7 INSTRUMENTATION AND CONTROL

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7 INSTRUMENTATION AND CONTROL

7.1 PROTECTIVE SYSTEMS

The protective systems, which consist of the Reactor Protective System and the Engineered Safeguards Protective System, perform the most important control and safety functions. The protective 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 Protective 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 Engineered Safeguards Protective 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 Protective System automatically trips the reactor to protect the reactor core under the following conditions:

a. The reactor power, as measured by neutron flux, reaches the limit set by the reactor coolant flow. The reactor coolant flow is determined by the number of operating reactor coolant pumps.

b. The reactor outlet temperature reaches an established maximum limit.

c. The reactor pressure reaches an established minimum limit.

The Reactor Protective System automatically trips the reactor to protect the reactor coolant system under the following condition:

a. The reactor pressure reaches an established maximum limit.

The Engineered Safeguards Protective System automatically performs the following vital functions:

a. Commands operation of the core emergency injection systems upon detection of abnormally low reactor coolant pressure~~, or low pressurizer level.~~ These conditions are indicative of a loss-of-coolant accident.

b. Commands operation of the reactor building cooling systems upon detection of an abnormally high reactor building pressure.

c. Commands closing of the reactor building isolation valves upon detection of an abnormally high reactor building pressure.

7.1.1.1.1 Nonvital Functions

The reactor protective system provides an anticipatory reactor trip when the reactor start-up rate reaches specified limits.

7.1.1.2 Principles of Design

The major design criteria are summarized as follows:

7.1.1.2.1 Single Failure

a. No single component failure shall prevent the protective systems from fulfilling their protective functions when action is required.

b. No single component failure shall initiate unnecessary protective system action, provided implementation does not conflict with the above criterion.

7.1.1.2.2 Redundancy

All protective system functions shall be implemented by means of redundant sensors, instrument strings, logic and action devices which combine to form the protective channels.

7.1.1.2.3 Independence

Redundant protective channels and their associated elements shall be electrically independent and packaged to provide physical separation.

Separate detectors and instrument strings are not, in general, employed for protective system functions and regulation or control. Sharing instrumentation for protection and control functions is accomplished within the framework of the stated criteria by the employment of isolation techniques to the multiple outputs of various instrument strings. This may be stated as a corollary to the design criteria, ie, a direct short, open circuit, ground fault or bridging of any two points at the output terminals of an instrument string having multiple outputs shall not result in a significant disturbance within more than one output.

7.1.1.2.4 Loss of Power

a. A loss of power in the reactor protective system shall cause the affected channel to trip.

b. Availability of power to the engineered safeguards protective system shall be continuously indicated. The loss of ac instrument power, ie, vital bus power, to the instrument strings and bistables will initiate a trip in the affected channels. System actuation requires control power from only one of the two engineered safeguards dc control power busses. Equipment is divided between the redundant engineered safeguards channels in such a way that the loss of one of the dc power busses does not inhibit the system's intended safeguards functions.

7.1.1.2.5 Manual Trip

Each protective 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 Protective System shall initiate a trip of the channel involved when modules, equipment, or subassemblies are removed. Engineered Safeguards Protective 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 protective systems to provide for the following:

a. Preoperational testing to give assurance that the protective 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 protective systems are those specified under vital functions together with interlocking functions.

The functional requirements of the Reactor Protective System are to trip the reactor when:

a. The reactor power, as measured by neutron flux, reaches an allowable limit set by the number of operating reactor coolant pumps.

b. The reactor outlet temperature reaches a preset maximum limit.

c. The reactor coolant pressure reaches a preset maximum limit.

d. The reactor coolant pressure reaches a preset minimum limit.

e. The reactor startup rate reaches a maximum limit while operating below a preset power level.

Interlocking functions of the Reactor Protective 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 Engineered Safeguards Protective System are to:

a. Start operation of the high pressure injection system upon detection of a low reactor coolant system pressure.

b. Start operation of the low pressure injection system upon detection of a very low reactor coolant system pressure. ~~or a low pressurizer level.~~

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 ~~120~~ 118 F. The Reactor Protective 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, and will function with lesser 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 Engineered Safeguards Protective 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 and control room is designed for continuous operation in an ambient of 140 F and 90 per cent relative humidity. The control room ambient will be maintained at the personnel comfort level; however, protective equipment in the control room will operate within design tolerance up to a temperature of 120 F.

7.1.2 SYSTEM DESIGN

7.1.2.1 System Description - Reactor Protective System

Figure 7-1 is a block diagram of the Reactor Protective System. The system consists of four identical protective channels, each terminating in a noninverting 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, de-energizing the reactor trip relay. Thus, for trip signals each channel becomes an OR gate.

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

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 Protective System (7-2B) and the Engineered Safeguards Protective System (7-2C). Figure 7-2B shows the circuit breakers controlling input power to the control rod drives and the manner in which the reactor trip relays trip these circuit breakers.

Reactor trip is accomplished by interrupting all three-phase input power to the control rod drive assemblies. Each control rod drive power supply receives its input power through two circuit breakers in series so that opening of either interrupts that source of power. As shown in Figure 3-59, control rod drive power supplies operate in parallel so that complementary supplies must be de-energized for the control rods to trip. Circuit breakers No. 1 and No. 2 control primary power to one set of power supplies and circuit breakers No. 3 and No. 4 control power to the other complementary set. Thus, reactor trip is accomplished by tripping one circuit breaker in each three-phase input.

The control rod drive 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 line side of the circuit breaker through a transformer.

Referring to circuit breaker No. 1, the undervoltage coil is 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, RS3 and RS2 or RS3 and RS4 become de-energized, circuit breaker No. 1 undervoltage coil will be de-energized and the circuit breaker will open. The trip relays which will cause circuit breaker No. 2 to open are RS1 and RS3, RS1 and RS4, RS2 and RS3 or RS2 and RS4. Thus any 2-out-of-4 trip relays will cause either circuit breaker No. 1 or circuit breaker No. 2 to open.

The 2-out-of-4 logic to trip circuit breaker No. 3 and circuit breaker No. 4 is identical.

The trip circuits and devices are redundant and independent. Each breaker is independent of each other breaker such that 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 means of 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 (ΣF), a "number of pump motors in operation" signal (P_n), and three isolated reactor power level signals (Φ).

The power/flow monitor continuously compares the ratio of the reactor neutron power to the reactor coolant flow. Should the reactor power as measured by the linear power range channels exceed 1.07 times the total reactor coolant flow, a reactor trip is initiated. All measurements are in terms of per cent full flow or full power. When the reactor is operating above a predetermined neutron power, X% FP, a reactor trip is initiated immediately upon the loss of a single pump. Below this power level a reactor trip is initiated when the reactor power to reactor coolant flow ratio exceeds 1.07. Thus below a predetermined reactor power there is opportunity for the control system to reduce the reactor power to an acceptable level without a reactor trip.

There are four combinations of logic functions within the power/flow monitor which may lead to a reactor trip; refer to Figure 7-2.

The purpose of (A1) is to compare the total reactor coolant flow against the number of operating pump motors, P_n. Normally, the loss of a pump will cause an instantaneous decrease in P_n with the flow signal lagging. Should the reverse ever occur, as might be indicative of a lost pump rotor, (A1) will initiate a reactor trip if the reactor power is greater than a predetermined value, X% FP (E1).

Below X% FP, the flux-flow comparator (D1) will trip the reactor when the flux to flow ratio exceeds 1.07.

The (B1) comparator compares the reactor coolant flow against the number of operating pumps to determine that not more than one pump has been coincidentally lost. Should (B1) detect the coincident loss of more than one pump, the logic is required to determine that the ratios of reactor power to operating pumps (C1) and reactor power to reactor coolant flow (D1) are both less than 1.07. If either of these conditions is not satisfied, a reactor trip results.

The (C1) comparator continuously compares the number of operating pump motors against the reactor power. A reactor trip is immediately initiated upon loss of a pump when the reactor power is above a predetermined value, X% FP (E1). Below this power level, (C1) will not actuate a trip unless the (B1) comparator detects the loss of more than one pump.

7.1.2.2 Description - Engineered Safeguards Protective System

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

a. Each protective action is initiated by two channels of 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 coincident 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.

Figure 7-3 shows typical control circuits for equipment serving safeguards functions. Each circuit provides for normal start-stop control by the 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 low pressure injection system pump is typical of the controller of a large pump started by switchgear. There are three low pressure injection system pumps, of which two are equipped with single control relays, CR1, powered from separate engineered safeguards channels. The third pump is equipped with two control relays, CR1 and CR2, each of which is powered from separate engineered safeguards channels. Energizing the control relays through their associated engineered safeguards channel energizes the pump circuit breaker closing coil and starts the pump.

The control circuit for a 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 engineered safeguards channel to CR1. If the valve is employed with a passive redundant check valve, then it is controlled by two engineered safeguards channels with CR1 and CR2 connected in an OR configuration.

The control relays, when energized by their associated engineered safeguards channel, 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 the control circuit 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.

The control of the Reactor Building spray pumps is by means of single control relays in each pump controller. 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. Each channel consists of a 2-out-of-3 coincidence network which is made up of pressure switches designed to sense directly the Reactor Building pressure. Each channel is powered from a separate engineered safeguard control power bus.

7.1.2.3 Design Features

7.1.2.3.1 Redundancy

The reactor protective 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 start-up rate channels and two intermediate range channels, each with its own independent sensor.

The engineered safeguards protective system is also redundant for all vital inputs and functions. Each input variable is measured three times 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 protective 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 engineered safeguards protective 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 protective system initiates trip action upon loss of power. All bistables operate in a normally energized state and go to a de-energized state to initiate action. Loss of power thus automatically forces the bistables into the tripped state. Figure 7-2B shows the system in a de-energized state.

The engineered safeguards protective system instrumentation strings terminate in bistable trip elements similar to those in the reactor protective system. Loss of instrument power, up to and including the bistables, forces the bistables into the tripped state. Each redundant channel of engineered safeguards protective system coincident logic and command circuits extending to the engineered safeguards equipment controllers is powered either from battery backed engineered safeguards control power bus No. 1 or bus No. 2. Engineered safeguards equipment such as pump and motor operators and their starting contactors are powered from the appropriate redundant ac power busses described in 8.2.2. Safeguards action is initiated by energizing command circuits rather than by de-energizing. Redundancy of power supply is discussed in 8.2.2.

7.1.2.3.4 Manual System Trip

The manual actuating devices in the protective systems are independent of the automatic trip circuitry, and are not subject to failures which make the

automatic circuitry inoperable. The manual trip devices are independent control switches for each power controller.

7.1.2.3.5 Equipment Removal

The removal of modules or subassemblies from vital sections of the Reactor Protective 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 protective 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 Engineered Safeguards Protective System provides for servicing without affecting the integrity of the redundant channels.

7.1.2.3.6 Testing

The protective 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 manually initiate trip signals 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 also 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 protective 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 between two 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 the true 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 protective system is the vital busses described in 8.2.2.6. The source of power for the measuring elements in the engineered safeguards protective system is also from the vital busses. Each redundant channel of engineered safeguards protective system coincident logic and command circuits extending to the engineered safeguards equipment controllers is powered either from battery backed engineered safeguards control power bus No. 1 or bus No. 2. Engineered safeguards equipment such as pump and motor operators and their starting contactors are powered from the appropriate redundant ac power busses described in 8.2.2.

7.1.2.3.9 Reliability

Design criteria for the reactor protective system and the engineered safeguards protective 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.

7.1.2.4 Summary of Protective Actions

The abnormal conditions which initiate a reactor trip are listed below:

Trip Variable	No. of Sensors	Normal Range	Trip Value or Condition for Trip
Neutron Flux	4	0-100%	107.5% of full power
Neutron Flux/Reactor Coolant Flow	4 Flux 16 Reactor Coolant Pump Monitors 2 Flow Tubes	1 to 4 pumps	(1) Number of operating coolant pump motors exceeds total coolant flow and reactor power exceeds predetermined level. (2) Ratio of reactor power to total reactor coolant flow exceeds 1.07.

Trip Variable	No. of Sensors	Normal Range	Trip Value or Condition for Trip
			(3) More than one reactor coolant pump motor is lost and reactor power exceeds remaining pump capability by more than 107%. (4) Reactor power exceeds number of operating pump motors and the reactor power exceeds predetermined level.
Startup Rate	2	0-2 Decades/min	5 Decades/min
Reactor Coolant Pressure	4	2,100-2,300 psig	2,350 psig 2,050 psig
Reactor Outlet Temperature	4	520-603 F	610 F

The reactor trip functions of the power/flow monitor logic are summarized as follows:

Trip Variable	No. of Sensors
Neutron Flux = $\emptyset$	4
Reactor Coolant Flow = ΣF	4
No. of Operating Pumps = P_n	16

Reactor Trip

(a) $(\emptyset > 1.07P_n)$ and $(\emptyset > X\%)$*

(b) $(\emptyset > 1.07P_n)$ and $(\Sigma F - P_n)$ = Loss of more than one pump

(c) $(\emptyset > X\%)$* and $(P_n - \Sigma F)$ = Abnormal relation of $P_n > \Sigma F$

(d) $(\emptyset > 1.07 \Sigma F)$

* Predetermined neutron power level to be specified during detail design.

Actions initiated by the engineered safeguards protective system:

Action	Trip Condition	Normal Value	Trip Point
High Pressure Injection	Reactor Coolant Pressure	2,100-2,300 psig	1,800 psig
Low Pressure Injection	Very Low Reactor Pressure	2,100-2,300 psig	200 psig

Action	Trip Condition	Normal Value	Trip Point
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 which have been established from the analyses described in Section 14. The setpoint for each input, which must initiate a trip of the reactor protective system, has been established at a level which will insure that control rods are inserted in sufficient time to protect the reactor core. Likewise, the setpoints for parameters initiating a trip of the engineered safeguards protective system are established at levels which 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 travel times and pump starting times have been considered in establishing the margin between the trip setpoints and the safety limits which have been derived.

The flux trip setpoint 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 Protective System

The reactor protective 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 protective channels are listed below. 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.

a. Temperature - 5 sec

b. Pressure - 0.5 sec

c. Flux - 0.3 sec

d. Pump monitor - 1.0 sec

The reactor protective system will limit the power which 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 protective 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.

A 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 secondary steam line rupture is reflected in a drop of reactor coolant pressure. The low reactor pressure trip shuts down the unit for such an occurrence.

7.1.3.2 Functional Capability - Engineered Safeguards Protective System

The engineered safeguards protective system is a graded protective system. The progressive actions of the injection systems as initiated by the Engineered

drift during the interval between heat balance checks of nuclear instrumentation calibration.

7.1.3 SYSTEMS EVALUATION

7.1.3.1 Functional Capability - Reactor Protective System

The Reactor Protective 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 protective channels are listed below. 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.

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a. Temperature - 5 sec

b. Pressure - 0.5 sec

c. Flux - 0.3 sec

d. Pump monitor - 1.0 sec

The Reactor Protective System will limit the power which 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 Protective 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.

A 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 secondary steam line rupture is reflected in a drop of reactor coolant pressure. The low reactor pressure trip shuts down the unit for such an occurrence.

7.1.3.2 Functional Capability - Engineered Safeguards Protective System

The Engineered Safeguards Protective System is a graded protective system. The progressive actions of the injection systems as initiated by the Engineered

Safeguards Protective 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 the high pressure injection system at 1,800 psig. If the high pressure injection system does not arrest the pressure drop, then the low pressure injection system starts upon a signal of 200 psig.

The key variable in the detection of an accident which 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.

7.1.3.3 Preoperational Tests

Valid testing of analog sensing elements associated with the protective systems can only be accomplished through the actual manipulation of the measured variable and then comparing the results against a standard.

Routine preoperational tests can be performed by the substitution of a calibrating signal for the sensor except for the Reactor Building spray pumps and valves. Simulated neutron signals may be substituted in each of the start-up, 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. Routine preoperational tests of the spray systems can be performed by applying the pressure signals direct to the pressure switches.

The final defense against sensor failure during operation will be the 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 protective 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. If the fault were of the "tripped," open circuit mode, then 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

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"tripped", open circuit fault, a second identical fault within the same variable would trip the reactor.

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 protective 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 Engineered Safeguards Protective 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 protective systems has been arranged to minimize the possibility of loss of power to either protective system. Each channel of the protective system will be supplied from one of the four vital busses described in 8.2.2.6. The operator can initiate a reactor trip, independent of the automatic protective action.

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

7.1.3.5 Operational Tests

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

Most inputs to the protective 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 protective 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.

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7.2 REGULATING SYSTEMS

7.2.1 DESIGN BASES

7.2.1.1 Compensation Considerations

Reactor regulation is based upon the use of movable poison (control rods) and chemical neutron poison (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 poison.

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.

At the beginning of first core life when the moderator temperature reactivity coefficient may be zero or slightly positive, the control rod drive assembly response is many times 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 coefficient 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 full 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 poison dilution is initiated manually and terminated automatically or manually. Manual rod control is used below 15 per cent of full power. Automatic or manual rod control may be used above 15 per cent of full 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 above 90 per cent are limited to 5%/min. Decreasing power transients between 100 and 20 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.

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 the rod of greatest worth fully withdrawn and typical level of soluble poison (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. (A discussion of the accidental withdrawal of the rod group of greatest worth is presented in Section 14.)

The maximum rate of pure water addition does not change reactivity worth more than 7×10^{-6} $\Delta k/k$/sec. Reactivity control may be exchanged between rods and soluble poison 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 station, 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 startup 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.

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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 rods (or groups), sufficient to provide 1 per cent shutdown margin during startup, is required to be withdrawn prior to a dilution cycle.

f. Minimum pressurizer water level conditions must be met prior to 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 full 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~~ a 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 rods and by soluble poison (boric acid) 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 poison 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 of each unit for the reactivity components and control distribution are listed in Table 3-5.

Twenty-five of the 69 control rods are assigned to automatic control of reactor power level. These control rods are arranged in four symmetrical groups which operate in sequence. The position of one automatic group is used as an index to soluble poison dilution. Soluble poison 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 rods. The group size is preset, and individual control rod assignments to a group are made at a control rod grouping panel. However, the operator can select any individual control rod and any control rod group for motion as required.

A typical control rod group withdrawal scheme is as follows:

Group 1	12 rods	
Group 2	12 rods	
Group 3	12 rods	
Group 4	8 rods	
Group 5	8 rods	Regulating Groups
Group 6	9 rods	Regulating Groups
Group 7	4 rods	Regulating Groups
Group 8	4 rods	Regulating Groups

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 poison 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 group to compensate for the reduction in poison concentration. When the control 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 $5.8 \times 10^{-5} \Delta k/k/sec$.

The maximum rate of reactivity addition with the soluble poison system, i.e., injecting unborated water from the makeup system at 70 gpm maximum, is $7.0 \times 10^{-6} \Delta k/k/sec$.

Table 3-6 shows a shutdown reactivity analysis. The rod worth provided gives a shutdown margin in excess of 4.0% $\Delta k/k$ under normal conditions, and a margin in excess of 1% $\Delta k/k$ with the rod of greatest worth stuck in the withdrawn position.

Under conditions where cooldown to reactor building ambient conditions is required, concentrated soluble poison 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 Controls

The reactor controls are 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 assembly 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 of the reactor coolant system average temperature deviation ($\overline{\Delta T}$) from the set point, according to the following equation:

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

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 actual reactor power level signal (N_i), which is derived from the nuclear instrumentation. The resultant error signal ($N_d - N_i$) 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 assembly 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 system. Power to each reactor coolant pump motor is monitored as an indication of reactor coolant flow. Logic units continuously

compare the number of energized pumps to the measured reactor power to sense that the flow is adequate for the operating power level. If the flow is low, the reactor power level demand is reduced by the Integrated Reactor-Boiler-Turbine Control System.

7.2.2.2 Integrated Reactor-Boiler-Turbine Control System

The Integrated Reactor-Boiler-Turbine Control System maintains constant average reactor coolant temperature and constant steam pressure in the unit during steady state and transient operation between 15 and 100 per cent full 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 Reactor-Boiler-Turbine Control System described in 7.2.2.1.4. Figure 7-8 shows the boiler-turbine control portion of the Integrated Reactor-Boiler-Turbine 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 full power with optional manual control. The reactor controls are designed for manual operation below 15 per cent full power and for automatic or manual operation above 15 per cent full 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 feedwater flow demand to the steam generator is a summation of the megawatt demand and the steam pressure error.

The Integrated Reactor-Boiler-Turbine Control obtains a load demand signal from the area load control or from the operator. A frequency loop is added to compensate for 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:

Maximum load limit	100%
Minimum load limit	15%
Rate limit	10%/min

The runback limiter acts to limit the load demand if one or more reactor coolant pumps are inoperative, or total feedwater flow lags total feedwater demand by more than 5 per cent.

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

7.2.2.2.1 Turbine Controls

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 Controls

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 controls act 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 auctioneer. At low loads below 15 per cent, the turbine bypass valves will operate to limit steam pressure rise.

The steam generator controls also provide ratio, limit, and runback actions as shown in Figure 7-8 which include:

a. <u>Steam Generator Load Ratio Control</u>

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 coolant temperatures during operation with more reactor coolant pumps in one loop than in the other.

b. Rate Limits

Rate limiters restrict loading or unloading rates to those which 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 is low, 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 Reactor-Boiler-Turbine 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 reactor controller electrical power reverts reactor control to the manual mode.

7.2.3.2 Interlocking

Control rod withdrawal is prevented on 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 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 rod drive control system initiates a reactor trip.

When emergency conditions arise which 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 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 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 Section 14.)

The features which permit continued operation under load rejection conditions include:

a. Integrated Reactor-Boiler-Turbine Control System

During normal operation the Integrated Reactor-Boiler-Turbine Control System (see Figure 7-8), controls the unit load in response to load demand from the automatic load 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.

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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 which is 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, causes 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 full 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 full 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.

The following additional features are pertinent to the nuclear instrumentation system:

a. Independent power supplies are included in each channel. Primary power originates from the vital busses described in 8.2.2.6. 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 protective 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 location of the neutron detectors is 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 full 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 8.2.2.6.

7.3.1.2.2 Reliability and Component Failure

The requirements established for the reactor protective 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 Protective Requirements

The relation of the power range channels to the Reactor Protective System has been described in Section 7.1. To maintain the desired accuracy in trip action, the total error from drift in the power range channels will be held to ±1/2 per cent at full 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 ±1/2 per cent of full 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 auxiliary reactor systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the 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 which 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 auxiliary reactor system drawings in Sections 5, 6, 9, and 11. Process variables are monitored as shown on the nonnuclear instrumentation and auxiliary reactor 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 by force balance transmitter devices. Pressures are measured in the reactor coolant system, the steam system, and the auxiliary reactor systems. Pressure signals for high and low reactor coolant pressures and high reactor building pressure are provided to the protective systems.

c. Reactor coolant pump motor operation is monitored as an indication of reactor coolant flow. This information is fed to the reactor controls and reactor protective system. In addition, reactor coolant flow signals are obtained from one calibrated flow tube and four flow transmitters installed in each reactor coolant loop. Buffered outputs from each of the flow transmitters are employed to provide analog flow signals for the reactor protective 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. ~~This level information is fed to the engineered safeguards protective system.~~ The pressurizer level is controlled by 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 pressurizer level control valve in the makeup line to maintain a constant level.

f. Pressurizer and reactor coolant system pressure is maintained by a control system which compares the reactor coolant system pressure with a setpoint, and then energizes pressurizer electrical heaters in banks at preset pressure values below 2,200 psia or actuates spray control valves if the pressure increases to 2,250 psia.

7.3.2.2 System Evaluation

Redundant instrumentation has been provided for all inputs to the protective 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 INSTRUMENTATION

7.3.3.1 Design Basis

The incore instrumentation system provides neutron flux and temperature 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, and temperature detectors measure the core temperature differential 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 instrumentation system consists of assemblies of self-powered neutron detectors, temperature detectors, and support tubes located at 51 preselected radial positions within the core. Twenty-nine assemblies are positioned within an eighth segment of the core to provide data for a detailed analysis of that segment while 22 additional assemblies are distributed throughout the balance of the core to provide data for overall core performance calculations. The incore monitoring locations are shown by Figure 7-12.

Each of the 51 incore detector assemblies consists of four local flux detectors, one background detector, two (inlet and outlet) temperature detectors, and a calibration tube. The local detectors are positioned at four 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. The temperature detectors, one located at the top of the fuel assembly and the other positioned just below the bottom of the fuel assembly, measure the temperature difference across the core to verify hot channel calculations and flow distribution in the core.

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

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 empty center tubes of 51 selected fuel assemblies. The center tube then serves as the guide for the incore detector assemblies. 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 position in the core, and the high pressure seals are secured.

7.3.3.2.2 Calibration Techniques

The nature of the proposed 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 full 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.

~~Calibration of temperature detectors is performed out of core. The relative temperature indication of each detector is verified when the reactor coolant system is isothermal.~~

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. They 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 which started 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 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.

The following preliminary conclusions have been 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:

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 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 units will be located in one 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 unit control or emergency functions.

7.4.1 GENERAL LAYOUT

The control room will be designed so that one man can supervise operation of both units 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 station. The control board is divided into relative areas to show the location of control stations and information display pertaining to various sub-systems.

7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTIONS

Consideration is given to the fact that certain systems normally require more attention from the operator. The integrated reactor-boiler-turbine control system will therefore be located nearest the center line of the boards (Sections 1, 2, and 3 on Figure 7-14).

On Section 1 of the control board, one indicator will be provided for each group of control rods. Fault detectors in the rod drive control system will be used to alert the operator should an abnormal condition exist for any individual control rod. Switches will be provided to allow the operator to monitor any single control rod position on a common indicator. Displayed in this same area will be limit lights for each control rod group and all nuclear instrumentation information required to start up and operate the reactor. Control rods are to be manipulated from the bench position of Section 1. Computer readout facilities for alarm monitoring and sequence monitoring will be located here to aid the operator in safe and reliable operation.

A process computer will be used on each unit for alarm monitoring, performance monitoring, data logging, sequence monitoring, and control of some functions during startup and shutdown of the turbine-generator. Any of the monitoring and display functions of the computer which deal with safety aspects of the nuclear steam supply system will be duplicated elsewhere in the control room. This scope of computer application has been successfully applied to units presently in operation on the Duke system. One or both of these computers will be used to perform on-site fuel management calculations.

Variables associated with operation of the secondary side of the station will be displayed and controlled from Section 2 of the control board. These variables include steam pressure and temperature, feedwater flow and temperature, electrical load, and other signals involved in the integrated control system.

Section 3 of the control board will contain provisions for indication and control of the reactor coolant system. Redundant indication is incorporated in the system design since pressure and temperature variables of the reactor coolant will be used to initiate safety features.

The engineered safeguards system will be controlled and monitored from Section 4 of the control board. Valve position indicating lights will be provided as a means of verifying the proper operation of the control and isolation valves

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following initiation of the engineered safeguards. Control switches located on this panel allow manual operation or test of individual units. Also located on this section will be the control switches, indicating lights, and meters for fans and pumps required for emergency conditions.

Control and display equipment for station auxiliary systems will be located on Section 5 of the control board.

Reactor coolant pump control located on Section 6 of the control board will consist of the pump controls and auxiliary instrumentation required for pump operation. Also mounted on this section will be auxiliary electrical system controls required for manual switching between the various power sources described in 8.2.2.3.

Controls and indications for all ventilation systems except the Reactor Building system will be located on Section 7 of the control board.

In order to maintain the desired accessibility for control of the station, miscellaneous recorders not required for station control will be located on the vertical recorder board where they will be visible to the operator. Radiation monitoring information will also be indicated there.

7.4.3 SUMMARY OF ALARMS

Visible and audible alarm units will be incorporated into the control board 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 and from the source range nuclear instrumentation. Audible alarms will be sounded in appropriate areas throughout the station if high radiation conditions are present.

7.4.4 COMMUNICATION

Station 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 station. Acoustic booths will be supplied in areas where the background noise level is high. Communication outside the station will be through the Southern Bell Telephone & Telegraph Company system and the Duke private telephone and microwave systems.

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 station operation.

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. Approximately 35 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 either 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, or by a-c motor starters which have individual control transformers.

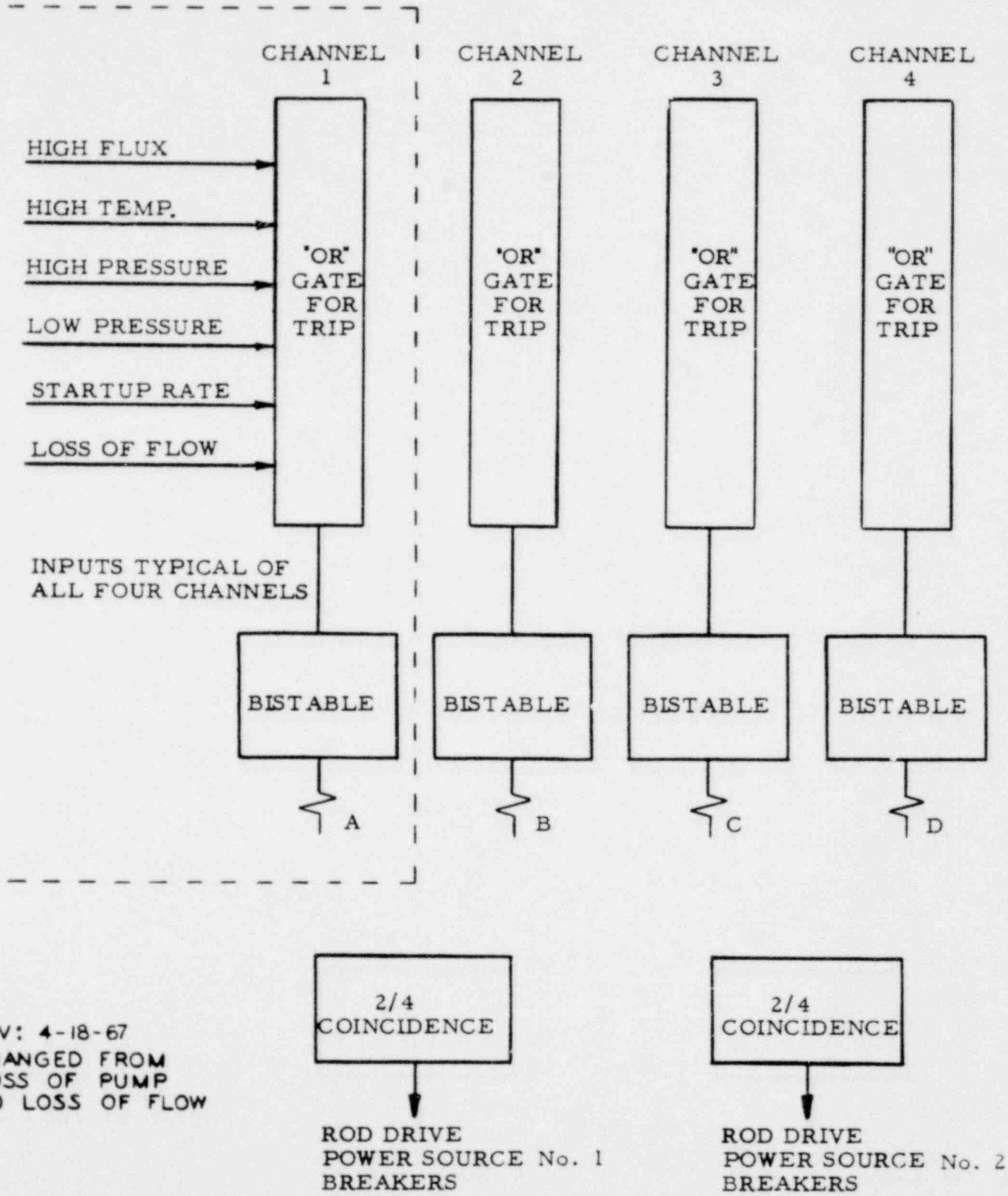
7.4.6 AUXILIARY CONTROL STATIONS

Auxiliary control stations will be provided where their use simplifies control of auxiliary systems equipment such as waste evaporator, sample valve selectors, chemical addition, etc. The control functions initiated from local control stations will not directly involve either the engineered safeguards system or the reactor control system. Sufficient indicators and alarms will be provided so that the central control room operator is made aware of abnormal conditions involving remote control stations.

7.4.7 SAFETY FEATURES

Primary objectives in the control room layout will be to provide the necessary controls to start, operate and shut down the units 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 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. The station computer will be used to perform high priority programs to verify proper operation of all engineered safeguards functions. Any deviations from predetermined conditions will be alarmed so that corrective action may be taken by the operator using redundant controls provided on the control panel.

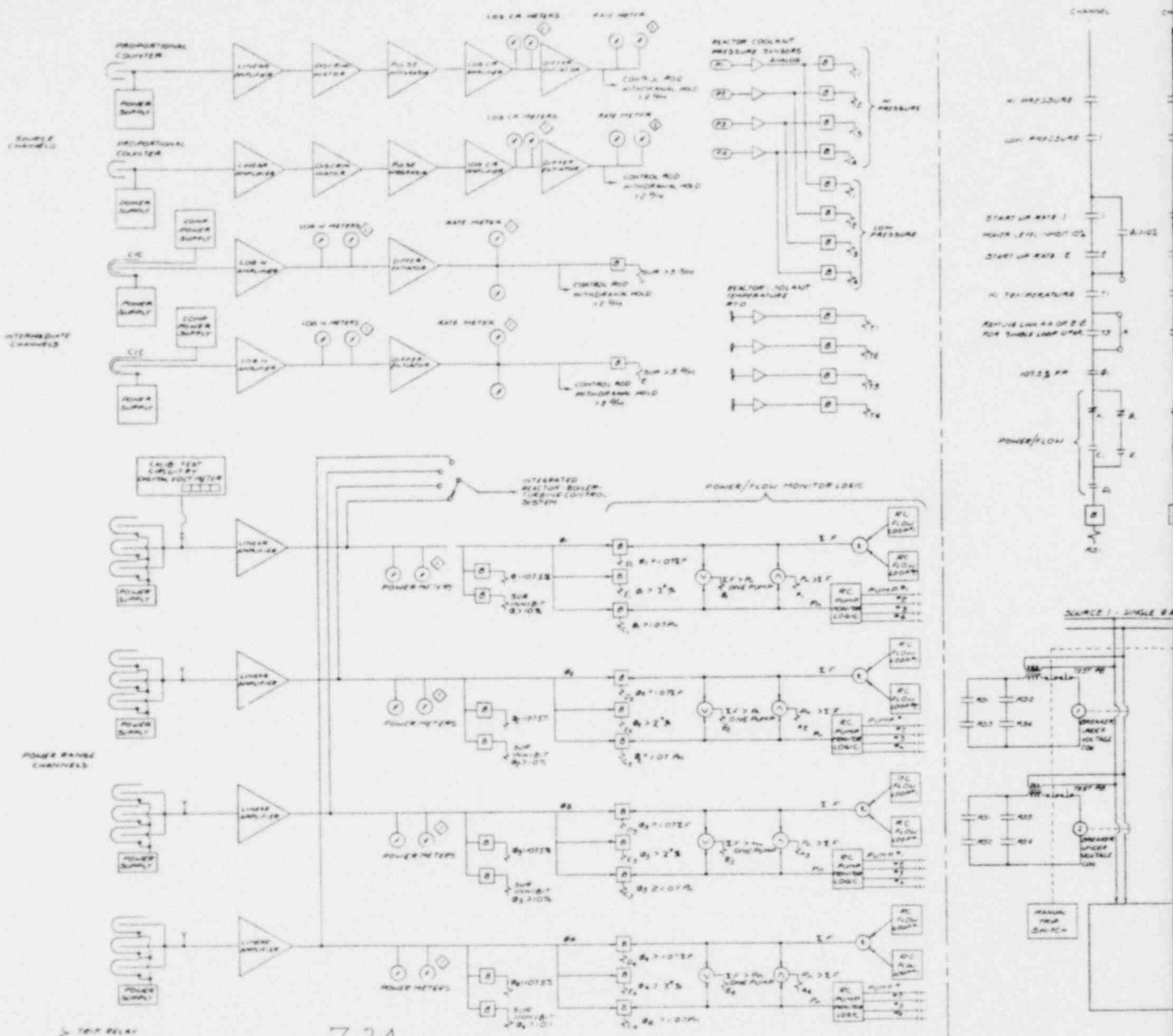


REV: 4-18-67
CHANGED FROM LOSS OF PUMP TO LOSS OF FLOW

FIGURE 7-1

REACTOR PROTECTIVE SYSTEM BLOCK DIAGRAM

NUCLEAR INSTRUMENTATION AND REACTOR PROTECTIVE SYST



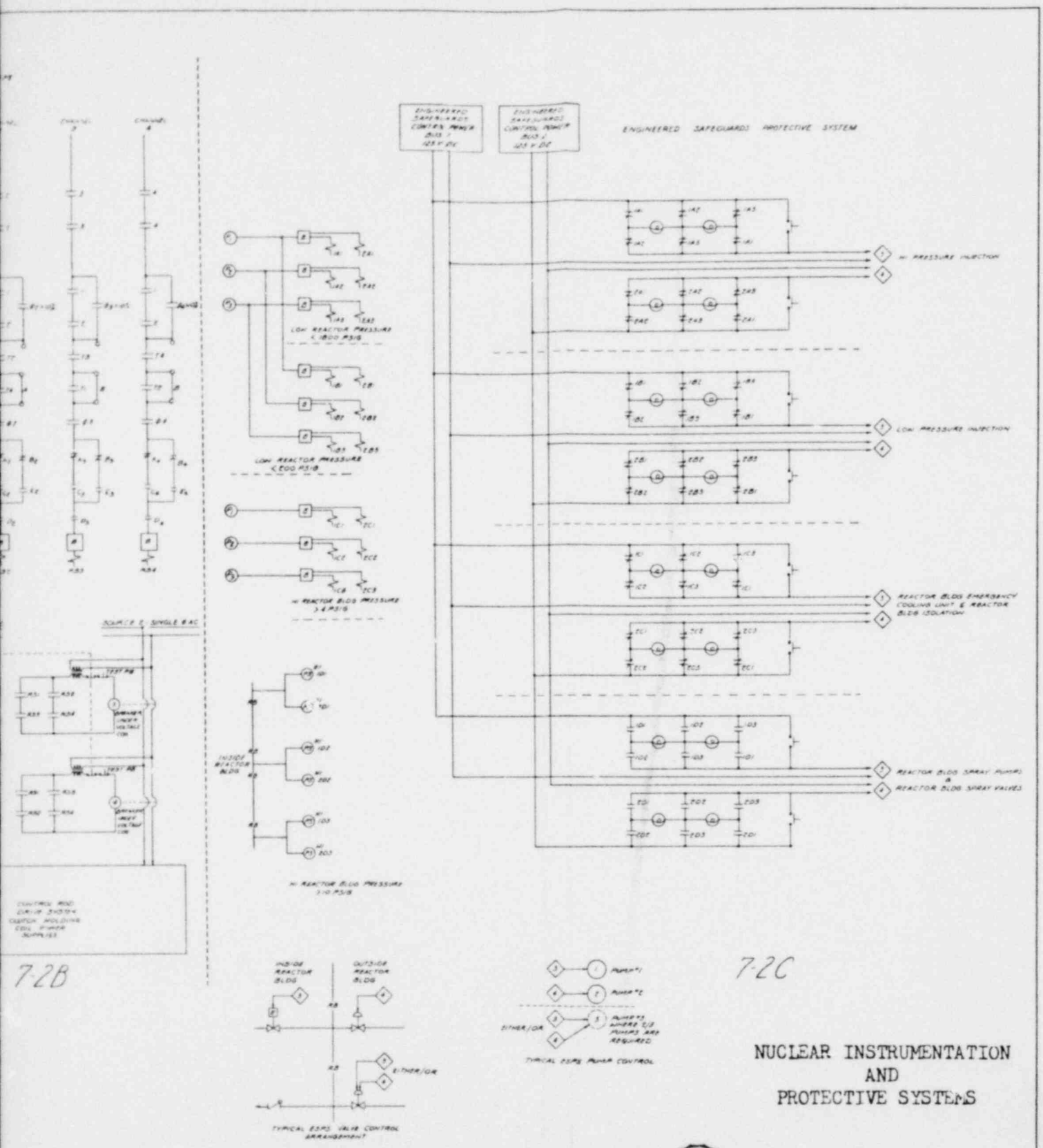
7-2A

REV: 4-1-87
(A) CHANGED ESFS FROM ENERGIZED TO DE-ENERGIZED SYSTEM
(B) DELETED PRESSURIZER LOW LEVEL TRIP FROM ESFS
(C) CHANGED R.B. SPRAY ACTUATION SENSORS
(D) CHANGED C.R.D. TRIP CIRCUIT
(E) ADDED ESFS CONTROL SYMBOLS

REV: 4-14-87
CHANGED PUMP MONITOR LOGIC TO INCLUDE POWER/FLOW CKT

REV: 4-29-87
CHANGED 3.8 POWER TO 1.8 FOR CONTROL ROD DRIVES CLUTCH HOLDING COIL

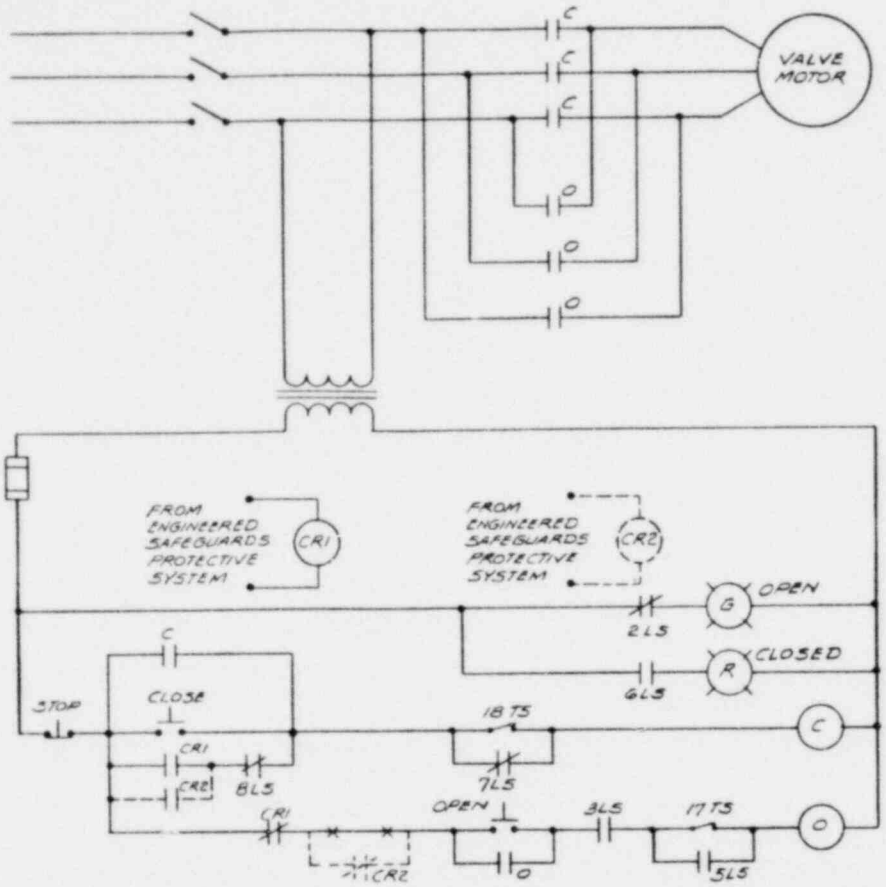
REV: 5-14-87
(A) CHANGED START-UP TO SOURCE
(B) DELETED FLUX AVERAGING CKT



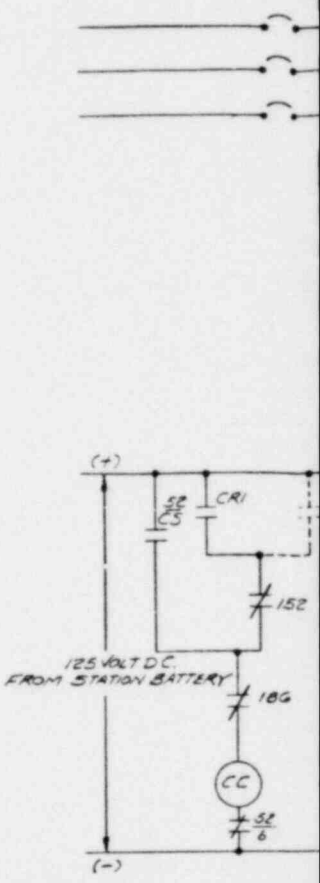
NUCLEAR INSTRUMENTATION
AND
PROTECTIVE SYSTEMS

OCONEE NUCLEAR STATION
FIGURE 7-2

TYPICAL REACTOR BUILDING ISOLATION VALVE



CONTROL CIRCUIT
(CIRC



SWITCHES AND CONTACTS SHOWN WITH VALVE IN FULL OPEN POSITION

C - MAIN CONTACTOR, CLOSING
O - MAIN CONTACTOR, OPENING
CR1 - CONTROL RELAY
CR2 - CONTROL RELAY
TS - TORQUE SWITCH
LS - LIMIT SWITCH

CONTACTS SHOWN I
CR1 - CONTROL RELAY
CR2 - CONTROL RELAY
CC - CIRCUIT BREAKER
TC - CIRCUIT BREAKER

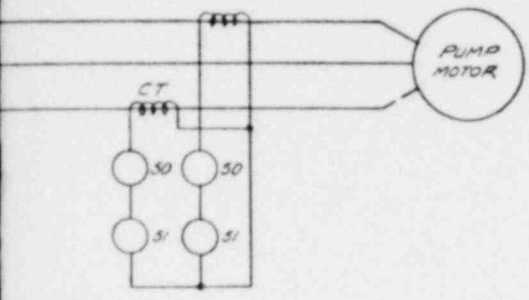
50 - INSTANTANEOUS O
51 - TIME OVERCURREN
52/CS - CIRCUIT BREAKER
152 - CIRCUIT BREAKER
18G - AUXILIARY TRIPPING

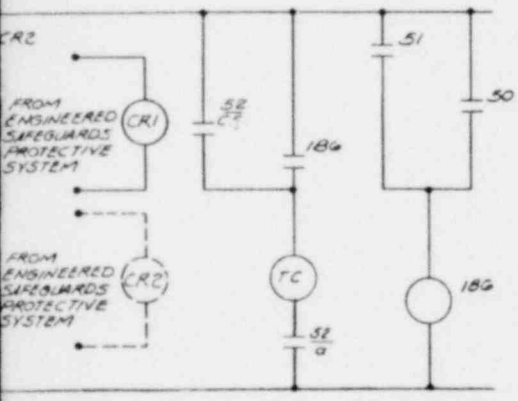
NOTE: CR2, SHOWN D
ONLY WHEN REDUNDA
REQUIRED

LIMIT SWITCH CONTACT DEVELOPMENT

CONTACT	VALVE FULL OPEN	INTERMEDIATE VALVE POSITION	VALVE FULL CLOSED	CONTACT FUNCTION
1	CLOSED	OPEN	OPEN	SPARE
2	CLOSED	OPEN	OPEN	OPEN IND LT
3	OPEN	CLOSED	CLOSED	OPEN LIMIT
4	OPEN	CLOSED	CLOSED	SPARE
5	OPEN	OPEN	CLOSED	TORQUE SW BYPASS
6	OPEN	OPEN	CLOSED	CLOSED IND LIGHT
7	CLOSED	CLOSED	OPEN	TORQUE SW BYPASS
8	CLOSED	CLOSED	OPEN	HOLD IN CIRCUIT
17	OPENING TORQUE SWITCH - OPENS ON MECHANICAL OVERLOAD IN OPENING DIRECTION			
18	CLOSING TORQUE SWITCH - OPENS ON MECHANICAL OVERLOAD IN CLOSING DIRECTION			

FOR LOW PRESSURE INJECTION PUMP
UIT BREAKER CONTROL)





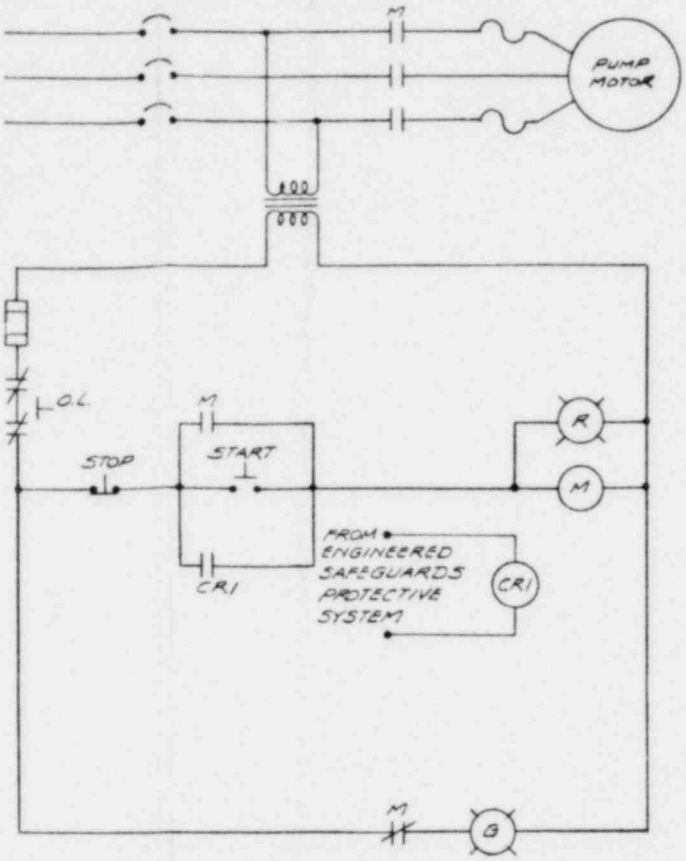
DEENERGIZED POSITION

CLOSING COIL
TRIP COIL

VERCURRENT RELAY
T RELAY
CONTROL SWITCH
UXILIARY SWITCH
RELAY

OTTED, IS USED
NT CONTROL IS

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



CONTACTS SHOWN IN DEENERGIZED POSITION

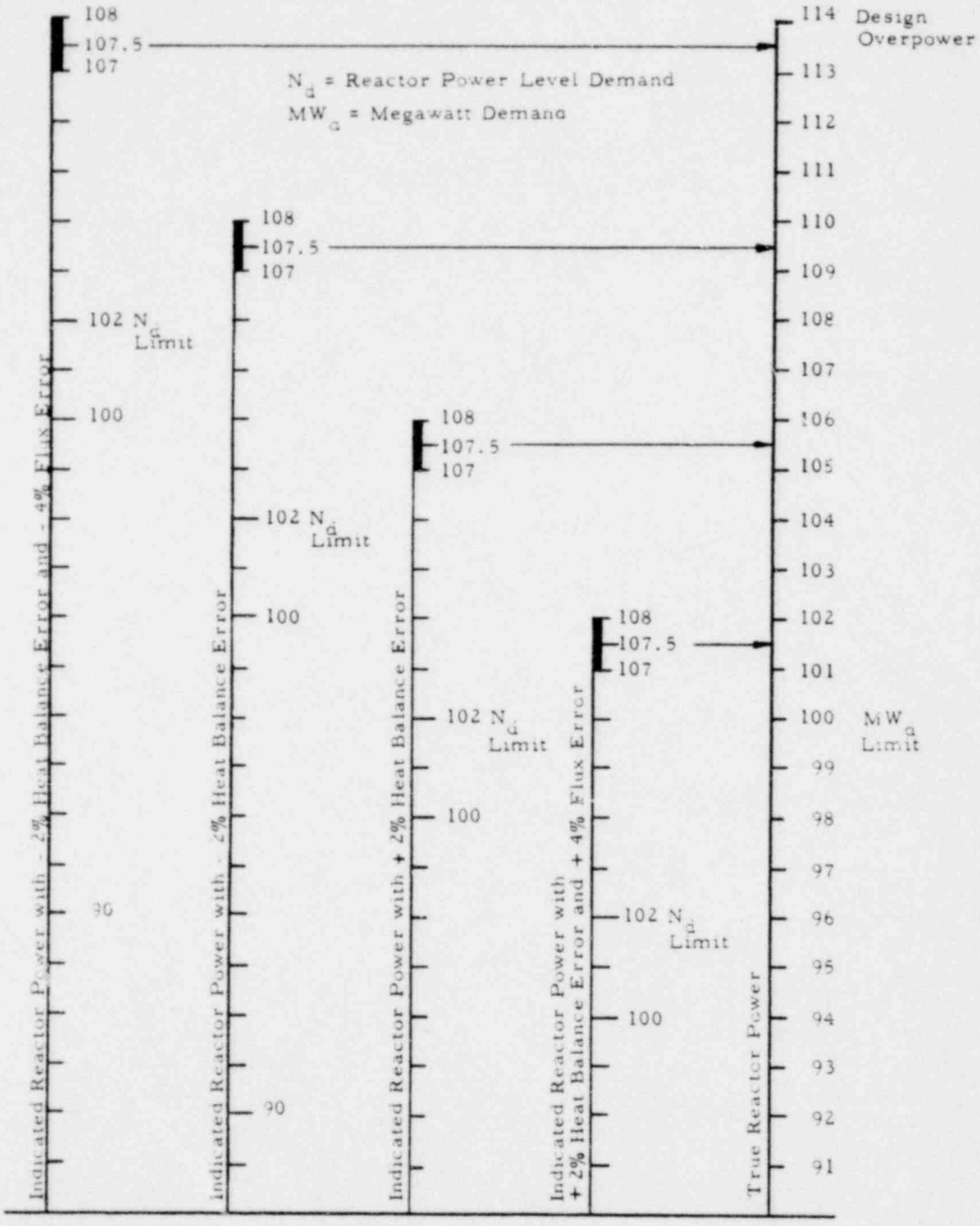
CR1 - CONTROL RELAY
M - MAIN CONTACTOR

REV: 4-1-67
CHANGED ESPS FROM ENERGIZED TO DE-ENERGIZED SYSTEM

TYPICAL CONTROL CIRCUITS
FOR ENGINEERED
SAFEGUARDS SYSTEM EQUIPMENT

DUKE POWER

OCONEE NUCLEAR STATION
FIGURE 7-3



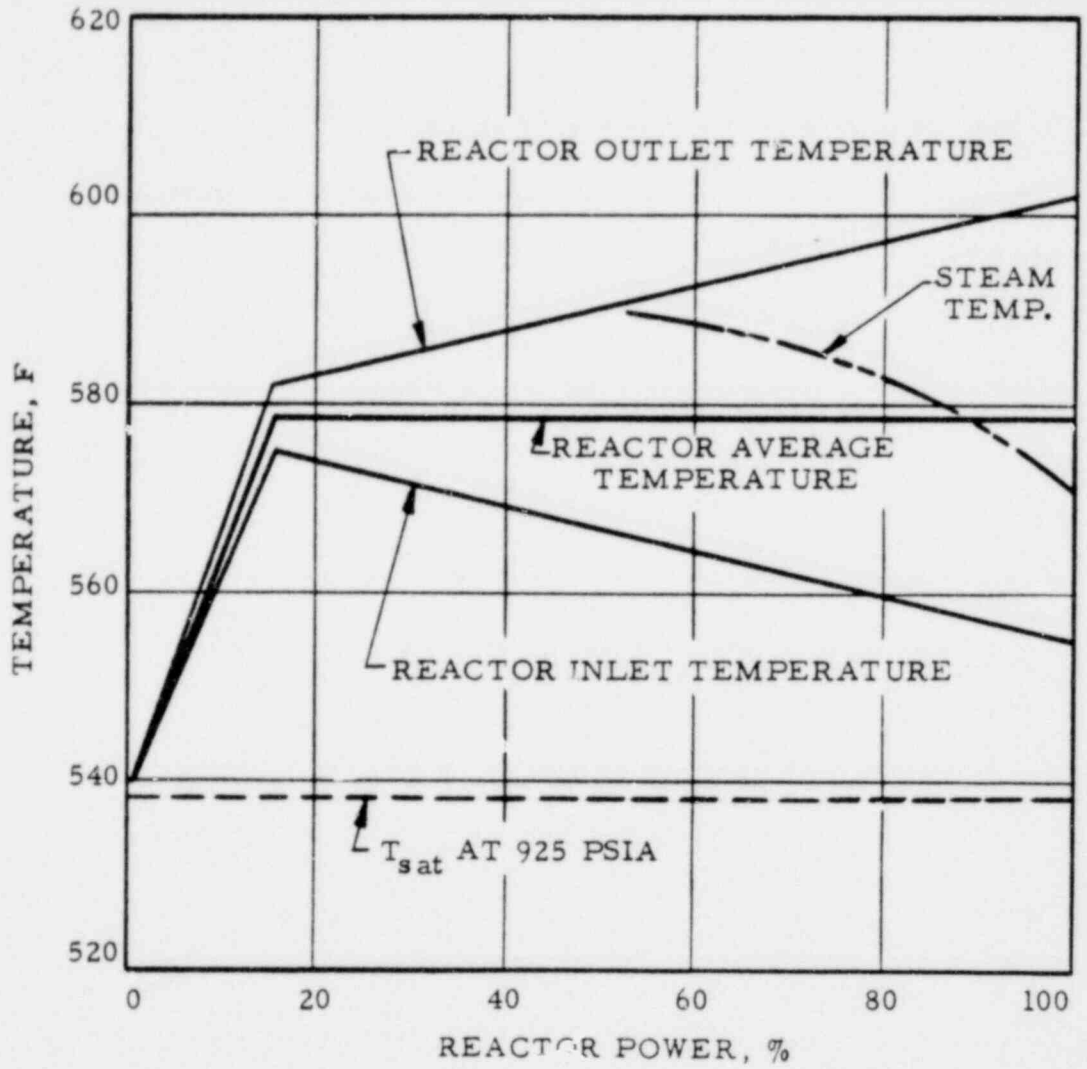
REACTOR TRIP at 107.5% ± 0.5% INDICATED POWER

REACTOR POWER MEASUREMENT ERRORS

AND CONTROL LIMITS

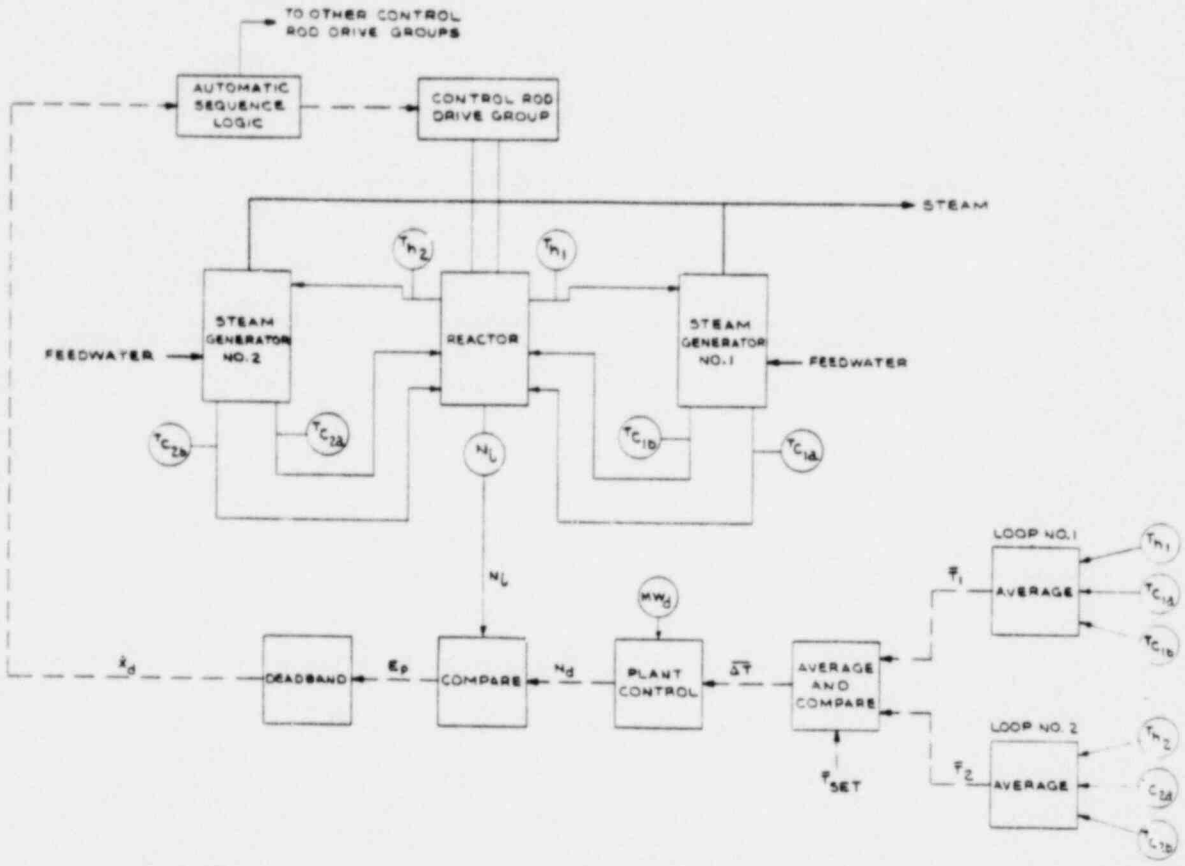
OCONEE NUCLEAR STATION

FIGURE 7-4



REACTOR AND STEAM TEMPERATURES
VERSUS REACTOR POWER

OCONEE NUCLEAR STATION
FIGURE 7-5



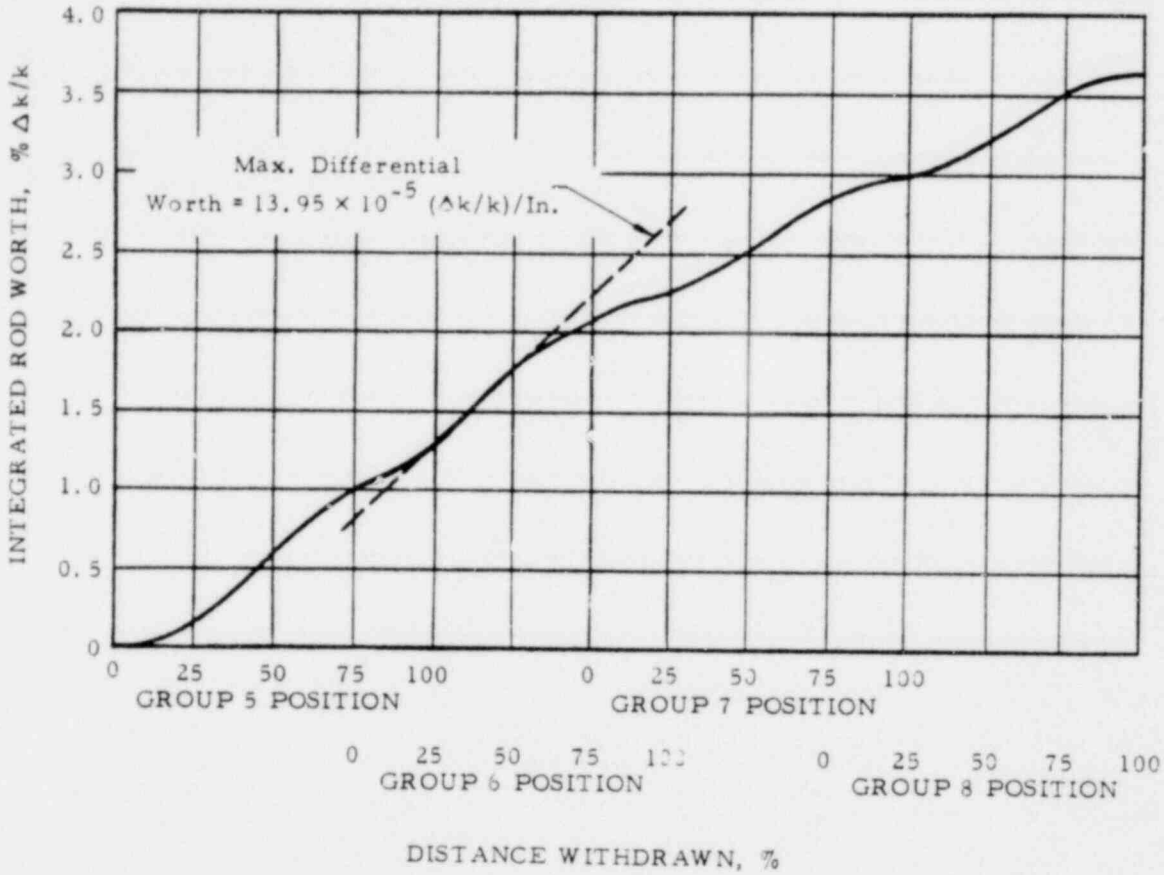
LEGEND

- REACTOR COOLANT SYSTEM
- STEAM SYSTEM
- CONTROL SYSTEM
- T_C = REACTOR COOLANT SYSTEM COLD LEG TEMPERATURE
- T_h = REACTOR COOLANT SYSTEM HOT LEG TEMPERATURE
- $\bar{T}$ = AVERAGE REACTOR COOLANT SYSTEM TEMPERATURE
- $\overline{\Delta T}$ = DEVIATION OF AVERAGE TEMPERATURE FROM SETPOINT
- E_p = REACTOR POWER LEVEL ERROR, $N_d - N_L$
- $\dot{x}_d$ = ROD VELOCITY DEMAND
- MW_d = MEGAWATT DEMAND
- N_L = REACTOR POWER LEVEL
- N_d = REACTOR POWER LEVEL DEMAND

REACTOR CONTROL DIAGRAM
INTEGRATED REACTOR-BOILER-TURBINE
CONTROL SYSTEM

Rev: 6-16-67
Deleted Flux Averaging Ckt

OCONEE NUCLEAR STATION
FIGURE 7-6

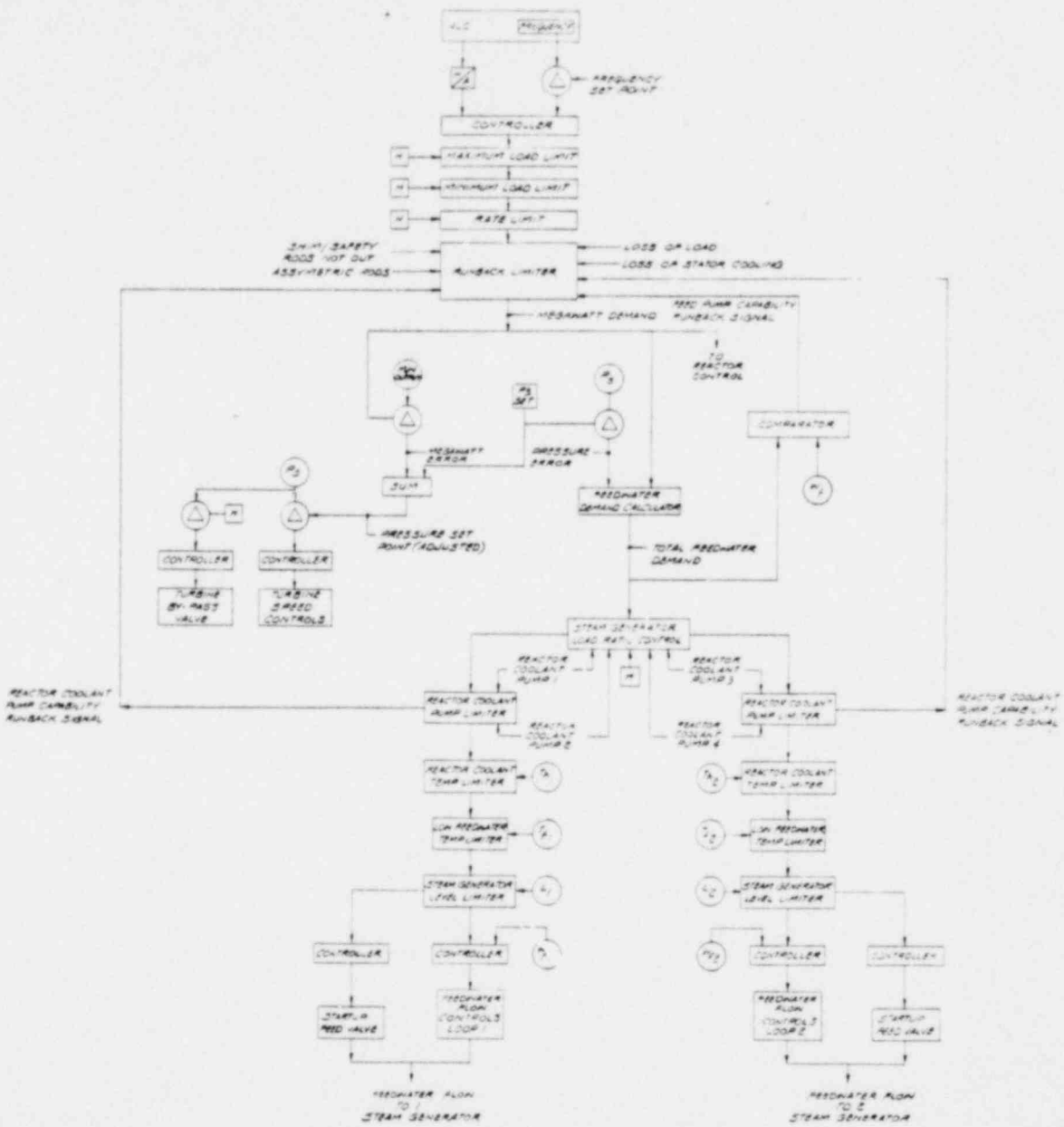


AUTOMATIC CONTROL ROD GROUPS

TYPICAL WORTH CURVE VERSUS DISTANCE WITHDRAWN

OCONEE NUCLEAR STATION

FIGURE 7-7



LEGEND

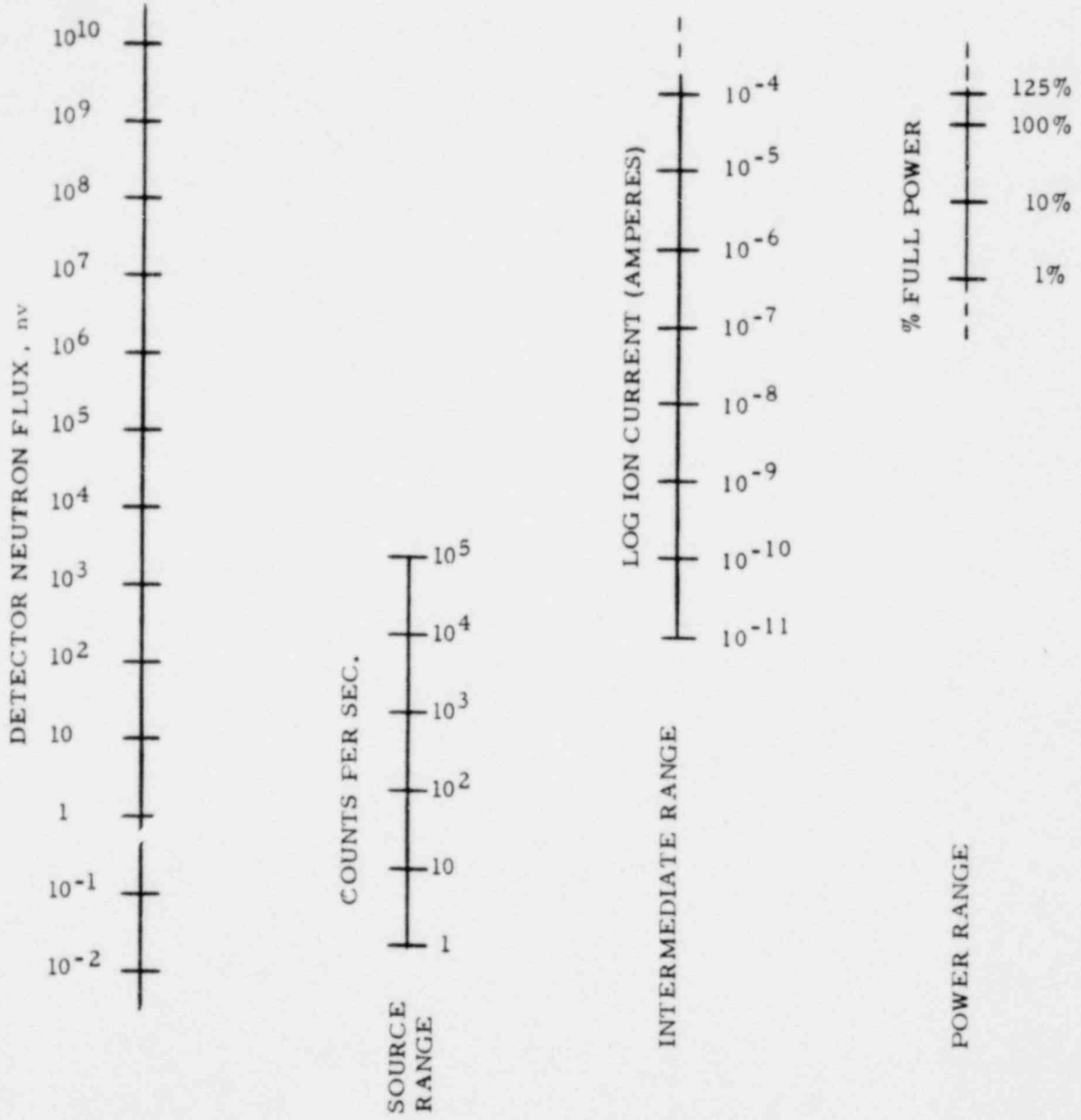
ALC - AREA LOAD CONTROL SIGNAL
△ - DIFFERENCE
P_s - STEAM PRESSURE
$P_{s(SET)}$ - STEAM SET PRESSURE
W_f - FEEDWATER FLOW
M - MANUAL SET POINT
M/A - MANUAL AUTOMATIC
L_s - STEAM GENERATOR LEVEL
T_c - REACTOR OUTLET TEMPERATURE
T_f - FEEDWATER TEMPERATURE

THE NUMBER SUBSCRIPTS REFER TO LOOP 1 & LOOP 2

BOILER-TURBINE CONTROL DIAGRAM
INTEGRATED REACTOR-BOILER-TURBINE
CONTROL SYSTEM

OCONEE NUCLEAR STATION

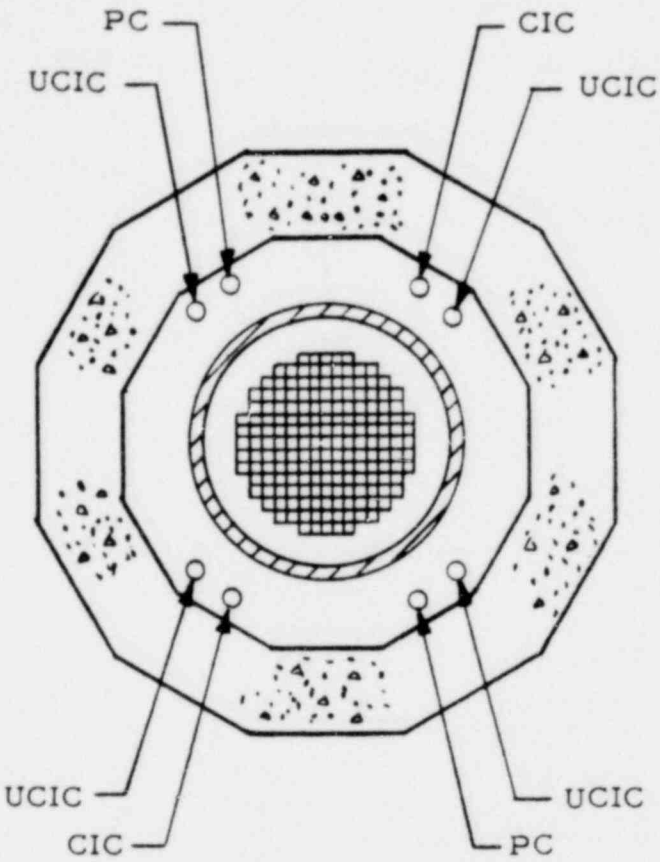
FIGURE 7-8



NUCLEAR INSTRUMENTATION FLUX RANGES

OCONEE NUCLEAR STATION

FIGURE 7-9



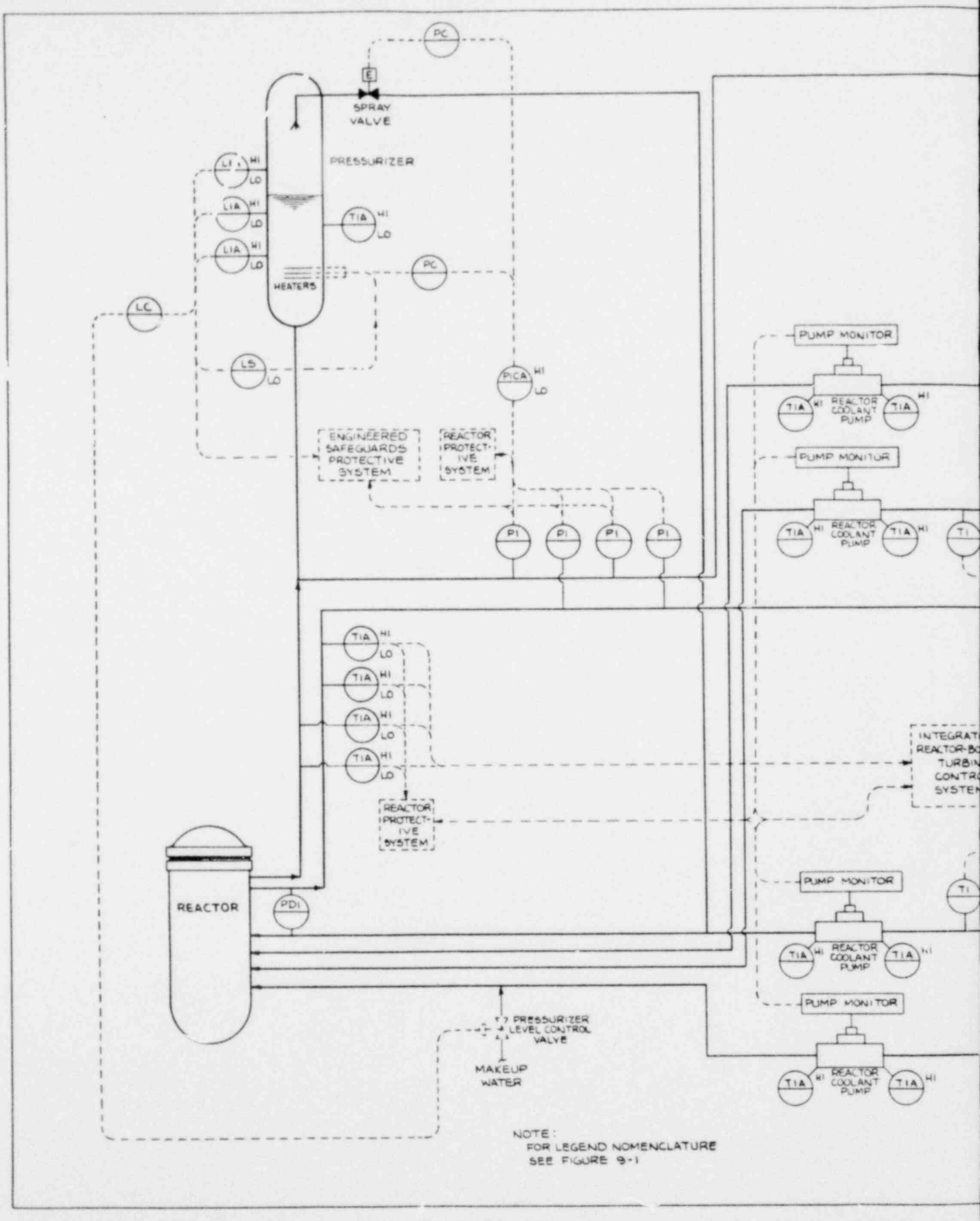
LEGEND

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

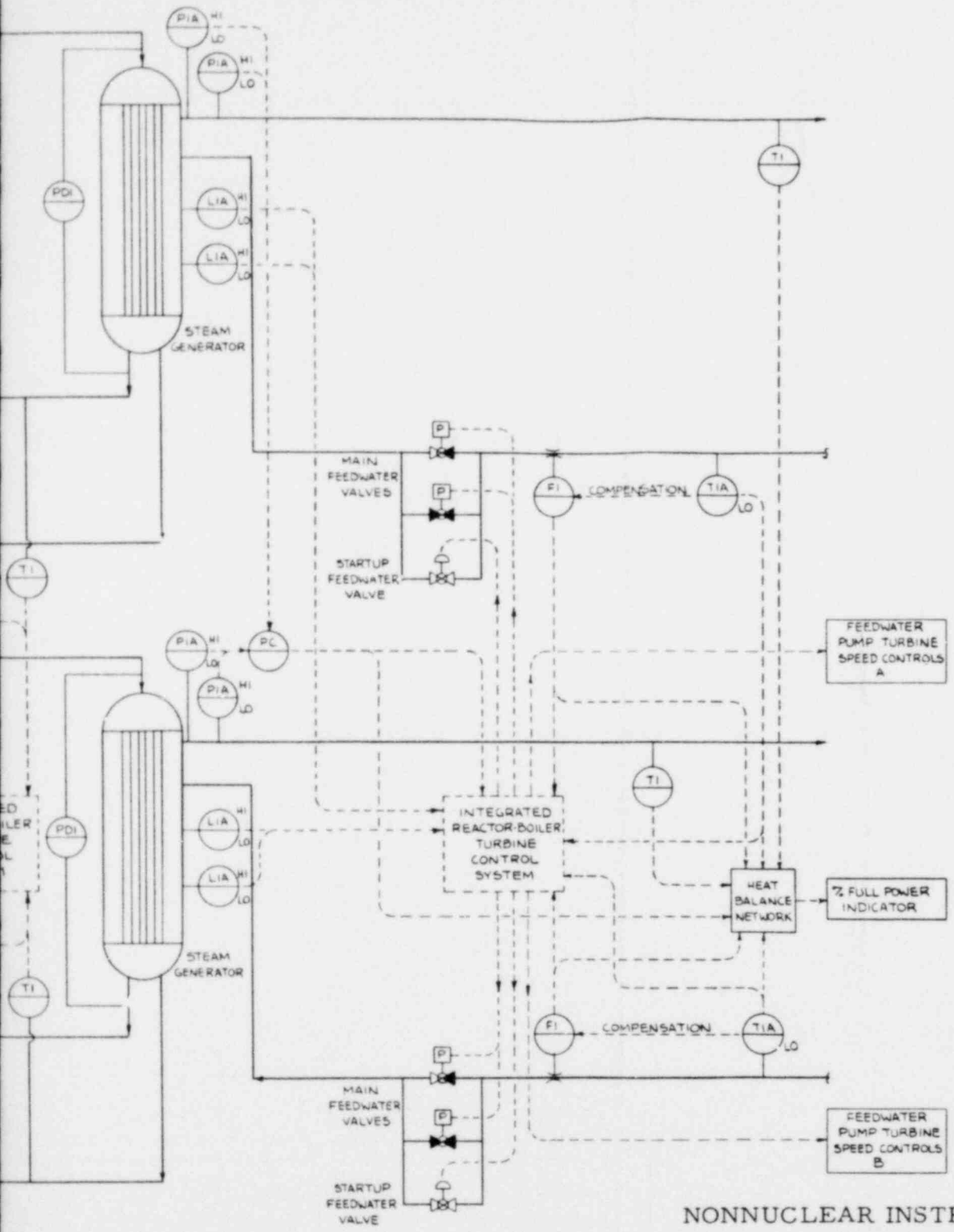
NUCLEAR INSTRUMENTATION DETECTOR LOCATIONS

OCONEE NUCLEAR STATION

FIGURE 7-10



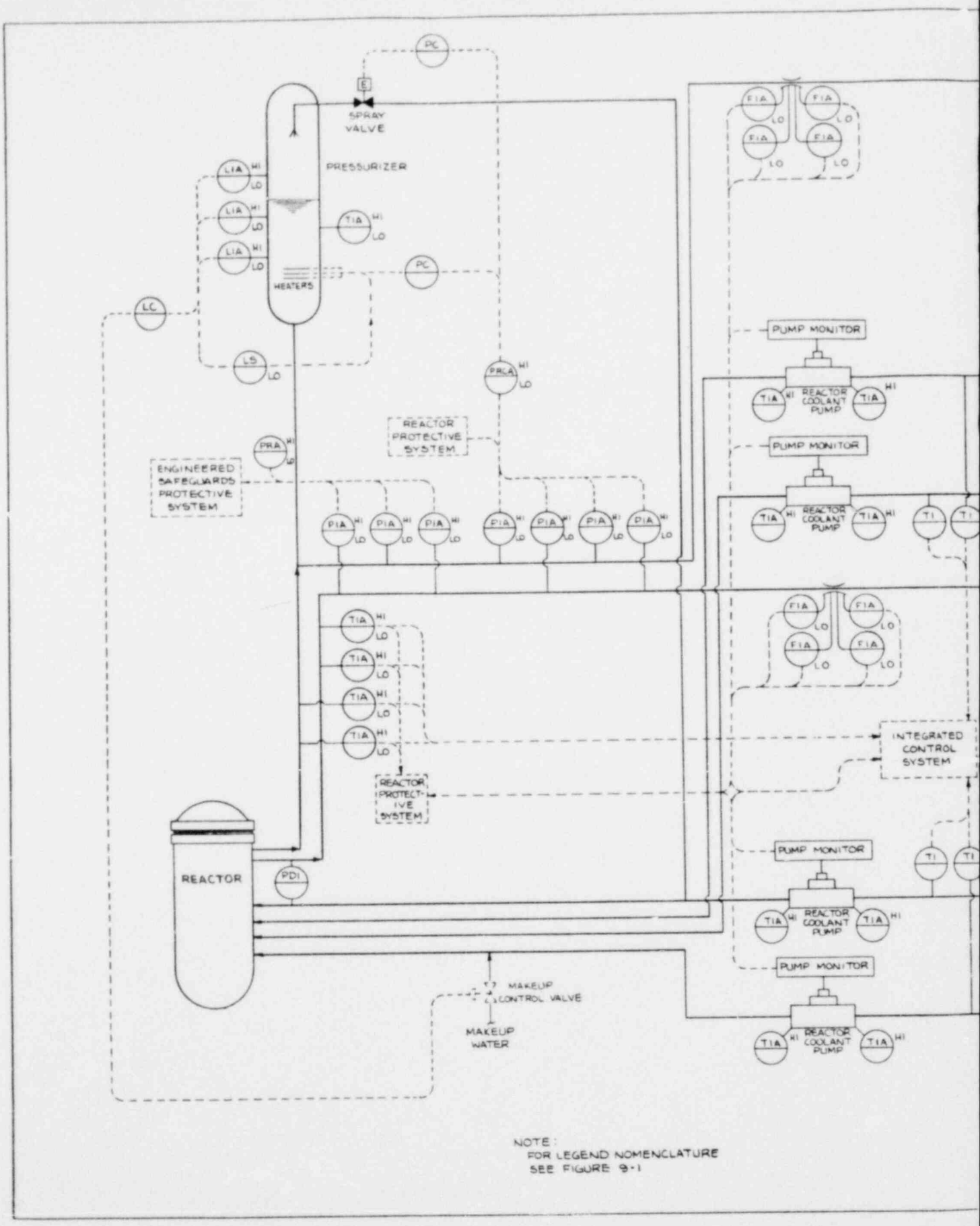
NOTE:
FOR LEGEND NOMENCLATURE
SEE FIGURE 9-1



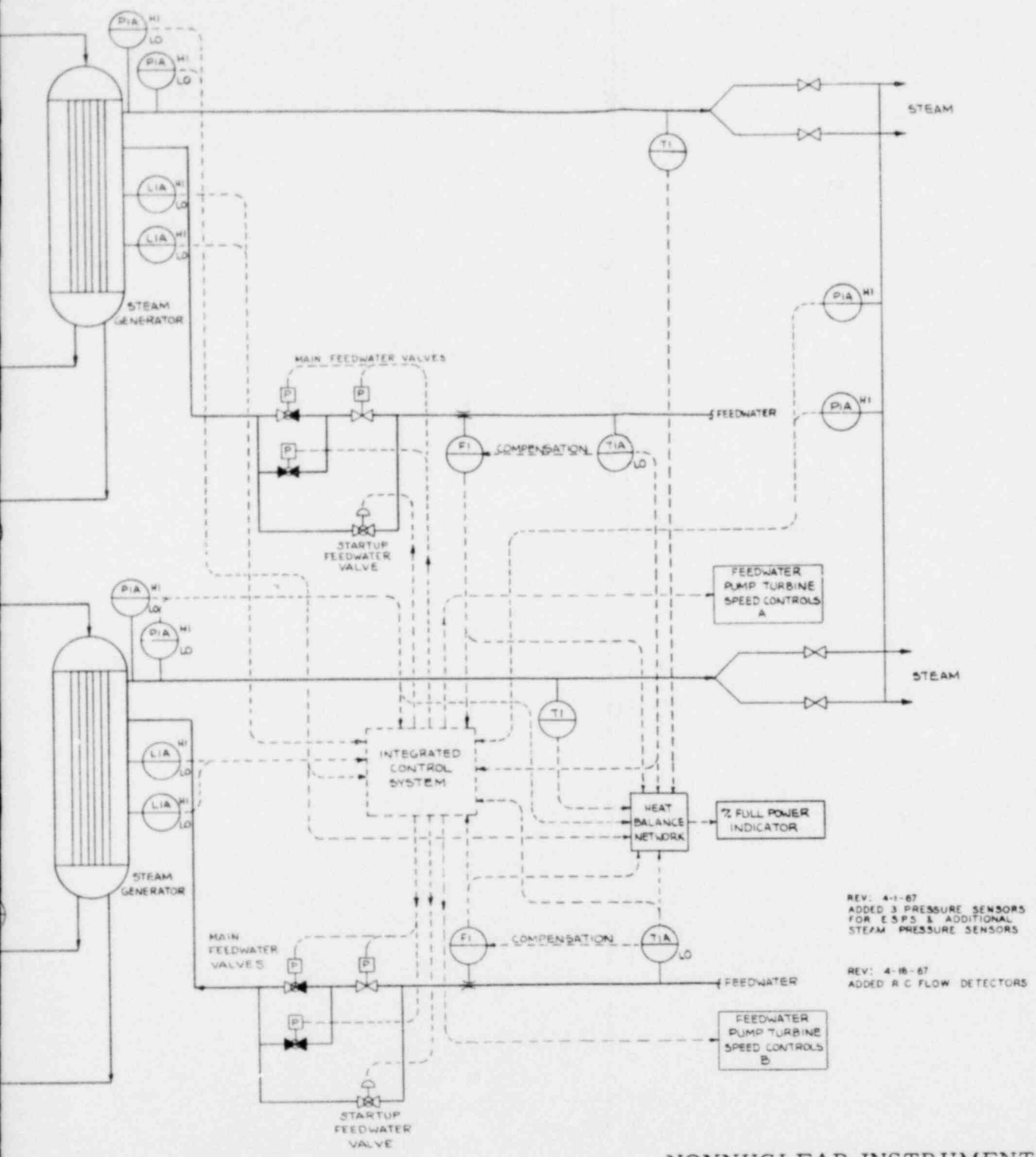
NONNUCLEAR INSTRUMENTATION SCHEMATIC

OCONEE NUCLEAR STATION

FIGURE 7-11



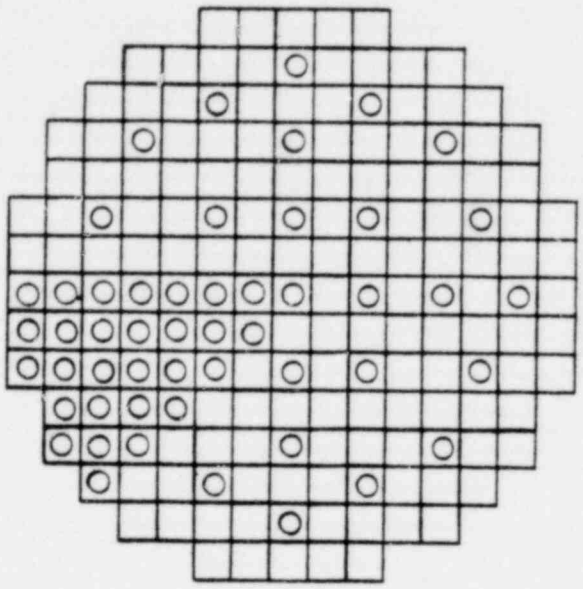
NOTE:
FOR LEGEND NOMENCLATURE
SEE FIGURE 9-1



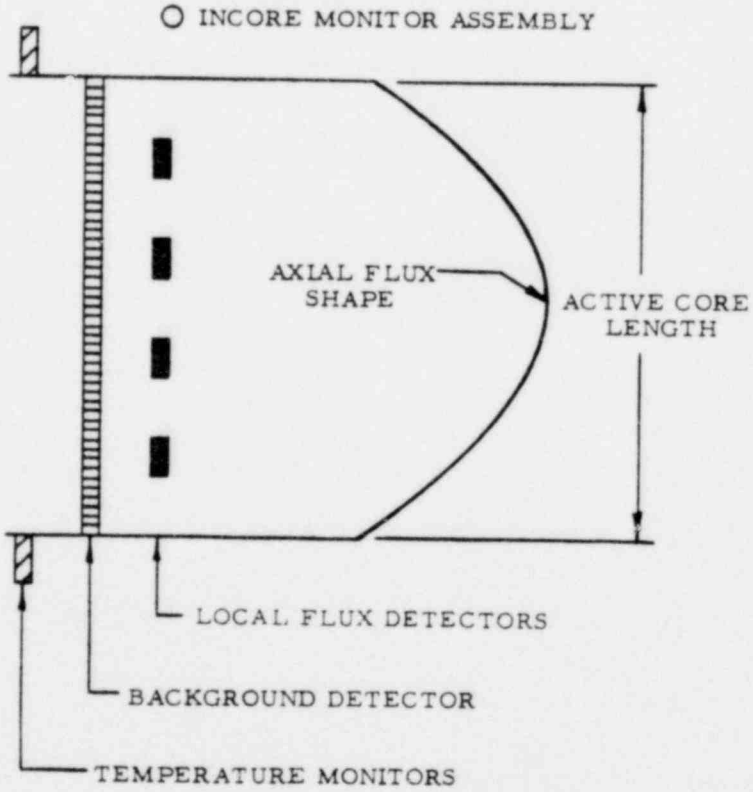
NONNUCLEAR INSTRUMENTATION SCHEMATIC

OCONEE NUCLEAR STATION

FIGURE 7-11



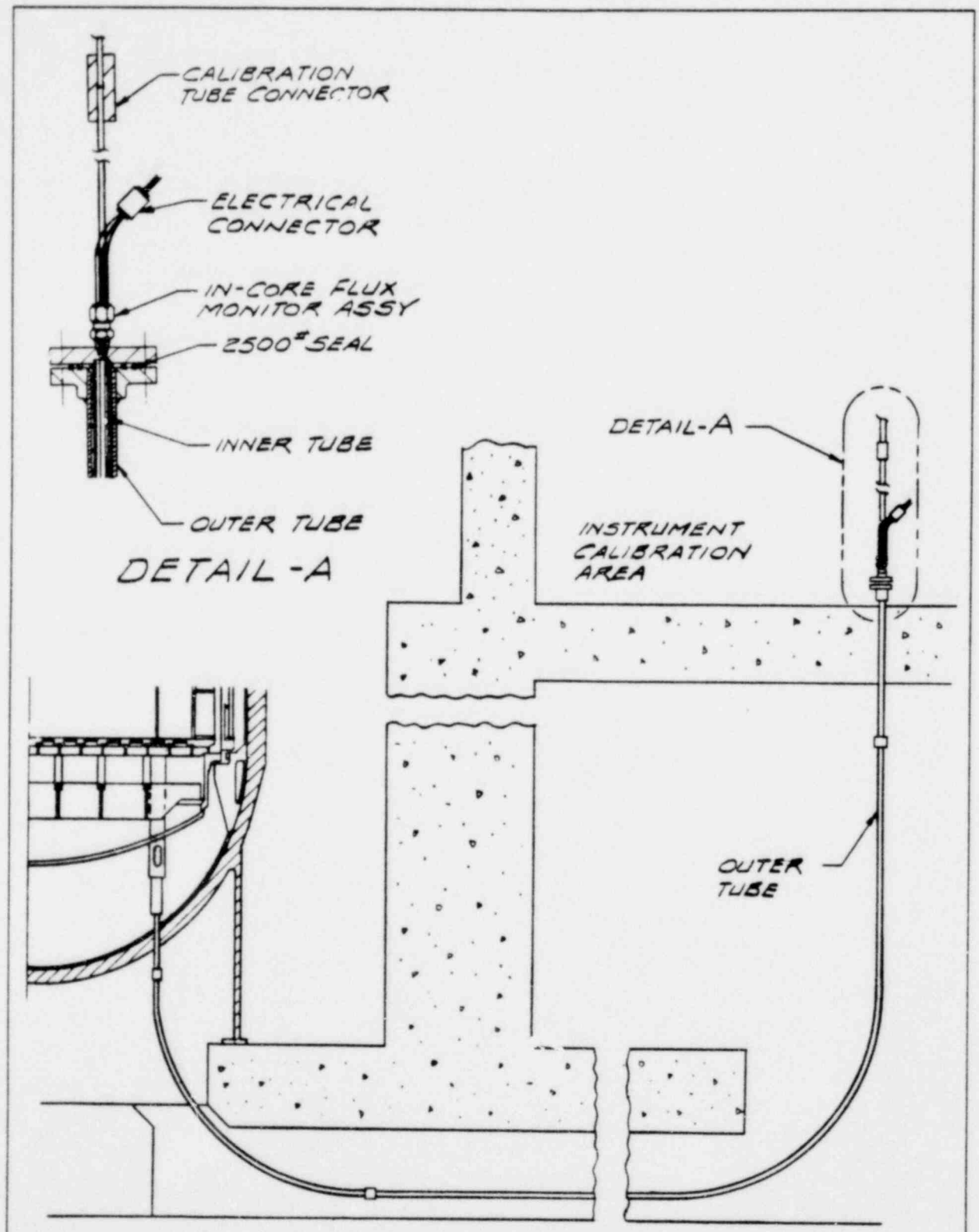
O INCORE MONITOR ASSEMBLY



INCORE DETECTOR LOCATIONS

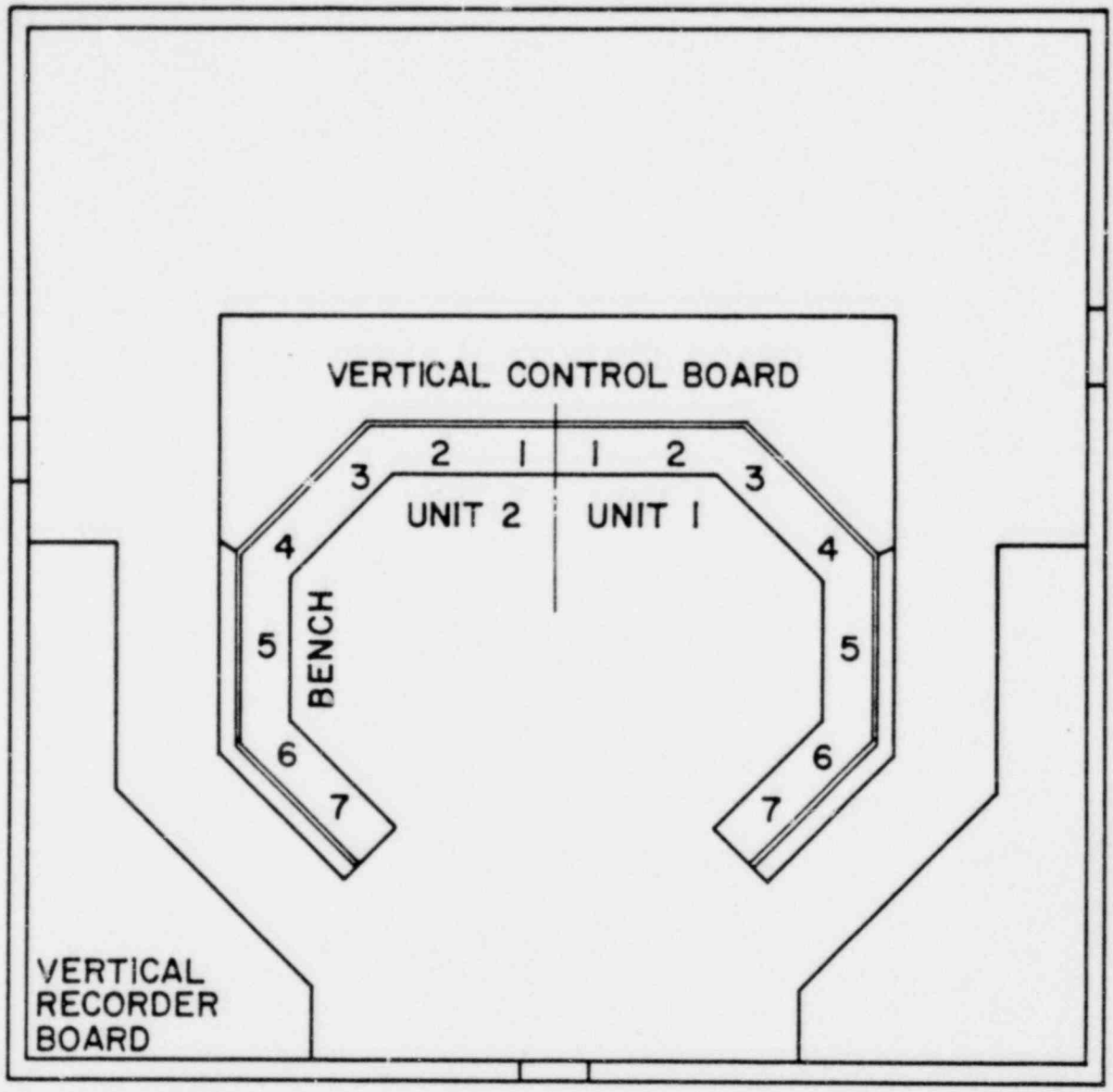
OCONEE NUCLEAR STATION

FIGURE 7-12



OCONEE NUCLEAR STATION

FIGURE 7-13



OCONEE NUCLEAR STATION
CONTROL BOARD LAYOUT

OCONEE NUCLEAR STATION
FIGURE 7-14