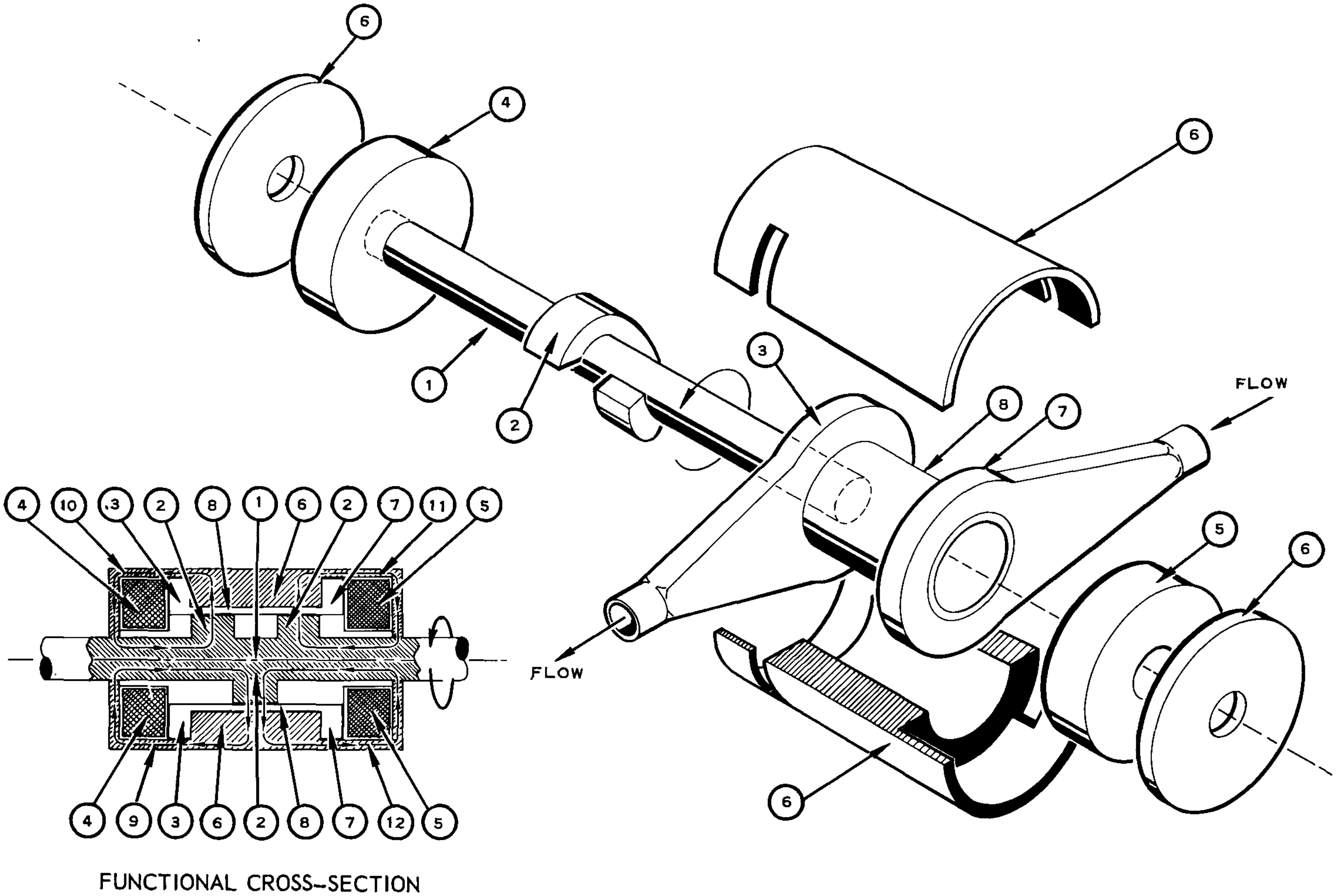


Fig. 3. Griscom-Russell Steam Generator Thimble Tube

7504-5407



7504-5404

Fig. 4. Spiral Rotor Electromagnetic Induction Pump

SODIUM REACTOR MATERIALS

R. L. Carter*

For SRE moderator graphite, the thermal neutron cross section, thermal conductivity from 300° to 700° C, and extent and nature of gas evolution during early reactor operation have been determined, thus permitting prediction of behavior and specification of graphite for sodium graphite reactors.

Studies of zirconium in contact with sodium of controlled degree of purity permit estimation of life of moderator cladding under various possible reactor operating conditions. Of crucial importance is the fatigue life which in turn depends on oxygen in oxide films and grain growth of zirconium and its alloys.

Irradiation tests of dilute uranium alloy fuel material are reported. A Th-7.6%-U^{235} alloy has been selected for the second experimental loading of SRE. To meet the need for still higher temperatures, uranium oxide fuel elements are under development.

* Group Leader, Materials Research

I. INTRODUCTION

The program of supporting research associated with the Sodium Graphite Reactor program may be classified into four general categories. These are, first, the fuel materials investigations; secondly, the investigations upon effects suffered by moderator materials in the reactor core; thirdly, studies on cladding materials. (These are the materials necessary to separate incompatible media in the core.) Lastly, there is general work upon the various minor core materials such as control poisons and high temperature lubricants.

In each of these areas of work the program can be regarded as divided into two parts. The first phase is concerned with the preparatory experiments which, prior to SRE completion, are bent on insuring that conditions met in the sodium reactor experiment are acceptable. These "pre-experiments" ensure that experiments performed in the reactor are useful and operate in a proper manner. The second phase of the experimental work in each program utilizes the SRE itself, or concurrent experiments during SRE operation, to develop sodium-graphite reactor technology.

II. FUEL

The fuel investigation programs are of a new type. They are new because the required conditions for SGR fuel usage are somewhat different than have been met in prior reactor concepts. The requirement of a high coolant temperature, one which runs from about 1000° to 1200° F, requires a higher fuel surface temperature than has previously been met in most reactor operations.

The moderately high cross section of sodium (of the order of one-half barn) requires that only a rather small quantity of this material be present in the core. This requirement can be met by virtue of the excellent thermal transfer capability of this material. A corollary to this condition is that a small fuel element surface area must be used for transfer of heat to the coolant. A further corollary is that a high internal fuel temperature must be tolerated. Specifically, an internal fuel temperature of the order of 1300° F or greater is desirable. The SRE will be the first reactor facility to yield statistically significant data on such usage of fuel.

As was mentioned earlier, the SGR fuel element development program falls into two phases: that of conducting pre-SRE screening experiments and that of evaluating fuel in the SRE. The latter will be carried out both through the medium of full core loadings and through the medium of individual experimental fuel elements.

The initial SRE fuel loading is alpha-rolled, beta-heat-treated uranium containing 2.78 per cent U^{235}. This material was selected for start-up loadings several years ago on the basis of the volume of experience in other reactors with this fuel. It has served for the critical and for the shake-down operation of the sodium reactor experiment. The first developmental loading will be made in the summer of 1958. It will be a full loading with a thorium base alloy containing 7.6 per cent fully enriched uranium. These fuel elements will be extruded and swaged. This material is expected to show improved dimensional stability over the uranium used in the first loading. Certain experimental slugs* will be inserted in all loadings of the SRE. Among the experimental materials will be dilute uranium base alloys with zirconium and with molybdenum. Uranium oxide will certainly be investigated in the immediate future; uranium carbide is being considered for investigation. There also will be uranium-base alloys formed in experimental shapes which are considered to be especially favorable. As an example, there is the hollow slug element described in last year's Forum.

The screening program for experimental elements has been underway in the Materials Testing Reactor for several years. The samples being subjected to screening tests are encapsulated with a thermal bond of sodium-potassium alloy in stainless steel. These are heavily enriched so as to be heated by fission to the required temperature. In the progress of irradiation at MTR, they suffer irradiation damage at an accelerated rate. Materials already subjected to such test include beta-heat-treated uranium, uranium base zirconium alloy, uranium base molybdenum alloy, and thorium base 10 per cent uranium-235 alloy. Measurements are to be made on changes of diameter and density, on the extent of burn-up, and on metallographic changes experienced by the fuel material. To date, tentative conclusions have been based on post-irradiation measurements of changes in density and diameter. These are backed up by a fairly large volume of data now available from work done at Bettis Field, Argonne National Laboratories, and in England.

* C. C. Woolsey, "Experimental Fuel Materials," *Proceedings of the SRE–OMRE Forum* (Los Angeles, November 1956), TID–7525 (NAA–SR–1804). January 15, 1957

In Fig. 1, a selected example of the changes which are experienced by a fuel element under irradiation is shown. In many tests, more complicated effects were observed. We observe that a change in density has been experienced by the element and we see in the table data illustrative of typical experience with two of the alloys tested.

Specimen	Burnup, Mwd/T	Temp, °F	Volume Change, %
U-1.2 w/o Mo	2000	1250	+22
Th-10.0 w/o U^{235}	3500	1230	+ 4.5

It is noted that experience with the thorium base alloy appears considerably more favorable than experience with the uranium base alloy. This conclusion must be regarded as tentative, since it is based upon a statistically small number of samples exposed. Generally speaking, the uranium base alloys have been unsatisfactory for the conditions of interest in the sodium graphite type reactor; that is, temperatures over 1000° F and burn-ups well in excess of 200 Mwd/T. Statistically significant testing of the thorium base alloy will be accomplished through the thorium-uranium loading of the Sodium Reactor Experiment to begin in the summer of 1958.

Exploratory work is proceeding in several directions. In the order of increasing speculativeness, these include:

a) Uranium oxide, the present gleaming hope because of its seeming ability to accommodate displaced fission gas atoms in open lattice structure.

b) Uranium-based, heavily-alloyed material containing 15 to 20 per cent molybdenum.

c) Uranium carbide

d) Cermet fuel suspensions

e) Constraining cladding or jacketing

A remote possibility which is being investigated is the introduction of fission gas nucleation centers which will lead the fission-product gas atoms to "hang-up" at centers in the strong crystals of fuel material rather than agglomerating at the grain boundaries. Such grain boundary agglomeration forces separation at the grain boundaries, causing a gross expansion in the fuel material.

III. MODERATOR GRAPHITE

Only nominal attention is being given to moderator graphite problems, since there is reason to believe that little difficulty will be encountered with SRE moderator graphite. Rather extensive irradiation damage experience that has been accumulated within the AEC leads one to expect little change in graphite as a consequence of irradiation in the SRE temperature range of from 500° to 1300° F. Radiation damage monitoring is being made, with expectation of finding a change of the order of 1 to 4 in the thermal conductivity and a change of less than 0.1 per cent in linear dimension. Negligible changes are anticipated in the other properties of the graphite so exposed. Specimens of graphite will be removed from the core at about 3000 Mwd/T and 15,000 Mwd/T integrated fuel burn-up levels. These samples have been pre-measured so that small changes can be observed.

The extent to which radiation-induced degassing of graphite takes place is important because of the proposed sealing of moderator cans in future SGR designs. Also of concern is the extent to which canning material strength might be impaired by active metal gettering of exuded gases. The graphite used in SRE is a specially-prepared low-gas-content graphite. It is believed that little gas will be evolved from this under the influence of radiation, although no prior experimental work has been done under the precise conditions which will be experienced in this case. There have been included in the Sodium Reactor Experiment special vented moderator cans with facilities to permit the measurement of the degree of pressure buildup. There are also gas-sampling cans through which gas is continuously circulated, permitting sampling of the evolved gas. Results obtained to date (fuel burnup of 50 to 75 Mwd/T) have shown only the normally expected thermal evolution of gas from the graphite.

IV. MODERATOR CLADDING

We move now to the subject of moderator cladding. The zirconium used on the graphite in the SRE was chosen several years ago. It was chosen in consideration of data then available on its oxidation characteristics, on its low thermal-neutron cross section, and on its strength at high temperatures. Subsequent work has indicated that zirconium is not the ultimate cladding material for use in high-temperature sodium-cooled reactors. In addition to problems of attack by oxygen impurity in the sodium, the difficulty with germinative grain growth shown in section in Fig. 2 was uncovered. This phenomenon occurs when slightly strained zirconium is held at temperatures in excess of 1000° F for times of the order of several months.

Research work now under way utilizes facilities such as that shown in Fig. 3. This is a pumped sodium loop (with thermal insulation removed) in which the sodium is treated by means of a hot trap and a cold trap in the desired experimental sequence. Samples are exposed in the large facility on the right of the figure. Work done with this loop has given confidence to predictions of the gettering capability of materials used in the hot trap. A section of an experimental hot trap is shown in Fig. 4. The trap filler has been exposed by removal of a thimble-shaped section which directs sodium flowing from the inner tube back through the channels between the corrugated layers of getter material. A device similar to this has been included in the SRE and will be used if needed to purge the last traces of oxygen from the process sodium.

The extent to which materials degradation may have occurred in the Sodium Reactor Experiment will be determined by the withdrawal of samples from the Materials Evaluation Facility which has been incorporated into the primary process sodium loop. New can designs free of fatigue concentration points will diminish concern in advanced SGR planning with the changes in notch sensitivity which may occur in zirconium as a consequence of grain growth or oxidation. Investigations are underway to select improved cladding materials for use in replacement elements and in subsequent sodium-cooled reactor designs. Among these are the austinetic stainless steels which are made acceptable nuclear-wise by use of very thin sheet, and which have shown tendency toward carburization only if contact between sodium and graphite occurs. Data on the extent and degree of the carburization as a function of temperature has shown no observable carburization at temperatures below about 1150° F. However, extrapolations suggest that this effect might become serious for prolonged times at temperatures as low as about 1050° F.

An excellent prospect as a cladding material for moderator as well as fuel is zircaloy-2. Zircaloy-2 is a 1.5 per cent tin, 0.15 per cent iron, 0.10 per cent copper, and 0.05 per cent nickel containing alloy. It exhibited an oxidation rate for identical conditions of immersion in sodium about a factor of five below that for unalloyed zirconium. Its neutron cross-section is comparable with that for zirconium but it is precipitation stabilized against recrystallization. No evidence has been found for germinative grain growth in zircaloy-2. Service test of this alloy will be made in the Sodium Reactor Experiment in the near future as the material for construction of thimbles for control and safety elements. Optimized zirconium-based alloys showing yet higher resistance to sodium-borne contaminants and more favorable high temperature strength are being sought.

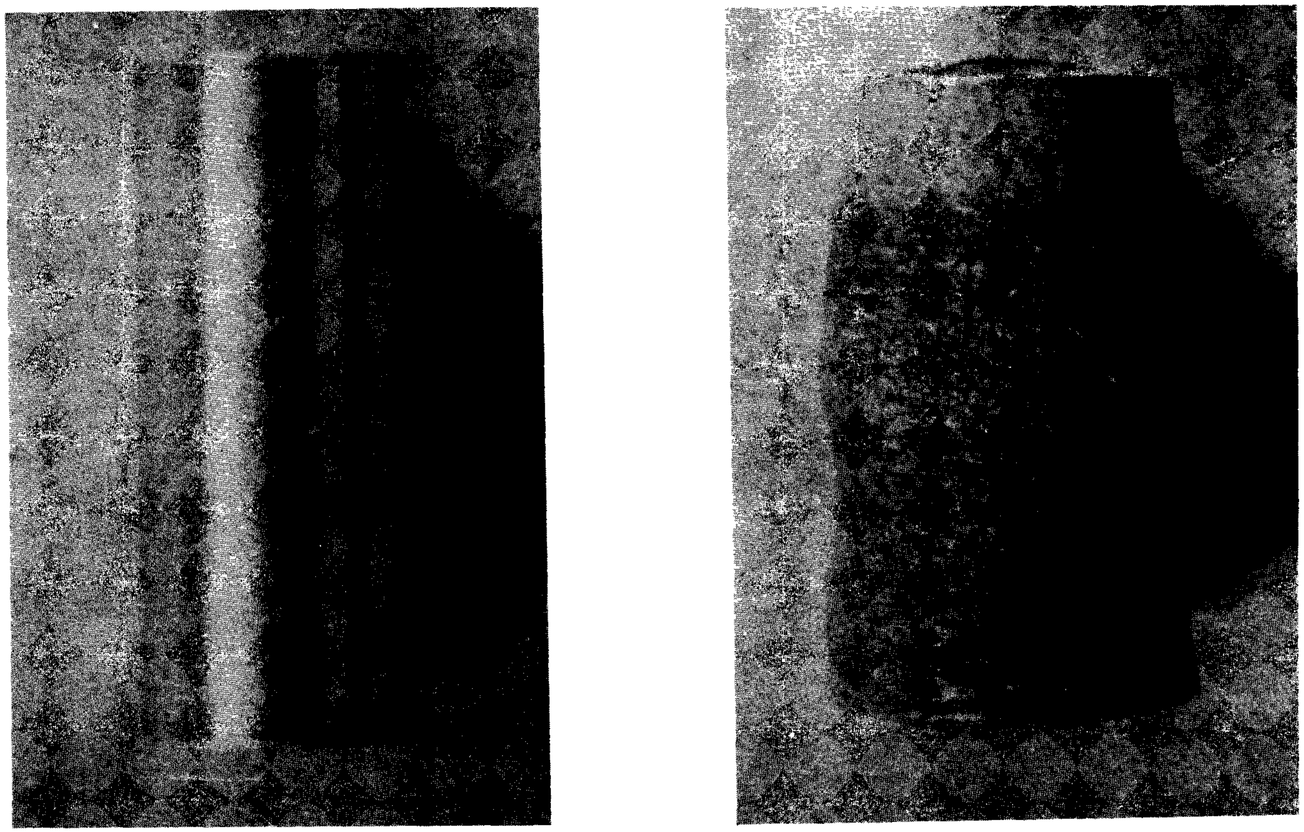
V. MISCELLANEOUS CORE MATERIALS

Among the other core materials which have received attention are the safety and control poison materials. Those in present use are boron-containing and exhibit a fission gas bloating not dissimilar to that found in fuel materials held at high temperatures. A survey of alternate poison materials has been completed and, as a consequence, certain experimental programs have been initiated. Among the earliest to be implemented will be one in which a boron carbide powder is enclosed in a double sleeve. The annulus will be vented to allow escape of gases generated in the boron carbide. This experiment will go into the SRE sometime in the summer of 1958. Also planned for insertion into SRE is a "Thyrex" experiment in which several toroid-shaped rare-oxide specimens will be tested for stability under irradiation. This is planned in connection with the Sheldon reactor development program.

Another phase of the control problem has been of finding adequate high-temperature bearing surfaces. The control rod drives used in SRE were pictured in previous papers with flush-mounted drive motors. These would operate through the use of screw and nut arrangements below the top shield in the high temperature region of the reactor core. Because of difficulties encountered in securing materials which would successfully endure wear at temperatures of the order of 1200° F, the present control rod drives are exterior to the reactor shield. The control mechanisms are supported by means of rods which hang into the reactor. Recent development work has indicated that graphite nuts can be used on steel screws at core temperature if molybdenum disulfide lubrication is used. Tests on using this principle are to be placed in the SRE in the future.

VI. SUMMARY

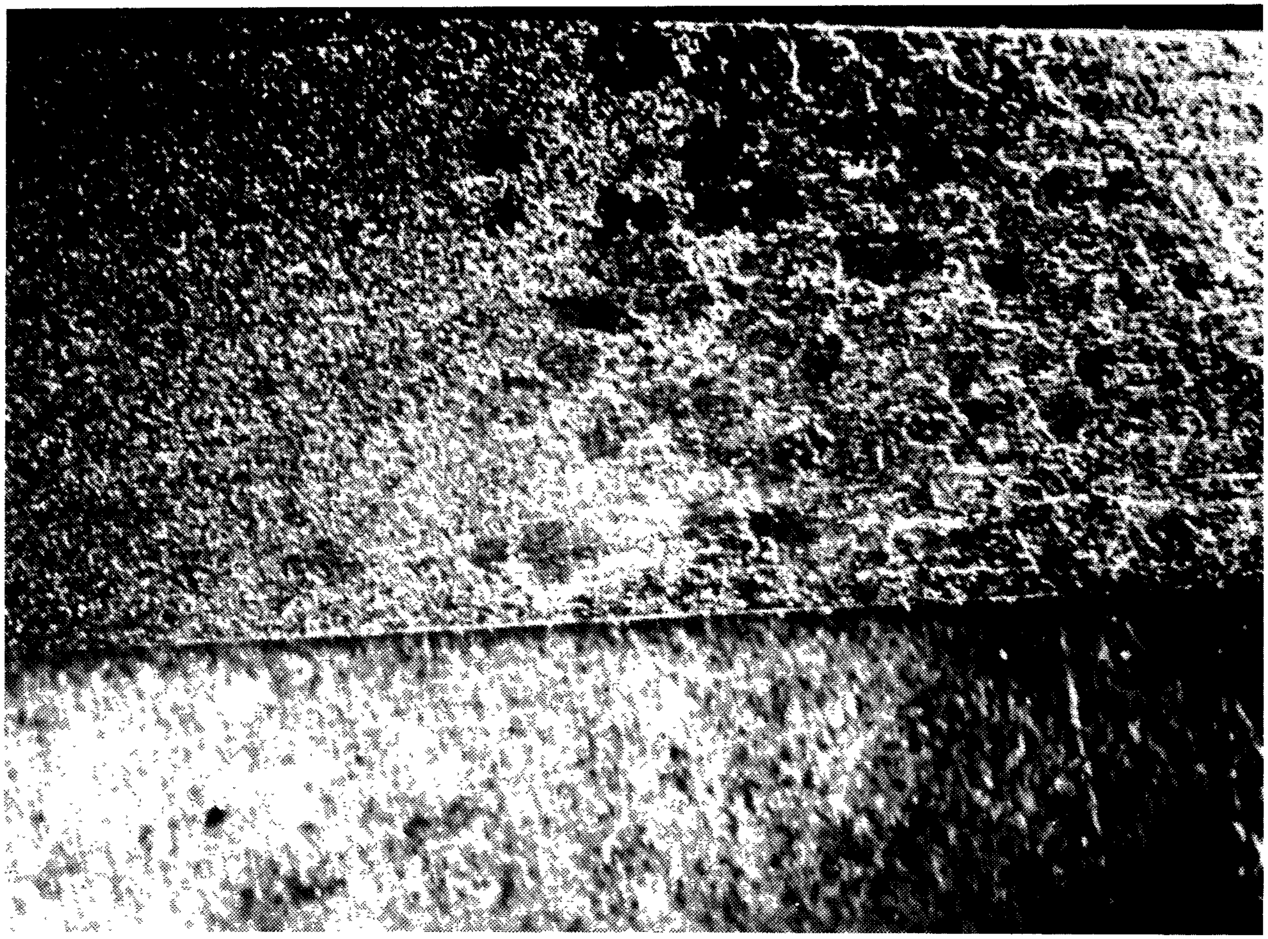
We have discussed the results obtained and the plan of work followed in investigating fuel, moderator, cladding, and miscellaneous core materials. We have discussed these with regard to the pre-SRE experiments and with regard to those experiments which are being done in the Sodium Reactor Experiment and simultaneously in other facilities.



BEFORE AFTER

Fig. 1. Typical Fuel Element Behavior

9693-4751 (a)



The region of exaggerated grain growth in zirconium strained at 500° F and annealed for 287 hours at 1050° F. The critical strain for this specimen is 5 per cent elongation.

Fig. 2. Germinative Grain Growth in Zirconium

9693-4752 (a)

Fig. 2. Germinative Grain Growth in Zirconium 9693-4752 (a)

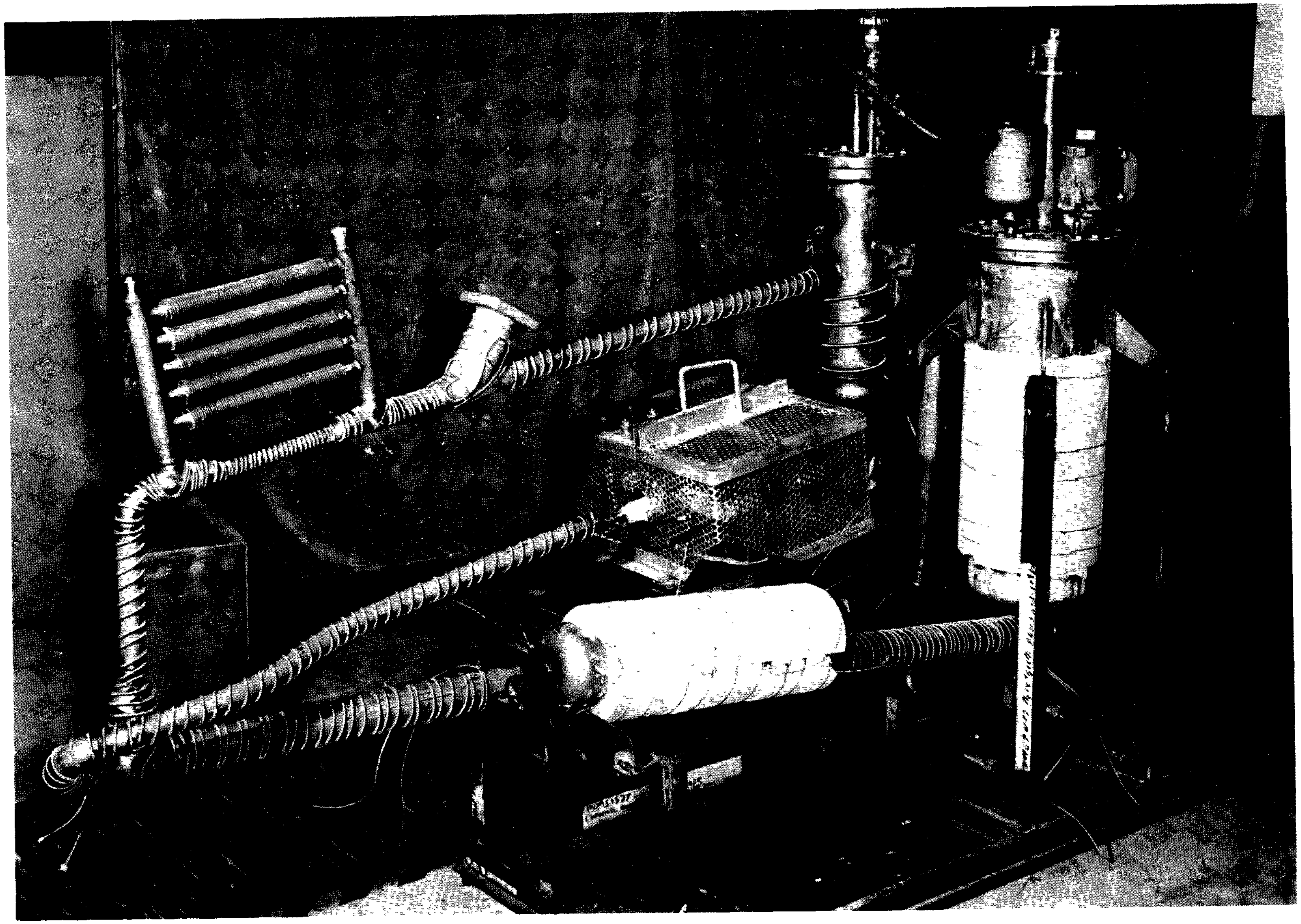


Fig. 3. Experimental Sodium Loop 9693-54254

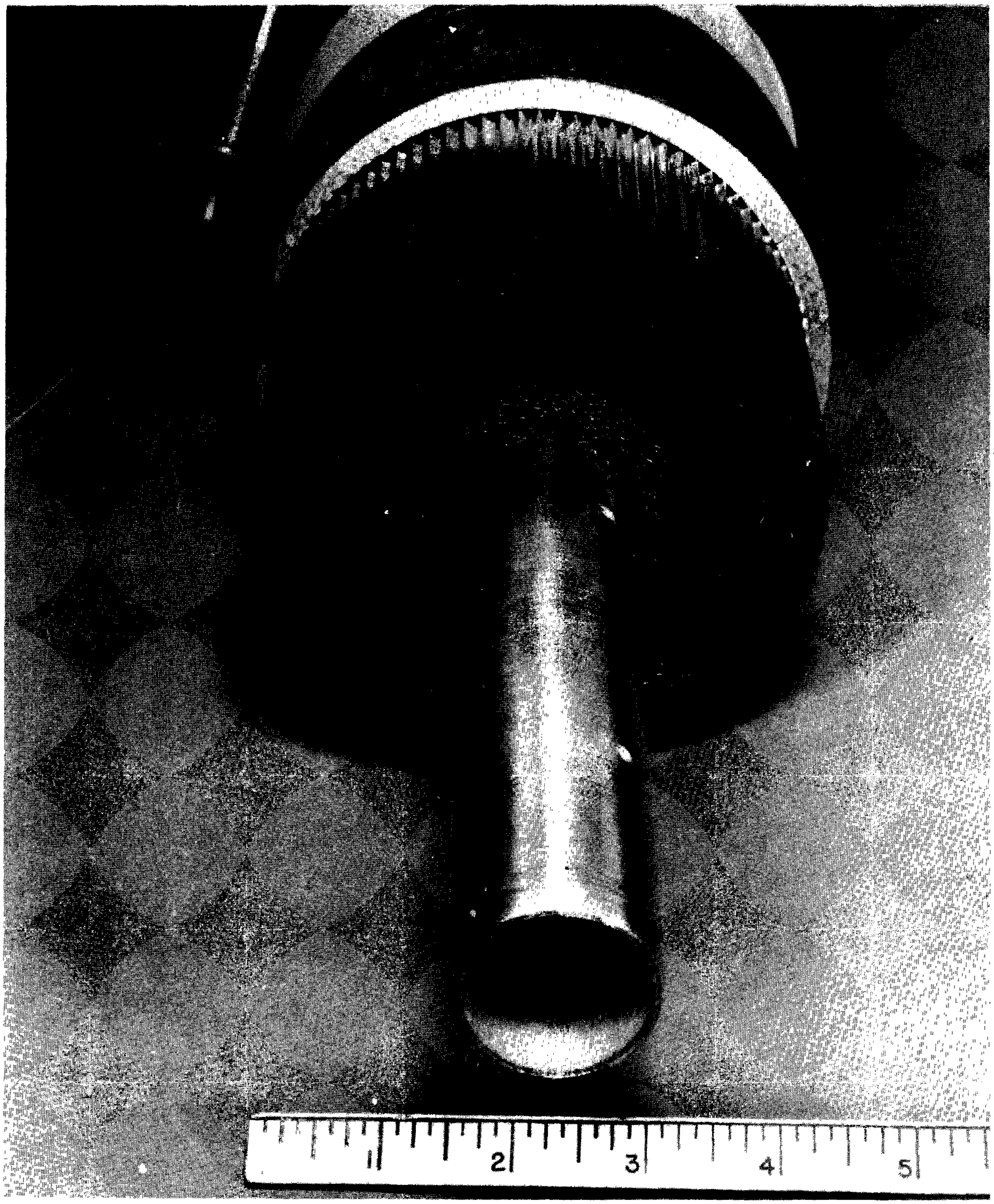


Fig. 4. Section of Experimental Hot Trap

9693-54477 A