NAA-SR-MEMO-9003

SRE SYSTEMS AND COMPONENTS EXPERIENCE

CORE II

By

E. N. PEARSON

Facsimile Price $ 3.60
Microfilm Price $ 1.13

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

Presented at the Sodium Components Development Program Information Meeting
Palo Alto, California; August 20–21, 1963

ATOMICS INTERNATIONAL

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

CONTRACT: AT(11-1)-GEN-8
ISSUED:

DISCLAIMER

CONTENTS

TABLES

FIGURES

FIGURES

SUMMARY

The general experience at the SRE has proved to be a useful tool for the development of sodium-cooled reactor concepts and has specifically provided information and support for the design and operation of the HNPF. The SRE will be converted in the near future to a high-temperature fuel irradiation plant (PEP Program), and it is expected that the experience and knowledge acquired will provide assurance for the successful attainment of the PEP Program objectives.

INTRODUCTION

The operation of the Sodium Reactor Experiment as both a full-scale power plant and a reactor systems experiment has put special demands on the system and its components. High reliability and performance is desired, but by virtue of the experimental nature of the work, this is not always possible.

As a power plant, the SRE since going critical in April of 1957, has generated 26.5×10^6 kwh of electricity. This was accomplished in ~23,500 reactor operating hours out of a theoretical time of 45,000 hr that was available. For Core II, out of 25,400 hr theoretically available, operation of the reactor was performed for 14,900 hr. Currently the reactor is operating at 12 Mw (60% of full-power) and over the past year the reactor has been at power ~80% of the time.

The operation with Core I was terminated in July of 1959[2] with the intervening period devoted to rehabilitation of the core following fuel element and moderator-can damage[3,4] and needed modifications to the system.[5]

The operation of the SRE Core II can be considered to have commenced with the critical loading of Core II fuel elements (Th-U) in September of 1960.

The SRE has been described elsewhere[1] and only those features pertinent to this paper will be discussed here. The major features of the reactor are shown in Figure 1, and a schematic of the sodium flow path in the primary and secondary loops are shown in Figure 2.

The components to be discussed in this report cover the following areas:

1) Fuel Elements
2) Moderator Cans
3) Hot Traps
4) Cold Traps
5) Control Elements
6) Pumps
7) Main Primary Heat Exchanger

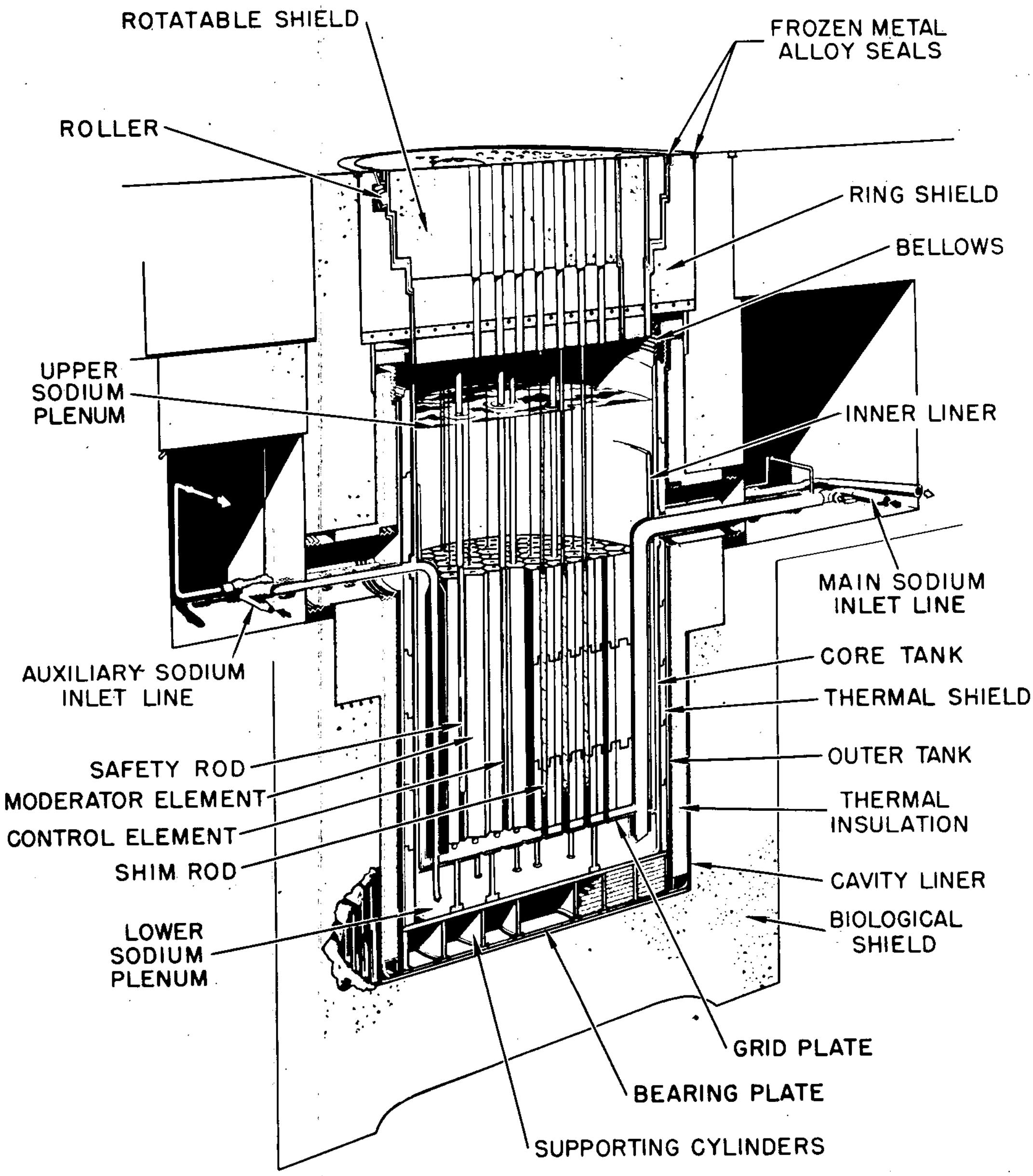


Figure 1. Cutaway View of SRE Reactor

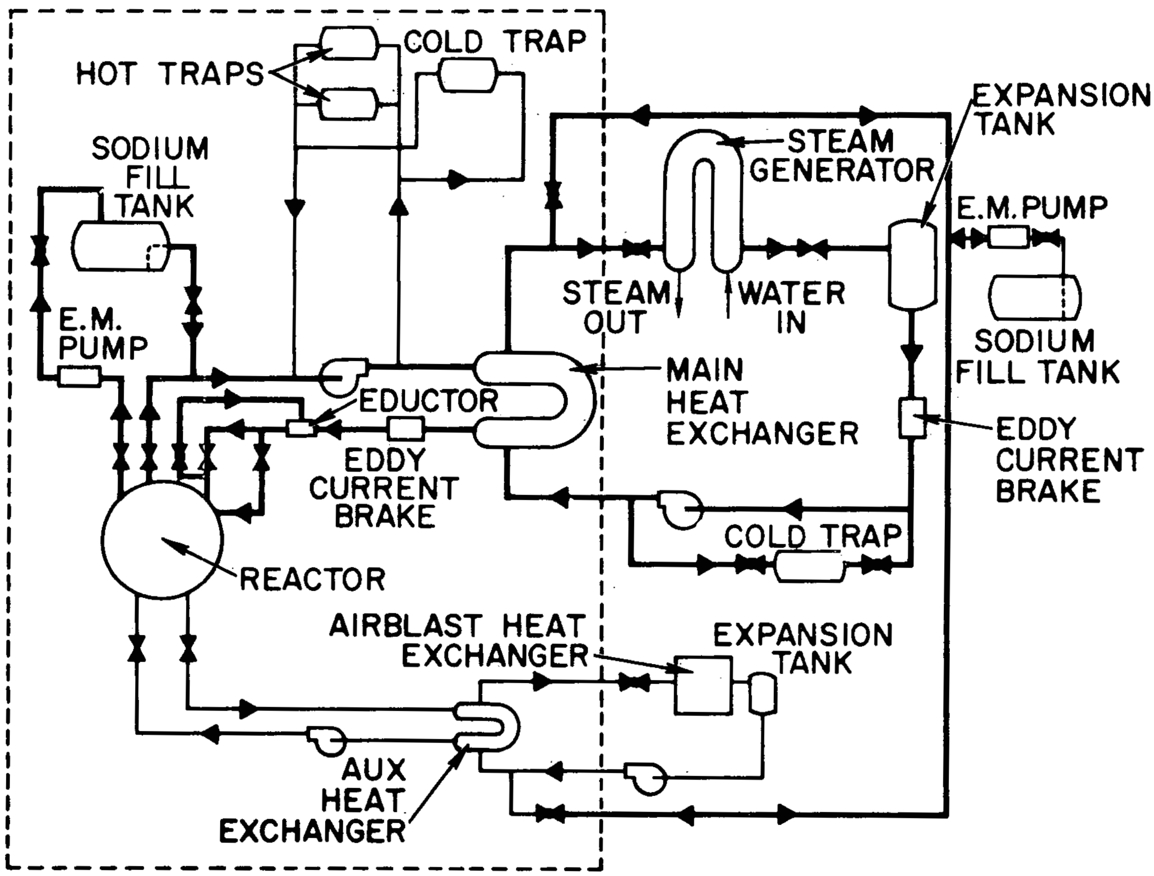


Figure 2. SRE Schematic Flow

8) Valves and Sodium Piping

9) Instrumentation

10) Miscellaneous

A. FUEL ELEMENTS

Core II fuel elements consist of a 0.75 in. in diameter cylindrical slug of Th-U fuel 7.6% enriched. Fuel slugs are clad in 0.010-in.-thick stainless steel tube 0.79-in. OD, with an 0.010-in. -thick NaK bond. Five fuel rods are assembled into a fuel cluster with a solid center rod as a structural member, see Figure 3. Core I fuel element differed not only in fuel (U metal, 2-1/2% enriched), but also in the number of rods,[7] and in the clearance within the moderator can process tube. This radial clearance was increased from a nominal 0.029 to 0.162 in. in order to reduce the possibility of scoring the zirconium process tube and of retaining foreign material on wire wraps.

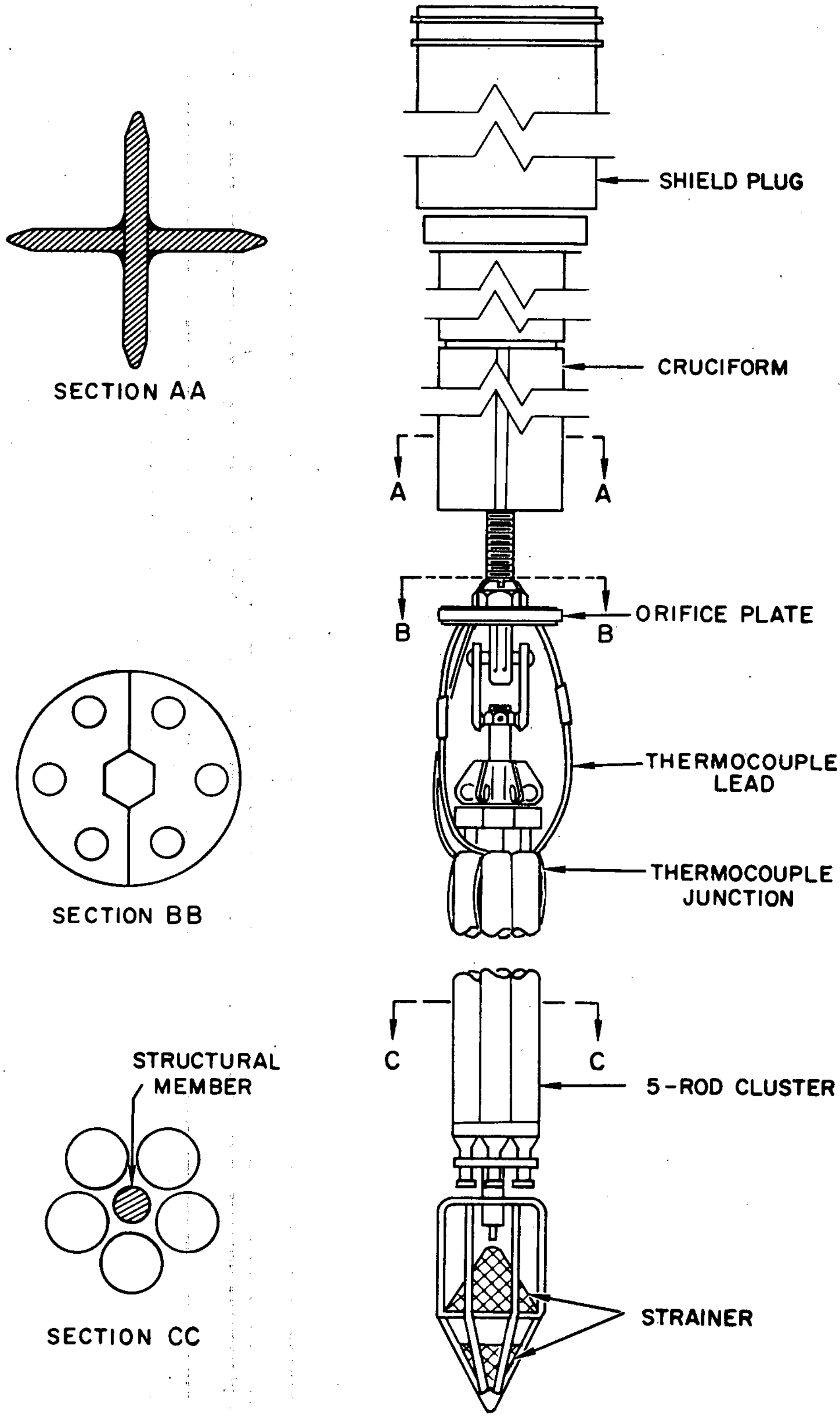


Figure 3. SRE Fuel Element Configuration

Initial operation with Core II showed that the critical mass, temperature coefficient, flux distribution, relative element worths, and zero-power coefficient were as expected and agreed with calculations. In December of 1960 the reactor power was increased and the reactor became difficult to control at ~2 Mw power. Measurement of the power coefficient showed a value of +9.0¢/Mw, rather than the calculated -5.0¢/Mw.[6] Special reactor tests and analysis indicated fuel rod bowing to be the cause. In order to clearly demonstrate this a specially designed element,[7] as shown in Figure 4, was installed in the reactor and by bowing the rods outward it was possible to add reactivity and cause the positive power coefficient.[8]

A modification to the fuel was designed and developed which resulted in restraint of the fuel rods. This modification took the form of stainless wires placed around the complete fuel rod assembly and an added wire in the inner region as shown in Figure 5. There was some concern that the addition of the inner wire wrap to the inner passage would result in undesirable temperature differences across the fuel rod. Preliminary calculations indicated this was not of major consequence.

In November of 1961, after the completion of wire wrapping of the total core (40 elements), the power coefficient was ~-2.0¢/Mw. This provided smooth and controllable reactor operation. As power was increased up to limits of 5 Mwt, several specially instrumented fuel elements indicated a temperature difference across the diameter of a fuel rod which could be as high as 250°F at full reactor power of 20 Mwt. A simple analytical model was developed to explain this difference and the results of the analysis and measurements are shown in Figure 6. This mismatch in temperature distribution was considered to be due to reduced flow in the inner subchannel of the fuel element, see Figure 5.

Approval to operate at power levels above 5 Mwt, was given in September of 1962. It was indicated that a new phenomenon was being seen in the temperature oscillations measured at the exit of the fuel-element process channels and also in-fuel thermocouples located at various elevations within the fuel.

Initial indications at lower powers (core ΔT's of <80°F) were that temperature fluctuations of <±5°F were present; however, this did not seem significant. But as the power and the temperature rise across the reactor was increased, the amplitude of the temperature oscillations increased to ±20°F at frequencies in

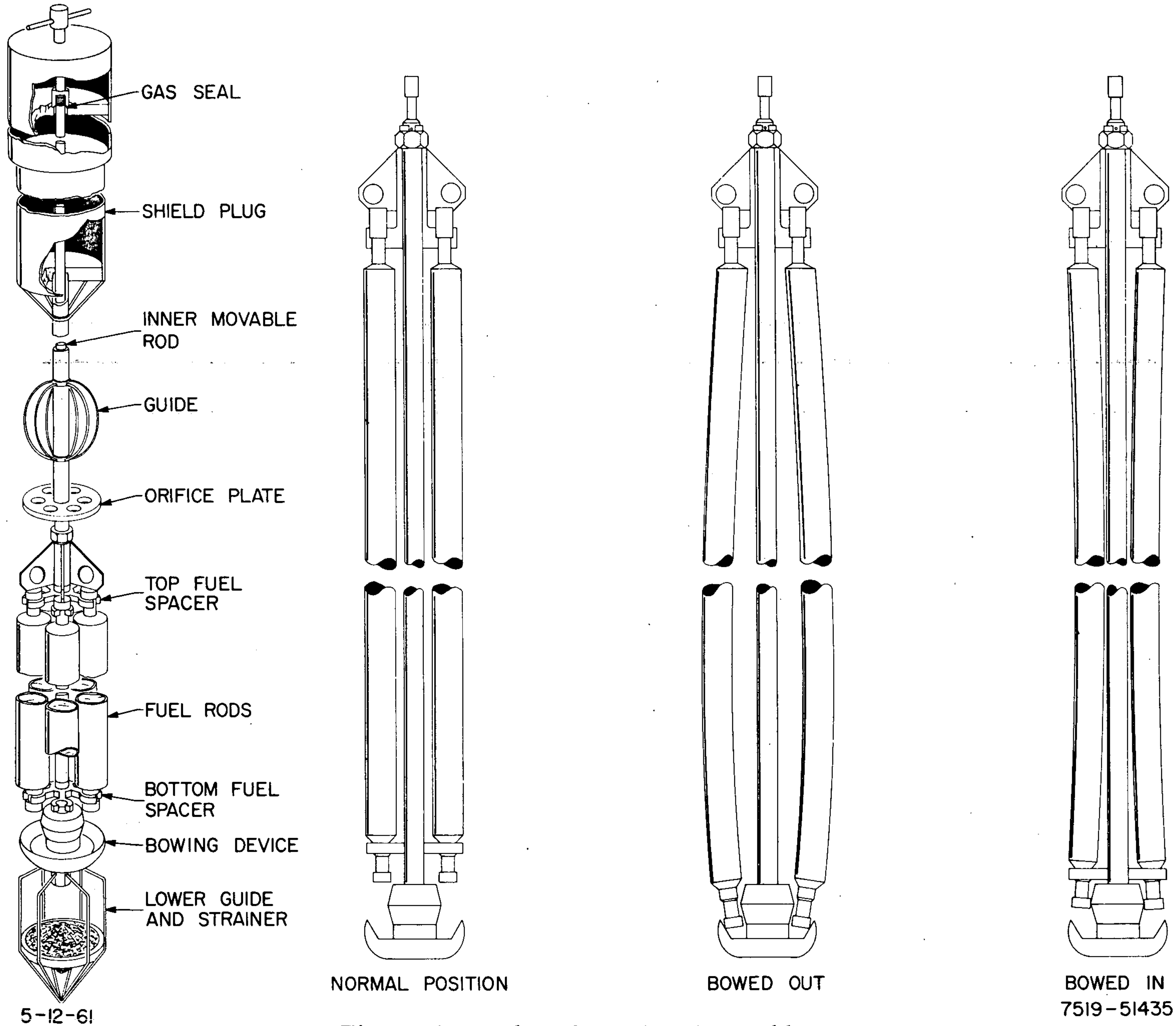


Figure 4. Fuel Rod Bowing Assembly

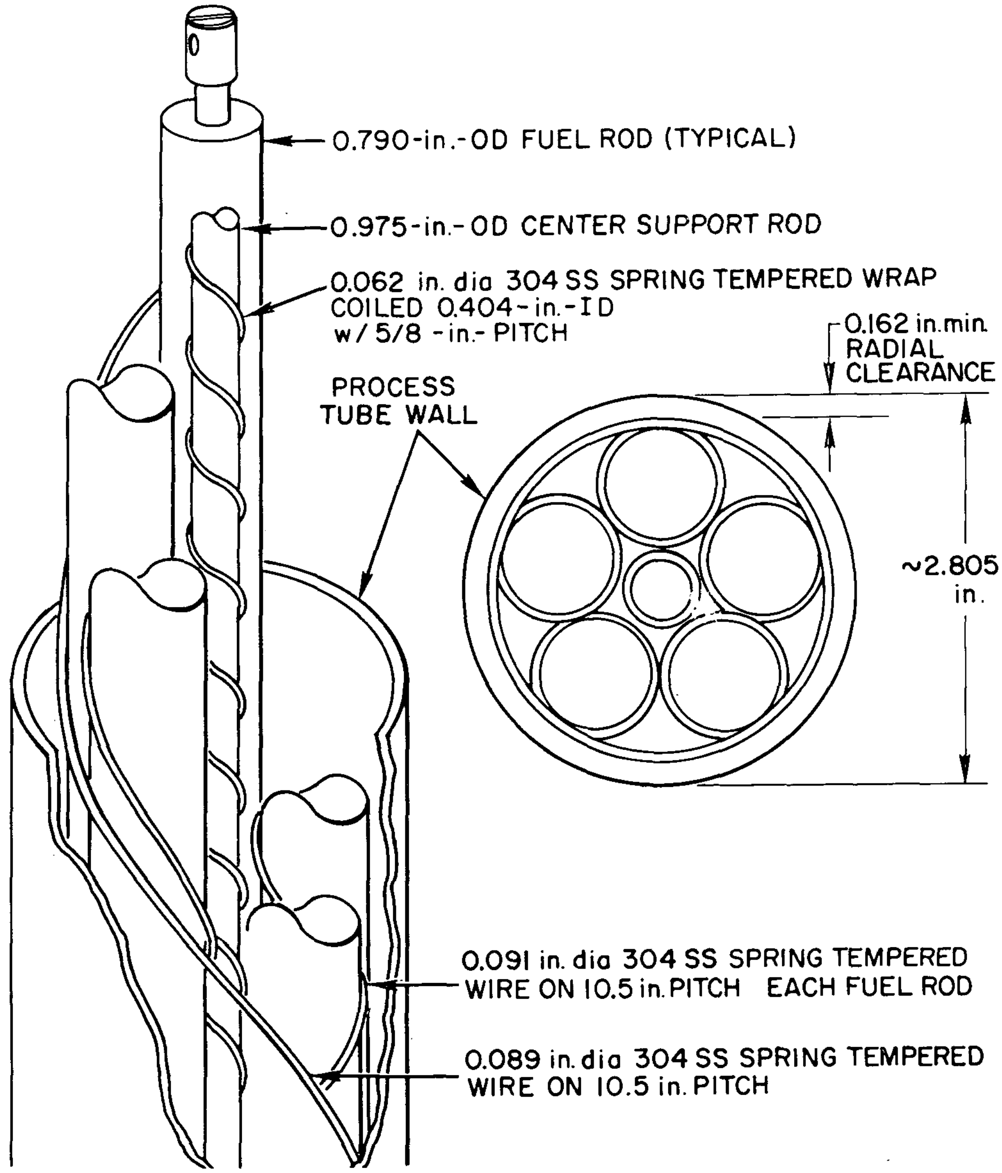


Figure 5. Wire-Wrapped Fuel Cluster in a Fuel Channel

the order of 0.2 to 0.5 cps. The occurrence of these oscillating temperatures has resulted in reevaluation of thermal fatigue resistance of the fuel element stainless steel cladding and also the zirconium moderator process tubes in order to prevent inadvertent fatiguing of these components. At the present time the oscillating temperature limitations of ±50° F are being applied to the fuel element cladding and ±35° F to the zirconium process tube. Of course the ability to measure such temperature oscillations is dependent upon having thermocouples with fast time constants (0.3 sec), which is the case with the thermocouples in the SRE elements.

Several experimental changes and fixes have been applied to the fuel element with very little success. One change has been performed on six elements with complete reduction in the amplitude of the temperature oscillation as shown in

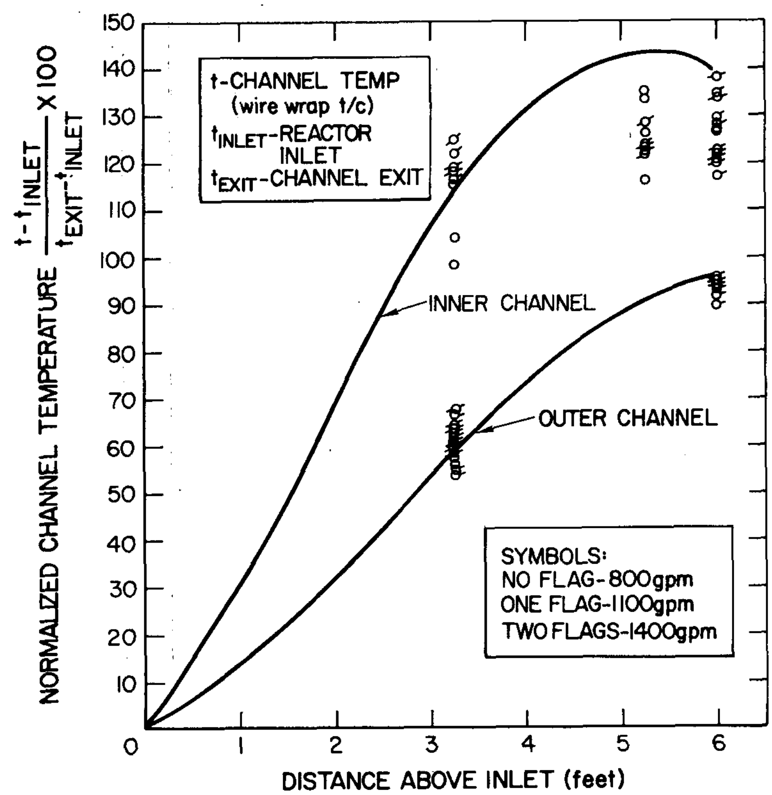


Figure 6. Normalized Channel Temperature Distribution, With W. W. Elements vs Element Height

Figure 7. This figure shows the replacement of the inner wire wrap with a "star-shaped spreader." This spreader bows out the individual fuel rods a maximum of 1/8 in. at the center of the fuel rods. It is thought that this improvement results from a combination of three effects:

1) A "stiffening" of the fuel cluster due to the bowing of the individual fuel element.

2) Increase in clearance between the fuel rods, thus permitting a clearer passage of fluid motion between inner and outer passage.

3) The reduced clearance between the fuel element and process tube wall.

Evidence of the presence of temperature oscillation is also shown by the process period recorder. Examination of Figure 8 shows that during the period of ~30 min, while the reactor temperature was increased from 160 to 210°F, the amplitude increased on the period recorder as well as the temperature oscillations.

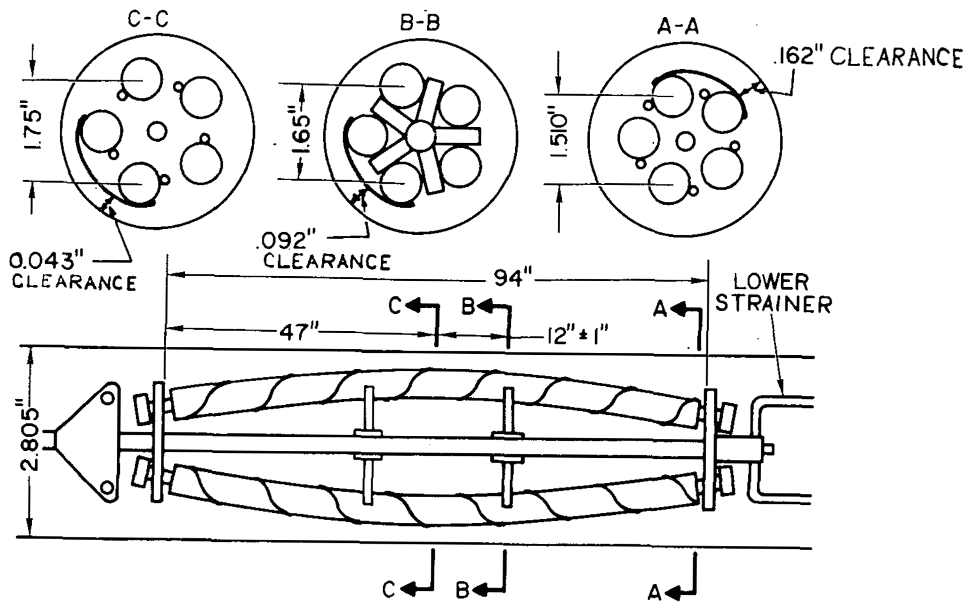


Figure 7. Schematic of SRE Element With "Spreaders"

A great deal of analytical and experimental work must be done before a clear understanding of the phenomenon can be presented.

Orificing of the elements is necessary to maintain constant exit temperatures within ±25°F. (At design temperature rise this is an individual element temperature rise of ±5%.) Maintaining moderator can head thermal stresses at a low level (see Figure 9) has been accomplished successfully. Relocating the orifice from the bottom of the fuel element to the top (the difference between Core I and Core II element orifice) did not affect several other variables that determine the ability to reorifice and predict the resulting temperature.

1) Changing the orifice (and the flow rate) in one channel which also changes the flow rate (in an opposite direction) in the six surrounding channels;

2) The effect of shim rod position on the perturbation of local flux levels and therefore power production in a given element;

3) The effect of suspected pressure gradients in the lower plenum and geometrical deviations that materially affect the flow at specific places within the core;

4) The influence of the cleanliness of the element. Under present conditions the accumulation of foreign material in the lower strainer baskets reduces the flow ~1% per month as indicated by temperature deviations before and after washing.

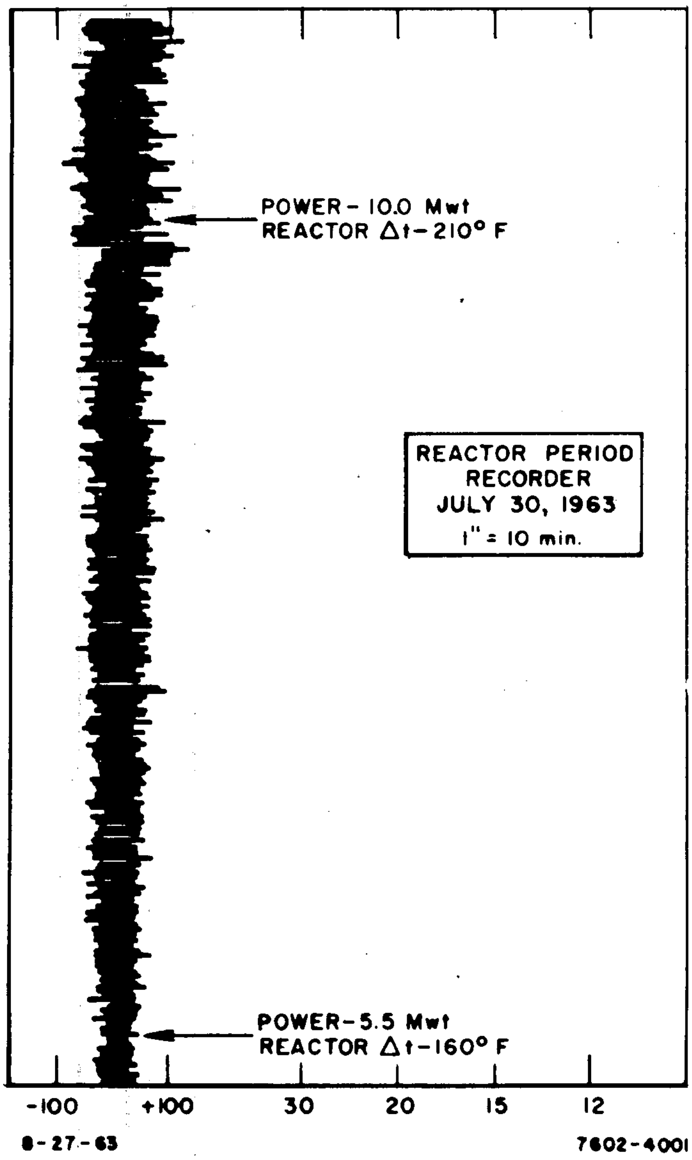
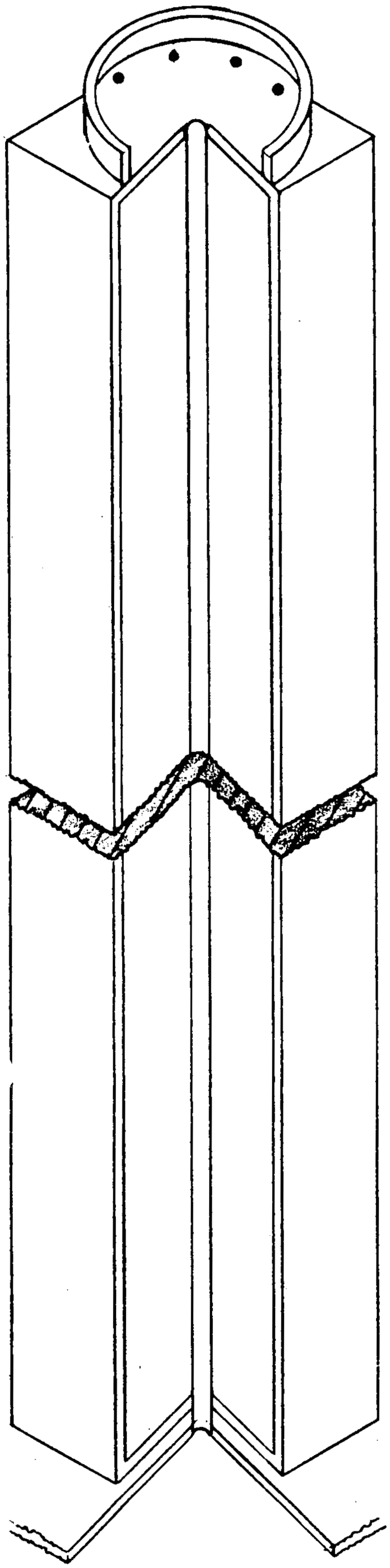
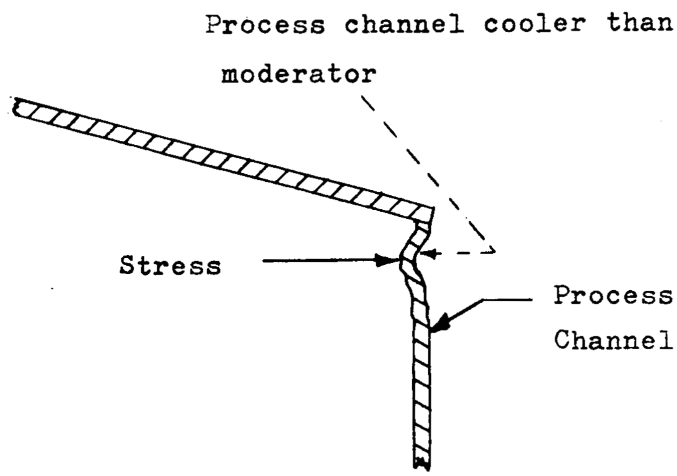


Figure 8. Variation of Reactor Period With Power





Where the stress is $\sigma = 128\Delta t$

Δt is the temperature difference between fuel channel exit and moderator exit

REFERENCE: W.F. Anderson and D.F. Casey, Re-evaluation of SRE moderator cans. Unpublished data.

Figure 9. Moderator Can Stress

The SRE has several experimental elements in the core. The current exposure of these elements compared to the standard Th-U element is as follows:

	Average	Peak
Th-U Core II	2386 Mwd/Tonne	3340 Mwd/Tonne
U^{10}-Mo	1910 Mwd/Tonne	2670 Mwd/Tonne
4 rod UC	400 Mwd/Tonne	555 Mwd/Tonne
5 rod UC	400 Mwd/Tonne	555 Mwd/Tonne

It is interesting that the current burnup on the Core II fuel is twice what had been achieved on Core I.

B. MODERATOR CANS

The use of zirconium and zirconium-2 clad moderator cans in the operation of Core II has proven to be satisfactory. A typical can is illustrated in Figure 10.

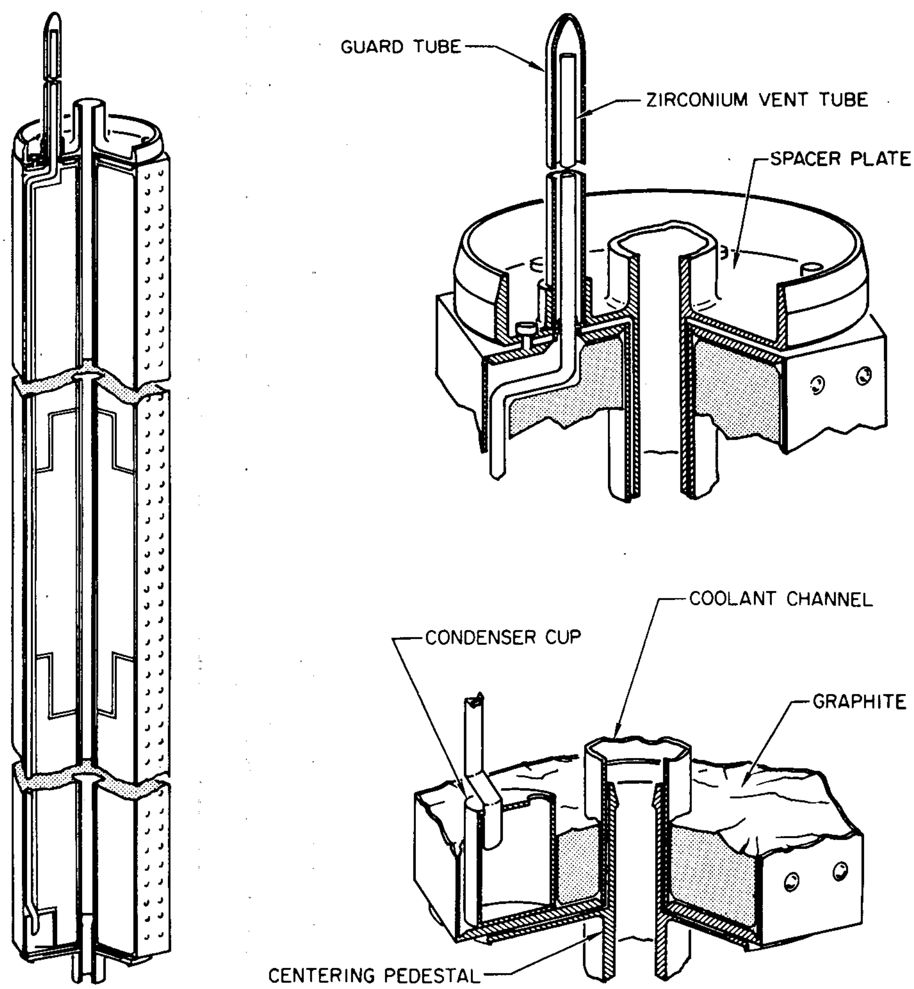


Figure 10. SRE Moderator Assembly

A review of the history of the SRE moderator cans reveals that 12 of 13 zirconium moderator cans removed during the initial core recovery operations during early 1960[4] contained portions of fuel elements that had been separated during the removal of the remainder of the elements. The thirteenth had a split at the seam between the top head and a corner-channel side panel. In August 1960, 3 additional moderator cans were found to be broken following a sodium circulation test and probing of moderator can heights. Two of these were examined closely and found to have broken "pumpout" tubes. These tubes are unprotected and it was assumed they were damaged during the recovery program. Subsequent reprobing and sodium circulation tests showed all the moderator cans to be sound.

On June of 1961, two additional zirconium cans were removed because they were found to be high and therefore suspected to be damaged. Examination revealed that one had a fracture at the top, while the second, which was a zirconium clad reflector type, had no visible defects. This was possibly due to the fact that the reflector can did not have a central process tube.

At present, 18 of the original 86 zirconium-clad moderator and reflector cans have been replaced with zirconium-2 cladding, after 25,000 hr of reactor operation, generating 5,000 Mw days of energy. This effort of moderator can replacement has demonstrated and proved that the design features of the SRE provide for direct, simple, and safe maintenance of the core.

C. HOT TRAPS

The operation of the hot traps as a mechanism for removing carbon from the primary system sodium has proceeded smoothly since November of 1961.[10]

There are two hot traps installed in parallel in the SRE primary systems: hot trap-A and -B. The flow through the hot trap (Figure 2) shows the supply to the hot trap coming from the discharge of the main primary pump and sodium returning to the primary system on the suction side of the pump. The nominal flow rate through the hot trap is $\sim$10 gpm or $<$1% of the main primary rate.

The hot traps are stainless steel tanks $\sim$80 in. high constructed of 16 in. Schedule 40 pipe. The carbon gettering material is Type 304 stainless steel strip assembled with alternating corrugated strips, to form a coil 15 in. in diameter and 3 in. high, see Figure 11.

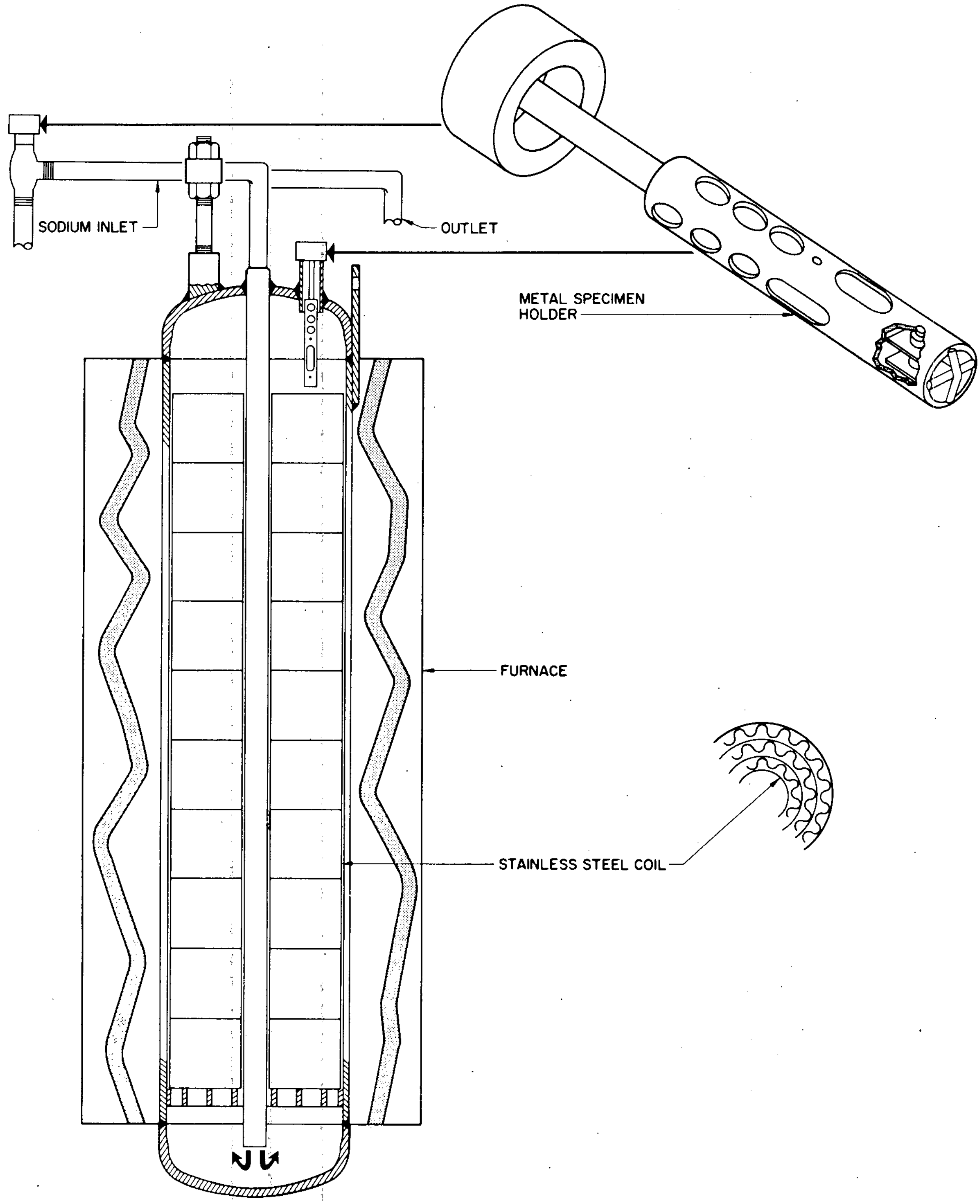


Figure 11. Hot Trap

Initial hot traps have foil material 0.004-in.-thick, but subsequent traps have had foil material 0.010 in.-mil-thick, the amount of stainless steel being increased from ~500 to 1140 lb.

Approximately six different hot traps have been operated in the SRE successfully reducing the carburizing potential of the sodium, see Figure 12. Table I presents data on the operation of only five traps. The missing hot trap A2 developed a leak around a sample holder, was removed and then replaced with hot trap A3.

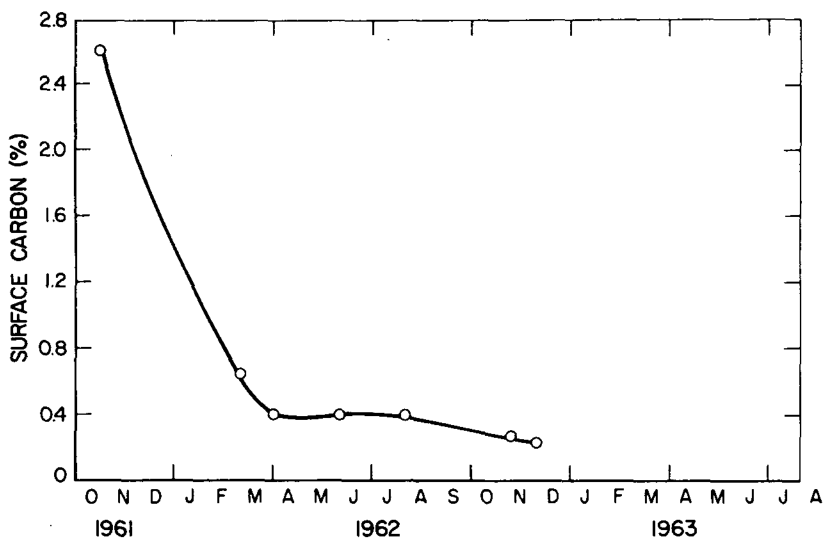


Figure 12. SRE Carburizing Potential History at 1200° F

The hot trap inlet and outlet tab holders are removed and replaced with the hot trap installed in the system at the completion of each hot trapping run. This penetration of the sodium lines has been done four separate times (excluding the five expended hot trap replacements) without difficulty in subsequent operation at temperatures >1200° F.

The heaters for these traps consist of heater elements in a ceramic type furnace, enclosing the vessel itself. On one occasion the heaters shorted out and it was not possible to operate the traps. On two other occasions a short developed in one bank of heaters, but the hot trapping proceeded using the remaining bank.

The operation of this system for more than 3000 hr since 1961 at temperatures >1200° F (see Table I), has shown a high degree of reliability in the structure and components, and provides confidence in operating sodium systems at

TABLE I

SUMMARY OF HOT TRAP OPERATION

Hot Trap	Operating Period	Total Volume Treated (gal)	Oxygen Concentration (ppm)	Surface Carbon in Hot Trap Sample Tabs: Inlet (%)	Surface Carbon in Hot Trap Sample Tabs: Outlet (%)	Carbon Concentration in Sodium: Inlet (ppm)	Carbon Concentration in Sodium: Outlet (ppm)	Carbon Removed (lb)	Equivalent* Hours (1200°F)	Average Hot Trapping Rate (lb/eq hr)
A1	10/18/61-10/24/61	73,000	50	2.5	1.0	39	27.5	5.5	240	0.022
A3	2/23/62-3/16/62	210,000	10	0.654	0.12	25.5	19	9.0	560	0.016
B1-1	3/17/62-4/10/62	160,000	10	0.4	0.065	23	17	5.8	805	0.007
B1-2	5/28/62-6/3/62	47,300	200	0.4	0.06	23	17	1.9	400	0.005
	7/27/62-8/6/62	92,000							503	
B1-3	8/6/62-8/12/62	5,100†	20	0.4	0.06	23	17	4.2	330	0.005
A4-1	11/10/62-11/12/62	30,000	<10	0.27	0.06	21.5	17	0.9	130	0.0069
B2-1	11/21/62-12/3/62	140,000	<10	0.23	0.06	21	17	3.7	410	0.009
A4-2	8/5/63-8/7/63	30,000							72	
B2-2	8/14/63-8/16/63	—	—	No Data Available		—	—	—	50	—

*Hot trap temperatures (T) vary from 1200 to 1300°F. Operating time (t_T) at temperature T is therefore normalized to equivalent hours at 1200°F by the formula:

$$t_{1200} = \frac{D_T}{D_{1200}} \times t_T$$

where: t_{1200} = equivalent time at 1200°F

t_T = actual time at operating temperature T

D_T = diffusion coefficient of carbon in stainless steel at temperature T

D_{1200} = diffusion coefficient of carbon in stainless steel at 1200°F

† This test was run at a flow rate of 0.5 gpm; all other tests at 7 gpm.

Note: Hot trap designation Bk-n, indicates the kth hot trap in the "B" location operated for the nth time.

these temperature levels. Similar experience had been observed during Core I, the hot traps at that time being used for oxygen removed, with zirconium as the gettering material, for an approximately similar length of time, at temperatures of 1200°F.

D. COLD TRAPS

In the SRE there are two circulating gas-cooled cold traps. These cold traps are in the primary system and in the main secondary system. The cold trap for the independent auxiliary secondary is a diffusion cold trap attached to the auxiliary secondary expansion tank.

The installation of the cold trap is similar to the installation of the hot traps discussed previously (Figure 2). Sodium enters the cold trap from the discharge of the main pump (similar flow paths for both main primary and main secondary), and returns to the suction side of the pump.

The design features of the primary and secondary cold traps are identical; the traps are constructed of 18-in. standard weight carbon steel pipe with 18-in. pipe caps as the heads, (see Figure 13).

Incoming sodium is cooled in the internal economizer as it flows down the annulus. The annulus is packed with stainless steel mesh (open volume 5 to 6 ft^3) to provide sites for the crystallization of sodium oxide, and to filter out sodium oxide precipitates from the sodium stream when the temperature is lowered below the saturation temperature. As mentioned previously, the traps are gas-cooled; in the primary system this gas is the gallery nitrogen atmosphere; in the secondary system it is air. Both traps are provided with a 10-hp blower.

Several difficulties were apparent with initial operation of these traps. The tendency was initially to reduce the internal temperature of the cold trap to ~250°F, regardless of the indicated plugging temperature. This resulted in shortening the life of the cold trap in that with high plugging temperatures, the oxide would deposit in the upper portion of the cold trap (because of counter-current cooling gas flow), thus effectively stopping cold trap flow. This was further complicated when the flow indicator for the cold trap did not go to zero, because the provision of a 1/8-in.-diameter venting hole in the internal economizer permitted bypassing the internals of the cold trap, thus giving an indication of flow through the cold trap while performing no cold-trapping. These

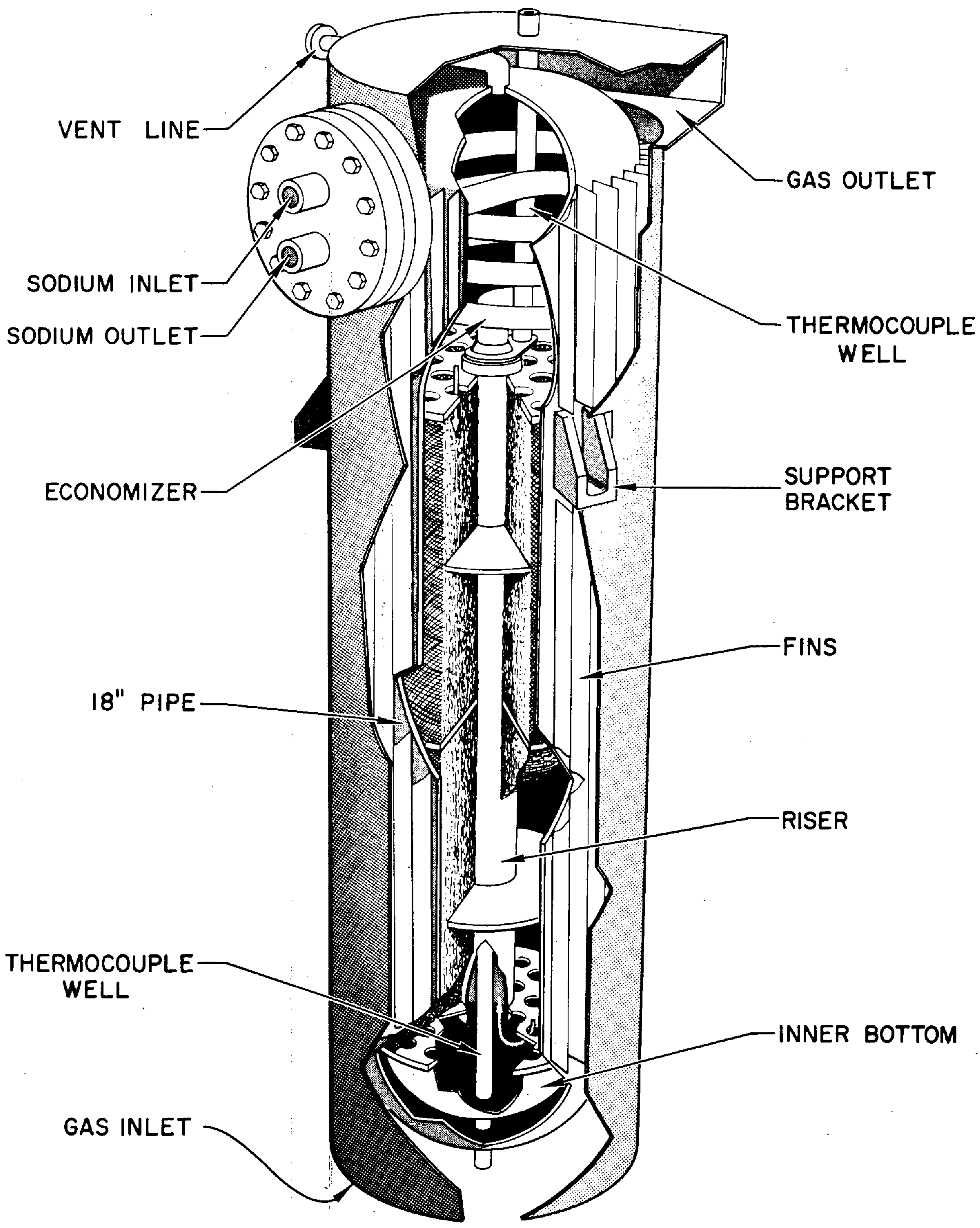


Figure 13. Circulating Cold Trap

problems have been solved by eliminating this hole and also by the operating procedure requirement that while cold trapping, the internal temperature of the trap be no greater than 50°F less than indicated plugging temperature.

The cold trap serves not only to reduce the oxide content of the sodium, but indications at the SRE are that it is also removing fission products. Plots of radiation at the surface of the cold trap (no contribution from Na^{24} activity), Figure 14, indicate an exponential rise with quantity of sodium processed through the cold trap. Furthermore, it is seen that the maximum radiation level in the cold trap has decreased with time during the past three years, indicating that the system is becoming progressively cleaner.

The operation of the MP cold trap at the SRE is dependent on the penetration (removal and repair of equipment, fuel element handling, etc.) into the primary system. Based on such activities and on variations in plugging temperature, it is possible to estimate the amount of sodium oxide retained in the trap. During the period November 1961 to August 1962, it was calculated that ~82 lb of oxide had been cold-trapped. With a capacity of 180 lb of oxide (based on a measured density of 0.36 lb/ft^3),[11] which is 1/4 of the crystalline density, it was felt that the trap was expended. Generally, the removal rate of sodium oxide is 2 to 6 lb Na_2O/day with the maximum amount of 10μ of oxide/day.

The decision for removal of the cold trap in the primary system has been made usually on the basis of its radiation level. Whereas over the past four years the SRE primary sodium system (50,000 lb of Na) has consumed seven cold traps, the main secondary system (8000 lb of Na) has used only two traps over the same period. Of course the latter system is not opened in the degree and frequency that the primary system is, and also it is not a source of radiation.

The SRE cold traps have demonstrated the ability to reduce the oxide content of the sodium in an efficient manner and have provided assurance of the performance of similar cold traps at HNPF.

E. CONTROL ELEMENTS

The control elements for the SRE consist of four control rods and four safety rods. Both types of rods are constructed similarly, i.e., composed of stacked rings of a 2% boron-nickel alloy suspended on a pull tube. The entire assembly operation is performed in helium-filled stainless-steel thimbles.

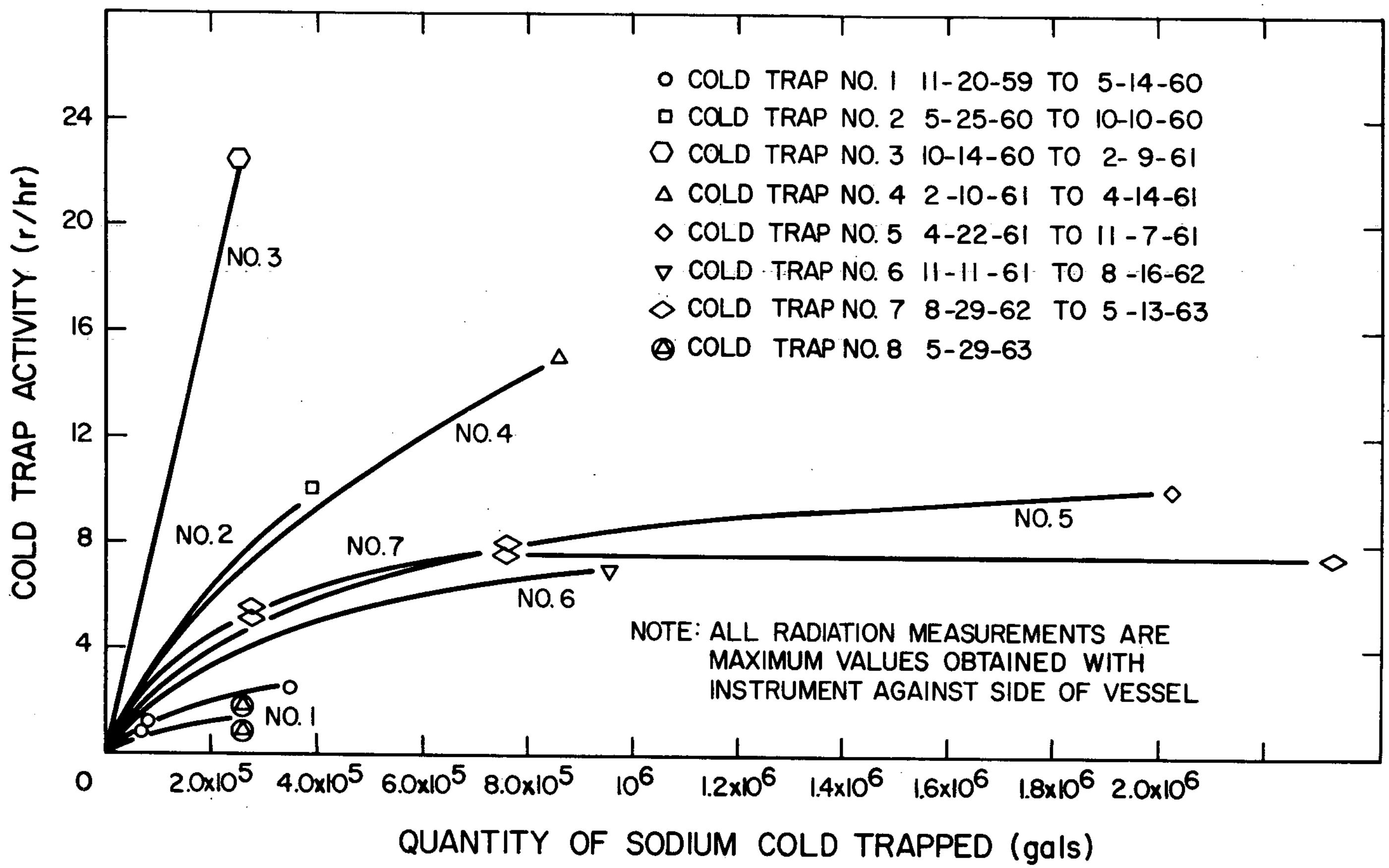


Figure 14. SRE Main Primary Cold Trap

The four control rods are usually partially inserted in the core while the reactor is operating. Following Core I (core exposure 2400 Mwd), the control rods were inspected in the SRE hot cell. The maximum diameter change was 0.006 in., with no evidence of wear. Continued operation with the same elements and drive mechanisms has been good. Maintenance on the drives consists of motor-brake replacements and redesign of an understrength limit-switch bracket.

The safety elements are completely withdrawn during reactor operation. These elements have had no difficulty other than that caused by recurring gas leaks due to a poor sealing design when the drive mechanism is installed on the rod. Several redesigns have been attempted but none were completely successful. An automatic device has been installed to measure the drop time and snubbing time to detect evidence of wear and/or sticking. Prestartup measurements recorded during the past year show that the free-fall time of 0.57 ± 0.03 sec and snubbing time of 0.6 ± 0.2 sec for a total drop time of 1.17 ± 0.23 sec have not changed over the past seven years.

F. PUMPS

The four sodium pumps in the SRE are of the freeze-seal type in which the liquid sodium is permitted to enter a narrow annulus between the shaft and housing to be frozen there by an auxiliary coolant (NaK). The sodium seal is continuously sheared by rotating of the shaft. A similar freeze seal is used between the pump case and housing. Figure 15 is a picture of the main primary and main secondary pumps. The primary and secondary pumps are of similar design except that the primary housing is extended vertically to permit operation of the drive motor above the shielded gallery.

The freeze seals of the pumps have been sources of operational difficulty due to shaft binding, sodium extrusion, and gas in-leakage. Initially, operation procedures required the shaft seal temperature to be <150°F. However, since it was found that a higher temperature provides smoother pump operation, this limit has been extended to ~175°F.

The reliability of the pumps has not been satisfactory. During the past six months the pumps have been 100% reliable, i.e., no pumps had to be removed for repair or maintenance. During the past year the secondary pumps have still been 100% reliable, but the primary pumps have had to be removed approximately five times. These latter events occurred prior to the modifications

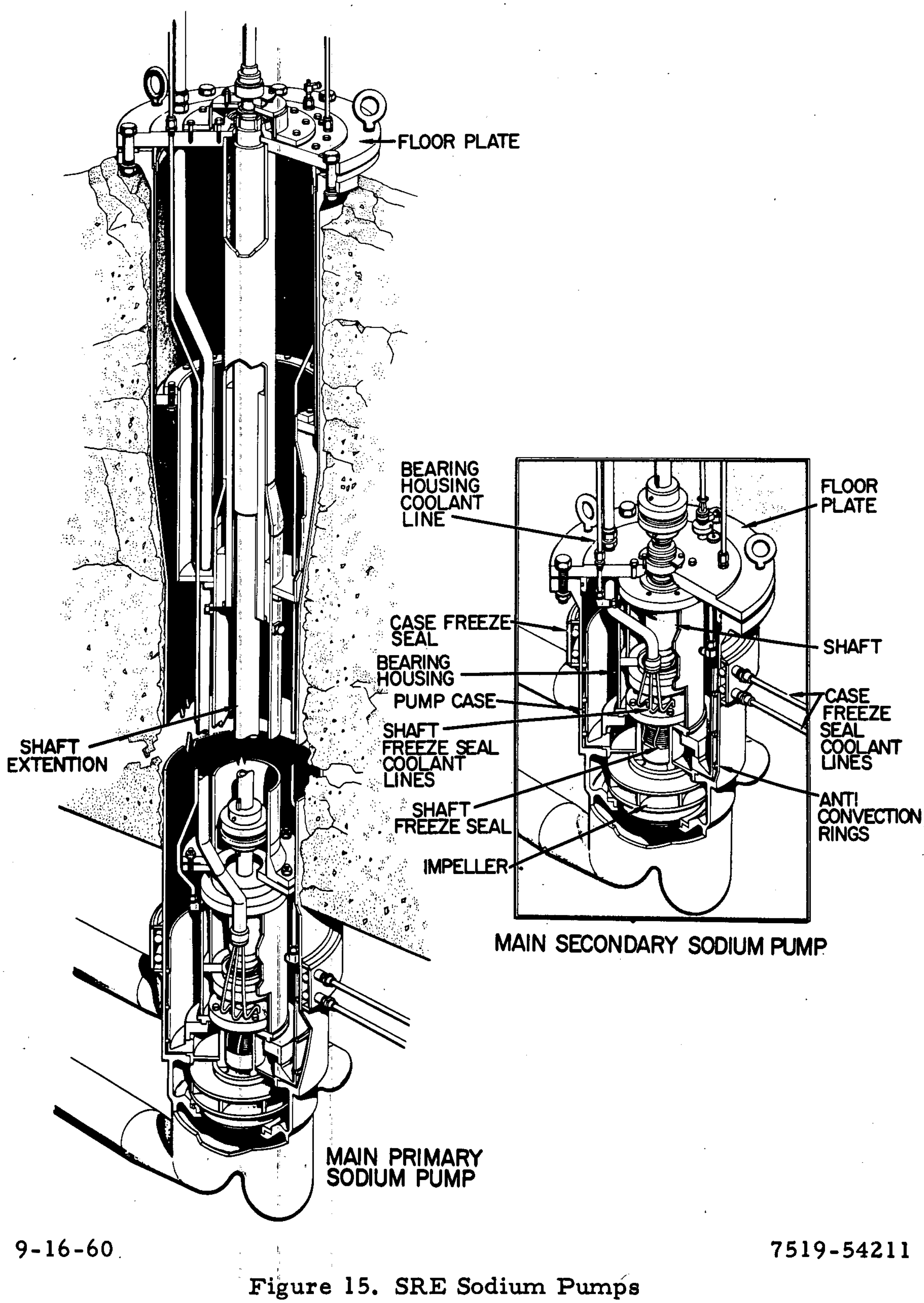


Figure 15. SRE Sodium Pumps

of the shaft seals on these pumps, therefore it is not definite that this problem is solved.

It has become possible to operate the auxiliary-secondary system on free convection without depending on the pump in this system. Experimental and analytical information presented in Reference 12 indicates that following the accidental loss of the MP and MS systems and scram from 20 Mwt, the increase in the average temperature of the core would be ~150°F, which is not excessive.

The difficulties with the SRE freeze pumps led to the adoption of free-surface pumps for HNPF.

G. MAIN PRIMARY HEAT EXCHANGER

The main primary heat exchanger has been in service since 1957 and has been subjected to >25,000 hr of sodium service during operation while handling 5000 Mwd of heat (see Figure 16).

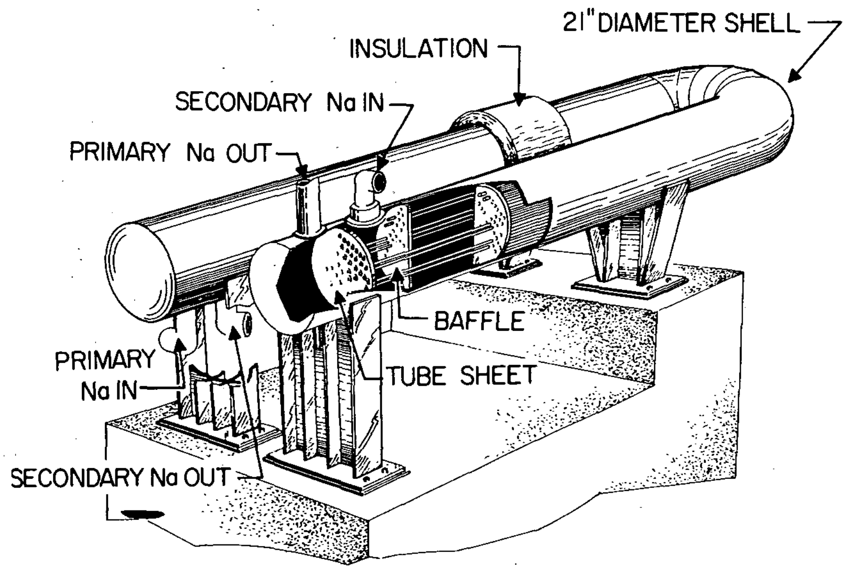


Figure 16. SRE Main Intermediate Heat Exchanger

The thermal performance of the SRE heat exchanger[13] resulted in two particular problems: (1) temperature stratification in the shellside of the exchanger, causing reverse flow in the tubes, at low flows (<100 gpm) following a scram, and (2) the log mean temperature difference was about 50% more than the design value (90 vs 60°F) because of poor baffling particularly in the bend area of the U tubes.

Neither of these conditions has been corrected with this unit. Continuous monitoring of the performance of this unit has shown that the overall heat trans fer coefficient has returned to its initially clean value, measured U of 750 Btu/hr-ft^2-°F vs a calculated 713 Btu/hr-ft^2-°F at an average sodium flow rate of 1300 gpm. This can be compared to data obtained in July 1959 when fouling of the heat exchanger had occurred, with the measured U of 436 vs calculated of 674 (at a flow rate of 1100 gpm) a difference of 35%.

Further evidence of fouling in the IHX is shown in Figure 17. These data were measured along the cold leg of the MIHX and it can be seen that there were large amounts of foreign material deposited in this region. The improvement with time is a consequence of system cleanup due to cold trapping and hot trapping.

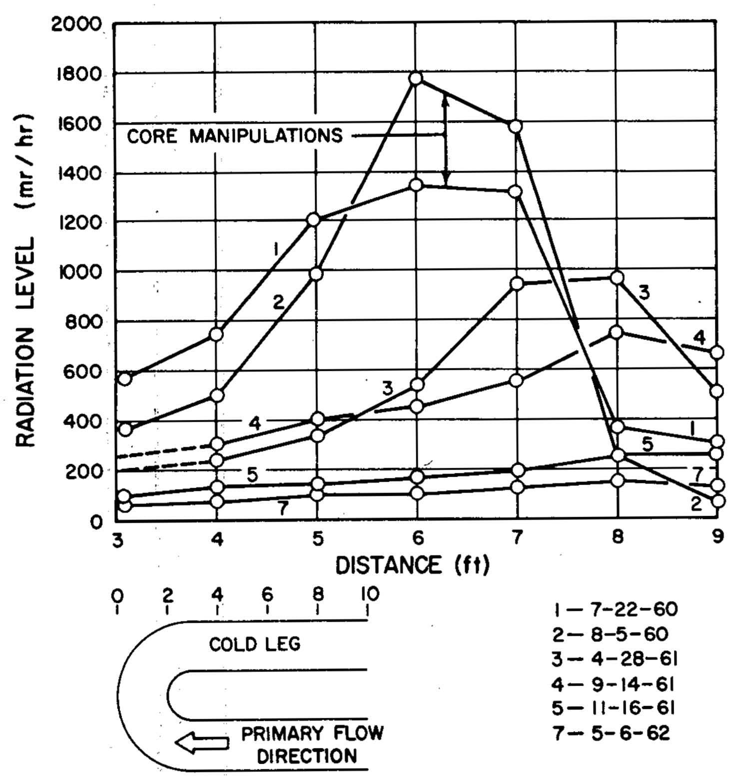


Figure 17. Radiation Intensity Along Cold Leg of MIHX

The difficulties with the SRE horizontally-mounted heat exchanger led to the vertically-mounted unit for HNPF.

H. VALVES AND SODIUM PIPING

As discussed in Reference 3, difficulty with the remaining bellows-seal sodium valves persists, but has diminished. Failures in 1960 were approximately 6 per year; whereas in 1962 there were 2 per year. Despite the use of preheat control curves, bellows failure usually occurs during reheating because of solid-sodium extrusion along the piping. The replacement valve is a freeze-stem type. Failures or sodium extrusion with these valves has not been a problem up to the present time. In the hot trap circuit freeze-stem valves have demonstrated successful operation after 1600 hr with sodium temperature over 1200° F.

At the present time the sodium valves in the MP and AP are freeze-stem valves, while in the primary sodium-service vault the valves are bellows-seal and freeze-stem valves. Flexible-shaft, remote-valve stem continues to be a problem. Difficulty occurs approximately four times a year and the problem is usually solved by going to larger components, and also by redesigning various components.

Eddy-current brakes control the post-scram convective flow (Reference 14). The brakes have operated faultlessly, but the automatic controls of this system have proved to be unreliable; they have been eliminated and the control system simplified. The modifications consisted of replacing the automatic circuitry with a simple control rheostat for the brakes. Upon scram initiation, the brakes come on automatically to a preset value and further adjustment of the flow for the desired temperature change of 60° F/hr is accomplished by manual adjustment of the brake current.

I. MISCELLANEOUS

A continuing problem has been seemingly random period spikes occurring every 3 to 7 days of reactor operation. These spikes, as seen on the operations process recorder, are usually 10 or 20 sec and the duration (indicated on special experimental recorders) is ~1 to 2 sec.

Initially it was felt that gas was being introduced into the primary system through the sodium service system. Modifications to various sections of the sodium service system were designed and installed to eliminate this possibility.

Subsequent experience has indicated these modifications did not completely eliminate the randomly occurring period spikes. Data have shown that these period spikes can be attributed to the operation of a valve (V-104A), ~50% of the time, dislodging gas into the sodium stream. The operation of this valve does not guarantee a period spike 50% of the time, but rather 50% of the period spikes can be attributed to operation of this valve.

An analysis of the reactor kinetics, using an analog computer supports the idea of gas passage through the core. The input was a positive reactivity addition at a rate proportional to the axial statistical weight and the average fluid velocity. This simulated the motion of a fixed size bubble through the core. The calculated data are very similar to changes seen on process instrumentation during such events, i. e., recorded periods, temperature increases and power increases.

Calculations made for full-power and using reactivity inputs three times larger than experienced (50¢ vs 15¢) indicate fuel temperature rise of only 60° F. This rise is not detrimental to the core.

This problem will be resolved as other more vexing areas are solved, i. e., temperature oscillations and carburizing potential.

By removing the temperature restrictions and returning to more normal operating conditions, it will be possible to operate at lower flow rates and thus reduce the potential for gas entrainment as well as decrease the number of occurrences of period spikes.

REFERENCES

1. Sodium Graphite Reactors by C. Starr and R. W. Dickinson
2. NAA-SR-4488, SRE Fuel Element Damage by A. A. Jarrett et al.
3. Experience With SRE by L. E. Glasgow, Nucleonics, Vol 20, No. 4, p 61-65
4. NAA-SR-6369, SRE Core Recovery Program by W. J. Freede
5. NAA-SR-5348, Design Modification to the SRE During FY 1960 by G. E. Deegan et al.
6. Analysis of the Sodium Reactor Experiment Prompt Power Coefficient by C. W. Griffin, Nuclear Science and Engineering, p 114, 304-311, 1962
7. NAA-SR-6878, Fuel Rod Bowing in the SRE by H. F. Donohue and R. W. Keaten
8. NAA-SR-7705, Detecting and Eliminating Fuel Rod Bowing in the SRE by C. W. Griffin, R. W. Keaten, and R. W. Woodruff
9. NAA-SR-6879, Stabilizing SRE Fuel Elements by H. F. Donohue and H. A. Vislay
10. NAA-SR-7804, Relationship of Carburizing Potential to Operating Temperature Limitations in SRE by E. N. Pearson and D. I. Sinizer
11. NAA-SR-3638, Control of Oxygen Concentration in a Large Sodium System by R. B. Hinze
12. Modification of the SRE Auxiliary Heat Transfer System, unpublished AI data and analysis, by M. D. Dermer, G. B. Kruger, and C. L. Peckinpaugh
13. NAA-SR-3775, Thermal Performance of the SRE Main Intermediate Heat Exchanger by K. W. Foster
14. SRE Decay – Heat Problem Solved by Eddy Current Brake by W. S. DeBear, Nucleonics, Vol 17, No. 6, p 108-113, June 1959