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PREFACE

More than 2100 papers were submitted by the nations, the specialized agencies, and the International Atomic Energy Agency, which participated in the Second United Nations International Conference on the Peaceful Uses of Atomic Energy. The number of papers was thus about twice that involved in the First Conference. Provision was therefore made to hold five concurrent technical sessions in comparison with the three that were held in 1955. Even so, the percentage of orally presented papers was less in 1958 than in 1955.

In arranging the programme, the Conference Secretariat aimed at achieving a balance, allowing adequate time for presentation of as many papers as possible and, nevertheless, leaving time for discussion of the data presented. Three afternoons were left free of programme activities so that informal meetings and discussions among smaller groups could be arranged. No records of these informal meetings were made.

A scientific editorial team assembled by the United Nations checked and edited all of the material included in these volumes. This team consisted of: Mr. John H. Martens, Miss L. Ourom, Dr. Walter M. Barss, Dr. Lewis G. Bassett, Mr. K. R. E. Smith, Martha Gerrard, Mr. F. Hudswell, Betty Guttman, Dr. John H. Pomeroy, Mr. W. B. Woollen, Dr. K. S. Singwi, Mr. T. E. F. Carr, Dr. A. C. Kolb, Dr. A. H. S. Matterson, Mr. S. Peter Welgos, Dr. I. D. Rojanski, Dr. David Finkelstein, Dr. Cavid Ergensoy (Dr. Ergensoy's services were furnished through the courtesy of the International Atomic Energy Agency), Dr. Vera J. Peterson, Dr. Paul S. Henshaw, Dr. Hywell G. Jones, Dr. Alvin Glassner and Mr. J. W. Greenwood.

The speedy publication of such a vast bulk of literature obviously presents considerable problems. The efforts of the editors have therefore been primarily directed towards scientific accuracy. Editing for style has of necessity been kept to a minimum, and this should be noted particularly in connection with the English translations of certain papers from French, Russian and Spanish.

The Governments of the Union of Soviet Socialist Republics and of Czechoslovakia provided English translations of the papers submitted by them. Similarly, the Government of Canada provided French-language versions of the Canadian papers selected for the French edition. Such assistance from Governments has helped greatly to speed publication.

The task of printing this very large collection of scientific information has been shared by printers in Canada, France, Switzerland, the United Kingdom and the United States of America.

The complete Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy are published in a 33-volume English-language edition as follows:

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1	Progress in Atomic Energy	1, 2, 23a, 23b, 23c
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3	Processing of Raw Materials	E-10, E-6 and E-7a
4	Production of Nuclear Materials and Isotopes	E-11, E-12, C-14, C-15
5	Properties of Reactor Materials	E-14, E-15
6	Basic Metallurgy and Fabrication of Fuels	E-13, E-17, E-18
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Session B-9

POWER REACTORS

LIST OF PAPERS

LIST OF PAPERS

(Continued)

Liquid Metal Cooled Reactors

Homogeneous Reactors

Operating Experience with the Sodium Reactor Experiment

By F. E. Faris, L. E. Glasgow, D. H. Johnson, R. W. Campbell, J. E. Owens and G. E. Deegan*†

The Sodium Reactor Experiment (SRE) was designed, built and is being operated by the Atomics International Division of North American Aviation, Inc., as part of a program with the United States Atomic Energy Commission for the development of the sodium graphite approach to economic nuclear power. Photographs of the completed plant, the loading face, and the control room console are given in Figs. 1, 2 and 3 respectively. Descriptions of the plant have been given elsewhere,[1,2,3] and will not be repeated here. The sodium graphite type reactor is particularly attractive because of its inherent safety and the efficiency which its high operating temperature makes possible. These features are associated with the chemical compatibility of the materials used in the core and the heat transfer systems and the high boiling point (1614°F at atmospheric pressure) of sodium, which permits a low pressure heat transfer system even at temperatures of interest in modern steam plant technology.

The SRE serves both as an experiment to determine the limits of performance of the original design concept and as a flexible developmental facility, which is used for a variety of tests and experiments and which is being continually modified to improve performance. The approach to full power operation of the reactor has purposely proceeded relatively slowly to provide ample time for a variety of tests to be performed and analyzed prior to the operation of the plant at the limits of which it is ultimately capable. This philosophy of operation is consistent with the concept of the SRE as a high temperature reactor experiment, from which the information needed for the design of a full-scale sodium graphite reactor is being obtained.

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OPERATING HISTORY

The construction of the SRE extended from December 1954, when site development was initiated, to March 1957, when responsibility for the plant was assigned to the operating personnel. In order to gain familiarity with the plant, operating personnel assisted in the supervision of construction.

Pre-operational testing of the components and systems was initiated by the same personnel prior to the end of the construction phase as the installation of each component and system was completed. Following functional tests on the overall plant, a "dry" (no sodium in core) subcritical loading of the reactor was carried out during the period from 16 March to 23 March 1957. The extrapolated critical mass at room temperature with the core dry was found to be 22 fuel elements. Subsequent to the dry subcritical loading sodium was transferred to the reactor and the heat transfer systems after preheating to 350°F. Sodium circulation tests, which proceeded without significant incidents, were carried out initially at 350°F and later prior to the first power run, at 670°F, which is approximately the average core temperature expected during full power operation.

Physics experiments with sodium in the core were initiated with the "wet" critical experiment (critical loading 33 fuel elements at 350°F), which was completed on 25 April 1957. Physics experiments at

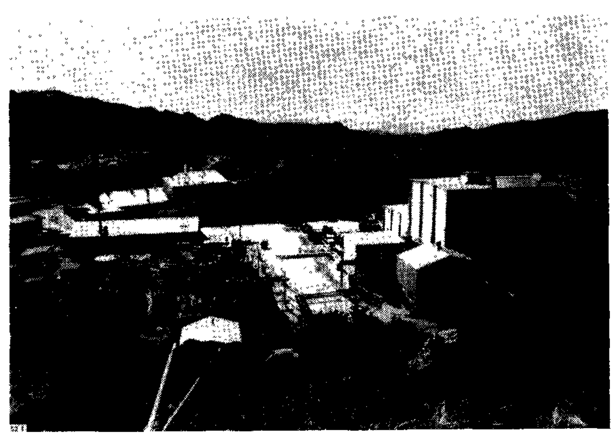


Figure 1. Overall plant

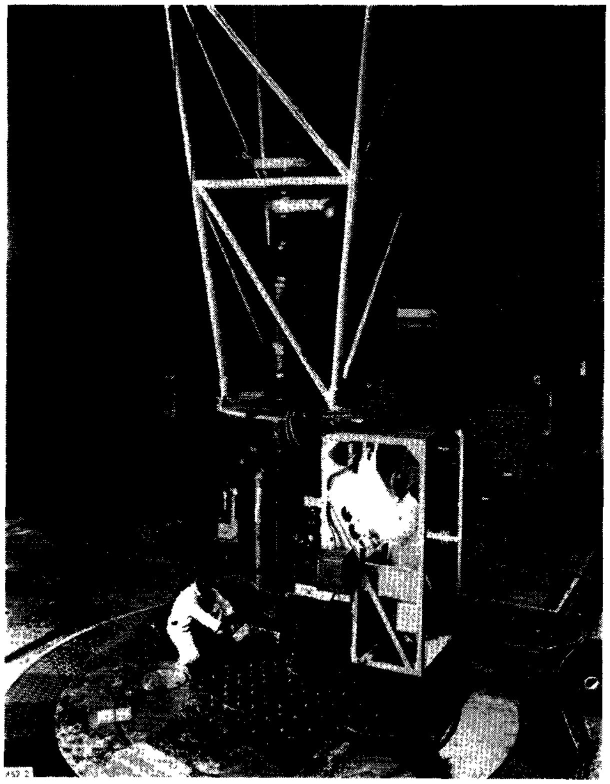


Figure 2. Reactor loading face and fuel handling cask

relatively low power levels to calibrate the control and safety rods, to measure the worth of various core elements, to measure the temperature coefficient, and to determine the neutron flux distribution were carried out intermittently until the middle of January 1958.

Several maintenance activities, including the replacement of a few bellows-seal valves in the sodium service system and the removal and the replacement of the disposable cold trap cartridge, were carried out in 1957. The most important maintenance operation performed during this period was the removal of a radioactive fuel element that had been damaged while being inserted into the core. These operations, which were performed with relative ease even though the sodium had become activated, provided confidence in the maintenance techniques that had been developed.

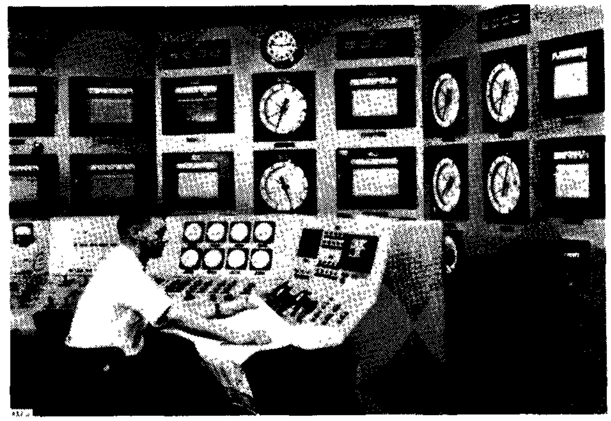


Figure 3. Control room console

Three power runs to demonstrate feasibility and to gain information considered important prior to full power operation were carried out during the latter half of 1957. Power levels during these runs were quite stable even when the automatic control system was not in operation. For example, during one 17-hr steady state period a timer connected to the regulating rod indicated only 2 min of integrated operation, and this included shim adjustments for xenon buildup. Operating information for these power runs, along with the data from the full power run, is given in Table 1. During the period the power runs were

Table 1. Operating Information from the First Four SRE Power Runs

Run number	1	2	3	4
Period for entire run	July 10–15 1957	July 25–26 1957	Nov. 7–20 1957	May 6–28 1958
Period of electrical power generation	July 12–15	July 25–26	Nov. 9–20	May 21–28
Number of scrams per week other than those purposely initiated	7.1	7 7	1.7	5.0
Number of hours turbine was "on line"	95	43.5	304.5	170
Kwhr of electricity produced	59,800	18,400	290,850	691,950
Electrical power, maximum	1.7 Mw	1 Mw	2 Mw	5.8 Mw
Thermal power, maximum	6.5 Mw	4 Mw	7.6 Mw	21 Mw
Main primary sodium flow rate	1000 gpm	780 gpm	1080 gpm	1300 gpm
Main secondary sodium flow rate	950 gpm	690 gpm	1040 gpm	1300 gpm
Steam flow rate	20,500 lb/hr	10,000 lb/hr	27,500 lb/hr	59,000 lb/hr
Reactor inlet temperature	481°F	475°F	505°F	576°F
Reactor outlet temperature	631°F	605°F	675°F	944°F
Steam generator sodium inlet temperature	605°F	530°F	638°F	875°F
Steam generator sodium outlet temperature	465°F	440°F	473°F	498°F
Steam temperature	508°F	590°F	628°F	800°F
Steam pressure	450 psig	460 psig	470 psig	580 psig

scheduled several scram tests were conducted. On the basis of information gathered, a decision was reached to perform certain modifications before operating the plant at full power. These modifications are as follows:

1. Installation of electromagnetic eddy current brakes in the main primary and secondary sodium systems so that flow decay may be matched to power decay following a scram. The possible need for these devices to minimize temperature changes was indicated by systems analysis studies during the design phase, but a decision regarding installation was postponed pending the accumulation of test data because of the many assumptions involved in the calculations.
2. Installation of an electromagnetic pump to transfer sodium from the moderator can cooling plenum to the main primary sodium system. This modification was desirable because of sodium leakage across the bottom grid plate.
3. Installation of a line from the discharge side of the main secondary pump to the main secondary expansion tank to provide for the removal of entrained helium in the system. During certain operating procedures the helium in the system caused gas binding of the pump.
4. Installation of a disposable cold trap of improved design.

The plant was shut down from mid-January to mid-April 1958 for the modification program.

Immediately following the completion of the plant modifications, a series of tests were conducted to evaluate the effectiveness of the design changes. These tests proved to be very successful and as a result full power operation was attempted and achieved on 21 May 1958. Upon completion of the power run the reactor and associated components were inspected and no problems from high power operations were evident. The plant is now preparing for several short term full power runs to obtain fuel irradiation data and to gain long term operating experience.

PHYSICS OF THE REACTOR

The critical mass of the SRE was determined both with and without sodium in the reactor. In the dry subcritical experiment the core was loaded to within two fuel elements of criticality. This loading was sufficiently high so that a critical mass could be determined accurately by extrapolation. With sodium in the core, the reactor was loaded to the critical condition and then excess reactivity added.

The results of the experiments yielded a dry critical mass of 22.2 fuel elements, or 42.4 kg of U^{235}, and a wet critical mass of 32.6 fuel elements, or 62.2 kg of U^{235}. These compare with theoretical values of 21.1 elements for the dry reactor and 28.3 for the wet case. The theory made use of data from an exponential experiment. Figure 4 shows the average inverse multiplication curve obtained from three symmetrically located counters for both the dry and wet case.

The volume fractions of materials in the SRE core at the end of the critical loadings were as follows:

	SRE Dry Core (without sodium coolant) 68°F	SRE Wet Core (with sodium coolant) 356°F
Uranium	0.0294	0.02967
Graphite	0.8702	0 8696
Stainless steel	0.00383	0.00337
Zirconium	0.0173	0 0173
NaK	0.00152	0.00131
Void	0 0785	0.0161
Sodium	—	0.0627

Experimentally measured neutron flux distributions were obtained in the dry subcritical and the wet critical reactor. In the latter case, both 35 and 43 element fuel loadings were studied. Measurements of the thermal flux were made using bare and cadmium covered gold foils loaded in graphite stringers. These were inserted through the top shield into special experimental thimbles located in or between moderator cans.

Flux plots in the dry core were in essential agreement with theory. For the wet core, Figs. 5 and 6 show the thermal flux obtained for the radial and axial cases respectively. In the radial plot of the core and reflector region agreement between experiment and theory is good. The disagreement in the axial reflector and sodium plenum is due largely to the neglect in theory of (1) the influence of the four safety and three control rods, four of which protrude about one foot into the reflector, and (2) the presence of fuel and control process tubes passing through the reflector.

The concrete biological shielding appears to be more effective than predicted from theory. The fast neutron relaxation length, indicated from the experimental curve, is 7.3 cm. This is to be compared to a value of about 10 cm calculated from removal theory. One possible explanation of this difference is that the concrete is still in the process of curing and contains excess amounts of water.

The control and safety rod worth was determined by the usual techniques of period measurements and

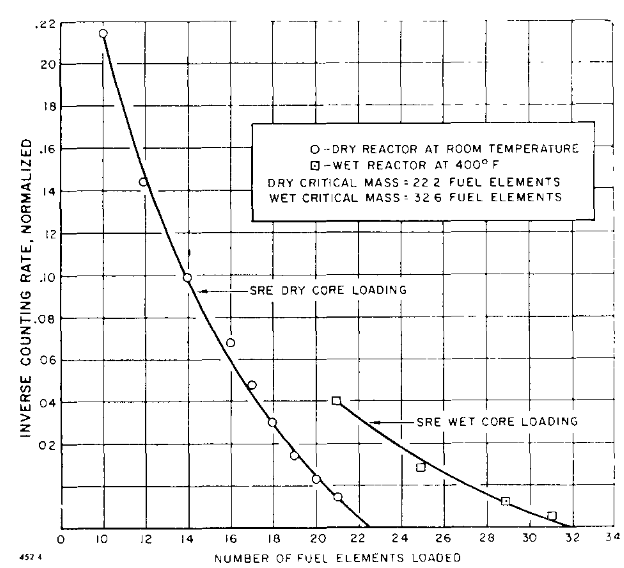


Figure 4. Critical loading

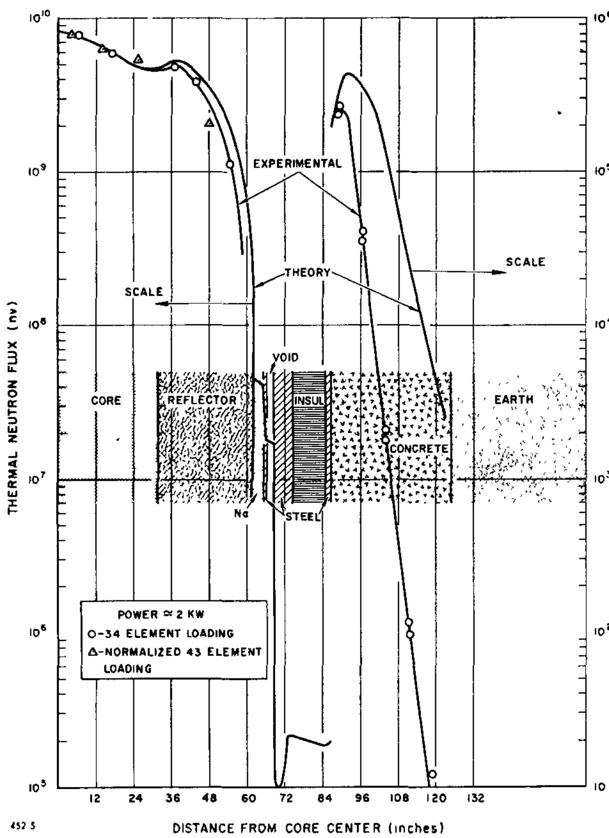


Figure 5. Thermal neutron flux plots across the radius

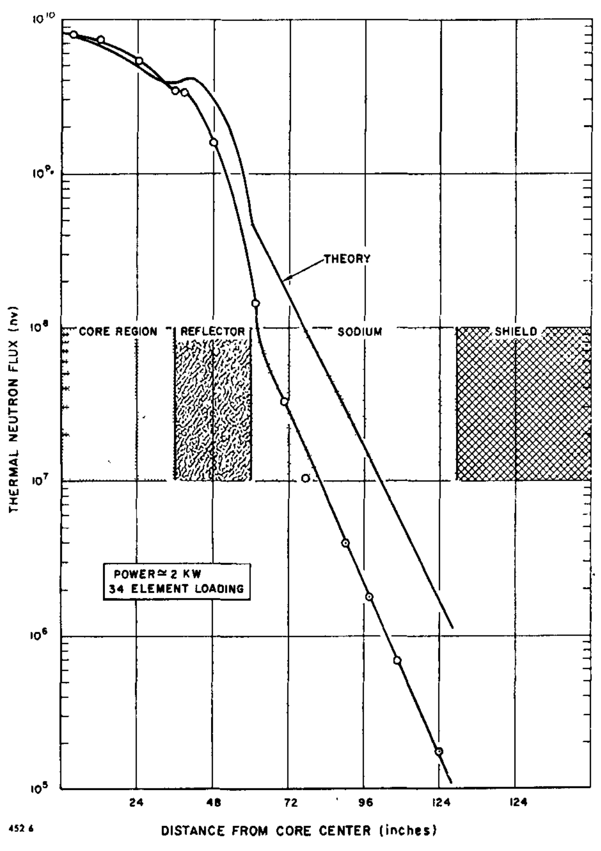


Figure 6. Thermal neutron flux plots along the axis

subcritical rod calibrations. The average reactivity worth of a control rod at a 43-element loading was found to be about 2.0%, compared to the theoretical estimate of 1.6%. From similar measurements, the worth of the first fuel element loaded beyond criticality (34th element) was found to be 0.4%. This compares with an expected worth of 0.3%. Safety rod worth, as determined by an inter-calibration with a control rod, was found to be about 1.7%. Safety rod drop measurements gave a single rod worth of 1.5%, and a ganged 4-rod insertion worth of 4.4%.

Danger coefficient measurements have been made on several experimental fuel element types, a stainless steel thimble and a graphite dummy fuel element. The results, which are somewhat preliminary, are summarized in Table 2. Danger coefficient measure-

Table 2. Danger Coefficient Measurements—34-Element Loading[a]

Element	*Worth Relative to Standard SRE Fuel*
1. Six-rod cluster of SRE fuel with gas-filled center rod	−0.035%
2. Hollow element of standard SRE fuel	+0.03%
3. Dummy fuel	−0.85%
4. Seven-rod cluster of natural uranium	−1.1%
5. Seven-rod cluster of thorium–uranium (7.6% fully enriched uranium)[b]	+0.176%
	Worth Relative to Dummy Fuel Element
6. Stainless steel thimble	−0.35% (theory 0.33%)

[a] Measurements made in fuel channel located a distance of one moderator can width from center of reactor.
[b] 43-element loading.

ments were also made for the radial worth of a standard fuel element to a dummy graphite element and are plotted in Fig. 7.

An isothermal temperature coefficient was measured over the range 177°C to 400°C. Nuclear heating was used to raise the reactor temperature. Reactivity measurements were made about every 30°C with the reactor in essentially an isothermal condition. Although a complete analysis of the results is not

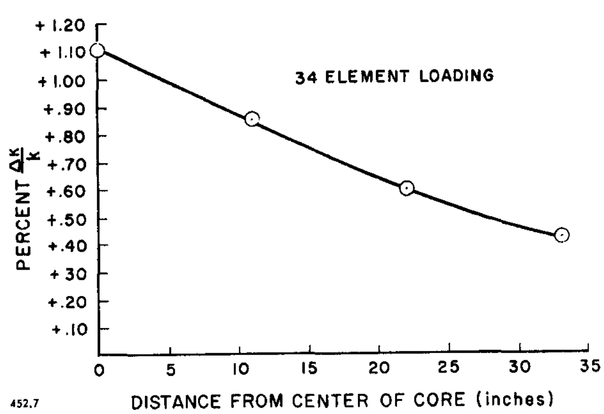


Figure 7. Radial worth of conventional fuel element to dummy graphite element

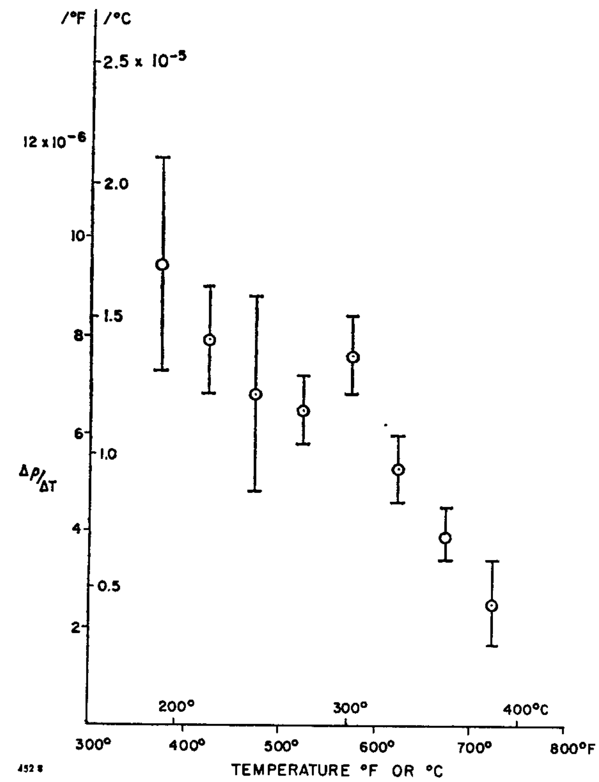


Figure 8. Overall isothermal temperature coefficient

yet available, a tentative isothermal temperature coefficient is shown in Fig. 8. The isothermal coefficient is positive, but there is a relatively long time-constant associated with this coefficient because the time-constant for diffusion of heat into the graphite is about two min. The prompt temperature coefficient ($-13 \times 10^{-6} \Delta k/k/°C$) (time-constant, two sec) which arises from heating of the uranium is negative and provides a stable reactor.

The power coefficient calculated from the measured values of the metal and overall isothermal temperature coefficients is negative and is equal to -6.8×10^{-5} $\Delta k/k$/Mw.

COMPONENT PERFORMANCE

In general, the heat loss calculations for the sodium piping and vessels were confirmed during operation. Some minor adjustments were made in the insulation thickness and input power to the heaters on some fill tanks. Some of the difficulty with non-uniform heating was due to insufficiently dry insulation in the secondary system, part of which is located outside the building. This was corrected with prolonged heating and additional weather proofing. The sodium piping has operated at 944°F and has evidenced no leaks. Head losses measured in flow tests on the system agree with the design calculations, with the exception that the suction pressure at the main primary pump is slightly below atmospheric (essentially 0 psig, design value was 1 or 2 psig). Oxide plugging has been experienced in some of the $1\frac{1}{2}$ in. and 1 in. sodium lines in the primary system, however, in the secondary system where the oxide content has been kept below 10 ppm no plugging has been observed.

As near as can be determined, ten bellows-sealed valves which leaked failed for operational reasons, rather than inherent weaknesses. A typical failure is thought to result from reheating a sodium-filled line, one end of which is plugged at a cold spot by oxide with the other end closed by the valve. The expanding sodium then collapses the bellows. A typical failure is shown in Fig. 9, and is compared with a normal bellow. There are nine valves in the system which have stem freeze seals. All of these have operated very satisfactorily, but some trouble has been experienced in closing the valves that have extension operators.

Three conduction pumps initially used for sodium transfer, which were obtained from a commercial supplier, tended to plug at the 0.5 in. by 0.035 in. throat section. They have been replaced with linear induction pumps, designed and built by Atomics International, which have proven to be satisfactory. Early in the sodium circulation tests the freeze seal was lost on the main primary pump, filling the case with 600 lb of sodium. The pump was pulled, cleaned and modified before it was reinstalled. The modifications included correcting the initially miswired series-shunt field of the motor to provide load-stable operation, and hydraulically rebalancing the pump impeller by increasing the number and size of the "weep" holes to decrease the pressure at the freeze seal from 15 psig to 6 psig. Since this modification was made, each of the four pumps has given 4000 hr of trouble-free operation.

Some gas binding has been experienced with the main pumps due to the piping arrangement. A continuous vent has been installed near the secondary pump to eliminate the problem. This arrangement is not possible on the primary side because of the possibility of convective flow interruption caused by the antisyphoning effect of a vent. Operational procedures will have to suffice here. Since this vent line was installed the pump has given no trouble from gas binding.

The sodium melt station worked well and permitted the melting of 200 barrels (55 gal, 400 lb each) of sodium at the rate of two hr per barrel.

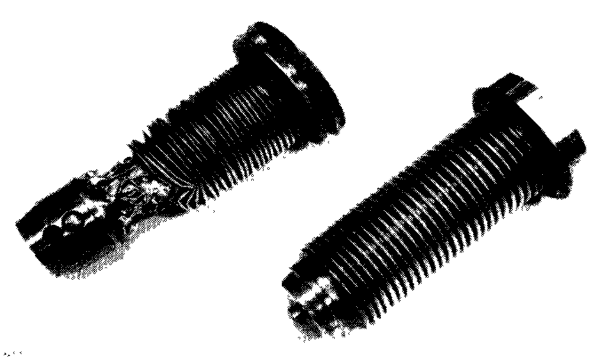


Figure 9. Valve failure

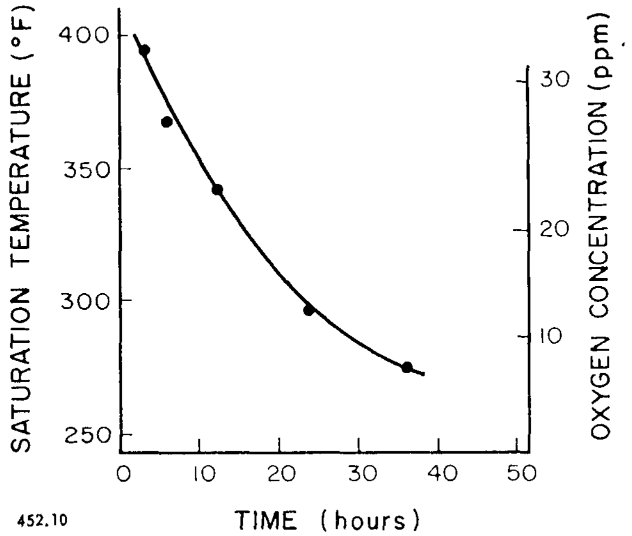


Figure 10. Cold trap performance

The original disposal cold trap appeared to work well if the bulk oxide was less than about 50 ppm. At this concentration the traps easily brought the indicated oxide concentration down to less than 10 ppm (Fig. 10); however, plugging of the traps occurred at higher bulk oxide concentrations. An improved disposable cold trap has been designed and installed and has operated successfully at the higher concentrations. This new trap, shown in Fig. 11, handles sodium flows four times as great as the original design and as a result it has a higher capacity for trapping oxides before plugging. A counter flow, forced-convection cooling jacket is utilized to distribute the temperature gradient over the entire cold trap length. In the original design, the trap was cooled by toluene at its boiling point, and a steep temperature gradient existed at the boiling surfaces. The original cold trap tended to plug in a short time due to oxide precipitation at the point of this temperature change. Consequently, only a very small volume of the trap was effective. The improved design utilizes tetralin as a coolant.

The boiling toluene primary process cold traps, which had been provided in the original design in addition to the disposable traps, were not too effective and have been removed from the system. Plugging meters have proven a satisfactory method of determining oxide concentration in sodium systems.

A complete evaluation of the tetralin cooling effectiveness has not been made at this time. The present indications are that the tetralin cooling is more than adequate, totally and in individual areas. The carbon steel pipe used for the tetralin system has shown some rusting, arising from moisture in the tetralin. The atmosphere cover gas in the surge tank has been replaced with dry nitrogen to eliminate the moisture source, and activated alumina has been installed in a bypass filter to dehydrate the tetralin.

Preparations for the power runs and the operations during the physics tests provided approximately 800 fuel transfer manipulations with the fuel handling cask. Although several minor modifications to the fuel handling system have been made and a few other improvements are visualized, the basic principle of the fuel handling operation remains the same and the unit fills the original design requirements. Time for one fuel element change, including the washing cycle, amounts to two hr.

Two Mark I and two Mark II control rods, and four Mark II safety rods are used to control and shut down the SRE. The Mark I control rods use a ball screw operating in the hottest part of the reactor just above the core. On the basis of prototype tests and the postulated operating time to be expected for a regulating rod, it appeared that the Mark I design would have a limited life. Therefore, it is used only for shim control. The Mark II control rods, having the ball screw drive above the floor level, appear to have practically unlimited life. (Tests show no apparent wear after 1000 cycles with a temperature of 1100°F on the rod.) These rods are used for regulation. Both types of control rod have performed very satisfactorily. However, operating experience has shown that the reactor system is so stable that it requires very little regulation. Even with the present design, the Mark I control rod has an estimated life of five years.

The Mark I safety rod retrieving system proved unreliable because of the complexity of the mechanism in the hot zone. It was replaced with the Mark II type which has the drive above floor level. Instead of the mechanical latch and ball screw retriever, the new design uses a chain drive and an electromagnet for raising and dropping the poison rings. The rod fall is

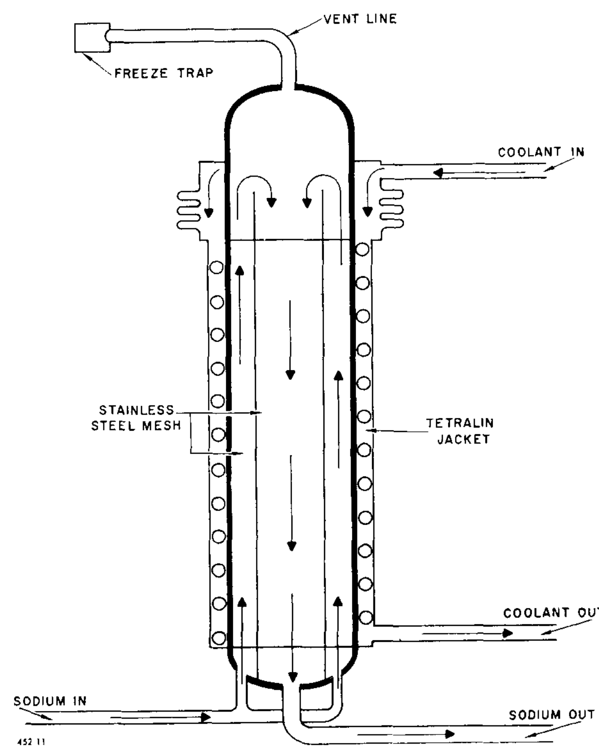


Figure 11. Modified cold trap

cushioned by a gas snubber. This design has functioned satisfactorily and reliably. Tests show that, with no afterglow heat present in the core, the safety rods will reduce reactor power from 100% to 1% in 0.5 sec after the scram is initiated. At this time the rods have entered the core region about 21 in.

The Atomics International designed and manufactured sodium instrumentation has performed satisfactorily. The sodium level gauges have adequately indicated the level of sodium in the various tanks in the SRE. The magnetic flowmeter signal-to-statistical-noise ratio is approximately 3% at full flow rates. It has been noted that on some of the sodium service magnetic flowmeters the signal sensitivity for a given flow has decreased as much as 20%. The main flowmeters, which were stabilized prior to installation, have not shown this drift. False alarms caused by moisture and mechanical shorts have been given by the sodium leak detectors; however, no sodium leaks in the tanks and piping have been encountered to date. The sodium leak tracings were repaired when the piping insulation was removed for the modification work. The electrical sodium pressure instrumentation has operated satisfactorily. Some zero drift was encountered with pneumatic sodium pressure instrumentation. The orifice on the pneumatic pressure transmitters and pilot valves has been redesigned to reduce zero drift in the main and auxiliary secondary sodium pump suction and discharge pressure instrumentation.

The fission counters and ionization chambers have operated satisfactorily in the specified range of their respective channels. No drift in sensitivity has been encountered. Double-shielded coaxial cable was installed between the neutron detectors and nuclear channels to improve signal-to-noise ratio. The period amplifiers in the count rate channels have proven to be ineffective because of the neutron statistical noise, and have been eliminated from the SRE safety system. Maximum sensitivity adjustment is required on the linear amplifiers to provide sufficient gain from the fission counters. The time constants have been modified in the Log N and period channels to eliminate false alarms which are caused by neutron statistical noise and voltage transients from the line power. The voltage regulation for the Log N period channels has been modified to decrease line power transients and reduce false period scrams.

The electromagnetic pump installed on the moderator coolant line to remove excess grid plate leakage operated quite well during the full power run.

The operating characteristics of the eddy current brakes, used for maintaining the reactor core temperature gradient after a scram for extended periods of time, proved to be entirely satisfactory. Maximum coil temperatures after three hr of continuous full current operation were about 100°F below the acceptable limit of 350°F for the wire insulation.

The principal operating characteristics of the steam plant during part of the May 1958 full power run is shown in Fig. 12. Initially, the control of the steam temperature was erratic and unstable operation

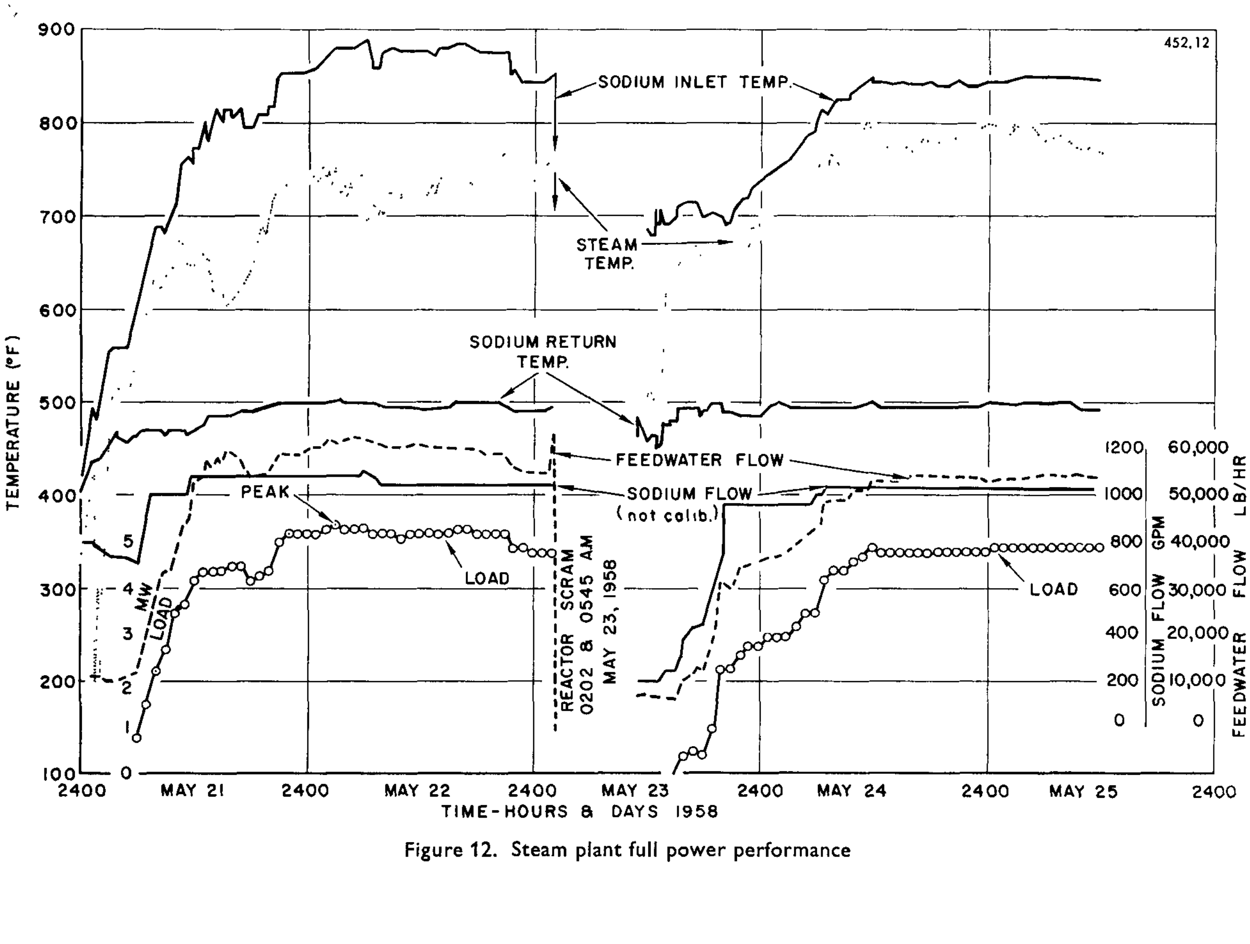


Figure 12. Steam plant full power performance

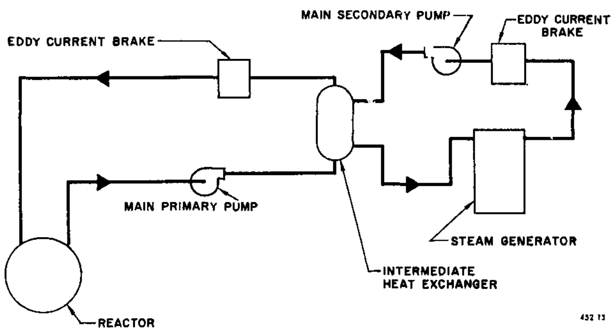


Figure 13. Eddy current brake installation

resulted until feedwater flow was reduced and the sodium outlet temperature was allowed to increase to about 500°F. The cause of this condition is attributed to uneven distribution of water flow in the tubes. Thermocouples on the steam outlet header show that saturated water is ejected from the bottom tubes and superheated steam from the top tubes. The tubes have been orificed at the inlet tube sheet in an attempt to provide more even flow distribution by increasing the pressure drop across the steam generator. The plant has performed very satisfactorily since this modification. Previous to the installation of the orifices the temperature difference across the steam outlet tube sheet at full power was about 200°F; with the modification the difference is only 30°F.

During the same power run two scrams occurred within a few hours of each other on 23 May. The recovery was slow and deliberate; however, the recovery time can be improved as evidenced by past experience. Two similar consecutive scrams took place during the November low power runs. Recovery from the first scram was accomplished after twelve hr, and from the second scram in six hr, showing the benefits of experience even though the operators were cautioned to use extreme care in restarting and no emphasis was placed on minimizing down time.

PLANT MODIFICATION

Eddy current brakes to control the thermal convective flow in both the primary and secondary main coolant loops after a scram have been installed as shown in Fig. 13. The device itself is shown in Fig. 14 and is simply a means of applying a variable dc magnetic field across a flattened section of pipe. The head loss at full flow is 1.5 psi. 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 is done automatically, using a thermocouple signal from the fuel coolant channel, so that the sodium coolant temperature will remain constant. The addition of this device will eliminate transient thermal stresses in the core tank nozzle and in the moderator cans. The brakes have been tested in a sodium test loop. The results of these tests are shown in Figs. 15 and 16, and compare well with the equation developed by R. H. Baker from Ref. 4, given in Fig. 17.

The time-constant of the braking action is approximately 2 sec, compared to a 3 min 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 the off-normal flow circuit and the coolant channel temperature monitor.

Control of the moderator flow by means of the valve initially provided in the moderator line proved ineffective because of the excessive leakage past the grid plate. Instead, this flow is now controlled by a

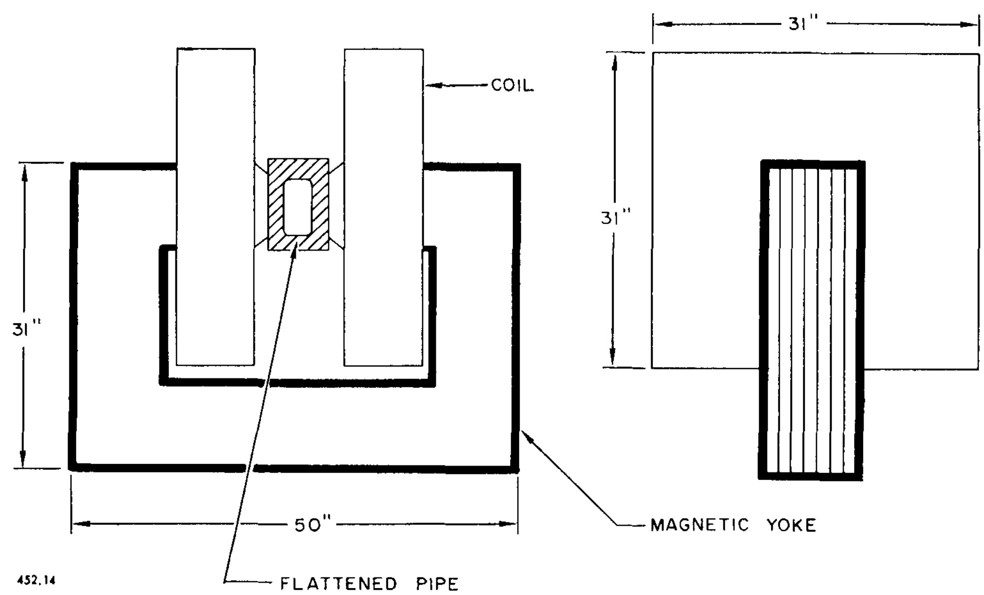


Figure 14. Eddy current brake

variable flow (60 gpm max) linear induction pump located in the gallery. The pump is able to transfer sodium in either direction, and by pumping backward through the moderator line any desired fraction of the grid plate leak may be removed from the moderator plenum. The pump is shown in Fig. 18.

Water in the galleries is being eliminated by recirculating the gallery atmosphere through a refrigeration dehydrator. The circulating gas is monitored continuously for water vapor content, oxygen concentration and radioactivity. Initially the system removed ten gal of water in the first two hr of operation and is presently removing about five gal in every 40 hr.

Studies of means of increasing the reactivity of the SRE core disclosed that the substitution of Zircaloy for stainless steel in the control and safety rod thimbles will provide the equivalent of five additional fuel clusters and permit the reactor to operate for at least 250 days at full power. The gain in the worth of the control rods is equivalent to one additional rod.

An off-normal flow circuit was developed which compares a set sodium flow with actual flow and pump speed.

The circuit is arranged to scram the reactor if the flow should depart $\pm 10\%$ of the set value.

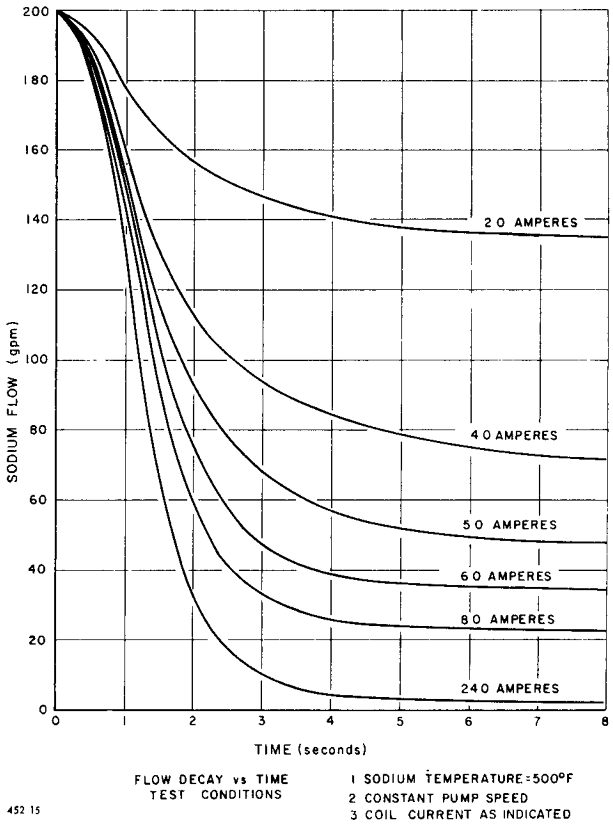


Figure 15. Flow vs. time characteristic for eddy current brake

ENGINEERING TEST RESULTS

During the power runs which preceded full power operation, a number of planned "scram" tests were carried out to observe the rates of temperature change throughout the system and to compare these changes with the values which were calculated in the system analysis studies. Uncertainties in these 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.

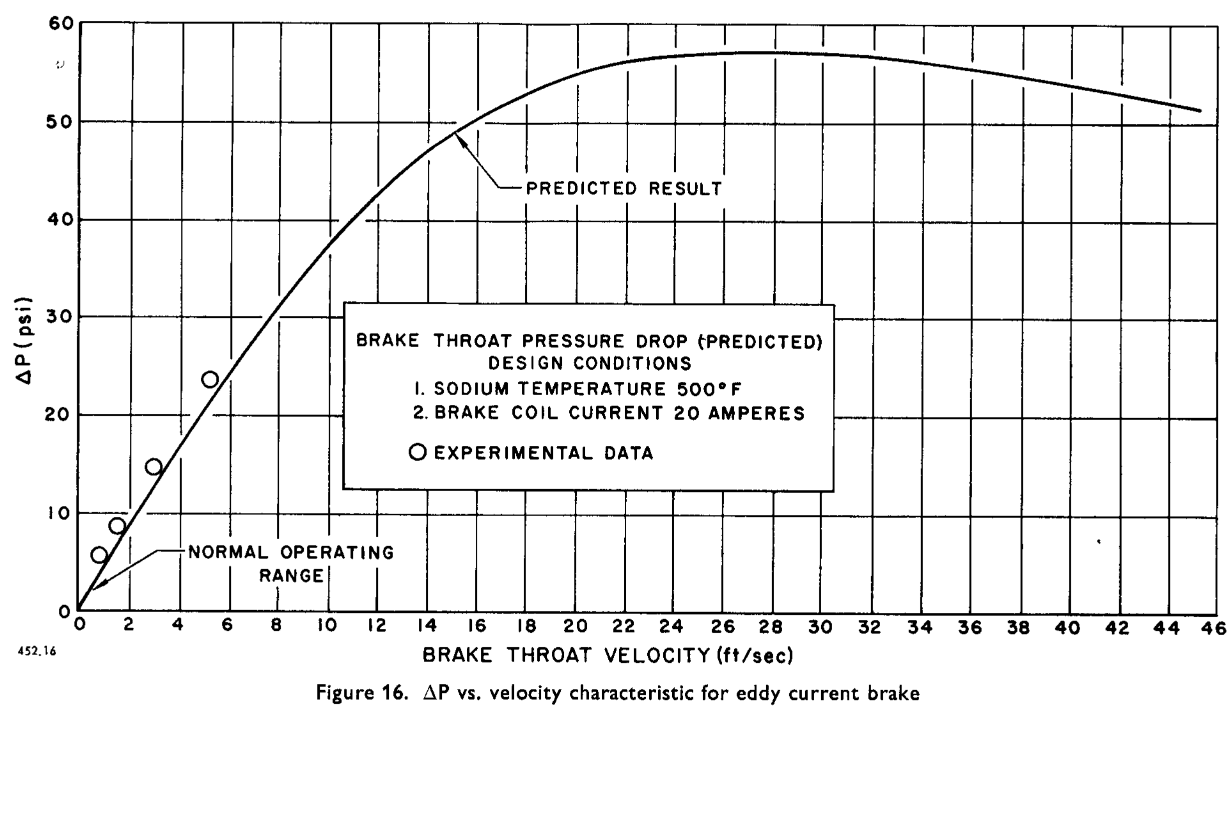
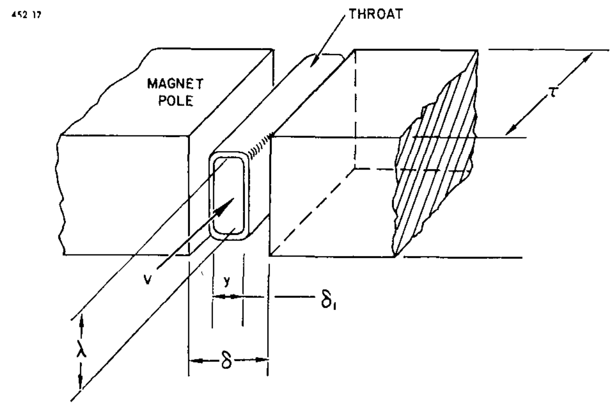


Figure 16. ΔP vs. velocity characteristic for eddy current brake



$$\Delta p = \frac{V\lambda h}{4\rho\tau}\left[\frac{\left(\frac{\lambda}{\tau}+\frac{\tau}{\lambda}\right)}{\left[\frac{4V\,\delta.\lambda}{\rho\,\delta}\right]^2+\left[\frac{\lambda}{\tau}+\frac{\tau}{\lambda}\right]^2}\right] \times B_g^2$$

V = FLUID VELOCITY

$\frac{h}{\tau}$ = NO OF POLES = 1.0

FLUX DISTRIBUTION IS ASSUMED AS

$B_x = B_{max} \sin \frac{\pi x}{\tau}$

$B_y = B_{max} \sin \pi \frac{y}{\lambda}$

$\rho = 16 \times 10^3$ abohm-cm (500°F Na)

B_g = AIR-GAP FLUX-DENSITY (GAUSS)

Δp = DRAGGING PRESSURE (dyne/cm²)

λ = 19.7 cm

τ = h = 25.4 cm

δ = 10.16 cm

δ_1 = 5.08 cm

h = LENGTH OF ACTIVE SURFACE

Figure 17. Rudenberg's equation

Two calculations were made of the reactor outlet temperature after a scram assuming first, complete mixing of the exit sodium from the reactor core with the sodium in the upper plenum, and secondly, complete stratification of the exit sodium from the core. The comparison of the test data with the theoretical calculations is shown in Fig. 19, and indicates that nearly complete mixing occurs. The test data were extrapolated from about one-third power to full power and full afterglow heat, as this is the more interesting case.

The time dependence of the core tank wall temperature and the reactor outlet temperature following a scram from full power are predicted in Fig. 20. It is interesting to note that the afterglow heat is not very effective in reducing the difference between the core tank and the outlet nozzle temperatures after a scram. Presuming the reactor outlet nozzle follows the

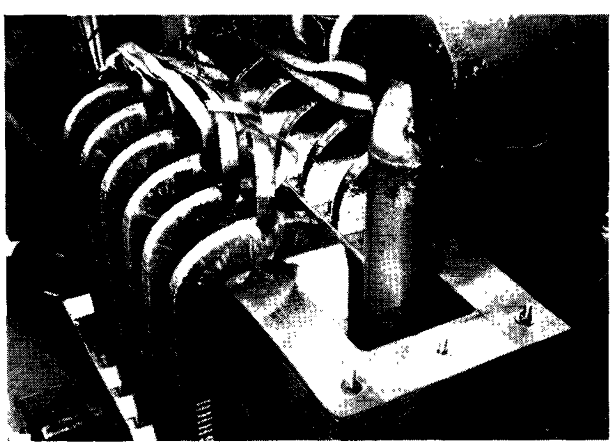


Figure 18. Moderator coolant line pump

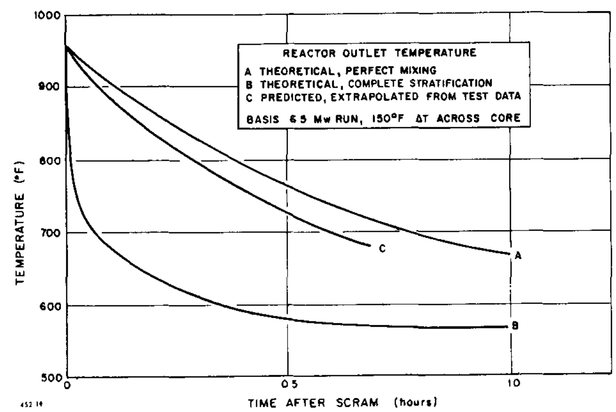


Figure 19. Degree of mixing in upper sodium plenum

temperature of the outlet sodium, the nozzle very quickly drops to a temperature which is about 100°F below the average core tank wall temperature. This temperature difference will cause the nozzle body to contract to a diameter smaller than the diameter at its attachment to the tank wall. Calculations based on this information indicate that the stress in the core tank nozzle under these conditions would exceed the yield strength by a factor of approximately 2.5. Control of the thermal convection flow following scram shutdowns has eliminated this stress.

In the course of the design and development of the reactor and its components particular attention was given to the temperature transients associated with the moderator cans. Early simulator studies revealed the can head-to-coolant tube junction developed excessive stresses following a scram, since the center channel fuel coolant temperature was predicted to drop rather rapidly compared to the corner channel moderator coolant temperature. This joint was redesigned to withstand the worst case of a step change of 460°F in the fuel coolant temperature while the moderator coolant temperature remained

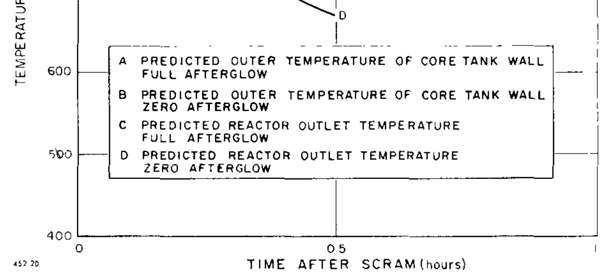


Figure 20. Core tank and core tank outlet nozzle temperature following a scram

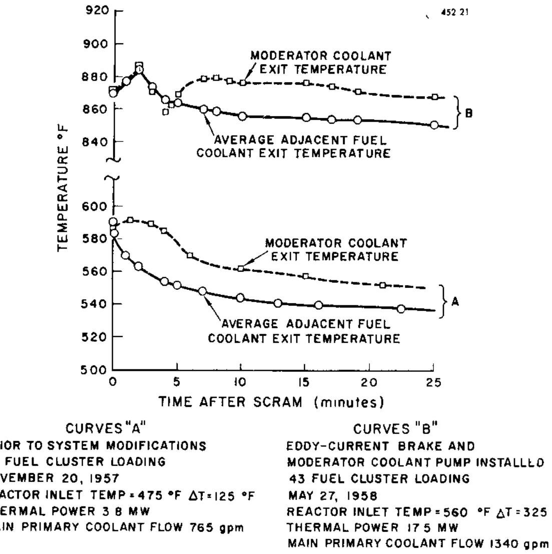


Figure 21. Moderator can temperature following a scram

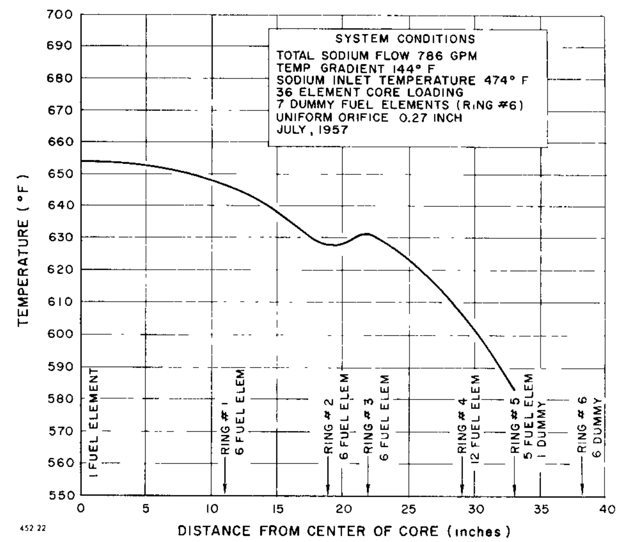


Figure 22. Core radial temperature plot

constant. Figure 21 shows the temperature transients experienced in the reactor following a scram before and after the eddy current brake and moderator pump were installed. When scram data taken before the modifications is extrapolated to full power scram conditions it leads to a prediction of about 100°F difference between two adjacent center and corner coolant channels. As can be seen from the high power scram data, the modifications hold the temperature difference to about 20°F which reduced these stresses to an insignificant level.

During the first power run using a 36-element core loading, a large, unexpected steady state temperature difference was observed between the fuel channel outlet sodium and the moderator coolant. Hydraulic tests performed subsequent to the power run, confirmed that at design flow rates of 1080 gpm in the main system, a moderator coolant flow existed even though the programming valve was closed and the flowmeter in the line indicated a zero signal. The theory that sodium leakage occurs across the grid plate has strong experimental support and it has been shown that during the power run the excess moderator flow amounted to 50% more than that required to produce a temperature match between the moderator exit sodium and the adjacent fuel channel exit sodium. This amount of excess cooling introduces a calculated stress at the junction of the scallop tube and the can head of three times the yield stress at full power. Since, in this case, the moderator coolant temperature is lower than the fuel channel sodium exit temperature, this thermal stress problem is similar but not identical to the problem studied during the design stages of the moderator can head. At full power conditions a temperature difference greater than 120°F will cause stresses in the can to exceed the yield point.

Later in the experimental program with a 43-element core loading, and flow rate in the fuel coolant channels of 1080 gpm, it was discovered that due to a lower core pressure drop the leakage rate alone approximately equaled the required moderator coolant flow rate. However, because the reactor is operated as an experimental plant, which dictates various core loadings and coolant flow rates, a moderator pump was installed to control the amount of sodium leakage past the grid plate under all conditions. During the full power run when the main coolant flow rate was significantly above 1080 gpm, the moderator coolant pump successfully removed excess grid plate leakage flow and thereby improved the temperature situation between moderator coolant exit temperature, as measured in two corner channel locations, and the adjacent fuel channel exit temperatures. During increasing power operation the moderator coolant temperature lagged the fuel coolant exit temperature

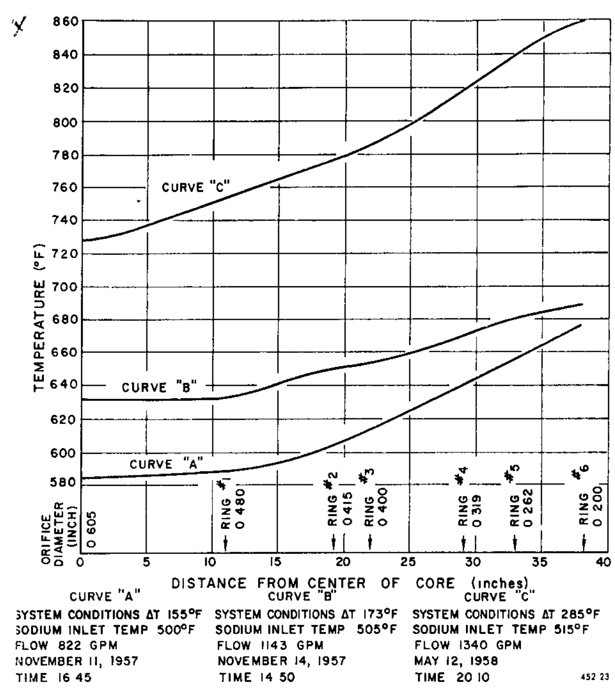


Figure 23. Radial temperature plot—43 element loading

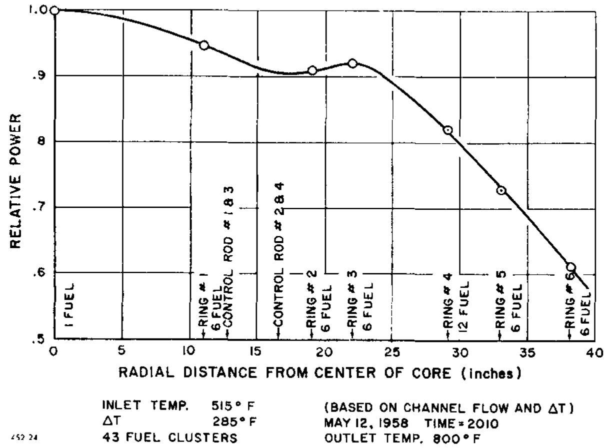


Figure 24. Relative power vs. radial distance

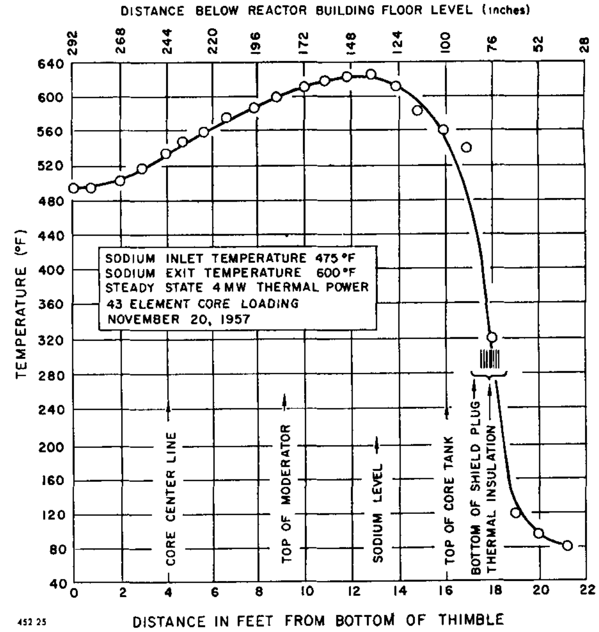


Figure 25. Core temperature vs. axial distance

by approximately 30°F. After a scram its temperature became about 20°F higher as seen in Fig. 21. During steady state operation these adjacent temperatures were matched within 5°F. Under these controlled conditions most of the moderator cans are virtually stress free.

The initial radial temperature profile for a 36-element fuel loading and uniform 0.27 in. diam orifice plates is plotted in Fig. 22. Since the flow through the six-holed orifice plates is uniform, the temperature curve is the same shape as the power curve. The estimated radial power distribution for the core with a 43-fuel-cluster loading was obtained by extrapolating the measured radial power distribution for the 36-fuel-cluster loading. From the extrapolated data new orifice plates were sized for a 43-cluster loading and full power operation in an attempt to flatten out the temperature distribution in the core.

Figure 23 shows the sodium exit temperatures from the fuel channels for two low power runs and one

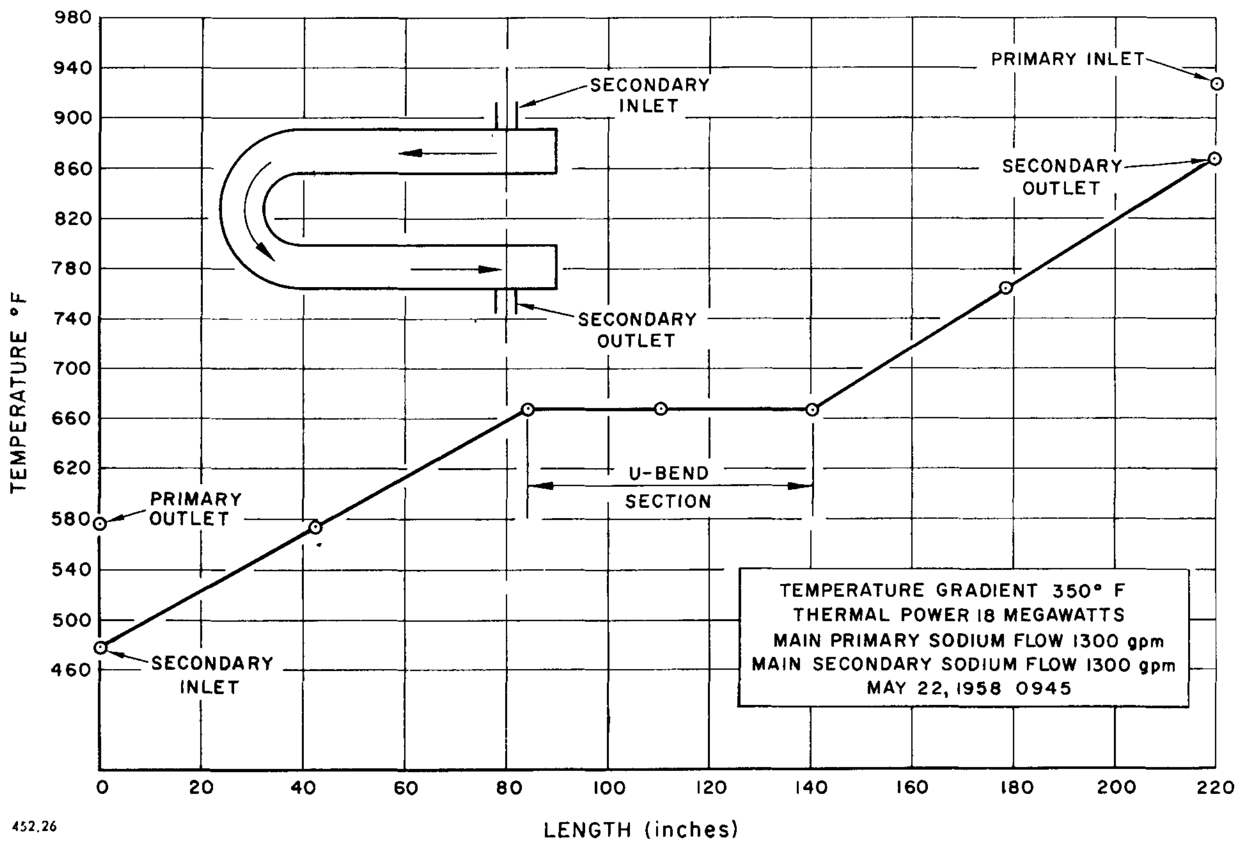


Figure 26. Heat exchanger shell side temperature distribution

high power run with the new orifice plates installed and with a fuel loading of 43 fuel clusters. The orifice plates at each radial distance from the center of the core are identical for each ring but vary from 0.605 in. for the center fuel ring to 0.200 in. for the outermost fuel ring. The two low power data indicated that the temperature distribution would flatten out as desired, however, this effect was not realized at higher power levels and additional orifice changes will be required.

From the temperature distribution and flow rates observed during the high power run, the relative power versus radial distance was calculated and is plotted in Fig. 24.

The axial temperature distribution in the reactor is shown in Fig. 25 and indicates a maximum gradient of 25°F/ft along the core tank wall. The resulting stress is less than 3000 psi compared to the yield stress of 16,000 psi for 304 stainless steel at 800°F. The maximum gradient of 185°F/ft occurs across the reflective insulation which is stress free.

An unexpected result from the first power run was the 50% 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 taken during the recent full power run is shown in Fig. 26. The temperature curve is seen to rise uniformly at the straight portions of the exchanger but it is 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 plus some bypass flow which takes place in the straight sections accounts for the high log mean temperature difference, which is 84°F for the experimental data shown versus 60°F for the design value.

Transient data recorded on this equipment after a scram test discloses that rather complete temperature stratification occurs in the exchanger. Investigation of this phenomenon verifies that the density differences in the primary headers cause internal circulation of sodium flow through the top and bottom tubes. The

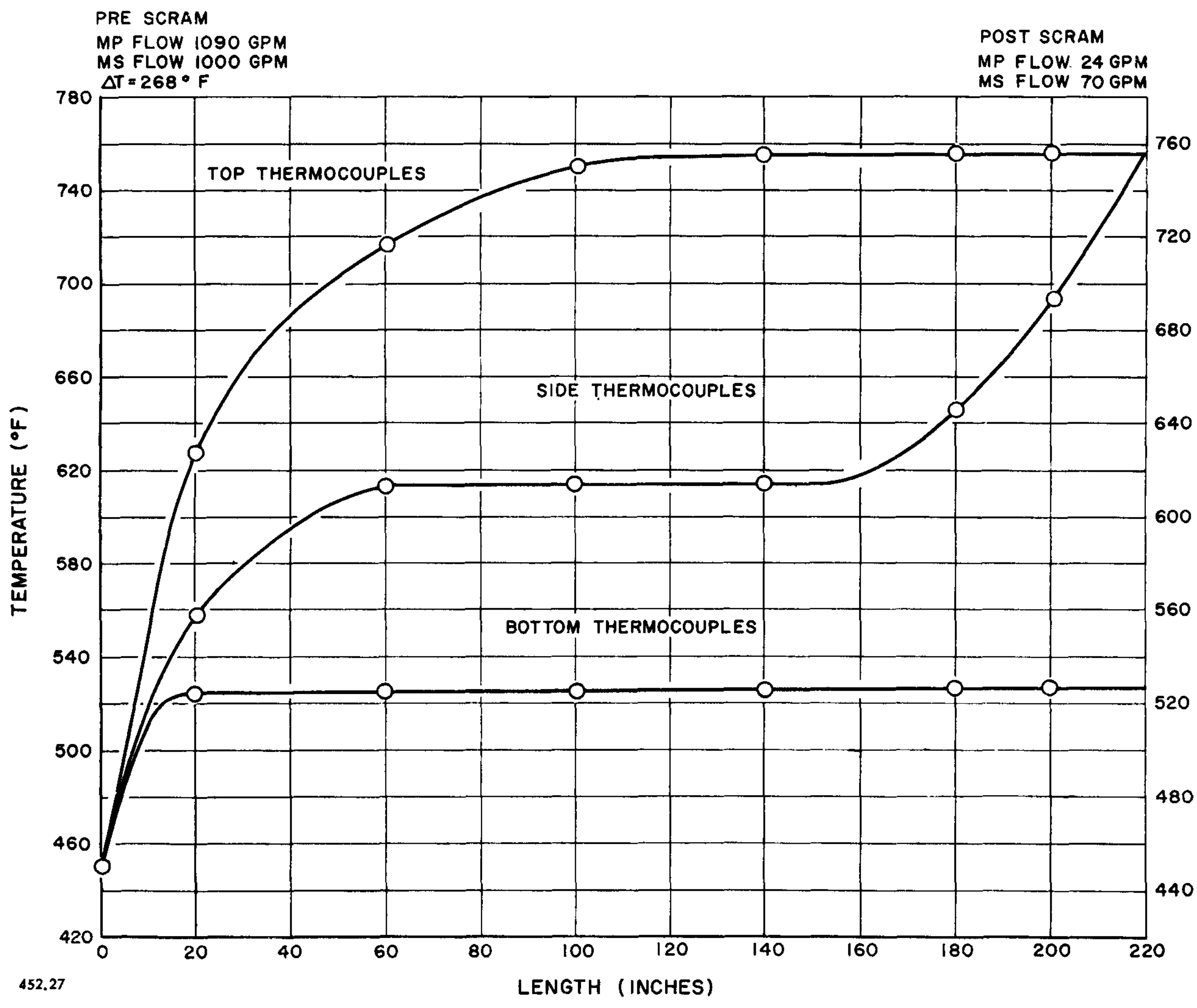


Figure 27. Stratification in main intermediate heat exchanger three min after a scram (12 May 1958)

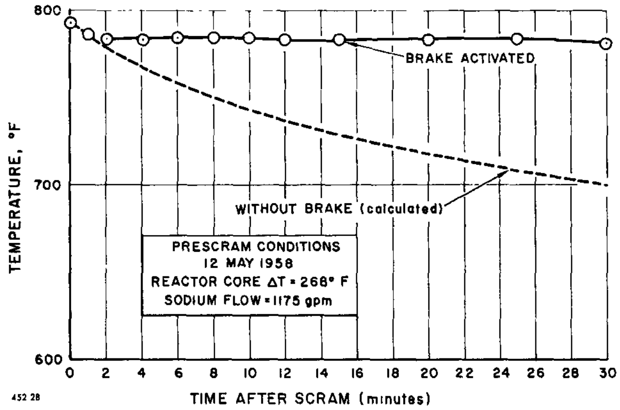


Figure 28. Brake effect on reactor outlet temperature after scram

magnitude of this flow varies with the temperature gradient across the exchanger and the convective flow in the primary system, but calculations show the direction of flow in the lower tubes is opposite to the initial flow. Recent data indicate that at a temperature gradient of about 200°F, reverse flow in the lower tubes occurs when the primary convective flow is below 50 gpm. The temperature data shown in Fig. 27 were taken three min after the scram was initiated. The log mean temperature difference at this time is about 145°F. 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.

Data from four scheduled reactor scram tests prior to full power operation have been studied in regard to the performance evaluation of the convection flow control system. These data provide a very satisfactory demonstration of the effectiveness of the recently installed flow control equipment for maintaining the reactor core temperature gradient after a scram. Temperature information obtained from the highest gradient, planned scram test is shown in Fig. 28 and indicates a drop in reactor outlet temperature

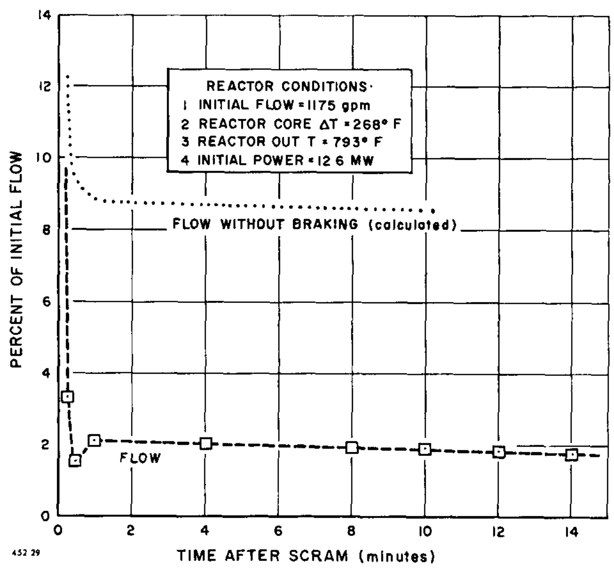


Figure 29. Primary system flow decay

at a rate of only 20°F/hr which is a factor of two better than the 50°F/ht limit based on the most conservative stress analysis.

Flow reductions due to the action of the eddy current brakes have been in accordance with expectations based on previous test information. The system temperature transients are minimized by matching the flows and power decay after a scram. In Fig. 29 the primary flow after a scram is compared to the predicted convection flow without the effect of the brake on the system. As can be seen, a flow reduction to about 22 gpm was achieved in the main primary loop as compared to a calculated value of 105 gpm due to an unimpeded convection driving head in the system. This flow reduction was accomplished with only 4.9 amp (2200 gauss) applied to the brake coils. The brake is capable of drawing 20 amp (8000 gauss).

Examination of the galleries after the first power run disclosed that water had condensed on the tetralin-cooled finned tubing used to space-cool the galleries, and had then dripped on the pipe insulation and puddled on the steel liner which covers the gallery floor and the lower three ft of the gallery walls. To determine the source and the approximate effusion rate of the water, five stainless steel sheaths, each 1.4 ft^2 in area, were sealed to the following sites: the concrete wall, the gap between the steel liner and the wall, one of the removable shielding blocks, and the pipe insulation. The sheaths covered dry-weighed bags of silica gel. These bags were weighed after the November power run and show a weight gain of approximately 1.0 lb of water per test sheath for the concrete tests, and approximately $\frac{1}{2}$ lb weight gain for the pipe insulation. Since the shield block does not come in contact with the ground and yet showed the same weight gain as the walls, it was concluded that the water was due to drying of the concrete which is accelerated by gamma heating. Examination of the construction records indicate that approximately 1400 ft^3 of water was used in the concrete mix. About half of this is expected to come out of the concrete over a one-year period. The recently installed dehumidification system will reduce the water content in the gallery atmosphere during this curing time.

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 had purposely 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, radiation dose rates in the order of 100 mr/hr were measured, as opposed to values of the order of 1000 mr/hr calculated.

OPERATING AND MAINTENANCE PROCEDURES

Start-up of the plant is quite direct and uncomplicated. A typical start-up begins with check out of the various reactor service systems such as the inert gas system and the reactor vent system. 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 neutron 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-sec period until reactor heat can replace the electrical heat. The overall temperature is raised isothermally to 500°F, and the feed-water 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% 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 over-speed tripping device. 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. To date, runs have usually been terminated with a planned scram to provide transient data. The startup time under the present conservative procedure is approximately eight hr.

Because of the large volume of sodium in the primary heat transfer system, relatively low concentrations of oxide, say 20–30 ppm, can provide enough precipitated oxide to plug $1\frac{1}{2}$ in. lines. Consequently, the precipitation temperature at a given concentration is an important operational parameter for the sodium piping. In addition to maintaining the sodium oxide concentration below 10 ppm, current practice is to drain all sodium lines that cannot be held above 350°F.

Fuel washing procedures are still being developed. One of the problems still under investigation is the time required for drying. This must be established experimentally, because the fuel normally cannot be examined for dryness.

The tetralin cooling system, which supplies all auxiliary cooling in lieu of water, has proven to be amenable to the operating procedures established for it. In general, successful operation of this system depends upon maintaining adequate cooling flow to critical items such as freeze seals. Originally proposed operating procedures called for a complete check of all auxiliary cooling flow rates as often as once each hour. This was found to be cumbersome, however, and the procedure now in effect is to check vital temperatures once each hour, resetting tetralin flows where necessary. Coolant flows to important areas are checked every four hr, and a complete survey of all auxiliary cooling flows is made at least once per week.

Procedures for the maintenance of sodium systems have been used for the replacement of a few bellows seal valves and the removal and replacement of the disposable cold trap cartridge. The pipe cutting and welding operations have been done on drained, partially drained, and full sodium piping on line sizes ranging from one in. to six in. in diameter. The only added precautions needed are to tape the ends of the pipe after the cutting operation and to purge them with argon or helium to minimize the oxidation of the frozen sodium.

Insofar as disposal of sodium is concerned, steam cleaning of most equipment, after the radiation level has decayed sufficiently, has become standard practice. As yet it has not been necessary to dispose of significant quantities of highly radioactive sodium.

A damaged fuel element was discovered in the core 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 7-rod cluster had remained in the core. Examination after the sodium level had been lowered showed that the cluster was protruding approximately 18 in. above the top of the moderator can and that the portion protruding was bent at approximately 30 deg to the vertical. The cluster, which had an activity corresponding to 200 r/hr at one ft, 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 in 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 top 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 not loaded unless there is a 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.

CONCLUSIONS

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

1. The plant has performed extremely well at all power levels including full power. The performance of the reactor is very stable.

2. 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 sodium cooled reactor free from these stresses.

3. With the exception of the boiling toluene cold traps, all of the components developed for the SRE have performed well.

4. For large sodium systems free flow through $1\frac{1}{2}$ in. and smaller lines is determined by the sodium oxide precipitation temperature as well as by the melting point of sodium.

5. Maintaining the oxygen concentration at or below an indicated level of 10 ppm can easily be accomplished under normal operating conditions.

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

7. 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.

8. The gamma heating of concrete structures greatly accelerates the dehydration rate.

9. If the SRE were treated as a power producing plant (rather than an experimental plant) the initial startup time would amount to two months from the time sodium is circulating in the heat transfer loops, including normal startup testing.

10. Reasonably good agreement has been obtained between the calculated and measured critical masses, individual worths of core elements and the flux distribution.

11. The slow acting overall isothermal temperature coefficient for the SRE is positive but small. The fast acting metal coefficient is negative and large.

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