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MEASUREMENT OF SOME REACTIVITY EFFECTS IN THE ENRICO FERMI REACTOR

G. T. SEELEY (a)
G. L. BALL
C. E. BRANYAN
R. E. HORNE
R. E. MUELLER
B. M. SEGAL (b)
H. A. WILBER (c)

(a) Purdue University, Lafayette, Indiana
(b) Stanford University, Palo Alto, California
(c) Power Reactor Development Company

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ABSTRACT

Various nuclear tests are described which followed initial criticality of the Enrico Fermi fast breeder reactor. Measurements were made of the reactivity effects of the permanent neutron source, retractable neutron source, temporary instrument thimble, variation in flow rate of the primary coolant, and variation in pressure of cover gas in the primary system.

Each test is reported separately and includes tabulated and graphical test data and discussions of purpose, apparatus and equipment, reactor plant conditions, measurements, experimental results and analysis, and conclusions.

FOREWORD

This is one of a series of reports on the low-power (up to 1 Mwt) and high-power (up to 200 Mwt) nuclear tests of the Enrico Fermi fast breeder reactor. The Nuclear Test Program is planned, directed, and evaluated by Atomic Power Development Associates, Inc., (APDA). The tests are conducted by Power Reactor Development Company (PRDC), which owns and operates the reactor. The steam generators and electrical generating facilities are owned by The Detroit Edison Company (DECo).

Many individuals have contributed to the nuclear testing of the Enrico Fermi Reactor. Listed below are those, exclusive of the authors, who have made significant contributions to the work described in this report.

Experimental Phases:

E. L. Alexanderson	PRDC
D. Erdman	PRDC
L. A. Haigh	PRDC
W. R. Hill	PRDC

Analytical Phases:

S. K. Kapil	Bhabha Atomic Research Centre, Bombay, India

SUMMARY

Shortly after initial criticality, tests were conducted to determine the reactivity effects of the permanent neutron source, retractable neutron source, temporary instrument thimble, variation of primary coolant flow rate and variation of the cover gas pressure in the primary system.

The source correction factor was found to be 234 watt-cents for the stationary source at the core-blanket interface, and 290 watt-cents for the retractable neutron source in position P03-P00, relative to August 27, 1963, and October 14, 1963, respectively. Power levels in excess of 1 kw were found to be sufficient for measurements requiring no source correction.

The worth of the antimony portion of the retractable source, when completely inserted into the core, was found to be -8.3 cents. The temporary instrument thimble, when completely withdrawn from position P03-P00, was found to have a slight positive reactivity effect of +1.4 cents.

There was no indication of any effects of coolant flow rate or cover gas pressure in the primary system and there was no detection of gas entrainment or structural deformation.

All measurements agreed closely with predicted values, and no safety hazards were presented by any of the effects investigated.

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I. PURPOSE OF TESTS

The purposes of the various tests included in this report are described on an individual test basis in report Sections III, IV, V, and VI. Other phases of each test are correspondingly included in the respective report sections so that each test has the status of a separate report.

II. DESCRIPTION OF THE ENRICO FERMI ATOMIC POWER PLANT

A. GENERAL DESCRIPTION

The Enrico Fermi Atomic Power Plant utilizes a fast breeder reactor, cooled by sodium and operated at essentially atmospheric pressure. The reactor is designed for a maximum capability of 430 Mwt; however, with the initial loading the maximum reactor power is 200 Mwt. The plant is comprised of the Reactor Building, in which are housed the reactor and primary coolant system; the Steam Generator Building; the Control Building; and the Turbine Building. Some distance removed from this building complex there are a Fuel and Repair Building, in which fresh and spent fuel are stored and processed; a Waste Gas Building and an Inert Gas Building, in which waste gas is processed and from which inert gas is supplied and recirculated; a Sodium Service Building, in which sodium is cold trapped and stored; and a Health Physics Building and Chemistry Laboratory.

B. REACTOR VESSEL AND ASSOCIATED STRUCTURES

The reactor vessel and its associated structures are shown in perspective in Fig. 1. The stainless steel reactor vessel is composed of four major parts: (1) lower reactor vessel, (2) transfer rotor container, (3) upper reactor vessel, and (4) rotating shield plug of steel and boronated graphite. The plug supports the control rod drive mechanism, the fuel handling mechanism, the sweep mechanism, and the subassembly holddown mechanism.

The wall of the reactor vessel is shielded with laminated stainless steel plates and solid steel bars, to attenuate the gamma and neutron fluxes incident on the wall. The shield plates also reduce the thermal stress in the wall resulting from changes in sodium temperature. The entire reactor vessel is contained in a carbon steel liner which, although not leaktight, is designed to direct any sodium leakage to detectors in the bottom of the liner. Both the vessel and the liner are enclosed in the primary shield tank. The primary shield, which fills the space between the vessel wall and the shield-tank wall, consists of boronated graphite and plain graphite neutron-shielding material. The primary shield tank is so sized and constructed that an adequate supply of sodium coolant will be maintained in the reactor should leakage develop in the vessel wall.

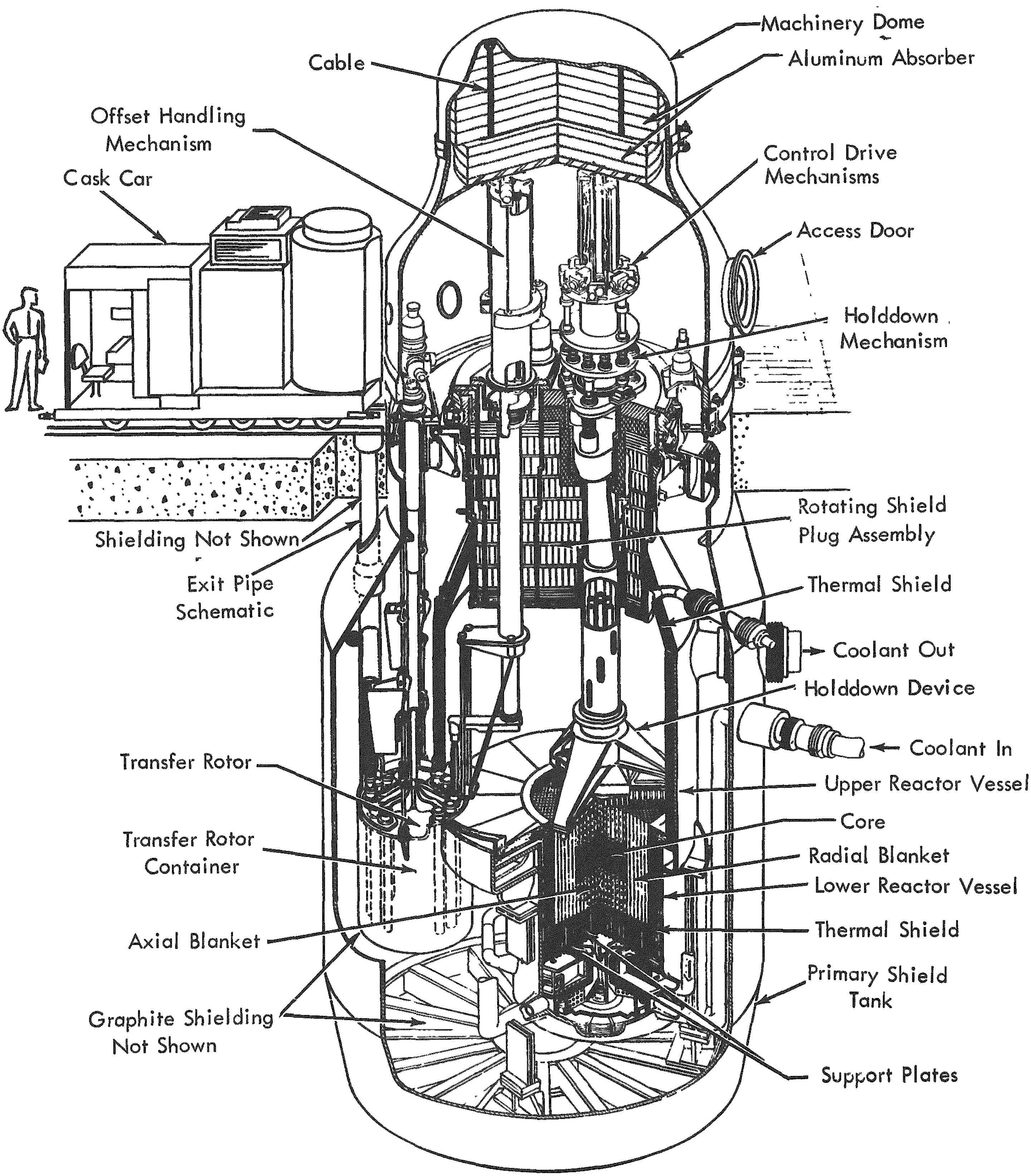

FIG. 1 PERSPECTIVE VIEW OF REACTOR

C. CORE AND BLANKET

The core and blanket are located in the lower portion of the reactor vessel and consist of 2.646-in.-square subassemblies containing fuel pins and blanket rods. The core, which is contained in the central portion of the subassemblies, approximates a right circular cylinder 31 in. in diam. and 31 in. high. It is axially and radially surrounded by breeder blankets, and the entire configuration of core and blanket subassemblies approximates a right circular cylinder 80 in. in diam and 70 in. high.

The fuel is in the form of zirconium-clad pins of 0.158-in. diam, containing U-10 w/o molybdenum alloy, with the uranium enriched to 25.6 w/o U-235. Each core subassembly contains 140 fuel pins having a total mass of approximately 4.75 kg of U-235. Each blanket subassembly contains 25 rods of 0.443-in. diam, consisting of depleted U-3 w/o molybdenum alloy.

A cross section of the reactor, shown in Fig. 2, indicates the placement of individual components within the reactor vessel*. Core and inner radial blanket subassemblies, the antimony-beryllium (Sb-Be) neutron source, and the 10 control rods and safety rods occupy 149 central lattice positions. These are cooled by sodium flowing upward from a high-pressure plenum.

The lattice positions surrounding the core and inner radial blanket subassemblies contain outer radial blanket subassemblies as well as stainless steel-filled subassemblies for thermal shielding. These positions are cooled by sodium flowing upward from a low-pressure plenum.

The sodium coolant system, described in more detail in Section II-E, consists, in part, of three primary coolant loops. The high- and low-pressure plena are fed by three primary sodium pumps through 14-in. and 6-in. lines, respectively. All sodium mixes above the core in the upper plenum and exits by gravity through three equispaced 30-in. lines to three intermediate heat exchangers. From there it returns to the suction side of the three primary pumps.

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* The coordinate system used to identify locations in the core lattice is shown in Fig. 3 and consists of two numbers. The first is the X-coordinate and the second the Y-coordinate. Positive values are represented by "P"; negative values are represented by "N"; the central location is designated P00-P00.

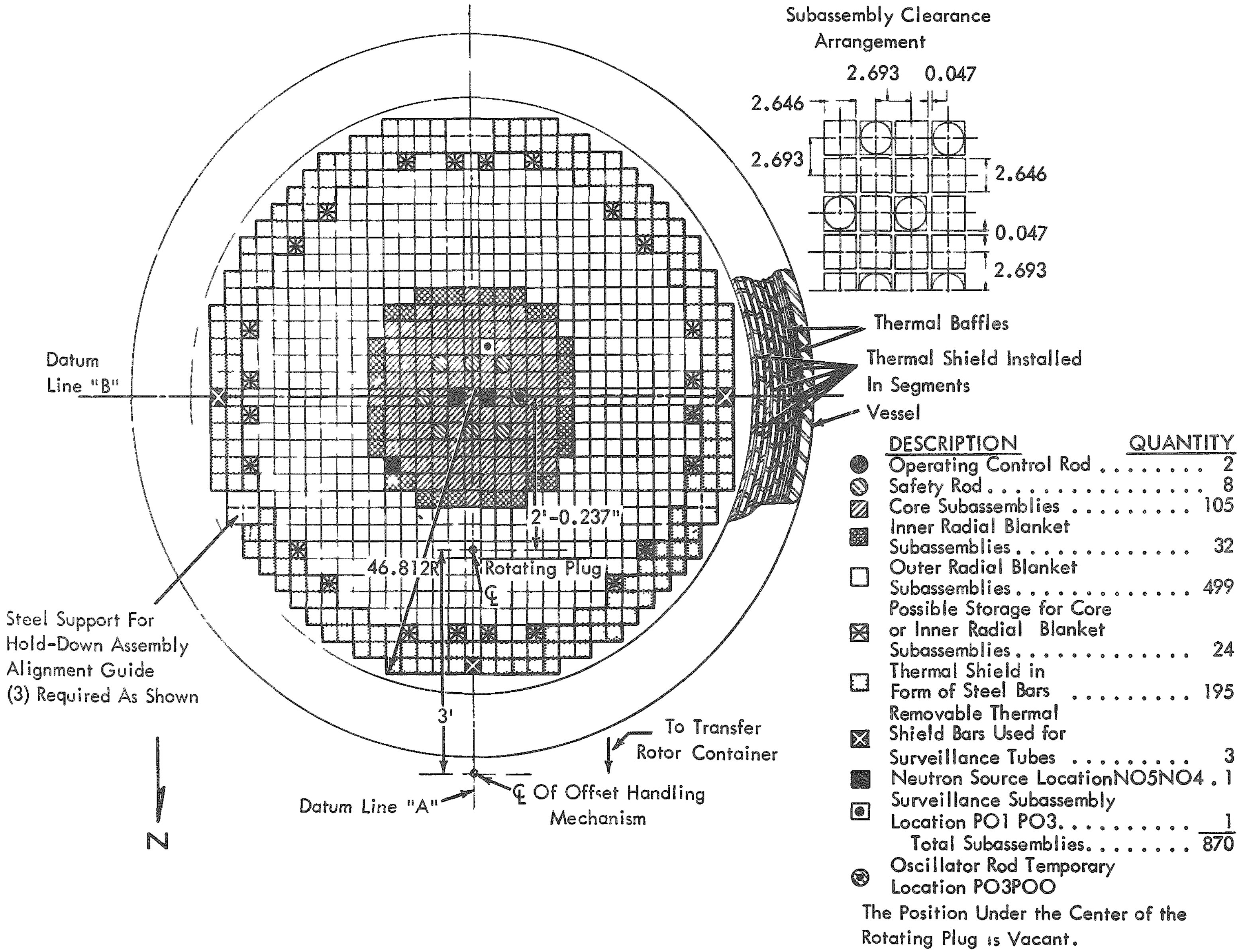

FIG. 2 REACTOR CROSS SECTION

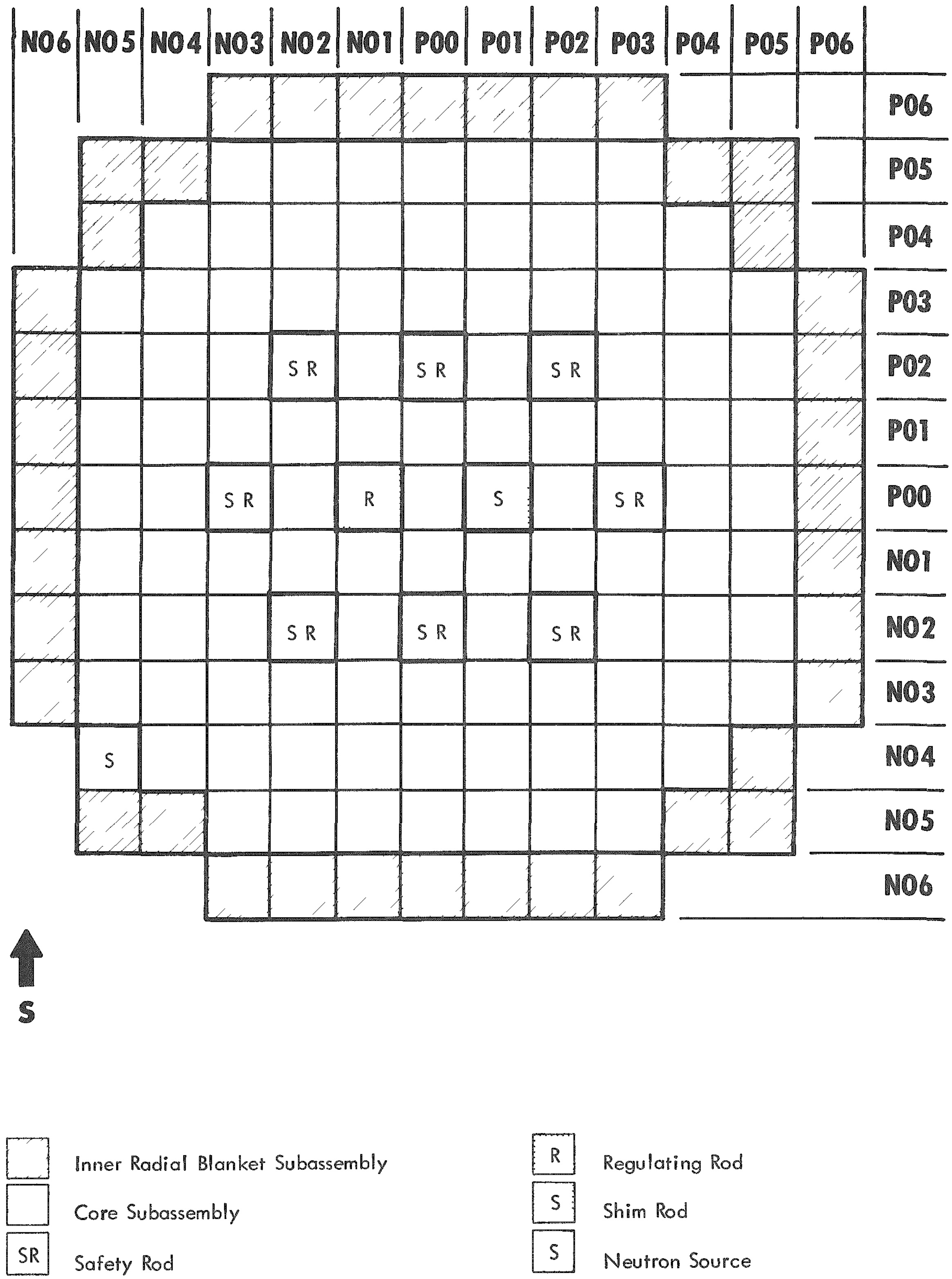

FIG. 3 REACTOR COORDINATE SYSTEM

The subassembly holddown mechanism is used to prevent the core and inner radial blanket subassemblies from floating or rising above the lower support plate due to the sodium pressure drop across the core during reactor operation. Mechanical holddown in addition to the normal weight of the outer radial blanket subassemblies is not necessary because they are cooled only by sodium flowing from the low-pressure plenum.

D. REACTOR CONTROL AND SAFETY SYSTEMS

The reactor is controlled by seven safety rods and two control rods, with provisions available for installation of an eighth safety rod. The rods contain boron carbide, with the boron enriched in boron-10. One control rod is for shimming, while the second provides regulation, and all rods are driven and actuated from the top. During a normal shutdown they are driven in by a motor; following a scram signal, they become delatched from their drives and are rapidly inserted into the core by the expansion of compressed springs.

There are 11 channels of permanent nuclear instrumentation, consisting of compensated and uncompensated ion chambers and fission chambers. The channels are located in six neutron - counter tubes embedded in the graphite neutron shield surrounding the reactor vessel and they are distributed in such a pattern that they will monitor the full flux level of power range during reactor operation. The locations of the neutron - counter tubes in the graphite shield are shown in Fig. 4. Summarized in Table 1 are the type of counter, channel number, neutron - counter tube location, and the specific purpose of each of the detectors.

The start-up instrumentation was used during the tests described in this report. Modifications in location and the type of detector were made to create the permanent plant instrumentation previously described.

Source range measurements were made by BF_3 proportional counters in the start-up instrumentation. In addition, an absolute fission chamber housed in the temporary instrument thimble (see Section II-F-2) was used during several of the tests. The thimble was located in safety rod position No. 5, and all other neutron detectors were housed in either No. 3 or No. 4 neutron - counter tube.

An antimony-beryllium neutron source is located in the reactor at the core-blanket interface (Fig. 2) to provide a neutron flux at the neutron detectors during start-up, and to provide a measureable neutron flux when the reactor is shut down.

The radioantimony portion of the source is made as a separate unit for easy replacement. It is a rod approximately 0.77 in. in diam, and it fits inside a hollow beryllium cylinder 27-in. long, which in turn fits inside a square steel can having the external dimensions of a normal subassembly.

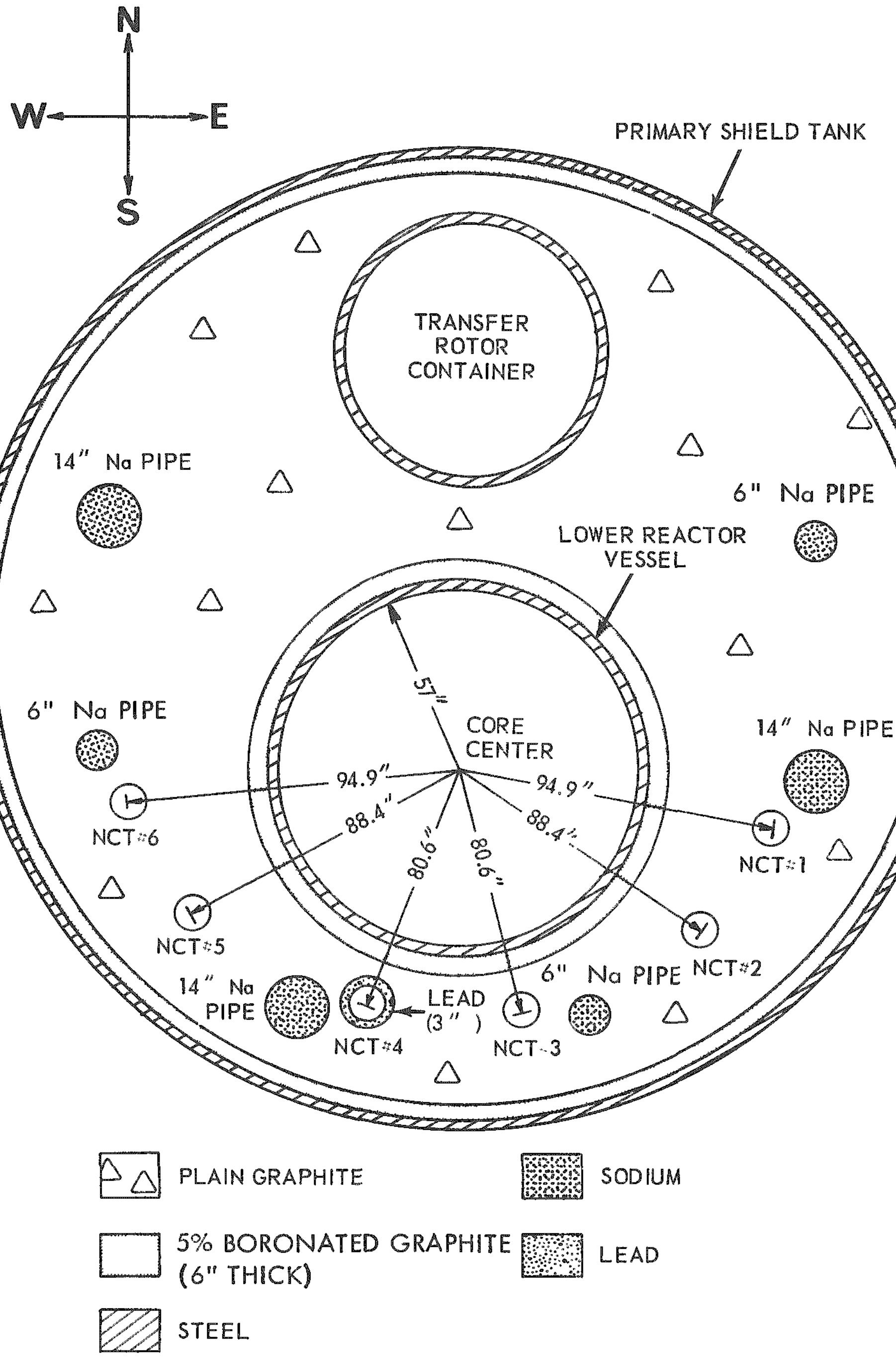

FIG. 4 LOCATION OF NEUTRON-COUNTER TUBES IN GRAPHITE SHIELD

TABLE 1 - PERMANENT PLANT INSTRUMENTATION

Channel Description	No.	Neutron-Counter Tube Number	Type of Detector	Power Range
Source Range	1	3	Fission Chambers	Shutdown - 12 kw
	2	3		
Intermediate Range	1	2	Compensated Ion Chambers	1.2 kw - 1200 Mwt
	2	2		
	3	5		
Power Range Operating	1	2	Uncompensated Ion Chambers	2 Mwt - 300 Mwt
	2	5		
Power Range Safety	1	1	Uncompensated Ion Chambers	2 Mwt - 300 Mwt
	2	1		
	3	6		
Fission Chamber For Intermediate Range Calibration		3	Fission Chamber	300 w - 30 Mwt

E. SODIUM COOLANT AND INERT GAS SYSTEM

The heat transport system, shown in Fig. 5, consists of three primary coolant loops and three secondary coolant loops. The sodium pumps, one per loop, are single-stage, centrifugal pumps. Heat is removed from the core and blanket by the primary sodium, transferred to the secondary sodium in three parallel intermediate heat exchangers located in the Reactor Building, and is finally transferred to steam and water in three once-through-type steam generators located in the Steam Generator building.

A primary sodium service system stores and purifies new sodium received by tank car and monitors and purifies a side stream of the highly radioactive primary coolant when required during operation. The secondary sodium is provided with its own system, which is unshielded because the secondary sodium is not radioactive.

Inert argon cover gas is provided for the entire primary and secondary sodium systems. Its primary purpose is to provide and maintain an inert atmosphere over all liquid-sodium systems, to prevent oxidation of the metal. Further, it provides a supply of purge gas for fuel handling operations when radioactive gas and liquid metal are present, as well as a supply of inert gas for liquid-metal transfer and remote operations. The gas is removed to the waste gas disposal system if it becomes radioactive.

F. ADDITIONAL INSTRUMENTATION

In addition to the 11 channels of nuclear instrumentation discussed in Section II-D, three other devices are pertinent to one or more subsequent sections of this report. They are the retractable neutron source, the temporary instrument thimble, and the temperature-sensing devices used during these tests.

1. Retractable Neutron Source

A retractable neutron source is desirable during low-power tests to permit elimination of the source reactivity contribution by removing the source after criticality is reached. Therefore, such a source was located in core position P03-P00 during several low-power tests.

The source is antimony-beryllium, and consists of a 0.77-in.-diam by 25-in.-long antimony rod located inside a hollow beryllium cylinder 27-in.-long. The beryllium cylinder fits inside a square steel can having the outside dimensions of a normal safety rod guide tube. Thus, the source can be inserted in a safety rod position, and the antimony section can be removed with the offset handling mechanism in the same way that fuel is removed. The beryllium can section, however, must be removed with special tools. The antimony section is the antimony section of the permanent neutron source.

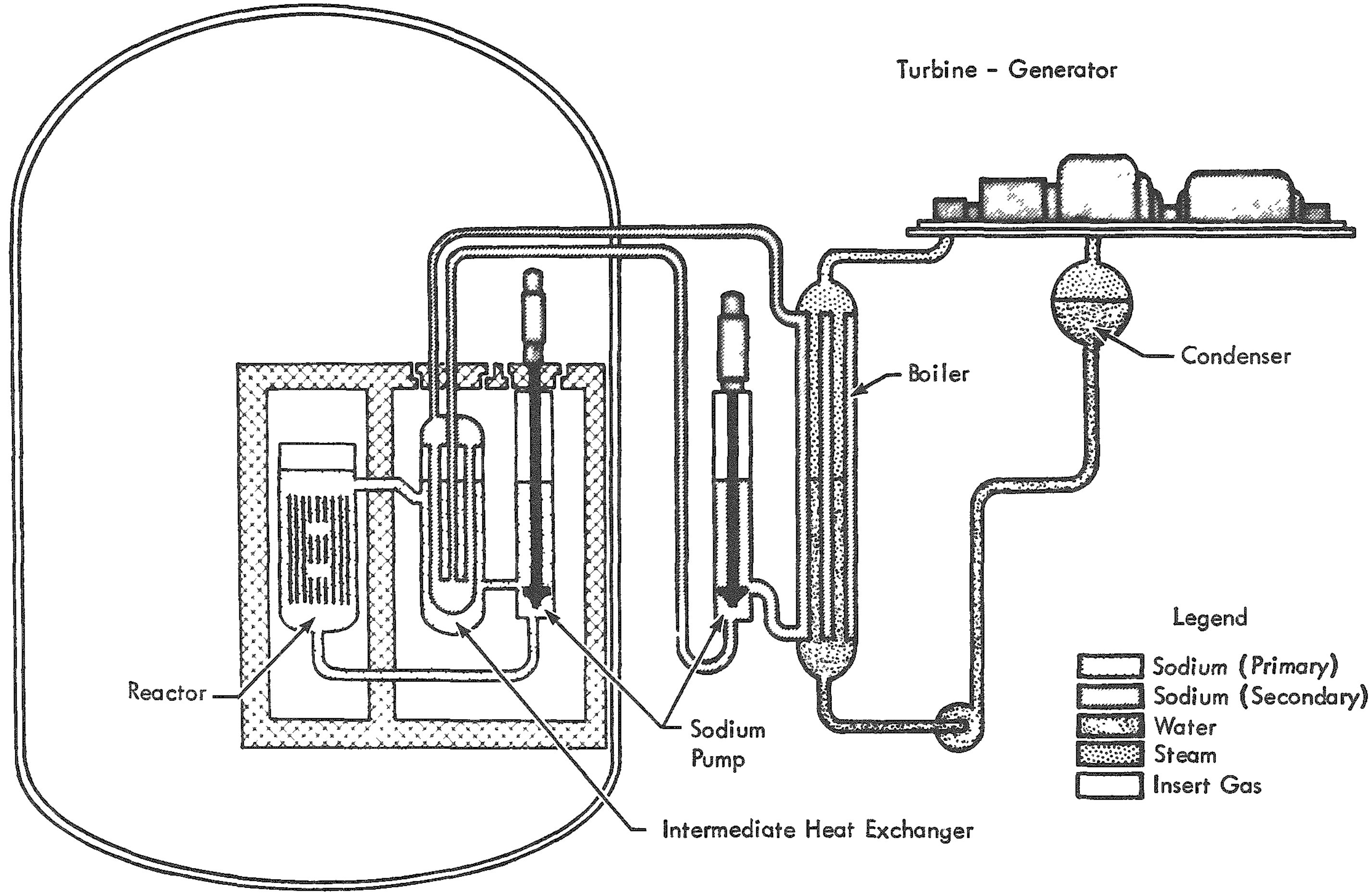

FIG. 5 HEAT TRANSPORT SYSTEM OF THE ENRICO FERMI ATOMIC POWER PLANT

2. Temporary Instrument Thimble

The temporary instrument thimble was provided to house in-core detectors during start-up and other low-power tests. When used, it was located in safety rod position No. 5 at location P03-P00. It is a 2-in. -ID gastight stainless steel tube, sealed at the lower end. It penetrates the rotating shield plug, extends downward through the core, and rests against the dash pot at the bottom of the safety rod guide tube.

3. Temperature Detectors

The temperature detectors used were iron-constantan thermocouples and platinum resistance temperature detectors connected to a high-sensitivity potentiometer and a resistance bridge, respectively. Their locations, types, and temperature-measuring functions are summarized in Table 2.

TABLE 2 - TEMPERATURE DETECTORS

Designation	Type	Location	Temperature Measured
TE 115-3 -15 -30	TC*	P00-P00 ‡ P00-P03 ‡ P00-P05 ‡	Core Outlet
TE 133-11	TC*	P00-P06 ‡	Inner Radial Blanket Outlet
TE 201-1 -2 -3	RTD†	In Elbow Of 30-in. Outlet Pipes	Reactor Outlet
TE 209-1 -2 -3	RTD†	In 6-in. Inlet to Low-Pressure Plenum	Reactor Inlet
TE 110-1	TC*	On Lower Support Plate	Lower Support Plate

* Iron-constantan thermocouple.
† Platinum resistance temperature detector.
‡ Refers to the holddown finger above the location indicated.

Throughout this report a distinction is made between temperatures in the power-producing portion of the reactor and temperatures at the inlet and outlet extremities of the reactor vessel. To make this distinction, the parameters concerned only with the power-producing portion are associated with the word "core" and those concerned with the inlet or outlet extremities are associated with "reactor". Thus, reactor and core inlet temperatures are separate entities, as are core and reactor outlet temperatures.

III. MEASUREMENT OF REACTIVITY WORTH OF PERMANENT NEUTRON SOURCE

A. PURPOSE OF TEST

The purpose of this test was to determine the effective reactivity contribution, as a function of reactor power, of the permanent antimony-beryllium neutron source located at the core-blanket interface (see Fig. 2). This information was necessary to permit source reactivity corrections to be applied to critical rod positions during reactivity measurements at any given power level. Further, this information permitted determination of the effective lower limit of the power range over which reactor period measurements could be made which would not require a source reactivity correction.

B. EXPERIMENTAL PROCEDURE

1. Apparatus and Equipment

The reactor power levels during this test were measured by two high-sensitivity BF_3 proportional counters and one B-10-lined ion chamber located in NCT-3 and NCT-4 (see Fig. 4). The two BF_3 detectors were connected to mechanical scalers located in the reactor control room and were used to supply the count rate and period signals to the two source range channels of the reactor safety system. The ion chamber was connected to a picoammeter recorder channel, also located in the control room. This recorder yielded period data and measured any power drift taking place during critical rod position measurements.

The temperatures at specific locations were measured during this test using the normal plant temperature-sensing devices. These consisted of iron-constantan thermocouples and platinum resistance temperature detectors, for which specific locations and functions are described in detail in Section II-F-3 and Table 2. During the test, the data from all temperature sensors were relayed to the temporary precision-readout station in the reactor control room. The thermocouples were connected to a high-sensitivity potentiometer, and the resistance detectors were connected to a resistance bridge. This equipment permitted temperature readout accuracy of $\pm$ 1F.

The permanent plant instrumentation was used to obtain the necessary data on sodium flow rates and control rod and safety rod positions. The primary sodium flow rates were measured in the control room for each loop by magnetic flowmeters capable of being read to within $\pm 0.05 \times 10^6$ lb/hr. The regulating rod and shim rod positions could be read to within $\pm$ 0.01 in. by means of Gilmore digital-readout position indicators located in the control room.

2. Reactor Plant Conditions

During this test, the temporary instrument thimble (see Section II-F-2) was in position P03-P00, and was used to house a U-235 absolute fission chamber. The neutron source was at its normal operating position at the core-blanket interface, position N05-N04. (See Fig. 2.)

The primary system was maintained at an isothermal condition of 410 F and primary flow was set at approximately 2.00×10^6 lb/hr/loop for three-loop operation. The coolant temperature was held constant by balancing the heat input from the primary sodium pump operation with the heat removal resulting from operation of the secondary sodium system, the feedwater system, and all auxiliary systems.

The normal reactor safety system was in operation during the test, with the exception that the sodium flow scram point was reduced from 75 to 40 per cent of the 200-Mwt design flow rate, because of the lower primary coolant flow rate during this test. The intermediate level scrams were set at a flux level corresponding to a power of approximately 1 Mwt, and the power range channels were adjusted to provide level scrams at their minimum set points (approximately 1 Mwt).

The reactor was loaded during this test to provide at least 32 inhours but less than 96 inhours excess reactivity in the shim rods and regulating rods, to overcome the source reactivity effects at low powers and to accomodate the desired power level changes during the test. The loading configuration used is shown in Fig. 6.

3. Description of Measurements

After the necessary circuits were modified and the flow rates were adjusted, the reactor was brought to a subcritical power level of approximately 15 watts. The reactor was subcritical due to the absorptive effect of the antimony portion of the source. The subcritical count rate was considered to have leveled out when successive counts on the count rate channels in equal time periods did not vary by more than two per cent of the total count rate.

To insure that the desired isothermal condition of 410 F existed, core and reactor inlet and outlet temperatures were monitored and recorded at 5-min intervals until the temperature of any one point did not vary more than 2 F between readings, and the ΔT across the reactor was 2 F or less.

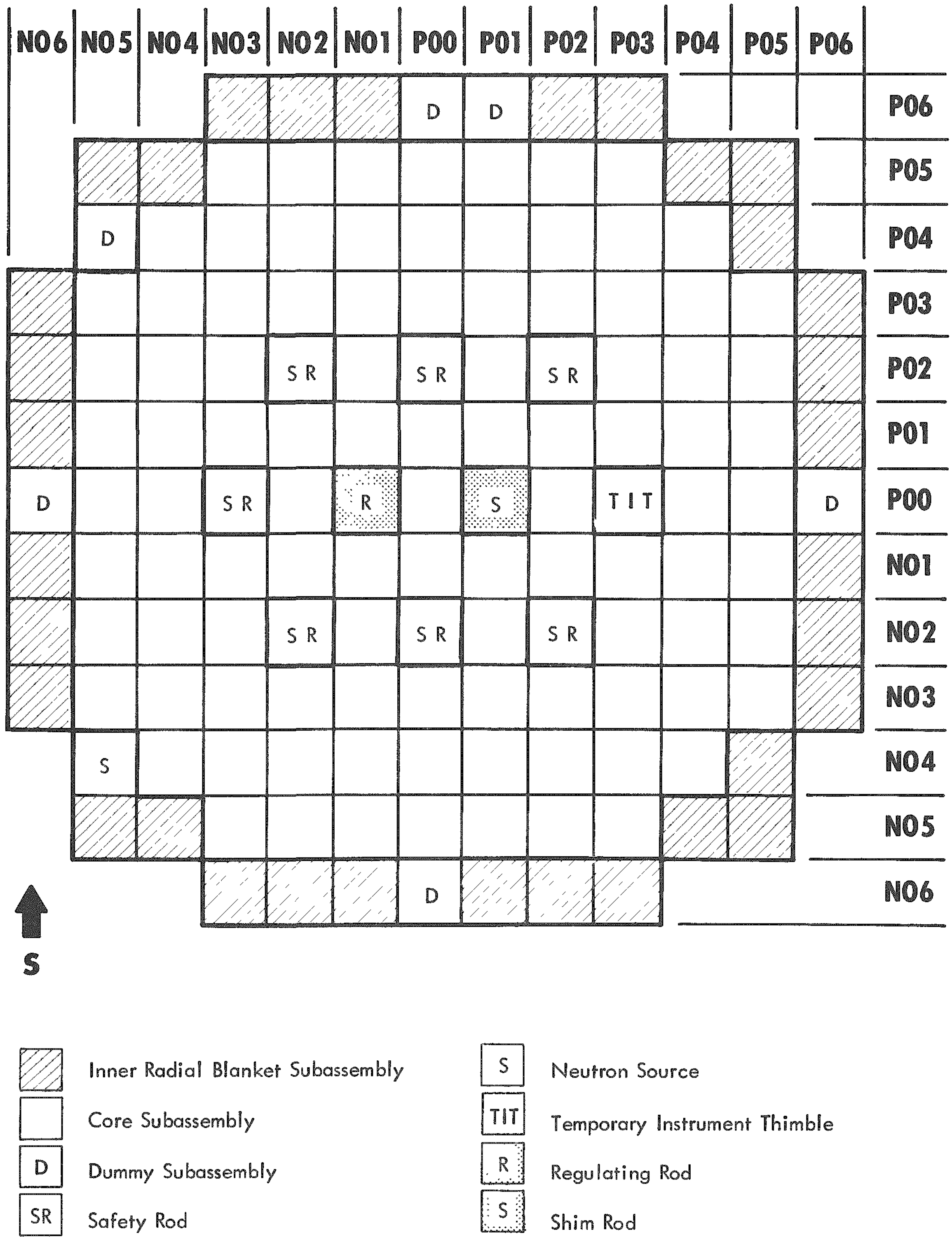

FIG. 6 REACTOR LOADING FOR PERMANENT SOURCE TEST

When the equilibrium conditions were satisfied, the following data were noted and recorded:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperature
- Primary sodium flow rate
- Integrated count rate on the neutron detector channels and the in-core fission chamber for 10,000 counts or 5 min
- *Shim rod and regulating rod position**

The regulating rod was repositioned to raise the power level, and the entire list of data was recorded again. Ultimately, the data were recorded at four subcritical power levels up to approximately 100 watts.

Following these measurements, a series of measurements was made in which the reactor was placed on various positive periods. Counts were recorded at 15-sec intervals to a power level of approximately 3000 watts, and the periods were determined from a graph of logarithm of power versus time, as well as from the picoammeter recorder channel. Measurements were recorded for six different reactor periods, using only one neutron-detector channel. At the conclusion of these measurements the test was completed.

C. EXPERIMENTAL RESULTS AND ANALYSIS

The experimental data for determination of the source correction factor are presented in Table 3. Counts were recorded with the shim rod position held constant, and the safety rods fully withdrawn. Counts were recorded using all four detector channels as well as the in-core fission chamber; however, the data from the in-core fission chamber were sufficient to determine the amount of subcritical reactivity.

The subcritical reactivity is determined by the method of subcritical multiplication, discussed in detail in the Appendix. Since the amount of reactivity (Δk) due to the source decreases as power (P) increases, $P \cdot \Delta k$ should be constant. The values for Δk and $P \cdot \Delta k$ as functions of power are given in Table 4 and the average of the four measured values is given as the source correction factor. The calculation of Δk assumed two important calibrations:

- The worth of the regulating rod between 4- and 9-in. withdrawn equals 3.55 cents per in., $\pm$ 2 per cent. The regulating rod calibration curve is shown in Fig. 7.

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* Safety rods were fully withdrawn throughout the test.

- 951 counts per min recorded by the U-235 (199 μgram) absolute fission chamber equals 1 watt of reactor power.

TABLE 3 - EXPERIMENTAL DATA -- DETERMINATION OF PERMANENT SOURCE CORRECTION FACTOR

Rod Positions, in.*		Count Rate From In-Core Fission Chamber, counts/min†	Calculated Power, watts‡
Regulating Rod	Shim Rod		
4.00	10.01	16,907	17.8
5.00	10.01	22,718	23.9
6 00	10.01	36,286	38.2
7.00	10.01	88,088	92.6

TABLE 4 - RESULTS OF DETERMINATION OF PERMANENT SOURCE CORRECTION FACTOR

Power, P, watts	Average Amount of Subcritical Reactivity, Δk, cents	$P \cdot \Delta k$
17.8	13.48	239.9
23.9	9.811	234.5
38.2	6.064	231.6
92.6	2.491	230.7

Average source correction factor = 234 $\pm$ 7 watt-cents

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* All measurements made with all safety rods fully withdrawn
† All measurements, average of two 1-min counts
‡ Assumes 951 counts per minute on fission chamber = 1 watt

Early calculations of the source correction factor were based on a different conversion factor, 863 counts/min/watt, rather than 951. Subsequent calculations which have been compared to heat balance tests have shown that the value of 951 is more accurate. Further, some initial calculations, which resulted in a source correction factor of 243 watt-cents (versus 234 quoted in Table 4) were based on a regulating rod calibration

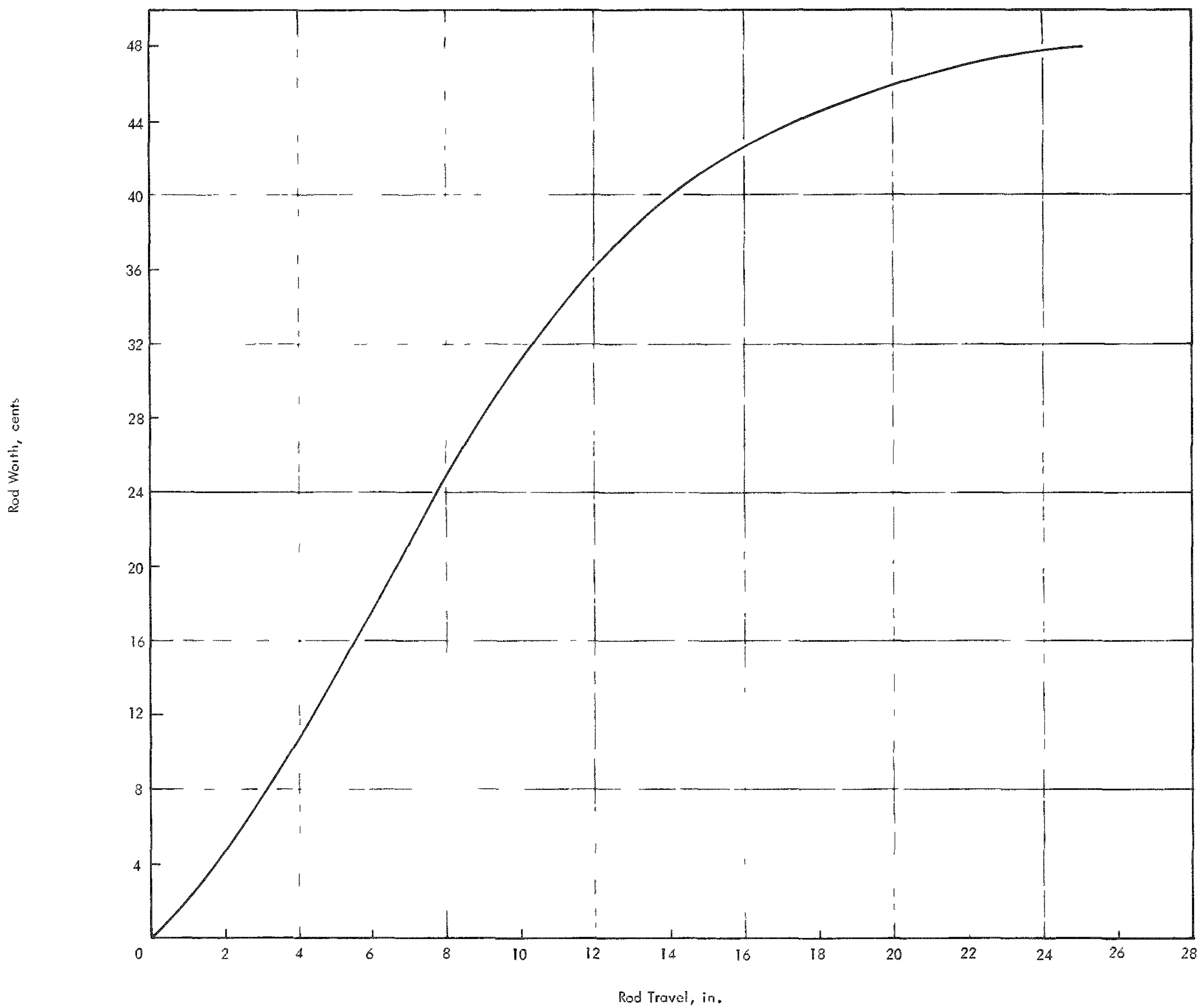

FIG. 7 REGULATING ROD CALIBRATION CURVE

curve which gave the worth of the regulating rod between 4- and 9-in. withdrawn as 3.35 cents per in., ± 2 per cent. This curve was found to be slightly less accurate than the one used in this calculation and shown in Fig. 7. The value of 234 ± 7 watt-cents for the permanent source correction factor given in Table 4 is therefore assumed to be more accurate than values which might be found elsewhere. In addition, the source factor will have decreased approximately 1 per cent per day relative to August 27, 1963 (when the value of source strength was established), due to decay of the radioantimony.

An analysis of errors present in the measurements and calculations showed that errors in measuring rod position and calibration and errors inherent in counting statistics were significant. The first two contributed 2 per cent error each, and counting statistics were assumed to be in error by ± 1 per cent. The most probable error was assumed to be the square root of the sum of the squares of the individual errors, or 3 per cent, and this was equivalent to the ± 7 watt-cents given in Table 4.

To determine the power range over which period measurements could be made which would require no source corrections, the reactor was placed on various positive periods and counts from the neutron detector recorded. Counts were recorded from only one neutron-detector channel, using a proportional counter, and the reactor period was determined from the graph of logarithm of power versus time. The reactor period was also measured by the picoammeter recorder channel, and in all cases the values agreed quite satisfactorily.

The count rate data were taken in 15-sec intervals, with the last 3 sec of each interval being used for recording the measured value and for resetting the counter. The counter stopped and started automatically, and the time intervals were constant within the limits of mechanical accuracy of the timer.

Power values for each time interval were obtained using a conversion factor for the neutron detector, based on subcritical counts recorded by the detector and by the in-core absolute fission chamber. Using the fact reported above that 951 counts per min on the fission chamber equals 1 watt of reactor power, a conversion factor was obtained for the neutron detector from counts recorded at the same power. This conversion factor was found to be 94 counts per min equals 1 watt.

A plot of the logarithm of power versus time after a reactivity insertion, (Fig. 8) will be nearly linear for powers high enough to minimize the source contribution. These curves are linear above a power of approximately 500 watts and remain linear to a power of at least 2 kw.

D. CONCLUSIONS

From the value of the source correction factor given in Table 4, it can be concluded that critical rod measurements should be made at power levels in excess of 1 kw. For example, at 10 watts the source correction would be 23 $\pm$ 1.2 cents, an appreciable amount, while at 1000 watts the source correction would be much less, 0.23 $\pm$ 0.012 cents, a negligible amount.

From the graphs shown in Fig. 8, it can be concluded that period measurements can be made at powers in excess of 500 watts with small source effects. However, to remain consistent with the above conclusions, period measurements should in the future be made at power levels above 1 kw if the permanent neutron source is in position.

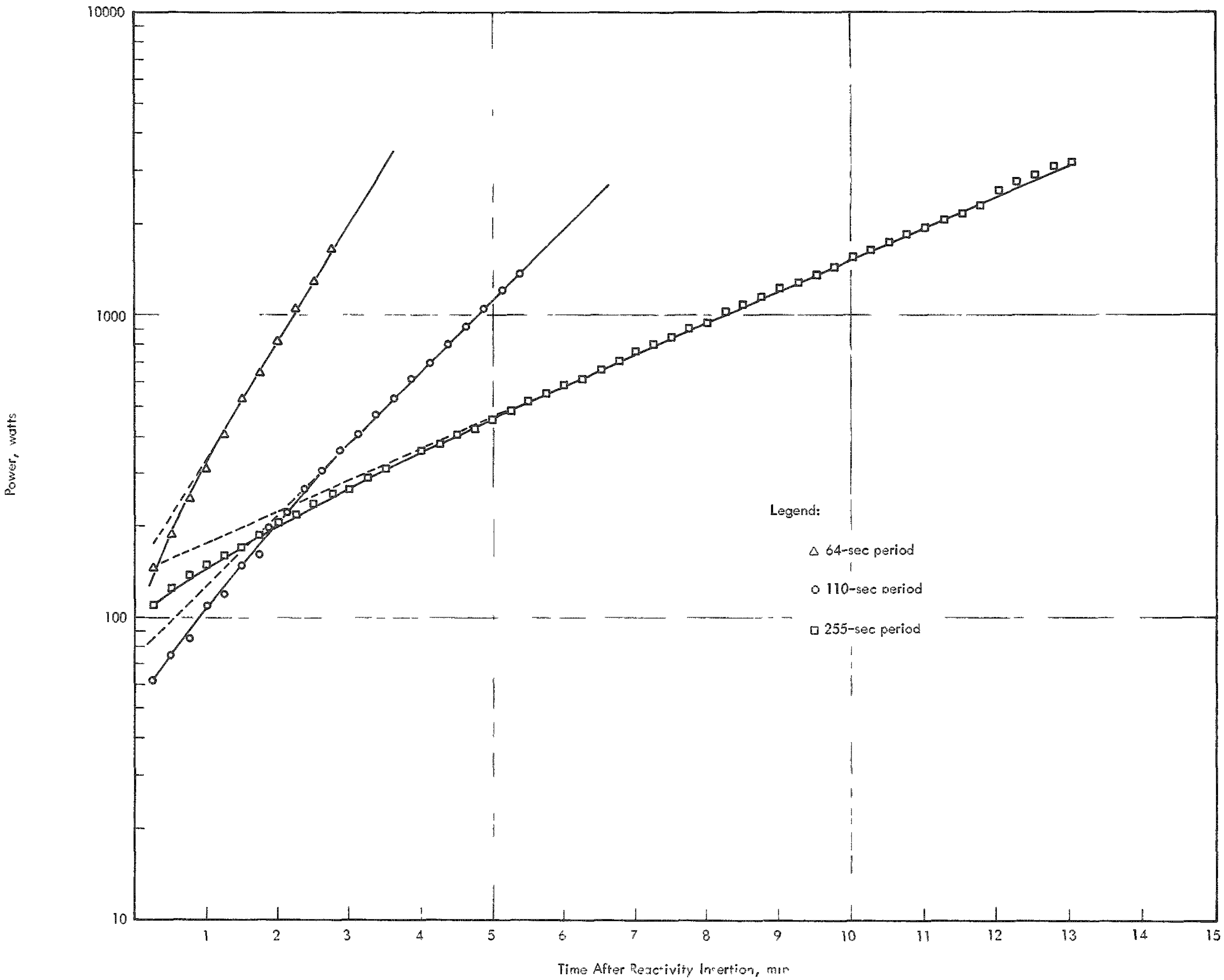

FIG. 8 SEMILOGARITHMIC PLOT OF REACTOR POWER AS A FUNCTION OF TIME AFTER REACTIVITY INSERTION

IV. MEASUREMENT OF REACTIVITY WORTH OF RETRACTABLE NEUTRON SOURCE

A. PURPOSE OF TEST

The purpose of this test was to measure the effect of inserting a retractable neutron source into the core, and the reactivity effects of the source following its installation.

For the majority of the low-power nuclear tests it was expedient to withdraw the stationary neutron source from its core edge position and to install a special neutron source in safety rod position No. 5 (core position P03-P00; see Fig. 2). This source had an active portion which could be retracted by the safety rod drive mechanism when desired. The retraction of this source after criticality had been reached would eliminate any considerations of source reactivity effects in subsequent measurements. This would circumvent the stationary source restriction that operating power levels exceed about 1 kw to minimize source reactivity effects. The very low operating power levels thus made possible would minimize the activation of core materials and coolant during the remaining low-power test program.

Two separate tests were required, however, for the retractable source. The first was the reloading and new approach to criticality following initial installation of the source, and the second was the actual reactivity measurements.

Since the reactivity effect of the source was unknown, the acceptable way to insert it into the core was to remove fuel to a configuration well below the critical loading, insert the source, and then to reload and approach criticality in a straightforward manner.

When installed, the retractable source has two reactivity effects. The first is the apparent reactivity resulting from the constant neutron emission rate of the antimony-beryllium assembly†, which is inversely

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† The reactions involved are:

$$Sb^{124*} \rightarrow Te^{124} + \beta^{-} + \gamma$$

$$Be^{9}\ (\gamma, n)\ Be^{8}$$

proportional to power. The second effect is an increase in reactivity upon withdrawal of the antimony portion of the source, which is an absorber. This effect is independent of power, but must be measured at a sufficiently high power to minimize the fact that, as the antimony is withdrawn, the source neutron emission rate decreases.

B. EXPERIMENTAL PROCEDURE

1. Apparatus and Equipment

The reactor power level during the test was measured by three high-sensitivity BF_3 proportional counters and six B-10-lined ion chambers, located in neutron-counter tubes NCT-3 and NCT-4 (see Fig. 4). Two of the proportional counters supplied signals to the two source range channels, while the third served as a monitoring channel. Each counter was connected to a mechanical scaler located in the reactor control room. Five of the ionization chambers provided signals to the three intermediate range channels and to two of the three power range channels in the safety system. The sixth ion chamber provided a signal to a picoammeter recorder channel.

The primary system temperatures were monitored with the normal plant temperature-sensing elements consisting of iron-constantan thermocouples and platinum resistance temperature detectors, connected to a high-sensitivity potentiometer and a resistance bridge, respectively. The sensing elements were positioned such that core and reactor inlet and outlet temperatures could be monitored. Their specific locations and functions are discussed in detail in Section II-F-3 and Table 2. The temperature data for the test were obtained with high precision ($\pm$ 1 F) at a temporary readout station in the reactor control room.

Sodium flow rate data were obtained by use of six primary sodium flow meters -- part of the permanent plant instrumentation. Readings were made and logged to the nearest 0.05×10^6 lb/hr. Gilmore digital-readout position indicators were used to determine the shim rod and regulating rod positions to the nearest 0.01 in. During the ganged withdrawal of the seven operating control rods, the position of safety rod No. 1 was read to the nearest 0.01 in., as indicated on the Gilmore position indicator.

The position of the antimony portion of the retractable source at any given time was indicated by the control console dial connected to the potentiometer on the source drive extension normally employed for safety rod No. 5. The source position was controlled by means of a temporary control station located in the control room.

2. Reactor Plant Conditions

a. Reload for Retractable Source

To begin the test, seven core edge fuel subassemblies, representing approximately two dollars of reactivity, were removed from the core region and placed in fuel storage positions. These vacated positions were then filled with dummy core and inner radial blanket subassemblies. Following this reduction in core loading, the temporary instrument thimble located in position P03-P00 was replaced by the retractable neutron source beryllium can and source drive extension. The beryllium can for the operating stationary source remained in its position at the core-blanket interface, N05-N04. Its radioantimony section was also left in the permanent neutron source location during the portion of this test involving the approach to criticality. Following criticality, the radioantimony section was transfered to the retractable source position. The reactor loading at this point indicates the location of the permanent and retractable neutron sources (see Fig. 9).

The fuel loading operations required that the primary sodium flow be reduced to pony motor flow rate of approximately 11 per cent of the 200-Mwt design flow rate to facilitate maintenance of an isothermal condition (at 517 F) without "floating" the dummy subassemblies in the core with the holddown mechanism in the raised position. Earlier calculations had indicated that the plugged-filter-type dummy subassemblies would float at sodium flow rates greater than 25 per cent of 200-Mwt design flow rate. The flow rate was increased to 200-Mwt design flow rate when criticality was reached and normal subassemblies were back in the core.

The conditions of the secondary sodium and feedwater systems were adjusted to maintain the desired isothermal condition at 517 F in the primary system.

b. Reactivity Worth Measurements of Retractable Source

The core was loaded for this phase of the test so that the excess reactivity was between 50 and 140 inhours. It was predicted that approximately 100 inhours of negative reactivity in the shim rod and regulating rods would be required to achieve the desired initial power level of approximately 50 watts for this portion of the test.

Since the reactor power level was limited to 1 Mwt or less, the intermediate range level scrams were set to scram the reactor at a flux level corresponding to that power level. The power range level scrams were set at minimum - - approximately 1 Mwt.

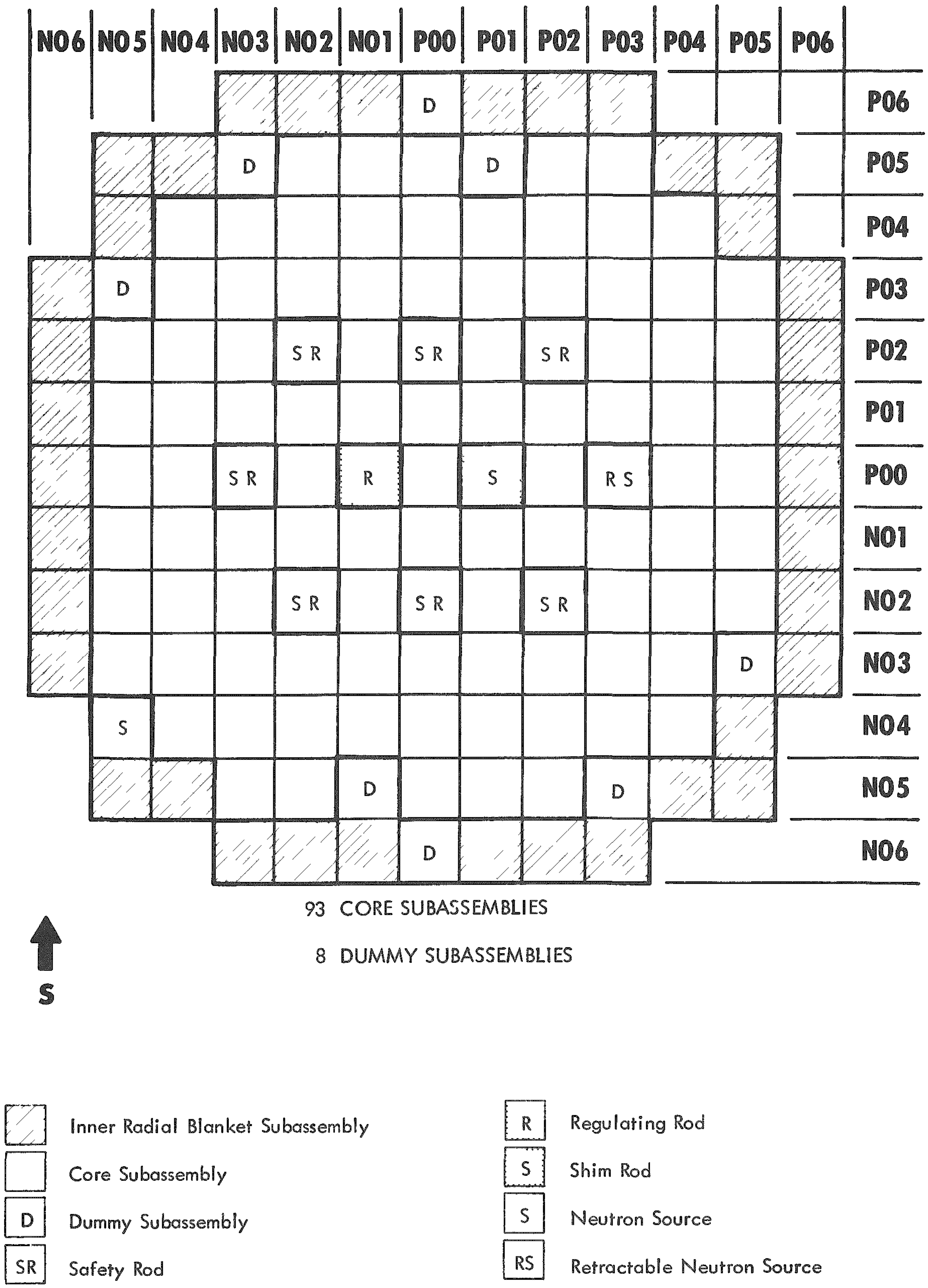


FIG. 9 REACTOR LOADING I FOR RELOAD FOR RETRACTABLE SOURCE

Primary sodium flow was set at the refueling flow rate of 52.5 per cent of 200-Mwt design flow, or 1.55×10^6 lb/hr/loop for three-loop operation. Further, the primary sodium flow rate scram point was set at 40 per cent of 200-Mwt design flow.

3. Description of Measurements

a. Reload for Retractable Source

When the desired configuration described in Section IV-B-2 was obtained, the approach to criticality was made in a manner similar to the procedure used in the initial approach to criticality.[1]

With an initial loading of 93 core subassemblies, the subcritical count rate was allowed to level-out. This was determined by taking successive counts until the counts accumulated in equal periods of time did not vary by more than 2 per cent of the total count rate, providing at least 10,000 counts were accumulated (or 5 min elapsed if the count rate was very low).

An isothermal condition was maintained at 517 F at each point. This was determined by recording temperatures at 5-min intervals until (1) the temperature at any given point did not vary between readings by more than 2 F, and (2) the temperature difference between any two points was 4 F or less.

When these criteria were satisfied, the following measurements were recorded:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperatures
- Primary sodium flow rate
- Retractable source position
- Total integrated counts on the three count rate channels for a minimum of 5 min, or 10,000 counts
- Ion chamber current as indicated on the picoammeter recorder channel
- Shim rod, safety rod, and regulating rod positions

These data were recorded at seven rod configurations, summarized in Table 5 and in Section IV-C-1.

TABLE 5 - EXPERIMENTAL DATA -- RELOAD FOR INSERTION OF RETRACTABLE NEUTRON SOURCE

Subassemblies In Core	Rod Positions, in. Safety	Shim	Regulating	Count Rate,[d] counts/min
93	IN	IN	IN	389
	19.65[a]	IN	IN	1,640
	21.22[b]	IN	IN	1,823
	31.56[c]	IN	IN	2,140
	OUT	IN	OUT	2,591
	OUT	OUT	IN	2,598
	OUT	OUT	OUT	3,349
96	IN	IN	IN	440
	19.59	IN	IN	2,611
	21.23	IN	IN	3,053
	31.60	IN	IN	3,827
	OUT	IN	OUT	5,405
	OUT	OUT	IN	5,527
	OUT	OUT	OUT	9,616
98-1/2[e]	IN	IN	IN	480
	19.55	IN	IN	5,046
	21.25	IN	IN	6,845
	31.63	IN	IN	10,807
	OUT	IN	OUT	58,012
	OUT	OUT	IN	80,901
	OUT	OUT	2.99	CRITICAL

a All safety rods were ganged. This is position of safety rod No. 1 at first subcritical stop position.
b Second subcritical stop position.
c Full-out position.
d Average of two 5-min counts on channel 1 and two 5-min counts on channel 2.
e 98 fuel subassemblies and one shim subassembly.

The magnitude of each reloading was determined by the degree of subcriticality, as estimated from the subcritical count rate for the various rod positions of the previous loading.

The sequence of loading increments was one loading increment of three fuel subassemblies followed by one loading increment of two fuel subassemblies and one shim subassembly. A shim subassembly has a worth approximately one-half that of a normal core subassembly. The last loading increment was chosen so that criticality would be reached with all but the regulating rod fully withdrawn. The reactor configurations after each of these two loading increments are shown in Figs. 10 and 11.

When criticality had been reached, the following data were recorded:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperatures
- Primary sodium flow rate*
- Critical regulating rod position
- Reactor flux level and drift rate as indicated by the picoammeter recorder channel

This concluded the reloading. The reactivity measurements were then begun.

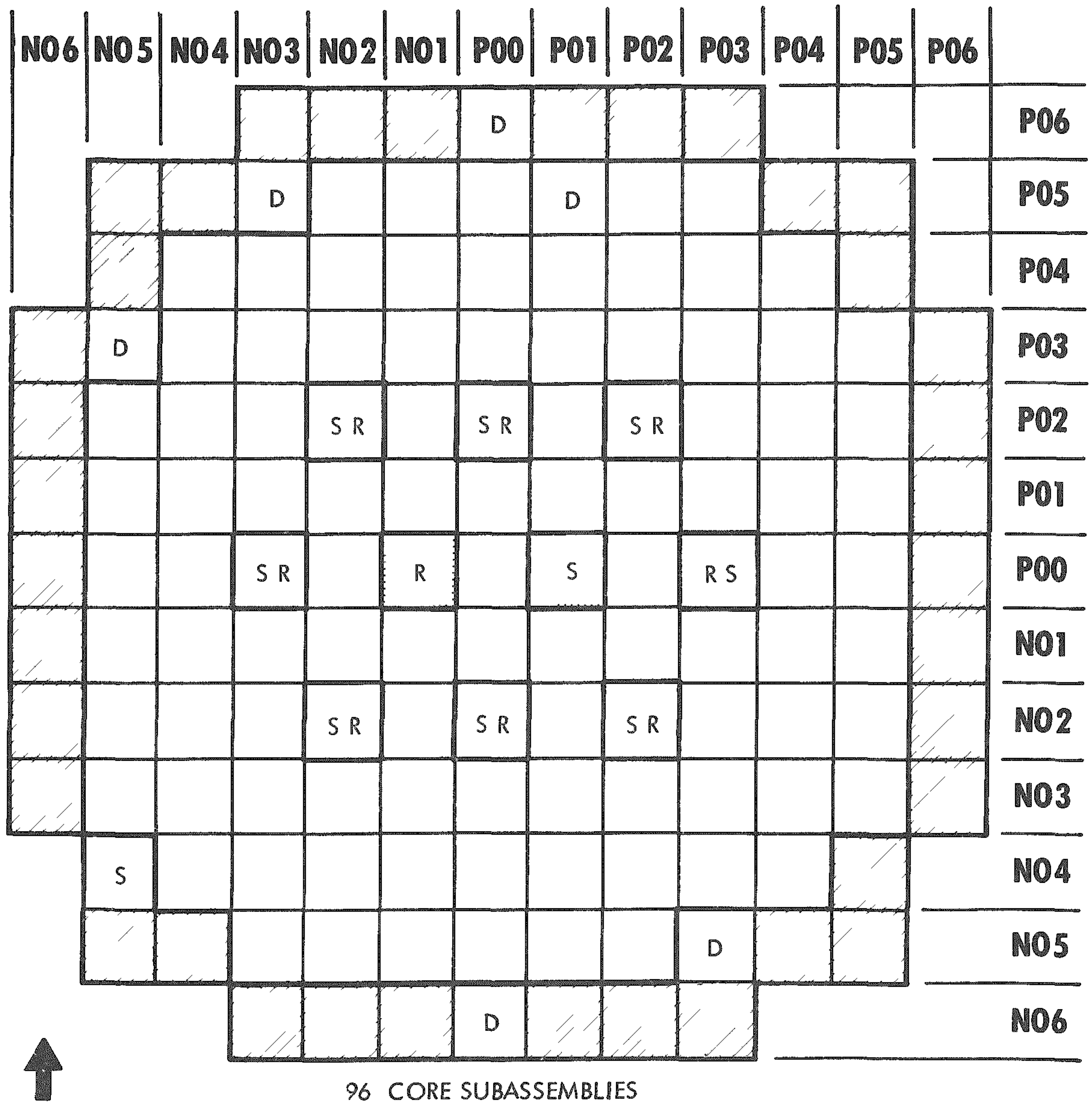
b. Reactivity Effects of Retractable Source

(1) Determination of the Reactivity Contribution of the Source Neutrons

The reactor was brought to a subcritical power level of approximately 70 watts to begin this test. The subcritical count rate was allowed to level out in the manner described earlier (Section IV-B-3-a) and the following measurements were recorded:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperature
- Subcritical count rate on the two count rate channels for 5 min, or a minimum of 10,000 counts
- Regulating rod position (shim rod position fixed at 13-in. withdrawn).

* At criticality this had been increased from pony motor flow to 200-Mwt design flow.



96 CORE SUBASSEMBLIES

5 DUMMY SUBASSEMBLIES

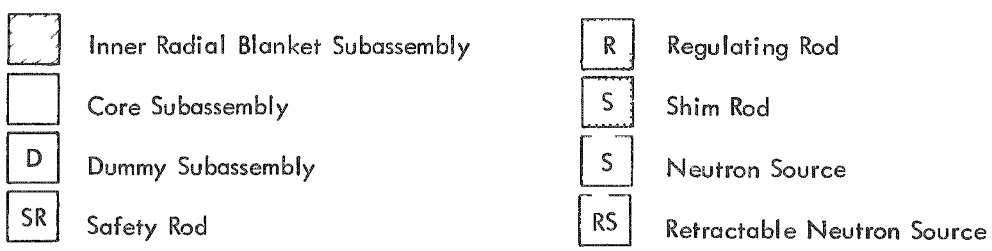
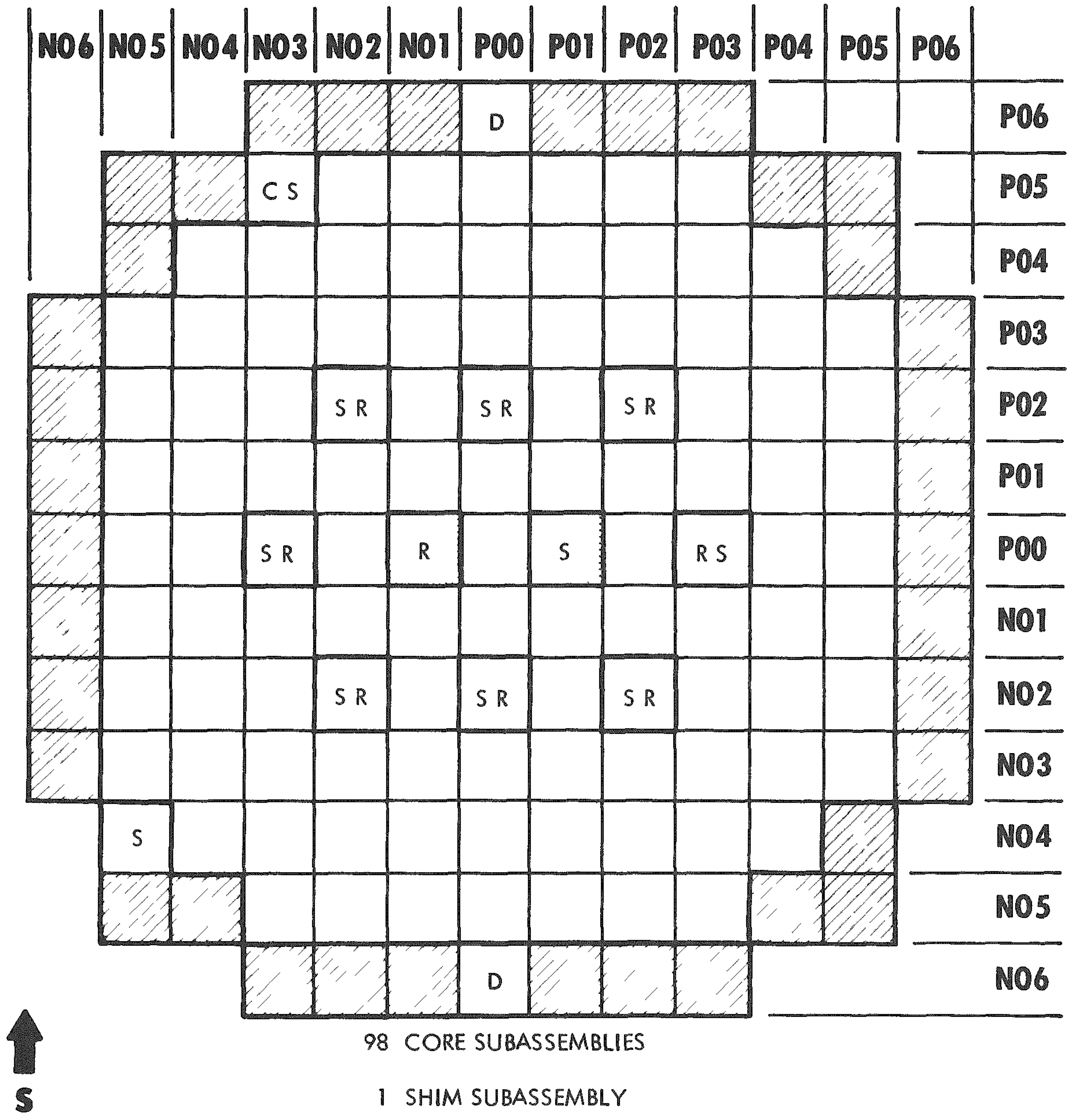


FIG. 10 REACTOR LOADING II FOR RELOAD FOR RETRACTABLE SOURCE



98 CORE SUBASSEMBLIES

1 SHIM SUBASSEMBLY

2 DUMMY SUBASSEMBLIES

	Inner Radial Blanket Subassembly	R	Regulating Rod
	Core Subassembly	S	Shim Rod
D	Dummy Subassembly	S	Neutron Source
CS	Core Shim Subassembly	RS	Retractable Neutron Source
SR	Safety Rod		

FIG. 11 REACTOR LOADING III FOR RELOAD FOR RETRACTABLE SOURCE

The preceding sequence of measurements was recorded at three more power levels up to a maximum of 1250 watts. It was observed that a power level of about 1 kw was sufficient for the second portion of the test (see Section III).

Using the regulating rod calibration curve shown in Fig. 7, and the method of subcritical multiplication described in the Appendix, the measured regulating rod positions and power levels as indicated by the picoammeter recorder channel were converted to reactivity values, to determine the source correction factor as a function of power.

(2) Determination of the Reactivity Worth of the Antimony Portion of the Retractable Source

This portion of the test was conducted at a fixed power of 1250 watts, for seven positions of the antimony rod ranging from fully inserted to fully withdrawn. The shim rod was held at 13-in. withdrawn and the critical regulating rod positions ranged from 4-1/2- to 7-1/2-in. withdrawn.

The following data were recorded at each position of the antimony rod:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperature
- Reactor flux level and drift rate, as indicated by the picoammeter recorder channel
- Critical regulating rod position

The seven positions measured were: fully inserted; withdrawn 10, 20, 25, 30, 40 in.; and fully withdrawn, or 53 in. This concluded the entire test.

C. EXPERIMENTAL RESULTS AND ANALYSIS

1. Reload for Retractable Neutron Source

The experimental data recorded during the approach to criticality following installation of the retractable neutron source are shown in Table 5. Counts were recorded on two source range channels for two 5-min intervals, and the average of the four values is given in Table 5. Measurements were made for all safety rods fully inserted, withdrawn to their first subcritical stop position, withdrawn to their second subcritical stop position, and fully withdrawn. With the safety rods fully withdrawn, further measurements were taken for the three shim rod and regulating rod positions noted in the table.

To approximate, initially, the amount of reactivity required in the next loading, a plot of inverse count rate versus number of subassemblies in the reactor was made. Criticality was desired for all but the regulating rod fully withdrawn, and the extrapolation of the curve through points for this configuration gave an estimate of the amount of fuel required to attain criticality. The plot of relative inverse count rate versus reactor loading is shown in Fig. 12, for which the data have been normalized to unity at 93 subassemblies in the reactor and all rods fully inserted.

The indicated loading with 96 subassemblies in the core was 98-1/2 subassemblies. This was attained by loading two normal fuel subassemblies and one shim subassembly which has a worth of approximately one-half that of a normal fuel subassembly. Criticality was achieved with 98-1/2 subassemblies in the reactor and all but the regulating rod fully withdrawn, as was desired (see Table 5).

The reactivity state of the reactor throughout the loading was determined from the subcritical count rate data. The data were analyzed in three ways: (1) by extrapolation of the curve of inverse count rate versus loading, Fig. 12; (2) by using the previously determined rod worths to derive a relationship between subcritical reactivity and the change in subcritical count rate upon rod withdrawal, using the method discussed in the Appendix; and, (3) by measuring the "source reactivity contribution" with all rods up.

Using these three methods, the amount of excess reactivity in the core after insertion of the final two and one-half subassemblies was computed to be 30 cents, 23 cents, and 35 cents, respectively.

The worth of the beryllium can was determined from the change in subcritical count rate which occurred after the can had been installed in the reactor, and also from the new critical mass with the can installed compared to the critical mass without the can. The worth of the can by these two methods is 37 cents and 35.5 cents, respectively.

2. Reactivity Worth of Retractable Neutron Source

The retractable source worth was measured as a function of power and as a function of position. The data for the experiment are shown in Table 6. From the regulating rod positions and the worth curve of the regulating rod, shown in Fig. 7, the amount of subcritical reactivity, Δk, at each power level, P, was calculated using the method of subcritical counting, described in the Appendix. The source correction factor was calculated for each power level and the average of four values is given in Table 6.

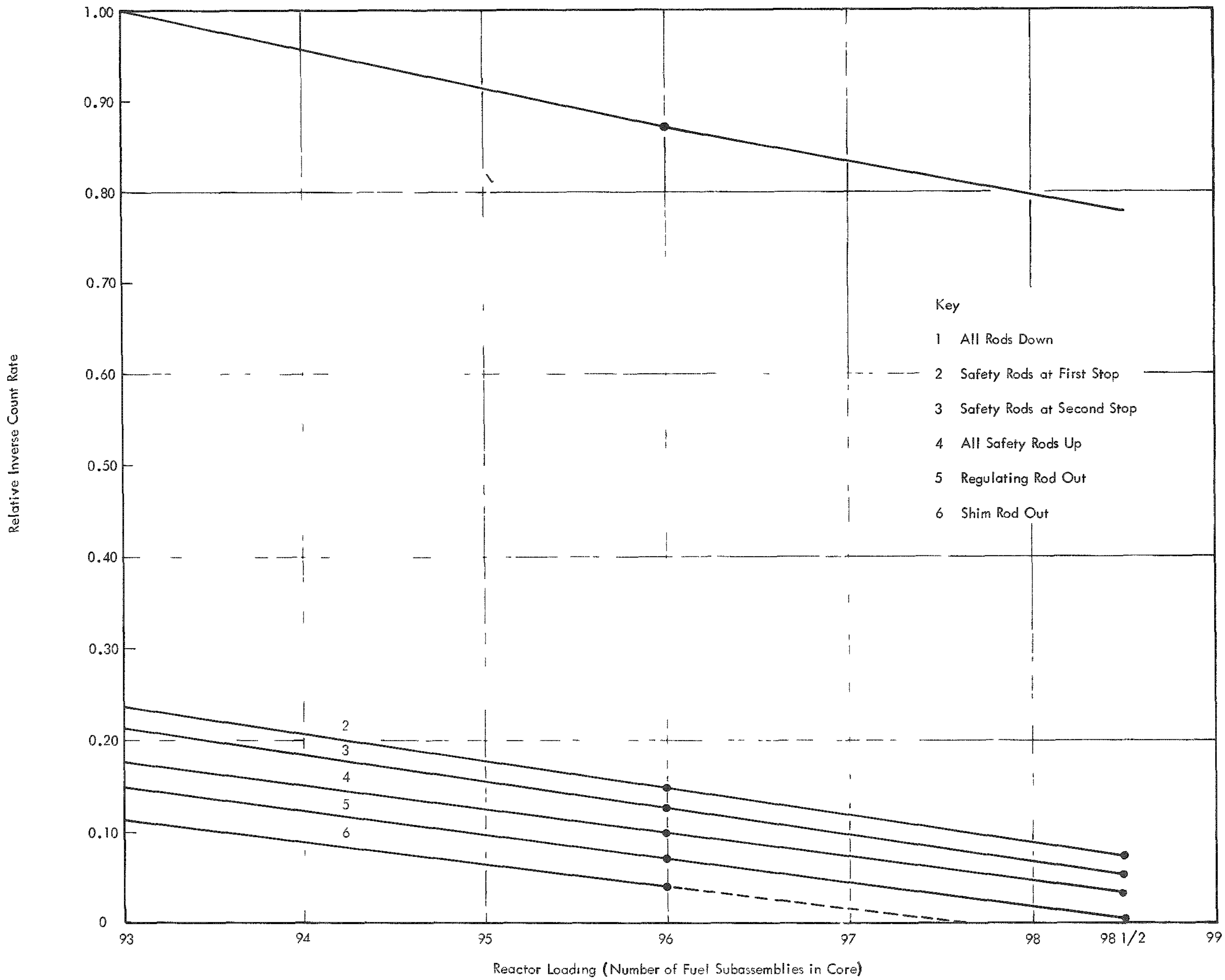


FIG. 12 RELATIVE INVERSE COUNT RATE AS A FUNCTION OF REACTOR LOADING DURING RELOAD FOR RETRACTABLE NEUTRON SOURCE

TABLE 6 - MEASUREMENTS FOR DETERMINATION OF WORTH OF RETRACTABLE NEUTRON SOURCE

Source Worth As A Function Of Power With Source Fully Inserted

Regulating Rod Position, in.	Power, P, watts*	Δk, cents*
6.17	70	4.09
6.68	131	2.35
7.16	367	0.798
7.29	1,250	0.223

* Average $P \cdot \Delta k$ = 291.4 watt-cents.

Source Worth As A Function Of Source Position And At Constant Power

Source Position, in.	Regulating Rod Position, in.	Source Worth,† cents
IN	7.26	
10	6.71	1.60
20	5.64	5.35
25	5.26	6.68
30	4.92	7.87
40	4.86	8.08
53-1/8 (OUT)	4.80	8.29

† Includes corrections for temperature and power drift.

Section III-C and Table 4 give the source correction factor as 234 $\pm$ 7 watt-cents for the source at the core-blanket interface. Preliminary calculations had predicted an increase in source effectiveness of a factor of about 3 in moving it to center position P03-P00. By the time this test was run, the source factor had decreased due to decay of the radioantimony to a measured value of 110 watt-cents, and the observed factor of increase was 2.6.

This calculation assumes two important calibrations. Absolute reactor power was determined on the basis that a reading on the picoammeter recorder channel of 6.0×10^{-9} picoamperes equals 1 kw of reactor power. Further, over the range of 5 to 7.5 in. of regulating rod withdrawal, it was assumed that the regulating rod was worth 3.5 cents per in., $\pm$ 2 per cent (see Fig. 7).

Reactivities of the retractable source as a function of source withdrawal position are given in Table 6. This test was run at a constant reading on the picoammeter of 0.75×10^{-8} picoamperes, or 1250 watts, to minimize the power-dependent source neutron contribution. At that power, a systematic correction of -0.23 cents was applied to all reactivity values as the source correction.

The worths were calculated, again, from the worth curve of the regulating rod. The values given assume that the regulating rod is worth 3.5 cents per in., $\pm$ 2 per cent, and they were corrected for temperature and power drift. A graph of the worth of the retractable source versus source position is shown in Fig. 13; this resembles the withdrawal curve for a safety rod.

The reactivity effect of completely withdrawing the antimony rod was calculated as a function of complete withdrawal of a standard safety rod (containing 535 grams of B-10) from the same position.[2] The resulting fraction of the safety rod worth was 1/13.5.

Safety rod No. 4, located in the mirror image of safety rod position No. 5 (where the antimony rod was located), contained 546 grams of B-10 and its worth was measured to be 1.19 dollars. Therefore, a standard rod would be worth 1.17 dollars and the predicted antimony rod worth would be

$$1.17 \text{ dollars}/13.5 = 8.6 \text{ cents}$$

The worth of the antimony rod at complete withdrawal, (as determined from Fig. 14 and Table 6), is 8.3 cents, in close agreement with prediction.

An investigation of sources of error revealed that errors arise from the source contribution to reactivity, $\pm$ 1 inhour; positioning the antimony rod, $\pm$ 0.2 inhour; measurement of reactivity $\pm$ 1.1 inhours, rod position error, $\pm$ 0.4 inhour; power drift, $\pm$ 0.1 inhour; and temperature drift, $\pm$ 1.0 inhour. The most probable error is assumed to be the square root of the sum of the squares of the errors due to source contribution, position of

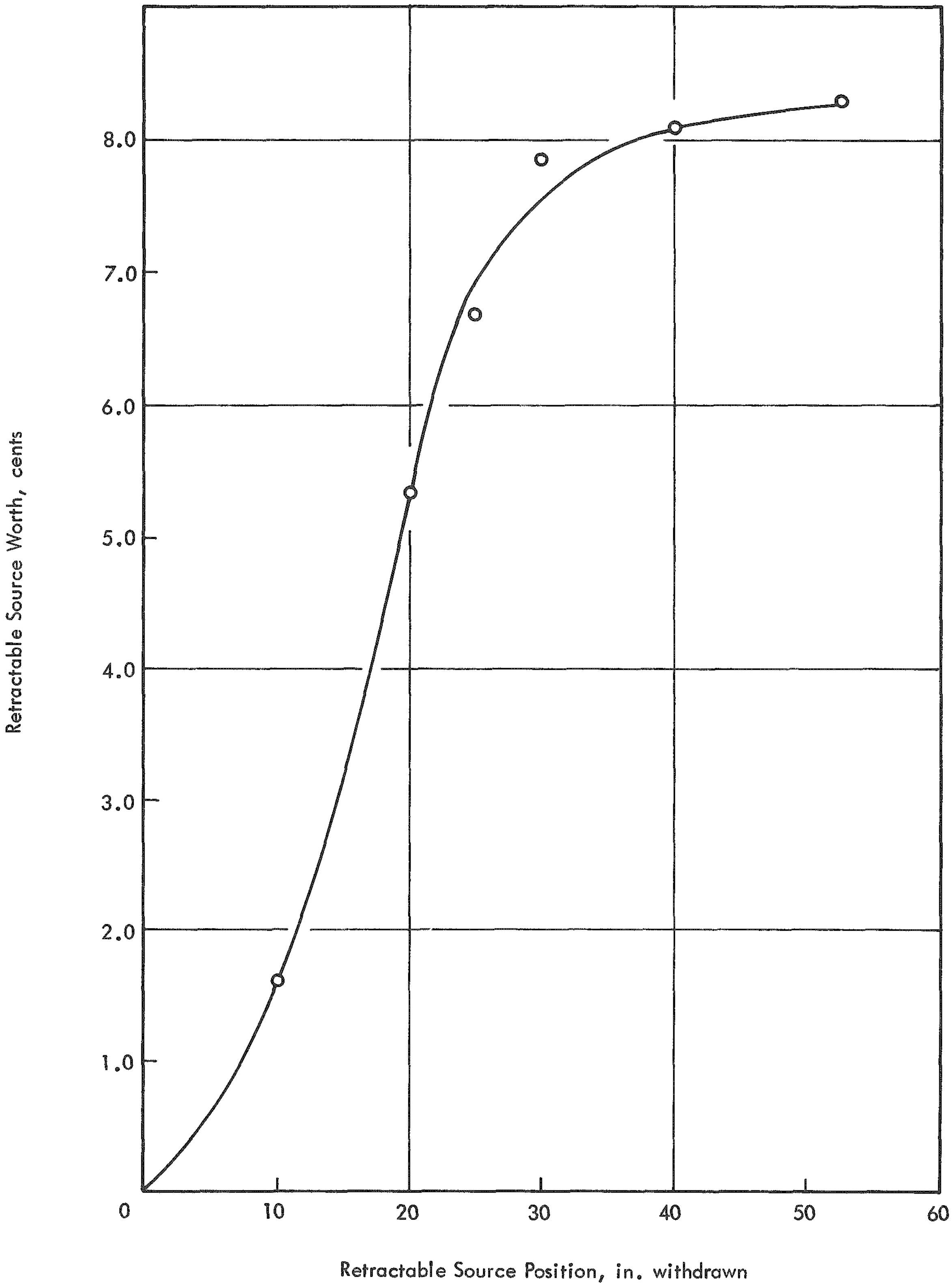


FIG. 13 REACTIVITY WORTH OF RETRACTABLE NEUTRON SOURCE AS A FUNCTION OF POSITION

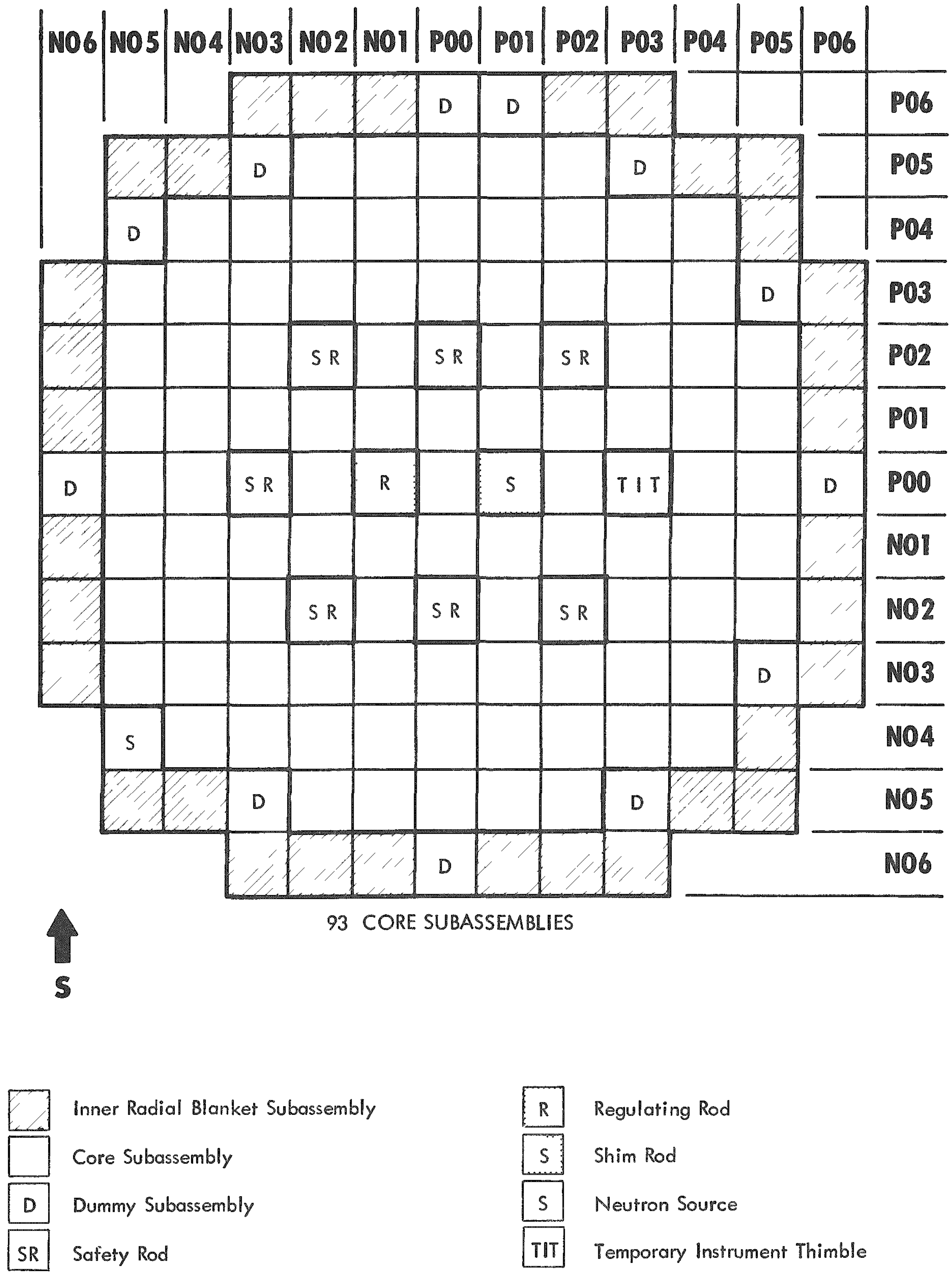


FIG. 14 REACTOR LOADING I FOR SUBCRITICAL COOLANT FLOW RATE TEST

the antimony rod, and reactivity measurements, with reactivity error being included twice, since it is present in the reference measurement. The computed value of probable error in this section of the test is $\pm$ 1.9 inhours, or $\pm$ 0.6 cent.

D. CONCLUSIONS

The measured value of source worth agrees quite well with predicted values at constant position and at constant power and variable position. With an overall reactivity increase of only + 8.3 cents upon complete withdrawal, the antimony rod withdrawal presents no significant safety problem to operation of the reactor and further use of the retractable neutron source.

V. REACTIVITY EFFECTS OF COOLANT FLOW RATE AND PRIMARY SYSTEM COVER GAS PRESSURE

A. PURPOSE OF TEST

The purpose of this test was to investigate possible reactivity effects resulting from changes in the primary coolant flow rate or the primary system cover gas pressure.

The measurements in which coolant flow rates were changed were intended to determine whether gas was introduced into the reactor coolant by operation of the overflow pump. Further, they were intended to detect any reactivity effects which may have been caused by structural deformation of the core or core support structure resulting from the flow-rate-dependent pressure differential across the core. This test was conducted primarily because gas entrainment problems had been experienced earlier at the British Dounreay fast reactor and the Russian BR-5 fast reactor.

All measurements in the test for investigation of cover gas pressure were made at one constant coolant flow rate. This was done to determine whether an apparent negligible or zero reactivity effect might have resulted from two compensating effects in the first test. Although improbable, the reactivity change due to compression of entrapped gas in the coolant could have been offset by structural deformations due to changes in primary coolant flow rate. These measurements, therefore, were made to investigate the gas effect by eliminating any flow rate consideration. Also, they would determine the volume of any entrapped gas by noting the reactivity effect of compressing it, and any other effects varying with the primary system cover gas pressure would be detected.

B. EXPERIMENTAL PROCEDURE

1. Apparatus and Equipment

The measurements involving coolant flow rate effects were made both at subcritical and at critical loadings. Thus, power level detectors consisted of a fission chamber in the temporary instrument thimble (see Section II-F-2), located in safety rod position No. 5, as well as three high-sensitivity BF_3 proportional counters and one ionization chamber, all located in NCT-3 and NCT-4 (see Fig. 4). The fission chamber and proportional counters were connected to mechanical scalers located in the control room and the ion chamber was connected to a picoammeter recorder channel, on which was recorded period and power drift data.

The primary sodium temperatures were monitored using iron-constantan thermocouples and platinum resistance temperature detectors, located to measure core and reactor inlet and outlet temperatures. The temperature data were read with high precision at a temporary readout station in the control room by connecting the thermocouples and temperature detectors to a high-sensitivity potentiometer and a resistance bridge, respectively. The locations and specific functions of the temperature detectors are discussed in detail in Section II-F-3 and are shown in Table 2.

The primary coolant flow rates were determined with the normal, permanent, plant instrumentation consisting of six magnetic flowmeters, which were read to the nearest 0.05×10^6 lb/hr. The shim rod and regulating rod positions were determined to the nearest 0.01 in., by use of Gilmore digital-readout position indicators. The sodium level in the reactor during the measurements, as measured in the sodium overflow tank, was recorded on an automatic level recorder to the nearest 0.1 ft.

The measurements involving pressure effects upon reactivity generally used the same instrumentation. However, since all the latter measurements were made at a critical loading, the fission chamber in the temporary instrument thimble was not used.

The reactor cover gas pressure for all measurements was regulated to within $\pm$ 0.5 in. of water. A special pressure indicator was employed for this test, having an overall accuracy of $\pm$ 0.5 psi.

2. Reactor Plant Conditions

The measurements to determine coolant flow rate effects on reactivity were made under the following conditions.

The permanent neutron source was at its normal core-blanket interface position, N05-N04, and the temporary instrument thimble was in safety rod position No. 5, P03-P00 (see Fig. 2).

The primary sodium flow rate was maintained at each of two different extreme values during the measurements: (1) 200-Mwt design flow (2.95×10^6 lb/hr/loop) and, (2) pony motor flow (0.32×10^6 lb/hr/loop). The primary system temperature was maintained at an isothermal condition of approximately 420 F, and all auxiliary systems such as the secondary sodium and the feedwater systems were operated as needed to maintain the desired isothermal condition.

The primary coolant flow rate scram systems were rendered inoperative for these measurements, and the intermediate and power range level scram systems were set at a flux level corresponding to a power of approximately 1 Mwt.

The primary system was maintained at an isothermal condition of approximately 517 F for the pressure coefficient of reactivity measurements, and the primary coolant flow rate was 1.55×10^6 lb/hr/loop (refueling flow rate) for three-loop operation.

The reactor was loaded such that it was possible to achieve criticality with the safety rods fully withdrawn, the shim rod fully inserted, and the regulating rod located in the middle third of its travel. This loading permitted the shim rod to remain at a constant location throughout the test.

Since the primary system cover gas pressure was to be raised 10 psi above its normal operating pressure of 15.1 psia, a check was made of the conditions of the seals in the primary system. Such maintenance and preparation as were feasible were performed to minimize the probability of failure upon an increase of pressure.

The primary coolant flow rate scram was set at 40 per cent of 200-Mwt design flow, and the intermediate and power range level scrams were set at a flux level corresponding to a power of approximately 1 Mwt.

3. Description of Measurements

a. Coolant Flow Rate Tests

A very large reactivity effect was conceivable before the test began for primary coolant flow rate. Therefore, the sequence of measurements was made with all safety rods inserted. When only a very small effect was found to be the case, the measurements were taken with all safety rods fully withdrawn, a condition which considerably enhances the accuracy of most measurements.

The test for reactivity effect due to flow rate variation was made with the reactor subcritical and with the reactor critical. The test for reactivity effect due to a change in primary system cover gas pressure was made only with the reactor critical. Two subcritical loadings were used, containing 93 and 97 subassemblies, respectively. The critical loading contained 99-1/2 subassemblies. The loadings are shown in Figs. 14, 15, and 16. At each loading during the approach to criticality the test was begun with the primary sodium flow rate at the 200-Mwt design flow of 2.95×10^6 lb/hr/loop. All safety rods and the regulating rod were fully withdrawn and the shim rod was fully inserted. The plant was allowed to settle-out to an isothermal condition of approximately 420 F. This was insured by logging temperature readings from each detector at 5-min intervals until two successive temperatures at any one point and also the ΔT across the reactor did not vary more than 2 F.

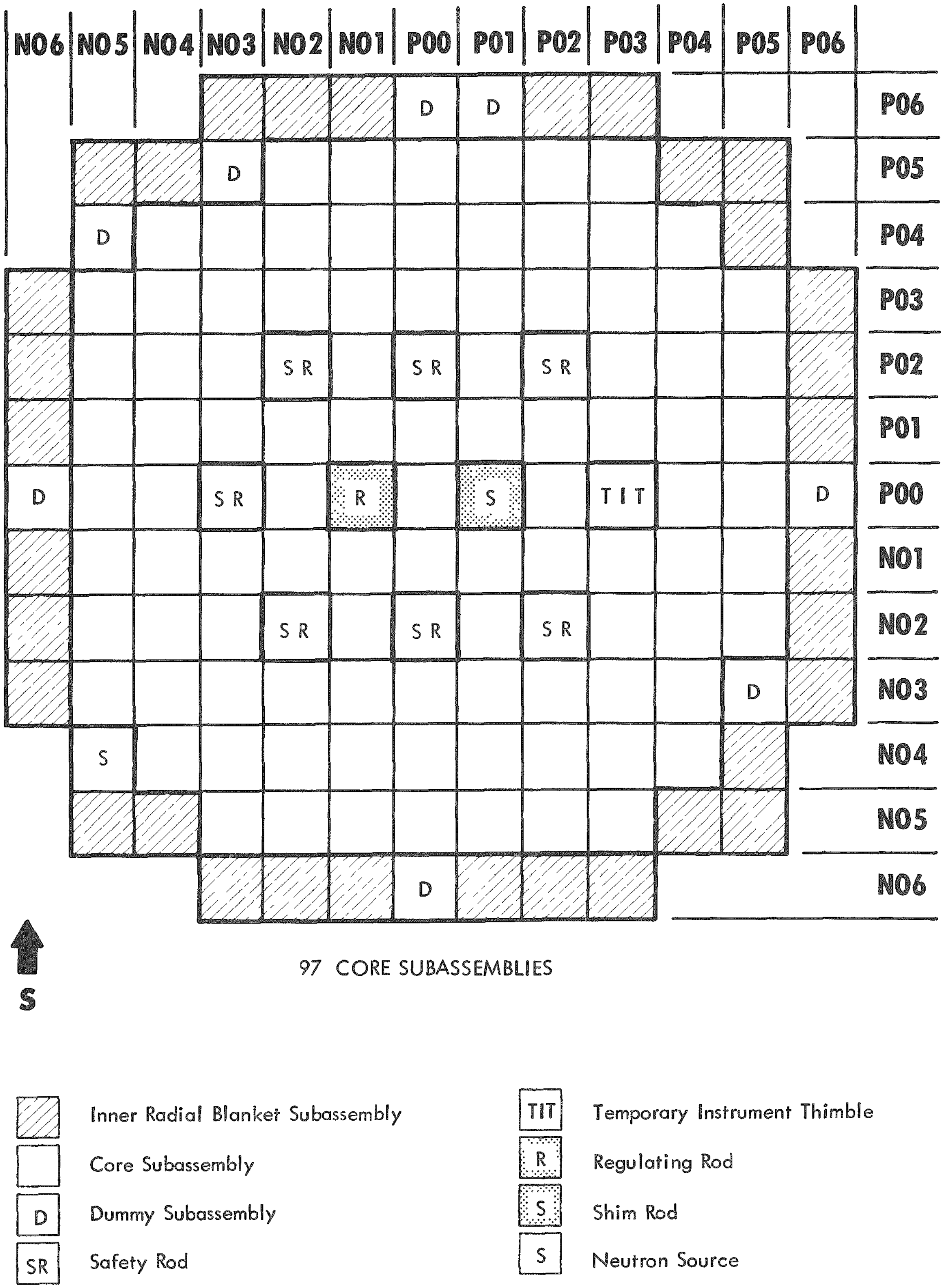


FIG. 15 REACTOR LOADING II FOR SUBCRITICAL COOLANT FLOW RATE TEST

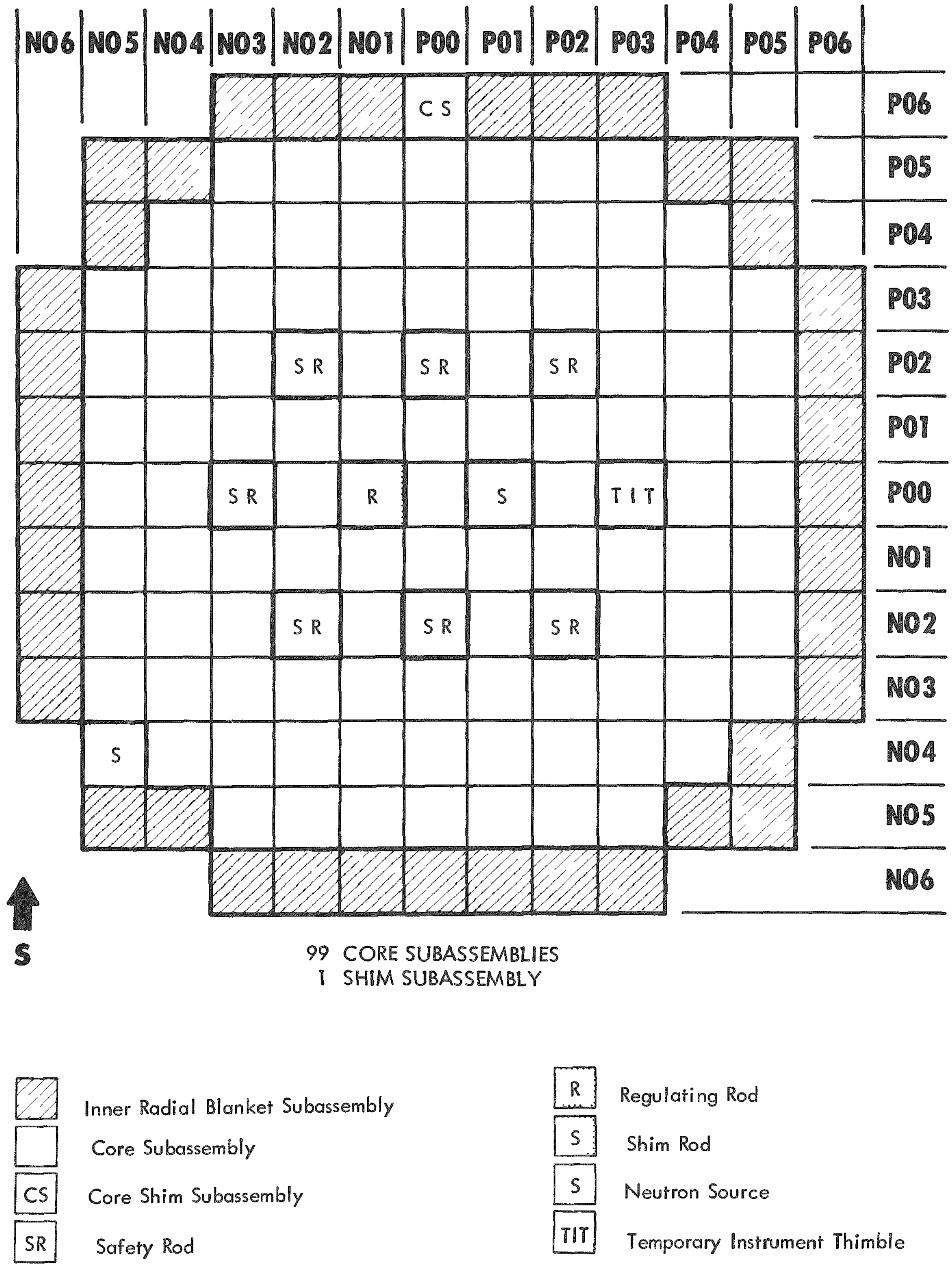


FIG. 16 REACTOR LOADING FOR CRITICAL COOLANT FLOW RATE AND PRESSURE TESTS

When these conditions were met the following data were logged:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperatures
- Primary sodium flow rate
- Reactor sodium level
- Reactor cover gas pressure
- Neutron counts from four scalers and from the fission chamber
- Safety rod, shim rod, and regulating rod position.

All the rods were then inserted, the plant was allowed to settle-out, and the entire set of measurements was relogged. Ultimately, the above measurements were made at eight configurations, summarized in Table 7. Any change in reactivity from one configuration to another was determined by the method of subcritical multiplication, discussed in detail in the Appendix.

TABLE 7 - CONFIGURATIONS FOR TESTS -- SUBCRITICAL COOLANT FLOW RATE

Test No.	Primary Flow Rate	Overflow Pump Status	Rod Positions
1	Full	Off	Shim In, Others Out
2	Full	Off	All In
3	Full	On	Shim In, Others Out
4	Full	On	All In
5	Pony Motor	On	All In
6	Pony Motor	On	Shim In, Others Out
7	Pony Motor	Off	All In
8	Pony Motor	Off	Shim In, Others Out

The critical test was begun by adjusting the reactor to a power level of approximately 1 kw, with full primary sodium flow and with the overflow pump off. Under these conditions, sodium flow rates, sodium temperatures, control rod positions, and flux level and drift rate were logged. This sequence was repeated with the overflow pump operating, and then at the same power level but with primary sodium flow reduced to pony motor flow rate. Any change in reactivity was noted as a change in critical rod position, provided that any changes in rod position were first corrected for flux drift and temperature changes.

b. Pressure Coefficient of Reactivity

The test for a reactivity effect due to variation of the primary system cover gas pressure was performed at an isothermal condition of 517 F and a refueling flow rate of 1.55×10^6 lb/hr/loop at three-loop operation. Once again, the isothermal condition was considered to have been reached when two successive temperature readings at any one point, taken 5 min apart, did not vary by more than 2 F, and when the ΔT across the reactor was 2 F or less.

The test was begun at the normal cover gas pressure of 15.1 psia. The reactor was brought to criticality at a power level of approximately 1 kw and allowed to settle-out to the desired isothermal condition. All safety rods were fully withdrawn, and the shim rod position was held constant throughout the entire test. The overflow pump was not operating. The loading used for this test is shown in Fig. 16.

When equilibrium was reached, the following data were recorded:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperature
- Primary sodium flow rate
- Reactor sodium level
- Reactor cover gas pressure
- Reactor flux level and drift as indicated by the picoammeter recording channel
- Shim rod, regulating rod, and safety rod positions

The regulating rod was then withdrawn to place the reactor on a positive period of approximately 85 sec. At this point the entire list of data above was again recorded, with reactor period replacing reactor flux level and drift rate. The procedure was repeated with the overflow pump operating, to determine if the operation of the overflow pump introduced any extra gas into the system under refueling flow conditions.

The reactor was then shut down for ten days for maintenance and the test at increased primary system cover gas pressure was conducted at the end of this period. The effect of this delay is discussed in Section V-C-2.

Following start-up of the reactor again, the cover gas pressure was raised to 10 psi above the normal operating pressure of 15.1 psia. The entire test was then repeated as described above, the conclusion of which completed the test.

C. EXPERIMENTAL RESULTS AND ANALYSIS

1. Coolant Flow Rate Tests

a. Subcritical Measurements

The data for the subcritical measurements are presented in Table 8. The in-core neutron-counting channel was a fission chamber inside the temporary instrument thimble, and the proportional counters and ion chamber discussed in V-B-1 comprised the remaining four neutron-counting channels. The channels were used to measure counts for two 5-min periods. These were then combined to present the averaged value of counts for 1 min from each channel.

The test was conducted with the shim rod fully inserted. The regulating rods and safety rods were ganged, i.e., their positions and movements were identical at all times.

Using the method of subcritical multiplication to determine reactivity (see Appendix), the amount of subcritical reactivity at each flow rate and overflow pump configuration was determined from the count rates from each channel. The five values for each configuration were then averaged to give an average subcritical reactivity for that particular configuration.

The apparent reactivity effect for a desired change was determined by appropriately combining the subcritical reactivities for each configuration. These results are shown in Table 9, and sources of error present in the measurements and calculations are discussed below. The combinations which were studied allowed investigation of flow rate variation with and without overflow pump effects, and also the effect of overflow pump operation at each extreme flow rate.

b. Critical Measurements

The data for the critical test are presented in Table 10. The subcritical tests were performed to insure that no large reactivity effect caused by flow rate variation would compromise safety considerations. When this was found to be the case, the critical measurements were made with the safety rods fully withdrawn. The shim rod remained fully inserted for this portion of the test.

TABLE 8 - EXPERIMENTAL DATA FOR TESTS -- SUBCRITICAL COOLANT FLOW RATE

No. of Subassemblies In Reactor	Primary Flow Rate	Overflow Pump Status	Safety Rod and Regulating Rod Positions*	Averaged Count Rates, counts per min†				
				Channel 1	Channel 2	Channel 3	Channel 4	In-Core
93	Full	Off	In	6,186	6,800	4,367	26,545	140
			Out	43,685	48,076	28,067	186,776	974
		On	In	6,209	6,843	4,328	26,595	144
			Out	43,444	47,794	27,946	186,316	982
	Pony	On	In	6,194	6,847	4,315	26,580	134
			Out	43,520	47,862	27,934	186,191	979
		Off	In	6,191	6,814	4,318	26,561	141
			Out	43,473	47,922	27,996	186,355	988
97	Full	Off	In	756	845	4,793	3,629	164
			Out	12,583	13,790	76,752	53,916	2,675
		On	In	763	831	4,954	3,279	165
			Out	12,502	13,736	76,273	53.448	2,696
	Pony	On	In	766	846	4,970	3,306	170
			Out	12,535	13,756	76,738	53,758	2,706
		Off	In	766	837	4,961	3,218	162
			Out	12,615	13,851	77,056	54,164	2,746

* All measurements taken with shim rod fully inserted.

† Measurements were taken for two 5-min intervals.

TABLE 9 - REACTIVITY CHANGES FOR TESTS -- COOLANT FLOW RATE

Number of Sub-assemblies In Reactor	Primary Flow Rate		Overflow Pump Status		Reactivity Change, inhours	Experimental Error, inhours
	Initial	Final	Initial	Final		
93	Full	Pony	Off	Off	+2.40	±6.4
	Full	Pony	On	On	+0.83	±6.4
	Full	Full	Off	On	-0.93	±6.4
	Pony	Pony	Off	On	-2.49	±6.4
97	Full	Pony	Off	Off	+0.72	±2.9
	Full	Pony	On	On	+1.16	±2.9
	Full	Full	Off	On	-2.20	±2.9
	Pony	Pony	Off	On	-1.81	±2.9
99-1/2 (Critical)	Full	Full	Off	On	+0.032	±1.5
	Full	Pony	On	On	-0.032	±1.5
	Full	Pony	Off	On	-0.032	±1.5

TABLE 10 - EXPERIMENTAL DATA FOR TESTS -- CRITICAL COOLANT FLOW RATE

Primary Flow Rate	Overflow Pump Status	Regulating Rod Position,* in.	Power Drift Rate, % power/min
Full	Off	8.00	- 1.0/3
Full	On	8.22	- 5.4/5
Pony	On	8.28	+ 0.76/7

* All measurements made with all safety rods fully withdrawn and shim rod fully inserted.

The ion chamber, located in NCT-4, was connected to the picoammeter recorder channel for this test, and was used to measure power and drift rate. Power was determined by assuming that a reading of 6.0 x 10^{-9} picoamperes on the picoammeter recorder channel equals 1 kw of reactor power.

The reactivity changes between configurations were determined by standard critical rod position techniques, with indicated critical rod positions first being corrected for power and temperature drift. The results are presented in Table 9.

An analysis of sources of error was made, taking into consideration counting statistics, rod positions, temperatures, and flux drift. Counting statistics errors, arising during subcritical measurements, contributed from 1.5 to 6.3 inhours, varying directly as a function of subcritical reactivity, and hence inversely as the number of subassemblies in the reactor. The error in rod position, $\pm$ 0.03 in., accounted for 0.12 inhour at most, and the temperatures, accurate to 1 F, contributed $\pm$ 0.93 inhour to the total error for subcritical measurements. Critical measurements contained errors due to flux drift as well as those errors discussed above, and it was estimated that this contribution was $\pm$ 0.12 inhour.

The experimental error was assumed to be the square root of the sum of the squares of all the individual errors, and the proper values are given in Table 9 for each loading.

2. Pressure Coefficient of Reactivity

The data for the pressure variation test are presented in Table 11. The test was conducted at two pressures and at two overflow pump conditions, and the safety rods were fully withdrawn throughout the test. Although the shim rod position was not held precisely constant, the variation was probably caused by noise fluctuations in the Gilmore position indicators and was insignificant.

Detector channels 3 and 4 were fed by proportional counters and the ion chamber was connected to the picoammeter recorder channel. The picoammeter was also used to measure power drift rates at criticality.

The reactivity change was investigated for the variations of conditions given in Table 12 and was calculated using both critical rod position and period methods. Reactivity as a function of period was calculated using the following formula:

TABLE 11 - EXPERIMENTAL DATA FOR TESTS -- PRESSURE COEFFICIENT OF REACTIVITY

Reactor System Cover Gas Pressure, psia	Overflow Pump Status	Control Rod Positions, * in.		Period Measurements, sec			Power Drift Rate, % power/ 5 min
		Regulating Rod	Shim Rod	Channel 3	Channel 4	Picoammeter Recorder Channel	
15.1	Off	19.92	6.18	87	87	86	
	On	19.94	6.20	87	85	85	
	Off	12.15	6.20				+0.65
	On	12.03	6.20				+0.33
25.1	Off	19.98	6.18	84	85.6	81	
	On	19.99	6.18	82.5	85.5	74	
	Off	12.19	6.18				+5.26
	On	12.08	6.18				+3.9

* All measurements were made with all safety rods fully withdrawn.

TABLE 12 - REACTIVITY CHANGES FOR TESTS -- PRESSURE COEFFICIENT OF REACTIVITY

Test Conditions		Reactivity Change, inhours		Experimental Error, inhours	
Pressure, psia	Overflow Pump Status	Critical Rod Measurement	Positive Period Measurement	Critical Rod Measurement	Positive Period Measurement
25.1	Off On	- 0.12	- 0.43	± 1.10	± 1.15
15.1	Off On	- 0.01	- 0.21	± 1.10	± 1.15
15.1 25.1	Off	+ 2.30	+ 2.33	± 2.35	± 2.40
15.1 25.1	On	+ 2.19	+ 2.11	± 2.35	± 2.40

$$\rho = \frac{\ell}{Tk_{eff}} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i T}$$

where,

ρ = reactivity, inhours

k_{eff} = effective multiplication factor

T = reactor period, sec

Weighted average values ℓ, λ_i, and β_i are as follows:

$\ell = 1.38 \times 10^{-7}$ sec

i	β_i	λ_i sec^{-1}
1	0.00021	0.01275
2	0.00127	0.03196
3	0.00123	0.12040
4	0.00262	0.32260
5	0.00104	1.50500
6	0.00026	3.94400

A graph of reactivity in inhours versus reactor period in seconds is shown in Fig. 17. The conversion to reactivity in cents was made on the basis that one cent equals 3.19 inhours.

An error analysis for this experiment included consideration of temperature, rod position, period, cover gas pressure, and reactor sodium level. Rod positions were accurate to ± 0.03 in., and at the maximum dk/dz this represented an error of ± 0.38 inhour. Period measurements were estimated to be accurate to 1 sec, which amounted to an error of ± 0.39 inhour; cover gas pressures, measured to ± 0.5 psi, contributed ± 0.1 inhour to the most probable error. Sodium level is important in considering the effect of the static sodium head on entrained gas. Detailed calculations[3] give a volume decrease of 0.28 liters due to increasing the cover gas pressure by 10 psi, and this corresponds to a reactivity increase of 2 inhours.

The most probable net error is assumed to be the square root of the sum of the squares of all the contributing errors, and these values are listed for each condition in Table 12. There is an increase in error value in two cases because there was actually a ten-day interval between the tests at 15.1 and 25.1 psia, during which the reactor was shut-down. After restarting, it was estimated that there was an error of 2.5 F instead of the previous 1 F in temperature readings, and this is reflected in larger experimental errors.

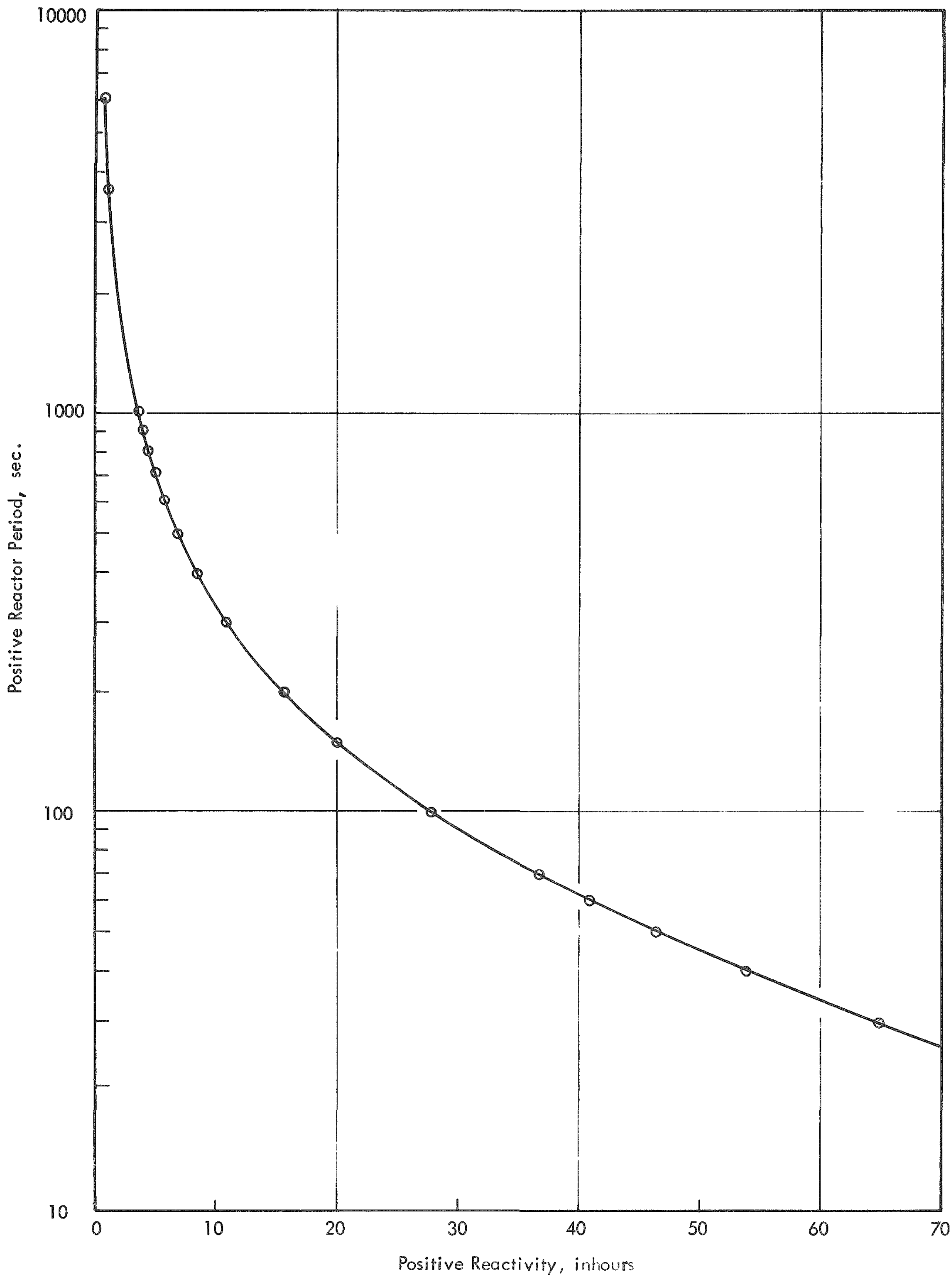


FIG. 17 POSITIVE REACTOR PERIOD AS A FUNCTION OF POSITIVE REACTIVITY

D. CONCLUSIONS

As indicated in Table 9 and Table 11, the reactivity effects for all changes fall within the limits of experimental error. From the results of the coolant flow rate tests it was concluded that the amount of gas, if any, which was introduced into the coolant by operation of the overflow pump was less than the detectable amount. Further, it was concluded that no noticeable effect occurred as a result of structural deformations in going from the low extreme to the high extreme of coolant flow rate.

This second conclusion would be valid only if certain results were found in the second test. From the results of the pressure test, it was concluded again that the overflow pump did not cause a detectable amount of gas to be introduced into the core. It was further concluded that the total amount of gas entrained in the coolant was less than the detectable amount. Therefore, in the absence of effects due to gas, it was concluded that there were no significant reactivity changes produced by structural deformation caused by flow rate variation.

VI. REACTIVITY EFFECT OF TEMPORARY INSTRUMENT THIMBLE

A. PURPOSE OF TEST

A temporary instrument thimble (TIT) was inserted into the core during the initial loading to criticality. The TIT, described in more detail in Section II-F-2, was located in safety rod position P03-P00, (see Fig. 3) where it could be withdrawn when desired, using the safety rod drive mechanism.

The major reason for installing the TIT was to be able to house in-core detectors and thus obtain enhanced counting rates at very low powers. During the initial loading to criticality, the TIT contained an absolute fission chamber and an aluminum filler-plug located immediately above the counter in the core region. The plug was included to minimize the volume available to a moderator material in the event of a leak, and also because the properties of aluminum are quite similar to those of sodium.

The TIT was designed to produce negligible reactivity effects when withdrawn from the core and replaced by sodium, and the purpose of this test was to verify the design effectiveness. This would further establish that no safety hazard would result from floating or retraction of the TIT while the reactor was operating. (Floating would have been possible only if the TIT were disconnected from its drive during full coolant flow.) Verification of an essentially zero TIT displacement reactivity effect was essential to the safety of several subsequent low-power tests involving the TIT.

B. EXPERIMENTAL PROCEDURE

1. Apparatus and Equipment

Reactor power was monitored by four high-sensitivity BF_3 proportional counters located in NCT-3 and NCT-4 (see Fig. 4) and connected to mechanical scalers. Two of these supplied count rate and period signals to the two source range channels, while the other two served as monitoring channels. A fission chamber with scaler readout was centered about the core centerplane in the temporary instrument thimble and was used to provide monitoring information.

Five ionization chambers, located in NCT-3 and NCT-4, provided signals to three intermediate range channels and to two of the three power range channels of the safety system. A sixth ion chamber, located in NCT-3, provided a signal to a picoammeter recorder channel.

The desired inlet and outlet temperatures were measured with the normal plant temperature sensors. These consisted of iron-constantan thermocouples and platinum resistance temperature detectors connected to a high-sensitivity potentiometer and a resistance bridge, respectively. These devices permitted highly accurate temperature readout to within 1F. The location and specific functions of each of the temperature sensors are described in detail in Section II-F-3 and Table 2.

The permanent plant instrumentation was used to determine flow rates and rod positions. Magnetic flowmeters capable of measuring flow rates to within 0.05×10^6 lb/hr were used to determine primary coolant flow rates in each loop, and shim rod and regulating rod positions were measured to within 0.01 in. by Gilmore digital-readout position indicators located in the control room. The position of the TIT was read from a scale attached to the drive rack in the Reactor Building.

2. Reactor Plant Conditions

The worth measurements of TIT reactivity were made shortly after initial criticality with a reactor loading of 99 fuel subassemblies. The loading is shown in Fig. 18.

The primary system temperature was maintained at approximately 405 F, with the primary sodium flow rate at approximately 2.00×10^6 lb/hr/loop. All secondary systems, such as secondary sodium and feedwater systems, were operated at those conditions necessary to maintain an isothermal condition of 405 F in the primary system.

The neutron source was located in its normal operating position at the core-blanket interface in location N05-N04, and the TIT was located in safety rod location No. 5, P03-P00.

The intermediate and power range level scrams were set at a flux level corresponding to a power of approximately 1 Mwt.

3. Description of Measurements

The test was begun with the TIT at its lowest position, or fully inserted. The safety rods were fully withdrawn and the shim rod was withdrawn 10.00 in., where it was held throughout the test. The reactor was then brought to criticality at a power level high enough to minimize source effects. (See Sections III and IV.)

The primary system temperature was allowed to stabilize at an isothermal condition of approximately 405 F, which was insured by logging temperatures at each point at 5-min intervals until the temperature at any one point and the ΔT across the reactor both were 2 F or less.

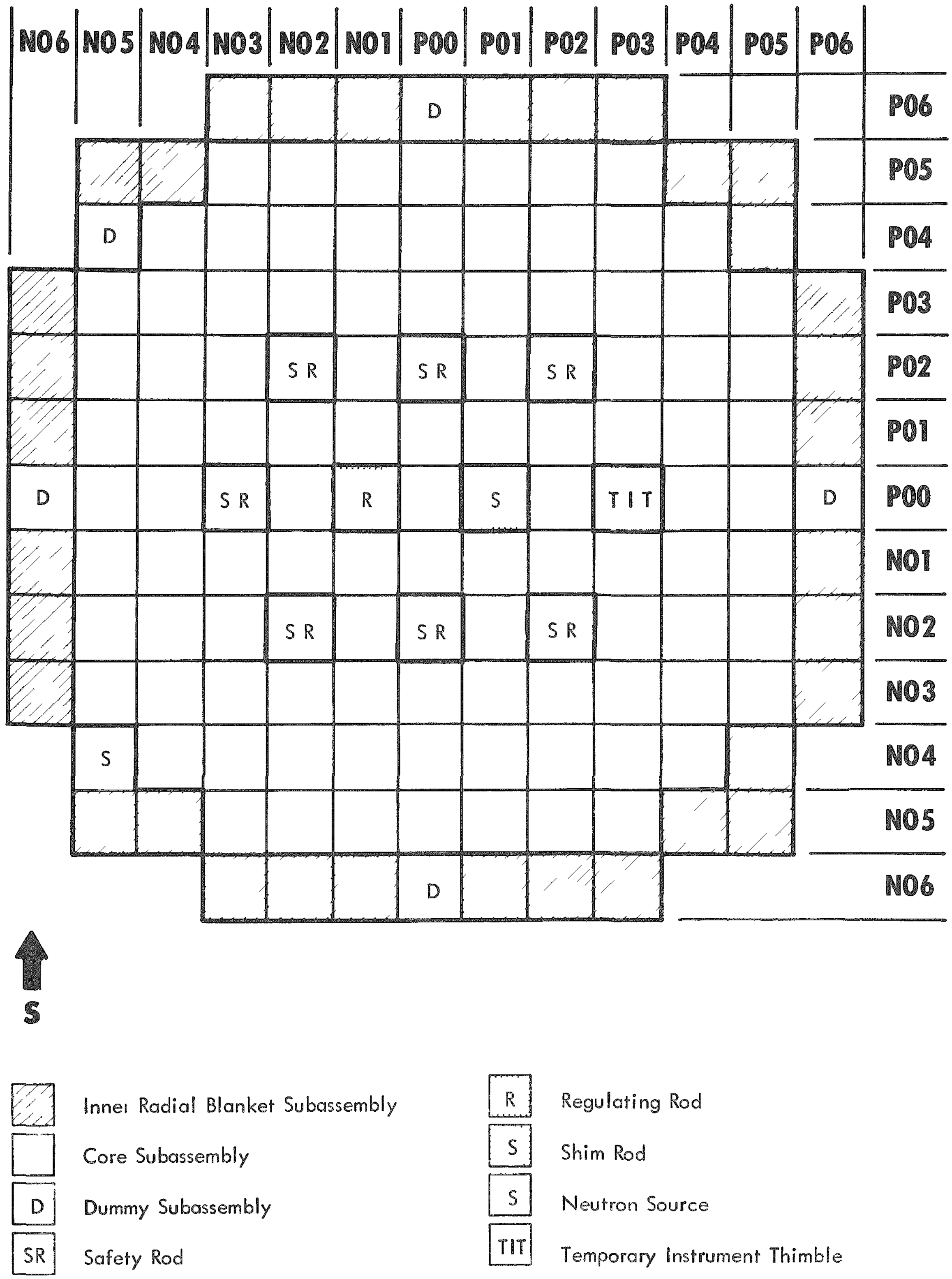


FIG. 18 REACTOR LOADING FOR TEMPORARY INSTRUMENT THIMBLE TEST

Once this condition was achieved, the following data were recorded:

- Core and blanket outlet temperatures
- Reactor inlet and outlet temperatures
- Lower support-plate temperature
- Primary sodium flow
- Shim rod, safety rod and regulating rod positions
- Reactor flux level and drift rate as indicated by the picoammeter recorder channel

The reactor was then placed on a safe positive period by appropriate withdrawal of the regulating rod. The period was measured by two of the channels being fed by proportional counters, as well as by the picoammeter recorder channel. The amount of reactivity was calculated from period measurements as well as from critical rod positions.

All rods were then inserted and the TIT was fully withdrawn to approximately 68 in. The entire test procedure was repeated, with the period measurements being made at the same regulating rod position as in the previous case. The critical rod positions were first corrected for power drift, and then used to compute one value of reactivity. Period measurements provided a check, and an average value for this configuration was compared to the value obtained for the initial configuration to determine the effect on reactivity of withdrawing the TIT. This concluded the test.

C. EXPERIMENTAL RESULTS AND ANALYSIS

The experimental data and results are shown in Table 13. All reactivity measurements were made with the safety rods completely withdrawn. The period measurements given in the table are the average of the values given by the neutron-detector channels and the picoammeter recorder channel for each run.

The reactivity worth calculated from critical rod positions assumes that over the range of 4 to 9 in. of regulating rod travel, the regulating rod is worth 3.55 cents per in., $\pm$ 2 per cent, as can be seen from Fig. 7. The relationship between reactor period and reactivity is expressed graphically in terms of positive reactor period and positive reactivity (Fig. 17).

The reactivity difference in the two TIT positions was determined for both methods of measurement (critical rod and period) and the worth of the TIT was determined to be -1.41 cents. This value was obtained as the average value of all the measurements, using the data of Table 13.

TABLE 13 - EXPERIMENTAL DATA AND RESULTS -- WORTH MEASUREMENT OF TEMPORARY INSTRUMENT THIMBLE

Temporary Instrument Thimble Position	Rod Positions, in.*		Period, sec† (Or Critical)	Reactivity Worth, cents
	Regulating Rod	Shim Rod		
Down	6.85	10.00	Critical	24.3
	8.89	10.00	129.3	7.08
	8.92	10.00	131.6	7.02
Up	7.26	10.01	Critical	25.8
	8.91	10.01	108.4	5.68

* All measurements made with all safety rods fully withdrawn.

† Period measurements given as average of picoammeter and neutron-detector values.

D. CONCLUSIONS

There is a slight increase in reactivity as the TIT is withdrawn and replaced in the core by sodium, and hence the temporary instrument thimble has a small, negative reactivity worth (-1.41 cents) when in the core. This worth, however, is not great enough to endanger the continuation of any experiments using the TIT, and any movement of the TIT can be accomplished with safety rods fully withdrawn without safety hazard.

REFERENCES

1. Mueller, R. E. et al, "Initial Loading to Criticality of the Enrico Fermi Reactor", APDA-NTS-1, May, 1964.

2. Segal, B. M., "Preliminary Evaluation of Enrico Fermi Nuclear Test Procedure No. 31, Reactivity Worth of Antimony Section of Retractable Neutron Source", APDA Internal Memorandum, October, 1963.

3. Wilber, H. A., "Enrico Fermi Nuclear Test Procedure No. 5, Flow Dependence of Reactivity", APDA Internal Memorandum, Appendix A, June, 1962.

APPENDIX: SUBCRITICAL MULTIPLICATION METHOD FOR DETERMINING REACTIVITY

When the reactor is subcritical it is frequently necessary to have a method for measuring reactivity to determine the degree of subcriticality. This appendix discusses the method used in the tests described in this report, based on subcritical count rate measurements.

If k_1 = subcritical reactivity at control rod configuration 1,
and k_2 = subcritical reactivity at control rod configuration 2,
then neutron multiplication, M, can be approximated:

$$M_1 = \frac{1}{1-k_1}, \qquad M_2 = \frac{1}{1-k_2}$$

Reactor power, P, is directly related to neutron multiplication by a constant, so

$P_1 = F \cdot M_1$ and $P_2 = F \cdot M_2$, where F is a constant.

$$P_1 = F \cdot \frac{1}{1 - k_1}, \qquad P_2 = F \cdot \frac{1}{1 - k_2}$$

$$\frac{P_1}{P_2} = \frac{1 - k_2}{1 - k_1}$$

$$\frac{P_1}{P_2} - 1 = \frac{1 - k_2}{1 - k_1} - 1$$

$$= \frac{k_1 - k_2}{1 - k_1}$$

$$\delta k = k_1 - k_2$$

$$\Delta k = 1 - k_1$$

$$\therefore \quad \Delta k = \frac{\delta k}{\dfrac{P_1}{P_2} - 1}$$, taking absolute value for convenience.

The amount the reactor is subcritical at rod position one is Δk, and δk is the difference in reactivity between the two rod positions, which is found from the rod worths. The ratio of power for the two rod positions is equal to the ratio of observed count rates. Thus, Δk can be calculated from rod worths and count rates at two different rod positions.