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TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>	
7	<u>INSTRUMENTATION AND CONTROL</u>	7-1	
7.1	<u>PROTECTION SYSTEMS</u>	7-1	
7.1.1	DESIGN BASES	7-1	
7.1.1.1	<u>Vital Functions</u>	7-1	
7.1.1.2	<u>Principles of Design</u>	7-2	
7.1.1.3	<u>Functional Requirements</u>	7-4	3
7.1.1.4	<u>Environmental Considerations</u>	7-5	
7.1.2	SYSTEM DESIGN	7-5	
7.1.2.1	<u>System Description - Reactor Protection System</u>	7-5	
7.1.2.2	<u>Description - Safeguards Actuation System</u>	7-5b	
7.1.2.3	<u>Design Features</u>	7-7	
7.1.2.4	<u>Summary of Protective Actions</u>	7-10	
7.1.2.5	<u>Relationship to Safety Limits</u>	7-11	
7.1.3	SYSTEMS EVALUATION	7-12	3
7.1.3.1	<u>Functional Capability - Reactor Protection System</u>	7-12	
7.1.3.2	<u>Functional Capability - Safeguards Actuation System</u>	7-13	
7.1.3.3	<u>Preoperational Tests</u>	7-13	
7.1.3.4	<u>Component Failure Considerations</u>	7-13a	3
7.1.3.5	<u>Operational Tests</u>	7-14	
7.2	<u>REGULATING SYSTEMS</u>	7-14	
7.2.1	DESIGN BASES	7-14	
7.2.1.1	<u>Compensation Considerations</u>	7-14	
7.2.1.2	<u>Safety Considerations</u>	7-15	

CONTENTS (Cont'd)

<u>Section</u>		<u>Page</u>
7.2.1.3	<u>Startup Considerations</u>	7-16
7.2.2	SYSTEM DESIGN	7-16
7.2.2.1	<u>Description of Reactivity Control</u>	7-16
7.2.2.2	<u>Integrated Control System</u>	7-19
7.2.3	SYSTEM EVALUATION	7-22
7.2.3.1	<u>System Failure Considerations</u>	7-22
7.2.3.2	<u>Interlocking</u>	7-23
7.2.3.3	<u>Emergency Considerations</u>	7-23
7.2.3.4	<u>Loss-of-Load Considerations</u>	7-23
7.3	<u>INSTRUMENTATION</u>	7-24
7.3.1	NUCLEAR INSTRUMENTATION	7-24
7.3.1.1	<u>Design</u>	7-25
7.3.1.2	<u>Evaluation</u>	7-26
7.3.2	NONNUCLEAR PROCESS INSTRUMENTATION	7-27
7.3.2.1	<u>System Design</u>	7-27
7.3.2.2	<u>System Evaluation</u>	7-28
7.3.3	INCORE MONITORING SYSTEM	7-28
7.3.3.1	<u>Design Basis</u>	7-28
7.3.3.2	<u>System Design</u>	7-29
7.3.3.3	<u>System Evaluation</u>	7-30
7.4	<u>OPERATING CONTROL STATIONS</u>	7-31
7.4.1	GENERAL LAYOUT	7-31
7.4.2	INFORMATION DISPLAY AND CONTROL FUNCTION	7-31
7.4.3	SUMMARY OF ALARMS	7-31
7.4.4	COMMUNICATION	7-32

CONTENTS (Cont'd)

<u>Section</u>		<u>Page</u>	
7.4.5	OCCUPANCY	7-32	
7.4.6	AUXILIARY CONTROL STATIONS	7-33	
7.4.7	SAFETY FEATURES	7-33	
7.5	SYSTEMATIC, NONRANDOM, COMMON MODE INSTRUMENTATION FAILURES	7-33	13

7-iii

10-31-69
Supplement No. 13

LIST OF FIGURES

(At rear of Section)

<u>Figure No.</u>	<u>Title</u>
7-1	Reactor Protection System Block Diagram
7-2	Nuclear Instrumentation and Protection Systems
7-3	Typical Control Circuits for Engineered Safeguards Equipment
7-4	Reactor Power Measurement Errors and Control Limits
7-5	Reactor and Steam Temperatures versus Reactor Power
7-6	Reactor Control Diagram - Integrated Control System
7-7	Automatic Control Rod Groups - Typical Worth Curve versus Distance Withdrawn
7-8	Steam Generator and Turbine Control Diagram - Integrated Control System
7-9	Nuclear Instrumentation Flux Ranges
7-10	Nuclear Instrumentation Detector Locations
7-11	Nonnuclear Instrumentation Schematic
7-12	Incore Detector Locations
7-13	Typical Arrangement - Incore Instrumentation Channel
7-14	Control Room Layout

7.1 PROTECTION SYSTEMS

The protection systems, which consist of the Reactor Protection System and the Safeguards Actuation System, perform the most important control and safety functions. The protection systems extend from the sensing instruments to the final actuating devices, such as trip circuit breakers and pump or valve motor contactors.

7.1.1 DESIGN BASES

The Reactor Protection System monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage caused by departure from nucleate boiling (DNB), and to protect against reactor coolant system damage caused by high system pressure. The Safeguards Actuation System monitors parameters to detect failure of the reactor coolant system and initiates reactor building isolation and engineered safeguards operation to contain radioactive fission products in the reactor building.

7.1.1.1 Vital Functions

The Reactor Protection System automatically trips the reactor to protect the reactor core under these conditions:

- a. When the reactor power, as measured by neutron flux, exceeds the limit set by the reactor coolant flow.
- b. Loss of any two reactor coolant pumps.
- c. The reactor outlet temperature reaches an established maximum limit.
- d. The reactor pressure reaches an established minimum limit.

The Reactor Protection System automatically trips the reactor to protect the reactor coolant system under this condition:

- a. The reactor pressure reaches an established maximum limit.

The Safeguards Actuation System automatically performs the following vital functions:

- a. Commands operation of injection emergency core coolant.
- b. Commands operation of the reactor building emergency cooling system and the reactor building spray system.
- c. Commands closing of the reactor building isolation valves.

The core flooding system is a passive system and does not require Safeguards Actuation System action.

7.1.1.1.1 Nonvital Functions

The Reactor Protection System provides an anticipatory reactor trip when the reactor startup rate reaches specified limits.

7.1.1.2 Principles of Design

The protection systems are designed to meet the requirements of the IEEE proposed "Standard for Nuclear Power Plant Protection Systems," dated September 13, 1966. Prototype and final equipment will be subject to qualification tests as required by the subject standard. The tests will establish the adequacy of equipment performance in both normal and accident environments.

The major design criteria are summarized in the following paragraphs.

7.1.1.2.1 Single Failure

- a. No single component failure shall prevent the protection systems from fulfilling their protective functions when action is required.
- b. No single component failure shall initiate unnecessary protection system action, provided implementation does not conflict with the criterion above.

7.1.1.2.2 Redundancy

All protection system functions shall be implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels.

7.1.1.2.3 Independence

Redundant protection channels and their associated elements will be electrically independent and packaged to provide physical separation.

Separate detectors and instrument strings are not, in general, employed for protection system functions and regulation or control. However, no more than one channel of protection is shared with control at any time so that failures in the single channel still have three complete protection channels which fully satisfy the single failure criteria of the IEEE standard for reactor protection systems. Sharing instrumentation for protection and control functions is accomplished within the framework of the stated criteria by the employment of isolation amplifiers in each of the multiple outputs of the analog protection system instrument strings.

The isolation amplifiers are precision operational amplifiers having a closed loop unity gain and a low dynamic output impedance. The effectiveness of the isolation amplifiers has been proven by actual test.

The isolation amplifiers will block a direct connection across their output of 410 vdc or peak ac, 300 v rms without perturbing the input signal.

3

This may be stated as a corollary to the design criteria: "a direct short, open circuit, ground fault, faulting to a power source of less than 410 volts, or bridging of any two points at the output terminals of a protection system analog instrument string having multiple outputs shall not result in a significant disturbance within more than one output."

3

Testing has demonstrated that the protection system design will meet the above criteria.

7.1.1.2.4 Loss of Power

- a. A loss of power in the Reactor Protection System shall cause the affected channel to trip.
- b. Availability of power to the Safeguards Actuation System shall be continuously indicated. The loss of instrument power, i.e., vital bus power, to the instrument strings and bistables will initiate a trip in the affected channels. System actuation requires control power from one of the two engineered safeguards d-c power busses so that loss of this power does not actuate the system. The system equipment is divided between the redundant engineered safeguards channels in such a way that the loss of one of the d-c power busses does not inhibit the system's intended safeguards functions.

7.1.1.2.5 Manual Trip

Each protection system shall have a manual actuating switch or switches in the control room which shall be independent of the automatic trip instrumentation. The manual switch and circuitry shall be simple, direct-acting, and electrically connected close to the final actuating device.

7.1.1.2.6 Equipment Removal

The Reactor Protection System shall initiate a trip of the channel involved when modules, equipment, or subassemblies are removed. Safeguards Actuation System channels shall be designed to provide for servicing a single channel without affecting integrity of the other redundant channels or without compromising the criterion that no single failure shall prevent actuation.

7.1.1.2.7 Testing

Manual testing facilities shall be built into the protection systems to provide for

- a. Preoperational testing to give assurance that the protection systems can fulfill their required functions.
- b. On-line testing to prove operability and to demonstrate reliability.

7.1.1.3 Functional Requirements

The functional requirements of the protection systems are those specified under vital functions together with interlocking functions.

The functional requirements of the Reactor Protection System are to trip the reactor under the following conditions:

- a. The reactor power, as measured by neutron flux, reaches an allowable limit set by the number of operating reactor coolant pumps.
- b. Two reactor coolant pumps are lost. (This also covers loss of three pumps and loss of all pumps) | 4
- c. The reactor outlet temperature reaches a preset maximum limit.
- d. The reactor coolant pressure reaches a preset maximum limit.
- e. The reactor coolant pressure reaches a preset minimum limit.
- f. The reactor startup rate reaches a maximum limit while operating below a preset power level.
- g. The reactor power as measured by neutron flux reaches an allowable point set by the measured reactor coolant flow. | 3

Interlocking functions of the Reactor Protection System are to

- a. Bypass the startup rate trip when the reactor power reaches a preset value.
- b. Inhibit control rod withdrawal on the occurrence of a predetermined startup rate, slower than the rate at which reactor trip is initiated.

The functional requirements of the Safeguards Actuation System are to

- a. Start operation of high pressure injection upon detection of a low reactor coolant system pressure or upon detection of a high reactor building pressure. | 3
- b. Start operation of low pressure injection upon detection of a very low reactor coolant system pressure, or upon detection of a high reactor building pressure.
- c. Operate the reactor building isolation valves upon detection of a moderately high reactor building pressure.
- d. Start the reactor building emergency cooling units upon detection of a moderately high reactor building pressure.

- e. Start the reactor building spray system upon detection of a high reactor building pressure.

7.1.1.4 Environmental Considerations

The operating environment for equipment within the reactor building will normally be controlled to less than approximately 110 F. The Reactor Protection System instrumentation within the reactor building is designed for continuous operation in an environment of 140 F, 60 psig, and 100 per cent relative humidity, but will function with less accuracy at the accident temperature.

The environment for the neutron detectors will be limited to 150 F with a relative humidity of less than 90 per cent. The detectors are designed for continuous operation in an environment of 175 F, 90 per cent relative humidity, and 150 psig.

The Safeguards Actuation System equipment inside the reactor building will be designed to operate under the accident environment of a steam-air mixture.

Protective equipment outside of the reactor building, control room, and relay room is designed for continuous operation in an ambient of 120 F and 90 per cent relative humidity. The control room and relay room ambients will be maintained at the personnel comfort level; however, protective equipment in the control room and relay room will operate within design tolerance up to an ambient temperature of 110 F.

7.1.2 SYSTEM DESIGN

7.1.2.1 System Description - Reactor Protection System

Figure 7-1 is a block diagram of the Reactor Protection System. The system consists of four identical protection channels, each terminating in a non-inverting bistable and reactor trip relay. In the normal untripped state, each channel functions as an AND gate, passing current to the terminating bistable and holding the reactor trip relay energized only if all channel inputs are in the normal energized (untripped) state. Should any one or more inputs to a channel become deenergized (tripped), the terminating bistable in that channel trips, deenergizing the reactor trip relay. Thus, for trip signals each channel becomes an OR gate.

Contacts from the four reactor trip relays (RS) are arranged into four identical 2-out-of-4 coincidence networks. Each pair of these coincidence networks controls the power to one of the two identical control rod drive power supplies. | 3

The reactor trip circuits are shown in more detail on Figure 7-2, which is an overall diagram showing the Nuclear Instrumentation System (7-2A), Reactor Protection System (7-2B), and the Safeguards Actuation System (7-2C). Figure 7-2B shows the circuit breakers controlling input power to each control rod drive clutch assembly and the manner in which the reactor trip relays trip these circuit breakers.

Reactor trip is accomplished by interrupting all input power to each drive clutch assembly. Each control rod drive clutch power supply received its input power through two circuit breakers in series so that opening of either interrupts that source of power. The two control rod drive clutch power supplies operate in parallel so that both must be deenergized for the control rods to trip. Circuit breakers No. 1 and No. 2 control primary power to one clutch assembly power supply, and circuit breakers No. 3 and No. 4 control power to the other. Thus, reactor trip is accomplished by tripping one circuit breaker in each of these pairs.

The control rod drive clutch holding coil power supply circuit breakers are equipped with undervoltage coils which must be energized for the circuit breaker to be closed or to remain closed. The holding voltage for the undervoltage coil of each circuit breaker is taken from the vital bus.

In each circuit breaker (nos. 1, 2, 3 and 4) the undervoltage coils are energized through contacts of trip relays RS1, RS2, RS3, and RS4 under normal conditions with all trip relays energized. If trip relays RS1 and RS2, RS1 and RS4, RS2 and RS4, or RS3 and RS4 become deenergized, each circuit breaker undervoltage coil will be deenergized, and the circuit breaker will open. Thus any 2-out-of-4 trip relays will cause each circuit breaker to open, removing power. 4

The trip circuits and devices are redundant and independent. Each breaker is independent of each other breaker, so a single failure within one trip circuit does not affect any other trip circuit or prevent trip. By this arrangement each breaker may be tested independently by the manual test switch. One segment of the manual reactor trip switch is included in each of the circuit breaker trip circuits to implement the "direct action in the final device" criterion.

The power/flow monitor logic details are also shown on Figure 7-2. There are four identical sets of power/flow monitor logic, one associated with each protective channel. Each set of logic receives an independent total reactor coolant flow signal (\dot{V}), a "number of pump motors in operation" signal (P_n), and an isolated reactor power level signals (ϕ). 3

Using a flux/flow comparator, one part of the power/flow monitor continuously compares the ratio of the reactor neutron power to the total reactor coolant flow. Should the reactor power as measured by the linear power range channels exceed 1.075 times the reactor coolant flow, a reactor trip is initiated. All measurements are in terms of per cent full flow or full (rated) power. The flux/flow comparator runs back the over flux trip level in step with a detected decreasing flow thus providing an opportunity for the control system to reduce the reactor power to an acceptable level without a reactor trip. 4

The second element in the power/flow monitor is the pump monitor logic. The pump monitor logic counts the number of pump motors in operation as indicated by the number of closed pump power breakers and initiates a reactor trip if less than three pumps are in operation.

Figure 7-2C shows the action initiating sensors, bistables, and logic for the Safeguards Actuation System. The major differences between this system and the Reactor Protection System are:

- a. Each protective action is initiated by either of two channels with 2-out-of-3 coincidence logic between input signals.
- b. Either of the two channels is independently capable of initiating the desired protective action through redundant Safeguards equipment
- c. Protective action is initiated by the application of power to the terminating control relays through the coincidence logic.

There are three independent sensors for each input variable. Each sensor terminates in a bistable device. The outputs of the three bistables associated with each variable are formed into two identical and independent 2-out-of-3 coincident logic networks or channels. Safeguards action is

initiated when either of the channels associated with a variable becomes energized through the coincident trip action of the associated bistables. The engineered safeguards equipment is divided between redundant actuation channels as shown in Figure 7-2C. The division of equipment between channels is based upon the redundancy of equipment and functions. Where two active safeguards valves are connected in redundant manner, each valve will be controlled by a separate engineered safeguards channel as shown in Figure 7-2C. When active and passive (check valve) safeguards valves are used redundantly, the active valve will be equipped with two OR control elements, each driven by one of the safeguards channels. Redundant safeguards pumps will be controlled in the same manner as redundant active valves. Figure 7-2C shows a typical control scheme for both safeguards valves and pumps. Separate dual logic channels for reactor building spray pumps and valves are provided. The separate logic permits testing the reactor building spray system without actually spraying water by starting the pumps with the valves closed and opening the valves with the pumps shut off.

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Figure 7-3 shows typical control circuits for equipment serving safeguards functions. Each circuit provides for normal start-stop control by the plant operator as well as automatic actuation. Normal starting and stopping are initiated by momentary contact pushbuttons or control switches.

The control circuit shown for a makeup pump is typical of the controller of a large pump started by switchgear. There are three makeup pumps, two are equipped with single control relays powered from separate safeguards actuation channels. The third pump is equipped with two control relays, CR1 and CR2, each of which is powered from separate safeguards actuation channels. Energizing the control relays through their associated safeguards actuation channel energizes the pump circuit breaker closing coil and starts the pump.

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The control of the reactor building spray pumps and decay heat removal pumps is by means of single control relays in each pump control circuit. Each pump is controlled by separate engineered safeguards channels. Safeguards action is initiated when the pump control relay is energized by its associated engineered safeguards channel.

The control circuit for a reactor building isolation valve is typical of a motor-operated valve which is required to close as its engineered safeguards action. If the valve is employed as one of two active redundant valves, then it is controlled by a single safeguards actuation channel to CR1. If the valve is employed with a passive redundant check valve, then the motor operated valve is controlled by two safeguards actuation channels with CR1 and CR2 connected in an "OR" configuration.

The control relays, when energized by their associated safeguards actuation channels, close the valve through contacts which duplicate the manual CLOSE pushbutton and at the same time override any existing signal calling for the valve to open. A valve limit switch opens just before the valve seats to permit torque closing.

Air-operated engineered safeguards valves automatically go to their engineered safeguards position upon loss of control air. Valves used with active redundant valves are equipped with a single electrical actuator for control by a single engineered safeguards channel as shown in Figure 7-2C. Valves used with redundant passive valves are equipped with two electrical actuators, each controlled by a single safeguards channel operating in an OR configuration. Engineered safeguards action is initiated when power is applied to the electrical actuator.

7.1.2.3 Design Features

7.1.2.3.1 Redundancy

The Reactor Protection System is redundant for all vital inputs and functions. Redundancy begins with the sensor. Each power range input variable is measured four times by four independent and identical instrument strings. Only one of the four is associated with any one protective channel. The total and complete removal of one protective channel and its associated vital instrument strings would not impair the function of any other instrument or protective channel.

There are two source range channels and two intermediate range channels, each with its own independent sensor.

The Safeguards Actuation System is also redundant for all vital inputs and functions. Each input variable is measured by three independent and identical instrument strings. The total removal of any one instrument string will not prevent the system from performing its intended functions.

7.1.2.3.2 Independence

The redundancy, as described above, is extended to provide independence in the Reactor Protection System. Each instrument string feeding into one protective channel is operationally and electrically independent of every other instrument string. Each protective channel is likewise functionally and electrically independent of every other channel.

Only in the coincidence output are the channels brought into any kind of common relationship. Independence is preserved in the coincidence circuits through insulation resistance and physical separation of the coincidence networks and their switching elements.

The Safeguards Actuation System instrumentation and control have electrically and physically independent instrument strings. The output of each bistable is

electrically independent of every other bistable. Independence is preserved in the coincidence networks through insulation resistance and physical separation of the switching elements.

7.1.2.3.3 Loss of Power

The Reactor Protection System initiates trip action upon loss of power. All bistables operate in a normally energized state and go to a deenergized state to initiate action. Loss of power thus automatically forces the bistables into the tripped state. Figure 7-2B shows the system in a deenergized state.

The Safeguards Actuation System instrumentation strings terminate in bistable trip elements similar to those in the Reactor Protection System. Loss of instrument power up to and including the bistables forces the bistables into the tripped state initiating safeguards action. The logic networks and the equipment control elements are powered from the Engineered Safeguards d-c power buses 1 and 2. Electrical safeguards equipment is powered from one of the Engineered Safeguards a-c power busses. Loss of engineered safeguards power to the logic networks or to the safeguards equipment does not initiate safeguards action as described in 7.1.1.2.4.

7.1.2.3.4 Manual System Trip

The manual actuating devices in the protection systems are independent of the automatic trip circuitry and are not subject to failures that make the automatic circuitry inoperable. The manual trip devices are independent control switches for each power controller. The independent control switches, however, are all actuated through a mechanical linkage to a common manual trip switch or pushbutton.

7.1.2.3.5 Equipment Removal

The removal of modules or subassemblies from vital sections of the Reactor Protection System will initiate the trip normally associated with that portion of the system. The removal criterion is implemented in two ways: (1) advantage is taken of the inherent characteristics of a normally energized system, and (2) interlocks are provided.

An inherent characteristic is illustrated by considering the power supply for one of the reactor protection channels. Removal of this power supply automatically results in trip action by virtue of the resulting loss of power. No interlock is required in such cases. Other instances require a system of interlocks built into the equipment to insure trip action upon removal of a portion of the equipment.

The Safeguards Actuation System provides for servicing without affecting the integrity of the redundant channels.

7.1.2.3.6 Testing

The protection systems will meet the testing criterion and its objectives. The test circuits will take advantage of the systems' redundancy, independence, and coincidence features which make it possible to initiate trip signals manually in any part of one protective channel without affecting the other channels.

This test feature will allow the operator to interrogate the systems from the input of any bistable up to the final actuating device at any time during reactor operation without disconnecting permanently installed equipment.

The test of a bistable consists of inserting an analog input and varying the input until the bistable trip point is reached. The value of the inserted test signal represents the true value of the bistable trip point. Thus the test verifies not only that the bistable functions but that the trip point is correctly set.

Prestartup testing will follow the same procedure as the on-line testing except that calibration of the analog instrument strings may be checked with less restraint than during reactor operation.

As shown in Figure 7-2B, the power breakers in the reactor trip circuit may also be manually tested during operation. The only limitation is that not more than one power supply may be interrupted at a time without causing a reactor trip.

7.1.2.3.7 Physical Isolation

The physical arrangement of all elements associated with the protection systems will reduce the probability of a single physical event impairing the vital functions of the system. For example, pressure measurements of reactor coolant pressure will be divided among four redundant pressure taps so as to reduce the probability of collective damage to all sensors by a single accident.

System equipment will be distributed between instrument cabinets so as to reduce the probability of damage to the total system by some single event.

Wiring between vital elements of the system outside of equipment housing will be routed and protected within the unit so as to maintain redundancy of the systems with respect to physical hazards.

7.1.2.3.8 Primary Power Source

The primary source of control power for the Reactor Protection System is the vital busses described in 8.2.2.7. The source of power for the measuring elements in the Safeguards Actuation System is also from the vital busses. Command circuits from the Safeguards Actuation System coincidence logic that extend to Engineered Safeguards Equipment controllers are powered from the Engineered Safeguards d-c busses. Engineered Safeguards equipment such as pump and motor operators and their starting contactors are powered from the Engineered Safeguards a-c busses.

7.1.2.3.9 Reliability

Design criteria for the Reactor Protection System and the Safeguards Actuation System have been formulated to produce reliable systems. System design practices, such as redundant equipment, redundant channels, and coincidence arrangements permitting in-service testing, have been employed to implement reliability of protective action. The best grades of commercially available components will be used in fabrication. A system fault analysis will be made considering the modes of failure and determining their effect on the system vital functions. Acceptance testing and periodic testing will be designed to insure the quality and reliability of the completed systems.

7.1.2.3.10 Instrumentation for Emergency Core Cooling Initiation

The instrumentation system makes use of both physical and electrical isolation. The high pressure and low pressure injection systems are activated by both low reactor coolant and reactor building pressure signals originating from three pressure transmitters measuring the reactor coolant system pressure, as shown in Figure 7-11, and three pressure transmitters measuring the reactor building pressure.

Two reactor coolant pressure transmitters are connected to one reactor pipe; the third transmitter is connected to the other reactor outlet pipe. Each transmitter has a separate tap on the reactor coolant piping inside the secondary shield. The transmitters are physically separated from each other and located outside the secondary shield inside the reactor building. The transmitters' electrical outputs leave the reactor building through separate penetrations.

The three reactor building pressure transmitters are connected to the reactor building through independent taps. The transmitters are physically separated from each other and are located outside the reactor building. The output of each transmitter provides isolated signals to its associated bistable trip devices. The bistable trip devices of a given logic function are physically separated by cabinet barriers. Each pressure transmitter and its associated bistable trips are powered from separate battery-backed vital bus power sources, the same power sources which power the reactor protection channels. Two, isolated 125 volt d-c engineered safeguards control power sources are used for the power to the engineered safeguards channels and logic, as shown in Figure 7-2. Each major function is, therefore, activated from two independent sources of control power.

The operation of the engineered safeguards channels and the trip relays forming the system logic is described in 7.1.2.2.

The high order of system redundancy assures compliance with the single failure criteria of 7.1.1.2.1.

7.1.2.4 Summary of Protective Actions

The abnormal conditions that initiate a reactor trip are as follows:

<u>Trip Variable</u>	<u>No. of Sensors</u>	<u>Steady State Normal Range</u>	<u>Trip Value or Condition for Trip</u>
Neutron Flux	4	0-100%	107.5% of full (rated) power
Neutron Flux/ Reactor Coolant Flow	4 Flux 16 Reactor Coolant Pump Monitors 2 Flow Tubes	3 to 4 pumps	(1) Ratio of reactor power to total reactor coolant flow exceeds 1.07. (2) Loss of two or more reactor coolant pump motors.

<u>Trip Variable</u>	<u>No. of Sensors</u>	<u>Steady State Normal Range</u>	<u>Trip Value or Condition for Trip</u>
Startup Rate	2	0-2 Decades/min	5 Decades/min
Reactor Coolant Pressure	4	2,120-2,250 psig	2,350 psig 2,050 psig
Reactor Outlet Temperature	4	520-603 F	610 F

Actions initiated by the Safeguards Actuation System are as follows:

<u>Action</u>	<u>Trip Condition</u>	<u>Steady State Normal Value</u>	<u>Trip Point</u>	
High Pressure Injection	Low Reactor Coolant Pressure or	2,120 - 2,250 psig	1,800 psig	3
	High Reactor Building Pressure	Atmospheric	10 psig	
Low Pressure Injection	Very Low Reactor Pressure or	2,120 - 2,250 psig	200 psig	3
	High Reactor Building Pressure	Atmospheric	10 psig	
Start Reactor Building Emergency Cooling Unit and Reactor Building Isolation	High Reactor Building Pressure	Atmospheric	4 psig	
Reactor Building Spray	High Reactor Building Pressure	Atmospheric	10 psig	

7.1.2.5 Relationship to Safety Limits

Trip setpoints tabulated in 7.1.2.4 are consistent with the safety limits that have been established from the analysis described in Section 14. The set point for each input, which must initiate a trip of the Reactor Protection System, has been established at a level that will insure that control rods are inserted in sufficient time to protect the reactor core. Likewise, the set points for parameters initiating a trip of the Safeguards Actuation System are established at levels that will insure that corrective action is in progress in sufficient time to prevent an unsafe condition. Factors such as the rate at which the sensed variable can change, instrumentation and

calibration inaccuracies, bistable trip times, circuit breaker trip times, control rod travel times, valve times, and pump starting times have been considered in establishing the margin between the trip set points and the safety limits that have been derived.

The flux trip set point of 107.5 per cent is based upon the tolerances and error bands shown in Figure 7-4. The incident flux error is the sum of the errors at the output of the measuring channel resulting from rod motion, and instrument drift during the interval between heat balance checks of nuclear instrumentation calibration.

7.1.3 SYSTEMS EVALUATION

7.1.3.1 Functional Capability - Reactor Protection System

The Reactor Protection System has been designed to limit the reactor power to a level within the design capability of the reactor core. In all accident evaluations the time response of the sensors and the protective channels are considered. Maximum trip times of the protection channels are listed below.

- a. Temperature - 5 sec
- b. Pressure - 0.5 sec
- c. Flux - 0.3 sec
- d. Pump Monitor - 1.0 sec

Since all uncertainties are considered as cumulative in deriving these times, the actual times may be only one-half as long in most cases. Even these maximum times, when added to control rod drop times, provide conservative protective action.

The Reactor Protection System will limit the power that might result from an unexpected reactivity change. Any change of this nature will be detected and arrested by high reactor coolant temperature, high reactor coolant pressure, or high neutron flux protective action.

An uncontrolled rod withdrawal from startup will be detected by the abnormally fast startup rate in the intermediate channels and high neutron flux in the power range channels. A startup rate trip from the intermediate-range channels is incorporated in the Reactor Protection System.

A rod withdrawal accident at power will immediately result in a high neutron flux trip.

Reduced reactor coolant flow results in a reduced allowable reactor power. The reactor coolant pump monitor operates to set the appropriate reactor power limit by adjusting the power level trip point. A total loss of flow results in a direct reactor trip, independent of reactor power level.

Two major measurements feed the power/flow monitor: (a) reactor coolant flow, and (b) neutron power level. The flow tubes which provide the reactor coolant flow measurement will exhibit no change during the reactor life. A periodic calibration of the flow transmitters will be made. The neutron power level signal will be recalibrated by comparison with a routine heat balance. The power range channels use detectors arranged to effectively average the measurement over the length of the core as described in 7.3.1.1.2. Therefore, their output is expected to be within 4 per cent of the calibrated value during normal regulating rod group position changes and the need for additional calibration thereby eliminated.

3

The loss of reactor coolant will result in a reduction of reactor coolant pressure. The low pressure trip serves to trip the reactor for such an occurrence.

A significant turbine-side steam line rupture is reflected in a drop of reactor coolant pressure. The low reactor pressure trip shuts down the plant for such an occurrence.

7.1.3.2 Functional Capability - Safeguards Actuation System

The Safeguards Actuation System is a graded protection system. The progressive actions of the injection systems as initiated by the Safeguards Actuation System provide sufficient reactor coolant under all conditions while minimizing the possibility of setting the entire system in operation inadvertently.

The key variable associated with the loss of reactor coolant is reactor pressure. In a loss-of-reactor-coolant accident, the reactor pressure will fall, starting high pressure injection at 1,800 psig. If high pressure injection does not arrest the pressure drop, then low pressure injection starts upon a signal of 200 psig. High reactor building pressure at 10 psig is used to provide diversification in actuation of both high pressure injection and low pressure injection.

3

The key variable in the detection of an accident that could endanger reactor building integrity is reactor building pressure. A reactor building pressure of 4 psig initiates operation of the reactor building emergency cooling unit and isolation of the building while a higher pressure of 10 psig initiates operation of the reactor building sprays, and high and low pressure injection systems.

3

7.1.3.3 Preoperational Tests

Valid testing of analog sensing elements associated with the protection systems will be accomplished through the actual manipulation of the measured variable and comparison of the results against a standard.

Routine preoperational tests will be performed by the substitution of a calibrating signal for the sensor. Simulated neutron signals may be substituted in each of the source, intermediate, and power range channels to check the operation of each channel. Simulated pressure, temperature, and level signals

may be used in a similar fashion. This type of testing is valid for all elements of the system except the sensors. The sensors should be calibrated against standards during shutdowns for refueling, or whenever the true status of any measured variable cannot be assessed because of lack of agreement among the redundant measurements.

The final defense against sensor failure during operation will be the plant operator. The redundancy of measurements provides more than adequate opportunity for comparative readings. In addition, the redundancy of the systems reduces the consequences of a single sensor failure.

7.1.3.4 Component Failure Considerations

The effects of failure can be understood through Figure 7-2B. In the Reactor Protection System, the failure of any single input in the "tripped" direction places the system in a 1-out-of-3 mode of operation for all variables. Failure of any single input in the "cannot trip" direction places the system in a 2-out-of-3 mode of operation for the variable involved, but leaves all other variables in the normal 2-out-of-4 coincidence mode. With a "tripped," open circuit fault in one channel, the system would be able to tolerate a minimum of two "cannot trip," short circuit failures within the same measured variable before complete safety protection of the variable were lost. With one "tripped," open circuit fault, a second identical fault within the same variable would trip the reactor. 3

A similar fault relationship exists between channels as a result of the 2-out-of-4 coincidence output. One "trip" faulted channel places the system in a 1-out-of-3 or single-channel mode. A "cannot trip" faulted channel places the system in a 2-out-of-3 mode.

At the final device, a "trip" faulted power breaker does not affect the protection channel mode of operation, reactor trip being dependent upon one of two breakers in the unaffected primary power supply to the control rod drives. A breaker faulted in the "cannot trip" mode leaves the system dependent upon the second breaker in the affected primary power supply.

The Safeguards Actuation System is a 2-out-of-3 input type of system. It can tolerate one fault of the "cannot trip" variety in each of the coincidence networks. For this type of fault, all remaining inputs must function correctly. A "tripped" input fault allows any one of the two remaining inputs to initiate action.

Primary power input to both protection systems has been arranged to minimize the possibility of loss of power to either protection system. Each channel of the protection system will be supplied from one of the four vital busses described in 8.2.2.7. The operator can initiate a reactor trip independent of the automatic protection action.

The engineered safeguards have been connected to multiple busses to minimize total loss of safeguard capability. The individual parts of the Safeguards Actuation System can be placed in operation through manual operator controls independent of the automatic protection equipment.

7.1.3.5 Operational Tests

The protection systems are designed and have the facilities for routine manual operational testing.

Most inputs to the protection systems originate from an analog measurement of a particular variable. Every input of this type is equipped with a continuous readout device. A routine check by the operator of each reading as compared to the other redundant readings available for each variable will uncover measurement faults. These elements plus the bistables and relays of the protection systems require a periodic dynamic test. Each system provides for routine testing. Each bistable may be manually tripped, and the results of that trip traced through the system logic and visually indicated to the operator. The trip point setting of each bistable may be verified by the application of an analog signal proportional to the measured variable, and that signal may be varied until the bistable element trips.

7.2 REGULATING SYSTEMS

7.2.1 DESIGN BASES

7.2.1.1 Compensation Considerations

Reactor regulation is based upon the use of movable control rod assemblies* and chemical neutron absorber (boric acid) dissolved in the reactor coolant.

Relatively fast reactivity effects including Doppler, xenon, and moderator temperature are controlled by the control rods, which are capable of rapid compensation. Relatively slow reactivity effects, such as fuel burnup, fission product buildup, samarium buildup, and hot-to-cold moderator deficit, are controlled by soluble boron.

It is possible to change the reactor coolant system boric acid concentration to "follow" xenon transients over approximately 70 per cent of each core cycle without control rod operation. However, to reduce waste handling requirements resulting from chemical shim operation, control rods are used throughout core life for xenon transient associated with normal power changes. Chemical shim is used in conjunction with control rods to compensate for equilibrium xenon conditions.

*Control rod, rod, and control rod assembly (CRA) are used interchangeably in this section and elsewhere in this report.

At the beginning of first core life when the moderator temperature reactivity coefficient may be zero or slightly positive, the control rod drive response is faster than necessary to maintain the power error within the allowed deadband. Analog computer analysis shows that the only change in control response when a positive moderator coefficient of reactivity exists is an increased frequency of control rod motion.

The reactor controls are designed to maintain a constant average reactor coolant temperature over the load range from 15 to 100 per cent of rated power. The steam system operates on constant pressure at all loads. The average reactor coolant temperature decreases over the range from 15 per cent load to zero load. Figure 7-5 shows the reactor coolant and steam temperatures over the entire load range.

Input signals to the reactor controls include reactor coolant average temperature, megawatt demand, and reactor power as indicated by out-of-core neutron detectors. The soluble boron dilution is initiated manually and terminated automatically or manually. Manual rod control is used below 15 per cent of rated power. Automatic or manual rod control may be used above 15 per cent of rated power.

Increasing power transients between 20 and 90 per cent power are limited to ramp changes of 10%/min and step increases of 10 per cent. Power increases from 15-20 per cent and above 90 per cent are limited to 5%/min. Decreasing power transients between 100 and 15 per cent power are limited to ramp changes of 10%/min and step decreases of 10 per cent. The turbine bypass system permits a load drop of 40 per cent, or a turbine trip from 40 per cent load without safety valve operation. The turbine bypass system and safety valves permit a 100 per cent load drop without turbine trip to satisfy "blackout" requirements as described in 14.1.2.8.2.

7.2.1.2 Safety Considerations

7.2.1.2.1 Shutdown Margin

The control rods are provided in sufficient number to allow a hot shutdown that is greater than 1 per cent subcritical with rod assembly of greatest worth fully withdrawn and a typical level of soluble boron (Figure 3-1).

7.2.1.2.2 Reactivity Rate Limits

The maximum average rate of change of reactivity that can be inserted by any group of rods does not exceed 5.8×10^{-5} ($\Delta k/k$)/sec. (The accidental withdrawal of the rod group of greatest worth is discussed in 14.1.2.2 and 14.1.2.3.)

The maximum normal rate of pure water addition does not change reactivity worth more than 3×10^{-6} ($\Delta k/k$)/sec. Reactivity control may be exchanged between rods and soluble boron consistent with the design bases listed above.

7.2.1.2.3 Power Peaking Limits

The nominal reactivity available to a power regulating control rod group is limited so that established radial and axial flux-peaking limits are not exceeded with the rod group in any position at power levels up to 100 per cent power.

7.2.1.2.4 Power Level Limits

The reactor automatic controls incorporate a high limit and a low limit of power level demand to the reactor. Limits are imposed on reactor megawatt demand by lack of feedwater flow capability and reactor coolant system flow capability.

7.2.1.3 Startup Considerations

Over the life of the nuclear unit, startup will occur at various temperature levels and after varying periods of downtime. Examples of regulating system design requirements as related to startup are

- a. Control rod and/or control rod group "withdraw inhibit" on high start-up rate (short period) in the source range and intermediate range.
- b. Reactor trip on high startup rate in the intermediate range.
- c. Startup control mode. This mode prevents automatic rod withdrawal below 15 per cent power.
- d. In startup control mode, the controls are arranged so that the steam system follows reactor power rather than turbine system power demand. The controls will limit steam dump to the condenser when condenser vacuum is inadequate.
- e. Sufficient control rod worth is provided to override peak xenon and return to power following a hot shutdown or hot standby. During cold shutdown it will be necessary to increase boron concentration to maintain shutdown margin. Following a cold shutdown, boron concentration changes will be made during startup. A number of rod assemblies (or groups), sufficient to provide 1 per cent shutdown margin during startup, are required to be withdrawn before a dilution cycle.
- f. Minimum pressurizer water level conditions must be met before and during startup.

7.2.2 SYSTEM DESIGN

7.2.2.1 Description of Reactivity Control

7.2.2.1.1 General Description

The reactor controls move control rods to regulate the power output of the reactor and maintain constant reactor coolant average temperature above 15 per cent rated power. As shown in Figure 7-6, the megawatt demand signal is added to the reactor coolant average temperature error to form a reactor power level demand signal. The reactor power level demand signal is compared to the average reactor power level measured by the power range detectors in the nuclear instrumentation. When the resulting reactor power level error signal exceeds the deadband, the output signal is a control rod drive "withdraw" or "insert" command to the controlling rod group. For reactivity control limits see 3.1.2.2.

7.2.2.1.2 Reactivity Control

Reactivity control is maintained by movable control rod assemblies and by soluble boron dissolved in the reactor coolant. The moderator temperature coefficient (cold to hot critical), as well as long-term reactivity changes caused by fuel burnup and fission product poisoning, are controlled by adjusting soluble boron concentration. Short-term reactivity changes caused by power change, xenon poisoning, and moderator temperature change from 0 to 15 per cent power are controlled by control rods.

First-cycle values for the reactivity components and control distribution are listed in Tables 3-4 and 3-5.

Twenty-one of the 69 control rod assemblies are assigned to automatic control of reactor power level during the first core cycle. Thereafter, 25 rod assemblies are used. These control rod assemblies are arranged in four symmetrical groups which operate in sequence. The position of one automatic group is used as an index to soluble boron dilution. Soluble boron adjustment is initiated manually and terminated automatically. The position of this group acts as a "permissive" to restrict the start of dilution to a "safe" rod position pattern. The position of the same group terminates dilution automatically.

During reactor startup, control rods are withdrawn in a predetermined sequence in symmetrical groups of four or more rod assemblies. The group size is preset, and individual control rod assembly assignments to a group are made at a control rod grouping panel. However, the operator can select any individual control rod assembly and any rod group for motion as required.

A typical control rod group withdrawal scheme is as follows:

	<u>First Cycle</u>		<u>Equilibrium Cycle</u>
Group 1	16 CRA's		12 CRA's
Group 2	12 CRA's		12 CRA's
Group 3	12 CRA's		12 CRA's
Group 4	8 CRA's		8 CRA's
Group 5	4 CRA's	} Regulating Groups	8 CRA's
Group 6	8 CRA's		9 CRA's
Group 7	5 CRA's		4 CRA's
Group 8	4 CRA's		4 CRA's

An automatic sequence logic unit is used for reactor control with four regulating rod groups in the power range. This unit allows operation of no more than one control rod group simultaneously except over the last 25 per cent travel of one group and the first 25 per cent travel of the next group when overlapping motion of two groups is permitted. This tends to linearize the reactivity insertion from group to group as shown in Figure 7-7.

As fuel burnup progresses, dilution of the soluble boron is controlled as follows:

When the partially withdrawn active control rod group reaches the fully withdrawn point, interlock circuitry permits setting up a flow path from a demineralized water tank, in lieu of the normal flow path of borated makeup, to the reactor coolant system. Demineralized water is fed to the reactor coolant system, and borated reactor coolant is removed.

The reactor controls insert the active regulating rod group to compensate for the reduction in boron concentration. When the control rod group has been inserted to the 75 per cent withdrawn position, the dilution flow is automatically blocked. The dilution cycle is also terminated automatically by a preset timing device, which is independent of rod position. Normally, a dilution cycle is required every several days.

7.2.2.1.3 Reactivity Worth

The maximum worth of any group of the four automatic control groups is approximately 1.2% $\Delta k/k$. At design speed, a group requires approximately 6 minutes to travel full stroke. This rate of control rod group travel results in a reactivity rate of 5.8×10^{-5} ($\Delta k/k$)/sec.

The maximum rate of reactivity addition with the soluble boron system, i.e., injecting unborated water from the makeup system at 70 gpm maximum, is 3.0×10^{-6} ($\Delta k/k$)/sec.

Table 3-5 shows a shutdown reactivity analysis. The rod worth provided gives a shutdown margin of 5.1% $\Delta k/k$ or more under normal conditions, and a margin in excess of 1% $\Delta k/k$ with the CRA of greatest worth stuck in the withdrawn position.

Under conditions where cooldown to reactor building ambient conditions is required, concentrated soluble boron will be added to the reactor coolant to produce a shutdown margin of at least 1% $\Delta k/k$. The reactivity changes from hot zero power to a cold condition, and the corresponding increases in boric acid concentration, are listed in Table 3-6.

7.2.2.1.4 Reactor Control

The reactor control is made up of analog computing equipment with inputs of megawatt demand, core average power, and reactor coolant average temperature. The output of the controller is an error signal that causes the control rod drive to be positioned until the error signal is within a deadband. A block diagram of the reactor control is shown in Figure 7-6.

First, reactor power level demand (N_d) is computed as a function of the megawatt demand (MW_d) and the reactor coolant system average temperature deviation (ΔT) from the set point, according to the following equation:

$$N_d = K_1 MW_d + K_2 \left(\Delta T + \frac{1}{\tau} \int \Delta T dt \right)$$

Megawatt demand is introduced as a part of the demand signal through a proportional unit having an adjustable gain factor (K_1). The temperature deviation is introduced as a part of the demand signal after proportional plus reset (integral) action is applied. For the temperature deviation, K_2 is the adjustable gain and τ is the adjustable integration factor.

The reactor power level demand (N_d) is then compared with the average reactor power level signal (\bar{N}_1), which is derived from the nuclear instrumentation. The resultant error signal ($N_d - \bar{N}_1$) is the reactor power level error signal (E_p).

When the reactor power level error signal (E_p) exceeds the deadband settings, the control rod drive receives a command that withdraws or inserts rods depending upon the polarity of the power error signal.

The following additional features are provided with the reactor power controller:

- a. An adjustable low limit on the megawatt demand signal (MW_d) to cut out the automatic reactor control action.
- b. A high limit on reactor power level demand (N_d).
- c. An adjustable low limit on reactor power level demand (N_d).

Separate from, but related to the automatic reactor control system is the reactor coolant flow signal. The reactor coolant flow is measured and limits the unit megawatt demand and respective steam generator feedwater demand to within the capability of the available reactor coolant flow.

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In addition to measuring the reactor coolant flow, digital logic units continually monitor the number of energized reactor coolant pumps and establish a maximum limit on the megawatt demand.

7.2.2.2 Integrated Control System

The Integrated Control System maintains constant average reactor coolant temperature and constant steam pressure in the nuclear unit during steady state and transient operation between 15 and 100 per cent rated power. Figures 7-6 and 7-8 show the overall system. The system is based on the Integrated Boiler-Turbine concept widely used in fossil-fuel-fired utility plants. It combines the stability of a turbine-following system with the fast response of a boiler-following system. Optimum overall unit performance is maintained by limiting steam pressure variations; by limiting the unbalance that can exist among the steam generator, turbine, and the reactor; and by limiting the total unit load demand upon loss of capability of the steam generator feed system, the reactor, or the turbine generator.

Figure 7-6 shows the reactor control portion of the Integrated Control System described in 7.2.2.1.4. Figure 7-8 shows the steam generator and turbine control portion of the Integrated Control System. This control receives inputs of megawatt demand, system frequency, and steam pressure, and supplies output signals to the turbine bypass valve, turbine speed changer, and steam generator feedwater flow controls with changing operating conditions.

The turbine and steam generator are capable of automatic control from zero power to rated power with optional manual control. The reactor controls are designed for manual operation below 15 per cent rated power and for automatic or manual operation above 15 per cent rated power.

The turbine is operated as a turbine-following unit with the turbine control valve pressure set point varied in proportion to megawatt error. The steam generator is operated as a boiler-following system in which the feed-water flow

7-19a

demand to the steam generator is a summation of the megawatt demand and the steam pressure error.

The Integrated Control System obtains a load demand signal from the system dispatch center or from the operator. A frequency loop is added to match the speed droop of the turbine speed controls. The load demand is restrained by a maximum load limiter, a minimum load limiter, a rate limiter, and a runback limiter. In normal operation the limits would be set as follows:

	<u>Megawatt Demand (MW_d)</u>
Maximum load limit	100%
Minimum load limit	15%
Rate limit	10%

The runbacks act to runback and/or limit the load demand on any of the following conditions:

- a. One or more reactor coolant pumps are inoperative.
- b. Total feedwater flow lags total feedwater demand by more than 5 per cent.
- c. The four shim/safety rod groups are not fully withdrawn.
- d. Assymmetric rod withdrawal patterns exist.
- e. The generator separates from the 500 kv bus.

The output of the limiters is a megawatt demand signal which is applied to the turbine control, steam generator control, and reactor control in parallel. The reactor control responds to the megawatt demand signal as described in 7.2.2.1.4.

7.2.2.2.1 Turbine Control

The megawatt demand is compared with the generator megawatt output, and the resulting megawatt error signal is used to change the steam pressure set point. The turbine valves then change position to control steam pressure. As the megawatt error reduces to zero, the steam pressure set point is returned to the steady state value. By limiting the effect of megawatt error on the steam pressure set point, the system can be adjusted to permit controlled variations in steam pressure to achieve any desired rate of turbine response to megawatt demand.

7.2.2.2.2 Steam Generator Control

Control of the steam generator is based on matching feedwater flow to megawatt demand with bias provided by the error between steam pressure set point and steam pressure. The pressure error increases the feedwater flow demand if the pressure is low. It decreases the feedwater flow demand if the pressure is high.

The basic control actions for parallel steam generator operation are:

- a. Megawatt demand converted to feedwater demand.

- b. Steam pressure compared to set pressure, and the pressure error converted to feedwater demand.
- c. Total feedwater demand computed from sum of a and b.
- d. Total feedwater flow demand split into feedwater flow demand for each steam generator.
- e. Feedwater demand compared to feedwater flow for each steam generator. The resulting error signals position the feedwater flow controls to match feedwater flow to feedwater demand for each steam generator.

For operation below 15 per cent load, the steam generator control acts to maintain a preset minimum downcomer water level. The conversion to level control is automatic and is introduced into the feedwater control train through an actuator. At low loads below 15 per cent, the turbine bypass valves will operate to limit steam pressure rise.

The steam generator control also provides ratio, limit, and runback actions as shown in Figure 7-8, which include

a. Steam Generator Load Ratio Control

Under normal conditions the steam generators will each produce one-half of the total load. Steam generator load ratio control is provided to balance reactor inlet coolant temperatures during operation with more reactor coolant pumps in one loop than in the other.

b. Rate Limits

Rate limiters are manually set to restrict loading or unloading rates to those that are compatible with the turbine and/or the steam generator.

c. Water Level Limits

A maximum water level limit prevents gross overpumping of feedwater and insures superheated steam under all operating conditions.

A minimum water level limit is provided for low load control.

d. Reactor Coolant Pump Limiters

These limiters restrict feedwater demand to match reactor coolant pumping capability. For example, if one reactor coolant pump is not operating, the maximum feedwater demand to the steam generator in the loop with the inoperative pump is limited to approximately one-half normal.

e. Reactor Outlet and Feedwater Low Temperature Limits

These limiters reduce feedwater demand when the reactor outlet temperature or the feedwater temperature is low.

f. Feedwater Pump Capability

A feedwater pump capability runback signal limits the megawatt demand signal whenever total feedwater flow lags total feedwater demand by 5 per cent.

7.2.3 SYSTEM EVALUATION

7.2.3.1 System Failure Considerations

Redundant sensors are available to the Integrated Control System. The operator can select any of the redundant sensors from the control room.

Manual reactivity control is available at all power levels.

Loss of electrical power to the automatic controller reverts reactor control to the manual mode.

7.2.3.2 Interlocking

Control rod withdrawal is prevented on the occurrence of a positive short period below 10 per cent power.

The automatic sequence logic sets a predetermined insertion and withdrawal pattern of the four regulating rod groups.

Control circuitry allows manually selected operation of any single control rod assembly or control rod group throughout the power range.

An interlock will prevent actuation of both withdrawal and insertion of control rods simultaneously with the insertion signal overriding the withdrawal.

Control rod drive switching circuits allow withdrawal of no more than a single control rod group in the manual mode.

The automatic sequence logic limits regulating rod motion to one group out of four at one time except at the upper and lower 25 per cent of stroke where operation of two groups is permitted to linearize reactivity versus stroke.

Maximum and minimum limits on the reactor power level demand signal (N_d) prevent the reactor controls from initiating undesired power excursions.

Maximum and minimum levels on the megawatt demand signal (MW_d) prevent the reactor controls from initiating undesired power excursions.

7.2.3.3 Emergency Considerations

Loss of power to the control rod drive magnetic clutch initiates a reactor trip.

When emergency conditions arise that exceed the capability of the control system, the operator can revert to the manual control mode.

7.2.3.4 Loss-of-Load Considerations

The nuclear unit is designed to accept 10 per cent step load rejection without safety valve action or turbine bypass valve action. The combined actions of the control system and the turbine bypass valve permit a 40 per cent load reduction or a turbine trip from 40 per cent load without safety valve action. The controls will limit steam dump to the condenser when condenser vacuum is inadequate, in which case the safety valves may operate. The combined actions of the control system, the turbine bypass valve, and the safety valves permit a 100 per cent load rejection without turbine trip. This permits the unit to ride through a "blackout" condition, i.e., sudden rejection of electrical load down to auxiliary load without turbine trip. (The "blackout" provisions are discussed in 14.1.2.8.2.)

The features that permit continued operation under load rejection conditions include

a. Integrated Control System

During normal operation the Integrated Control System (see Figure 7-8), controls the unit load in response to load demand from the system dispatch center or from the operator. During normal load changes and small frequency changes, turbine control is through the speed changer to maintain constant steam pressure.

During large load and frequency upsets, the turbine governor takes control to regulate frequency. For these upset conditions, frequency error at the input to the integrated control system becomes more important in providing load matching.

b. 100 Per Cent Relief Capacity in the Steam System

This provision acts to reduce the effect of large load drops on the reactor system.

Consider, for example, a sudden load rejection greater than 10 per cent. When the turbine generator starts accelerating, the governor valves and the intercept valves begin to close to maintain set frequency. At the same time the megawatt demand signal is reduced, which reduces the governor speed changer setting, feedwater flow demand, and reactor power level demand. As the governor valves close, the steam pressure rises and acts through the control system to reinforce the feedwater flow demand reduction already initiated by the reduced megawatt demand signal. In addition, when the load rejection is of sufficient magnitude, the turbine bypass valves open to reject excess steam to the condenser, and the safety valves open to exhaust steam to the atmosphere. The rise in steam pressure and the reduction in feedwater flow cause the average reactor coolant temperature to rise which reinforces the reactor power level demand reduction, already established by reduced megawatt demand, to restore reactor coolant temperature to set value.

As the turbine generator returns to set frequency, the turbine controls revert to steam pressure control rather than frequency control. This feature holds steam pressure within relatively narrow limits and prevents further large steam pressure changes which could impose additional load changes of opposite sign on the reactor coolant system. As a result, the reactor, the reactor coolant system, and the steam system run back rapidly and smoothly to the new load level.

7.3 INSTRUMENTATION

7.3.1 NUCLEAR INSTRUMENTATION

The nuclear instrumentation system is shown in Figure 7-2A. Emphasis in the design is placed upon accuracy, stability, and reliability. Instruments are redundant at every level. The design criteria stated in 7.1.1.2 have been applied to the design of this instrumentation.

7.3.1.1 Design

The nuclear instrumentation has eight channels of neutron information divided into three ranges of sensitivity: source range, intermediate range, and power range. The three ranges combine to give a continuous measurement of reactor power from source level to approximately 125 per cent of rated power or ten decades of information. A minimum of one decade of overlapping information is provided between successive higher ranges of instrumentation. The relationship between instrument ranges is shown in Figure 7-9.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity proportional counters. These channels are used over a counting range of 1 to 10^5 counts/sec as displayed on the operator's control console in terms of log counting rate. The channels also measure the rate of change of the neutron level as displayed for the operator in terms of startup rate from -1 to +10 decades/min. No protective functions are associated with the source range because of inherent instrumentation limitations encountered in this range. However, one interlock is provided, i.e., a control rod withdraw hold and alarm on high startup rate in either channel.

The intermediate range instrumentation has two log N channels originating in two identical electrically gamma-compensated ion chambers. Each channel provides seven decades of flux level information in terms of log ion chamber current and startup rate. The ion chamber output range is from 10^{-11} to 10^{-4} amperes. The startup rate range is from -1 to +10 decades per minute. Protective action on high startup rate is provided by these channels. A high startup rate on either channel causes a reactor trip. Prior to a reactor trip, high startup rate in either channel will initiate a control rod withdraw hold interlock and alarm.

The power range channels have four linear level channels originating in 12 uncompensated ion chambers. The channel output is directly proportional to reactor power and covers the range from 0 to 125 per cent of rated power. The system is a precision analog system which employs a digital technique to provide highly accurate signals for instrument calibration and reactor trip set point calibration. The gain of each channel is adjustable, providing a means for calibrating the output against a reactor heat balance. Protective action on high flux level consists of reactor trip initiation by the power range channels at preset flux levels.

Additional features pertinent to the nuclear instrumentation system are as follows:

- a. Independent power supplies are included in each channel. Primary power originates from the vital busses described in 8.2.2.7. Where applicable, isolation transformers are provided to insure a stable, high-quality power supply.
- b. The proportional counters used in the source range are designed to be secured when the flux level is greater than their useful operating range. This is necessary to obtain prolonged operating life.
- c. The intermediate range channels are supplied with an adjustable source of gamma-compensating voltage.

7.3.1.1.1 Test and Calibration

Test and calibration facilities are built into the system. The test facilities will meet the requirements outlined in the discussion of protection systems testing.

Facilities for calibration of the various channel amplifiers and measuring equipment will also be a part of the system.

7.3.1.1.2 Power Range Detectors

Twelve uncompensated ionization chambers are used in the power range channels. Three chambers are associated with each channel, i.e., one near the bottom of the core, a second at the midplane, and a third toward the top of the core. The outputs of the three chambers are combined in their respective linear amplifiers. A means is provided for reading the individual chamber outputs as a manual calibration and test function during normal operation.

7.3.1.1.3 Detector Locations

The physical locations of the neutron detectors are shown in Figure 7-10. The power range detectors are located in four primary positions, 90 degrees apart around the reactor core.

The two source range proportional counters are located on opposite sides of the core adjacent to two of the power range detectors.

The two intermediate range compensated ion chambers are also located on opposite sides of the core, but rotated 90 degrees from the source range detectors.

7.3.1.2 Evaluation

The nuclear instrumentation will monitor the reactor over the 10-decade range from source to 125 per cent of rated power. The full power neutron flux level at the power range detectors will be approximately 10^9 nv. The detectors employed will provide a linear response up to approximately 4×10^{10} nv before they are saturated.

The intermediate range channels overlap the source range and the power range channels in an adequate manner, providing the continuity of information needed during startup.

The axial and radial flux distribution within the reactor core will be measured by the incore neutron detectors (7.3.3). The out-of-core detectors are primarily for reactor safety, control, and operation information.

7.3.1.2.1 Loss of Power

The nuclear instrumentation draws its primary power from redundant battery-backed vital busses described in 8.2.2.7.

7.3.1.2.2 Reliability and Component Failure

The requirements established for the reactor protection system apply to the nuclear instrumentation. All channel functions are independent of every other channel, and where signals are used for safety and control, electrical isolation is employed to meet the criteria of 7.1.1.2.

7.3.1.2.3 Protection Requirements

The relation of the power range channels to the Reactor Protection System has been described in 7.1. To maintain the desired accuracy in trip action, the total error from drift in the power range channels will be held to $\pm 1/2$ per cent at rated power over a 30-day period. Routine tests and recalibration will insure that this degree of deviation is not exceeded. Bistable trip set points of the power range channels will also be held to an accuracy of $\pm 1/2$ per cent of rated power. The accuracy and stability of the equipment will be verified by vendor tests.

7.3.2 NONNUCLEAR PROCESS INSTRUMENTATION

7.3.2.1 System Design

The nonnuclear instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam system, and reactor auxiliary systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the nuclear unit are indicated, recorded, and controlled from the control room. The quantity and types of process instrumentation provided will insure safe and orderly operation of all systems and processes over the full operating range of the plant. The amounts and types of various instruments and controllers shown are intended to be typical examples of those that will be included in the various systems when final design details have been completed. The nonnuclear process instrumentation for the reactor coolant is shown in Figure 7-11 and on the reactor auxiliary system drawings in Sections 5, 6, 9, and 11. Process variables are monitored as shown on the nonnuclear instrumentation and reactor auxiliary system drawings and are as follows:

- a. In general, resistance elements are used for temperature measurements. Fast-response resistance elements monitor the reactor outlet temperature. The outputs of these fast-response elements supply signals to the protective system.
- b. Pressures are measured in the reactor coolant system, the steam system, and the reactor auxiliary systems. Pressure signals for high and low reactor coolant pressures and high reactor building pressure are provided to the protection systems.
- c. The number of reactor coolant pump motors in operation is monitored. In addition, there is a direct measurement of reactor coolant flow. This information is fed to the reactor controls and reactor protection system.

- d. Flow in the steam system is obtained through the use of calibrated feedwater flow nozzles. Flow information is utilized for control and protective functions in the steam system. Steam generator level measurements are provided for control and alarm functions.
- e. Pressurizer level is measured by differential pressure transmitters calibrated to operating temperature and pressure. The pressurizer level is a function of the reactor coolant system makeup and letdown flow rate. The letdown flow rate is remote manually controlled to the required flow. Pressurizer level signals are processed in a level controller whose output positions the makeup control valve in the makeup line to maintain a constant level.
- f. Reactor coolant system pressure is maintained by a control system that energizes pressurizer electrical heaters in banks at preset pressure values below 2,175 psig or actuates spray control valves if the pressure increases to 2,230 psig.

7.3.2.2 System Evaluation

Redundant instrumentation has been provided for all inputs to the protection systems and vital control circuits.

Where wide process variable ranges are required and precise control is involved both wide-range and narrow-range instrumentation are provided.

Where possible, all instrumentation components are selected from standard commercially available products with proven operating reliability.

All electrical and electronic instrumentation required for safe and reliable operation will be supplied from redundant vital a-c instrumentation busses.

7.3.3 INCORE MONITORING SYSTEM

7.3.3.1 Design Basis

The incore monitoring system provides neutron flux detectors to monitor core performance. No protective action or direct control functions are performed by this system. All high pressure system connections are terminated within the reactor building. Incore, self-powered neutron detectors measure the neutron flux in the core to provide a history of power distributions and disturbances during power operating modes. Data obtained will provide measured power distribution information and fuel burnup data to assist in fuel management decisions.

7.3.3.2 System Design

7.3.3.2.1 System Description

The incore monitoring system consists of assemblies of self-powered neutron detectors located at 52 preselected radial positions within the core. The incore monitoring locations are shown on Figure 7-12. In this arrangement, an incore detector assembly, consisting of six local flux detectors and one background detector, is installed in the instrumentation tube of each of 52 fuel assemblies (Figure 3-62). The local detectors are positioned at six different axial elevations to provide the axial flux gradient. The outputs of the local flux detectors are referenced to the background detector output so that the differential signal is a true measure of neutron flux. | 3

As shown in Figure 7-12, seventeen detector assemblies are located to act as symmetry monitors. The remaining 35 detector assemblies, plus five of the 17 symmetry monitors, provide monitoring of every type of fuel assembly in the core when quarter core symmetry exists.

Readout for the incore detectors is performed by the computer system rather than by individual indicators. This system sounds alarms if local flux conditions exceed predetermined values. | 3

When the reactor is depressurized, the incore detector assemblies can be inserted or withdrawn through guide tubes which originate at a shielded area in the reactor building as shown in Figure 7-13. These guide tubes, after completing two 90 degree turns, enter the bottom head of the reactor vessel where internal guides extend up to the instrumentation tubes of 52 selected fuel assemblies. The instrumentation tube then serves as the guide for the incore detector assembly. The incore detector assemblies are fully withdrawn only for replacement. During refueling operations, the incore detector assemblies are withdrawn approximately 13 feet to allow free transfer of the fuel assemblies. After the fuel assemblies are placed in their new locations, the incore detector assemblies are returned to their fully inserted positions in the core, and the high pressure seals are secured.

7.3.3.2.2 Calibration Techniques

The nature of the detectors permits the manufacture of nearly identical detectors which will produce a high relative accuracy between individual detectors. The detector signals must be compensated for burnup of the neutron sensitive material. The data handling system integrates each detector output current and generates a burnup correction factor to be applied to each detector signal before printing out the corrected signal in terms of per cent of rated power. The data handling system computes an average power value for the entire core, normalized to the reactor heat balance. This average power value is compared to each neutron detector signal to provide the core power distribution pattern.

7.3.3.3 System Evaluation

7.3.3.3.1 Operating Experience

The AECL has been operating incore, self-powered neutron detectors at Chalk River since 1962. They have been successfully applied to both the NRX and NRU reactors and have been operated at fluxes beyond those expected in normal pressurized water reactor service.

7.3.3.3.2 B&W Experience

Self-powered, incore neutron detectors have been assembled and irradiated in The Babcock & Wilcox Company Development Program that began in 1964. Results from this program have produced confidence that self-powered detectors used in an incore instrument system for pressurized water reactors will perform as well as, if not better than, any system of incore instrumentation currently in use.

The B&W Development Program includes these tests:

- a. Parametric studies of the self-powered detector.
- b. Detector ability to withstand PWR environment.
- c. Multiple detector assembly irradiation tests.
- d. Background effects.
- e. Readout system tests.
- f. Mechanical withdrawal-insertion tests.
- g. Mechanical high pressure seal tests.
- h. Relationship of flux measurement to power distribution experiments.

Preliminary conclusions drawn from the results of the test programs at the B&W Lynchburg Pool Reactor, the B&W Test Reactor, and the Big Rock Point Nuclear Power Plant are as follows:

- a. The detector sensitivity, resistivity, and temperature effects are satisfactory for use.
- b. A multiple detector assembly can provide axial flux data in a single channel and can withstand reactor environment. An assembly of six local flux detectors, three background detectors, and two thermocouples has been successfully operating in the Big Rock Point Reactor since May 1966.
- c. Data collection systems are successful as read-out systems for incore monitors.
- d. Background effects will not prevent satisfactory operation in a PWR environment.

Irradiation of detector assemblies and evaluation of performance data are continuing to provide detailed design information for the incore instrumentation system.

7.4 OPERATING CONTROL STATIONS

Following proven power station design philosophy, all control stations, switches, controllers, and indicators necessary to start up, operate, and shut down the nuclear unit will be located in the control room. Control functions necessary to maintain safe conditions after a loss-of-coolant accident will be initiated from the centrally located control room. Controls for certain auxiliary systems may be located at remote control stations when the system controlled does not involve power generation control or emergency functions.

Since there will be no flammable materials in the control room except for charts, records, and some of the electronics, a fire resulting in loss of redundant control systems is not considered possible. Shielding around the control room is provided to insure operator safety during any MHA. No incident or combination will result in loss of control room functions.

7.4.1 GENERAL LAYOUT

The control room will be designed so that one man can supervise operation during normal steady state conditions. During other than normal operating conditions, other operators will be available to assist the control operator. Figure 7-14 shows the control room layout for the unit. The control board is divided into relative areas to show the location of control stations and information display pertaining to various subsystems.

7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTION

The necessary information for routine monitoring of the nuclear unit and the plant will be displayed on the control room benchboard cubicles in the immediate vicinity of the operator. Information display and control equipment frequently employed on a routine basis, or protective equipment quickly needed in case of an emergency, will be mounted on the benchboards. Recorders and radiation monitoring equipment will be mounted on the vertical panel sections of the cubicles. Infrequently used equipment, such as indicators and controllers used primarily during startup or shutdown, will be mounted on side panel sections of the cubicles.

A plant computer will be available in the control room for alarm monitoring, performance monitoring, and data logging. On-demand printout is available to the operator at his discretion in addition to the computer periodic logging of plant variables.

7.4.3 SUMMARY OF ALARMS

Visible and audible alarm units will be incorporated into the control room to warn the operator if unsafe conditions are approached by any system. Audible reactor building evacuation alarms are to be initiated from the radiation monitoring system or manually by the operator. Audible alarms will be sounded in appropriate areas throughout the plant if high radiation conditions are present.

7.4.4 COMMUNICATION

Plant telephone and paging systems will be provided with redundant power supplies to provide the control room operator with constant communication with all areas of the plant. Acoustic booths will be supplied in areas where the background noise level is high. Communication outside the plant will be through the Bell System and the Arkansas Power & Light Company microwave system.

7.4.5 OCCUPANCY

Safe occupancy of the control room during abnormal conditions will be provided for in the design of the auxiliary building. Adequate shielding will be used to maintain tolerable radiation levels in the control room for maximum hypothetical accident conditions. The control room ventilation system will be provided with radiation detectors and appropriate alarms. Provisions will be made for the control room air to be recirculated through absolute and charcoal filters. Emergency lighting will be provided.

The potential magnitude of a fire in the control room will be limited by the following factors:

- a. The control room construction will be of noncombustible materials.
- b. Control cables and switchboard wiring will be constructed such that they have passed the flame test as described in Insulated Power Cable Engineers Association Publication S-61-402 and National Electrical Manufacturers Association Publication WC 5-1961.
- c. Furniture used in the control room will be of metal construction.
- d. Combustible supplies such as logs, records, procedures, manuals, etc., will be limited to the amounts required for plant operations.
- e. All areas of the control room will be readily accessible for fire extinguishing.
- f. Adequate fire extinguishers will be provided.
- g. The control room will be occupied at all times by a qualified person who has been trained in fire extinguishing techniques.

The only flammable materials inside the control room will be:

- a. Paper in the form of logs, records, procedures, manuals, diagrams, etc.
- b. The coaxial cables required for nuclear instrumentation.
- c. Small amounts of combustible materials used in the manufacture of various electronic equipment.

The above list indicates that the flammable materials will be distributed to the extent that a fire would be unlikely to spread. Therefore, a fire, if

started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the ventilation system.

Essential auxiliary equipment will be controlled by stored energy, closing-type, air-circuit breakers which will be accessible and can be manually closed in the event d-c control power is lost.

7.4.6 AUXILIARY CONTROL STATIONS

Auxiliary control stations will be provided where their use simplifies control of auxiliary systems equipment such as sample valve selectors, chemical addition, etc.

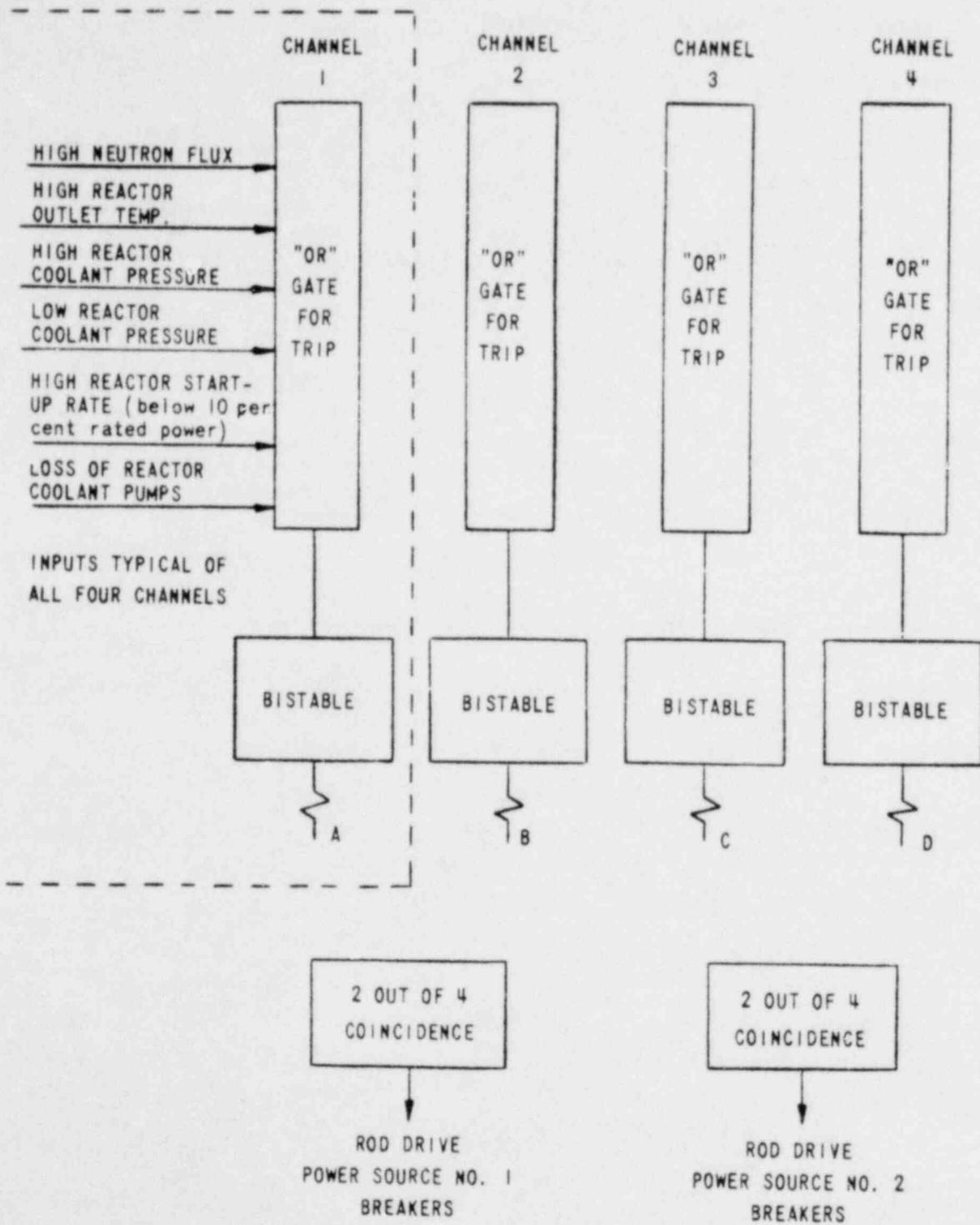
7.4.7 SAFETY FEATURES

The primary objectives in the control room layout are to provide the necessary controls to start, operate, and shut down the nuclear unit with sufficient information display and alarm monitoring to insure safe and reliable operation under normal and accident conditions. Special emphasis will be given to maintaining control integrity during accident conditions. The layout of the engineered safeguards section of the control board will be designed to minimize the time required for the operator to evaluate the system performance under accident conditions. Any deviations from predetermined conditions will be alarmed so that the operator may take corrective action using the controls provided on the control panel.

7.5 SYSTEMATIC, NONRANDOM, COMMON MODE INSTRUMENTATION FAILURES

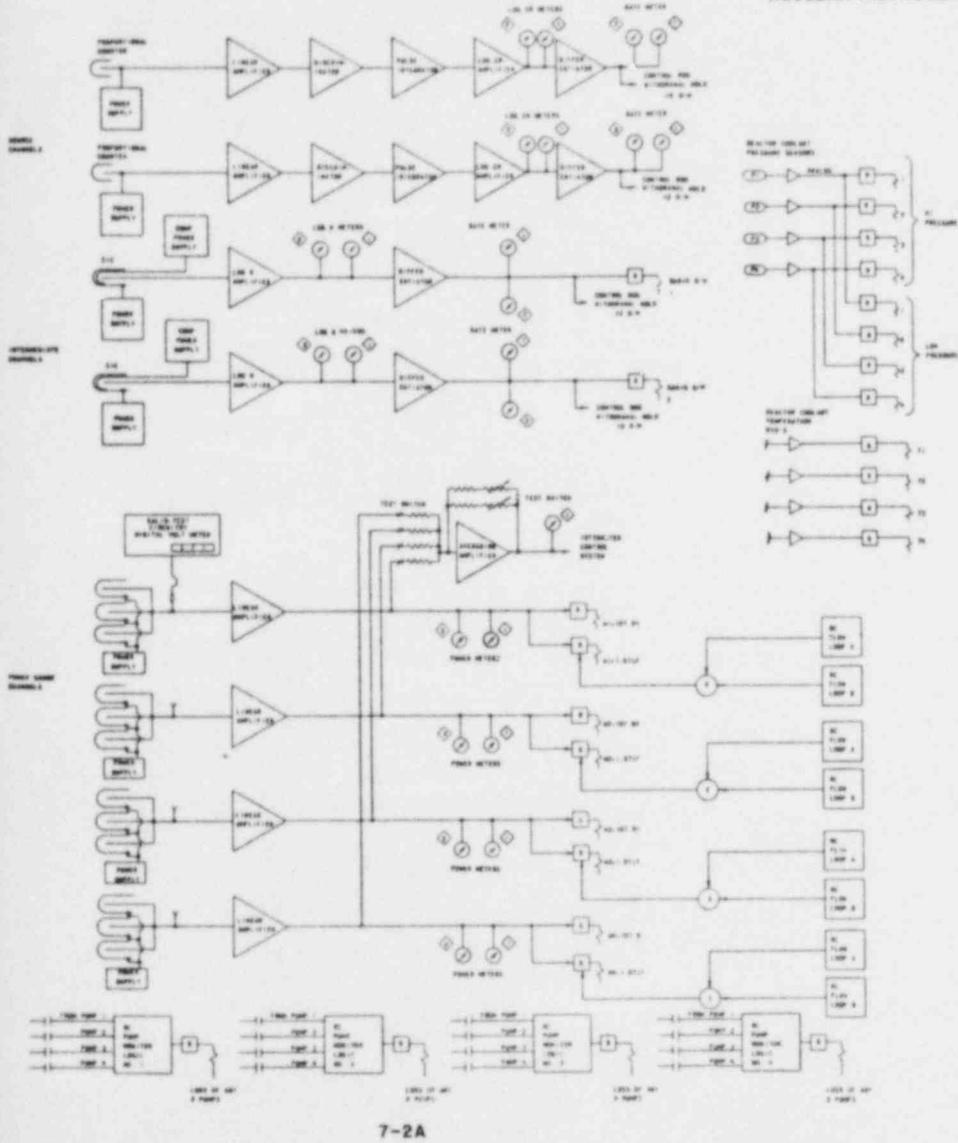
Arkansas Power & Light Company is following the progress of Babcock & Wilcox in this area. A Babcock & Wilcox Topical Report will be submitted on this subject in early 1970. We will incorporate this report or applicable sections thereof in the FSAR.

13

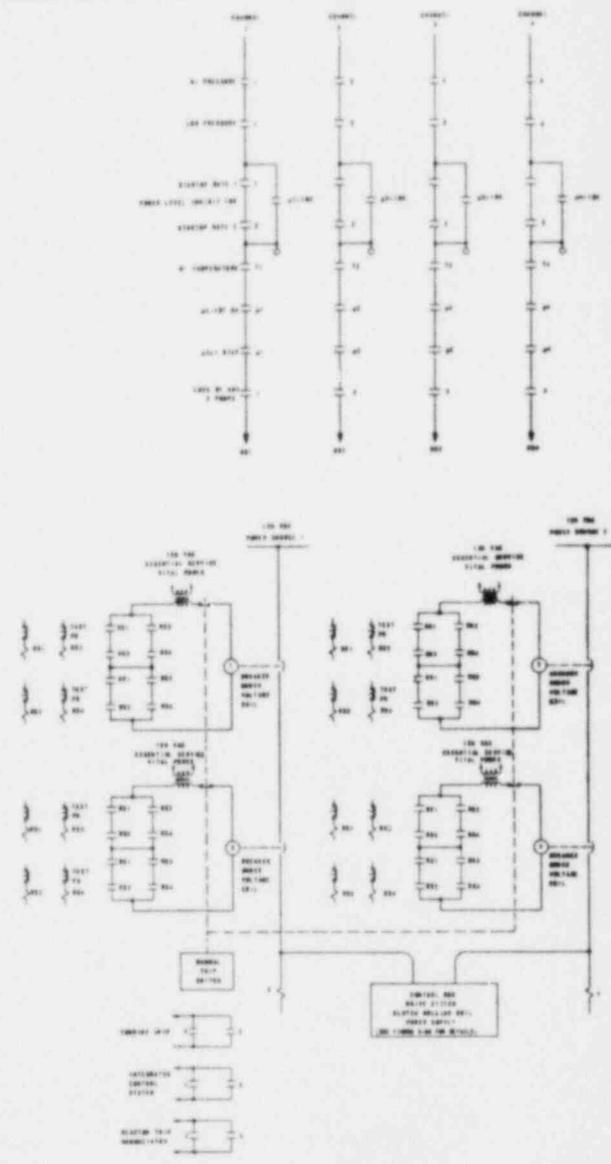


REACTOR PROTECTION SYSTEM BLOCK DIAGRAM

NUCLEAR INSTRUMENTATION AND REACTOR PROTECTION SYSTEM

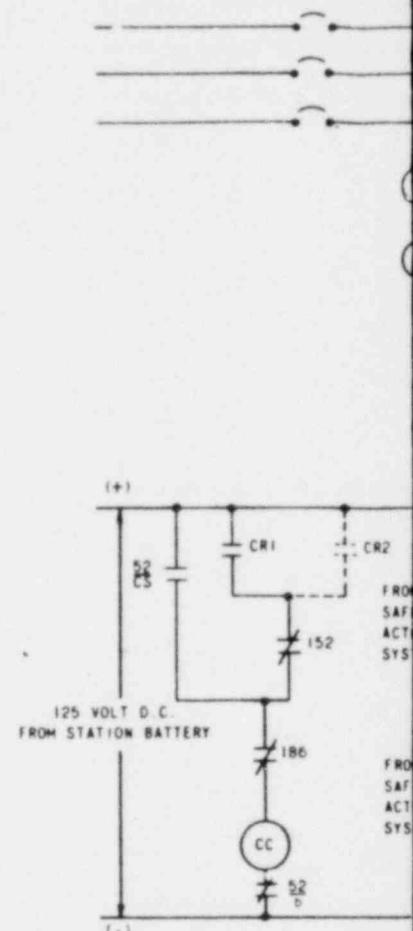
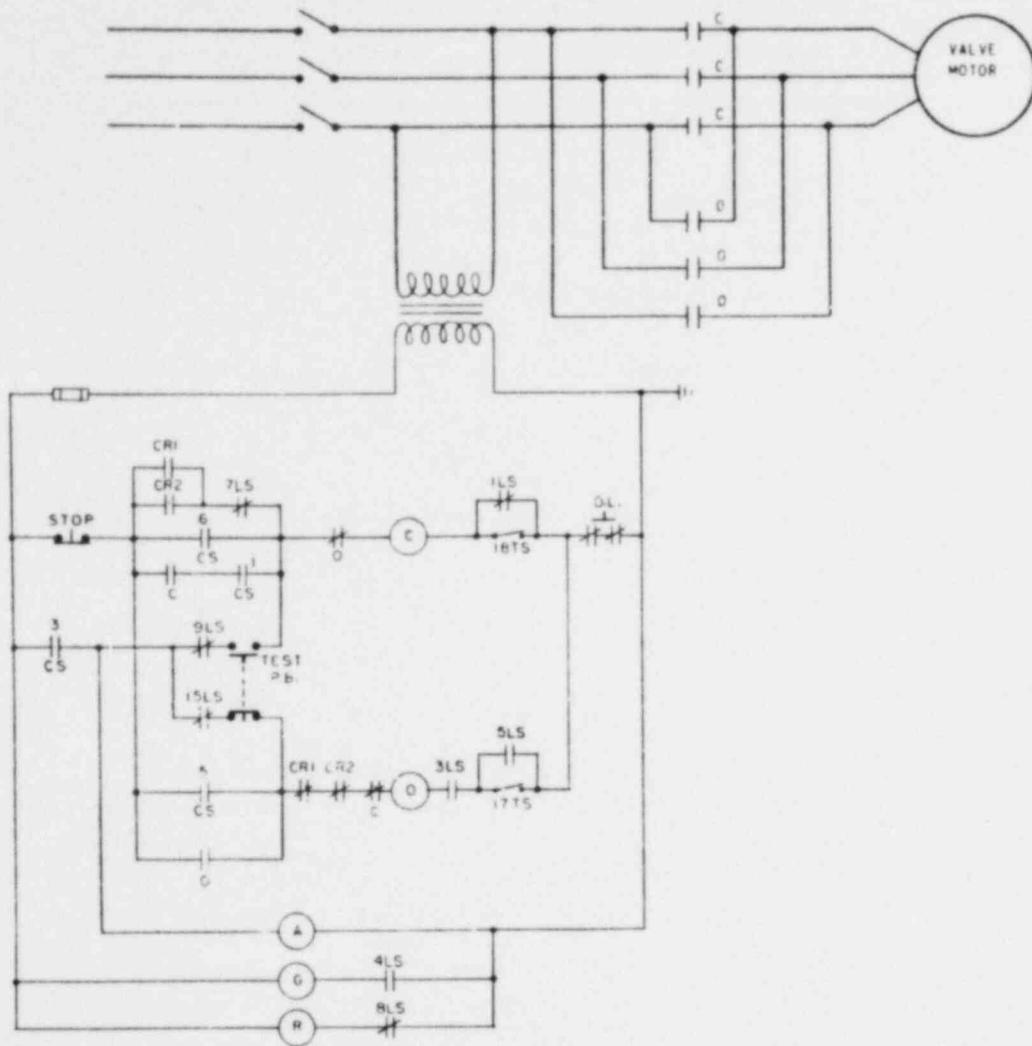


7-2A



TYPICAL REACTOR BUILDING ISOLATION VALVE

CONTROL CIRCUIT



CONTROL SWITCH, CS

CONTACTS	POSITION		
	OPEN	NORMAL	CLOSE
1			
2	} OPENS AT HANDLE OUT POS.		
3	} CLOSSES AT HANDLE OUT POS.		
4			
5	X		
6			X
7	X		
8			X

LATERAL STAGE, HANDLE PULLOUT FROM NORMAL POSITION ONLY

X INDICATES CONTACT IS CLOSED
 SPRING RETURN FROM CLOSE (OPENS) NORMAL
 NO SPRING RETURN FROM OUT POSITION ON
 LOCKED AGAINST TURNING WHEN HANDLE IS OUT POS.

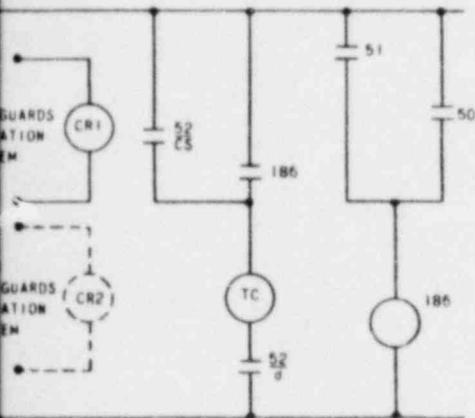
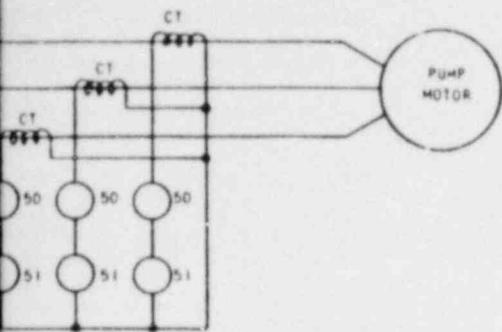


LIMIT SWITCH CONTACT DEVELOPMENT			
CONTACTS	FULLY OPEN	FULLY CLOSED	FUNCTION
1	█		IBTS BYPASS
2	█		
3		█	OPEN LIMIT
4		█	GREEN LIGHT
5		█	17TS BYPASS
6		█	
7		█	AUTO CLOSE
8		█	RED LIGHT
9		█	TEST CLOSE
10		█	
11		█	
12		█	
13		█	
14		█	
15		█	TEST RETURN
16		█	
17TS		█	OPEN TORQUE SWITCH - OPENS ON MECHANICAL OVERLOAD IN OPENING DIRECTION
18TS		█	CLOSING TORQUE SWITCH - OPENS ON MECHANICAL OVERLOAD IN CLOSING DIRECTION
█ DENOTES LIMIT SWITCHES CLOSED			

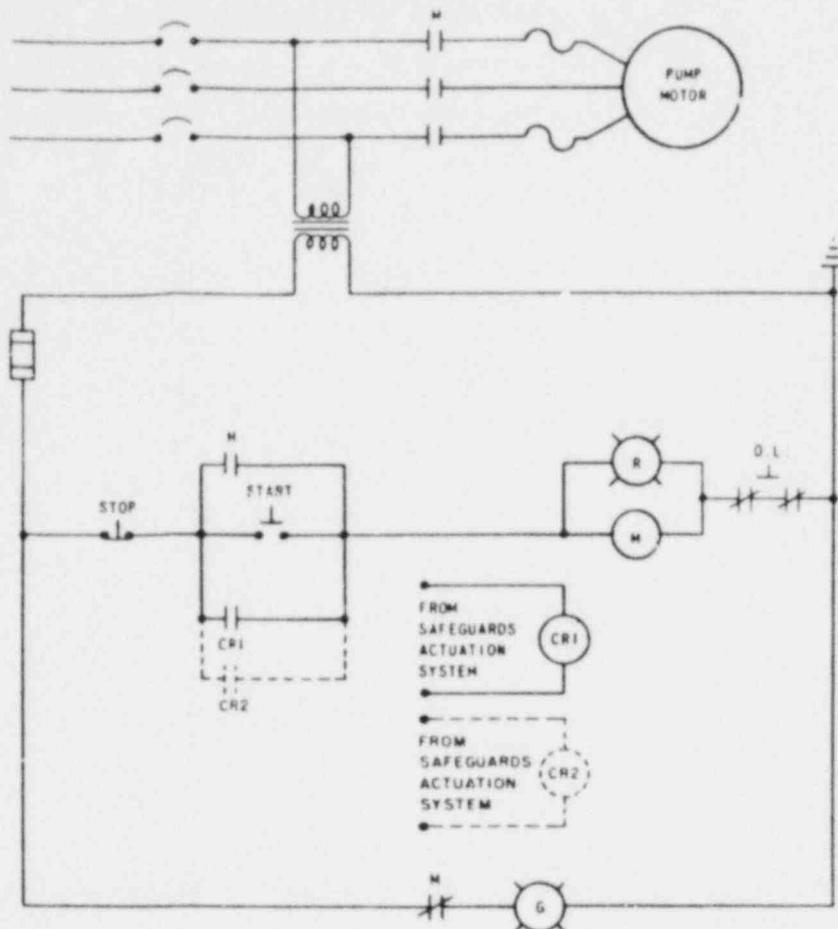
CONTACTS SHOWN IN DEENERGIZED POS.
 CR1 - CONTROL RELAY
 CR2 - CONTROL RELAY
 CC - CIRCUIT BREAKER CLOSING COIL
 TC - CIRCUIT BREAKER TRIP COIL
 50 - INSTANTANEOUS OVERCURRENT RE
 51 - TIME OVERCURRENT RELAY
 52 - CIRCUIT BREAKER CONTROL SWIT
 CS - CIRCUIT BREAKER AUXILIARY S
 186 - AUXILIARY TRIPPING RELAY

NOTE: CR2, SHOWN DOTTED, IS USED ONLY WHEN REDUNDANT CONTROL IS REQUIRED.

CIRCUIT FOR MAKEUP PUMP
(Circuit Breaker Control)



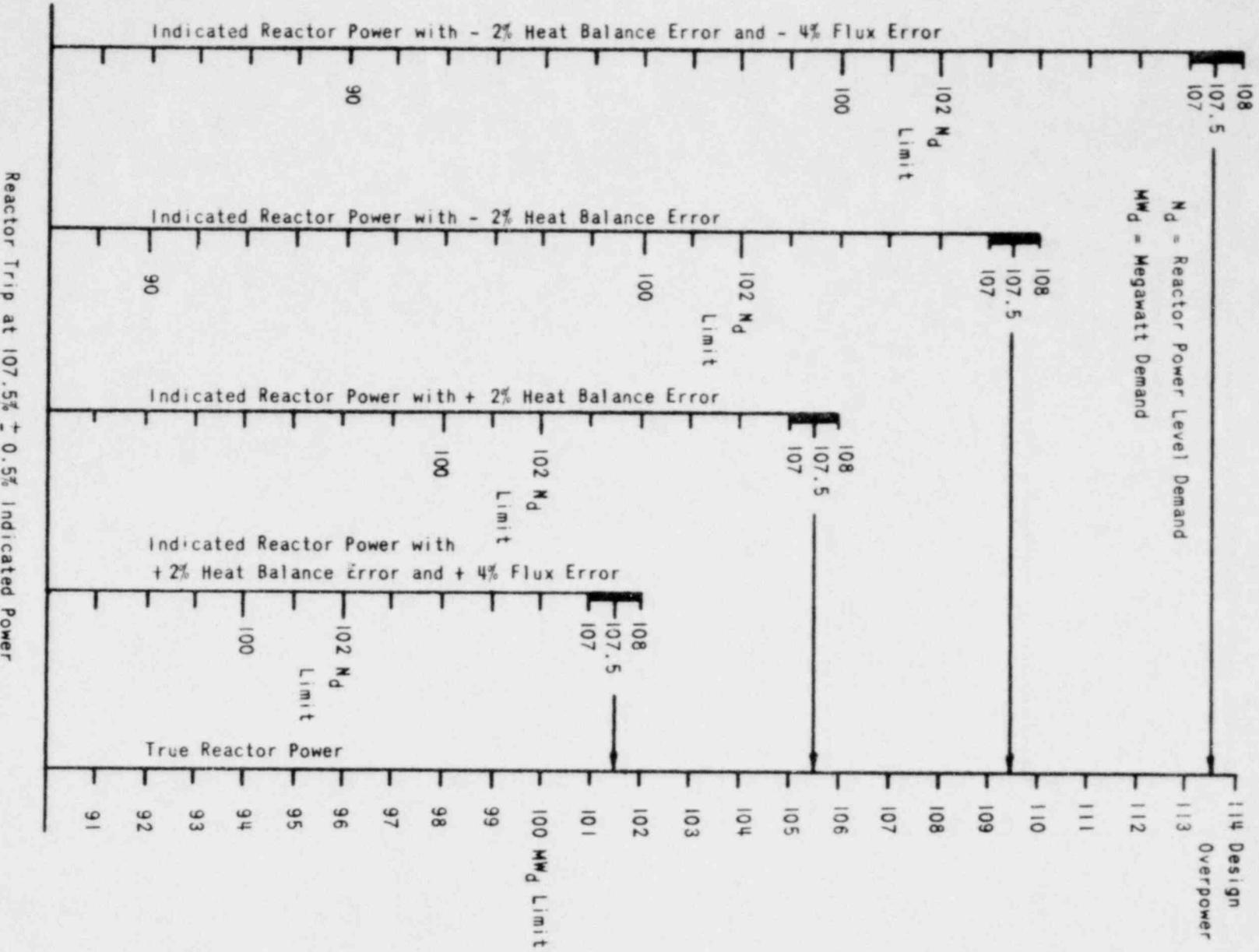
CONTROL CIRCUIT FOR REACTOR BUILDING SPRAY PUMP
(MOTOR STARTER CONTROL)



CONTACTS SHOWN IN DEENERGIZED POSITION
CR1 - CONTROL RELAY
M - MAIN CONTACTOR

TYPICAL CONTROL CIRCUITS
FOR ENGINEERED
SAFEGUARDS EQUIPMENT

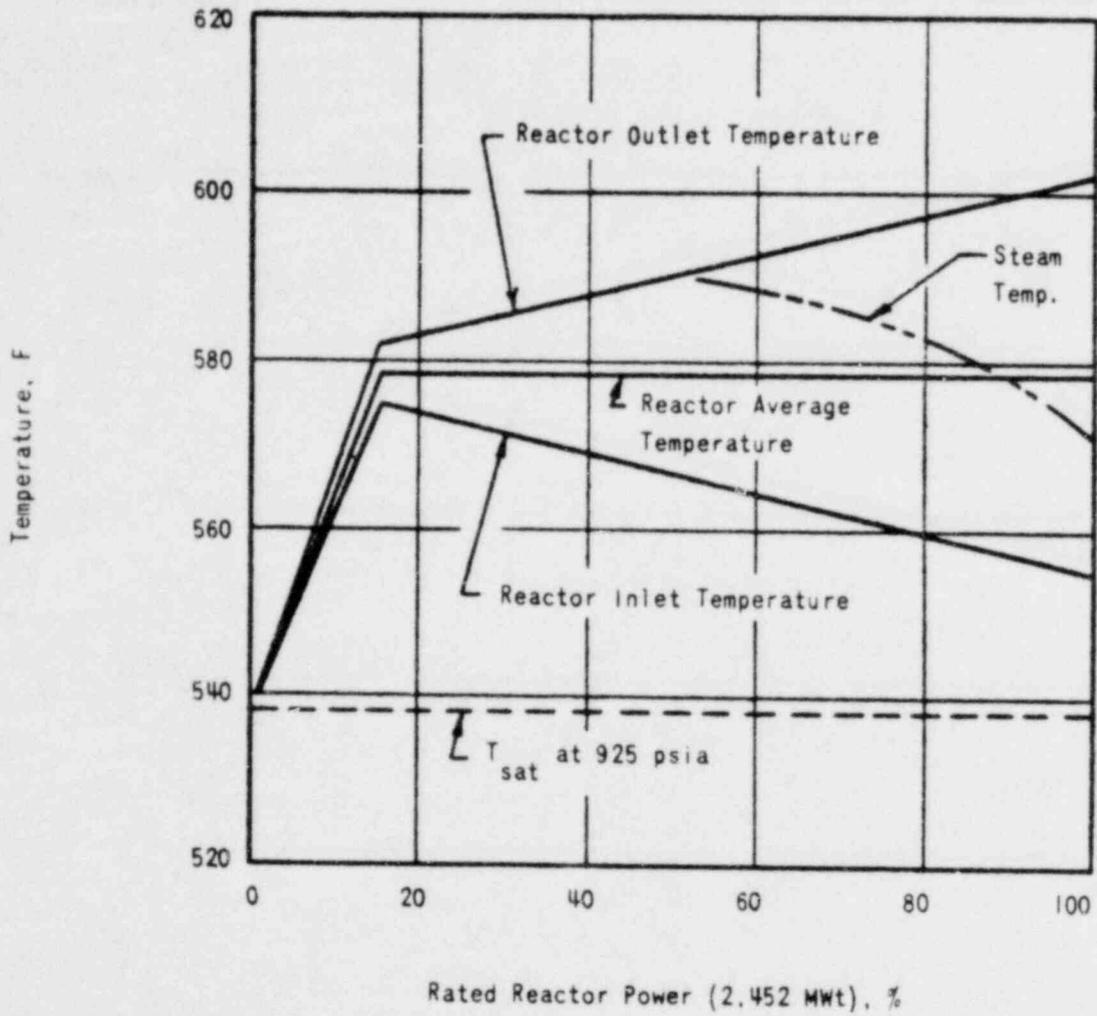
Figure 7-3



REACTOR POWER MEASUREMENT
 ERRORS AND CONTROL LIMITS

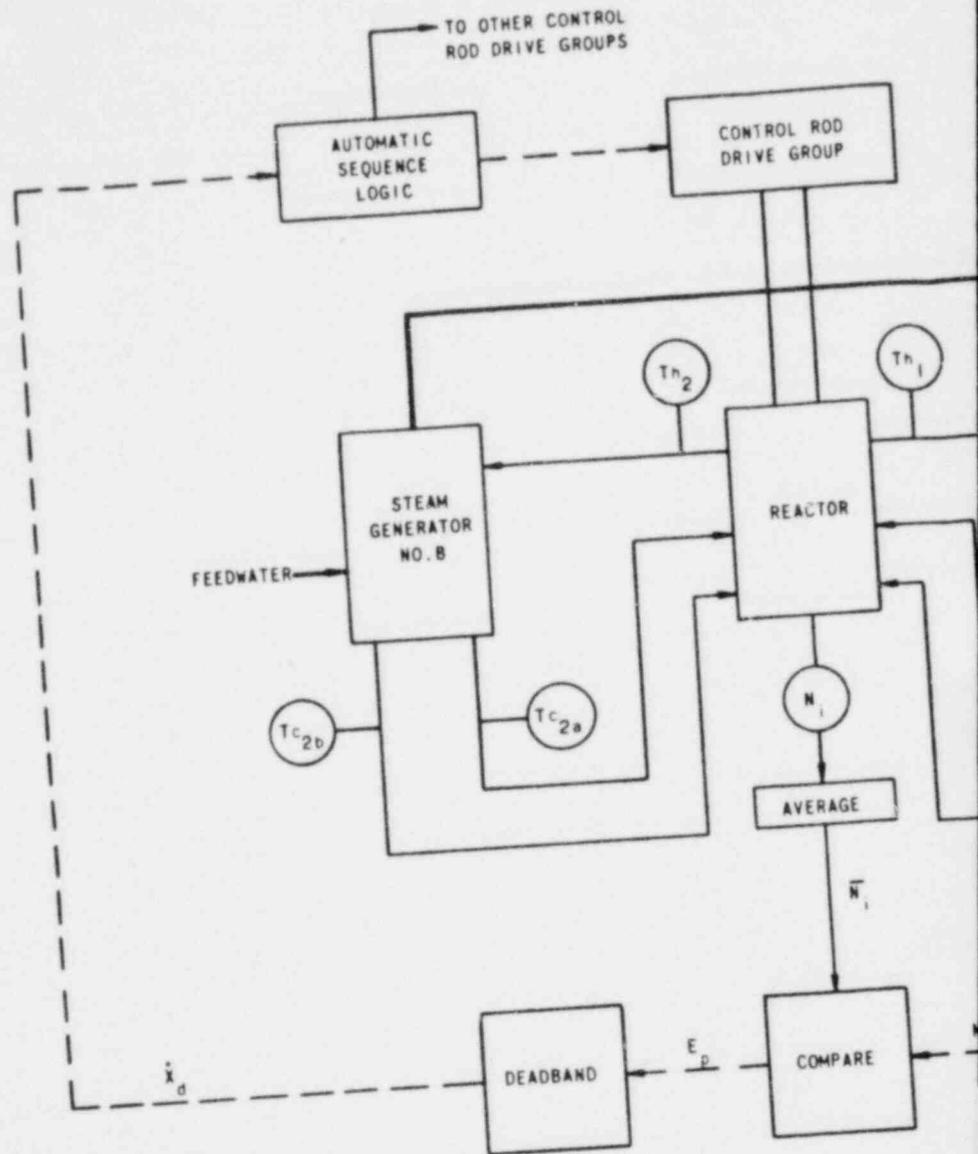
Figure 7-4

149



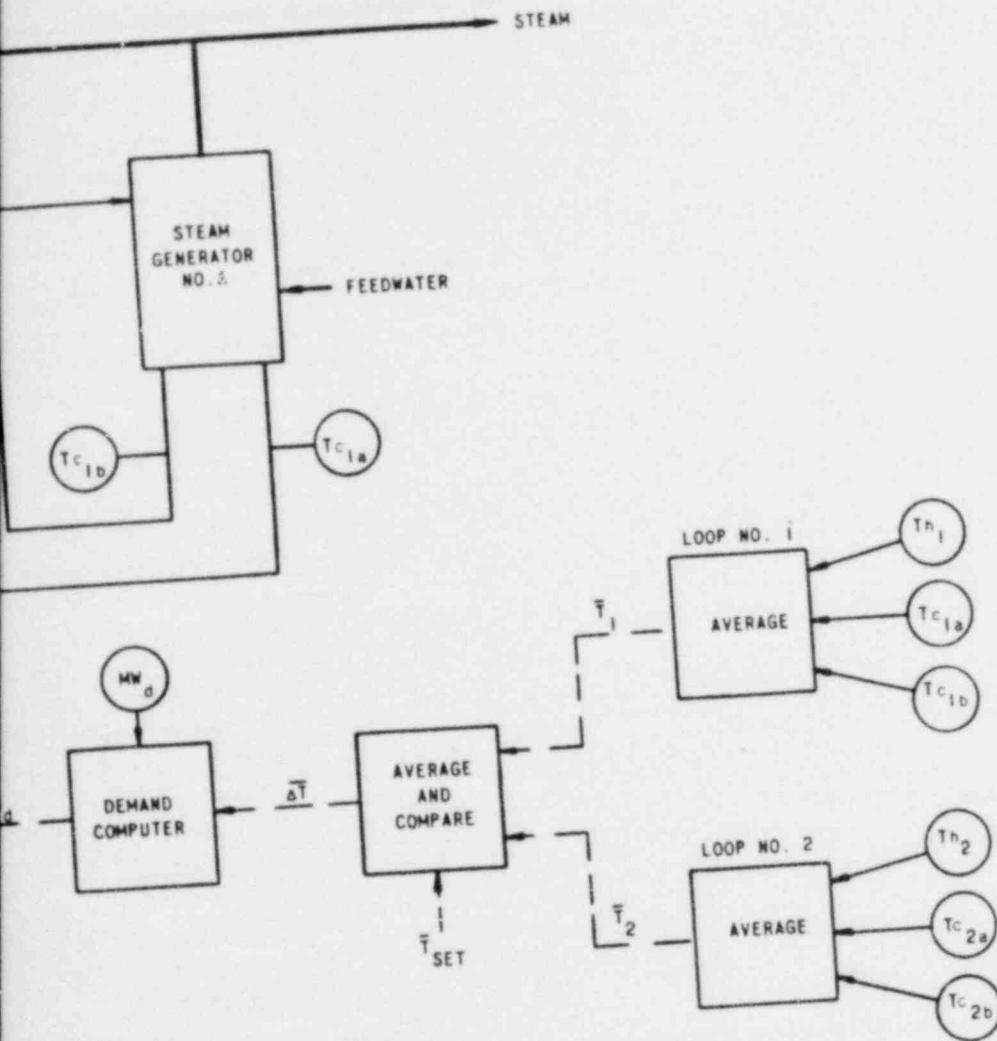
REACTOR AND STEAM TEMPERATURES
VERSUS REACTOR POWER

Figure 7-5



LEGEND

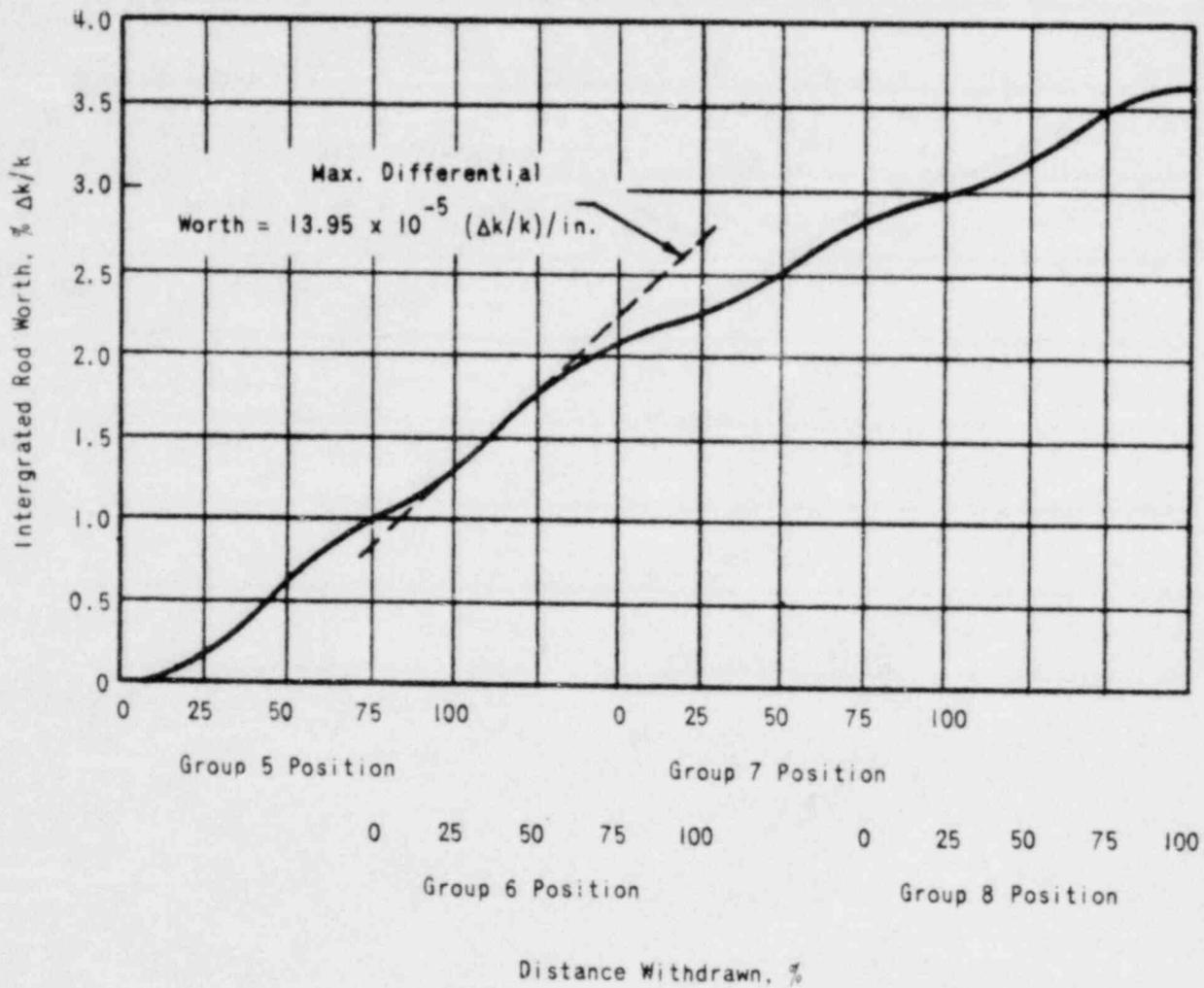
- REACTOR COOLANT SYSTEM
- STEAM SYSTEM
- - - CONTROL SYSTEM
- T_c - REACTOR COOLANT SYSTEM COLD
- T_h - REACTOR COOLANT SYSTEM HOT
- \bar{T} - AVERAGE REACTOR COOLANT SYSTEM
- ΔT - DEVIATION OF AVERAGE TEMPERATURE
- E_p - REACTOR POWER LEVEL ERROR, MW
- \dot{x}_d - ROD VELOCITY DEMAND
- MW_d - MEGAWATT DEMAND
- \bar{N}_1 - REACTOR POWER LEVEL
- N_d - REACTOR POWER LEVEL DEMAND



REACTOR TEMPERATURE
 REACTOR TEMPERATURE
 REACTOR TEMPERATURE
 REACTOR TEMPERATURE FROM SETPOINT
 $\bar{T}_d - N_i$

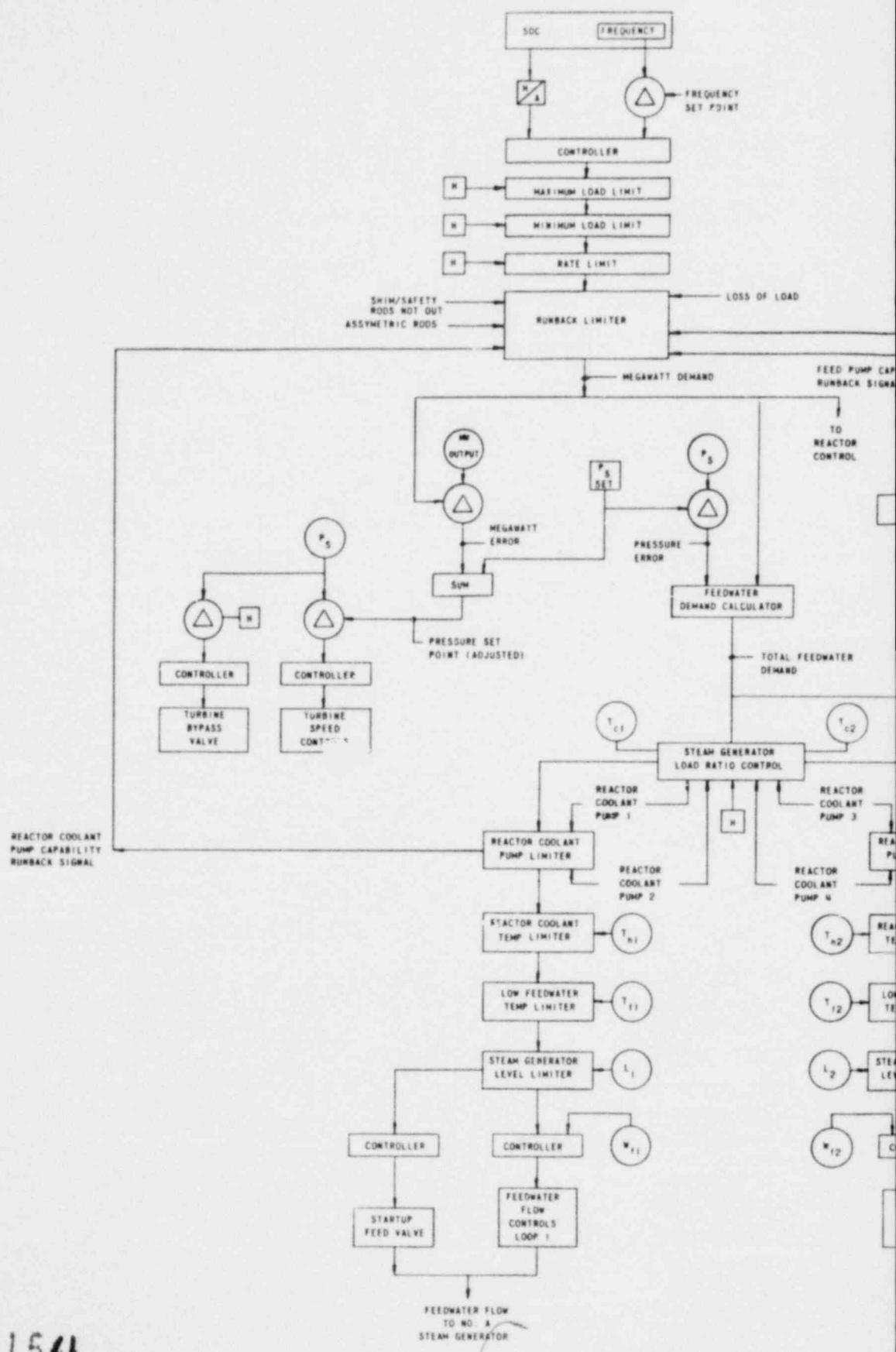
REACTOR CONTROL DIAGRAM - INTEGRATED CONTROL SYSTEM

Figure 7-6

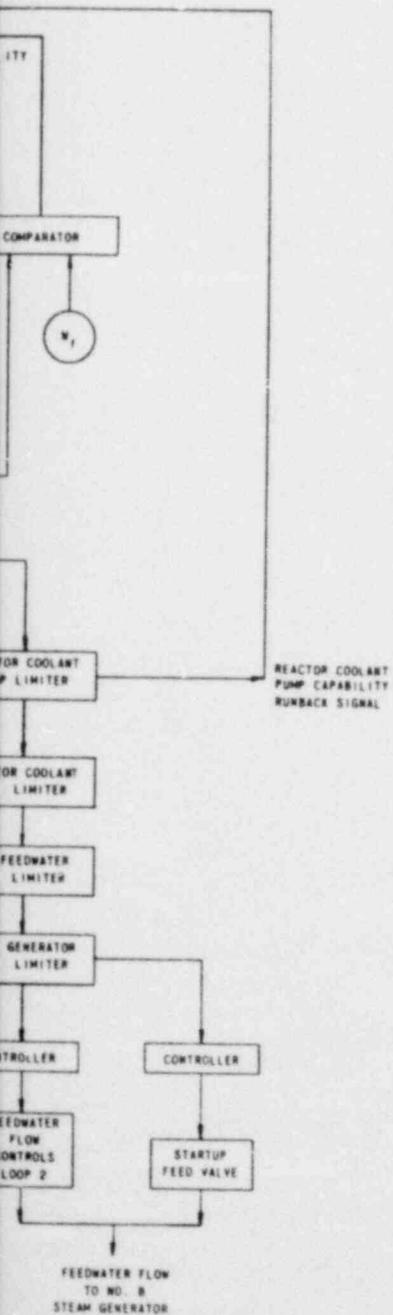


AUTOMATIC CONTROL ROD GROUPS -
TYPICAL WORTH CURVE VERSUS DISTANCE WITHDRAWN

Figure 7-7



154

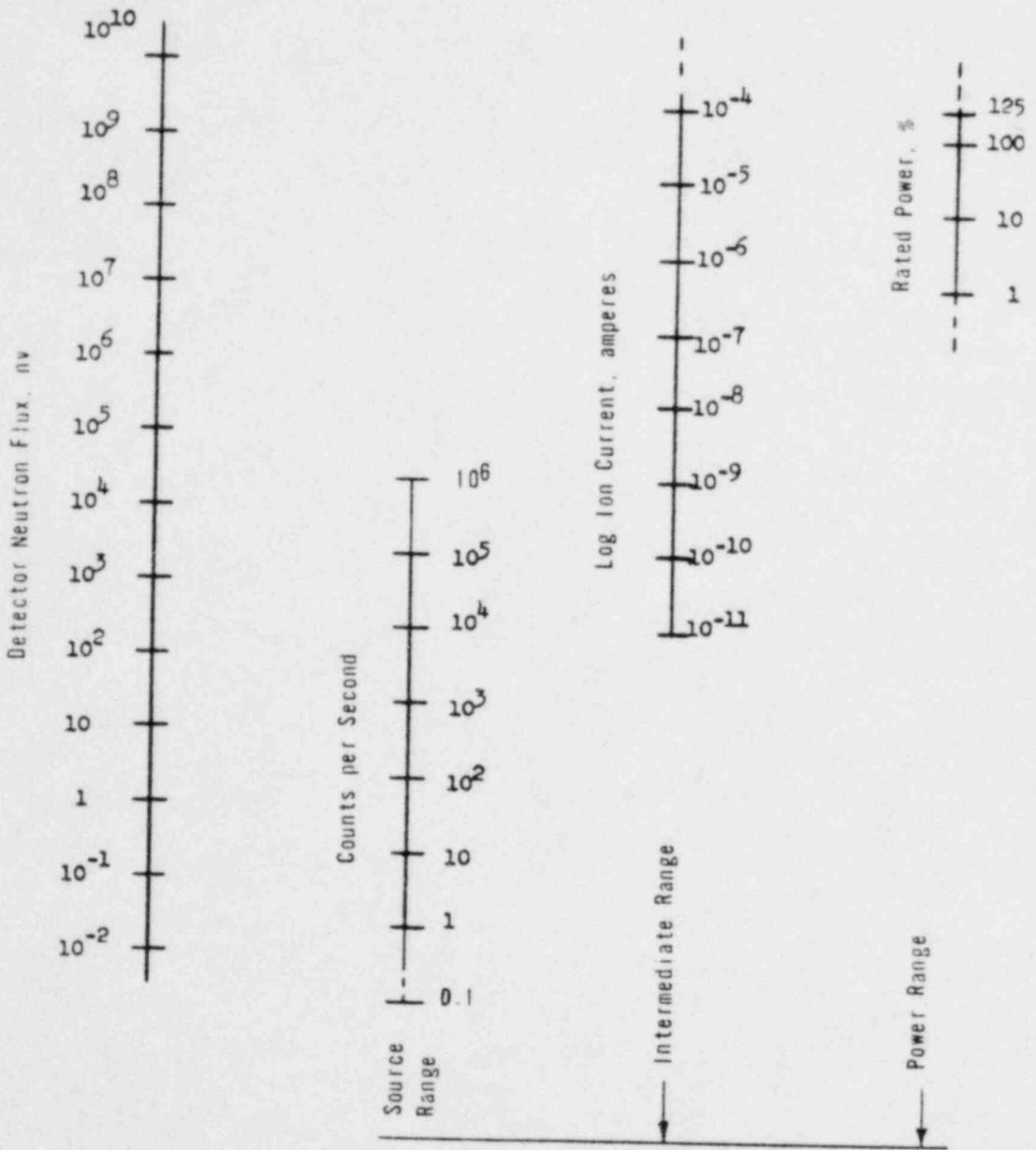


LEGEND:

- SDC - SYSTEM DISPATCH CENTER
- A - DIFFERENCE
- P - STEAM PRESSURE
- P_S SET - STEAM SET PRESSURE
- W₁ - FEEDWATER FLOW
- W - MANUAL SET POINT
- H/A - MANUAL/AUTOMATIC
- L - STEAM GENERATOR LEVEL
- T_{OUT} - REACTOR OUTLET TEMPERATURE
- T_{IN} - REACTOR INLET TEMPERATURE
- T_F - FEEDWATER TEMPERATURE

THE NUMBER SUBSCRIPTS REFER TO LOOP 1 AND LOOP 2

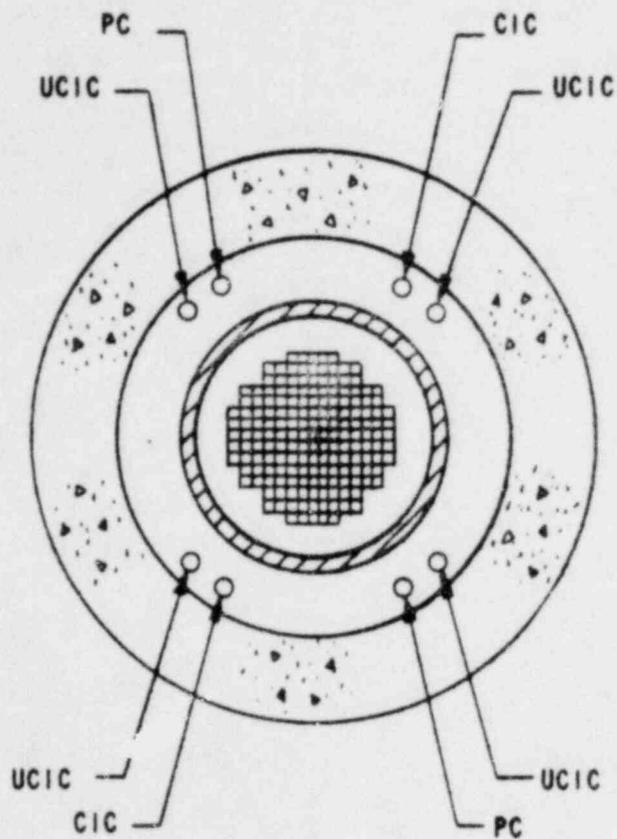
STEAM GENERATOR AND TURBINE CONTROL DIAGRAM -
INTEGRATED CONTROL SYSTEM



NUCLEAR INSTRUMENTATION FLUX RANGES

156

Figure 7-9
Supplement 3

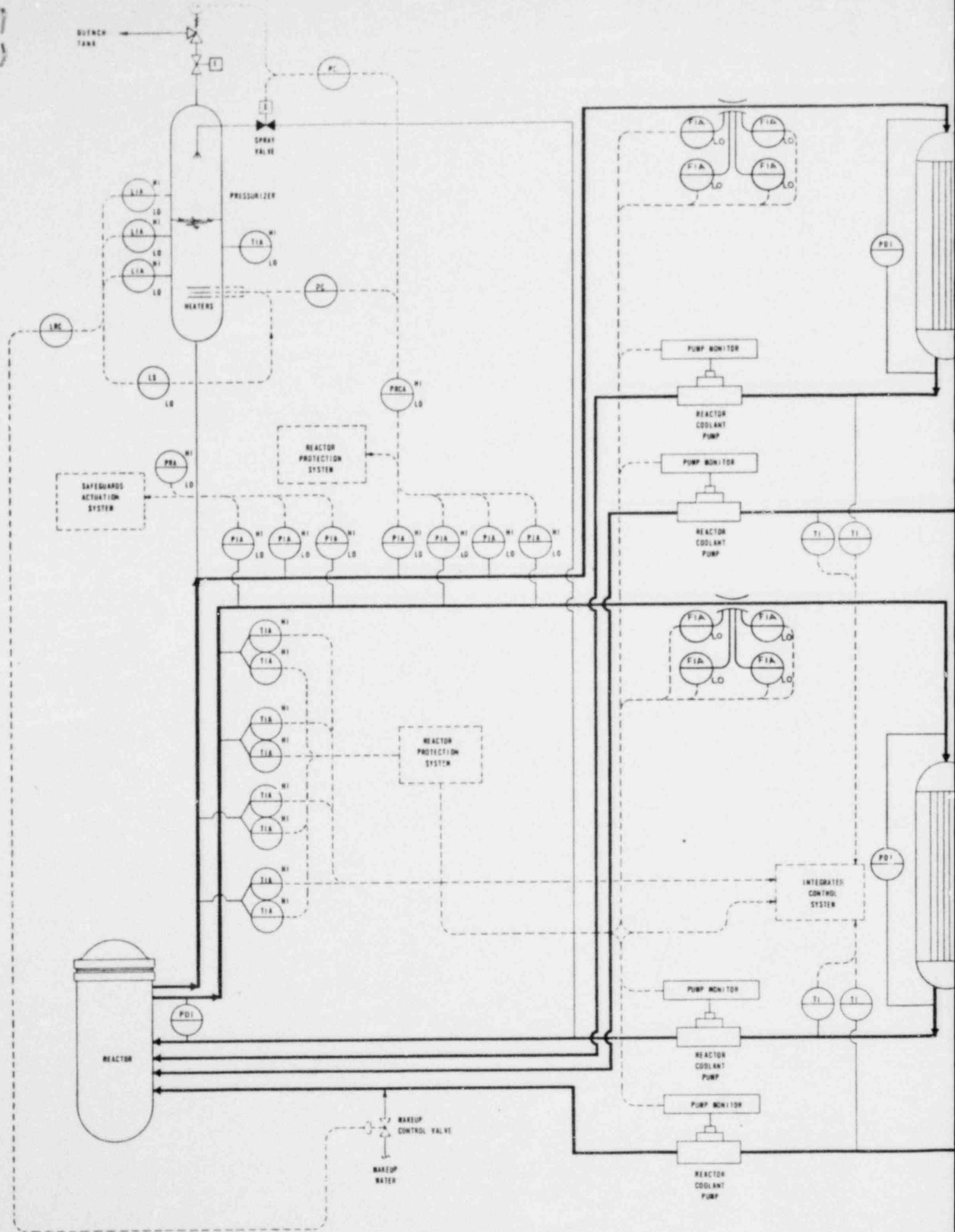


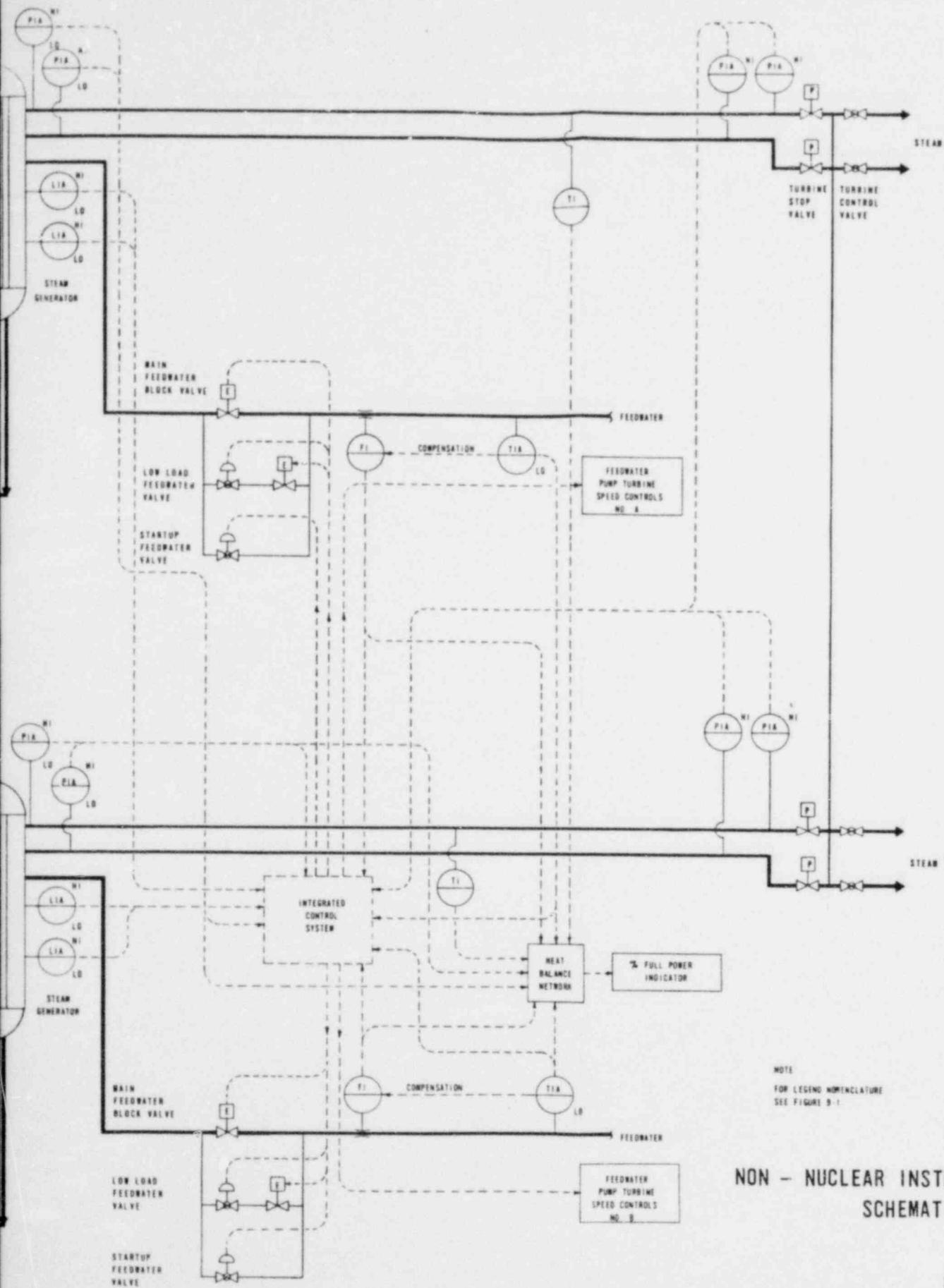
LEGEND

- PC - PROPORTIONAL COUNTER - SOURCE RANGE DETECTOR
- CIC - COMPENSATED ION CHAMBER - INTERMEDIATE RANGE DETECTOR
- UCIC - UNCOMPENSATED ION CHAMBER - POWER RANGE DETECTOR

NUCLEAR INSTRUMENTATION DETECTOR LOCATIONS

Figure 7-10

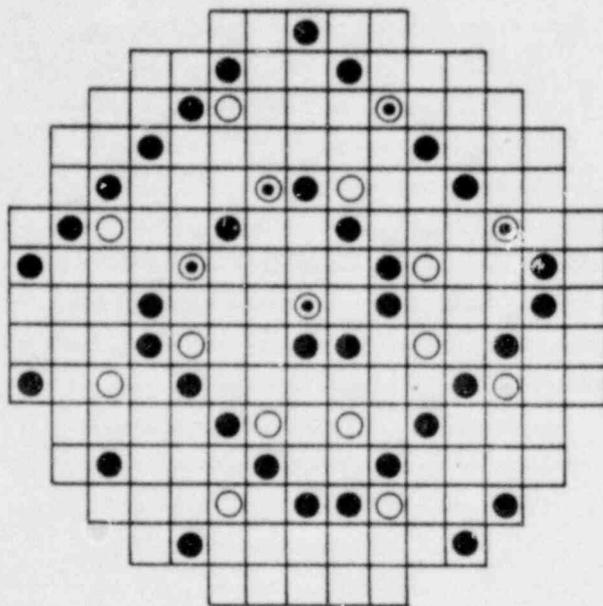




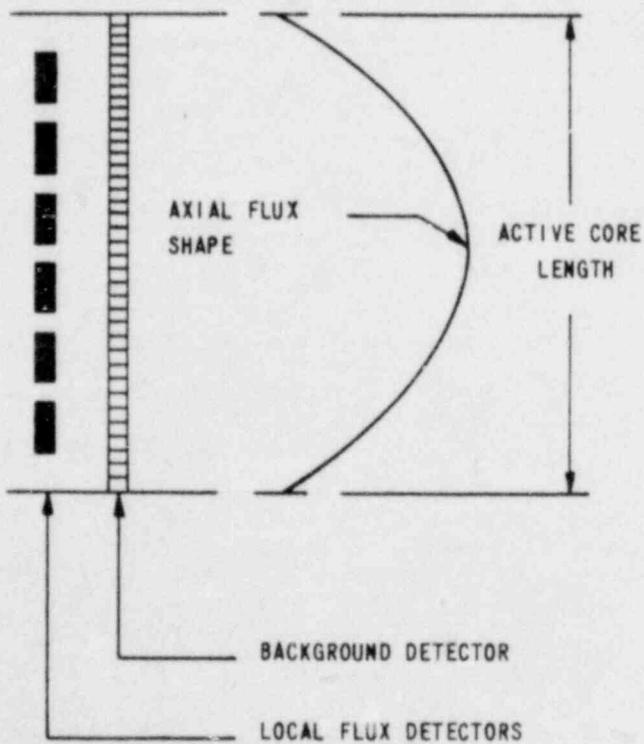
NOTE
FOR LEGEND NOMINCLATURE
SEE FIGURE 9-1

**NON - NUCLEAR INSTRUMENTATION
SCHEMATIC**

Figure 7-11
Supplement 3

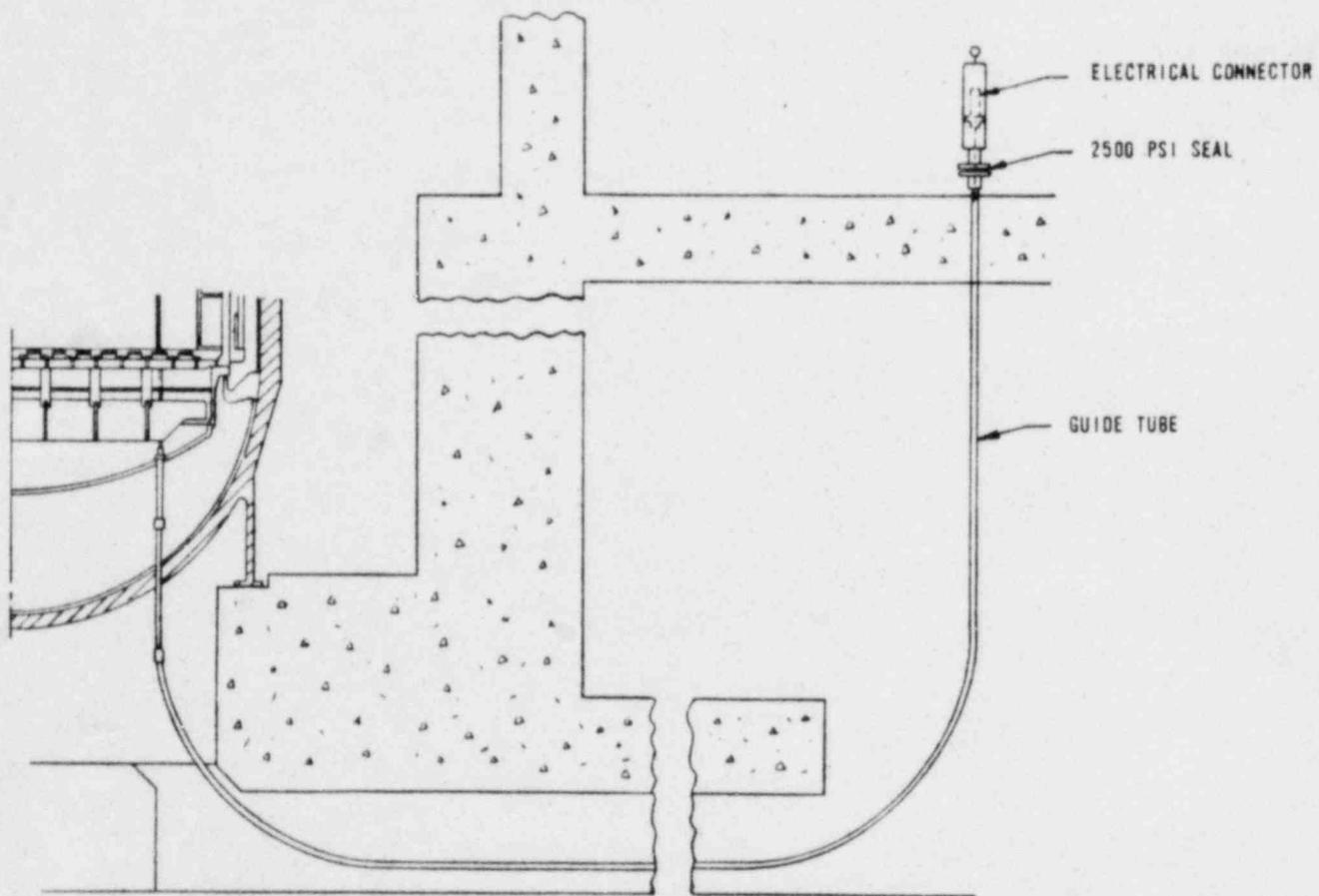


- Total core monitors based on 1/4 core symmetry.
- Symmetry monitors.
- ⊙ Combination total core and symmetry monitors.

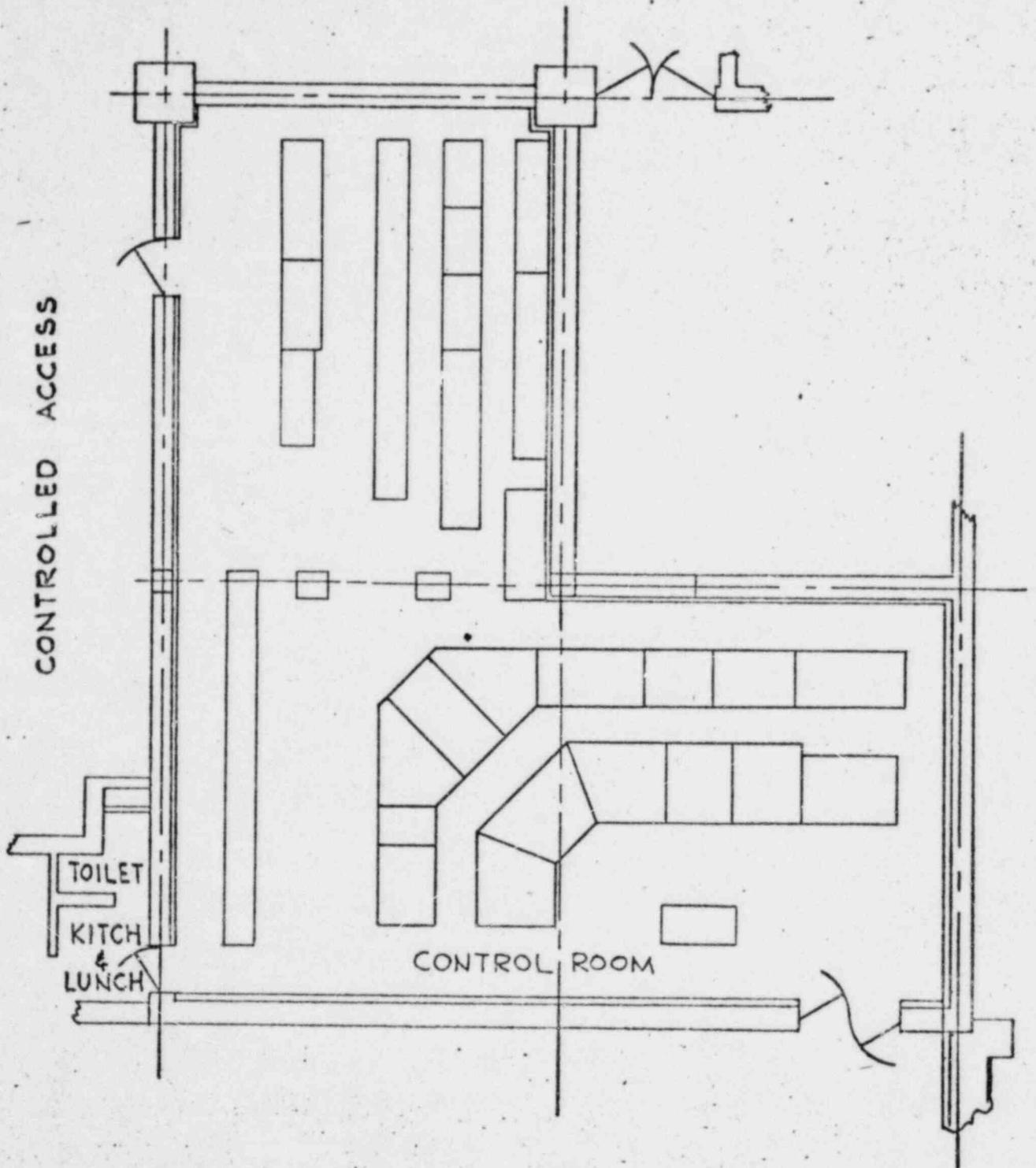


INCORE DETECTOR LOCATIONS

Figure 7-12
Supplement 3



TYPICAL ARRANGEMENT -
INCORE INSTRUMENTATION CHANNEL



ARKANSAS NUCLEAR ONE
CONTROL BOARD LAYOUT
FIGURE 7-14