7 INSTRUMENT AND CONTROL

0118

8003210 664

TABLE OF CONTENTS

0119

CONTENTS (Cont'd)

CONTENTS (Cont'd)

Section		Page

LIST OF FIGURES

(At rear of Section)

7

0122

7 INSTRUMENTATION AND CONTROL

7.1 PROTECTION SYSTEMS

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

7.1.1 DESIGN BASES

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

7.1.1.1 Vital Functions

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

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

b. Loss of all reactor coolant pumps.

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

d. The reactor pressure reaches an established minimum limit.

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

a. The reactor pressure reaches an established maximum limit.

The Safeguards Actuation System automatically performs the following vital functions:

a. Commands operation of injection emergency core coolant.

b. Commands operation of the reactor building emergency cooling system and the reactor building spray system.

c. Commands closing of the reactor building isolation valves.

0123

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

7.1.1.1.1 Nonvital Functions

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

7.1.1.2 Principles of Design

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

The major design criteria are summarized in the following paragraphs.

7.1.1.2.1 Single Failure

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

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

7.1.1.2.2 Redundancy

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

7.1.1.2.3 Independence

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

Separate detectors and instrument strings are not, in general, employed for protection system functions and regulation or control. Sharing instrumentation for protection and control functions is accomplished within the framework of the stated criteria by the employment of isolation amplifiers in each of the multiple outputs of the analog protection system instrument strings.

The isolation amplifiers are precision operational amplifiers having a closed loop unity gain and a low dynamic output impedance. The effectiveness of the isolation amplifiers has been proven by actual test. 3

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

This may be stated as a corollary to the design criteria: "a direct short, open circuit, ground fault, faulting to a power source of less than 410 volts, or bridging of any two points at the output terminals of a protection system analog instrument string having multiple outputs shall not result in a significant disturbance within more than one output." 3

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

7.1.1.2.4 Loss of Power

a. A loss of power in the Reactor Protection System shall cause the affected channel to trip.

0125

b. Availability of power to the Safeguards Actuation System shall be continuously indicated. The loss of instrument power, i.e., 120v a-c Essential Services bus power, to the instrument strings and bistables will initiate a trip in the affected channels. System actuation requires control power from one of the two engineered safeguards dc power busses so that loss of this power does not actuate the system. The system equipment is divided between the redundant engineered safeguards channels in such a way that the loss of one of the dc power busses does not inhibit the system's intended safeguards functions.

7.1.1.2.5 Manual Trip

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

7.1.1.2.6 Equipment Removal

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

7.1.1.2.7 Testing

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

a. Preoperational testing to give assurance that the protection systems can fulfill their required functions.

b. On-line testing to prove operability and to demonstrate reliability.

7.1.1.3 <u>Functional Requirements</u>

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

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

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

b. The loss of all reactor coolant pumps.

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

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

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

0126

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

Interlocking functions of the Reactor Protection System are to

a. Bypass the startup rate trip when the reactor power reaches a preset value.

b. Inhibit control rod withdrawal on the occurrence of a predetermined startup rate, slower than the rate at which reactor trip is initiated.

The functional requirements of the Safeguards Actuation System are to

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

b. Start operation of low pressure injection upon detection of a very low reactor coolant system pressure.

c. Operate the reactor building isolation valves upon detection of a moderately high reactor building pressure.

d. Start the reactor building emergency cooling units upon detection of a moderately high reactor building pressure.

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

7.1.1.4 Environmental Considerations

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

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

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

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

7.1.2 SYSTEM DESIGN

7.1.2.1 System Description - Reactor Protection System

The system as shown in Figures 7-1 and 7-2b consists of four identical protective channels. Each channel controls one RS relay in four identical 2 out of 4 logics. Each channel forms an AND gate to energize its respective RS relays. Should any one or more inputs to a given channel initiate a trip the RS relays associated with that channel will de-energize. Thus, from a trip standpoint, each channel is an OR gate. Should any two of the four protection channels de-energize their respective RS relays all four of the 2 out of 4 logics will open, tripping the control rod drive system circuit breakers which in turn removes power from the drive motors permitting the rods to move into the core shutting down the reactor. 7

The control rod drive circuit breakers, Figure 3-66, form a logic which is just short of a full 2-out-of-4 coincidence. The specified breaker combinations which initiate a reactor trip can best be stated in logic notation as: AB + ADF + BCE + CDEF. This is a 1 out of 2 logic used twice and is referred to as a 1-out-of-2 X 2 logic. It should be noted that when any 2-out-of-4 protective channels trip, all 2-out-of-4 logics trip, commanding all control rod drive breakers to trip.

The undervoltage coils of the control rod drive breakers receive their power from the protective channel associated with each breaker. The manual reactor trip switch is interposed in series with each 2-out-of-4 logic and the assigned breakers undervoltage coil.

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

The power/flow monitor logic details are shown on Figure 7-2b. There are four identical sets of power/flow monitor logic, one associated with each protection channel. Each set of logic received an independent total reactor coolant flow signal (ΣF), a "number of pump motors in operation" signal, and two isolated reactor power level signals (ϕ).

0128

Paragraphs Deleted. 7

Using a flux/flow comparator, one part of the power/flow monitor continuously compares the ratio of the reactor neutron power to the total reactor coolant flow. Should the reactor power as measured by the linear power range channels exceed 1.075 times the reactor coolant flow, a reactor trip is initiated. All measurements are in terms of per cent full flow or full (rated) power. The flux/flow comparator runs back the over flux trip level in step with a detected decreasing flow thus providing an opportunity for the control system to reduce the reactor power to an acceptable level without a reactor trip. 3

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

7.1.2.2 Description - Safeguards Actuation System

Figure 7-2c shows the action initiating sensors, bistables, and logic for the Safeguards Actuation System. The major differences between this system and the Reactor Protection System are: 3

a. Each protective action is initiated by either of two channels with 2-out-of-3 coincidence logic between input signals.

b. Either of the two channels is independently capable of initiating the desired protective action through redundant safeguards equipment.

c. Protective action is initiated by the application of power to the terminating control relays through the coincidence logic.

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

0129

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 Plant operator as well as automatic actuation. Normal starting and stopping are initiated by momentary contact pushbuttons or control switches.

The control circuit shown for a decay heat removal pump is typical of the con- 5
troller of a large pump started by switchgear. There are two decay heat removal pumps; they are equipped with single control relays powered from separate safeguards actuation channels.

channels. Energizing the control relays through their associated safeguards actuation channel, energizes the pump circuit breaker closing coil and starts the pump.

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

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

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

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.

7.1.2.3 Design Features

7.1.2.3.1 Redundancy

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

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

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

0131

7.1.2.3.2 Independence

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

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

The Safeguards Actuation System instrumentation and control have electrically and physically independent instrument strings. The output of each bistable is electrically independent of every other bistable. Independence is preserved in the coincidence networks through insulation resistance and physical separation of the switching elements.

7.1.2.3.3 Loss of Power

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

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

7.1.2.3.4 Manual System Trip

The manual actuating devices in the protection systems are independent of the automatic trip circuitry and are not subject to failures that make the automatic circuitry inoperable. The manual trip devices are independent control switches for each power controller. The independent control switches, however, are all actuated through a mechanical linkage to a common manual trip switch or pushbutton. 1

7.1.2.3.5 Equipment Removal

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

An inherent characteristic is illustrated by considering the power supply for one of the reactor 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 Safeguards Actuation System provides for servicing without affecting the integrity of the redundant channels.

7.1.2.3.6 Testing

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

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

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

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

As shown in Figure 7-2b and 3-66, 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

7.1.2.3.7 Physical Isolation

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

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

Wiring between vital elements of the system outside of equipment housing will be routed and protected within the unit so as to maintain 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 Protection System is the 120v a-c Essential Services busses described in 8.2.2.7. The source of power for the measuring elements in the Safeguards Actuation System is also from the 120v a-c Essential Services busses. Command circuits from the Safeguards Actuation System coincidence logic that extend to Engineered Safeguards Equipment controllers are powered from the Engineered Safeguards d-c busses. Engineered Safeguards equipment, such as pump and motor operators and their starting contactors, are powered from the Engineered Safeguards a-c busses.

7.1.2.3.9 Reliability

Design criteria for the Reactor Protection System and the Safeguards Actuation System have been formulated to produce reliable systems. System design practices, such as redundant equipment, redundant channels, and coincidence arrangements permitting in-service testing, have been employed to implement the 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's vital functions. Acceptance testing and periodic testing will be designed to insure the quality and reliability of the completed systems.

7.1.2.3.10 Instrumentation for Emergency Core Cooling Initiation

The instrumentation system makes use of both physical and electrical isolation.
The high pressure and low pressure injection systems are activated by both low 3
reactor coolant and reactor building pressure signals originating from three pressure transmitters measuring the reactor coolant system pressure, as shown in Figure 7-11, and three pressure transmitters measuring the reactor building pressure.

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

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

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

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

0134

7.1.2.4 Summary of Protective Actions

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

Trip Variable	No. of Sensors	Steady State Normal Range	Trip Value or Condition for Trip	
Neutron Flux	4	0-100%	107.5% of full (rated) power	
Neutron Flux/ Reactor Coolant Flow	4 Flux 16 Reactor Coolant Pump Monitors 2 Flow Tubes	3 to 4 pumps	(1) Ratio of reactor power to total reactor coolant flow exceeds 1.075. (2) More than one reactor coolant pump motor is lost.	3
Startup Rate	2	0-2 Decades/min	5 Decades/min	
Reactor Coolant Pressure	4	2,120-2,250 psig	2,350 psig 2,050 psig	
Reactor Outlet Temperature	4	520-603 F	610 F	

0135

Actions initiated by the Safeguards Actuation System are as follows:

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

7.1.2.5 Relationship to Safety Limits

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

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

0136

7.1.3 SYSTEMS EVALUATION

7.1.3.1 Functional Capability - Reactor Protection System

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

a. Temperature - 5 sec.

b. Pressure - 0.5 sec.

c. Flux - 0.3 sec.

d. Pump monitor - 1.0 sec.

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

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

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

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

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

Two major measurements feed the power/flow monitor: (a) reactor coolant flow,
and (b) neutron power level. The flow tubes which provide the reactor coolant 3
flow measurement will exhibit no change during the reactor life. A periodic calibration of the flow transmitters will be made. The neutron power level signal will be recalibrated by comparison with a routine heat balance. The power range channels use detectors arranged to effectively average the measurement over the length of the core as described in 7.3.1.1.2. Therefore, their output is expected to be within 4 per cent of the calibrated value during normal regulating rod group position changes and the need for additional calibration thereby eliminated.

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

7.1.3.2 Functional Capability - Safeguards Actuation System

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

The key variable associated with the loss of reactor coolant is reactor pressure. In a loss-of-reactor-coolant accident, the reactor pressure will fall, starting high pressure injection at 1,800 psig. If high pressure injection does not arrest the pressure drop, then low pressure injection starts upon a signal of 200 psig. The injection systems are initiated by a detected reactor building high pressure trip of 10 psig acting in parallel with the reactor coolant low pressure trips.

0138

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

7.1.3.3 Preoperational Tests

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

Routine preoperational tests will be performed by the substitution of a calibrating signal for the sensor. Simulated neutron signals may be substituted in each of the source, intermediate, and power range channels to check the operation of each channel. Simulated pressure, temperature, and level signals may be used in a similar fashion. This type of testing is valid for all elements of the system except the sensors. The sensors should be calibrated against standards during shutdowns for refueling, or whenever the true status of any measured variable cannot be assessed because of lack of agreement among the redundant measurements.

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

7.1.3.4 Component Failure Considerations

The effects of failure can be understood through Figure 7-2B. In the Reactor Protection System, the failure of any single input in the "tripped" direction places the system in a 1-out-of-3 mode of operation for all variables. Failure of any single input in the "cannot trip" direction places the system in a 2-out-of-3 mode of operation for the variable involved, but leaves all other variables in the normal 2-out-of-4 coincidence mode. If the fault were in 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 "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 Safeguards Actuation System is a 2-out-of-3 input type of system. It can tolerate one fault of the "cannot trip" variety in each of the coincidence networks. For this type of fault, all remaining inputs must function correctly. A "tripped" input fault allows any one of the two remaining inputs to initiate action.

Primary power input to both protection systems has been arranged to minimize the possibility of loss of power to either protection system. Each cahnnel of the protection system will be supplied from one of the four 120v a-c Essential Services busses described in 8.2.2.7. The operator can initiate a reactor trip independent of the automatic protection action.

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

7.1.3.5 Operational Tests

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

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

0140

7.2 REGULATING SYSTEMS

7.2.1 DESIGN BASES

Reactor output is regulated by the use of movable control rod assemblies and 7
soluble boron dissolved in the coolant. Control of relatively fast reactivity effects including Doppler, xenon, and moderator temperature effects, is accomplished by the control rods. The control response speed is designed to overcome these reactivity effects. Relatively slow reactivity effects, such as fuel burnup, fission product buildup, samarium buildup, and hot-to-cold moderator reactivity deficit, are controlled by soluble boron.

Control rods are normally used for control of xenon transients associated with normal reactor power changes. Chemical shim shall be used in conjunction with control rods to compensate for equilibrium xenon conditions. Reactivity control may be exchanged between rods and soluble boron consistent with limitations on power peaking.

Reactor regulation is a composite function of the Integrated Control System and Rod Drive Control System. Design data for these subsystems are given in the following sections.

7.2.2 ROD DRIVE CONTROL SYSTEM

The rod drive control system (RDC) includes drive controls, power supplies, position indicators, operating panels and indicators, safety devices, and enclosures.

7.2.2.1 Design Basis

The rod drive control system design bases are catagorized into safety considerations, reactivity rate limits, startup considerations and operational considerations.

7.2.2.1.1 Safety Considerations

a. The control rod assemblies (CRA) shall be inserted into the core upon receipt of protective system trip signals. Trip command has priority over all other commands.

b. No single failure shall inhibit the protective action of the rod drive control system.

7.2.2.1.2 Reactivity Rate Limits

The speed of the mechanism and group rod worth provide the reactivity change rates required. For design purposes the maximum rate of change of reactivity that can be inserted by any group of rods has been set at 1.1×10^{-4} $\Delta K/K/s$. The drive controls, i.e., the drive mechanims and rods combination, have an inherent speed-limiting feature.

7.2.2.1.3 Startup Considerations

The rod drive control system design bases for startup are as follows:

a. Reactor regulation during startup shall be a manual operation.

b. Control rod "out" motion shall be inhibited when a high startup rate (short period) in the source range or intermediate range is detected.

7.2.2.1.4 Operational Considerations

For operation of the reactor, functional criteria related to the rod drive control system are:

a. CRA Positioning

The rod drive control system provides for controlled withdrawal, controlled insertion and holding of the control rod assemblies (CRAs), to establish and maintain the power level required for a given reactor coolant boron concentration.

b. Position Indication

Continuous rod position indication, as well as full-in and full-out position indication, shall be provided for each control rod drive.

c. System Monitoring

The rod drive control system design includes provisions for routinely monitoring conditions that are important to safety and reliability.

7.2.2.2 System Design

The rod drive control system provides for withdrawal and insertion of the control rod assemblies to maintain the desired reactor output. This is achieved either through automatic control by the Integrated Control System discussed in Section 7.2.3, or through manual control by the operator. As noted previously, this control compensates for short term reactivity changes. It is achieved through the positioning in the core of sixty-one control rod assemblies and eight axial power-shaping rod assemblies. The sixty-one rods are grouped for control and safety purposes into seven groups. Four groups function as safety rods, and three groups serve as regulating rods. An eighth group serves to regulate axial power peaking due to xenon poisoning. Seven of the eight groups may be assigned from four to twelve control rod assemblies. Eight rod assemblies are used in group eight.

Control rods are arranged into groups at the control rod drive control system patch panel. Typically, twenty-eight rods, including the axial power shpaing rods, are assigned to the regulating groups, and forty-one rods are assigned to the safety rod groups. A typical rod grouping arrangement is shown below:

Safety Rods	Regulating Rods	Axial Power Shaping Rods
Group 1 - 8	Group 5 - 12	Group 8 - 8
Group 2 - 12	Group 6 - 4	
Group 3 - 9	Group 7 - 4	
Group 4 - 12		

During startup the safety rod groups are withdrawn first, enabling withdrawal of the regulating control groups. The sequence allows operation of only one regulating rod group at a time except where reactivity insertion rates are low (first and last 25% of stroke), at which time two adjacent groups are operated simultaneously in overlapped fashion. These insertion rates are shown in Figure 7-7.

As fuel is used, dilution of soluble boron in the reactor coolant is necessary. When Group 6 is more than 95% withdrawn, interlocks permit dilution. The reactor controls insert Group 6 to compensate for the reduction in boron concentration by dilution. The dilution is automatically terminated by a pre-set volume measuring device. Interlocks are also provided on Group 6 rod position to terminate dilution at a pre-set insert limit.

7.2.2.2.` System Equipment

The rod drive control system consists of three basic components: (1) control rod drive motor power supplies; (2) system logic; (3) trip breakers. The power supplies consist of four group power supplies, an auxiliary power supply, and two holding power supplies. The group power supplies are of a redundant six-phase half-wave rectifier design. In each half of a group power supply, rectification and switching of power is accomplished through the use of Silicon Controlled Rectifiers (SCRs). This switching sequencially energizes first two, then three, then two of the six CRA motor stator windings in stepping motor fashion, to produce a rotating magnetic field for the control rod assembly motor to position the CRA. Switching is achieved by gating the six SCRs on for the period each winding must be energized. As each of the six windings utilize SCRs to supply power, six gating signals are required.

Gating signals for the group power supplies are generated by a motor driven programmer, consisting of a 60-cycle, reversible synchronous motor driving a multichannel photo-optic encoder. The coded light beam excites photo-detectors, generating signals which are amplified to form the Silicon Controlled Rectifier gating signals. The programmer is redundant (except for motor and gears), thus providing separate but synchronized gating signals to the dual power supply units. Command signals to position the control rod drive are introduced at the programmer motor.

Identical power supplies are used for the regulating (control) groups and for the auxiliary power supply. Each half of each group power supply is capable of driving up to 12 drive mechanisms--the maximum number that may be in any one group. The power supplies have dual power inputs, each half fed from separate power sources and each half being capable of carrying the full load.

Unlike the control group power supplies, the holding power supply is used to maintain the safety rods fully withdrawn; consequently, switching is not required. A six-phase d-c power supply is used for this purpose. Two holding power supplies are provided. Each is rated to furnish power to one winding of 48 mechanisms. 7

The auxiliary power supply is used to position the safety rod groups and to provide single rod control. The safety rod groups are maneuvered with the auxiliary power supply, and then, when fully positioned, are transferred to the holding busses described above. After positioning the safety rods, the auxiliary power supply is available to the regulating groups, through transfer relays, to serve either as a single rod controller, should repositioning of a single rod be necessary, or, as a spare group controller, should one of the group control power supplies require maintenance. The system logic encompasses those functions which command control rod motion in the manual or automatic modes of operation, including CRD sequencing, safety and protection features, and the manual trip function. Major components of the logic system are the Operator's Control Panel, CRA position indication panels, automatic sequencer, and relay logic.

Switches are provided at the operators control panel for selection of the desired rod control mode. Control modes are: (1) Automatic mode--where rod motion is commanded by the Integrated Control System; and (2) Manual mode--where rod motion is commanded by the operator. Manual control permits operation of a single rod or a group of rods. Alarm lamps on the RDC panel alert the operator to the systems status at all times.

The sequence section of the logic system utilizes rod position signals to generate control interlocks which regulate rod group withdrawal and insertion. The sequencer operates in both automatic and manual modes of reactor control, and controls the regulating groups only. Analog position signals are generated by the reed switch matrix on the CRA, and an average group position is generated by an averaging network. This average singal serves as input to electronic set point trip units which are activated at approximately 25 and at 75 per cent of group rod withdrawal. Two bistable units are provided for each regulating group. Outputs of these bistables actuate "enable" relays which permit the rod groups to be commanded in automatic or manual mode.

In addition to the sequencer, relay logic monitors are provided in the "enable" circuits which prohibit out of sequence conditions. The selection of manual control mode or sequence bypass mode functions permit intentional out-of-sequence conditions. This condition is indicated to the operator.

Inputs to the system logic from the Reactor Protection System and the Integrated Control System provide interlock control over rod motion. These interlocks cause rod motion command lines and control mode selection to be inhibited.

In the rod drive control system, two methods of position indication are provided; an absolute position indicator and a relative position indicator. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity. As the lead screw (and the control rod assembly) moves, the switches operate sequentially, producing an analog voltage proportional to position. The accuracy of the analog signal is approximately ±1.1 per cent of full scale (139 in.) and the readout has approximately ±2.1 per cent of full scale accuracy. Other reed switches included in the same tube with the position indicator matrix provide full-in and full-out limit indications. 7

The relative position transducer is a small pulse-stepping motor, driven from the power supply for the rod drive motor. This small motor is coupled to a potentiometer with an output signal accuracy of ±0.7 per cent of full scale position, producing a readout with an accuracy of ±1.7 per cent of full scale.

Rod drive control system trip breakers are provided to interrupt power to the control rod assembly motors. When power is removed, the roller nuts disengage from the lead screw and a gravity free-fall trip of the CRA occurs. Two series trip methods are provided for removal of power to the CRA motors. First, a trip is initiated when Reactor Protection System logic interrupts power to the undervoltage (UV) coil of the main AC feeder breaker. Secondly, a trip is initiated when the Silicon Control Rectifier gating power and the DC holding power is interrupted. As parallel power feeds are provided on both holding and gating power, interruption of both feeds is required for trip action in either method of trip. Trip circuitry is shown in Figure 3-66.

AC power feed breakers are of the three-pole, stored-energy type, and are equipped with instantaneous undervoltage trip coils. Each AC feed breaker is housed in a separate metal-clad enclosure. The secondary trip breakers are also of the stored-energy type with two parallel-connected instantaneous undersoltage trip coils consisteing of two 2-pole breakers mechanically ganged to interrupt DC busses. All breakers are motor-driven-reset to provide remote reset capability. Each undervoltage trip coil is operated from the Reactor Protection System.

7.2.2.3 System Evaluation

7.2.2.3.1 Safety Considerations

A reactor trip occurs whenever power has been removed from the rod drive motors. The design provides two stored energy breakers which do not require power to interrupt the electrical feeds to rod drive control power supplies, and a second set of circuit-interrupting devices in series on the output of the power supplies. All devices have interrupting capacity of sufficient rating to open under any group load configuration. Reactor trip is further assured by providing series trip devices, split buses and provisions for periodic testing. Trip redundancy is provided by series breakers while availability and testability are provided through dual power sources. Redundant power supplies permit testing of the trip action of each power-interrupting device without loss of plant availability.

Reactivity shutdown margin provided by the safety rods is assured by diversification of their power buses. This feature, as shown in Figure 7-1, utilizes four separate buses, each having a separate trip device, to power the safety rods. A failure in one bus does not reflect into the other buses, therefore, a single failure in the distribution system for the safety rods does not prevent a plant shutdown.

In summary, series redundant trip devices having adequate rating, testability and a "split bus" arrangement insure safety of reactor trip circuits. 7

7.2.2.3.2 Reactivity Rate Limits

The desired rate of change of CRA reactivity insertion and uniform reactivity distribution over the core are provided for by the control rod drive and power supply design, and the selection of rods in a group. The motor, lead screw and power supply designs are fixed to provide a uniform rate of speed of 30 in./min. The reactivity change is then controlled by the rod group size. To insure flexibility in this area, a patch panel has been included in the power supply to enable the interchange of rod worth between rod groups. Any rod may be patched into any group with the exception of Group 8.

Uniform and symmetrical group insertion rate is provided for by synchronous withdrawal of all rods in that group. Such synchronous withdrawal is achieved by the design of the power supply. A group power supply operates synchronously by having its load (4 to 12 CRA motor windings) connected in parallel on the output of the SCR's. As the programmer gates on the SCR's, all rods in a group have the same motor winding energized simultaneously, producing synchronous motion of the entire group.

Monitors are provided to sense asymmetric rod patterns. These monitors alarm the condition to the operator, computer, and the ICS. Depending upon the power setting, action is initiated by the ICS to insert rods and reduce power.

7.2.2.3.3 Startup Considerations

The rod drive controls receive interlock signals from the ICS and nuclear instrumentation (NI). These inputs are used to inhibit automatic mode selection below 15% rated power and to inhibit out motion for high startup rates, respectively.

In addition to the startup considerations, dilution controls, to permit removal of reactor shutdown concentrations of boron in the reactor coolant, are provided. This control bypasses the normal reactor coolant dilution controls, described in Section 7.2.2.2, providing all safety rods are withdrawn from the core and the operator initiates a continuous feed and bleed cycle. An additional interlock on rod Group 5 inhibits the use of this circuit when rod Group 5 is more than 80% withdrawn.

7.2.2.3.4 Operational Considerations

The control rod assembly positioning system provides the ability to move any rod to any position required consistent with reactor safety. As noted in Section 7.2.2.3.2 a uniform speed is provided by the drive system. A fixed rod position when motion is not required is obtained by the power supply ability to energize two adjacent windings of the CRA motor stator. This static energizing of the windings maintain a latched stator and fixed rod position.

Position Indication

As previously described, two separate position indication signals are provided. The absolute position sensing system produces signals proportional to CRA position from the reed switch matrix located on each CRA mechanism. The relative position indicat on system produces a signal proportional to the number of CRA motor power pulses from a stepping motor and precision potentiometer for each CRA mechanism. 7

Position indicating readout devices mounted on the operator's console consist of 69 single rod position meters and 4 control group average position meters. The operation of a selector switch permits either relative or absolute position information to be displayed on the single rod meters.

The control-group-average meters display the arithmetic average of the relative position signals of all CRA's in a group. A selector switch on the operator's console permits the group meters to display either the positions of all safety rod groups (Groups 1-4) or the positions of all regulating rod groups (Groups 5-7) and the axial power shaping rod groups (Group 8).

Indicator lights are provided on the single-rod meter panel to indicate when each rod is; (1) fully inserted, (2) fully withdrawn, (3) under control and (4) whether a fault is present. Indicators on the operator's console show full insertion, full withdrawal, under-control and fault indication for each of the eight control rod groups.

Failures which could result in unplanned control rod withdrawal are continuously monitored by fault detection circuits. When failures are detected, indicator lights and alarms alert the operator. Fault indicator lights remain on until the fault condition is cleared by the operator. A list of indicated faults is shown below:

(1) Asymmetric rod patterns.
(2) Motor rotation faults.
(3) Sequence faults.
(4) Rod position sensor faults.
(5) Trip faults.
(6) Safety rods not withdrawn.
(7) Programmer lamp faults.

7.2.3 INTEGRATED CONTROL SYSTEM

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

Figure 7-6 shows the reactor control portion of the Integrated Control System described in 7.2.3.3. Figure 7-8 shows the steam generator and turbine control portion of the Integrated Control System. This control receives inputs of megawatt demand, system frequency, and steam pressure, and supplies output signals to the turbine bypass valve, turbine speed changer, and steam generator feedwater flow controls with changing operating conditions. 7

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

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

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

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

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

a. One or more reactor coolant pumps are inoperative.

b. Total feedwater flow lags total feedwater demand by more than 5 per cent.

c. Assymetric rod withdrawal patterns exist.

d. The generator separates from the 500 kv bus.

The output of the limiters is a megawatt demand signal which is applied to the turbine control, steam generator control, and reactor control in parallel. The reactor control responds to the megawatt demand signal as described in 7.2.3.3.

7.2.3.1 Turbine Control

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

7.2.3.2 Steam Generator Control

Control of the steam generator is based on matching feedwater flow to megawatt demand with bias provided by the error between steam pressure set point and steam pressure. The pressure error increases the feedwater flow demand if the pressure is low. It decreases the feedwater flow demand if the pressure is high. 7

The basic control actions for parallel steam generator operation are

a. Megawatt demand converted to feedwater demand.

b. Steam pressure compared to set pressure, and the pressure error converted to feedwater demand.

c. Total feedwater demand computed from sum of a and b.

d. Total feedwater flow demand split into feedwater flow demand for each steam generator.

e. Feedwater demand compared to feedwater flow for each steam generator. The resulting error signals position the feedwater flow controls to match feedwater flow to feedwater demand for each steam generator.

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

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

a. Steam Generator Load Ratio Control

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

b. Rate Limits

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

c. Water Level Limits

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

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

d. Reactor Coolant Pump Limiters

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

7

e. Reactor Outlet and Feedwater Low Temperature Limits

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

f. Feedwater Pump Capability

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

7.2.3.3 Reactor Control

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

First, reactor power level demand (N_d) is computed as a function of the megawatt demand (MW_d) and the reactor coolant system average temperature deviation ($\overline{\Delta T}$) 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 setttings, the control rod drive receives a command that withdraws or inserts rods depending upon the polarity of the power error signal.

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

a. An adjustable low limit on the megawatt demand signal (MW_d) to cut out the automatic reactor control action.

b. A high limit on reactor power level demand (N_d).

c. An adjustable low limit on reactor power level demand (N_d).

Separate from, but related to, the automatic reactor control system is the reactor coolant flow signal 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 Control System. 7

7.2.3.4 System Failure Considerations

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

Manual reactivity control is available at all power levels.

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

7.2.3.5 Interlocking

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

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

Control circuitry allows manually selected operation of any single control rod 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.

7.2.3.6 Loss-of-Load Considerations

The nuclear unit is designed to accept 10 per cent step load rejection without safety valve action or turbine bypass valve action. The combined actions of the control system and the turbine bypass valve permit a 40 per cent load reduction or a turbine trip from 40 per cent load without safety valve action. The controls will limit steam dump to the condenser when condenser vacuum is inadequate, in which case the safety valves may operate. The combined actions of the control system, the turbine bypass valve, and the safety valves permit a 100 per cent load rejection without turbine trip. This permits the unit to ride through a "blackout" condition, i.e., sudden rejection of electrical load down to auxiliary load without turbine trip. (The "blackout" provisions are discussed in 14.1.2.8.2.) 7

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

a. Integrated Control System

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

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

b. 100 Per Cent Relief Capacity in the Steam System

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

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

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

0152

7.3 INSTRUMENTATION

7.3.1 NUCLEAR INSTRUMENTATION

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

7.3.1.1 Design

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

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

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

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

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

a. Independent power supplies are included in each channel. Primary power originates from the 120v a-c Essential Services busses described

0153

in 8.2.2.7. Where applicable, isolation transformers are provided to insure a stable, high-quality power supply.

b. The proportional counters used in the source range are designed to be secured when the flux level is greater than their useful operating range. This is necessary to obtain prolonged operating life.

c. The intermediate range channels are supplied with an adjustable source of gamma-compensating voltage.

7.3.1.1.1 Test and Calibration

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

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

7.3.1.1.2 Power Range Detectors

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

7.3.1.1.3 Detector Locations

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

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

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

7.3.1.2 Evaluation

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

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

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

7.3.1.2.1 Loss of Power

The nuclear instrumentation draws its primary power from redundant battery-backed 120v a-c Essential Services busses described in 8.2.2.7.

7.3.1.2.2 Reliability and Component Failure

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

7.3.1.2.3 Protection Requirements

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

7.3.2 NONNUCLEAR PROCESS INSTRUMENTATION

7.3.2.1 System Design

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

a. In general, resistance elements are used for temperature measurements. Fast-response resistance elements monitor the reactor outlet temperature. The outputs of these fast-response elements supply signals to the protective system.

b. Pressures are measured in the reactor coolant system, the steam system, and the reactor auxiliary systems. Pressure signals for high and low reactor coolant pressures and high reactor building pressure are provided to the protection systems.

c. Reactor coolant pump motor operation is monitored as an indication of reactor coolant flow. This information is fed to the reactor controls and reactor protection system. In addition, reactor coolant flow signals are obtained and indicated by continuous measurement of the pressure drops across the reactor coolant side of each steam generator.

d. Flow in the steam system is obtained through the use of calibrated feedwater flow nozzles. Flow information is utilized for control and protective functions in the steam system. Steam generator level measurements are provided for control and alarm functions.

e. Pressurizer level is measured by differential pressure transmitters calibrated to operating temperature and pressure. The pressurizer level is a function of the reactor coolant system makeup and letdown flow rate. The letdown flow rate is remote manually controlled to the required flow. Pressurizer level signals are processed in a level controller whose output positions the makeup control valve in the makeup line to maintain a constant level.

f. Reactor coolant system pressure is maintained by a control system that energizes pressurizer electrical heaters in banks at preset pressure values below 2,175 psig or actuates spray control valves if the pressure increases to 2,230 psig.

7.3.2.2 System Evaluation

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

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

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

All electrical and electronic instrumentation required for safe and reliable operation will be supplied from redundant 120v a-c Essential Services busses.

7.3.3 INCORE MONITORING SYSTEM

7.3.3.1 Design Basis

The incore monitoring system provides neutron flux detectors to monitor core performance. No protective action or direct control functions are performed by this system. All high pressure system connections are terminated within the reactor building. Incore, self-powered neutron detectors measure the neutron flux in the core to provide a history of power distributions and disturbances during power operating modes. Data obtained will provide measured power distribution information and fuel burnup data to assist in fuel management decisions. 1

7.3.3.2 System Design

7.3.3.2.1 System Description

The incore monitoring system consists of assemblies of self-powered neutron detectors and calibration tubes located at 52 preselected radial positions within the core. The incore monitoring locations are shown on Figure 7-12. In this arrangement, an incore detector assembly, consisting of seven local flux detectors, one background detector, and a calibration tube, is installed in the instrumentation tube of each of 52 fuel assemblies (Figure 3-62). The local detectors are positioned at seven 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. 1

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

Readout for the incore detectors is performed by the unit computer 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 instrumentation tubes of 52 selected fuel assemblies. The instrumentation tube then serves as the guide for the incore detector assembly. The incore detector assemblies are fully withdrawn only for replacement. During refueling operations, the incore detector assemblies are withdrawn approximately 13 feet to allow free transfer of the fuel assemblies. After the fuel assemblies are placed in their new locations, the incore detector assemblies are returned to their fully inserted positions in the core, and the high pressure seals are secured.

7.3.3.2.2 Calibration Techniques

The nature of the detectors permits the manufacture of nearly identical detectors which will produce a high relative accuracy between individual detectors. The detector signals must be compensated for burnup of the neutron sensitive material. The data handling system integrates each detector output current and generates a burnup correction factor to be applied to each detector signal before printing out the corrected signal in terms of per cent of 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.

7.3.3.3 System Evaluation

7.3.3.3.1 Operating Experience

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

7.3.3.3.2 B&W Experience

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

The B&W Development Program includes these tests:

a. Parametric studies of the self-powered detector.

b. Detector ability to withstand PWR environment.

c. Multiple detector assembly irradiation tests.

d. Background effects.

e. Readout system tests.

f. Mechanical withdrawal-insertion tests.

g. Mechanical high pressure seal tests.

h. Relationship of flux measurement to power distribution experiments.

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

a. The detector sensitivity, resistivity, and temperature effects are satisfactory for use.

b. A multiple detector assembly can provide axial flux data in a single channel and can withstand reactor environment. An assembly of six local flux detectors, three background detectors, and two thermocouples has been successfully operating in the Big Rock Point Reactor since May 1966.

c. Data collection systems are successful as read-out systems for incore monitors.

d. Background effects will not prevent satisfactory operation in a PWR environment.

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

7.4 OPERATING CONTROL STATIONS

Following proven power station design philosophy, all control stations, switches, controllers, and indicators necessary to start up, operate, and shut down each nuclear unit 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 power generation control or emergency functions.

7.4.1 GENERAL LAYOUT

The control room will be designed so that one man can supervise operation of the Plant during normal steady-state conditions. During other than normal operating conditions, other operators will be available to assist the control operator. The control room will be arranged to include operating benchboard cubicles to house frequently used and emergency indicators and controllers at close proximity and visibility to the operator. Vertical panel sections of the cubicles will house less frequently used controllers and informational displays.

7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTION

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

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

7.4.3 SUMMARY OF ALARMS

Visible and audible alarm units will be incorporated into the control room to warn the operator if unsafe conditions are approached by any system. Audible reactor building evacuation alarms are to be initiated from the radiation monitoring system or manually by the operator. Audible alarms will be sounded in appropriate areas throughout the Plant if high radiation conditions are present. | 1

0159

7.4.4 COMMUNICATION

Independent Plant telephone and paging systems will be furnished to provide the control room operator with constant "hands free" communication with all areas of the Plant. Acoustical phones will be supplied in areas where the background noise level is high. Communication outside the Plant will be through the full period leased lines of the Florida Telephone Company. 1

7.4.5 OCCUPANCY

Safe occupancy of the control room during abnormal conditions will be provided for the design of the control room. 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 HEPA 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. Materials used in the control room construction will be nonflammable.

b. Control cables and switchboard wiring will be constructed of materials that have passed the flame test as described in Insulated Power Cable Engineers Association Publication S-61-402 and National Electrical Manufacturers Association Publication WC 5-1961.

c. Furniture used in the control room will be of metal construction.

d. Combustible supplies such as logs, records, procedures, manuals, etc., will be limited to the amounts required for Plant 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. The coaxial cables required for nuclear instrumentation.

c. Small amounts of combustible materials used in the manufacture of various electronic equipment.

The above list indicates that the flammable materials will be distributed to the extent that a fire would be unlikely to spread. Therefore, a fire, if started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the ventilation system.

Essential auxiliary equipment will be controlled by 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.

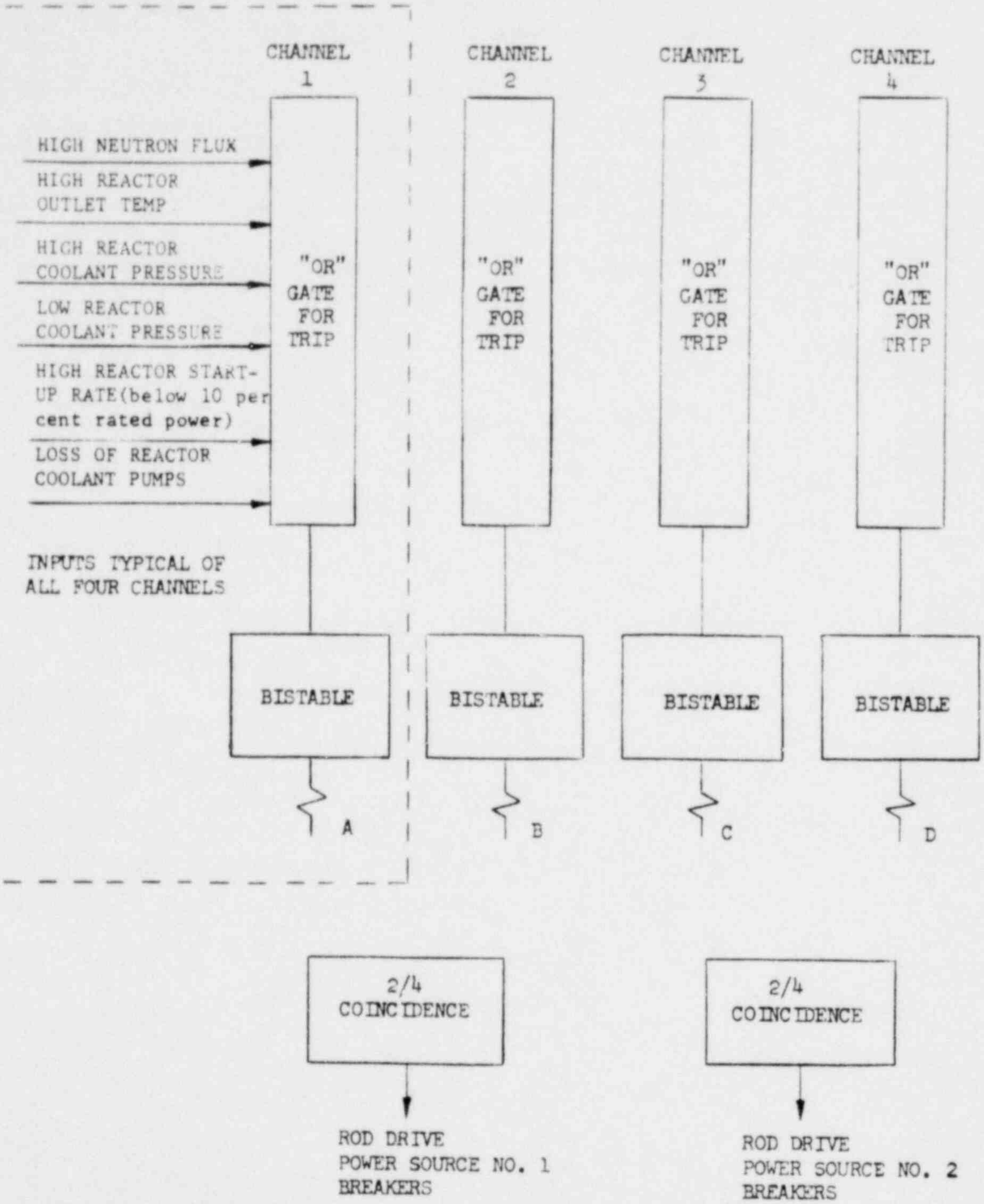
7.4.6 AUXILIARY CONTROL STATIONS

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 equipment 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 auxiliary systems equipment controlled by local stations.

7.4.7 SAFETY FEATURES

The primary objectives in the control room layout are to provide the necessary controls to start, operate, and shut down the nuclear 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 integrity during accident conditions. The layout of the engineered safeguards section of the control board will be designed to minimize the time required for the operator to evaluate the system performance under accident conditions. Any deviations from predetermined conditions will be alarmed so that the operator may take corrective action using the controls provided on the control panel.

0161



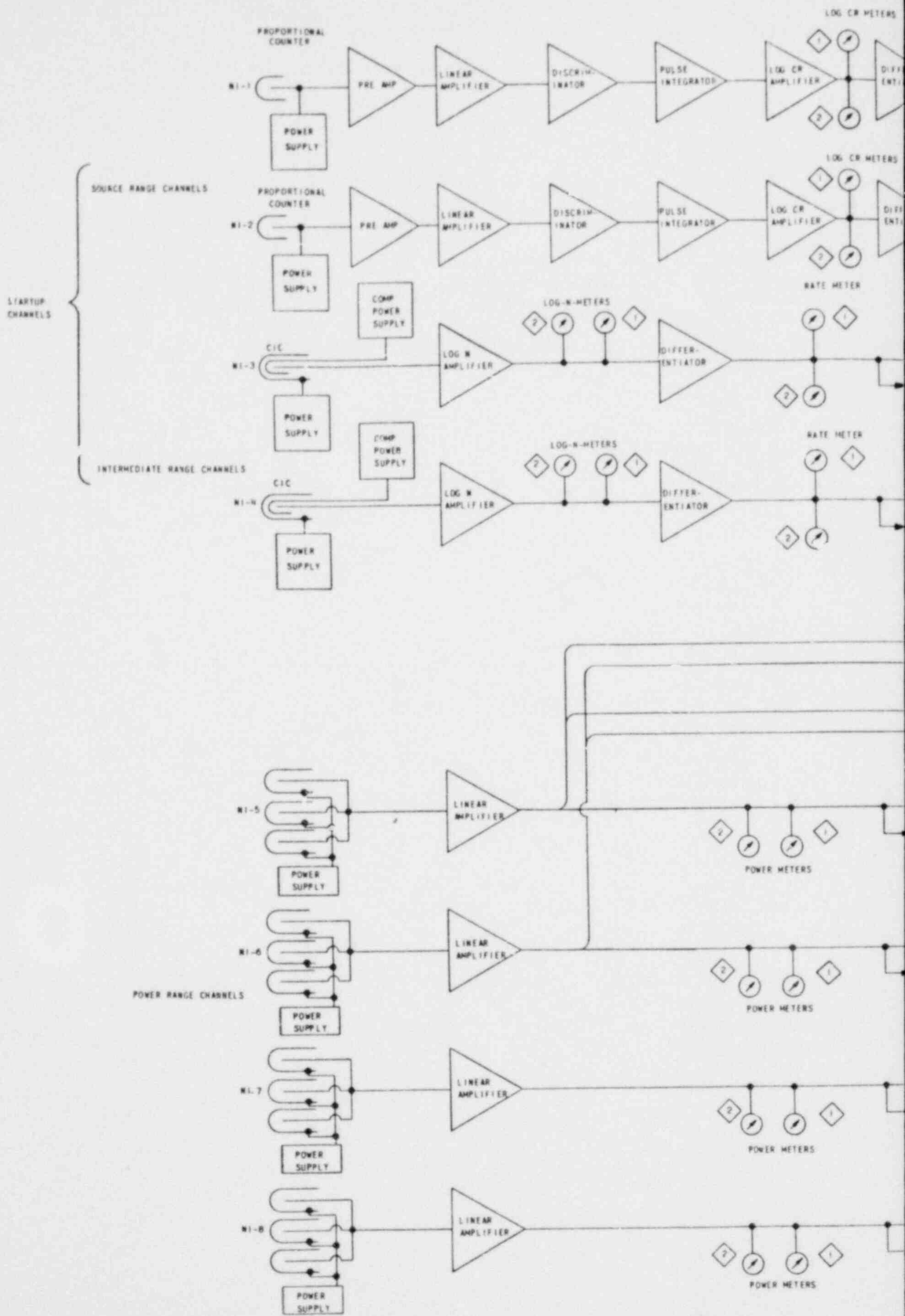
REACTOR PROTECTION SYSTEM BLOCK DIAGRAM

CRYSTAL RIVER UNITS 3 & 4

0162

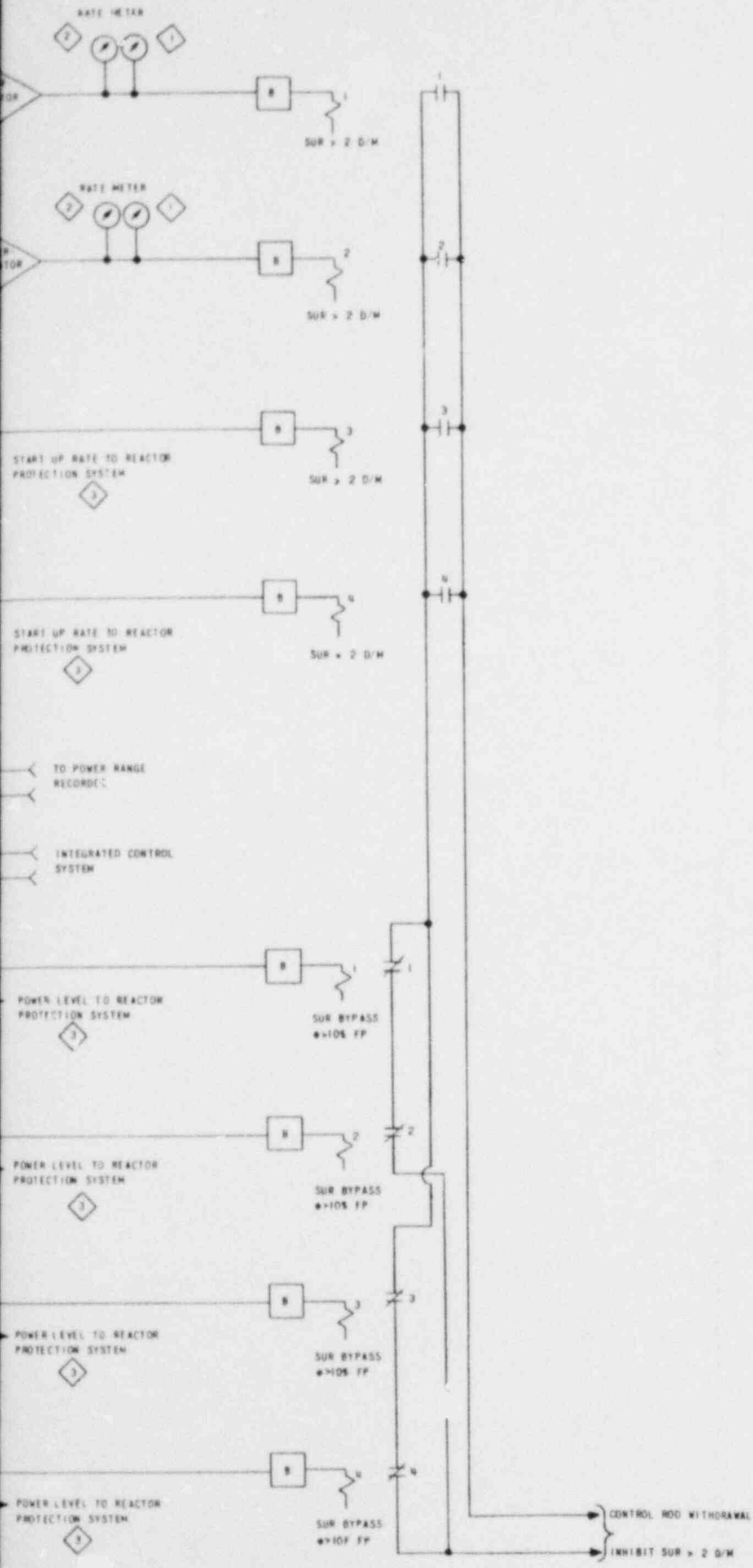
FLORIDA POWER CORPORATION

FIGURE 7-1



LOG CR METERS
PROPORTIONAL COUNTER
NI-1
PRE AMP
LINEAR AMPLIFIER
DISCRIMINATOR
PULSE INTEGRATOR
LOG CR AMPLIFIER
POWER SUPPLY
SOURCE RANGE CHANNELS
NI-2
STARTUP CHANNELS
COMP POWER SUPPLY
LOG-N-METERS
RATE METER
CIC
NI-3
LOG N AMPLIFIER
DIFFERENTIATOR
INTERMEDIATE RANGE CHANNELS
NI-4
NI-5
POWER METERS
NI-6
POWER RANGE CHANNELS
NI-7
NI-8

0163



NUCLEAR INSTRUMENTATION SYSTEM

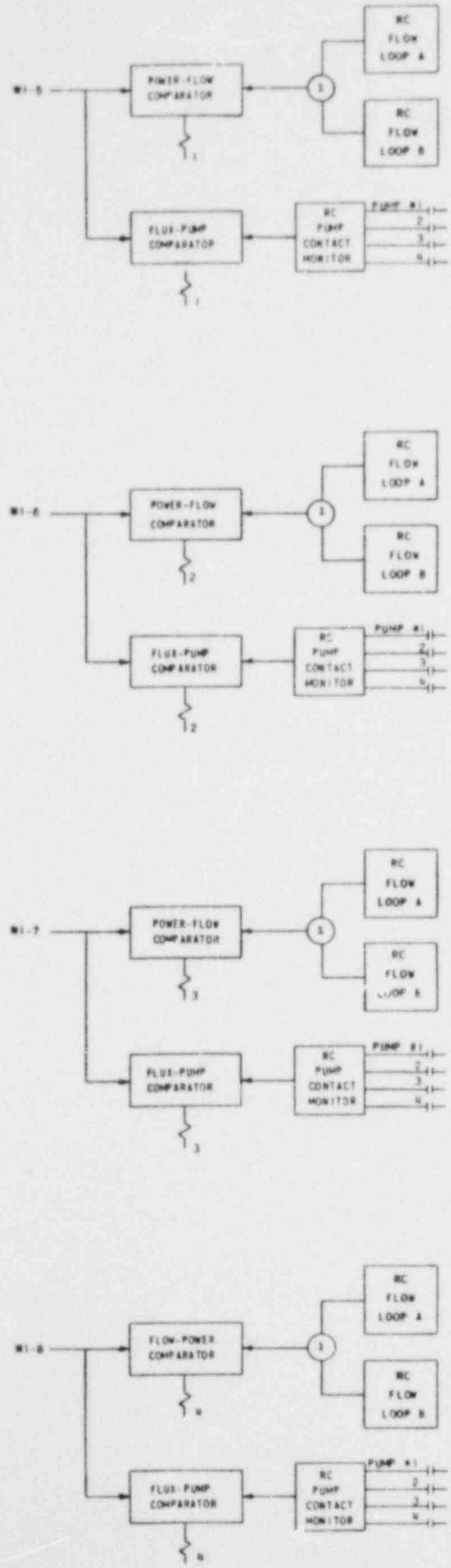
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

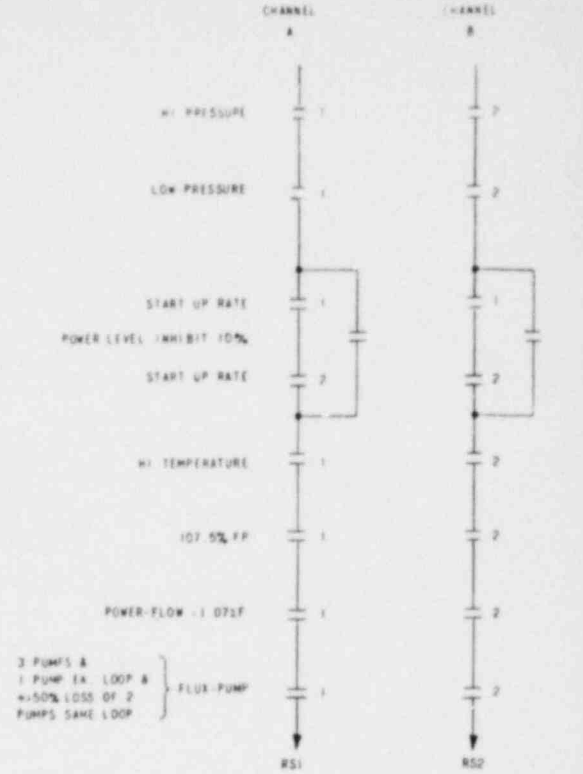
FIGURE 7-2a

AMEND. 7 (7-15-69)

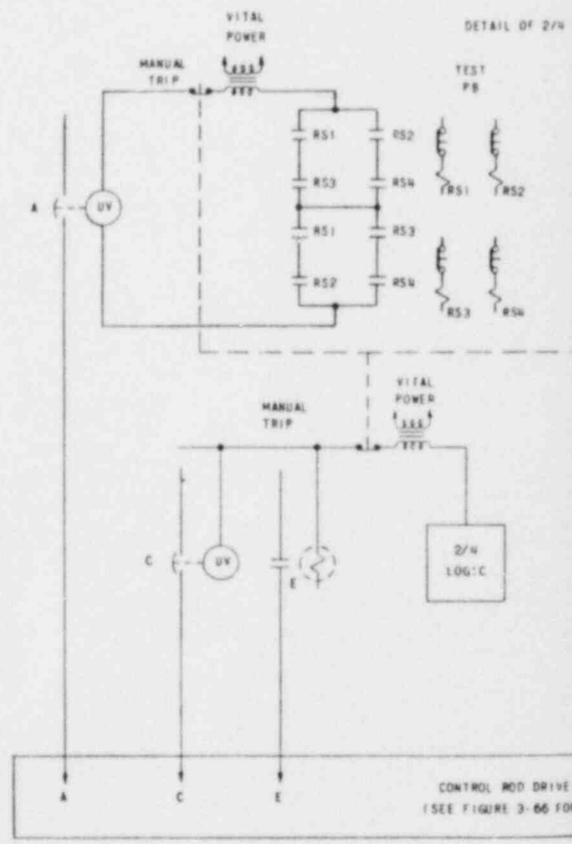
0164



WI-5
POWER-FLOW COMPARATOR
RC FLOW LOOP A
RC FLOW LOOP B
FLUX-PUMP COMPARATOR
RC PUMP CONTACT MONITOR
PUMP #1
WI-6
POWER-FLOW COMPARATOR
FLUX-PUMP COMPARATOR
WI-7
POWER-FLOW COMPARATOR
FLUX-PUMP COMPARATOR
WI-8
FLOW-POWER COMPARATOR
FLUX-PUMP COMPARATOR

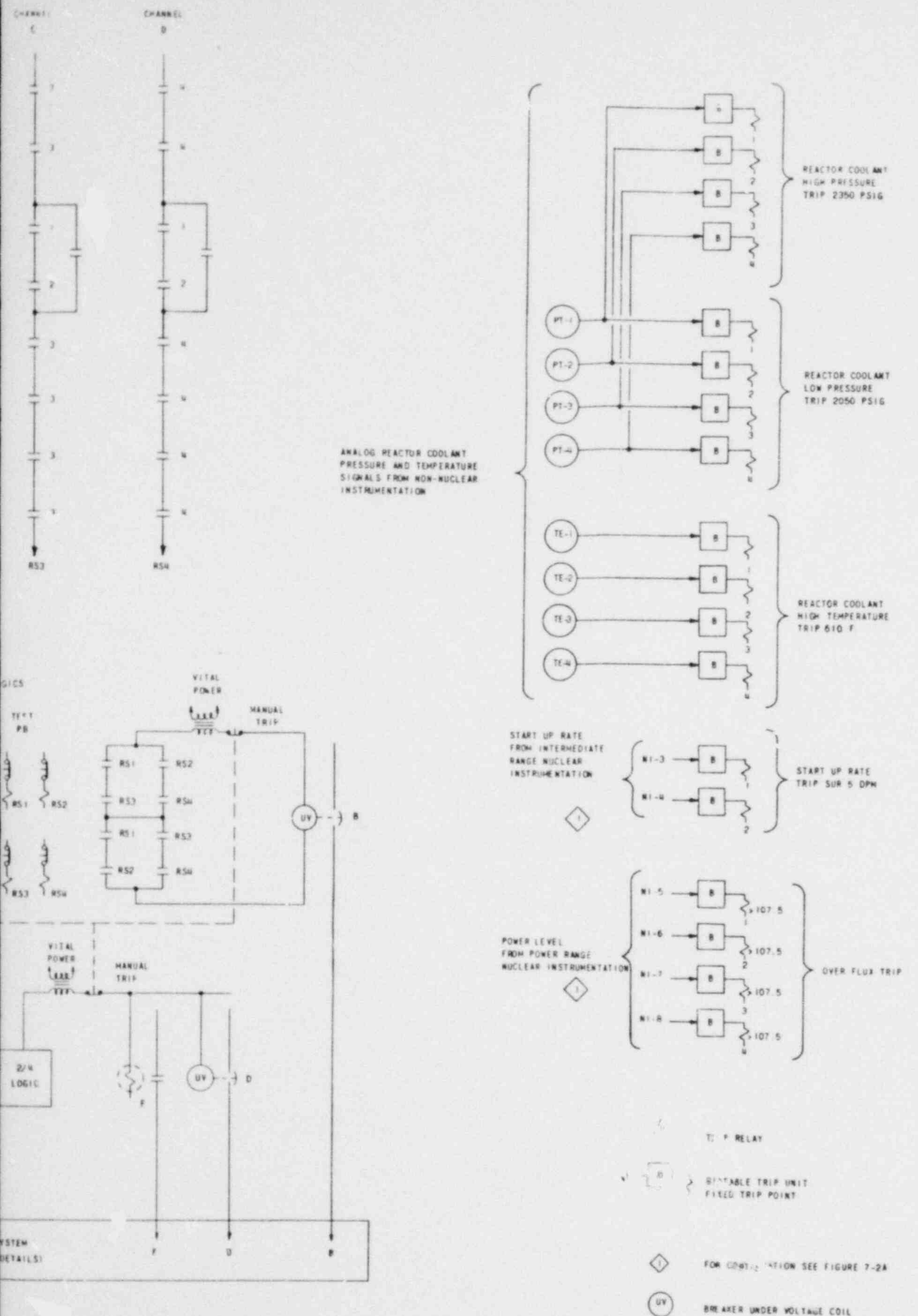


CHANNEL A
CHANNEL B
HI PRESSURE
LOW PRESSURE
START UP RATE
POWER LEVEL INHIBIT 10%
START UP RATE
HI TEMPERATURE
107.5% FP
POWER-FLOW
3 PUMPS &
1 PUMP EA. LOOP &
+50% LOSS OF 2
PUMPS SAME LOOP
FLUX-PUMP
RS1
RS2



VITAL POWER
DETAIL OF 2/4 L
MANUAL TRIP
TEST PB
RS1
RS2
RS3
RS4
UV
2/4 LOGIC
CONTROL ROD DRIVE
(SEE FIGURE 3-66 FOR

0165



0166

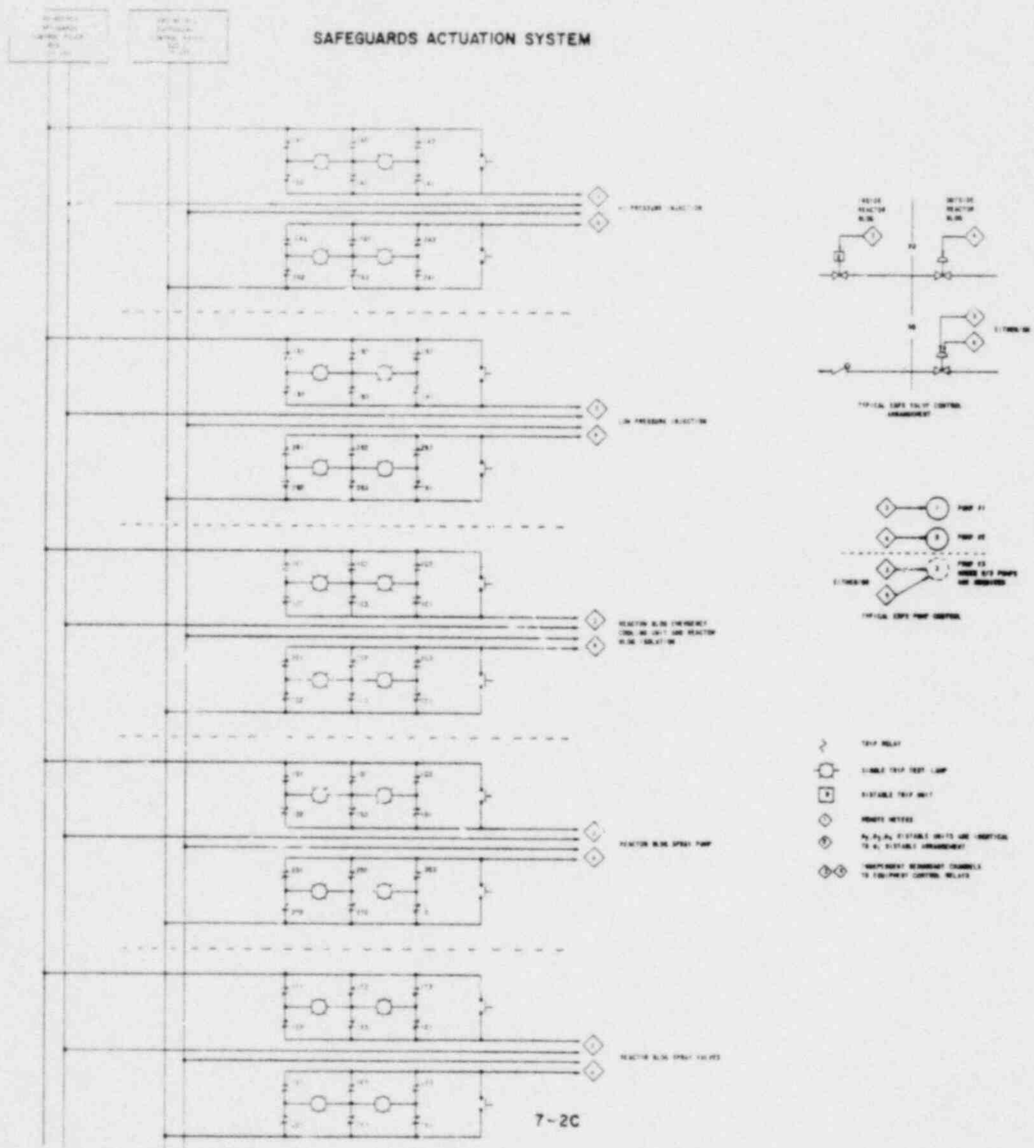
REACTOR PROTECTION SYSTEM

CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

FIGURE 7-2b

AMEND. 77



SAFEGUARDS ACTUATION SYSTEM

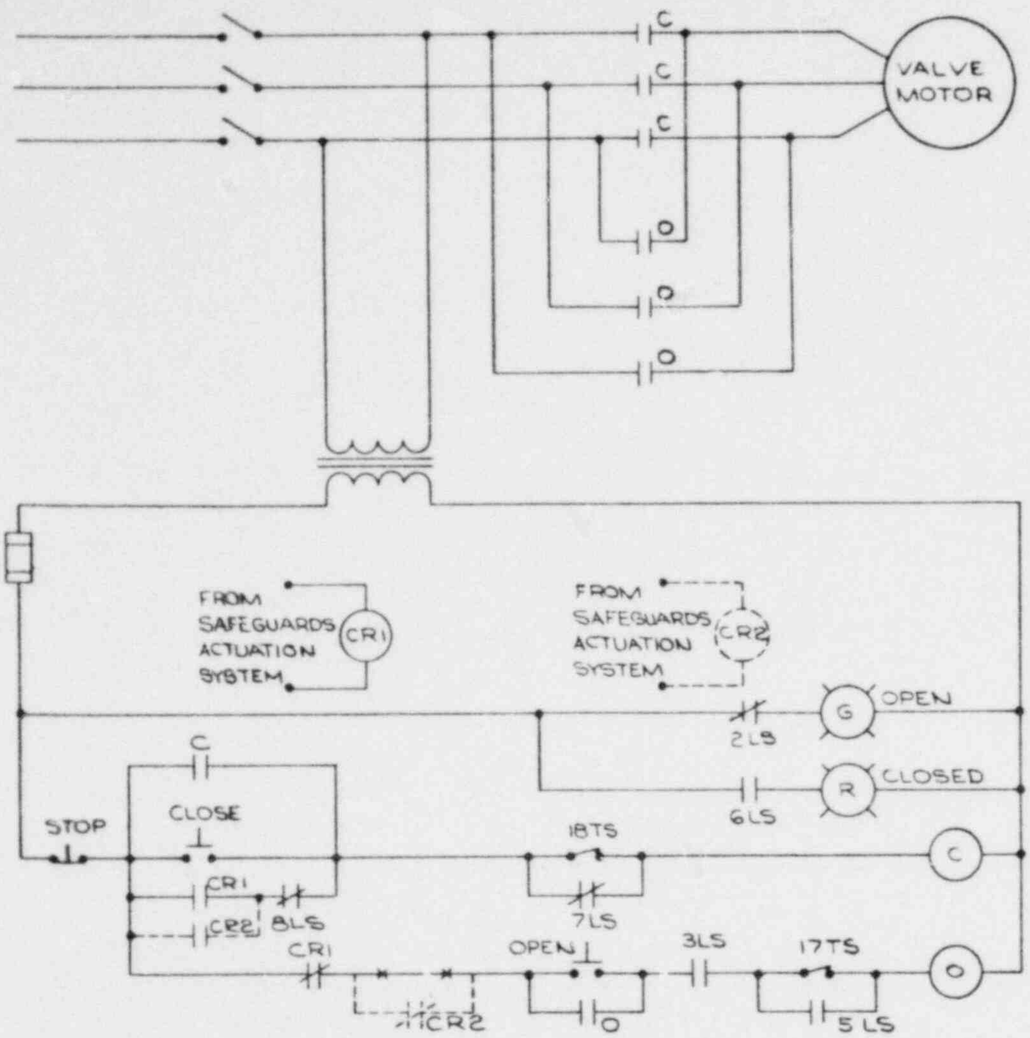
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

FIGURE 7-2c

AMEND. 7 (7-15-69)

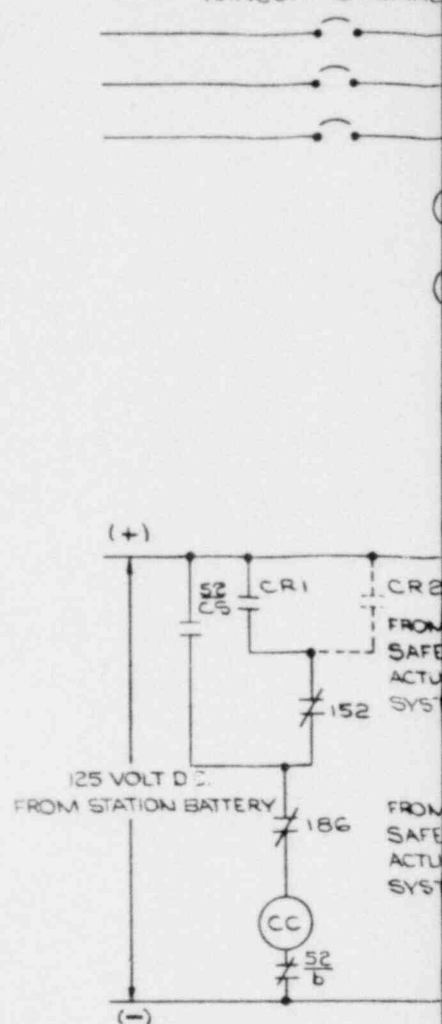
TYPICAL REACTOR BUILDING ISOLATION VALVE



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

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			

CONTROL CIRCUIT FOR D
(CIRCUIT BREAKE



CONTACTS SHOWN IN D

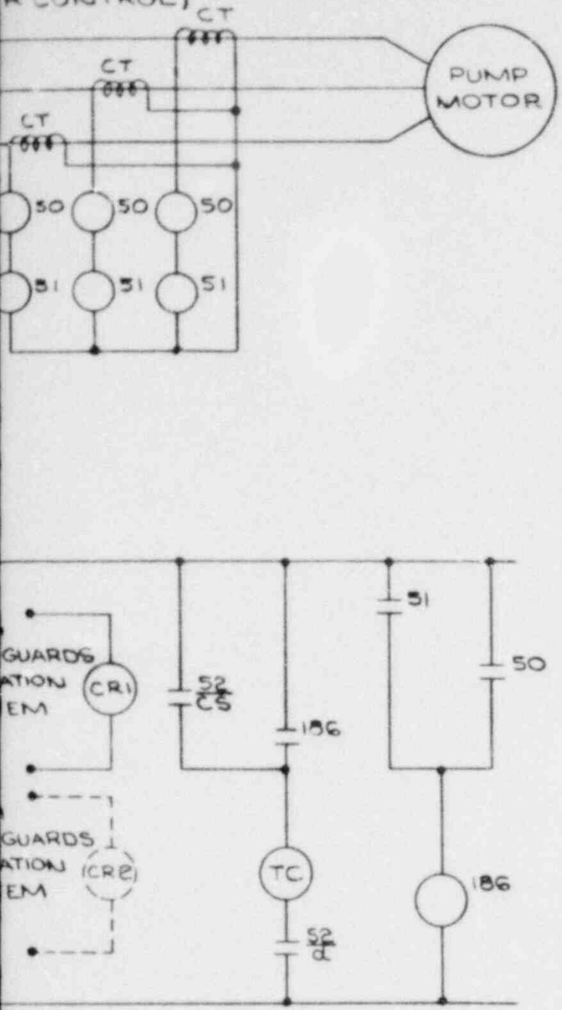
CR1 - CONTROL RELAY
CR2 - CONTROL RELAY
CC - CIRCUIT BREAKER CLOS
TC - CIRCUIT BREAKER TRIP

50 - INSTANTANEOUS OVER
51 - TIME OVERCURRENT
52/CS - CIRCUIT BREAKER CON
152 - CIRCUIT BREAKER AUX
186 - AUXILIARY TRIPPING

NOTE: CR2, SHOWN DOTTE
ONLY WHEN REDUNDANT
REQUIRED

0168

ECAY HEAT REMOVAL PUMP
R CONTROL)



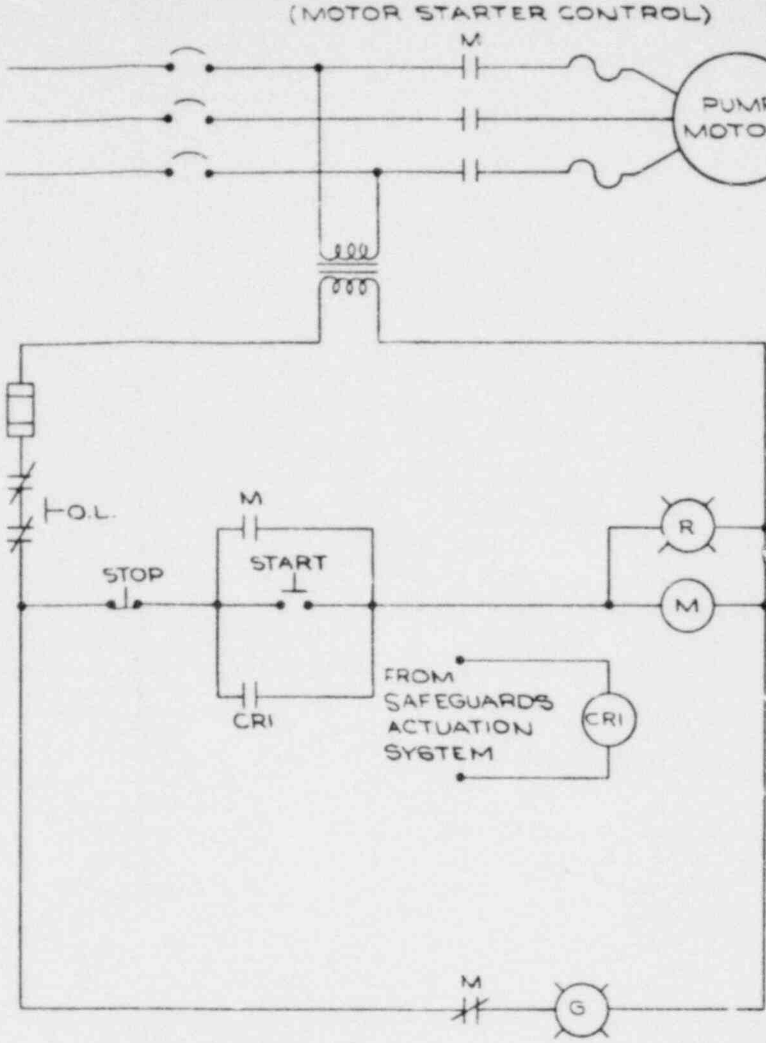
EENERGIZED POSITION

NG COIL
COIL

CURRENT RELAY
RELAY
TROL SWITCH
ILIARY SWITCH
RELAY

D, IS USED
CONTROL IS

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



CONTACTS SHOWN IN DEENERGIZED POSITION

CR1-CONTROL RELAY
M-MAIN CONTACTOR

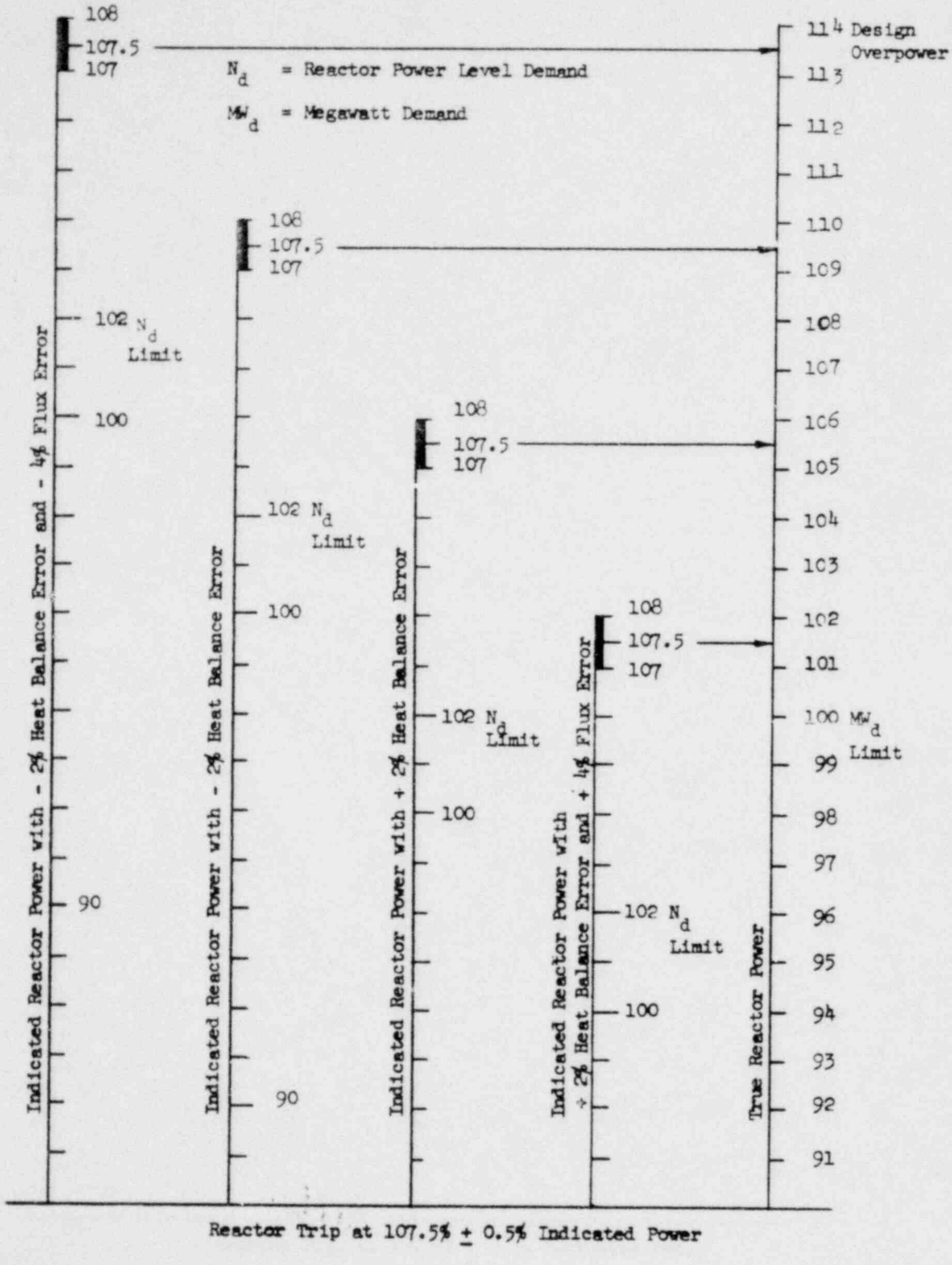
TYPICAL CONTROL CIRCUITS FOR
ENGINEERED SAFEGUARDS EQUIPMENT

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 7-3

0169



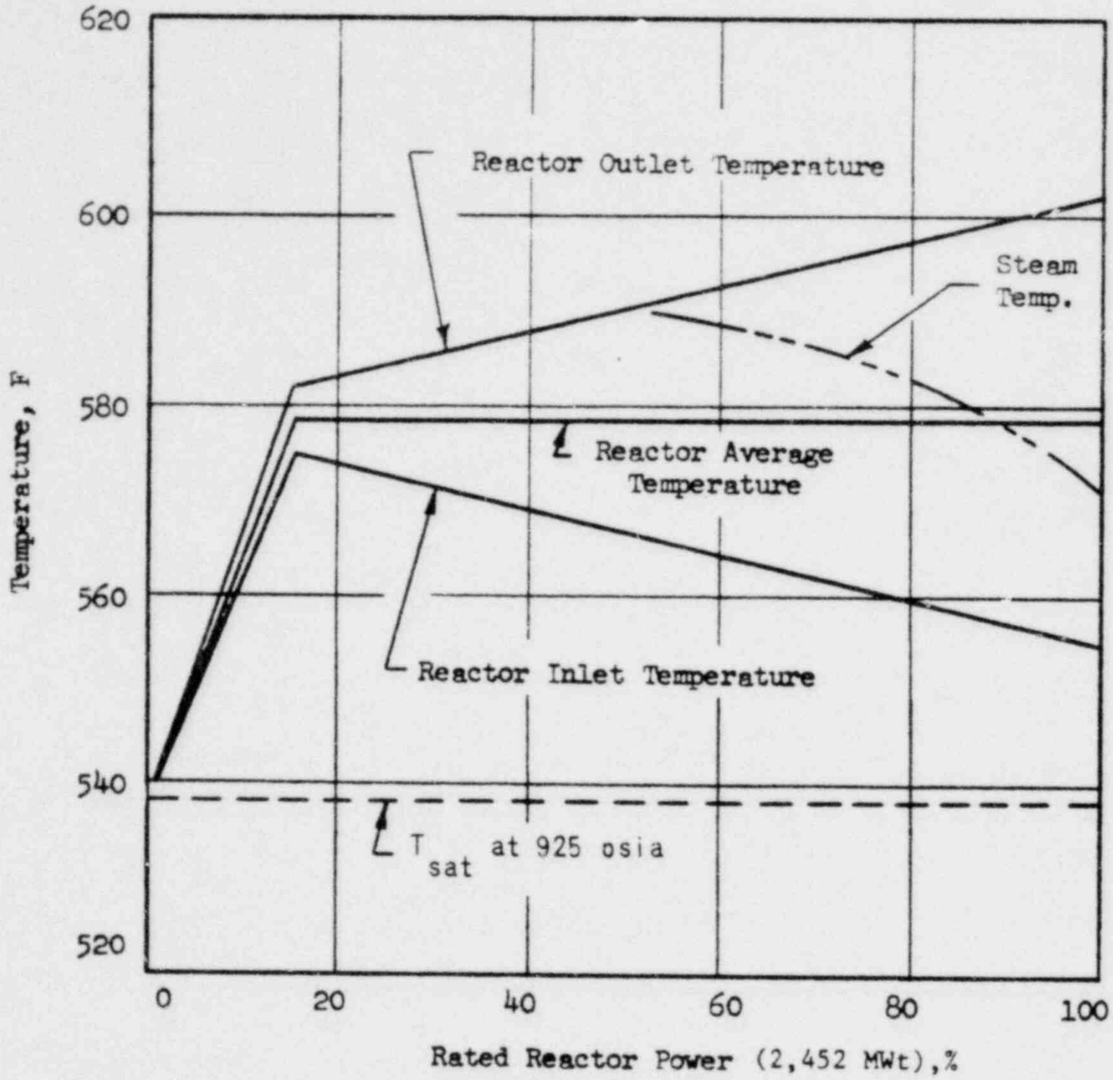
REACTOR POWER MEASUREMENT ERRORS & CONTROL LIMITS

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 7-4

0170



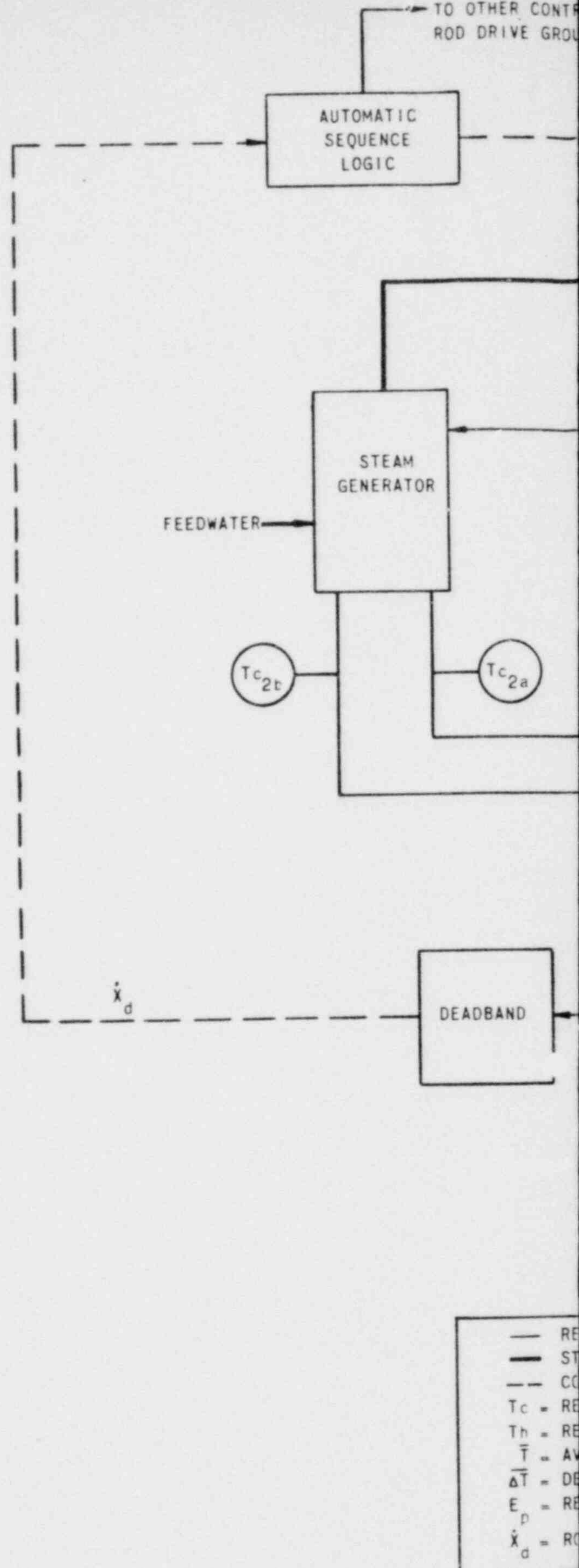
REACTOR & STEAM TEMPERATURES
VERSUS REACTOR POWER

CRYSTAL RIVER UNITS 3 & 4

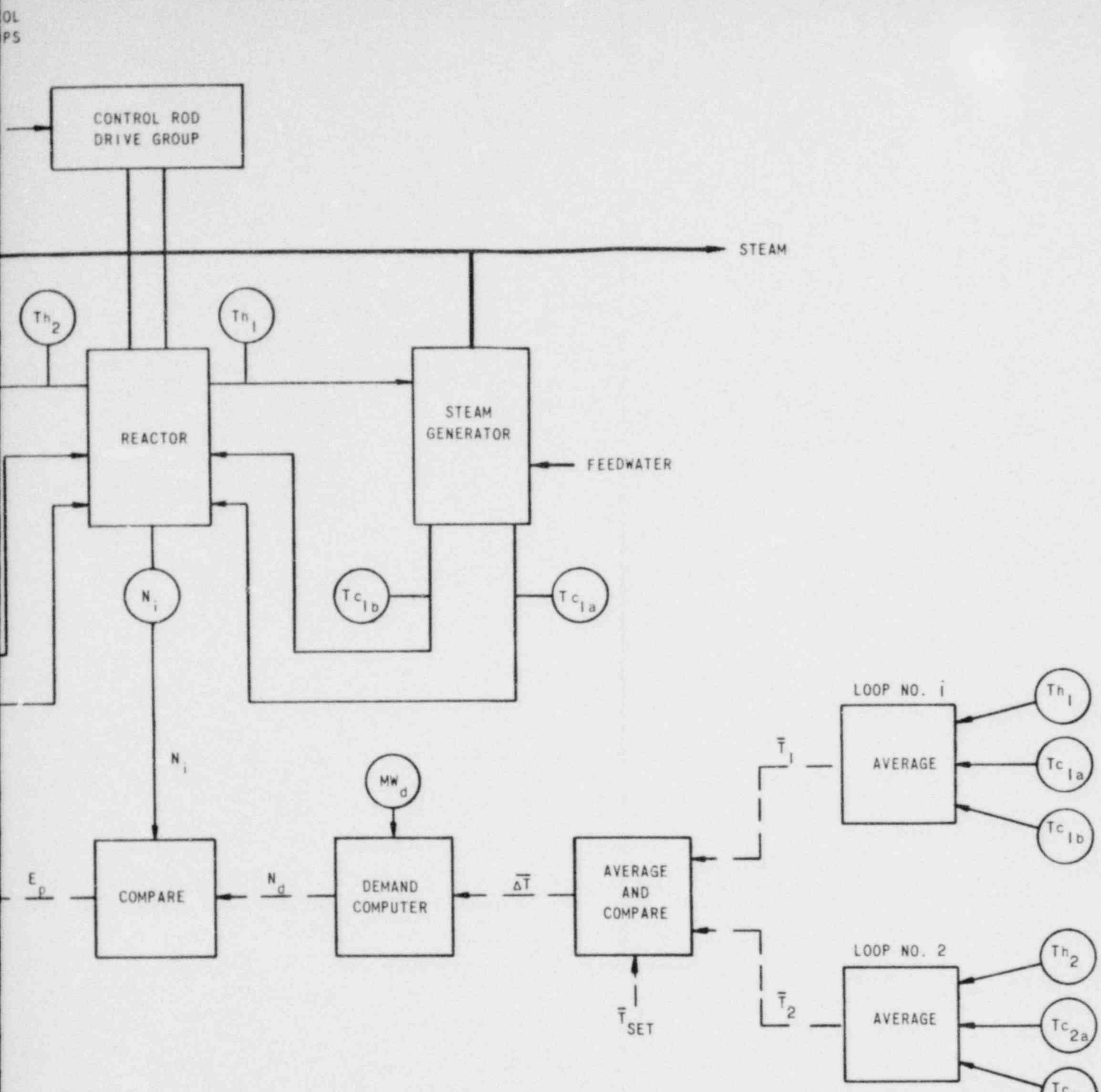
FLORIDA POWER CORPORATION

FIGURE 7-5

0171


TO OTHER CONTR
ROD DRIVE GROU
AUTOMATIC SEQUENCE LOGIC
STEAM GENERATOR
FEEDWATER
Tc_{2b}
Tc_{2a}
$\dot{X}_d$
DEADBAND
Tc -
Th -
$\bar{T}$ -
$\overline{\Delta T}$ -
E_p -
$\dot{X}_d$ -
MW_d -
N_i -
N_d -

0172



LEGEND

ACTOR COOLANT SYSTEM
EAM SYSTEM
NTROL SYSTEM
ACTOR COOLANT SYSTEM COLD LEG TEMPERATURE
ACTOR COOLANT SYSTEM HOT LEG TEMPERATURE
ERAGE REACTOR COOLANT SYSTEM TEMPERATURE
VIATION OF AVERAGE TEMPERATURE FROM SETPOINT
ACTOR POWER LEVEL ERROR, $N_d - N_i$
D VELOCITY DEMAND
GAWATT DEMAND
ACTOR POWER LEVEL
ACTOR POWER LEVEL DEMAND

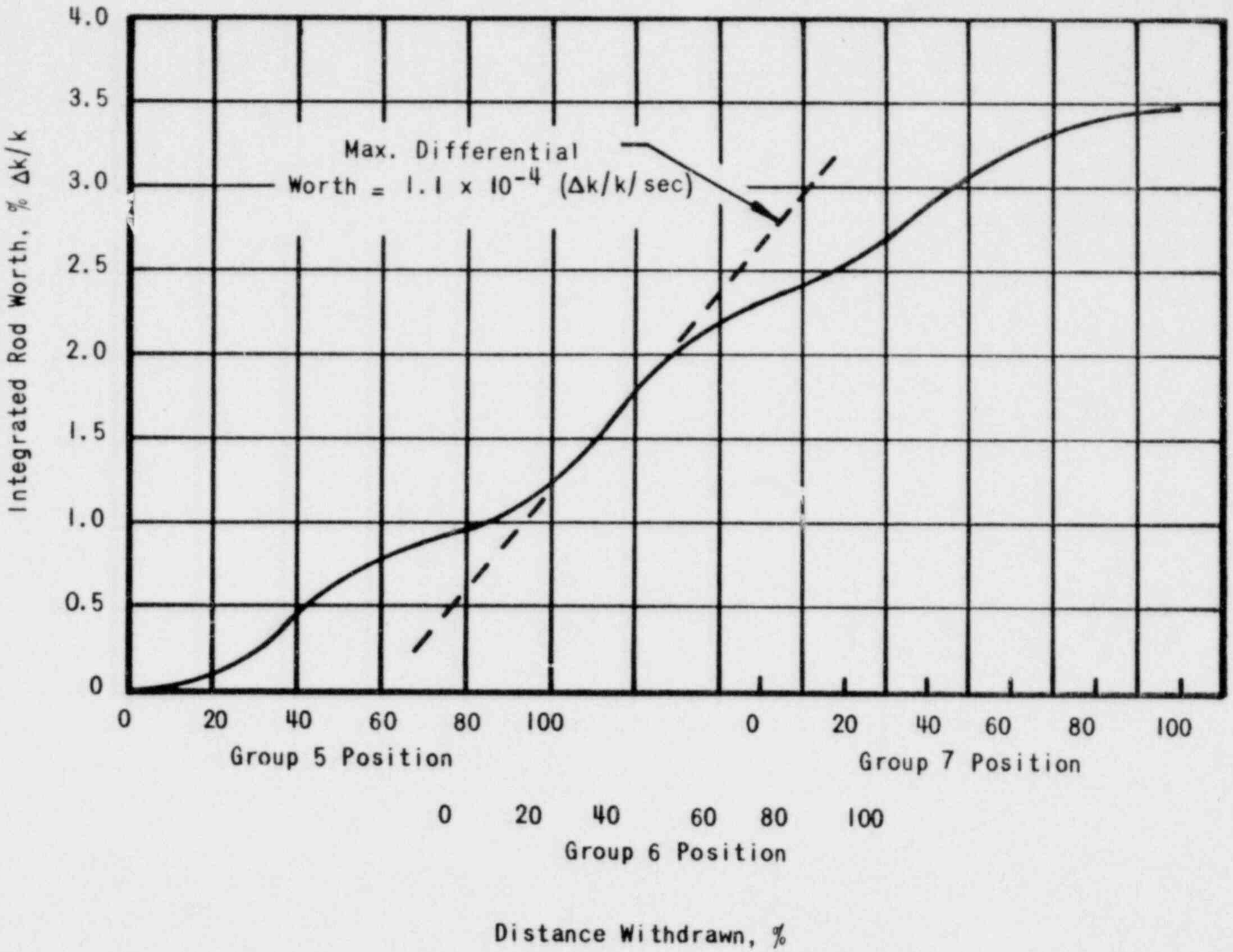
REACTOR CONTROL DIAGRAM
INTEGRATED CONTROL SYSTEM

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 7-6

AMEND. 3 (3-1-68)

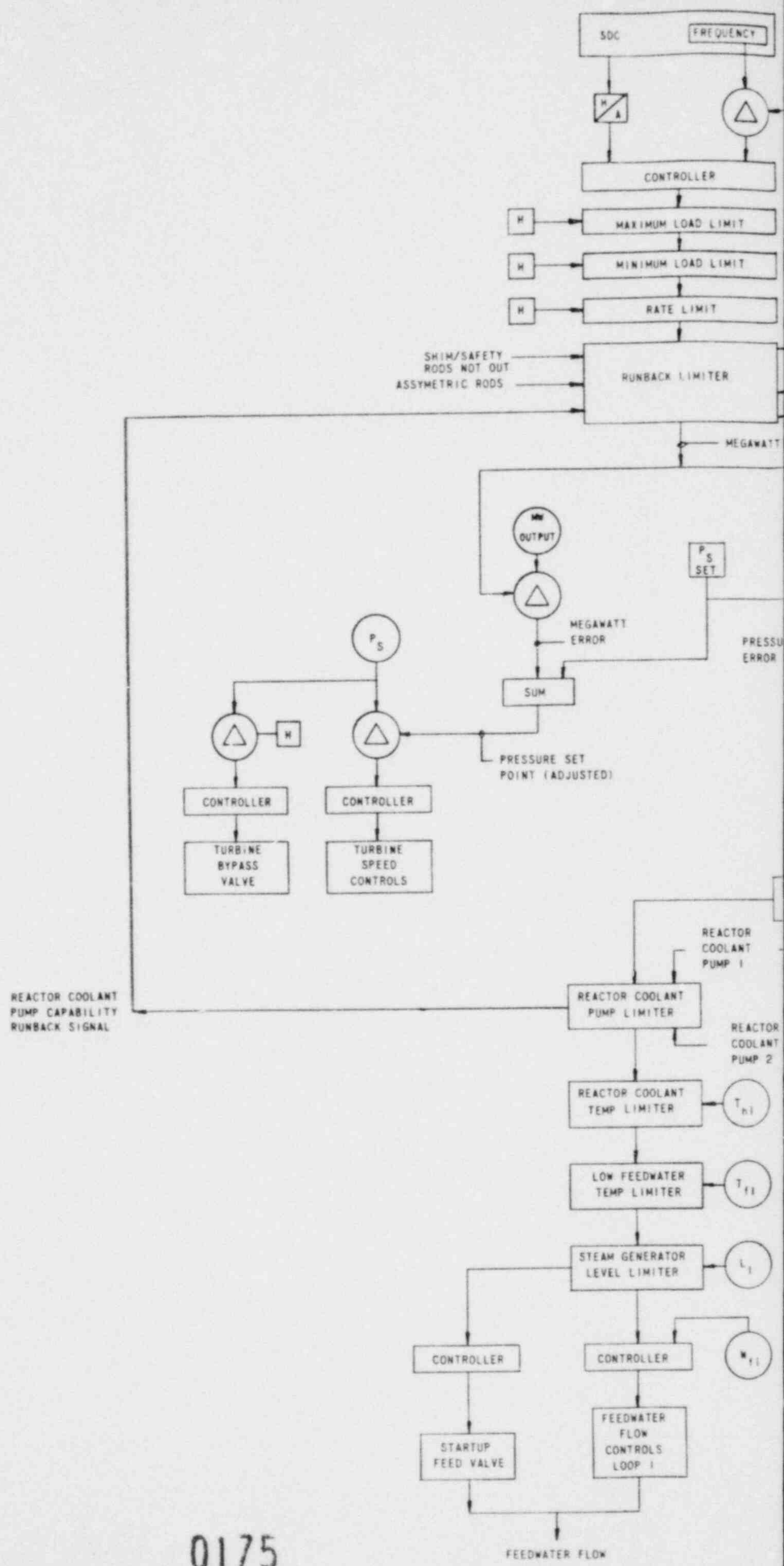


AUTOMATIC CONTROL ROD GROUPS
TYPICAL WORTH CURVE VERSUS
DISTANCE WITHDRAWN
CRYSTAL RIVER UNIT 3

FIGURE 7-7

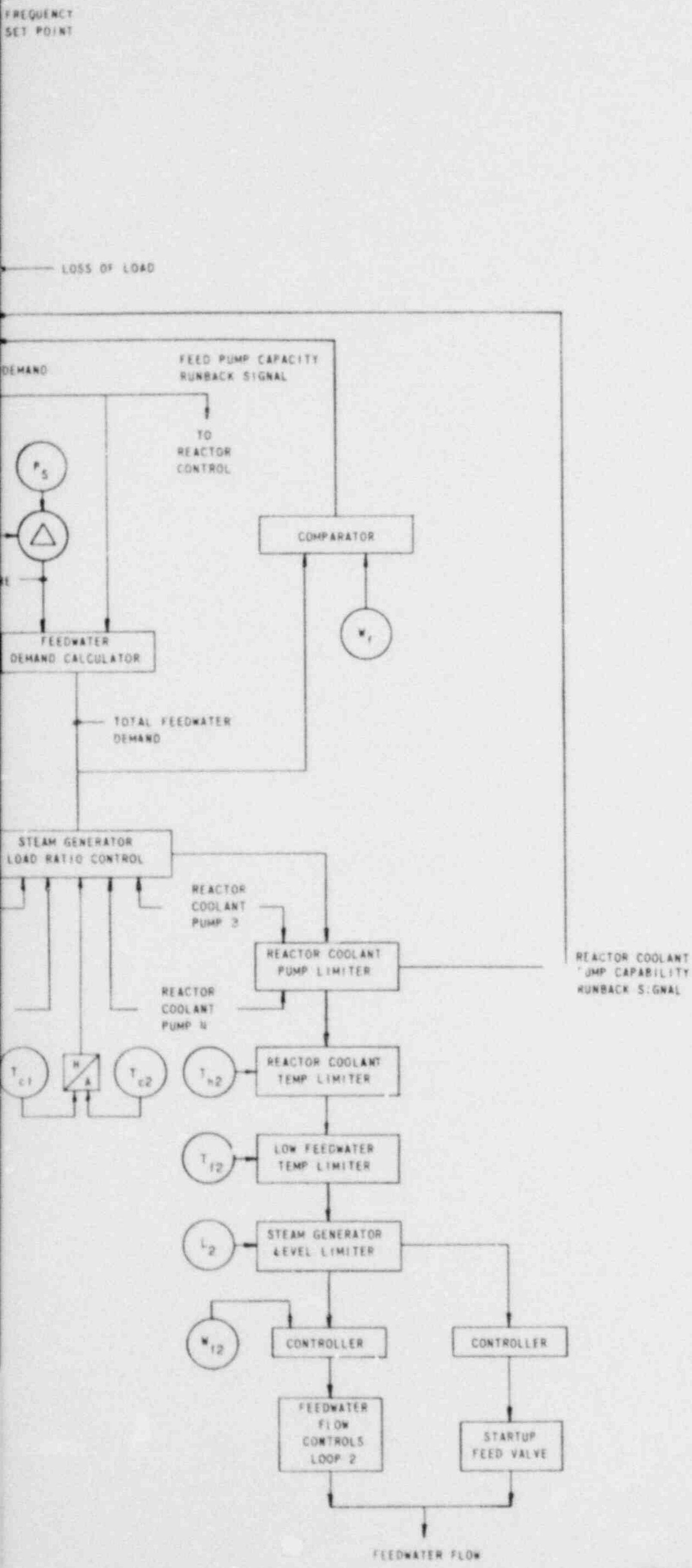
AMEND. 7 (7-15-69)

0174



SDC
FREQUENCY
CONTROLLER
H
MAXIMUM LOAD LIMIT
H
MINIMUM LOAD LIMIT
H
RATE LIMIT
SHIM/SAFETY RODS NOT OUT
ASSYMETRIC RODS
RUNBACK LIMITER
MEGAWATT
MW OUTPUT
P_S SET
MEGAWATT ERROR
PRESSU
ERROR
P_S
SUM
H
PRESSURE SET POINT (ADJUSTED)
CONTROLLER
CONTROLLER
TURBINE BYPASS VALVE
TURBINE SPEED CONTROLS
REACTOR COOLANT PUMP 1
REACTOR COOLANT PUMP CAPABILITY RUNBACK SIGNAL
REACTOR COOLANT PUMP LIMITER
REACTOR COOLANT PUMP 2
REACTOR COOLANT TEMP LIMITER
T_{h1}
LOW FEEDWATER TEMP LIMITER
T_{f1}
STEAM GENERATOR LEVEL LIMITER
L_1
CONTROLLER
CONTROLLER
W_{f1}
STARTUP FEED VALVE
FEEDWATER FLOW CONTROLS LOOP 1
FEEDWATER FLOW

0175



LEGEND:

SDC = SYSTEM DISPATCH CENTER
Δ = DIFFERENCE
P_S = STEAM PRESSURE
P_SSET = STEAM SET PRESSURE
W_f = FEEDWATER FLOW
H = MANUAL SET POINT
H/A = MANUAL/AUTOMATIC
L = STEAM GENERATOR LEVEL
T_h = REACTOR OUTLET TEMPERATURE
T_c = REACTOR INLET TEMPERATURE
T_f = FEEDWATER TEMPERATURE

THE NUMBER SUBSCRIPTS REFER TO LOOP 1 AND LOOP 2

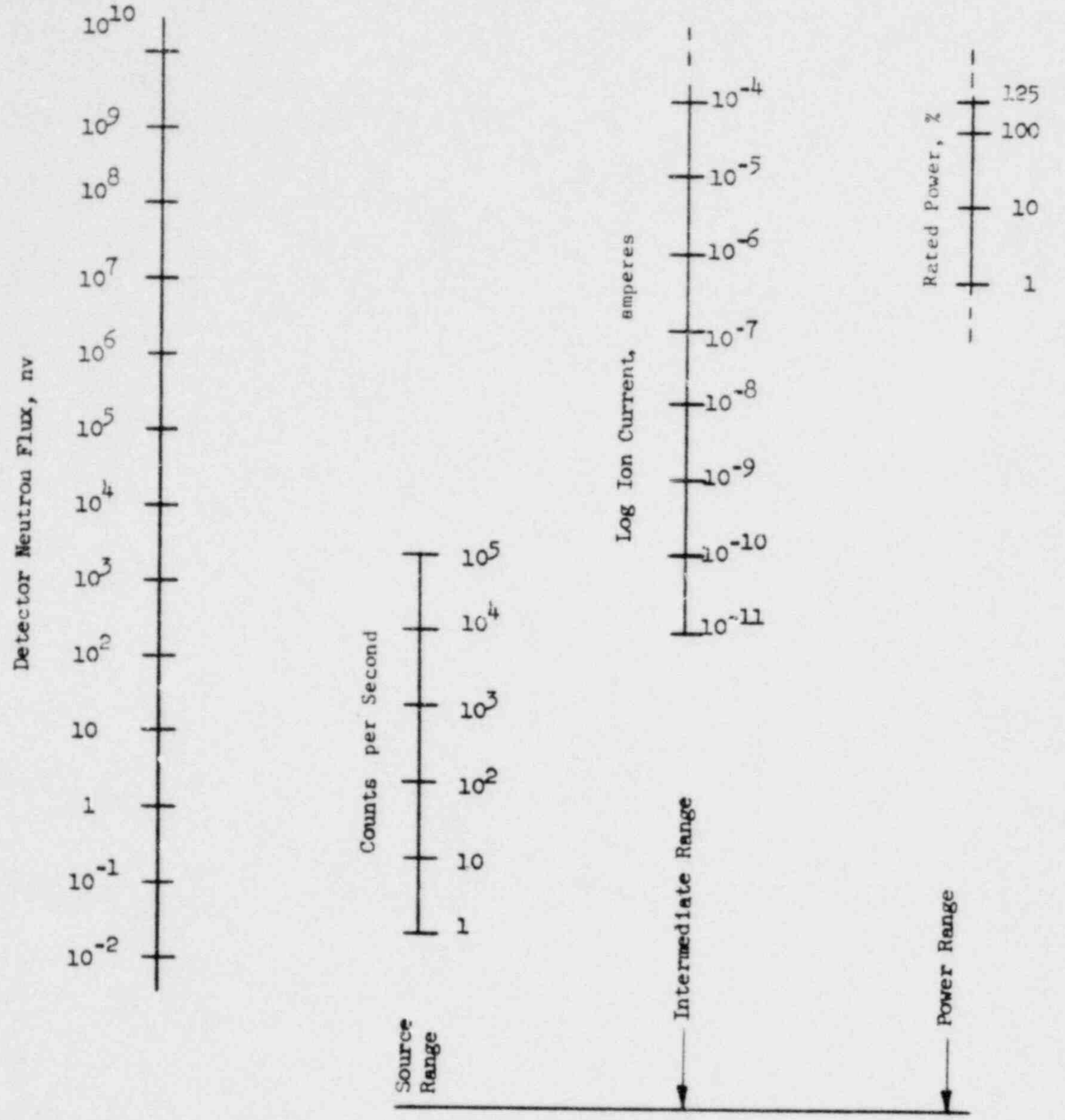
0176

STEAM GENERATOR AND TURBINE CONTROL DIAGRAM
INTEGRATED CONTROL SYSTEM

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 7-8



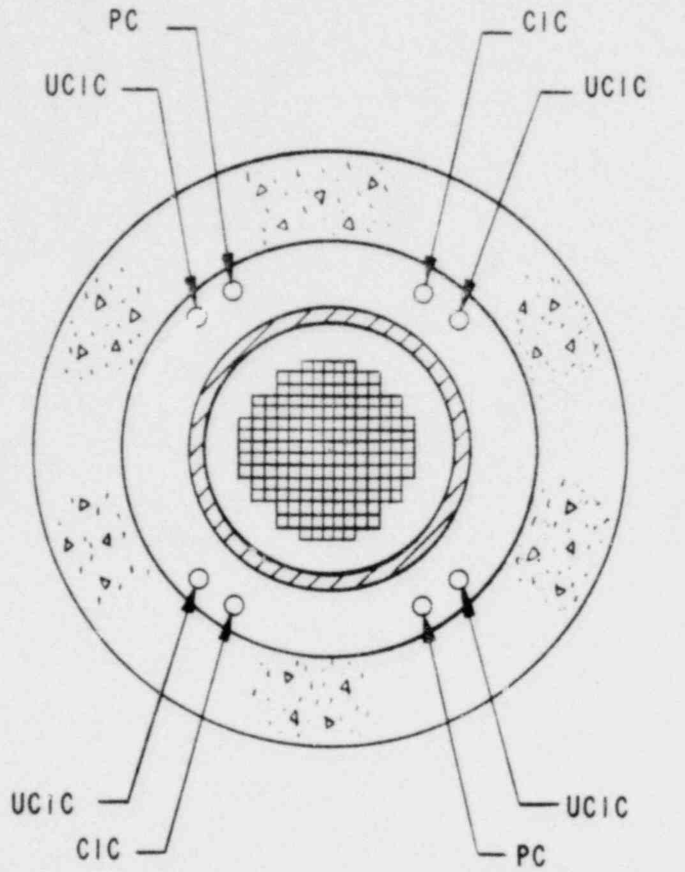
NUCLEAR INSTRUMENTATION FLUX RANGES

CRYSTAL RIVER UNITS 3 & 4

0177

FLORIDA POWER CORPORATION

FIGURE 7-9



LEGEND

PC - PROPORTIONAL COUNTER - SOURCE RANGE DETECTOR

CIC - COMPENSATED ION CHAMBER - INTERMEDIATE RANGE DETECTOR

UCIC - UNCOMPENSATED ION CHAMBER - POWER RANGE DETECTOR

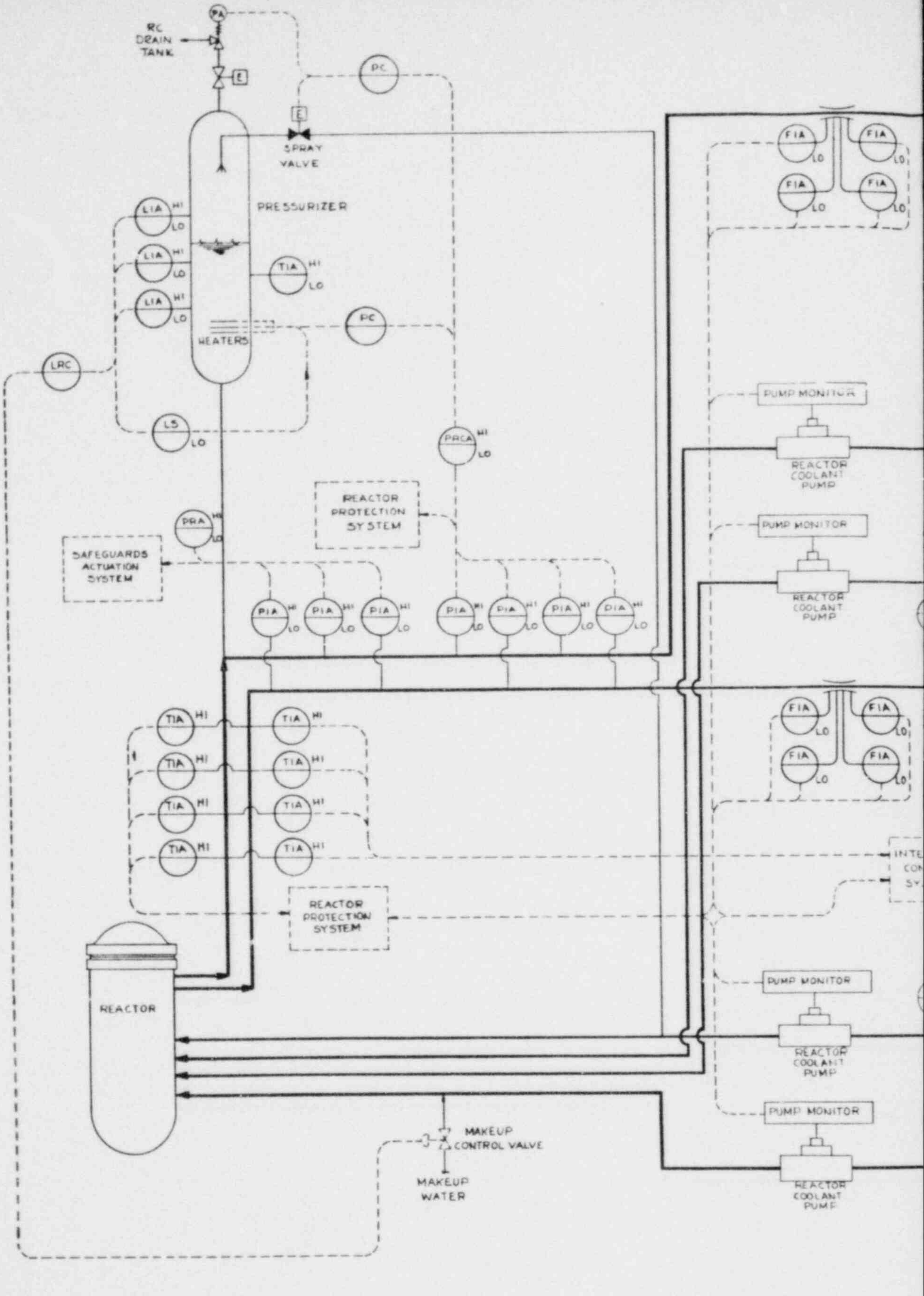
NUCLEAR INSTRUMENTATION DETECTOR LOCATIONS

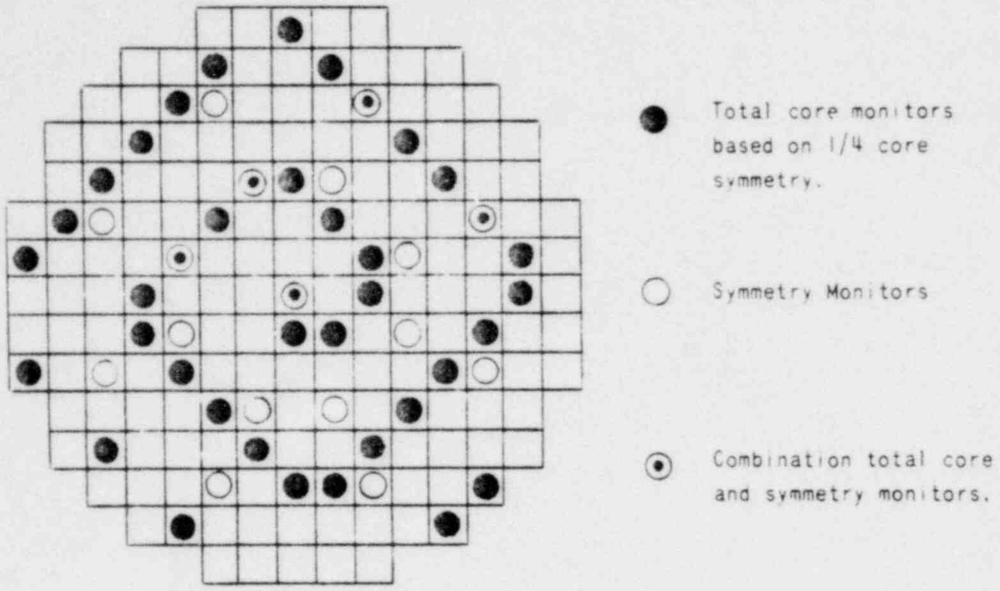
CRYSTAL RIVER UNITS 3 & 4

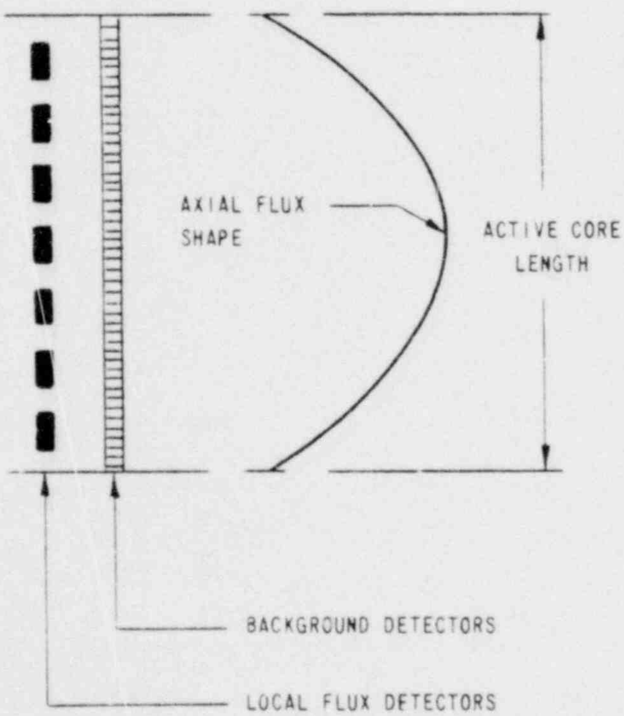
FLORIDA POWER CORPORATION

FIGURE 7-10

0178 [redacted]


RC DRAIN TANK
PA
E
PC
SPRAY VALVE
PRESSURIZER
LIA
HI
LO
TIA
HEATERS
LRC
LS
PRCA
PRA
REACTOR PROTECTION SYSTEM
SAFEGUARDS ACTUATION SYSTEM
PIA
FIA
PUMP MONITOR
REACTOR COOLANT PUMP
REACTOR
MAKEUP CONTROL VALVE
MAKEUP WATER



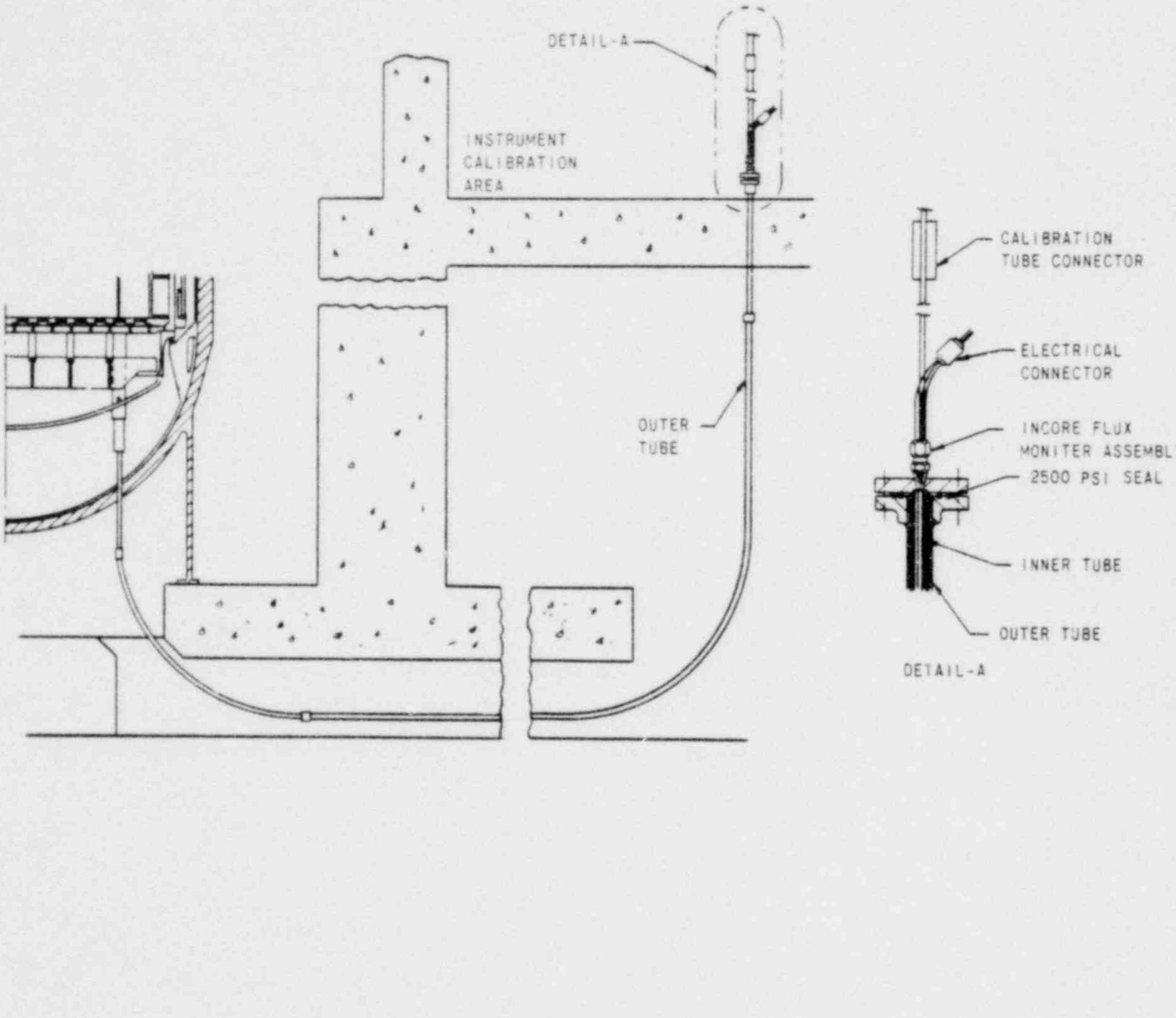


INCORE DETECTOR LOCATIONS
CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 7-12

AMEND. 1 (1-15-68)



TYPICAL ARRANGEMENT
INCORE INSTRUMENTATION CHANNEL

CRYSTAL RIVER UNITS 3 & 4

0182

FLORIDA POWER CORPORATION

FIGURE 7-13