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Niagara Mohawk Power Corporation

Presentation to the USAEC Regulatory Staff

August 26, 1965

Contents

I. Introduction

During our last meeting of May 19, 1965 we expressed our intent to meet with you from time to time as the work progresses to submit more comprehensive reports on various aspects of the Nine Mile Point design. We are prepared today to cover two subjects which we only briefly touched upon during our last meeting: (1) An entirely in-core neutron monitoring system, and (2) Incorporation of flow restrictors in the two main steam lines leading from the reactor out of the drywell. We have also included a brief status report on the principal topics of the May meeting, the core and containment spray systems.

II. Neutron Monitoring Systems

A. General Description

The decision to use all in-core nuclear instrumentation at Nine Mile Point resulted from:

(1) Shielding effects of the large core which diminished the monitoring adequacy of instrumentation located outside of the core.

(2) Neutron attenuation caused by the thickness of the downcomer water annulus made the capability of low level out-of-vessel instrumentation systems unacceptable.

(3) Recent developments which indicate that the accuracy and monitoring capability of in-core instrumentation will equal or exceed comparable out-of-core systems.

Figure 1 illustrates the ranges and overlap of the Nine Mile Point in-core instrumentation systems. The source range monitoring system (SRM) provides the information needed for efficient and knowledgeable reactor startup and low-level operations, monitors the neutron flux level from the source level ($\sim 3 \times 10^4$ nv) to about 10^9 nv with the chambers in the startup position. A chamber retraction capability is provided such that the SRM can monitor the neutron flux level up to about 10 percent of rated power (3×10^{12} nv). The SRM also continuously displays the reactor period at these low neutron flux levels.

RANGES OF NEUTRON MONITORING SYSTEMS

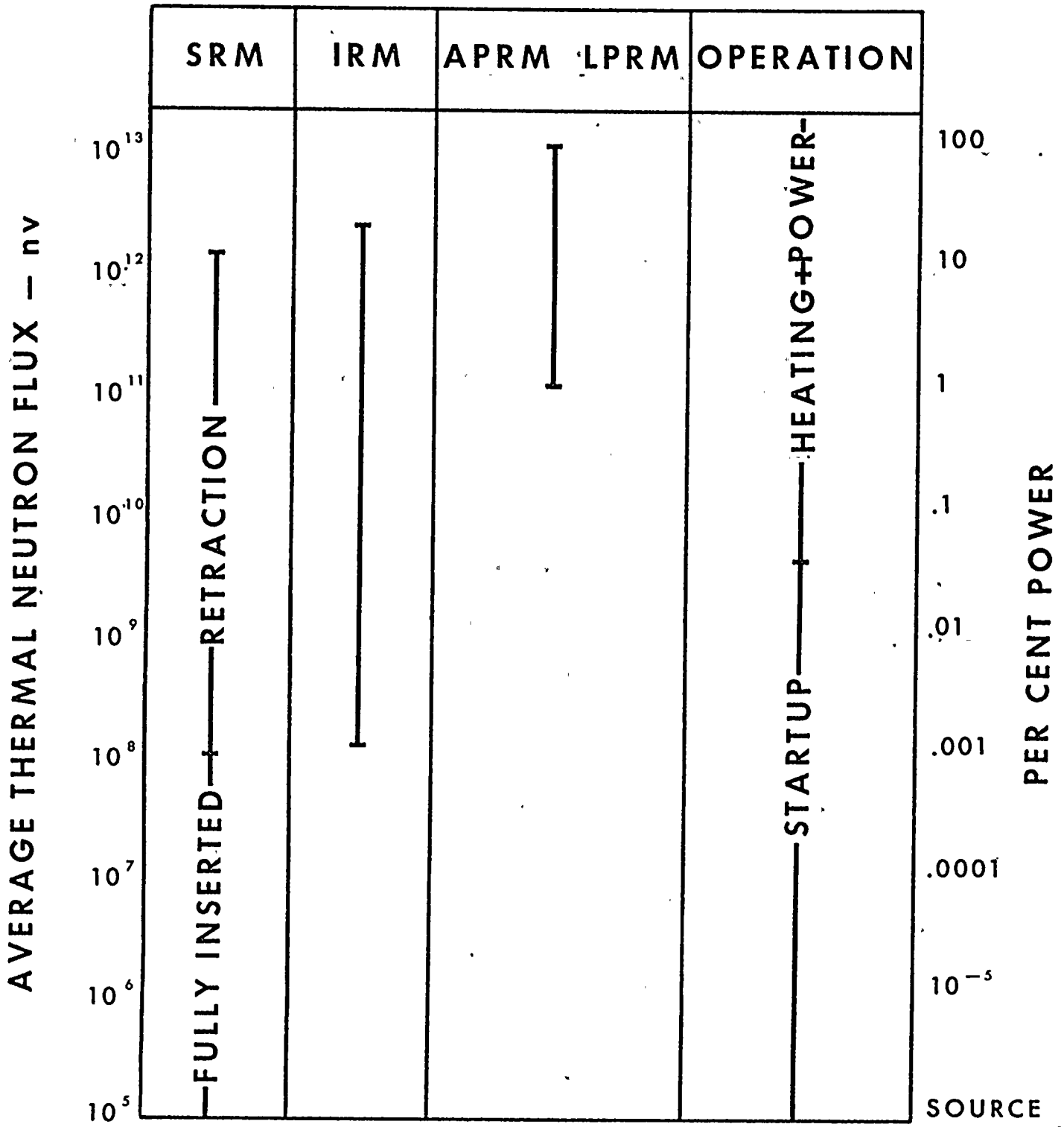

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FIGURE 1

Also shown on Figure 1 is the neutron-level coverage provided by the intermediate range instrumentation (IRM) which monitors the range between source level and power operation. This system has a level scram feature to provide the necessary automatic safety protection. The IRM range overlaps the SRM range on the lower end by about one decade, providing meaningful data at flux levels above about 10^8 nv. The range of the IRM extends to about 3×10^{12} nv or about 10 percent of rated power with the chambers in the inserted position. After the power range monitoring system is safely established on scale (about 5 to 10 percent of rated power), the IRM chambers are withdrawn to a low-flux position beneath the core using mechanical retraction mechanisms. A new feature of the IRM, described later, is the application of voltage-variance measurement techniques.

The power range system utilizes about 120 fixed in-core ion chambers similar to those used in other operating boiling water reactors. The outputs for these chambers are used for automatic safety protection in the power range as well as for power distribution and power level monitoring. Power distribution monitoring is accomplished by evaluating the output signals from the individual in-core chambers (LPRM). Power level monitoring and automatic safety protection is accomplished by averaging the output signals from selected groups of in-core chambers. This latter system of averaging

is called the average power range monitoring (APRM) system, and will be described in detail later.

The interaction of these systems can be illustrated by a typical startup sequence: Prior to withdrawal of control rods, the startup and intermediate range instrumentation chambers are inserted to the startup position and the mode selector switch is placed at START. The IRM range selector switches are set down to the lowest range. Rods are then withdrawn and the approach to critical is monitored using the signal from the SRM. Upon achieving critical, the power is allowed to rise exponentially at about a 60-second period to the heating range. During this power rise the period is monitored using the SRM initially in its fully-inserted position and subsequently in retracted positions. The IRM becomes effective before the SRM chambers are moved, and the IRM range selector switches are upranged as necessary during the approach to heating power. After heating power is reached, the SRM is retracted to the low-flux position beneath the core. After rated temperature and pressure are reached, the turbine is synchronized. When the APRM indication is above about 5 percent, the mode selector switch is switched from START to RUN and the IRM chambers are withdrawn to the storage position.

B. Source Range Monitoring System (SRM)

The design criteria and objectives require that the SRM provide sufficient information for knowledgeable and efficient reactor startup and low-level operations. Further, during initial startup the neutron emitting sources and SRM chambers will provide a minimum signal to noise ratio of 3/1, and an initial minimum count rate of 3 counts per second with all rods fully inserted. After the initial startup, these conditions will be met before the reactivity of the core exceeds the reactivity which existed at initial startup with all rods fully inserted. With all chambers operative, the SRM will show a measurable increase in output signal from at least one chamber before the neutron flux multiplication exceeds a factor of 2000 for the worst physically possible startup rod-withdrawal error. One or more chambers may be permitted out of service, since the output signals from the remaining chambers will increase substantially as the core-average neutron multiplication increases during a normal startup operation in which rods are withdrawn according to specified sequence. With the chamber in the startup position, the SRM signal will overlap the signal from the IRM as much as necessary to make insignificant the neutron level uncertainty resulting from transition between systems.

The SRM consists of miniature, neutron-sensitive fission chambers and neutron-emitting sources located inside

the core of the reactor. Figure 2 shows the tentative locations selected for the four chambers and twelve sources to be used at Nine Mile Point. Analyses indicate that this arrangement satisfies the criteria and objectives described above.

The neutron sensitivity of the detectors has been experimentally determined and found to vary with the fission product induced gamma field. During the initial startup, the power level of the reactor will be determined within acceptable accuracy using the signal from the SRM, the SRM sensitivity data and the analytically predicted nuclear characteristics of the core. After the power level has been raised to the power range, an approximate relationship between core power level and SRM output will be experimentally established.

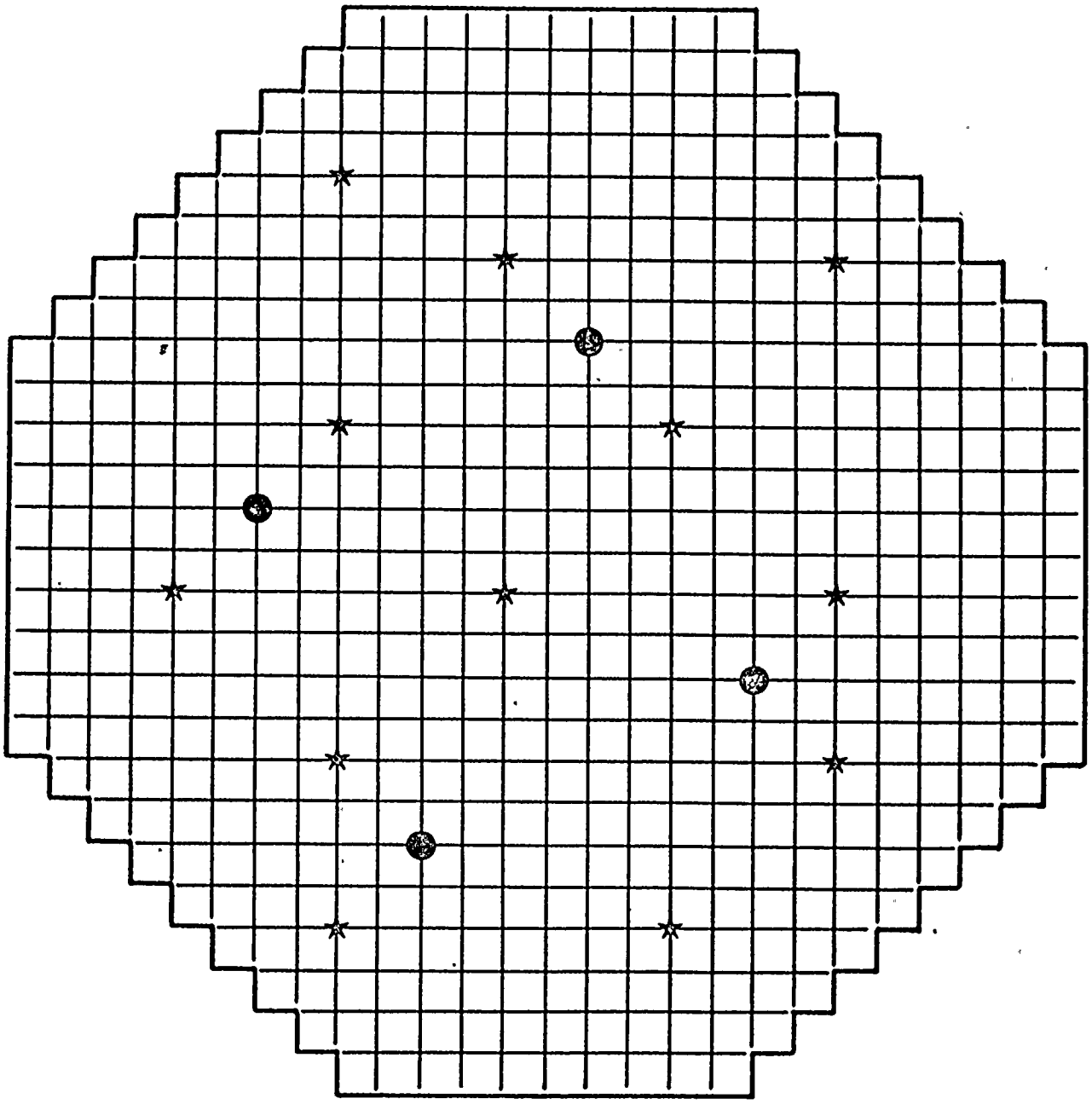
The SRM readout equipment consists of preamplifiers located in the reactor building, and meters and recorders located in the control room. The control room readout equipment displays the count rate and period to the operator. Recorders are used to provide records of the count-rate output signal. Automatic rod block signals prevent rod withdrawal for startup unless the chambers are inserted to the startup position. In addition, automatic interlock features prevent the erroneous withdrawal of the chambers into low-flux regions before the IRM is on scale.

C. Intermediate Range Monitoring and Safety System (IRM)

The design criteria and objectives require that the IRM system provide both sufficient information for safe, knowledgeable

SOURCE RANGE MONITORING SYSTEM

CHAMBER AND SOURCE ARRANGEMENT



● CHAMBERS

★ SOURCES

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FIGURE 2

and efficient reactor operations in the intermediate power range, and automatic low-level scram protection. Trip signals will prevent fuel damage due to any single operating error or equipment malfunction in the intermediate power range. For the worst possible combination of instrument bypass and startup rod withdrawal error the IRM system will generate a scram signal before the bulk fission power of the reactor exceeds 0.1 percent of rated power and before the power density of any region in the reactor exceeds 5 percent of the average power density at rated power.

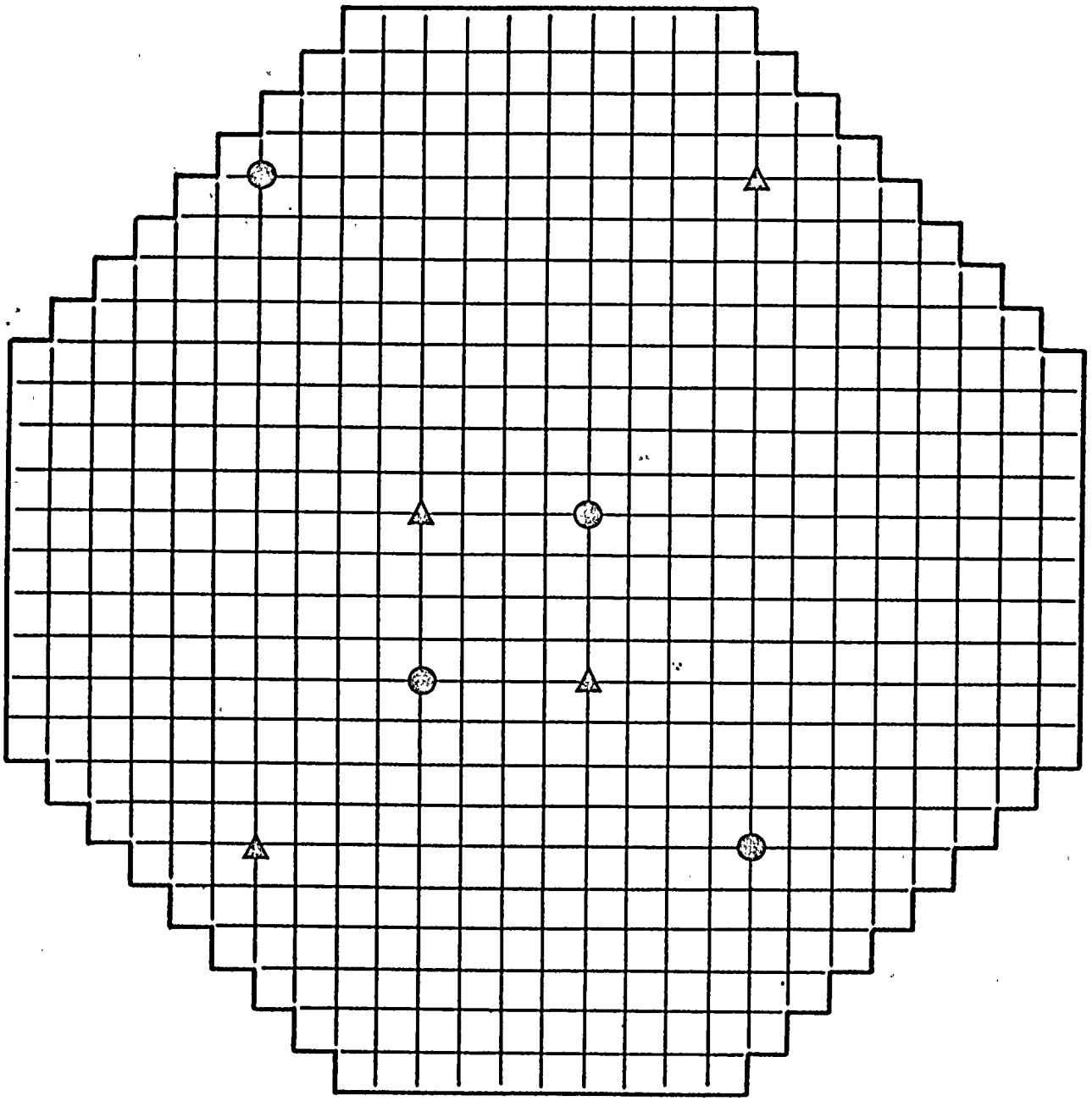
The IRM consists of miniature chambers, electronic control and readout equipment, and mechanical retraction mechanisms. Figure 3 shows the tentative location of the eight chambers in the Nine Mile Point reactor. Analyses indicate that this arrangement meets the criteria and objectives described above.

The chambers used in the IRM system are miniature fission chambers similar to the chambers used in the SRM and LPRM systems. A new measurement technique, called voltage-variance, will be used in the IRM system. This technique is based upon "Campbell's Theorem" (1). As applied in the IRM,

(1) GEAP-4747, "Theory of the Campbell System of Reactor Instrumentation", J. P. Neissel, October 1964.

INTERMEDIATE RANGE SAFETY SYSTEM

CHAMBER ARRANGEMENT



● SAFETY CHANNEL 1

▲ SAFETY CHANNEL 2

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FIGURE 3

the theorem states that the mean square of the variance of the output signal from the detectors is proportional to the fission rate in the detectors. Extensive analyses of the dynamic characteristics of the system have been performed and the validity of the analytical models have been verified experimentally. Measurements were made of the variation of output from a "Campbell" system using a scintillation crystal exposed to a rapidly varying gamma field. These experimental data were in excellent agreement with the analytically predicted output signal variation. Measurements have also been made of the neutron-flux transient during shutdown of a reactor with the "Campbell" system. The measured transient was in good agreement with the transient measured using a conventional D-C ionization system. The static sensitivity of the system has been measured for a range of neutron fluxes and for different chambers, and has proven to be predictable and well understood.

The neutron sensitivity of the chambers used in the intermediate range system has been experimentally determined, and does not vary significantly over the relevant range of reactor conditions. As with the SRM system, the power level of the reactor can be determined within reasonable accuracy during the initial startup using the signal from the intermediate range instrumentation and the predicted nuclear characteristics of the core. An experimental relationship between core power

and the IRM output signal will be determined and periodically checked as is routinely done at other operating reactors.

As shown in Figure 3, the eight IRM chamber signals are fed, four each, into both channels of the dual channel safety system. One trip in each of the channels is required to actuate a scram. We believe it is acceptable that one chamber in each of the two channels may be bypassed at any time except that mechanical interlock will prevent bypassing two chambers in a single reactor quadrant. Range switches are provided to allow the operator to uprange the system manually as necessary during reactor startup and approach to power. Downscale trips which actuate rod blocks will assure proper range switching, and upscale trips will actuate a reactor scram upon excessive level in any selector-switch range. Trips are also incorporated to prevent rod withdrawal unless the IRM chambers are inserted to the operating position.

Period information in the intermediate power range is obtained by proper retraction of the SRM chambers. Automatic scram protection against excessively fast periods is provided by the level scramming capability of the IRM. Figure 4 presents the results of measurements of the time-to-trip versus reactor period for out-of-vessel compensated ionization chamber IRM systems operating in level-scramming and period-scramming modes. The level-scramming system is faster than the period-

INTERMEDIATE RANGE SAFETY SYSTEM

MEASURED RESPONSE TIMES OF SAFETY INSTRUMENTATION

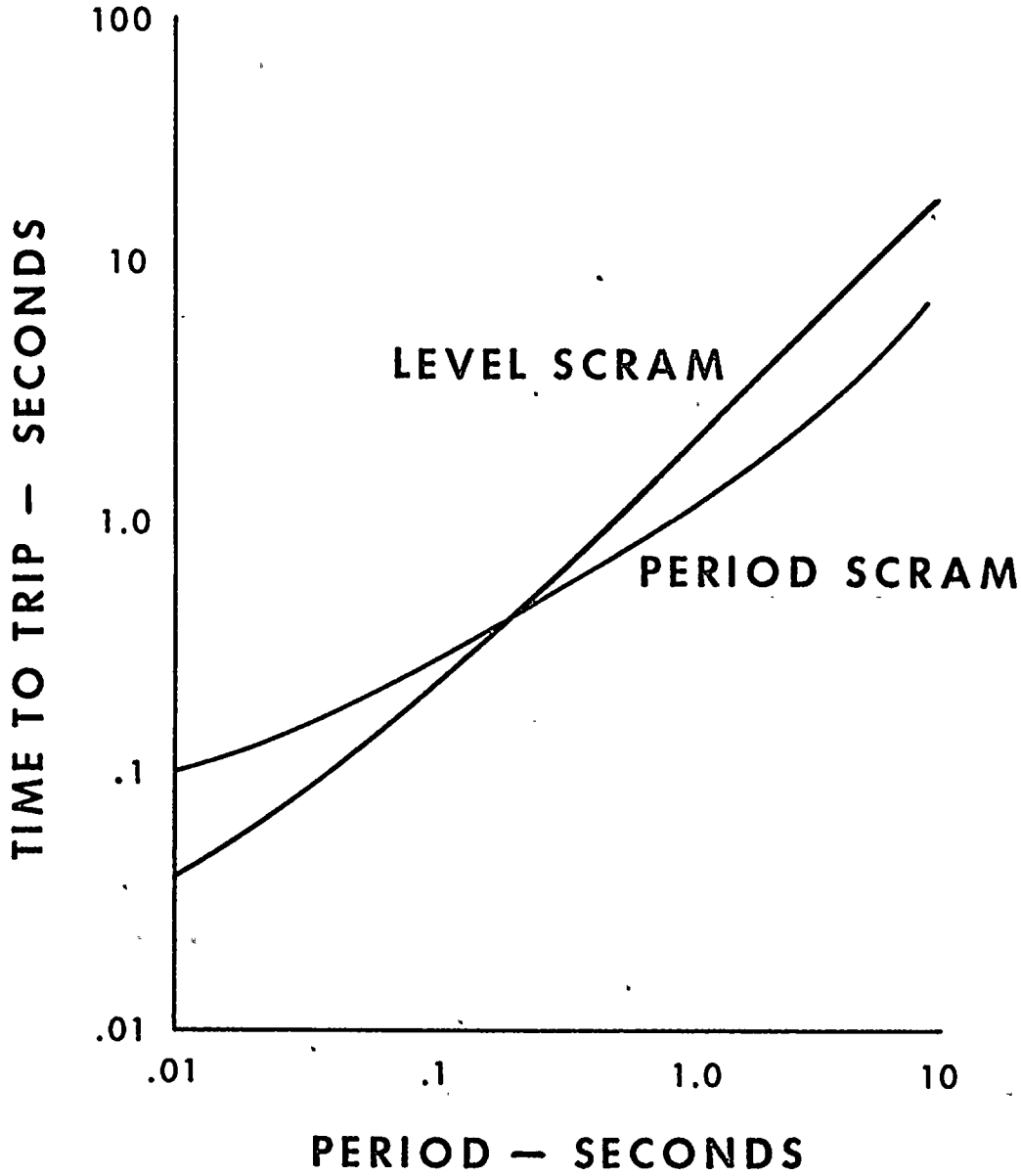

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FIGURE 4

scramming system for periods shorter than 200 milliseconds. For periods longer than 200 milliseconds, the period-scramming system is faster. Taking the worst case, the delay allows a maximum increase of about a factor of four in the power level at scram. However, a factor of four increase in the power level at scram for these longer periods is of no safety or operating significance because the peak power levels reached are quite low. Thus, the addition of period scramming capability would not significantly improve the safety of the reactor.

The mean-square-voltage preamplifiers used to condition the output signals from the IRM chambers are located in the reactor building. The other signal conditioning and readout equipment is located in the control room. The final output signal from each of the chambers is displayed and is automatically recorded. The bypass switches enabling the operator to bypass the trip signals from one chamber in each of the two safety channels in the event of equipment malfunction are located on the console.

D. Power Range Monitoring Systems

The power range instrument systems are described under three separate subheadings: Local Power Range Monitoring (LPRM), Average Power Range Monitoring (APRM) which serves as the power range safety system, and Traveling In-Core Probe (TIP) which is the system used to calibrate the power range instruments.

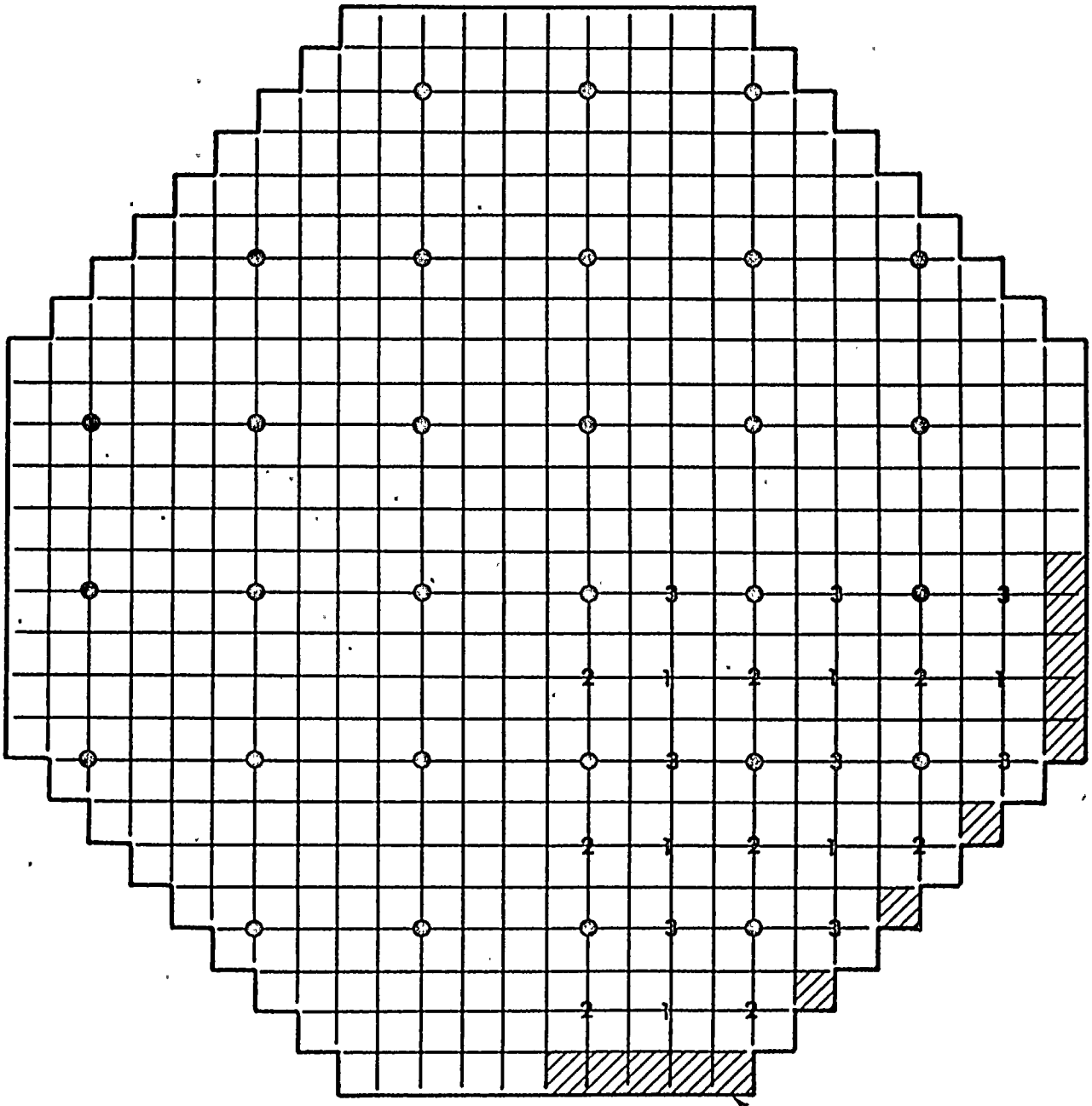
Local Power Range Monitoring (LPRM)

The LPRM system is used to monitor continuously the local neutron flux and to alarm if predetermined limits are exceeded. It is also used to obtain data which, when combined with heat balance determinations and analytical peaking factors, will permit detailed evaluation of power distributions and burnout ratios within the core. These capabilities will facilitate efficient management of the core and readily demonstrate compliance within established operating limits.

The LPRM system consists of 120 miniature ionization chambers distributed in fixed positions throughout the core, as shown in Figure 5, and associated readout equipment located in the control room. Each chamber thimble shown in plan view on Figure 5 contains four LPRM chambers at three-foot intervals to provide uniform coverage of the axial dimension. Figure 5 also illustrates by rotation of all the chamber locations to the SE quadrant that the system provides relatively complete monitoring of the core for symmetric neutron flux conditions. A calibration tube is also provided in each chamber thimble to permit scanning the axial neutron-flux distribution with the TIP calibration system. Figure 6 illustrates the detailed layout of these instruments within the core.

The readout of the individual in-core chambers, as well as the controls and readout for the TIP calibration system

POWER RANGE MONITORING SYSTEM



UNMONITORED REGION

ARRANGEMENT BY QUADRANT
1. ROTATED NW QUADRANT
2. ROTATED SW QUADRANT
3. ROTATED NE QUADRANT

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FIGURE 5

POWER RANGE MONITORING SYSTEM

DETAILED LAYOUT

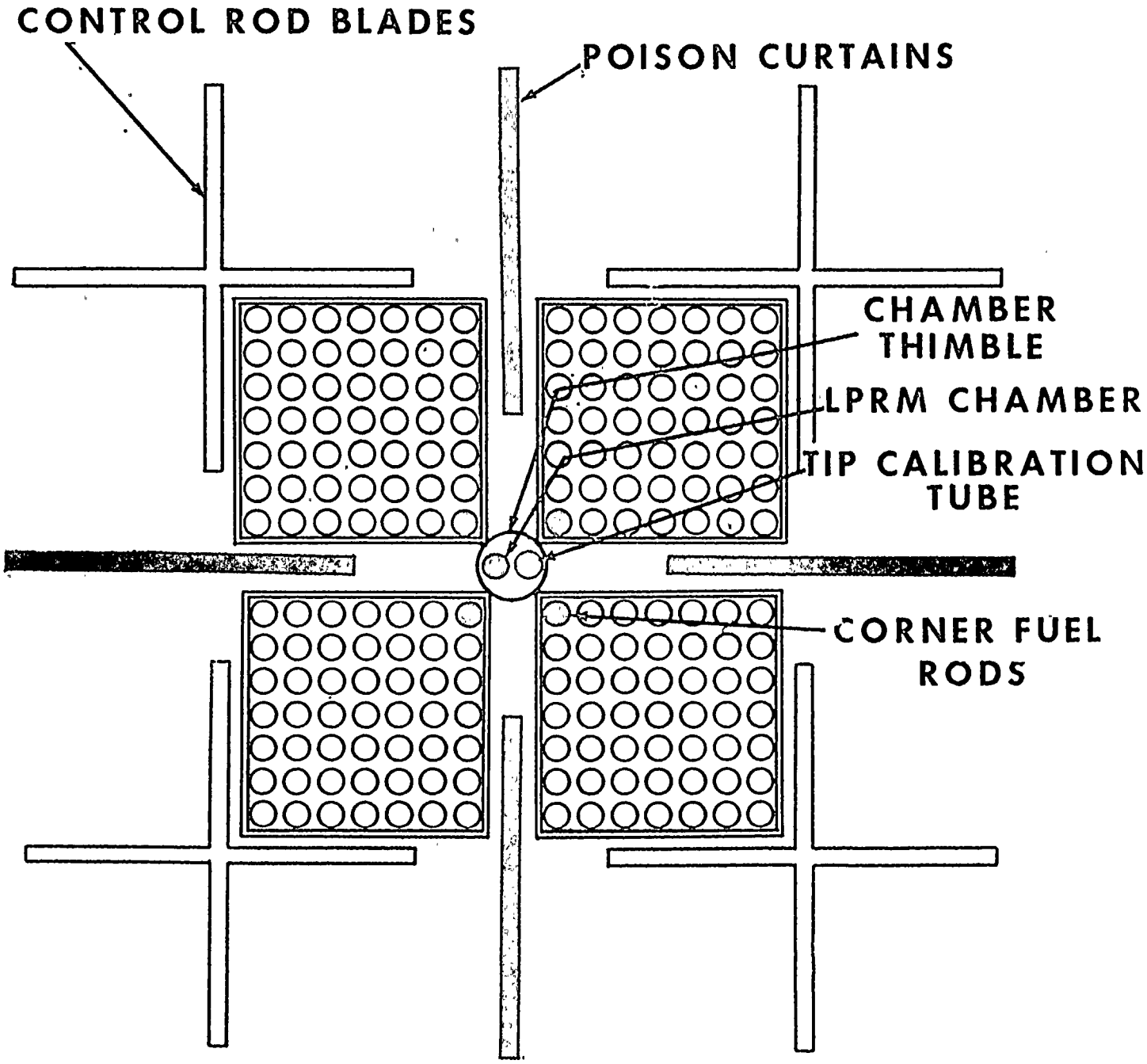

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FIGURE 6

are located in the control room to permit immediate and direct observation of core conditions by the operator. High-level trips are provided on the readout of each LPRM chamber to alarm automatically if predetermined local neutron-flux conditions are exceeded.

Average Power Range Monitoring (APRM)

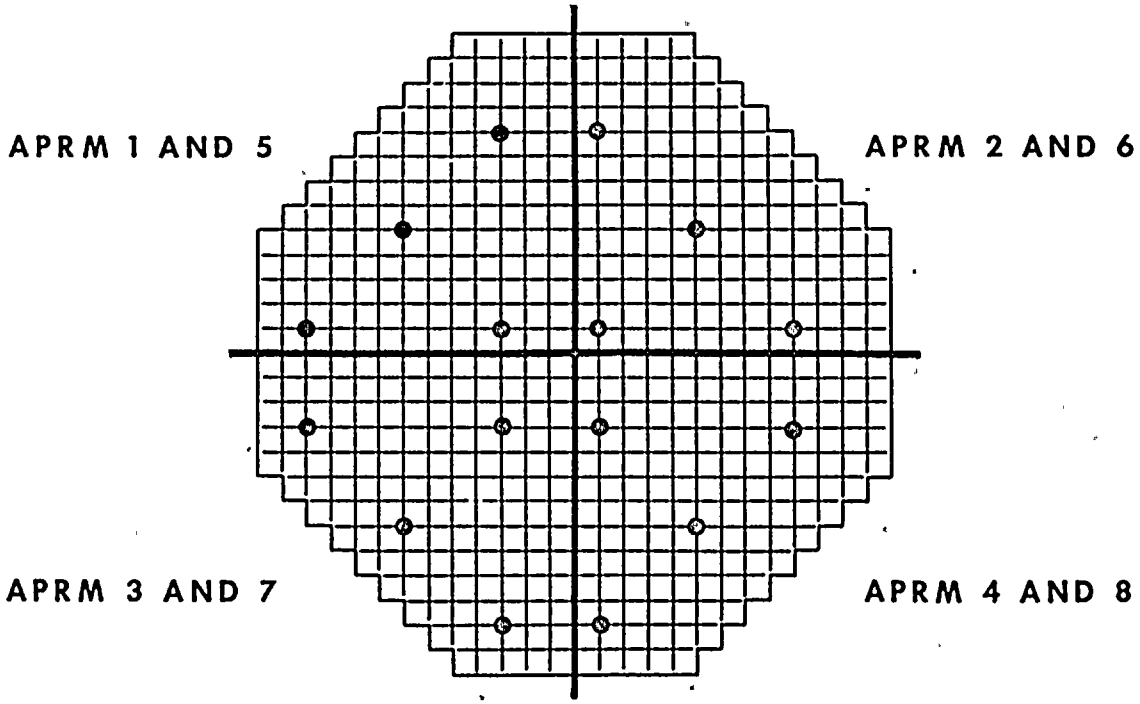
The APRM system utilizes the output signals from selected groups of LPRM in-core ionization chambers to provide an indication of the bulk thermal power of the reactor and to serve all necessary safety functions.

An important feature of the APRM system is the redundancy criteria incorporated to accommodate a reasonable degree of chamber failures without compromising reactor safety or operating efficiency. It is anticipated that two of the eight individual APRM groups provided may be bypassed except that bypassing of two groups in one core quadrant is mechanically prohibited. With two APRM groups bypassed, the applicable criteria are that the remaining system be capable of generating a scram signal during bulk neutron-flux level transients before the actual bulk level exceeds predetermined values and that fuel damage be prevented during local power disturbances caused by the worst single rod withdrawal error or equipment malfunction. These criteria apply to all permissible power and flow conditions.

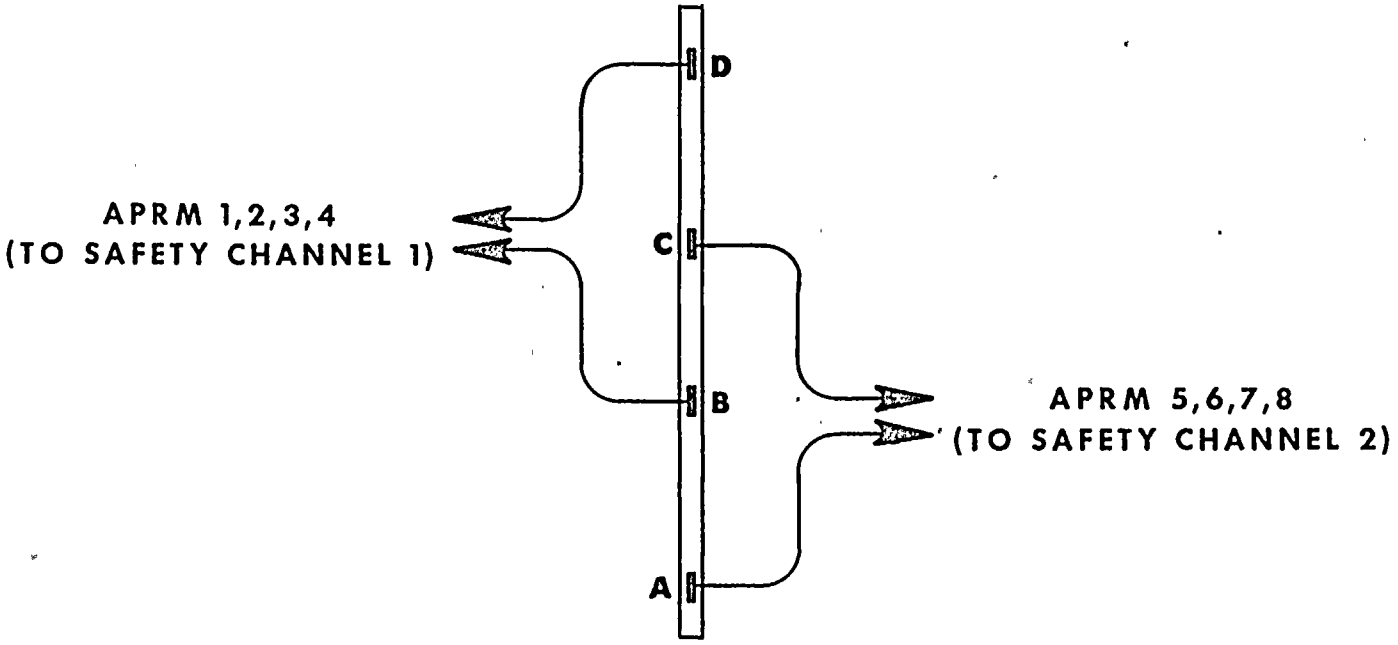
Figure 7 shows the plan whereby 64 LPRM in-core chambers are arranged into eight groups of eight chambers each. Figure 8 illustrates the averaging system employed to produce the eight individual output signals which feed into the dual-channel safety system, four to each separate channel. Analyses have been performed of APRM system behavior during normal bulk power-level maneuvering with both flow control and control rods. The results of these analyses are shown in Figures 9 and 10.

Two important safety functions are provided: A reactor scram on excessive bulk power-level transients and an upscale trip to automatically block control-rod withdrawal upon excessive local power conditions. The scram function is predicated upon tripping of one APRM group in each channel of the dual-channel safety system. It is designed to produce a scram signal before the actual bulk neutron level of the reactor exceeds 120 percent of rated at rated flow or the value which results in an adequate margin under partial flow conditions. Upscale trips on each of the eight individual APRM groups are used to block automatically control-rod withdrawal if the set point is exceeded. No coincidence is required. This trip is automatically varied with flow as necessary to meet the design criteria under all permissible flow and power conditions. Figure 11 illustrates a typical variation of the upscale trip points with flow.

ARRANGEMENT OF CHAMBERS FOR APRM



PLAN ARRANGEMENT OF
CHAMBER THIMBLES EMPLOYED



ELEVATION ARRANGEMENT OF
CHAMBER THIMBLES EMPLOYED

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FIGURE 7

APRM SYSTEM (NW QUADRANT OF CORE)

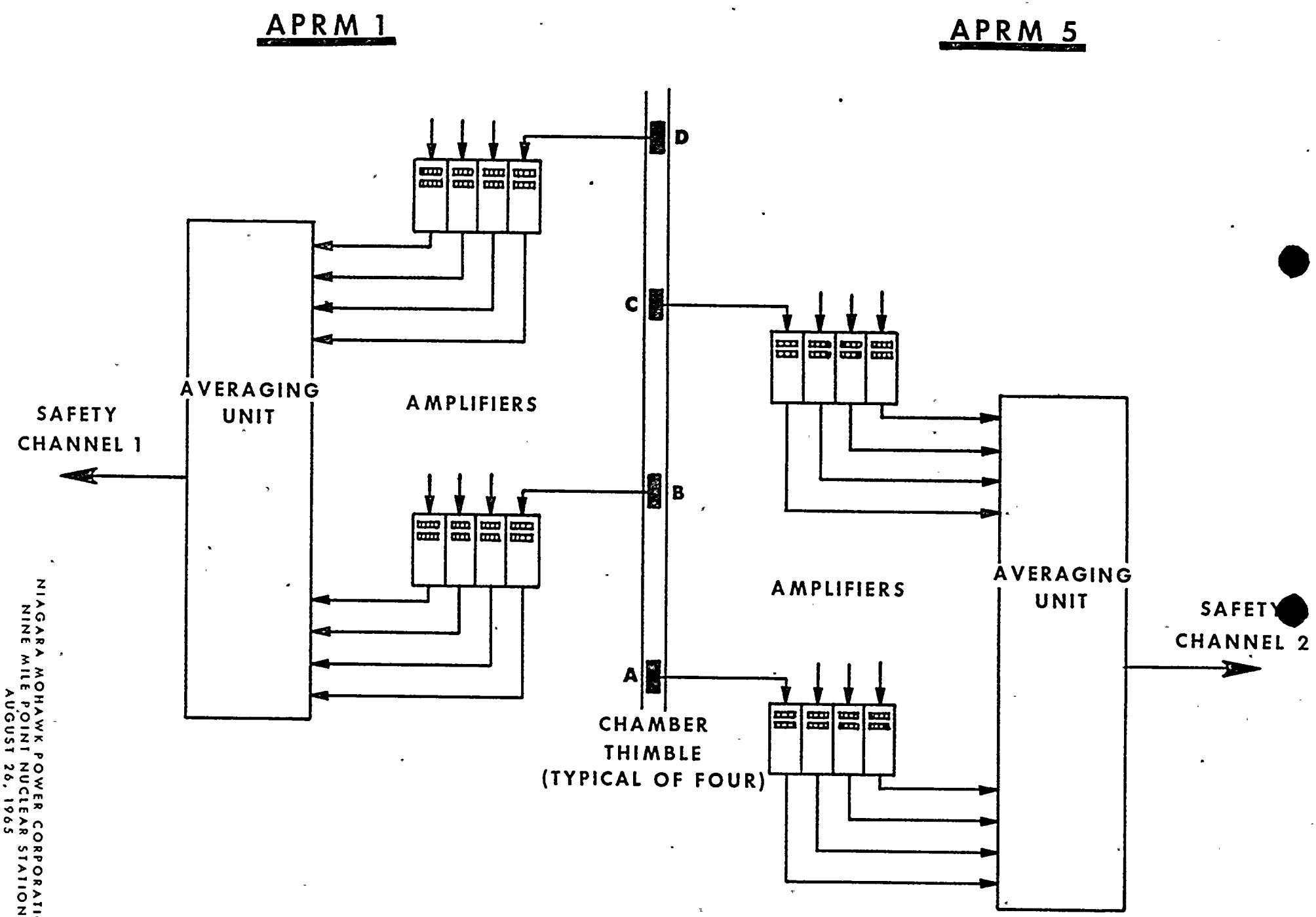

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FIGURE 8

APRM RESPONSE DURING POWER LEVEL MANEUVERING WITH CONTROL RODS

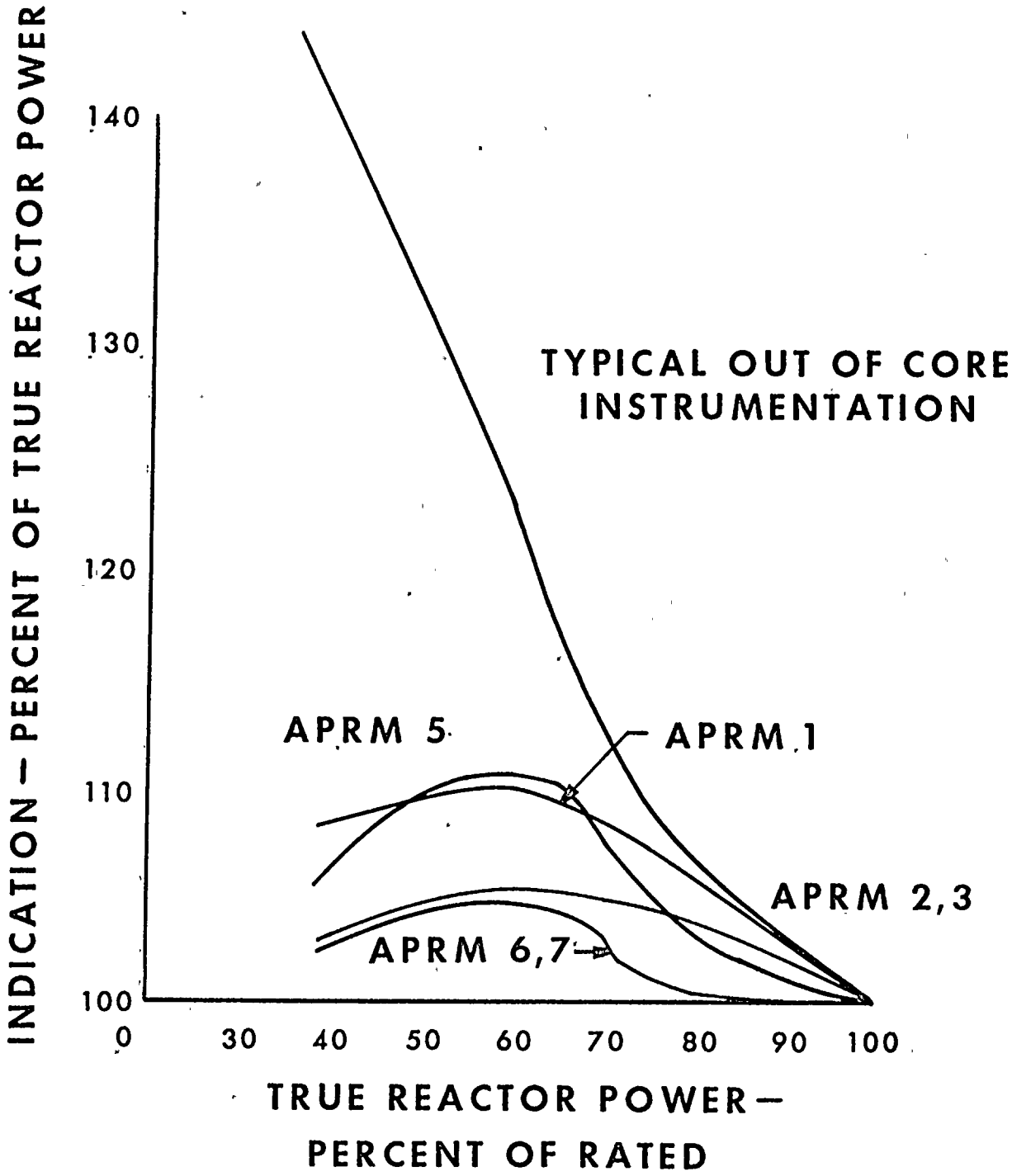

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FIGURE 9

APRM RESPONSE DURING POWER LEVEL MANEUVERING WITH FLOW

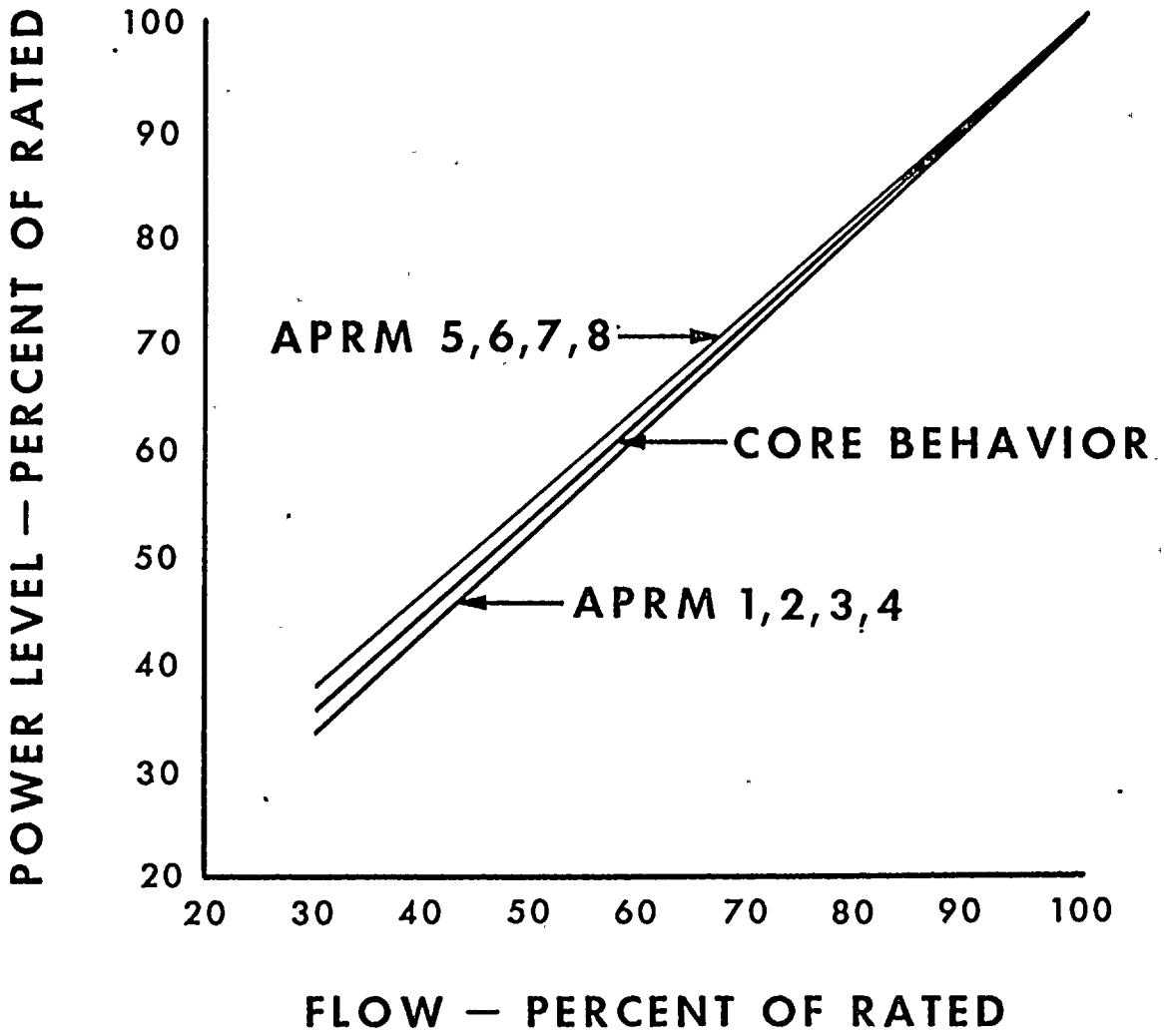

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FIGURE 10

TYPICAL APRM HIGH FLUX INTERLOCK

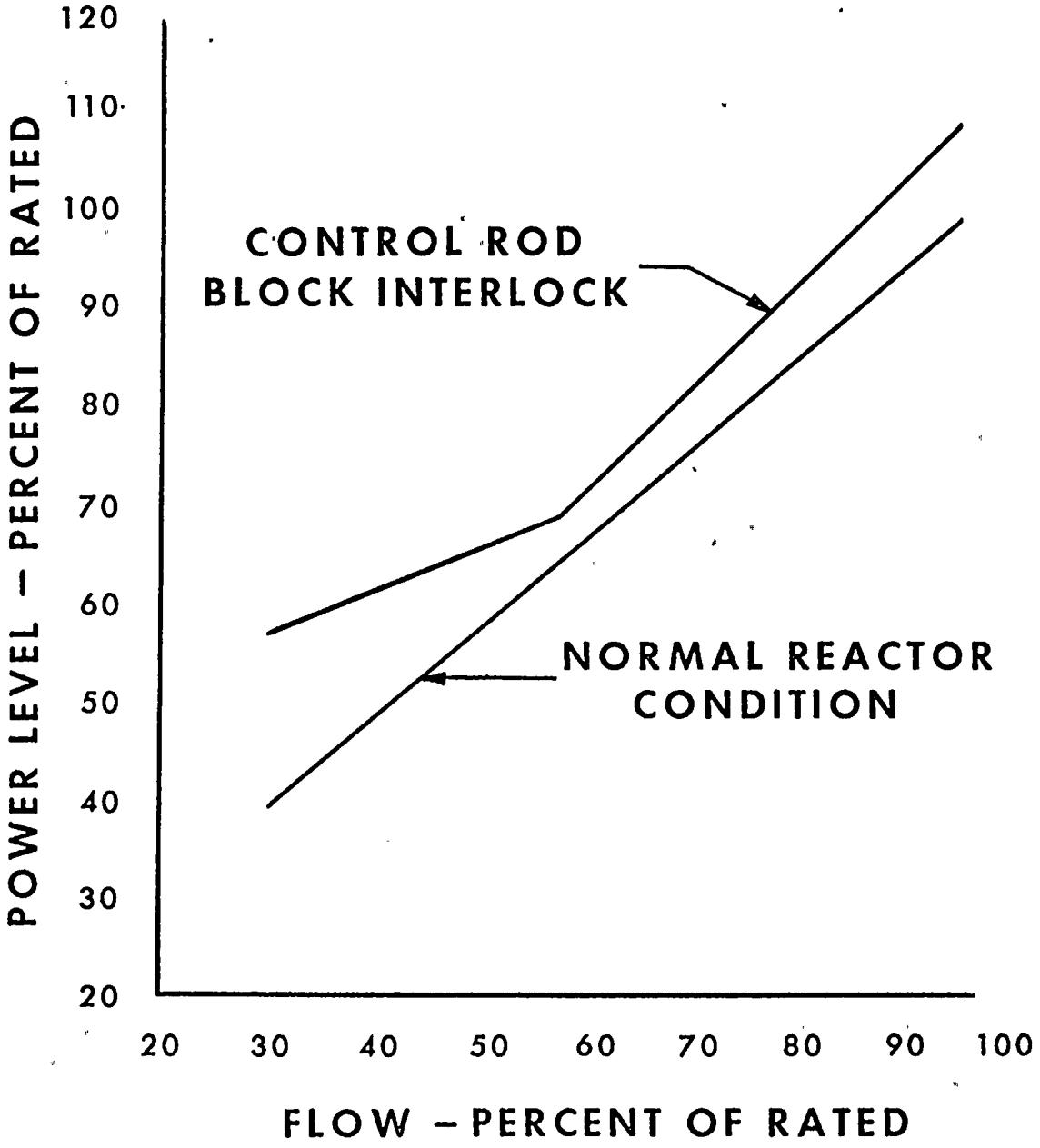

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FIGURE 11

The readout and associated equipment for the APRM system is located in the control room. Outputs from the recirculation flow sensors are used to provide the reference flow information. Operational amplifiers designated as averaging units in Figure 10 are used to average the signals from the individual LPRM amplifiers in each of the eight APRM groups. The output signals from the eight APRM groups are displayed to the operator and are automatically recorded. The bypass switches previously described are located on the console. It may also be necessary to provide two manually operated range switches on the console to permit ranging down the APRM trips when a power level reduction is made by control rod insertion.

The APRM will be calibrated using conventional heat-balance techniques to determine the thermal-power level of the reactor. Calibration will be accomplished using an adjustment at the output of the averaging unit shown in Figure 10. This adjustment will be made in the control room and is not readily accessible to the operator. The output signals on the individual LPRM chambers are not affected by this calibration of the APRM.

Traveling In-Core Probe System (TIP)

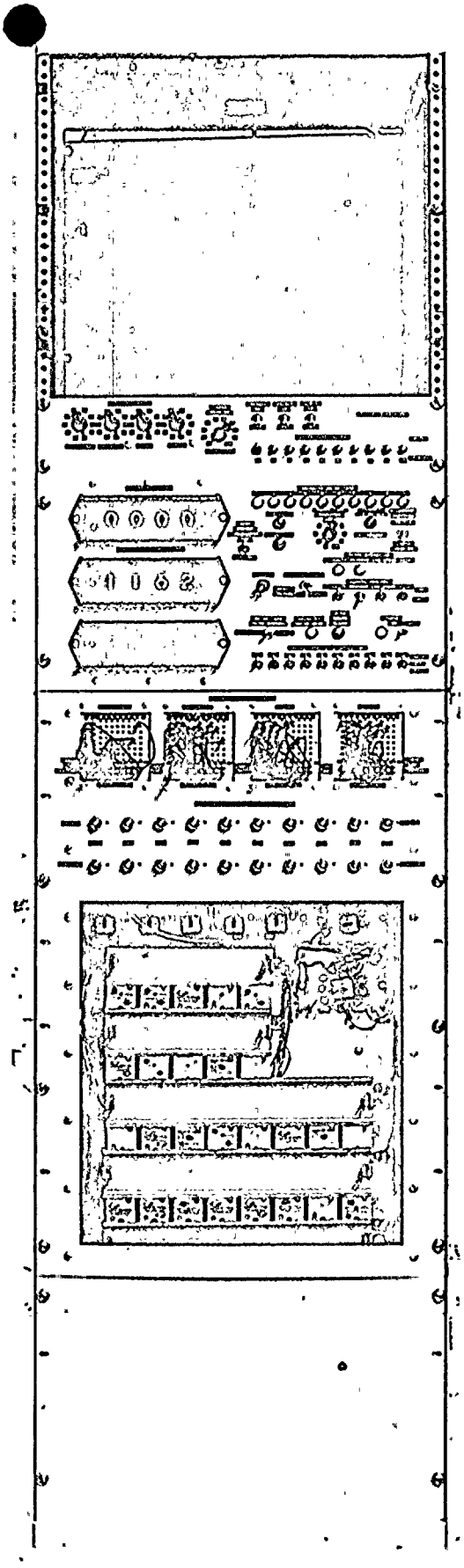
The traveling in-core probe (TIP) calibration system is designed to permit rapid and accurate calibration of the LPRM system. The TIP system consists of four miniature ionization chambers similar to the chambers used in the LPRM system.

Each of these chambers will obtain axial flux profile data in eight of the chamber thimbles shown in plan view in Figure 5. A ninth and common position will be traversed by each of the four TIP chambers to permit normalization of data.

In addition to the chambers, the TIP system contains cable runs, driving mechanisms used to move the chambers, indexing mechanisms used to select the thimble to be traversed, and control and readout equipment to operate the equipment and record the data. Figures 12 through 15 are photographs of a prototype TIP system which has been constructed and operated at General Electric's San Jose facilities.

E. Status of Testing

Experiments have been performed and others are in progress to establish the effects of neutron exposure on the sensitivity of both SRM and IRM chambers. This work at Big Rock Point has already demonstrated the capability of the SRM chambers to operate properly after a total integrated neutron exposure of 10^{19} nvt and of the IRM chambers to an exposure of 10^{20} nvt. Additional tests at Dresden indicate that the IRM chambers may be capable of satisfactory performance after exposures in excess of 10^{21} nvt. Tests have also been performed in the GETR to verify the performance of the SRM chambers in the high gamma fields which exist during the worst hot restart condition minutes after a shutdown. The



PROTOTYPE TIP CONTROL CABINET

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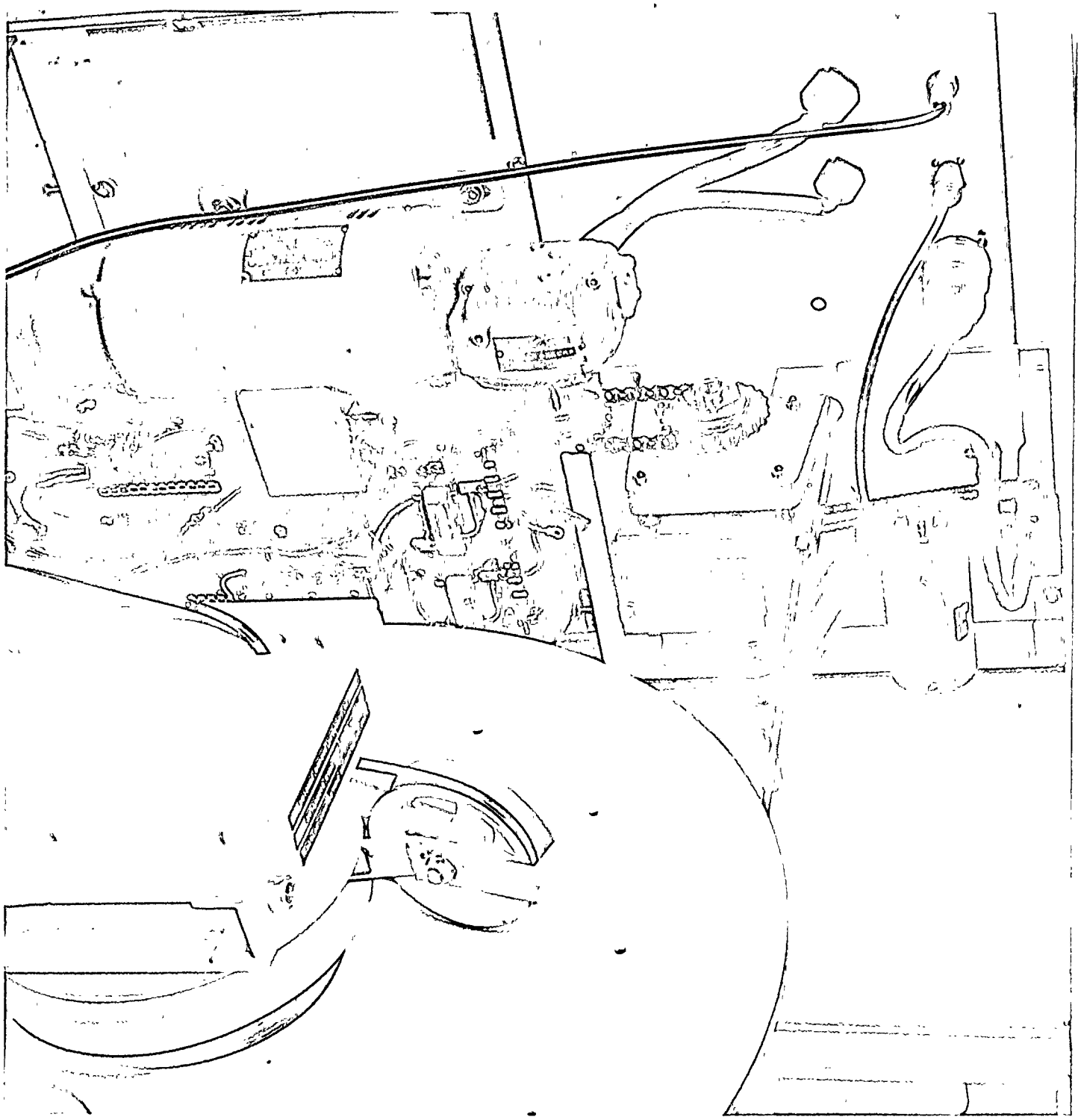
FIGURE 12



PROTOTYPE TIP CABLE DRIVE AND TAKEUP REEL CABINET INTERNALS

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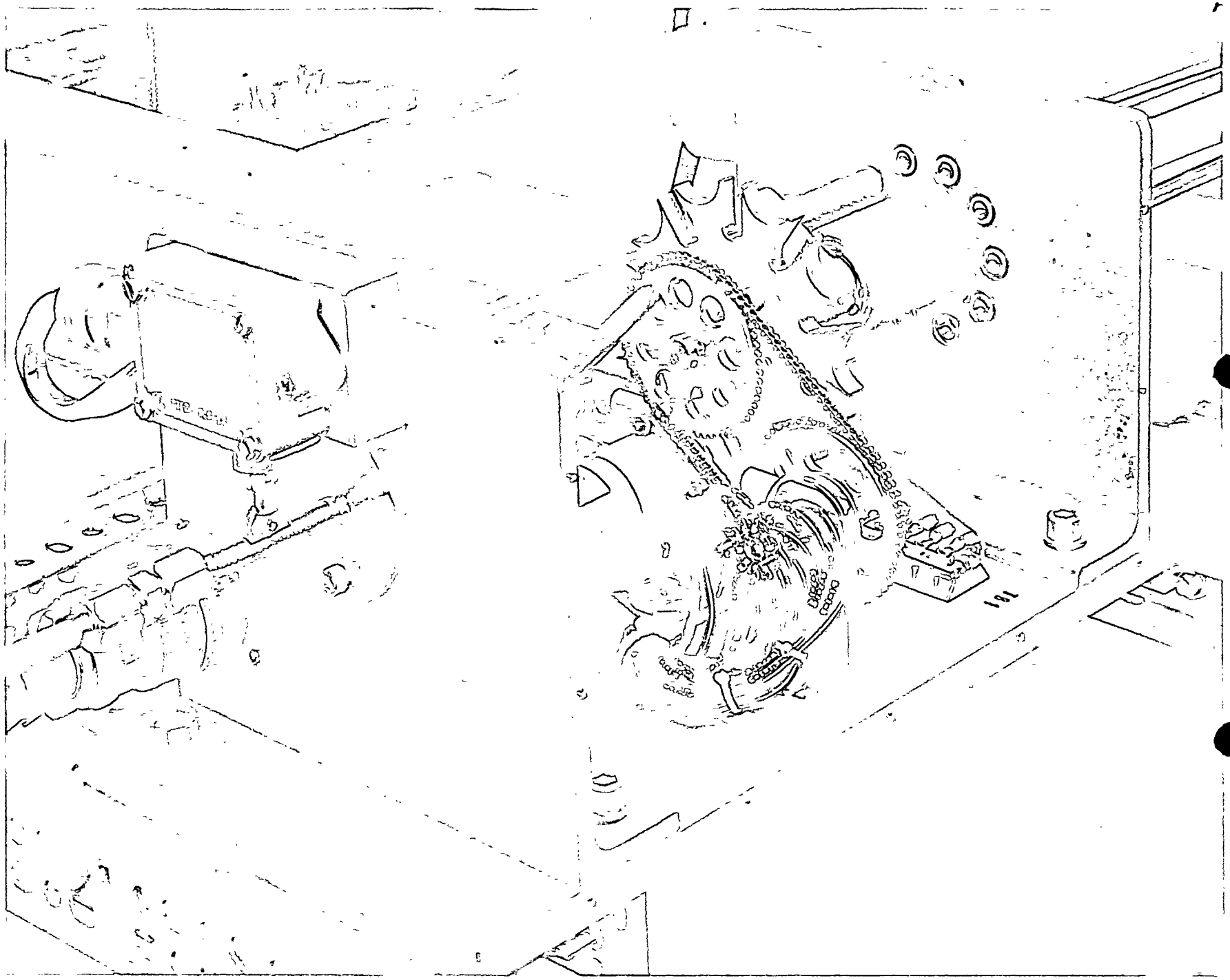
FIGURE 13



PROTOTYPE TIP INDEXING MECHANISM

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FIGURE 14

PROTOTYPE TIP GUIDE TUBE RUN AND REACTOR TEMPERATURE SIMULATOR

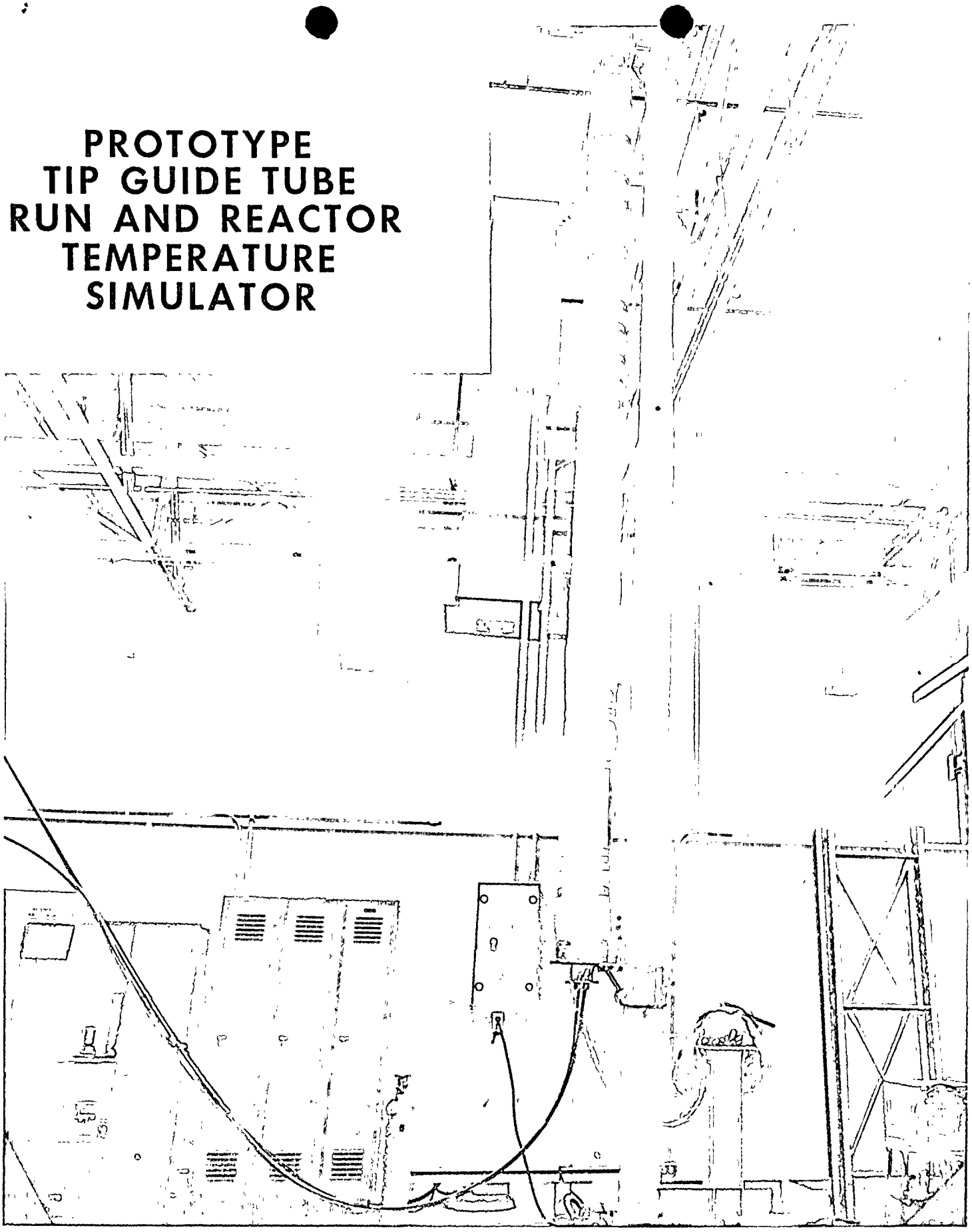

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FIGURE 15

LPRM chamber to be employed will be similar to those used in other operating boiling-water reactors. Extensive experience is available to demonstrate their reliability and adequacy as power-range monitors. These programs have already produced sufficient data and experience to confidently incorporate in-core chambers for the startup, intermediate and power range neutron instrumentation systems at Nine Mile Point. Further tests are planned, however, in a continuing effort to improve the performance of these chambers through modifications and to establish upper limits of performance.

Retraction mechanisms are incorporated in the design to minimize neutron exposure of the SRM and IRM chambers. A test facility has been constructed to verify the mechanical performance of these retraction mechanisms. The test mechanism has been operated successfully through about 10,000 cycles under simulated reactor conditions for the mechanical components. This is far in excess of the number of cycles expected during the life of the station.

The IRM system will use a new, voltage-variance, measurement technique. Extensive analytical and test work has been done to establish the validity and reliability of this technique.[2]

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(2) GEAP-4862, "Reactor Control System Based on Counting and Campbelling Techniques".

The APRM system is a new extension of the established in-core ionization chamber measurement system (LPRM) utilizing components which have been separately proven in other applications. Therefore, no equipment test work is planned except for the normal performance checkouts. For example, checkout tests are planned during the power test program at Nine Mile Point to verify the response of the APRM system to power-level maneuvering with control rods and with other power-level control mechanisms, to demonstrate the reliability of the methods used to predict the performance of the APRM system and to verify that the specified trip settings are adequate.

The TIP system is new and the test facility to verify the performance of the system has already been described. All reactor and relevant conditions except for the neutron and gamma flux are simulated in the test facility. The results obtained by cycling the test equipment through several thousand operating cycles demonstrate the mechanical reliability of the system. Operating experience will be obtained from other reactors employing this system before the startup of the Nine Mile Point reactor.

III. Main Steam Line Safeguards

The design criteria presented in the PHSR to minimize the potential hazard associated with a main steam line break outside the drywell included two automatic closing isolation valves in each steam line. The closure times for these valves must be set to minimize the loss of reactor coolant during blowdown, thereby preventing fuel-clad failures due to overheating. Preliminary studies indicated that valve closure times in the range of 10 to 30 seconds would be adequate to meet these criteria. Subsequent studies which took into account friction losses in the piping as presently designed, back-feeding of the break by the unbroken line through the turbine stop-valve header, and a swelling correction show the coolant loss to be greater than reported in the PHSR. As a result, we stated during our May meeting that Venturi-type flow nozzles will be utilized in each of the two main steam lines inside the drywell to limit the blowdown flow rate.

These nozzles will have a Beta ratio of approximately 0.5. In the event of a line break the nozzles will restrict the maximum flow rate during blowdown to approximately 200 percent of rated steam flow as compared to about 600 percent without these devices. To limit further the loss-of-coolant during blowdown it has been found feasible to utilize faster closing isolation valves than previously reported. The isolation valves outside the drywell will operate hydraulically or pneumatically with adjustable closure times

from 3 to 10 seconds. The inside isolation valves will be A-C motor operated. Discussions with suppliers indicate that it will be possible to obtain closure times in the range of 8 to 20 seconds.

Recent studies of coolant loss for various valve-closure times are presented in Figure 16. These calculations are based on swell data collected at EVESR which indicate that the reactor water level would rise at a rate such that water would reach the steam line flow nozzles approximately three seconds after the postulated steam line break. The resultant steam-water mixture will not prevent closure of the steam line isolation valves as stated in the PHSR's First Supplement (Page VII-8). The results of these studies show that a closure time of ten seconds will limit coolant loss to approximately 80,000 pounds which is less than the volume of water over the core. In addition, pressure transients resulting from a three-second isolation-valve closure time can be accommodated by the power-operated pressure-relief valves without actuating the reactor safety valves.

The combination of faster isolation valve closures and flow limiters is expected to reduce the radioactivity released due to a steam line break by approximately a factor of two from the values reported in the PHSR.

COOLANT LOSS FOR MAIN STEAM LINE RUPTURE

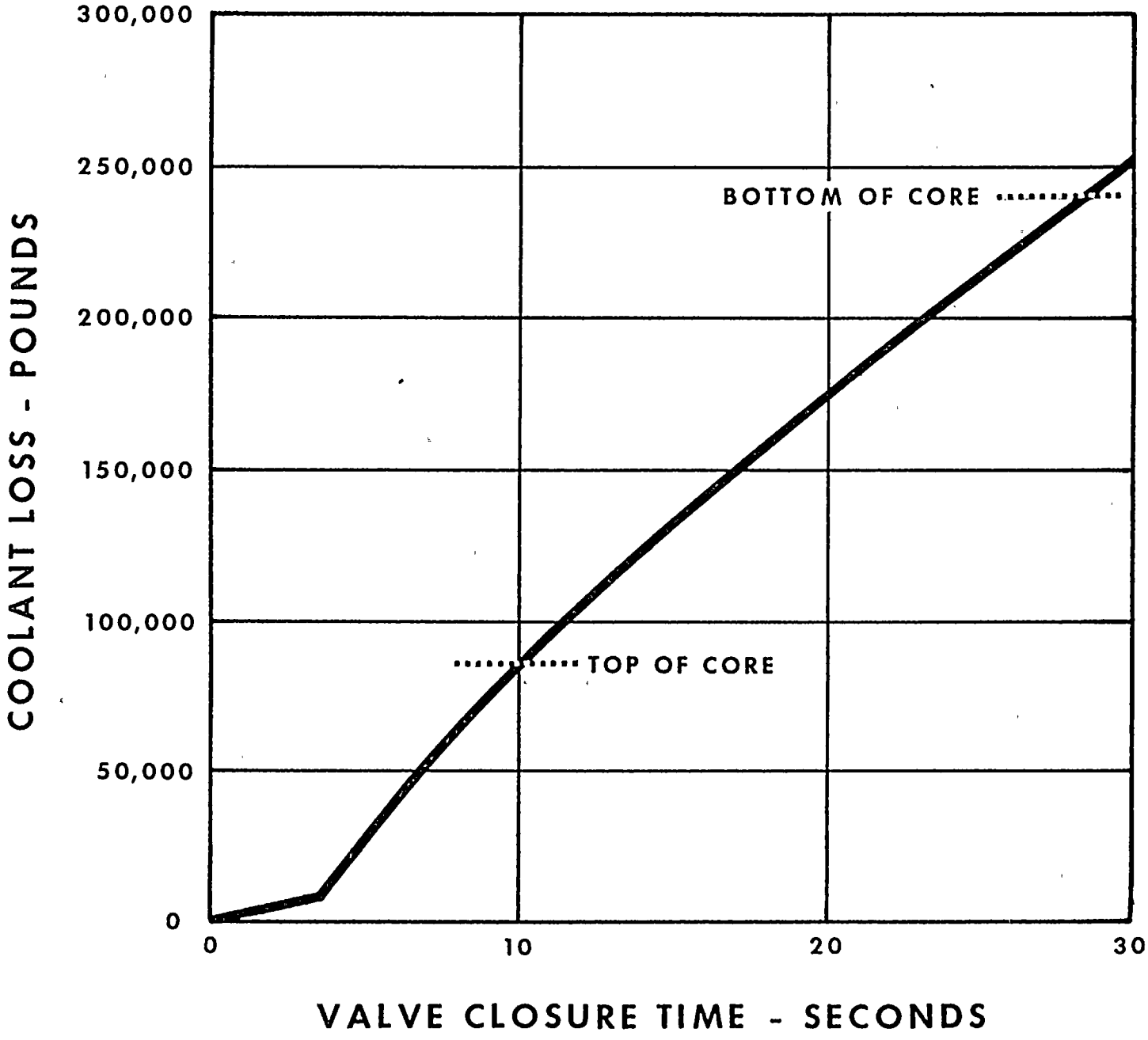

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FIGURE 16

IV. Core and Containment Spray Systems

The principal features of the core and containment spray systems as described at our meeting of May 19, 1965, remain unchanged and specification of major components is now nearing completion.

We have also initiated detailed studies along the lines suggested by the discussions during our last meeting to minimize vulnerability of these systems to crud and debris and to provide complete testing capability. In addition to the external testing provisions described during our last meeting, means will be provided to test the spray nozzles and headers inside the drywell and reactor to assure proper flow.

All equipment within the drywell necessary for safe and orderly shutdown is being designed to remain operational upon an inadvertent operation of the containment spray system. Further, certain other criteria are being established primarily to protect equipment. These include automatic trip of the reactor recirculation pump upon spray actuation, use of metallic piping insulation throughout and careful attention to proper spray distribution. It is our intent to describe fully these criteria and design details to you as soon as the studies are complete.