3.0 PROCESS INSTRUMENTATION AND CONTROL SYSTEMS

The systems discussed in this chapter are those that have to do with process instrumentation or process control. The control systems used to alter reactor core reactivity are discussed in Chapter 7. The process instrumentation and control systems are shown in Figure 3.0-1 and include the following systems: the Reactor Vessel Instrumentation System, the Electro Hydraulic Control System, and the Feedwater Control System.

3.0.1 Reactor Vessel Instrumentation System (Section 3.1)

The Reactor Vessel Instrumentation System provides information concerning reactor vessel water level, reactor vessel pressure, reactor vessel temperature, and core flow rate. This information is used for control and automatic trip functions.

3.0.2 Electro Hydraulic Control System (Section 3.2)

The Electro Hydraulic Control System maintains a constant reactor pressure for a given reactor power level, controls the speed and load on the turbine generator, and provides protection for the main turbine.

3.0.3 Feedwater Control System (Section 3.3)

The Feedwater Control System regulates the flow of feedwater to the reactor vessel in order to maintain reactor water level. The Feedwater Control System measures and uses total steam flow, total feedwater flow, and reactor vessel level signals to carry out its function.

3.0.4 Composite BWR Control Systems

Figure 3.0-2 shows a composite drawing of BWR control systems. In addition to the Electro Hydraulic Control System and Feedwater Control System, which are discussed in this chapter, some Chapter 7 reactivity control systems are also shown to assist in understanding overall plant response. The three Chapter 7 systems included in the drawing are the Reactor Manual Control System, Recirculation Flow Control System, and Reactor Protection System.

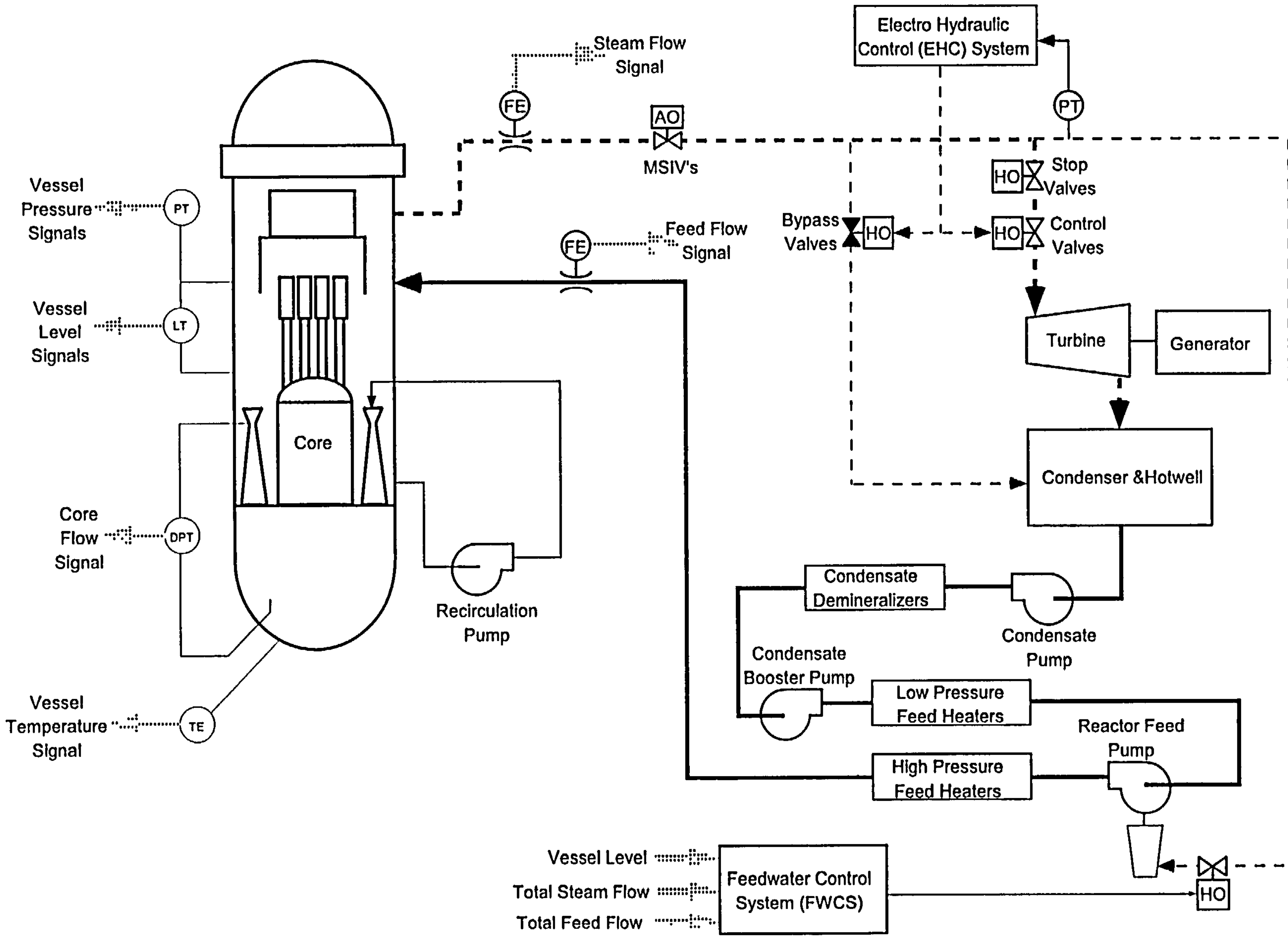


Figure 3.0-1 Process Instrumentation & Control Systems

0102

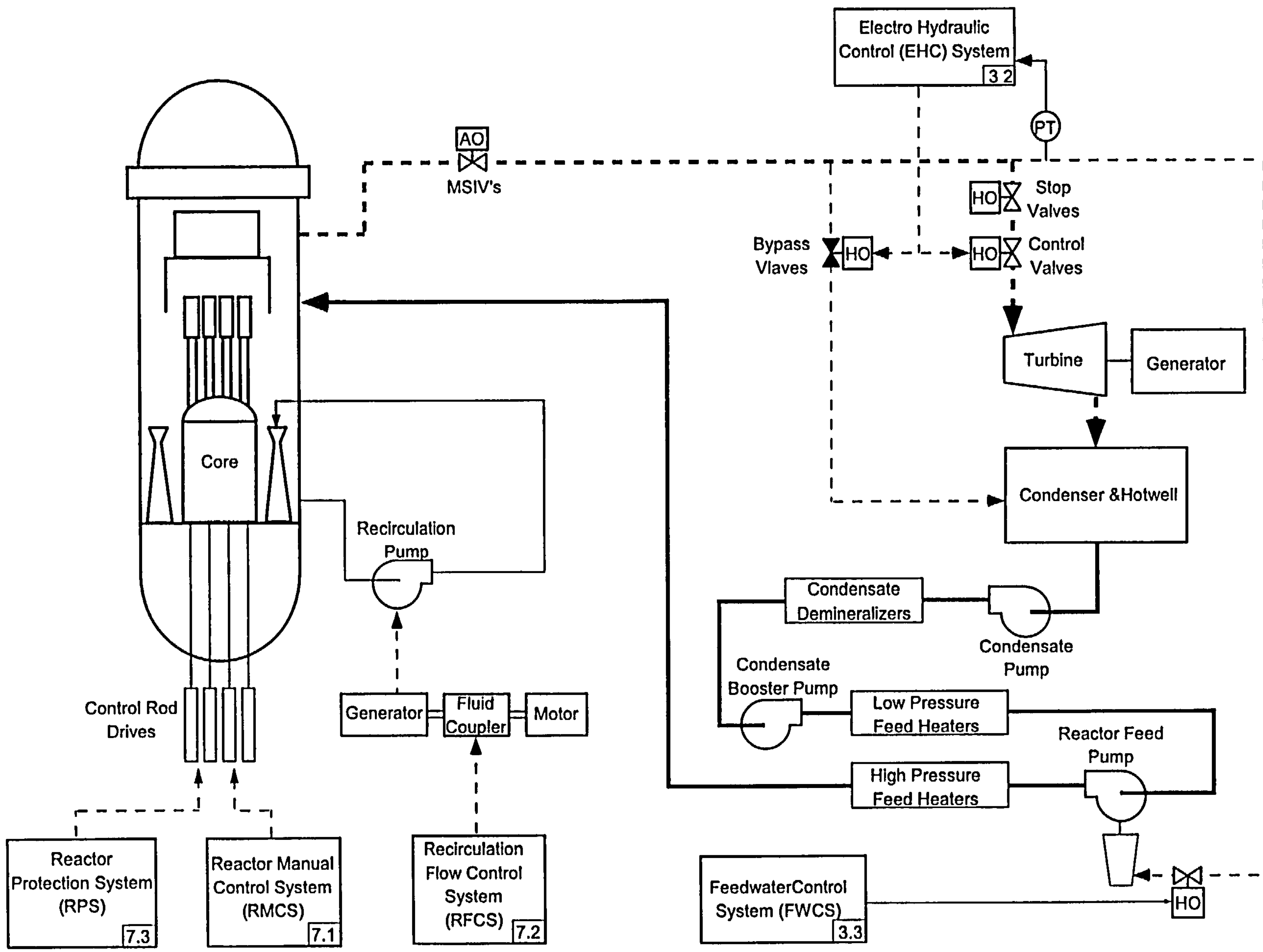


Figure 3.0-2 BWR Control Systems

Table of Contents

List of Tables

List of Figures

3.1 REACTOR VESSEL INSTRUMENTATION SYSTEM

The purpose of the Reactor Vessel Instrumentation System is to provide sufficient information concerning reactor vessel water level, reactor vessel pressure, reactor vessel temperature, and core flow rate to allow safe plant operation.

The functional classification of the Reactor Vessel Instrumentation System is that of a safety related system, although some portions are strictly for power generation.

3.1.1 System Description

Reactor vessel instrumentation consists of several individual subsystems that monitor reactor parameters such as water level, pressure, flow and temperature.

Reactor vessel water level is measured in the reactor vessel downcomer annulus. This parameter is monitored and displayed for operation on four different ranges.

Reactor vessel pressure is measured in the vessel steam space and displayed to aid the operator in safe plant operation. Both narrow and wide range pressure indications are provided for normal plant operation and for full range pressure coverage.

The plant power output capability should be proportional to the ability to remove the heat generated, so accurate core coolant flow measurements are required to evaluate core thermal behavior. Since the total flow that passes through the core must also pass through the jet pumps, the flow through each jet pump is measured and summed to yield total core flow.
Reactor vessel instrumentation supplies needed information to several systems such as the Reactor Protection System (RPS), Primary Containment Isolation System (PCIS), Feedwater Control System(FWCS) and the Emergency Core Cooling Systems (ECCS).

3.1.2 Component Description

The major components of the Reactor Vessel Instrumentation System are discussed in the paragraphs which follow.

3.1.2.1 Reactor Vessel Water Level Instrumentation

Reactor vessel water level is obtained through sensors which compare the weight of water in a reference column to the height (weight) of the water in the reactor downcomer annulus. Condensing chambers, external to the reactor vessel, are used to provide a constant reference column of water.

During normal reactor operation, reactor water level is maintained approximately 17 feet above the top of the active fuel (Figure 3.1-1). Maintaining an acceptable water level in the reactor vessel ensures that a sufficient quantity of reactor coolant is available to dissipate the heat generated by the core and the reactor is operating within the initial conditions assumed for the various analyzed accidents.

The level sensors, most of which indicate locally, are located throughout the reactor building at instrument racks. From the instrument racks, the level is transmitted to nine separate reactor vessel water level indicators in the control room and various trip circuits as shown on Figure 3.1-1 and Table 3.1-1.

3.1.2.2 Reactor Vessel Pressure

Reactor Vessel Pressure is sensed in the steam dome area using the same instrument piping that exists for vessel level instrumentation. The reactor vessel pressure instruments contain numerous pressure transmitters (PT), pressure switches (PS) and pressure indicators (PI). A summary of reactor pressure trips is shown in Table 3.1-2.

3.1.2.3 Core Flow Instrumentation

To evaluate reactor core power level and core thermal characteristics, accurate core flow measurements are required. Since all core flow, except control rod drive cooling water, must pass through the jet pumps, the flow through each jet pump is measured and summed to yield total core flow.

All 20 jet pumps have a pressure tap on the pump throat which is compared to core inlet plenum area pressure. The square root of this differential pressure provides a signal representing jet pump flow. As indicated on Figure 3.1-2, five jet pump flow signals are summed and then added to another five jet pump flow signals to yield loop flow. The two loop flow signals are then summed to yield total core flow.

During normal plant operation with both recirculation pumps operating, the loop flows are simply added together in a summation network. However, if one recirculation pump is off and the other is operating, the inactive loop will have reverse flow. The jet pump flow transmitters are not capable of distinguishing the direction of flow through the jet pumps. As a result, a relay logic system senses recirculation pump status to subtract the idle loop flow from the operating loop flow to yield an accurate total core flow output signal.

3.1.2.4 Reactor Vessel Temperature Instrumentation

The reactor vessel metal temperature is measured and monitored to provide temperature data representative of thick, thin, penetration, and transitional sections of the vessel. The temperature monitoring system is designed to map temperature gradients during startup and shutdown conditions. The data is recorded by a multipoint recorder and a two pen recorder, providing the basis to establish the rate of heating or cooling performed on the vessel.

3.1.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

3.1.3.1 Bases for Level Setpoints

Vessel level instrumentation used to initiate safety systems, cause operational trips, and provide control system inputs, are listed in Table 3.1-1. The reactor vessel water level trip setpoints are referred to as numbered levels. These levels and their elevation referenced to instrument zero are: Level 1 (-132.5"), Level 2 (-38"), Level 3 (+12.5"), Level 4 (+33.5"), Level 5 (approximately +37"), Level 7 (+40.5"), and Level 8 (+56.5"). The bases for the various level setpoints are discussed in the paragraphs which follow.

3.1.3.1.1 Level 8 (+56")

The trip of the main turbine is to protect it against the occurrence of gross carryover of moisture and subsequent damage to the turbine bladeing. The reactor feed pump turbines are tripped to prevent overfilling the reactor vessel. The reactor core isolation cooling and the high pressure coolant injection turbines are tripped, in the event these systems have activated, to prevent flooding of steam lines.

3.1.3.1.2 Level 7 Alarm (+40")

While operating at full power, the high level alarm annunciates at the reactor vessel water level above which moisture carryover in the steam is expected to increase at a significant rate. The alarm warns the operator of this undesirable condition.

3.1.3.1.3 Normal Operating Level (+37")

Reactor vessel water level can be controlled at any point between the high and low level alarms. However, the Feedwater Control System is usually set to maintain vessel level at +37 inches.

3.1.3.1.4 Level 4 Alarm Trip (+33")

The low water level alarm annunciates at the reactor vessel water level below which steam carryunder in the water will begin affecting the reactor recirculation flow rate significantly at full power because of recirculation pump cavitation. A water decrease to this point, coincident with a reactor feed pump trip, causes the recirculation pumps (Section 7.2) to runback to a predetermined speed to reduce thermal power output within the capacity of the remaining reactor feed pump(s).

3.1.3.1.5 Level 3 Trip (+12")

The low level scram function is for protection against high moisture carryover because of steam bypassing the dryer under the seal skirt. The scram occurs while the water level is above the bottom of the dryer seal skirt. The level selection also results in a quantity of reserve coolant between this level and the top of the active fuel to account for evaporation (decay heat boil off) losses, steam void collapse, and other coolant losses from the reactor vessel following a loss of feedwater flow, without the vessel water level decreasing to -132.5", which would initiate the Emergency Core Cooling Systems. This selected quantity of reserve coolant assumes the Reactor Core Isolation Cooling (RCIC) System is providing design flow rate. A decrease of reactor vessel inventory to this level also causes actuation of the Primary Containment Isolation System.

3.1.3.1.6 Level 2 Trip (-38")

This setpoint is selected to be low enough so that the RCIC and High Pressure Coolant Injection (HPCI) Systems will not be initiated on low level after a reactor scram unless feedwater flow has been terminated. The setpoint accounts for the expected level decrease caused by steam void collapse which occurs following any scram. The setpoint is selected high enough so that the RCIC System design flow is sufficient, taking into account system startup time following a loss of feedwater flow, to recover reactor vessel water level and prevent a level decrease to 132.5" with a subsequent initiation of emergency systems. The various system isolations are to prevent or limit the loss of reactor coolant and the release of radioactive products to the atmosphere assuming that the vessel water level decrease was due to a leak from one or more of the effected systems.

The recirculation pumps are tripped to insert negative reactivity using subsequent void formation, in the unlikely event that the reactor did not scram on a reactor vessel low water level signal. This even is referred to as an anticipated transient without scram - recirculation pump trip (ATWS-RPT).

3.1.3.1.7 Level 1 Trip (-132")

This level setpoint is selected to be high enough above the top of active fuel to initiate the ECCS thus providing adequate time for the ECCS to function in the event of a Loss of Coolant Accident (LOCA) to provide adequate core cooling and prevent fuel damage.

3.1.3.2 Bases for Reactor Pressure Setpoints

A summary of reactor pressure trips is given in Table 3.1-2 and discussed in the paragraphs which follow.

A reactor pressure of 1120 psig trips the recirculation pumps to insert negative reactivity by means of void formation, assuming the reactor failed to scram on high pressure. This event is referred to as an anticipated transient without scram - recirculation pump trip (ATWS-RPT).

The reactor scram setpoint (1043 psig) prevents reactor vessel overpressurization and, in conjunction with safety/relief valve operation, provides sufficient margin to the maximum allowable reactor coolant boundary pressure. Also, when not in the RUN mode, and below this setpoint, the scrams from Main Steam Isolation Valve closure and low condenser vacuum are bypassed to allow operation in Hot Standby at normal reactor pressure.

The high pressure alarm (1025 psig) alerts the operator to abnormal system pressure.

Water injection by the Core Spray (CS) and Residual Heat Removal (RHR) Systems is delayed until reactor vessel pressure is reduced to 465 and 338 psig to prevent reverse flow and overpressurization of these emergency core cooling systems.

As a part of the RHR initiation logic, the recirculation pump discharge valves close when pressure decreases to 310 psig to ensure RHR water enters the reactor vessel on a recirculation suction line LOCA. Delaying valve closure to this pressure will ensure the valve will be able to close since it is designed to close with a maximum differential pressure of <200 psi.

3.1.4 System Interfaces

The interfaces this system has with other plant systems are listed in Tables 3.1-1 and 3.1-2.

3.1.5 Summary

Classification:
Safety related system

Purpose:
To provide sufficient information concerning reactor vessel water level, reactor vessel pressure, reactor vessel temperature, and core flow rate, to allow for proper plant operation.

Components:
Reactor vessel level instrumentation; reactor vessel pressure instrumentation; reactor vessel temperature instrumentation; and core flowrate.

System Interfaces:
Reactor Vessel System; Recirculation System; Main Steam System; Reactor Core Isolation Cooling System; Feedwater Control System; Recirculation Flow Control System; all Emergency Core Cooling Systems.

TABLE 3.1-1 REACTOR VESSEL LEVEL SETPOINTS AND FUNCTIONS

LEVEL	FUNCTIONS
+56"	High Level Trips Main Turbine and Reactor Feed Pump Turbines. Closes RCIC and HPCI Steam Supply Shutoff Valves.
+40"	High Level Alarm
+37"	Normal Operating Level
+33"	Low Level Alarm, RFC Runback
+12"	Reactor Scram, Primary Containment Isolation, Start Standby Gas Treatment System, ADS Permissive
-38"	Initiate HPCI, RCIC Systems, Recirculation Pump Trip (ATWS-RPT)
-132"	Initiate Core Spray, Residual Heat Removal, ADS and Diesel Generators, and Main Steam Line Isolation

TABLE 3.1-2 REACTOR VESSEL PRESSURE SETPOINTS AND FUNCTIONS

Pressure Setpoint	Functions
1120 psig	Trip Recirculation Pump
1043 psig	Reactor Scram
1025 psig	High Pressure Alarm
465 psig & 338 psig	Core Spray, RHR Initiation & Valve Interlocks
310 psig	Recirculation Pump Discharge Valve Closure on LPCI Initiation Signal
125 psig	RHR Isolation (Shutdown Cooling Mode)

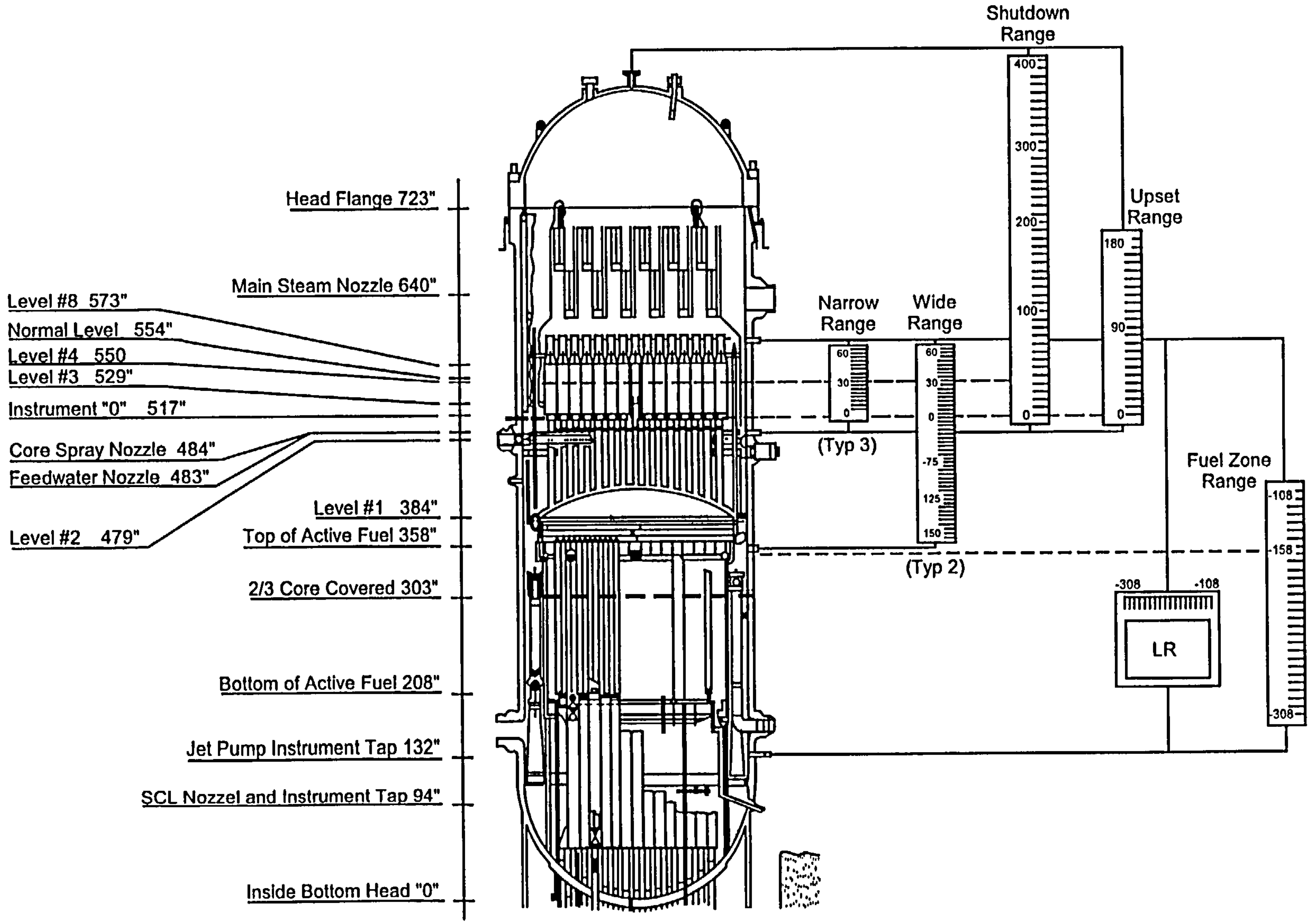


FIGURE 3.1-1 Vessel Level Instrumentation Ranges

0399

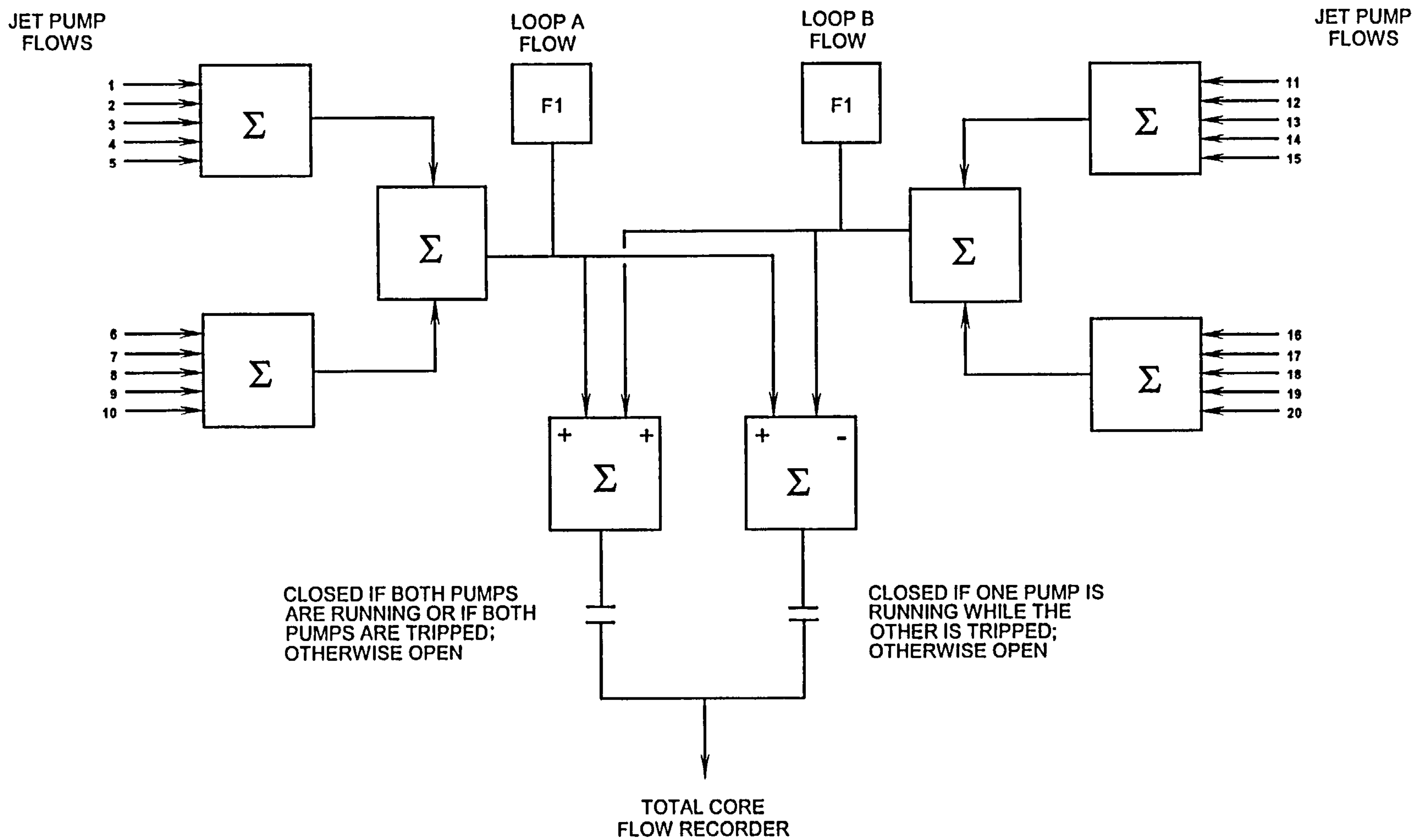


Figure 3.1-2 CORE FLOW SUMMING NETWORK

0102

Table of Contents

List of Figures

3.2 ELECTRO HYDRAULIC CONTROL SYSTEM

The purposes of the Electro Hydraulic Control (EHC) System are to:

1. Provide normal reactor pressure control by controlling the steam flow consistent with reactor power.

2. Provide the ability to conduct a plant cooldown.

The functional classification of the EHC System is that of a power generation system.

3.2.1 System Description

Pressure changes in a direct cycle boiling water reactor can have a pronounced effect on reactor power. If pressure is increased in a BWR during power operation, steam voids, which have significant reactivity effects on the core during power operation, collapse, increasing core moderator content. This increases neutron moderation resulting in more thermal neutrons being available, more fissions occuring and, increasing reactor power. As reactor power increases, pressure tends to increase even further, and a snowball effect is produced.

If reactor vessel pressure decreases, some of the moderator flashes to steam because the reactor vessel is in a saturated state. This flashing increases the void content in the reactor core resulting in more neutron leakage, fewer fissions, and a reduction in reactor power. The power reduction tends to decrease reactor pressure even further.

Because of the effects mentioned above, a pressure control system, the Electro-Hydraulic Control System (EHC), was developed. The EHC System requires reactor power to be changed first, followed by a change in turbine generator output. An increase in reactor power causes an increase in both reactor vessel and turbine throttle pressure. The pressure increase is due to increased heat generation by the reactor core producing more steam without a subsequent increase in steam flow rate. The throttle pressure increase is sensed by the pressure control system and the pressure control system signals the turbine control valves and/or bypass valves to open wider, accommodating the increased steam production. This increase in turbine steam flow compensates for the reactor vessel pressure rise.

Reducing reactor power decreases reactor vessel pressure and turbine throttle pressure. The control system responds to the decrease in throttle pressure by throttling the turbine control valves and/or bypass valves in the closed direction, decreasing turbine steam flow. Reducing steam flow stops the steam pressure decrease and lowers generator output. Using this control system, the turbine follows or is slaved to the reactor.

3.2.2 Component Description

Only the major EHC logic sections are discussed in the paragraphs which follow.

3.2.2.1 Pressure Control Unit

The pressure control unit consists of two pressure regulators and pressure-percent steam flow converters (Figure 3.2-2).

The pressure regulators are the proportional type which require a 30 psi difference between turbine inlet pressure and the pressure setpoint (pressure error) to open the control valves to the 100 percent position. Therefore, the pressure at the turbine inlet varies 30 psi from 0 percent power to full power, or 3.33% flow/psi .

This effect is shown in Figure 3.2-1. Also shown in this figure is a curve for reactor vessel pressure. The curve is not linear primarily because of pressure drops across the flow restrictors, MSIV's, and steam line piping which are proportional to the flow squared. The relationship between pressure error and steam flow was determined by experimentation and given a rapid response which is relatively stable.

The two pressure regulators compare the turbine inlet or throttle pressure with the pressure setpoint, normally set at 920 psi, and generate a steam flow demand based on the error. The controlling regulator is selected by applying +10 psi bias to one of the regulators. The regulator containing the +10 psi is called the backup regulator and will take over if the primary regulator output fails downscale (decreasing valve position demand).

At 100% power conditions, pressure inputs to both regulators would consist of the 920 psig pressure setpoint and a throttle pressure of 950 psig. The primary pressure regulator output would be 30 psi error; with the backup regulator having an output of 20 psi. The 30 psi signal is converted to 100% steam flow demand in the pressure-steam flow converter which is fed to a high value gate (HVG). Both regulator outputs are transmitted to the HVG, which allows the highest input signal to pass. The highest signal is then fed to a demand circuit and the pressure/ load low value gate (LVG).

3.2.2.2 Load Control Unit

The major part of the load control unit is the load set motor. The load set motor is used to set the desired maximum load. Once the load set motor has been adjusted up or down, the load set value will remain constant. The operator controls the position of the load set motor by using the load selector increase or decrease pushbuttons. The output of the load set motor is summed with the speed control unit output and transmitted to the pressure/load LVG. The load set motor is usually adjusted to yield a 100% output signal.

3.2.2.3 Speed Control Unit

The speed control unit consist of two separate speed/acceleration controllers. Each speed/acceleration controller receives a speed signal from a turbine shaft speed pickup unit and compares it to an operator selected speed to produce a speed error signal. The shaft speed signal is differentiated to produce an acceleration signal which is compared to an operator selected acceleration reference signal. The resulting acceleration error signal is then integrated to produce an equivalent speed error signal. The lowest value of the two speed errors and the two integrated acceleration errors is selected and summed with the load set signal. The speed set point and acceleration set point are selected by the operator using pushbuttons.

3.2.2.4 Valve Control Unit

The valve control unit establishes the steam flow demand signals to the control valves and/or the bypass valves. Within the valve control unit is the pressure/load low value gate (LVG) which receives the output from the pressure control unit, the load limit and the combined output from the speed control unit and the load control unit.
The load limit value establishes the maximum amount of rated reactor steam flow which is allowed to go to the turbine and is normally set at 100%. The value of the load limit is determined by a manually adjusted potentiometer. The maximum combined flow limits the total steam flow that can be passed by the control and bypass valves. The maximum combined flow signal is

normally set at 125% by the control room operator using a manual potentiometer.

The output of the pressure/load LVG is the control valve demand. This LVG is the device which makes the turbine a slave to the reactor. The bypass valve demand is established by subtracting the pressure/load LVG output from the pressure control unit output. Any difference passes through the circuit as a bypass valve demand.

3.2.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

3.2.3.1 Normal Operation

The following parameter and controller setpoint values are listed for a reactor power output of 100%.

Reactor power	3283 MWt
Generator power	880 MWe
Reactor pressure	1010 psig
Turbine inlet pressure	950 psig
Pressure setpoint	920 psig
Pressure error	30 psig
Max. Combine flow limit	115%
Load limit	100%
Turbine speed	1800 rpm
Load selector	965 MWe (110% of rated)

The turbine stop valves are open fully, and the control valves are positioned to pass 100% turbine steam flow. The combined intermediate valves (the intercept valve and the stop valve) are fully open supplying low pressure steam to the low pressure turbines.

Assume that the control room operator desires to reduce generator load to 80%. The control room operator starts reducing recirculation flow to reduce core flow. As core flow is decreased, more boiling occurs in the core which causes reactor power and the steam generation rate to decrease. With the control valves still passing 100% steam flow and the reactor producing less that 100% steam flow, reactor pressure decreases. The decrease in reactor pressure causes a decrease in turbine inlet pressure. As turbine inlet pressure decreases the pressure error signal decreases, causing the control valves to begin closing. Finally at 944 psig turbine inlet pressure, the pressure error is reduced to 24 psig which calls for 80% steam flow (24 psig x 3.33/1 = 80%).

3.2.3.2 Plant Shutdown and Cooldown

During a plant shutdown, reactor power is decreased to a point that steam production is within the capacity of the bypass valves to pass, usually about 10% generator load, and then the turbine is tripped by the operator. The turbine stop, control, and combined intermediate valves all trip closed. The bypass valves open in response to the signal generated from the bypass control summer. With a "+10" signal being summed with the "0" signal from the control valve demand, a "+10" steam flow signal is generated and transmitted to the bypass valves. The bypass valves then control reactor pressure during the power decrease.

When the reactor is shutdown, the bypass valves will remain open to control reactor pressure at 920 psig by removing steam at the rate of decay heat output of the reactor. To accomplish a reactor vessel cooldown rate the bypass jack is used to further open the bypass valves. By using the bypass jack to open more bypass valves, more steam is routed to the condenser, thus depressurizing the reactor at a controlled rate determined by the operator. If rate of pressure decrease is controlled in a saturated system, then the temperature decrease is controlled (cooldown rate).

3.2.4 System Interrelations

The interrelations this system has with other plant systems are discussed in the paragraphs that follow.

3.2.4.1 Main Steam System (Section 2.5)

The EHC System senses main steam equalizing header pressure (turbine inlet pressure/turbine throttle pressure) within the main steam system and positions the control valves and/or bypass valves in order to control reactor pressure.

3.2.4.2 Recirculation Flow Control System (Section 7.2)

The EHC System is capable of supplying a load demand signal to the Recirculation Flow Control (RFC) System if the RFC System master controller is in the automatic mode. This arrangement provides for load following capability in the BWR design.

3.2.5 Summary

Classification:

Power generation system

Purpose:

To provide normal reactor pressure control by controlling the steam flow consistent with reactor power and to provide the ability to conduct a plant cooldown.

Components:

Pressure control unit, speed control unit, desired load control unit, valve control unit, hydraulic power unit, Emergency Trip System

System Interrelations:

All systems providing turbine trip signals, Main Steam System, Condensate and Feedwater System, Reactor Feedwater Control, System, RPS, EACPS, DCPS, TBCW System.

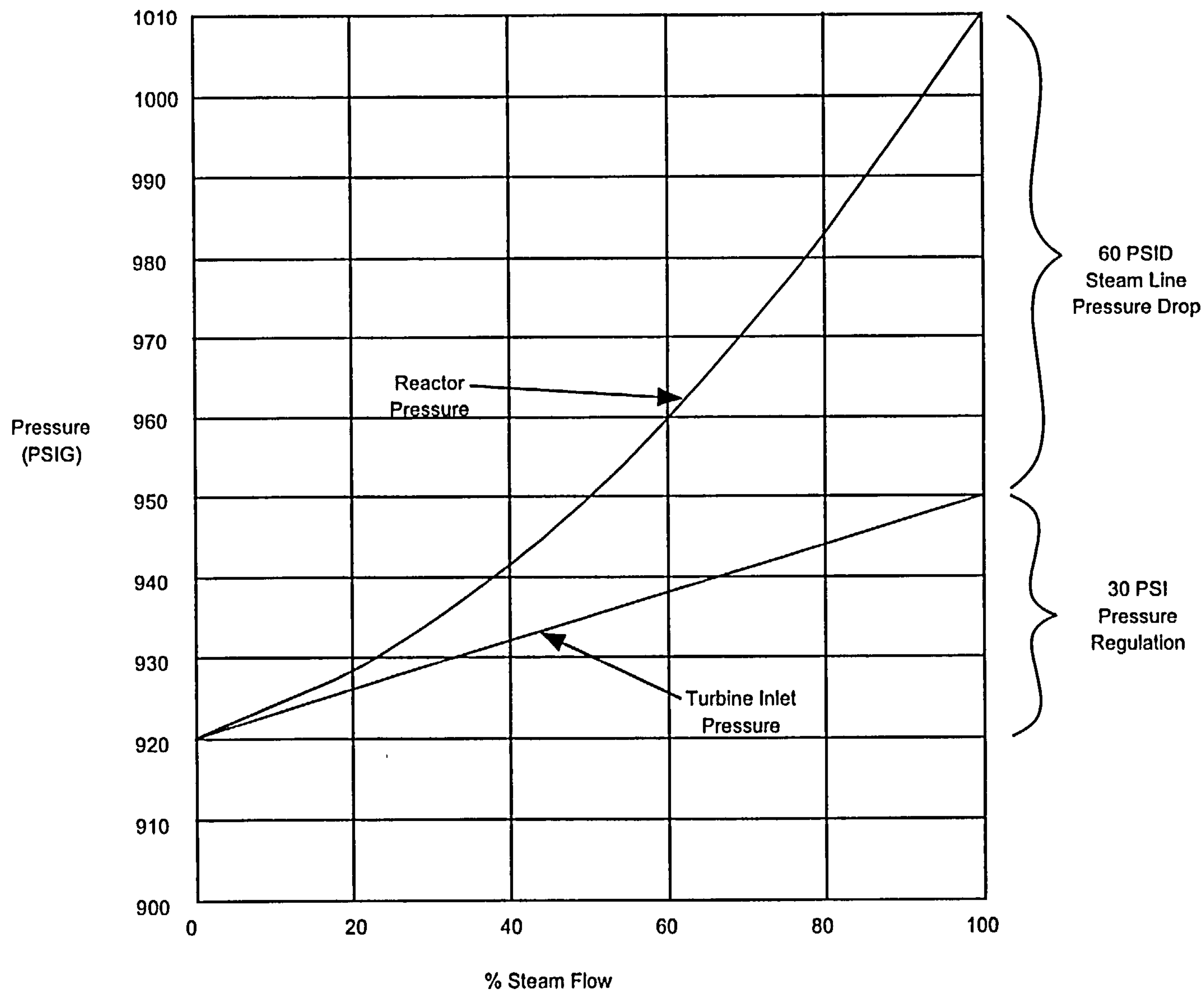


Figure 3.2-1 Pressure Control Spectrum

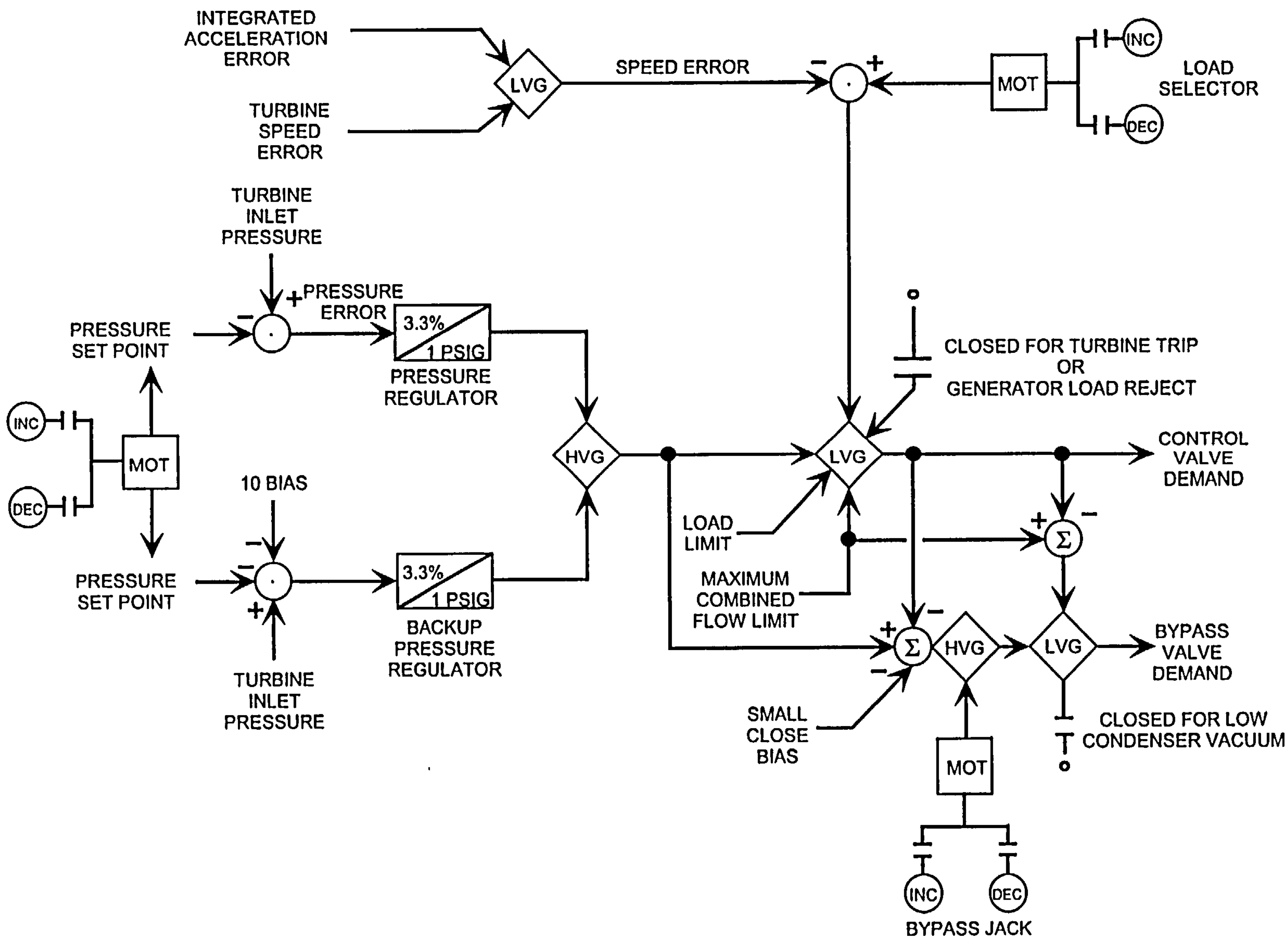


Figure 3.2-2 ELECTRO HYDRAULIC CONTROL SYSTEM LOGIC

0102

Table of Contents

List of Figures

3.3 FEEDWATER CONTROL SYSTEM

The purpose of the Feedwater Control System (FWCS) is to control the rate of feedwater flow, to maintain the proper reactor vessel water level.

The functional classification of the FWCS is that of a power generation system.

3.3.1 System Description (Figure 3.3-1)

During normal operation the Feedwater Control System regulates reactor vessel water level by measuring 3 different parameters: the mass flow rate leaving the reactor vessel (steam), the mass flow rate returning to the vessel (feedwater), and the mass inventory of water in the reactor vessel (level). By measuring the 3 different parameters a signal can be generated to regulate the opening of the feed pump turbine control valves. Opening or closing of the feed pump turbine control valves regulates turbine speed, controlling the pumping effort of the turbine driven reactor feed pumps.

During startup and shutdown operation below 300 psig the rate of feedwater flow to the reactor vessel is controlled with a feed pump bypass valve (startup valve). Due to the low steam flow conditions, reactor vessel water level is the only controlling parameter.

3.3.2 Component Description

The components of the Feedwater Control System consists of the necessary sensors, summers, controllers and circuitry needed to control reactor water level during all modes of reactor operation.

3.3.2.1 Reactor Water Level Instrumentation

Reactor water level is measured by three independent level transmitters (LT) with a range of 0 to 60 inches above instrument zero, and each with an indicator displayed in the control room. Level indicators A, B, and C are used to provide a reliable system for the high water level trip of the main turbine and reactor feed pump turbines. Any two out of the three level instruments must be tripped to satisfy the logic coincidence. Either level indicator A or B may also be selected as an input, via a level selector switch, to the flow error/level summer.

3.3.2.2 Total Steam Flow

Steam flow is calculated in each of the four steam lines by measuring the differential pressure across a flow restrictor. The calculated steam flow signal is then fed into a four input summer which develops a total steam flow signal.

The total steam flow signal is used as an input to the steam flow/feed flow summer and the Rod Worth Minimizer System (Section 6.1).

3.3.2.3 Total Feedwater Flow

Feedwater flow is measured by venturi flow elements located in the two feedwater lines penetrating the drywell (Section 2.6). The output signals from flow transmitters are sent to a feedwater flow summer which generates a total feedwater flow signal. The total feedwater flow signal is used as an input to the steam flow/feed flow summer, and is used by:the Recirculation Flow Control System (Section 7.2) and the Rod Worth Minimizer (Section 6.1).

3.3.2.4 Steam Flow/Feed Flow Summer

The feedwater flow summer output (-signal) and steam flow summer output (+ signal) are sent to the steam flow/feed flow summer where they are summed to produce a base signal for the FWCS. If steam flow and feed flow are not equal, this summer will produce a signal either greater than or less than the base signal. The algebraic signs are such that when steam flow exceeds feedwater flow, the output signal will modify the level signal to indicate the need for additional feedwater flow. Thus, an anticipatory signal is developed which will correct for projected changes in level due to process flow changes. This anticipatory signal corrects feedwater flow to lessen the effect of changes on reactor level due to a change in steam demand.

3.3.2.5 Level/Flow Summer

The output of the steam flow/feed flow summer (a flow error signal) is compared with the selected reactor water level signal to produce an output signal referred to as the modified level signal. The flow error signal provides anticipation of the change in the reactor vessel water level that will result from a change in load. The level signal provides a reference for any mismatch between the steam flow and feed flow that causes the level to rise or fall.

3.3.2.6 Master Level Controller

The master level controller is provided to control any one or all three reactor feed pumps to achieve the desired feed water flow. Both single element and three element control modes of operation are available as determined by the FWCS mode selector switch.

3.3.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

3.3.3.1 Single Element Control

Single element control is used during a startup, shutdown, or malfunction of the normal three element control. With single element selected, the selected level signal is compared to the desired level of either the master level controller or the startup level controller. The controllers process the signal in the same manner as described in Section 3.3.3.2. Single element control is less responsive to changes because the anticipatory response provided by the steam flow/feed flow error signal has been removed.

Unlike the master level controller ,the startup level controller only uses the single element control mode to position a startup valve.

3.3.3.2 Three Element Control (Automatic Mode)

The modified three element level control signal is compared to the desired level established by the control room operator using the level setpoint adjustment located on the face of the master controller. The master controller then produces the required level signal which is transmitted to all three reactor feed pump manual/auto controllers.

The reactor feed pump manual/auto controllers receive the required level signal and send it directly to the feed pump turbine speed control circuit. The feed pump turbine speed control circuit receives the input signal and uses it to open or close steam admission valves (control valves) to change turbine/pump speed.

3.3.4 System Interfaces

A short discussion of interfaces this system has with other plant systems is given in the paragraphs which follow.

3.3.4.1 Condensate and Feedwater System (Section 2.6)

The Feedwater System flow elements provide the feedwater flow input to the FWCS. The FWCS controls the speed of the RFPT's, position of the startup bypass valve, and provides a trip signal to the RFPT's.

3.3.4.2 Main Steam System (Section 2.5)

The Main Steam System steam line flow restrictors provide the steam flow input to the FWCS. The FWCS provides a trip signal to the main turbine.

3.3.4.3 Recirculation Flow Control System (Section 7.2)

The Recirculation Flow Control System receives water level and feedwater flow interlock signals from the FWCS.

3.3.4.4 Reactor Vessel Instrumentation System (Section 3.1)

The FWCS receives reactor vessel water level input signals from the Reactor Vessel Instrumentation System.

3.3.4.5 Rod Worth Minimizer (Section 6.1)

The FWCS provides total steam flow and total feed flow signals to the Rod Worth Minimizer bypass circuits.

3.3.5 Summary

Classification:

Power generation system

Purpose:

To control the rate of feedwater flow, to maintain the proper reactor vessel water level.

Components:

Reactor water level instrumentation; total steam flow; total feed flow; steam flow/feed flow summer; level/flow summer; master level controller; startup level controller.

System Interfaces:

Condensate and Feedwater System; Main Steam System; Recirculation Flow Control System, Rod Worth Minimizer.

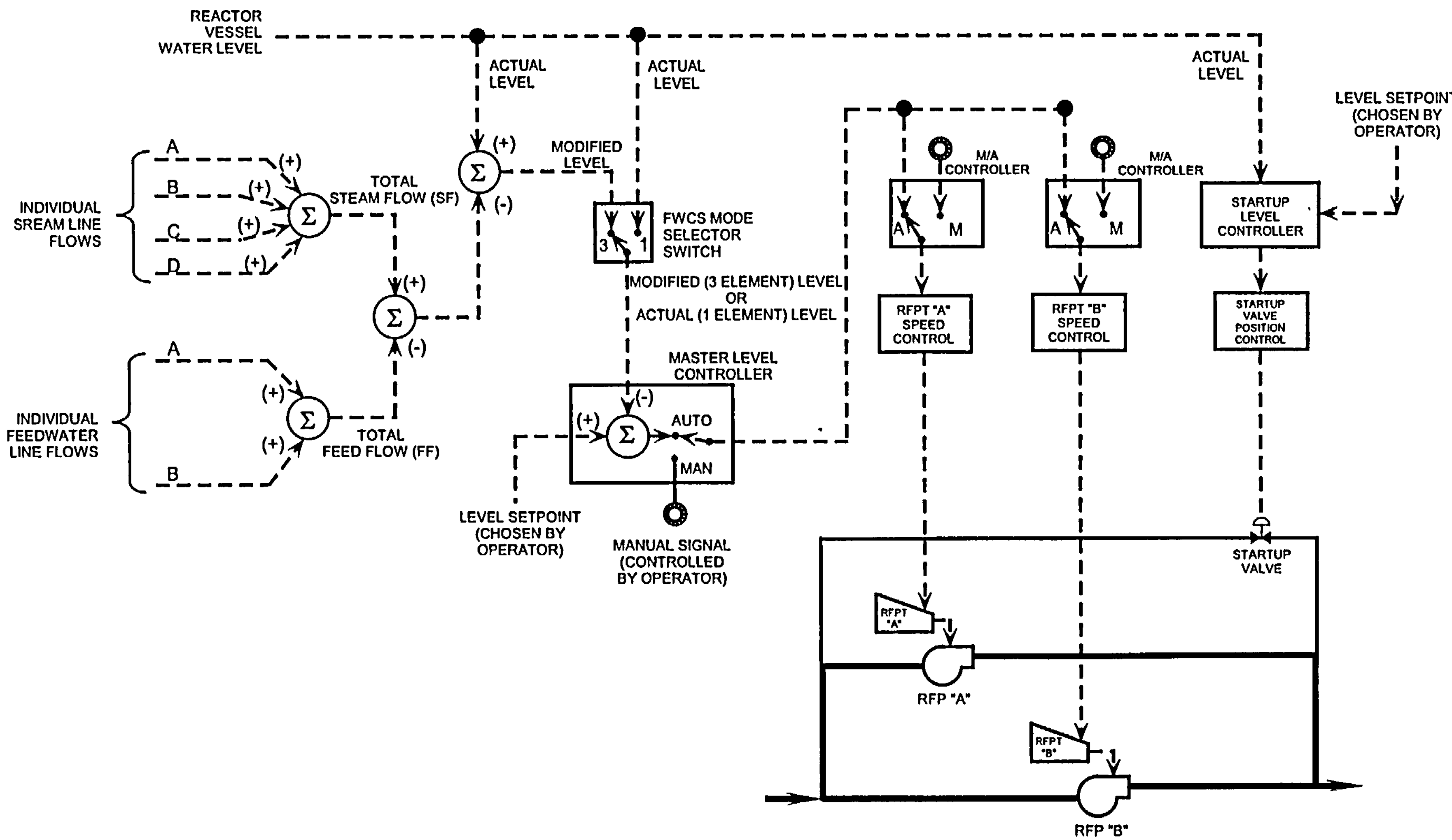


Figure 3.3-1 FEEDWATER CONTROL SYSTEM

0102

4.0 CONTAINMENT SYSTEMS

The containment systems shown in Figure 4.0-1 provide a multibarrier pressure suppression type containment. The fuel, fuel cladding, and reactor coolant system form initial barriers to the release of fission products. This chapter describes a containment system which encloses the Reactor Coolant System and is composed of a Primary Containment (the pressure suppression system), a Secondary Containment (the reactor building), a Standby Gas Treatment System and a Primary Containment Isolation System.

4.0.1 Primary Containment System (Section 4.1)

The Primary Containment consists of a drywell which encloses the reactor vessel and recirculation system, a pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and the suppression chamber, isolation valves and containment atmosphere control systems.

4.0.2 Secondary Containment System (Section 4.2)

The Secondary Containment consists of the reactor building, which serves as the secondary containment membrane; and low leakage dampers and valves used to isolate the secondary containment. The reactor building serves as a dilution and holdup volume for fission products which may leak from the primary containment following an accident.

The reactor building houses most of the reactor auxiliary and support systems including the Emergency Core Cooling Systems, the Reactor Water Cleanup System, the Control Rod Drive System, and the refueling and support facilities.

4.0.3 Standby Gas Treatment System (Section 4.3)

The Standby Gas Treatment System processes exhaust air from the secondary containment under accident conditions to limit radiation dose rates to less than 10 CFR 100 guidelines. The Standby Gas Treatment System is also used to purge the primary containment and leak test the secondary containment.

4.0.4 Primary Containment Isolation System (Section 4.4)

The Primary Containment Isolation System is used to automatically isolate the primary and secondary containments and reactor vessel during abnormal or accident conditions. This is done to prevent the release of radioactive materials in excess of specified limits.

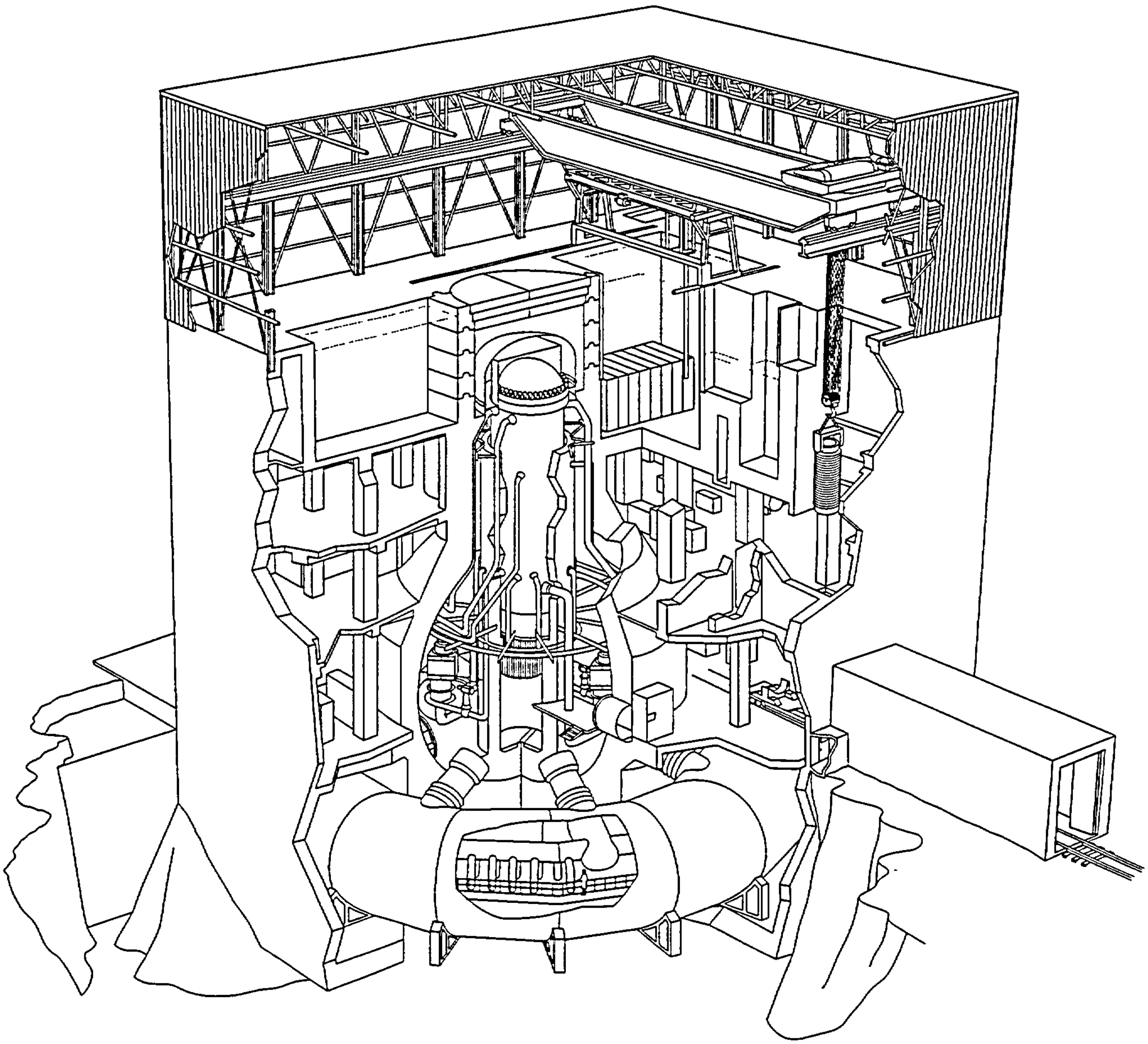


Figure 4.0-0 Mark I Containment

Table of Contents

List of Tables

List of Figures

4.1 PRIMARY CONTAINMENT SYSTEM

The purposes of the Primary Containment are to:

1. Contain fission products released from a loss of coolant accident (LOCA) so that off site radiation dose limits are not exceeded.

2. Provide a heat sink for certain safety related equipment.

3. Provide a source of water for Emergency Core Cooling Systems and the Reactor Core Isolation Cooling System.

The functional classification of the Primary Containment System if that of a safety related system. Its' regulatory classification is that of an engineered safety feature (ESF) system.

4.1.1 System Description

The primary containment consist of a pressure suppression type containment which houses the reactor vessel, the Recirculation System and other branch connections to the reactor coolant system boundary.

In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell increasing drywell pressure. This increase in pressure will force a mixture of steam, water, and drywell atmosphere through vents into a pool of water that is stored in the suppression chamber. The steam entering the suppression pool condenses, thus limiting the drywell pressure rise. Noncondensibles that enter the suppression pool accumulate in the air volume above the pool, increasing the suppression chamber pressure. When the suppression chamber pressure increases to a value greater than drywell pressure, the noncondensibles are vented back to the drywell.

Cooling systems are provided to remove heat from the drywell and suppression pool during normal and accident conditions.

Isolation valves are actuated during abnormal conditions, by the Primary Containment Isolation System (Section 4.4), to ensure containment of radioactive materials.

The primary containment system also serves as a heat sink for the 13 safety/relief valves, Reactor Core Isolation Cooling System (Section 2.7) and the High Pressure Coolant Injection System (Section 10.1). Besides serving as a heat sink for various systems, the suppression pool also serves as a source of water for all the Emergency Core Cooling Systems (Chapter 10) and the Reactor Core Isolation Cooling System.

4.1.2 Component Description

The components which comprise the Primary Containment System are discussed in the following paragraphs.

4.1.2.1 Drywell

The drywell, shown in Figure 4.1-1, is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion with a removable head. The drywell is enclosed in reinforced concrete for shielding and for additional resistance to deformation and buckling over areas where the concrete backs up the steel shell. Above the foundation transition zone, the drywell is separated from the reinforced concrete by a gap of approximately two inches for thermal expansion. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

Located within the drywell, below the drywell head and encompassing the reactor vessel, is a one inch steel bulkhead plate (Figure 4.1-1). During periods of fuel transfer requiring flooding of the reactor cavity, the bulkhead prevents leakage into the containment. Cooling in the upper head region of the drywell is provided by two ventilation supply and return air ducts during normal power operation. In addition to the drywell head, one double-door personnel air lock Figure 4.1-2 and two bolted equipment hatches are provided for access to the drywell. The locking mechanisms on each airlock door are designed so that a tight seal will be maintained when the doors are subjected to internal pressure. The doors are mechanically interlocked so that a door may be operated only if the other door is closed and locked. Handwheels are provided inside the airlock and exterior to the airlock within the drywell and the reactor building, to open or close either door. The seals on the doors are capable of being tested for leakage.

4.1.2.2 Pressure Suppression Chamber (Torus)

The pressure suppression chamber, shown in Figure 4.1-1 , is a steel pressure vessel in the shape of a torus located below and encircling the drywell. The suppression chamber is held on supports which transmit vertical and seismic loading to the reinforced concrete foundation slab of the reactor building. Eight circular vent pipes form a connection between the drywell and the pressure suppression chamber. The 81 inch diameter vent pipes are provided with expansion joints to accommodate differential motion between the drywell and suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces which might accompany a pipe break in the drywell. The vent pipes exhaust into a 57 inch diameter continuous vent header, from which 96 downcomer pipes extend into the suppression chamber pool. Each downcomer pipe has a 24-inch inside diameter. The pipe exit is approximately three feet below the pool water level.

The pressure suppression chamber serves not only as a heat sink for blowdown from the drywell after an accident but also as a source of water or heat sink for the following systems:

1. Core Spray System injection and source of water for testing (Section 10.3).
2. Low pressure coolant injection mode of Residual Heat Removal System and source of water for testing (Section 10.4).
3. HPCI and RCIC Pumps alternate source of water (HPCI, Section 10.1 and RCIC, Section 2.7).
4. Heat sink for steam blown down from safety/relief valves (Section 2.5).
5. Heat sink for HPCI turbine and RCIC turbine exhaust steam.

4.1.2.3 Vacuum Relief System

Protection of the primary containment from exceeding the design maximum external pressure of 2 psi is provided by a system of self-actuating swing check vacuum relief valves.

Twelve (12) vacuum relief valves, shown in Figure 4.1-1 are located on the torus vent header with direct flow into the drywell.These drywell/suppression chamber valves are intended to bleed non-condensible gases from the suppression chamber atmosphere into the drywell and will completely open within one second after a 0.5 psi differential pressure is applied across the seat.

Two valves in series, shown in Figure 4.1-3, are used in each of two lines from the reactor building atmosphere to the air space above the suppression pool. The first valve outboard of the suppression chamber is a self actuating check valve. The second is an air operated butterfly type vacuum breaker. The reactor building/suppression chamber valves are intended to bleed air from the Reactor Building into the suppression chamber after a differential pressure of 0.5 psi is reached.

4.1.2.4 Penetrations

Primary containment penetrations are essential for plant operation. All penetrations are designed and constructed to withstand the drywell pressures and temperatures without loss of leak tight integrity of the primary containment structure.

4.1.2.5 Isolation Valves

Isolation valves are provided to assure containment of radioactive materials which might be released from the containment during the course of an accident. The basic criterion is to provide a minimum of two isolation valves between the reactor core and the environmental surroundings. Further information on the types of isolation valves and the Primary Containment Isolation System is contained in Section 4. 4.

4.1.3 System Features

A short discussion of system features is given in the paragraphs which follow.

4.1.3.1 Normal Operation

Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment, which are not required to be open during accident conditions, are closed. These valves maybe opened to perform necessary opertional activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolation position.

4. All blind flanges and manways are closed.

In general, primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel.

4.1.3.2 Primary Containment Atmosphere

During normal plant operation, the drywell and suppression chamber are isolated to provide a closed atmosphere. The Primary Containment atmosphere consists of more than 96% nitrogen, less than 4% oxygen and is less than 150°F.

4.1.3.3 Nitrogen Inerting

The drywell and suppression chamber free air space are inerted with nitrogen to minimize the possibility of a hydrogen explosion during or following a loss of coolant accident. The containment is purged of air by introducing pure nitrogen via the liquid nitrogen storage tanks, a purge vaporizer, a pressure reducing valve and controller (Figure 4. 1-3).

Gases purged from the containment are vented through the Standby Gas Treatment System (Section 4.3) or the Reactor Building Ventilation System (Section 4.2). To reduce the containment oxygen level to less than 4% requires approximately 5 hours and about 5 containment atmosphere volumetric changes.

4.1.3.4 Temperature Control

Temperature control of the drywell is provided to ensure proper operation of instrumentation, valves and sensors. The temperature of the drywell is maintained by ten drywell cooling units shown in Figure 4.1-3. The cooling units consist of a fan and heat exchanger cooled by the Reactor Building Closed Cooling Water System (Section 11.1).

4.1.3.5 Drywell/Suppression Chamber Differential Pressure Control

To reduce the consequences of upward and downward loads on the suppression chamber during the initial vent header and downcomer pipe clearing phenomena of a LOCA, a differential pressure is maintained between the drywell and suppression chamber. The differential is maintained between 1.3 and 1.5 psid, by an air compressor removing nitrogen from the suppression chamber and pumping it to the drywell.

4.1.3.6 Shutdown Ventilation

During certain plant conditions, it may be necessary to establish ventilation flow to the primary containment. This is accomplished with a series of supply and exhaust fans as shown in Figure 4.1-3.

The Reactor Building supply fans can be used to supply filtered and tempered outdoor air to the containment for purge and ventilation purposes, allowing personnel access and occupancy during reactor shutdown and refueling.

The purge, or exhaust air, is discharged to atmosphere through the reactor building ventilation stack via the reactor building exhaust fans. If the exhaust air is contaminated, the containment air is exhausted through the Standby Gas Treatment System (Section 4.3) for cleanup, prior to release to the atmosphere.

4.1.3.7 Containment Atmosphere Dilution

Following a loss-of-coolant-accident, hydrogen and oxygen are evolved within the containment from two significant sources:hydrogen and oxygen produced from radiolysis of water, and hydrogen produced by a metal-water reaction between Zircaloy and water. If the concentrations of hydrogen and oxygen were to continue unchecked, a combustible gas mixture could be produced despite the initially inert atmosphere. To ensure that a combustible mixture is not formed, the oxygen concentration is kept below 5% by volume and the hydrogen concentration is kept below 4% by volume via the Containment Atmospheric Dilution(CAD)System.
The CAD System nitrogen supply facilities include two trains, each of which is capable of supplying nitrogen through separate piping systems to the drywell and suppression chamber (Figure 4.1-3). Each train includes a liquid nitrogen supply tank, an ambient vaporizer, an electric liquid heater, a manifold with branches to the primary containment, and pressure, flow, and temperature controls.

Each nitrogen storage tank has a normal capacity of 3000 gallons which is adequate for the first 7 days of CAD operation.

CAD system operation is performed manually by the control room operator. Following a LOCA, hydrogen and oxygen concentrations are recorded along with pressures inside of the primary containment to calculate the production rates of hydrogen and oxygen. Nitrogen additions are made periodically as needed to maintain the oxygen and hydrogen levels below the explosive limits. Gas releases are also performed periodically to ensure containment pressure remains below 30 psig.

4.1.3.8 Containment Response to a LOCA

Accidents that could cause a release of radioactive material directly into the primary containment result from postulated nuclear system pipe breaks inside the drywell. All possibilities for pipe break sizes and locations are analyzed including the severance of small pipe lines, the main steam lines upstream and downstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop pipelines.
The following describes how the pressure suppression system functions during a major piping failure within the drywell.

The reactor water and steam would be released into the drywell air space and cause the pressure to increase. As the drywell pressure rises to the isolation setpoint, the isolation valves close to seal off the drywell. The pressure buildup then drives the mixture of nitrogen, steam, and water down through the vents and into the downcomers. The steam is condensed as it bubbles through the water, thus causing a rapid reduction in pressure. The noncondensibles gather in the top of the suppression chamber. As pressure is reduced in the drywell due to condensation of the steam atmosphere in the drywell, it is possible to create a negative pressure in the drywell exceeding the design external pressure. Also, as drywell pressure falls below torus pressure, the expansion of the gas in the torus tries to force the water up the downcomer and into the drywell. To prevent this, the pressure suppression chamber-to-drywell vacuum breakers open and vent to the drywell and limit the pressure differential.

As the temperature in the suppression chamber pool and the pressure in the drywell steady out, the containment cooling mode of the Residual Heat Removal (RHR) System (Section 10.4) could be used to cool the system. The pool temperature may be reduced by direct recirculation and the drywell pressure reduced by spraying.

4.1.4 System Interfaces

A short discussion of interfaces this system has with other plant systems is given in the paragraphs which follow.

4.1.4.1 Standby Gas Treatment System (Section 4.3)

The Standby Gas Treatment System can be aligned to take suction on the drywell or suppression chamber free air volume.

4.1.4.2 Main Steam System (Section 2.5)

The Main Steam System penetrates the drywell and the suppression pool serves as a heat sink for the safety/ relief valves.

4.1.4.3 Reactor Core Isolation Cooling System (Section 2.7)

The suppression pool serves as a heat sink for the Reactor Core Isolation Cooling (RCIC) turbine exhaust steam and as an alternate water source for the RCIC pump.

4.1.4.4 High Pressure Coolant Injection System (Section 10.1)

The suppression pool serves as a heat sink for the High Pressure Coolant Injection (HPCI) turbine exhaust steam and as an alternate water source for the HPCI pump.

4.1.4.5 Core Spray System (Section 10.3)

The suppression pool provides the water source for the Core Spray System.

4.1.4.6 Residual Heat Removal System (Section 10.4)

The suppression pool provides the water source for the Residual Heat Removal (RHR) System. The RHR System has containment spray and suppression pool cooling modes of operation.

4.1.4.7 Primary Containment Isolation System (Section 4.4)

The Primary Containment System receives isolation demand signals from the Primary Containment Isolstion System.

4.1.5 Summary

Classification:

Safety related system. Engineered safety feature system.

Purpose:

1. To contain fission products released from a loss of coolant accident (LOCA) so that off site radiation doses limits are not exceeded.
2. To provide a heat sink for certain safety related equipment.
3. To provide a source of water for Emergency Core Cooling Systems and the Reactor Core Isolation Cooling System.

Components:

Drywell; suppression chamber; vacuum relief valves; CAD; containment air coolers; vent and purge components.

System Interfaces:

Main Steam System; High Pressure Coolant Injection System; Core Spray System; Residual Heat.Removal System; Reactor Core Isolation Cooling System; Standby Gas Treatment System.

TABLE 4.1-1 MARK I CONTAINMENT TYPICAL SPECIFICATIONS

Drywell	
Diameter (Spherical Section)	67 feet
Diameter (Cylindrical Section)	38 feet 6 inches
Height	115 feet
Free Air Volume	159,000 ft^3
Maximum Internal Design Pressure	62 psig
Maximum External Design	2 psig
Design Temperature	281°F
Suppression Chamber	
Major Diameter	111 feet
Minor Diameter	31 feet
Water Volume	135,000 ft^3
Free Air Volume	119,000 ft^3
Maximum Internal Design Pressure	62 psig
Maximum External Design Pressure	2 psig
Vent Pipes (8) Drywell To Suppression Pool	81 inches
Design Temperature	281°F

Primary Containment - .5% of free air volume per day
Design Leak Rate (at 62 psig)

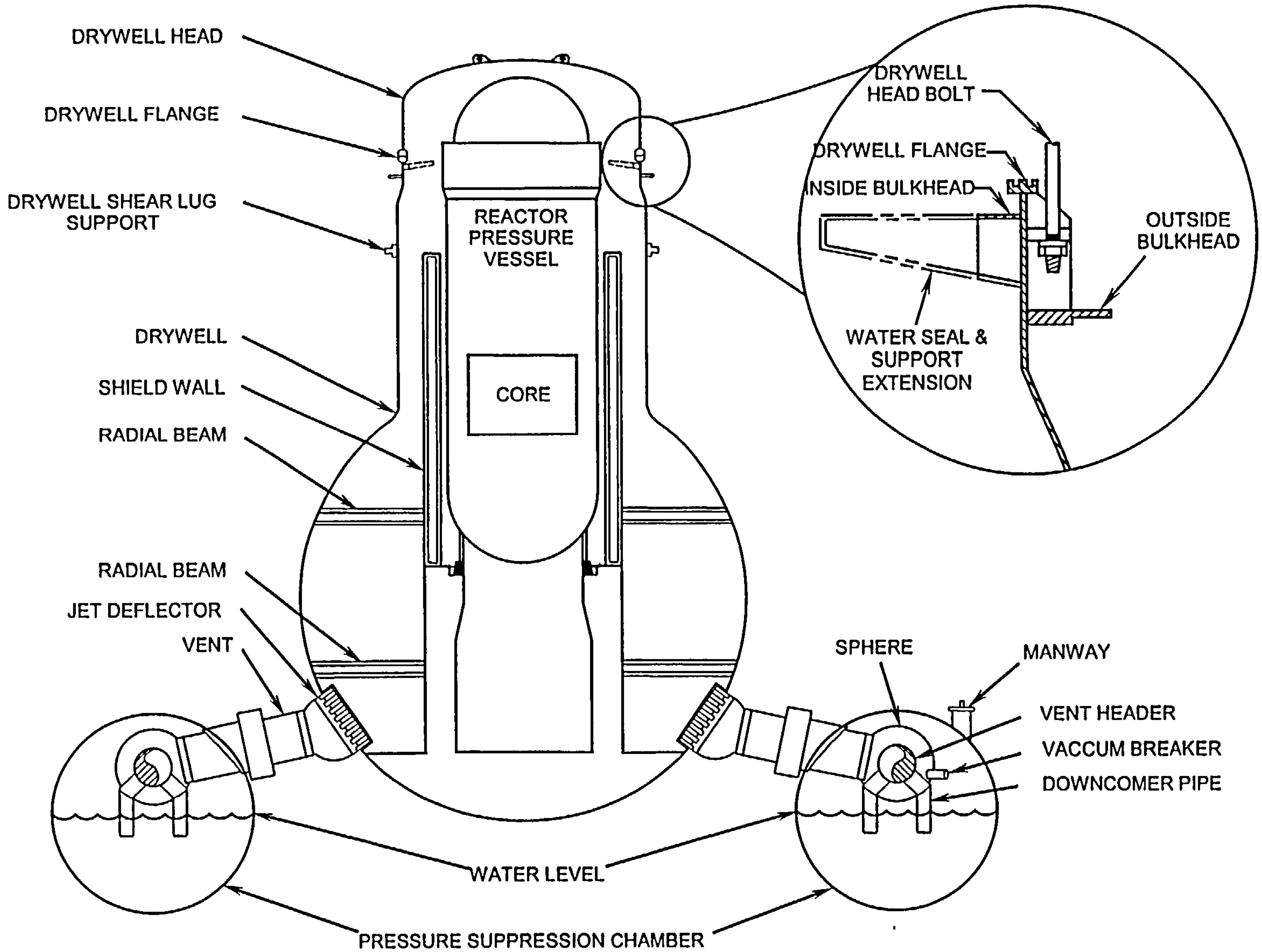
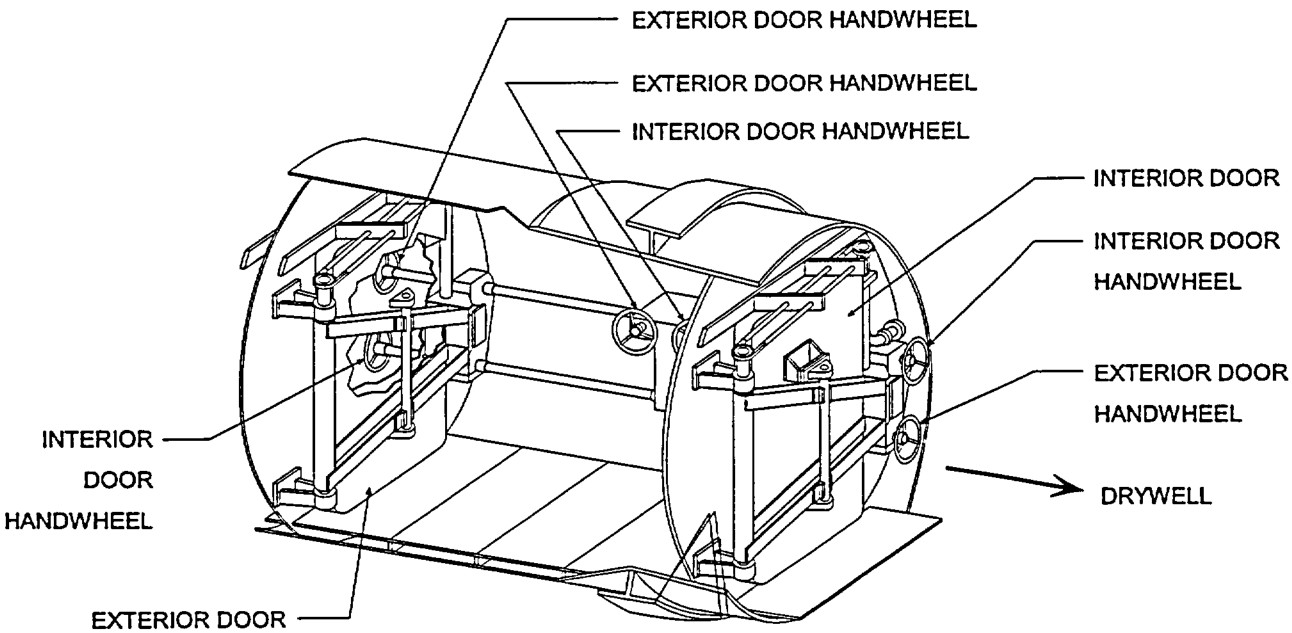
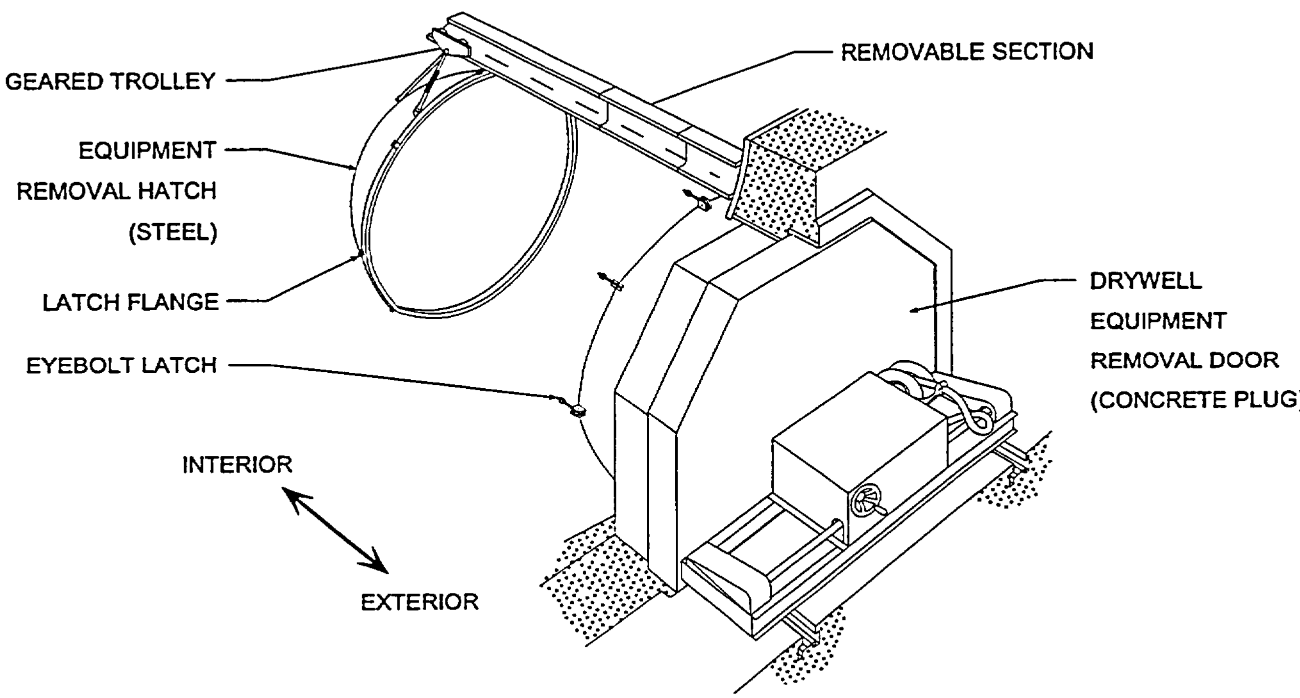


Figure 4.1-1 Primary Containment System

1197-4



DRYWELL PERSONNEL AIR LOCK



DRYWELL EQUIPMENT HATCH

Figure 4.1-2 Drywell Access Penetrations

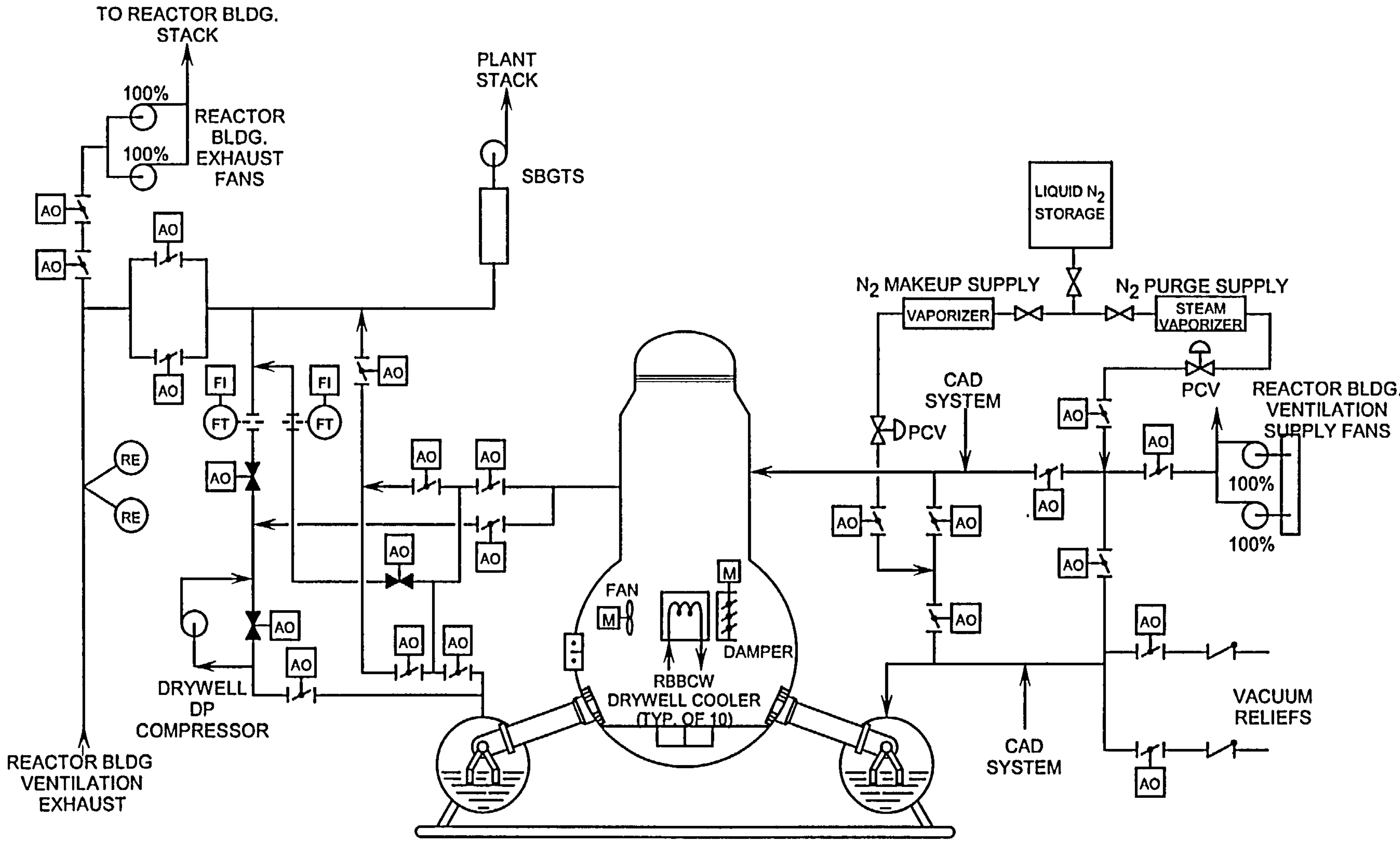


Figure 4.1-3 Primary Containment Ventilation and Inerting System

0102

Table of Contents

List of Figures

4.2 SECONDARY CONTAINMENT SYSTEM

The purposes of the Secondary Containment are:

1. To minimize the ground level release of airborne radioactive materials following an accident.

2. To serve as the primary containment during refueling operations.

The functional classification of the Secondary Containment System is that of a safety related system. Its regulatory classification is that of an engineered safety feature (ESF) system.

4.2.1 System Description

The secondary containment (reactor building), shown in Figure 4.2-1, is a physical boundary which encloses the primary containment. The reactor building also houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, Emergency Core Cooling Systems, Reactor Water Cleanup System, Standby Liquid Control System, Control Rod Drive System, instrumentation for the Reactor Protection System and electrical components.

During normal operation the reactor building atmosphere is maintained at a 0.25 inch of water vacuum to assure any leakage would be inleakage. Outside air is brought in through isolation valves and filters by reactor building supply fans and then distributed to the various rooms within the reactor building. Reactor building exhaust fans take suction on the various rooms and discharge through isolation valves to an elevated release stack.

The isolation valves are actuated during abnormal conditions by the Primary Containment Isolation System (Section 4.4), to ensure containment of radioactive materials.

4.2.2 Component Description

The components which comprise the Secondary Containment System are discussed in the following paragraphs.

4.2.2.1 Reactor Building

The reactor building substructure consists of poured in place reinforced concrete which extends up to and including the refueling floor. The superstructure of the reactor building, above the refueling floor, is a structural steel frame.

The reinforced concrete exterior walls and the structural steel for the superstructure are designed for tornado considerations and missile protection. In addition, relief (blowout) panels are installed to prevent excessive internal reactor building pressure.

4.2.2.2 Air Locks and Penetrations

All entrances and exits to and from the reactor building are through double door personnel and equipment air locks. Each access door is equipped with weather strip type rubber construction seals and are electrically interlocked so that only one door, of a pair, may be open at a time.

Secondary containment piping penetrations are provided with a means for isolating the piping to prevent and/or minimize any release to the environment.

4.2.2.3 Ventilation

Reactor building ventilation (Figure 4.2-2) is provided by equipment needed to process the influent air supply such as filters, hot water coils and coolers; supply and exhaust fans, dampers and controllers that ensure sufficient air flow and maintain the reactor building atmosphere at a quarter inch negative pressure; and isolation valves to minimize radioactivity release.

To ensure proper operation of the Emergency Core Cooling Systems located within the secondary containment, air cooling units are provided to remove the heat generated by the motors, pumps and piping. There is one air cooling unit provided for each Residual Heat Removal Pump and one for each Core Spray System.

4.2.3 System Features

A short discussion of system features is given in the paragraphs which follow.

4.2.3.1 Normal Operation

Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. The Standby Gas Treatment System is operable.
2. All automatic ventilation system isolation valves are operable or secured in the isolated position.
3. At least one door in each access opening is closed.

In general, secondary containment integrity must be maintained at all times and can be broken only under specified conditions described in the Technical Specifications. The following typical requirements must be met when secondary containment integrity is broken:

1. Reactor subcritical by a specified amount.
2. The reactor cooled down below 212°F and the reactor coolant system vented.
3. No activity is being performed that will reduce the shutdown margin below that specified.
4. Handling of spent fuel is prohibited.

4.2.3.2 Abnormal Operation

In the event of a loss of coolant accident or refueling accident, all isolation ventilation dampers automatically close and remain closed for the duration of the isolation signal(s). Supply and exhaust fans trip with the same signal that starts the Standby Gas Treatment System. To mitigate the consequences of the postulated accidents three different features are utilized.

The first feature is the negative internal pressure being maintained in the secondary containment, which minimizes the ground level release of fission products by exfiltration. The secondary containment structure along with the Primary Containment Isolation System and the Standby Gas Treatment System together provide this feature.

The second feature is a low leakage containment volume, which provides a holdup time for fission product decay prior to release.

The third feature is the removal of particulates and iodines by filtration prior to release. This feature is provided by isolating the normal ventilation supply and exhaust, and routing the secondary containment atmosphere to the Standby Gas Treatment System.

4.2.4 System Interfaces

The Secondary Containment interfaces with other plant systems which are discussed in the following paragraphs.

4.2.4.1 Standby Gas Treatment System (Section 4.3)

The Standby Gas Treatment System processes the secondary containment atmosphere under isolated conditions and provides leak testing capabilities.

4.2.4.2 Primary Containment (Section 4.1)

The secondary containment serves as the primary containment under specified conditions.

4.2.4.3 Primary Containment Isolation System (Section 4.4)

The Secondary Containment receives isolation signals from the Primary Containment Isolation System.

4.2.5 Summary

Classification:

Safety related system, engineered safety feature.

Purpose:

To minimize the ground level release of airborne radioactive materials following an accident. To serve as the primary containment during refueling operations.

Components:

Reactor Building; ventilation supply and exhaust fans; air cooling units; isolation dampers and valves.

System Interfaces:

Primary Containment; Standby Gas Treatment System; Primary Containment Isolation System.

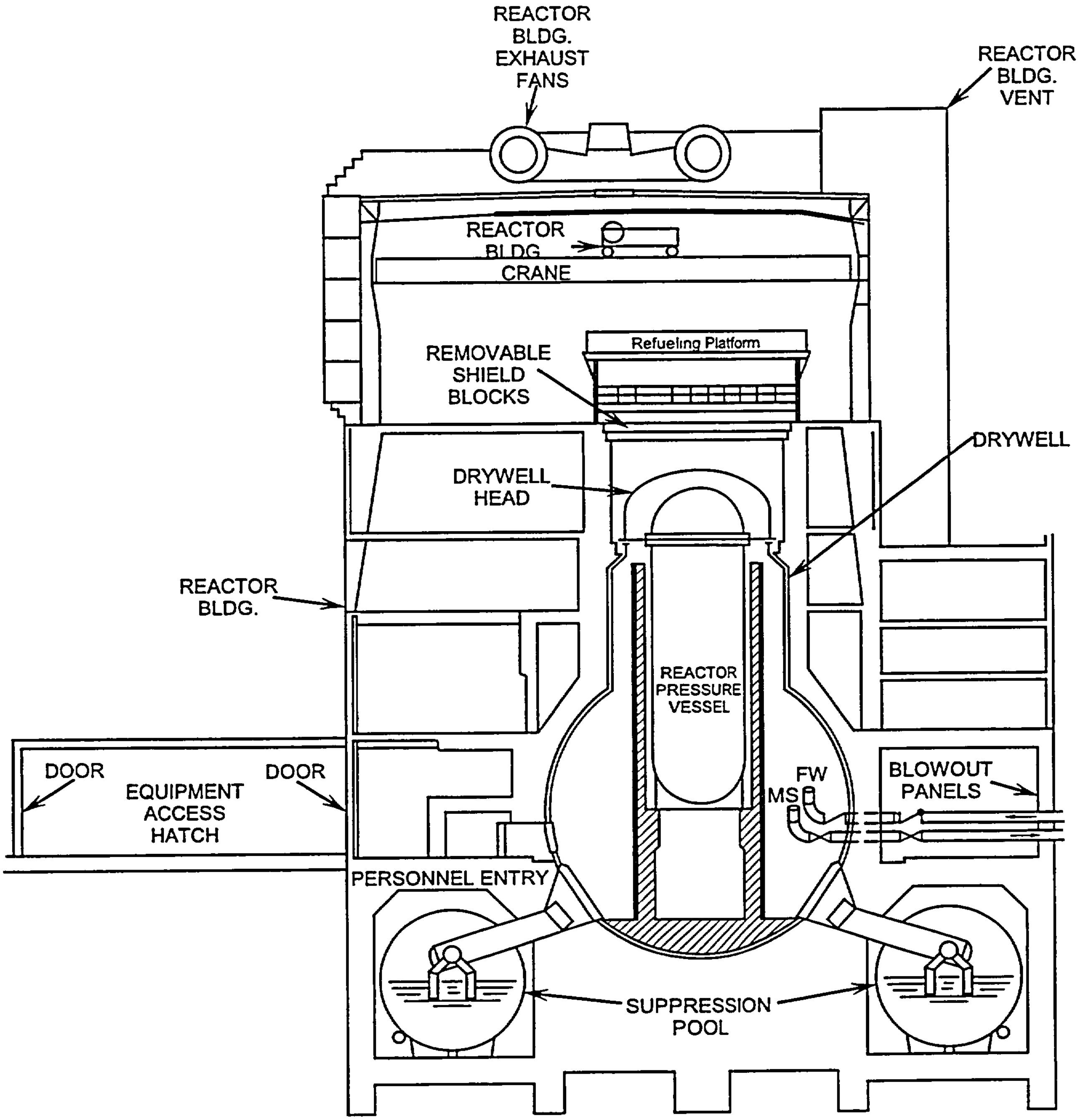


Figure 4.2-1 Secondary Containment

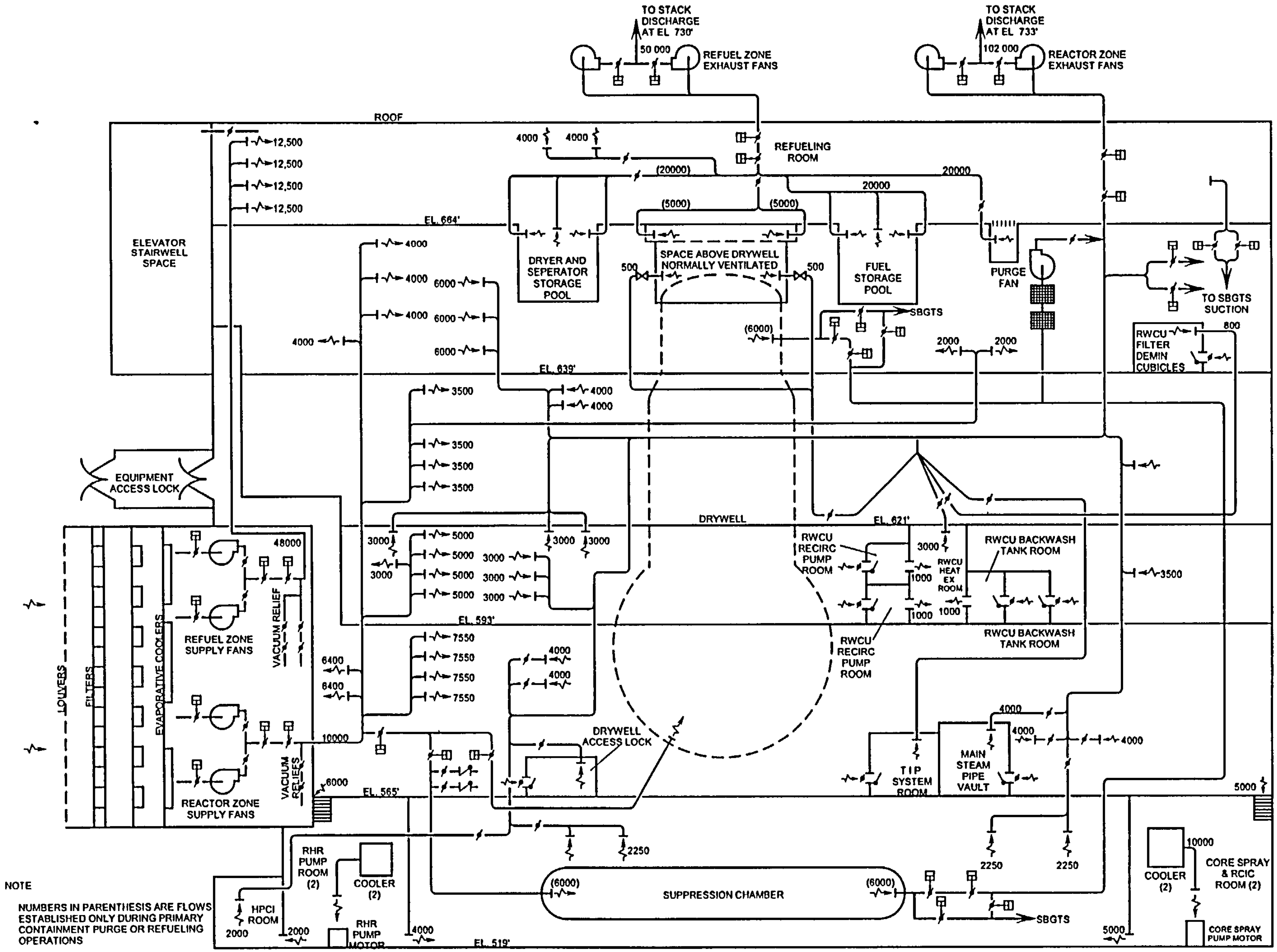


Figure 4.2-2 Secondary Containment Ventilation System

1297-4

Table of Contents

List of Figures

4.3 STANDBY GAS TREATMENT SYSTEM

The purposes of the Standby Gas Treatment System are:

1. To process secondary containment atmosphere prior to release under accident conditions.

2. To provide a means of venting the primary containment.

3. To perform leak tests of the secondary containment.

The functional classification of the Standby Gas Treatment System is that of a safety related system. Its' regulatory classification is that of an engineered safety feature (ESF) system.

4.3.1 System Description

The Standby Gas Treatment System (SGTS), Figure 4.3-1, is an engineered safety feature consisting of suction ducting, three SGTS filter trains, and discharge vent ducting. Each train includes a moisture separator, a heating element, a prefilter, two high efficiency filters, a charcoal adsorber, and a blower.

The SGTS provides for treatment of the effluent atmosphere from both the primary and secondary containment. The system is arranged to utilize the normal ventilation system exhaust ducting from the primary and secondary containment. The system is actuated following the indications of a loss of coolant accident and/or high radiation levels in the secondary containment ventilation system exhaust, to treat and maintain the secondary containment atmosphere at a negative internal pressure.

4.3.2 Component Description

The major components of the Standby Gas Treatment System are discussed in the paragraphs which follow.

4.3.2.1 Suction Path

The SGTS is capable of processing the gaseous streams from three different systems:

1. Primary Containment System (Section 4.1)

2. Secondary Containment System (Section 4.2)

3. High Pressure Coolant Injection System (Section 10.1)

4.3.2.2 Moisture Separator

The first element in the SGTS train is a moisture separator which is designed to remove any free droplets of water or mist which may be entrained in the incoming air stream. The moisture separator consists of six woven mesh modules which remove 99.9% of the intrained moisture particles 2 microns or larger.

4.3.2.3 Heating Element

The second component of each filter train is a 40-kW electric fin tube type air heater. The heating element is regulated by a humidity controller to maintain the air flow entering the prefilter at less than 70 percent relative humidity.If the relative humidity exceeds 70 percent, the charcoal adsorption efficiency will be reduced (Section 4.3.2.6).

4.3.2.4 Prefilter

The prefilter is constructed of a replaceable dry tube, extended fiberglass media, and is located in the exhaust plenum of the electric heating coil. The prefilter is designed to remove atmospheric dust and particulate matter to prevent clogging of the high efficiency filters, extending their life.

4.3.2.5 High Efficiency Particulate Filter

Immediately following the prefilter and upstream of the charcoal adsorber is the HEPA (high efficiency particulate air) filter bank. This HEPA filter, which has an efficiency of 99.97 percent for particles 0.3 microns or larger in diameter, removes particulates which may enter the plenum from the suction inlets and protects the charcoal adsorber from fouling. The HEPA filter bank is composed of 6 cells constructed of fiberglass media.

4.3.2.6 Charcoal Adsorber

The charcoal adsorber bed is located between the two high efficiency filters. The charcoal adsorber is impregnated with stable iodine and has a minimum capacity of removing 99.9 percent of iodines with 5% in the form of methyl iodine (CH_3I) when operating at <70 percent humidity.

4.3.2.7 Downstream HEPA Filter

Following the charcoal adsorber bed is the second HEPA filter bank. This downstream HEPA filter, which has the same design parameters as the upstream HEPA filter prevents any possible charcoal fines originating in the charcoal filter from exiting the plenum and being carried out of the plant exhaust.

4.3.2.8 Blower

The last unit in the SGTS filter train is a blower. The blower is a heavy duty industrial fan unit with a flow rate of ~9000 standard cubic feet per minute(SCFM). The SGTS blower provides the motive force required to pull the air stream through the SGTS train and then discharges the treated stream up the plant stack.

4.3.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

4.3.3.1 Normal Operation

During normal plant operation, the SGTS is in a standby status ready to be started either manually or automatically. The SGTS is capable of processing air flow from the primary containment (Section 4.1), the secondary containment (Section 4.2), and the High Pressure Coolant Injection System gland exhaust blower (Section 10.1).

4.3.3.2 Automatic Initiation

All three trains of the SGTS automatically start and dampers are alignedto process the air from the secondary containment, upon receipt of any of the following signals:

1. Low reactor water level
2. High drywell pressure
3. Refueling floor area high radiation
4. Reactor Building ventilation exhaust high radiation.

In addition, upon receipt of any of the above signals, the normal ventilation system isolation dampers for the secondary containment will automatically shut and the ventilation fans will trip.

4.3.3.3 Inspection and Testing

The secondary containment inleakage rate is determined by isolating the normal ventilation system and operating the SGTS. The SGTS flow is adjusted to <12000 SCFM, and the secondary containment is verified to have a pressure greater than 0.25 inches water gauge below the pressure outside of the containment.

4.3.4 System Interfaces

The SGTS interfaces with other plant systems discussed in the paragraphs which follow.

4.3.4.1 Primary Containment (Section 4.1)

The Primary Containment can use the SGTS for purging operations.

4.3.4.2 Secondary Containment (Section 4.2)

The Secondary Containment uses the SGTS for testing its integrity and for maintaining >0.25 inches of water negative internal pressure during containment isolation.

4.3.4.3 High Pressure Coolant Injection System (Section 10.1)

The High Pressure Coolant Injection System turbine gland seal leakoff condenser exhaust blower discharges to the SGTS where the noncondensibles are processed and released.

4.3.4.4 Primary Containment Isolation System (Section 4.4)

The Primary Containment Isolation System provides the automatic initiation signals for the SGTS.

4.3.4.5 Plant Stack

The SGTS discharges out the 600 ft. plant stack.

4.3.5 Summary

Classification:

Safety related system, Engineered Safety System.

Purpose:

To process secondary containment atmosphere prior to release under accident conditions. To provide a means of venting the primary containment. To perform leak tests of the secondary containment.

Components:

Dampers; moisture separators; heating elements; prefilter; high efficiency filters; carbon filters; blower.

System Interfaces:

High Pressure Coolant Injection System; Secondary Containment; Primary Containment Isolation System; Plant Stack.

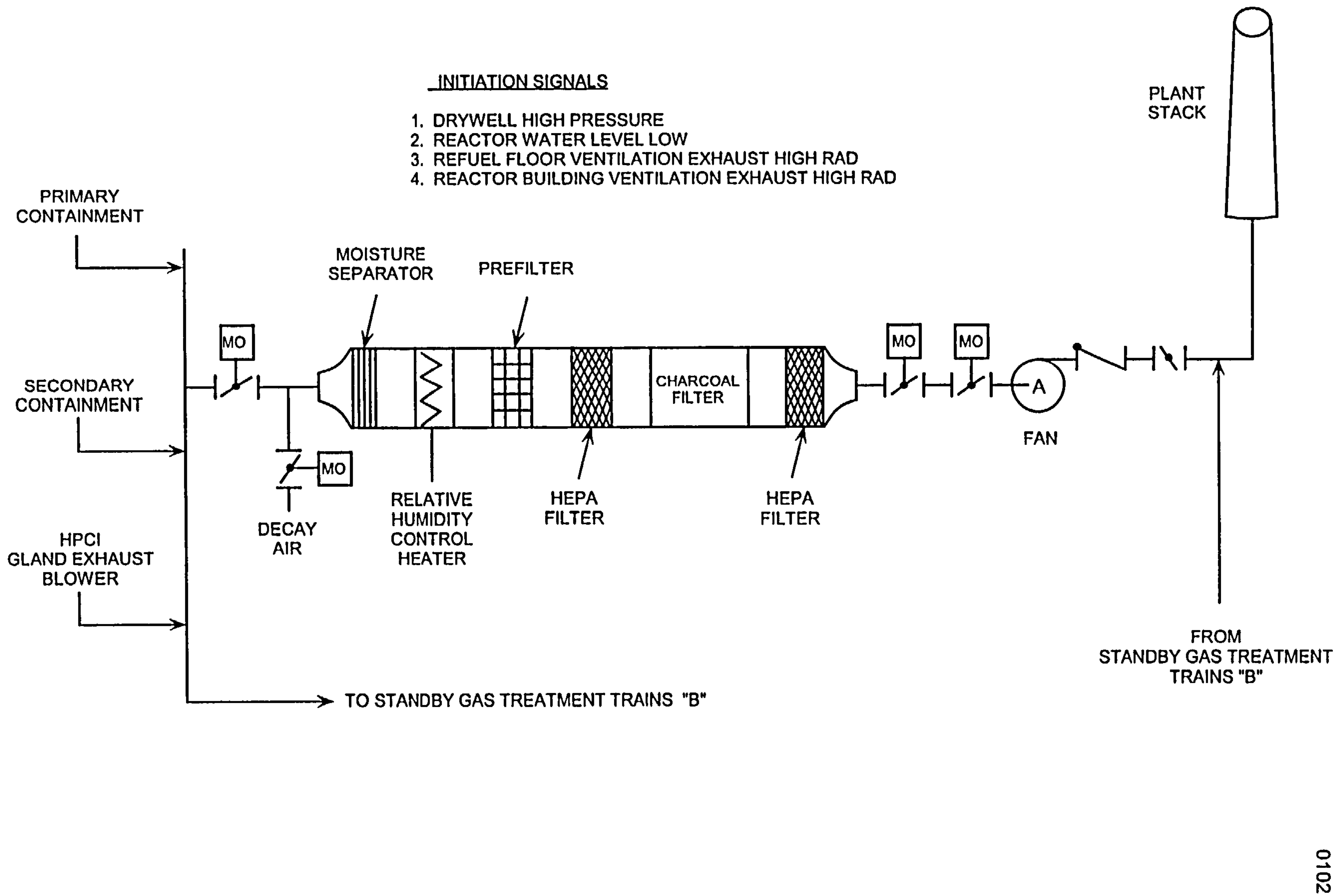


Figure 4.3-1 STANDBY GAS TREATMENT SYSTEM

Table of Contents

List of Tables

4.4 PRIMARY CONTAINMENT ISOLATION SYSTEM

The purpose of the Primary Containment Isolation System (PCIS) is to isolate the primary and secondary containments during accident conditions to limit the release of radioactive materials.

The functional classification of the PCIS is that of a safety related system. Its regulatory classification is that of an engineered safety feature (ESF) system.

4.4.1 System Description

The Primary Containment Isolation System determines, from information provided by reactor plant process instrumentation, which systems should be isolated during various accident or abnormal conditions. The isolation functions are coordinated to ensure integrity of the containments by meeting specific design features:

1. The system shall be fail safe.

2. Any one failure will not impair the functional ability to isolate when required.

3. The power supplies on a set of valves shall be reliable and from different sources (i.e.: AC-DC).

4. Redundancy and separation of sensors and cables.

5. The system shall be testable during normal plant operation.

6. Operation must be automatic with a manual reset.

The control circuitry is arranged in a dual trip system with a one-out-of-two-twice logic. In general, all piping penetrations to the primary containment, high or low energy gaseous or liquid streams, have isolations. The isolations are grouped according to valve classifications and/or the isolation signals.

4.4.2 Component Description

The major components of the Primary Containment Isolation System are discussed in the paragraphs which follow.

4.4.2.1 Isolation Valves

Isolation valves are provided to assure containment of radioactive materials which might be released from the containment during the course of an accident. Pipes that penetrate the drywell and connect to the nuclear steam supply system are double valved, as are those that open into the drywell free space. In the case of lines forming a part of the reactor coolant pressure boundary, one valve is located inside the drywell and the other outside, as close as practical to the containment. Normally closed double valves in lines to the drywell free space, such as the vacuum relief from the Reactor Building and the suppression chamber nitrogen makeup header, are both located outside the containment. In lines forming a closed loop inside the drywell, one remote, manually-controlled, motor-operated valve is generally provided outside the drywell.

4.4.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

4.4.3.1 Isolation Valve Grouping

The PCIS logic is divided into discrete groups according to the class of isolation valve as discussed in Section 4.4.2.1.

4.4.3.2 Main Steam System Isolation

The main steam line isolation and drain valves automatically close in response to signals which indicate a breach of the reactor coolant system boundary, a gross failure of fuel cladding, or a failure of the electro hydraulic control system logic. Table 4.4-1 lists the isolation signals, any additional requirements needed, probable causes of signals, and the reason for isolation by each signal.

4.4.3.3 Other System Isolations

When a low reactor water level or a high drywell pressure is detected, the systems listed below will automatically isolate to control the loss of coolant from the reactor vessel and the radioactivity release from the primary or secondary containments.

1. Drywell equipment and floor drain system.

2. Torus drain system.

3. Primary containment vent and purge sys tem.

4. Secondary containment ventilation system.

5. Residual Heat Removal System, shutdown cooling mode.

The systems listed below will also isolate upon detection of leakage from the appropriate system. For the specific isolation signals and their isolation valves refer to the sections indicated.

1. Reactor Core Isolation Cooling System (Section 2.7).

2. Reactor Water Cleanup System (Section 2.8).

3. High Pressure Coolant Injection System (Section 10.1).

4.4.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

4.4.4.1 Main Steam System (Section 2.5)

The PCIS isolates the Main Steam System when the isolation logic is satisfied.

4.4.4.2 Primary Containment System (Section 4.1)

The PCIS isolates the primary containment vent and purge lines when the isolation logic is satisfied.

4.4.4.3 Secondary Containment System (Section 4.2)

The PCIS isolates the secondary containment ventilation lines when the isolation logic is satisfied.

4.4.4.4 Residual Heat Removal System (Section 10.4)

The PCIS isolates the shutdown cooling mode when the isolation logic is satisfied.

4.4.4.5 Reactor Core Isolation Cooling System (Section 2.7)

The PCIS isolates the Reactor Core Isolation Cooling System when the isolation logic is satisfied.

4.4.4.6 Reactor Water Cleanup System (Section 2.8)

The PCIS isolates the Reactor Water Cleanup System when the isolation logic is satisfied.

4.4.4.7 High Pressure Coolant Injection System (Section 10.4)

The PCIS isolates the High Pressure Coolant Injection System when the isolation logic is satisfied.

4.4.5 Summary

Classification:

Safety related system. Engineered safety feature system.

Purpose:

To isolate the primary and secondary containments during accident conditions to limit the release of radioactive materials.

Components:

Isolation valves; logic.

System Interfaces:

Main Steam System; Primary Containment System; Secondary Containment System; Residual Heat Removal System; Reactor Core Isolation Cooling System; Reactor Water Cleanup System; High Pressure Coolant Injection System.

Table 4.4-1 PCIS Signals for Main Steam System

Condition Causing Isolation	Additional Requirements	Possible Cause	Reason for Isolation
Main Steam Line High Radiation	N/A	Gross Fuel Cladding Failure	Minimize Radiological Release to Environment
Main Steam Tunnel High Temperature	N/A	Steam Leak in Tunnel	Isolate the Steam Leak
Main Steam Line High Flow	N/A	Individual Steam Line Break	Attempt to Isolate the Steam Line Break
Main Steam Line Low Pressure	Mode Switch in "RUN"	EHC Pressure Regulator Failure Upscale	Prevent Excessive Reactor Vessel Cooldown Rate
Reactor Water Level Low Low	N/A	Large Loss of Coolant Accident	Conserve Reactor Vessel Water Inventory
Manual	N/A	Manual Isolation Control Switches in Closed Position	Operator Action Initiated the Isolation

5.0 NEUTRON MONITORING SYSTEMS

The purposes of the Neutron Monitoring Systems are:

1. To monitor reactor core neutron flux and provide indication during all modes of reactor operation.

2. To provide trip signals to the Reactor Manual Control System (RMCS) and the Reactor Protection System (RPS).

The safety objective of the Neutron Monitoring Systems is to detect conditions in the core that threaten the overall integrity of the fuel barrier due to excessive power generation and provide signals to the Reactor Protection System (RPS), so that release of radioactive material from the fuel cladding does not occur.

The power generation objective of the Neutron Monitoring Systems is to provide information for the efficient, expedient operation and control of the reactor. Specific power generation objectives of the Neutron Monitoring System are to detect conditions that could lead to local fuel damage and to provide signals that can be used to prevent such damage, so that plant availability is not reduced.

The Neutron Monitoring System (NMS) consists of a collection of six major systems:

Source Range Monitoring (SRM) System (Section 5.1)

Intermediate Range Monitoring (IRM) System (Section 5.2)

Local Power Range Monitoring (LPRM) System (Section 5.3)

Average Power Range Monitoring (APRM) System (Section 5.4)

Rod Block Monitoring (RBM) System (Section 5.5)

Traversing Incore Probe (TIP) System (Section 5.6)

Figure 5.0-1 illustrates the relationship among the individual NMS systems as a function of core power/flux and operating conditions. Figure 5.0-2 illustrates the radial distribution and Figure 5.0-3 the axial distribution of the incore detectors of these systems.

Figure 5.0-4 illustrates in block diagram form the systems used during shutdown, startup, and heatup operation. Figure 5.0-5 depicts the systems used during power operation.

5.0.1 Source Range Monitoring System (Section 5.1)

The Source Range Monitoring (SRM) System provides neutron flux information for display and initiation of rod withdraw blocks from shutdown source conditions to that point where neutron flux overlaps the intermediate range monitors.

5.0.2 Intermediate Range Monitoring System (Section 5.2)

The Intermediate Range Monitoring (IRM) System provides neutron flux information during startup and heatup from the upper portion of the source range to the lower portion of the power range.

5.0.3 Local Power Range Monitoring System (Section 5.3)

The Local Power Range Monitoring (LPRM) System provides signals proportional to local neutron flux at various radial and axial incore locations for use by the Average Power Range Monitoring System and Rod Block Monitors.

5.0.4 Average Power Range Monitoring System (Section 5.4)

The Average Power Range Monitoring (APRM) System monitors core average thermal power by computing the core average flux from the LPRM detector inputs.

5.0.5 Rod Block Monitor (Section 5.5)

The Rod Block Monitoring (RBM) System monitors local thermal power by computing local thermal flux in the vicinity of a control rod to be withdrawn, and compares it to the core average thermal power. Rod withdraw movement is then blocked if local power becomes excessive.

5.0.6 Traversing Incore Probe System (Section 5.6)

The Traversing Incore Probe (TIP) System provides a means of measuring axial neutron flux over 41 fixed locations in the core. The measured axial neutron flux is used to calibrate the local power range monitor detectors and to calculate core power distribution.

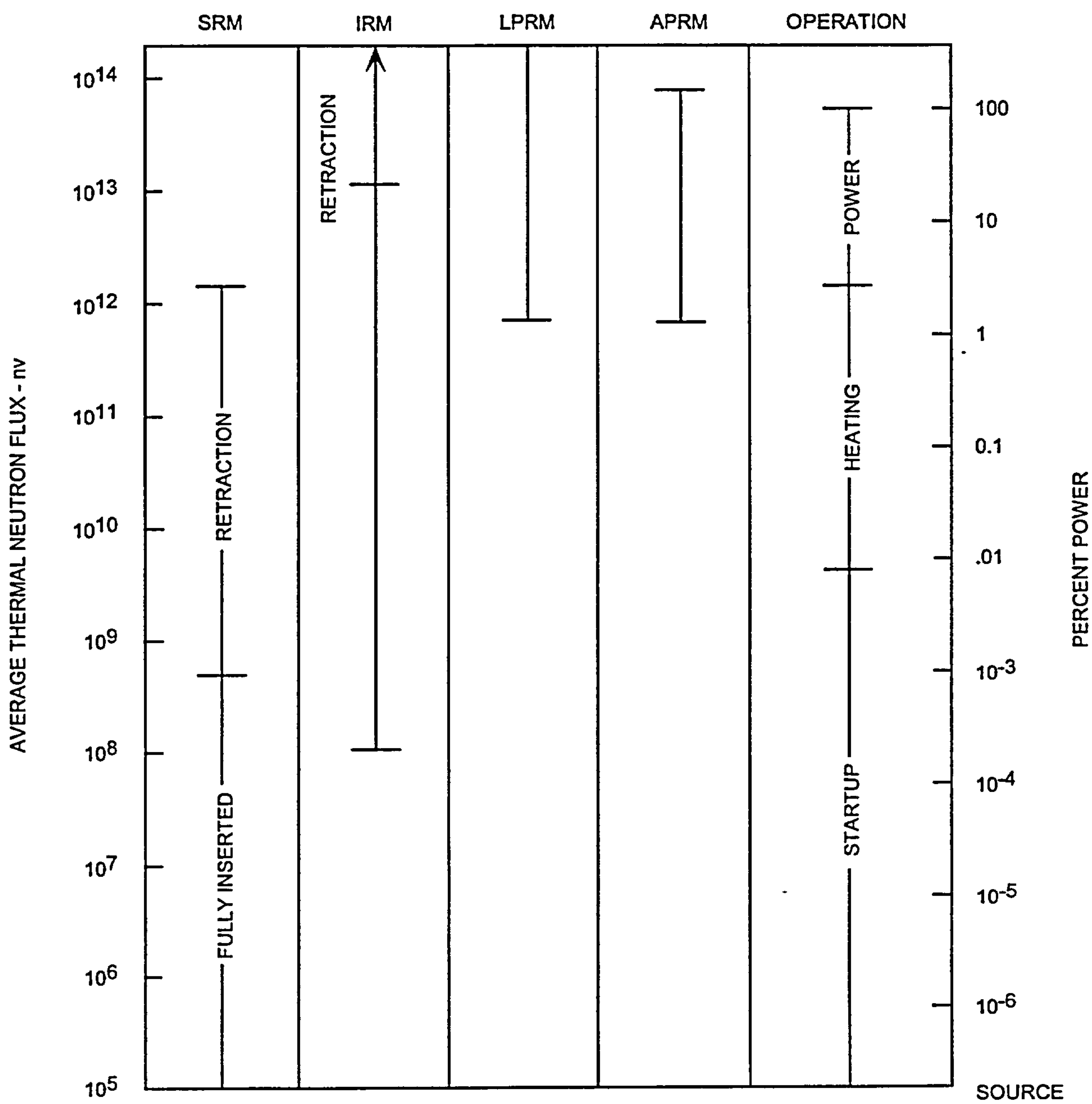


Figure 5.0-1 Neutron Monitoring System Ranges

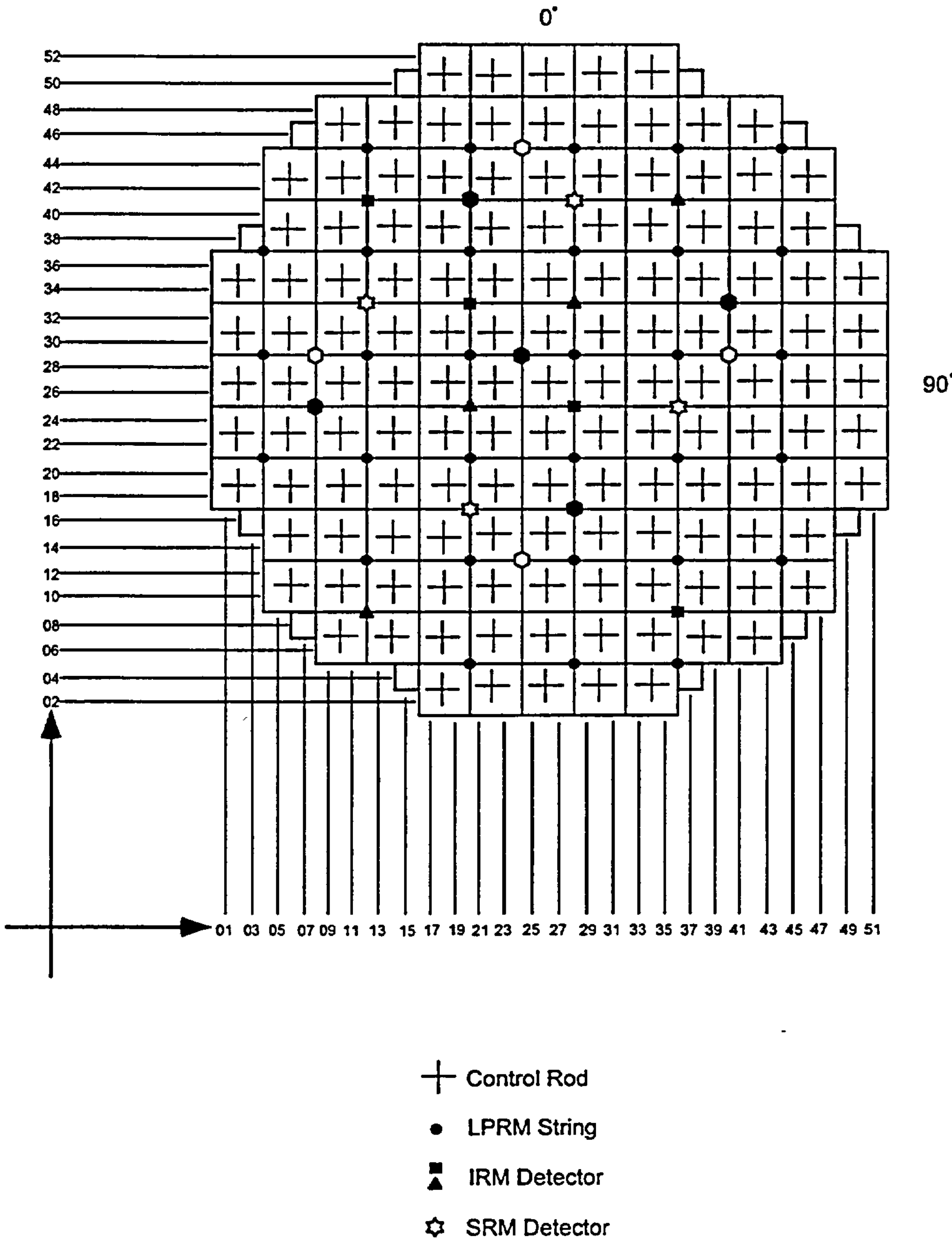


Figure 5.0-2 Detector and Control Element Arrangement

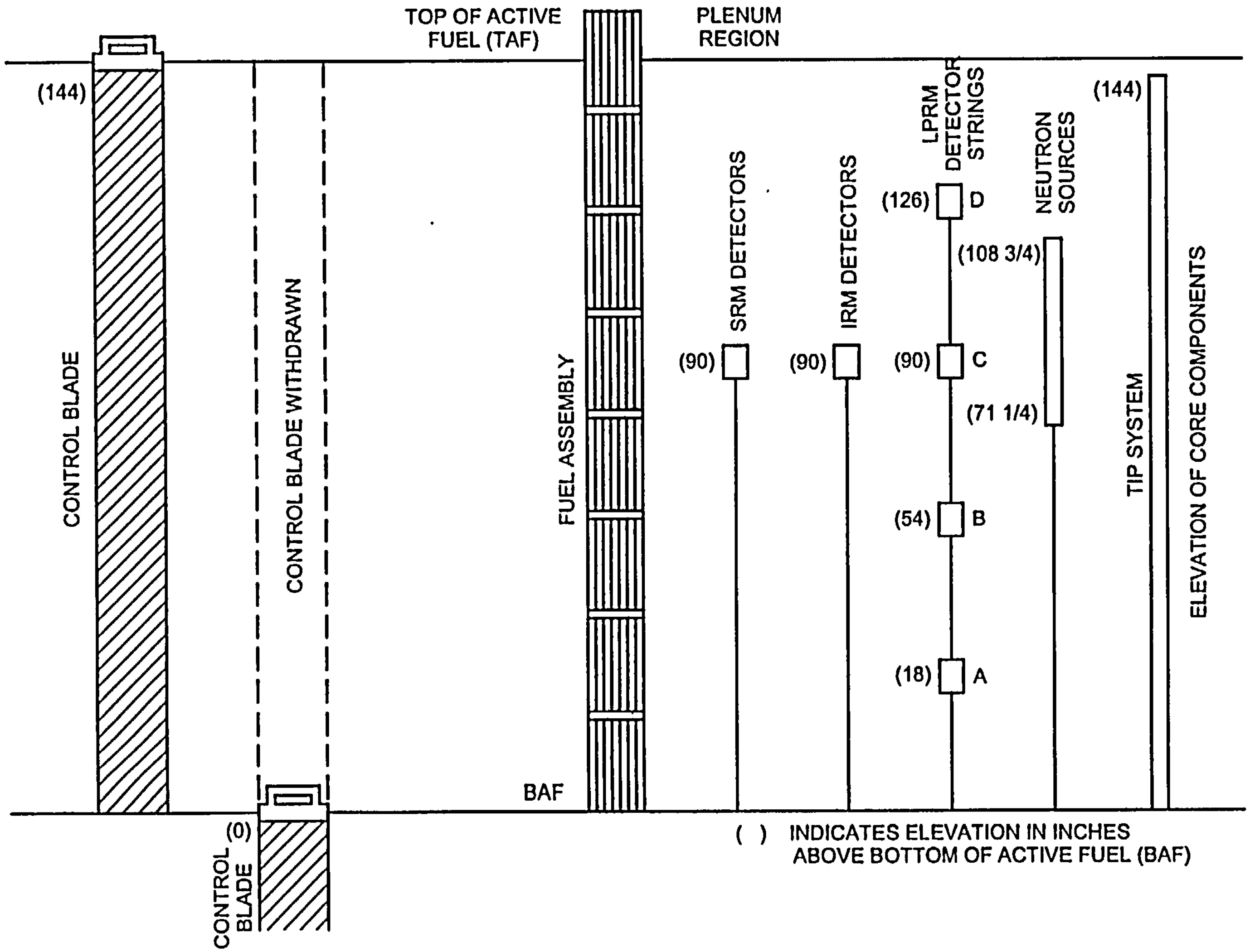
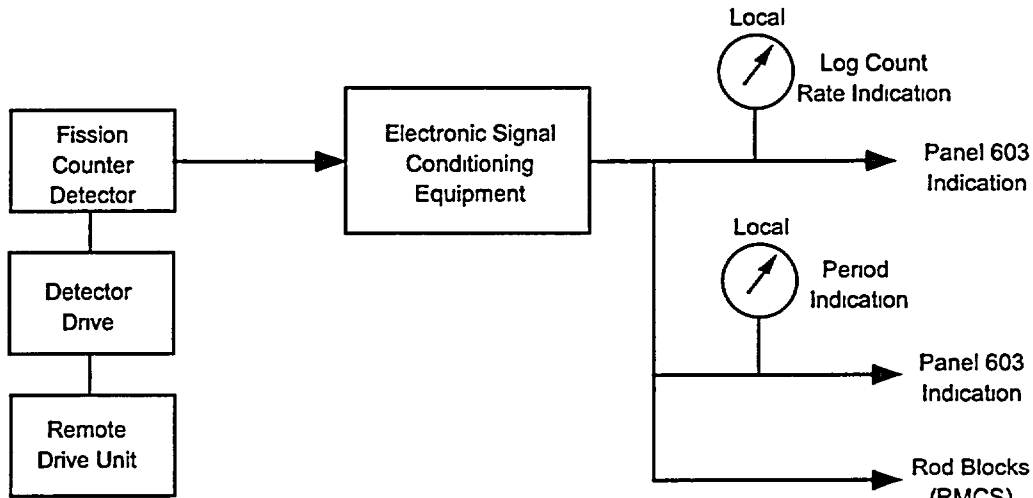
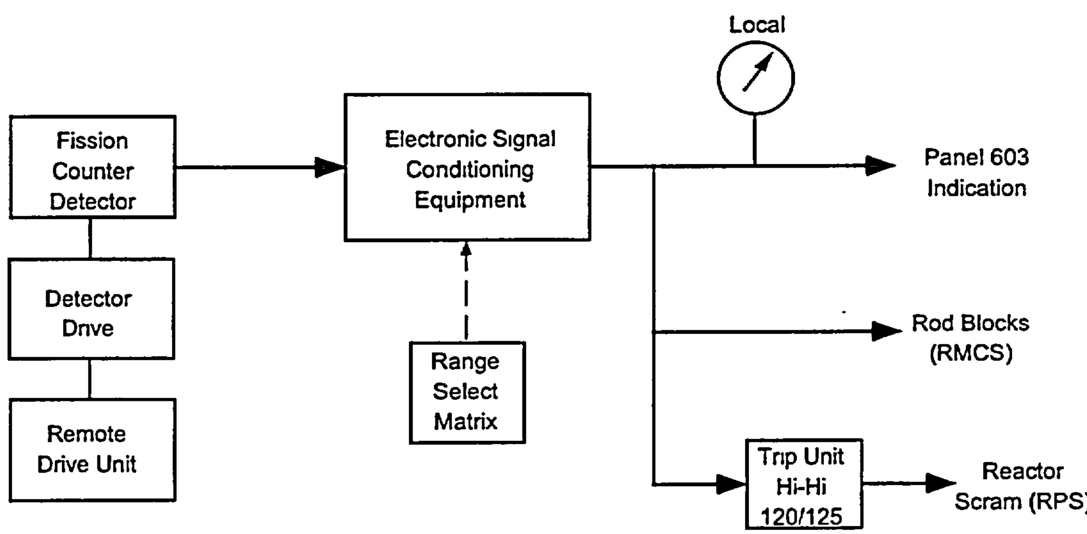


Figure 5.0-3 Axial Location of Neutron Monitoring System Components

0102



Source Range Monitor



Intermediate Range Monitor

Figure 5.0-4 Shutdown, Startup and Heatup Operations

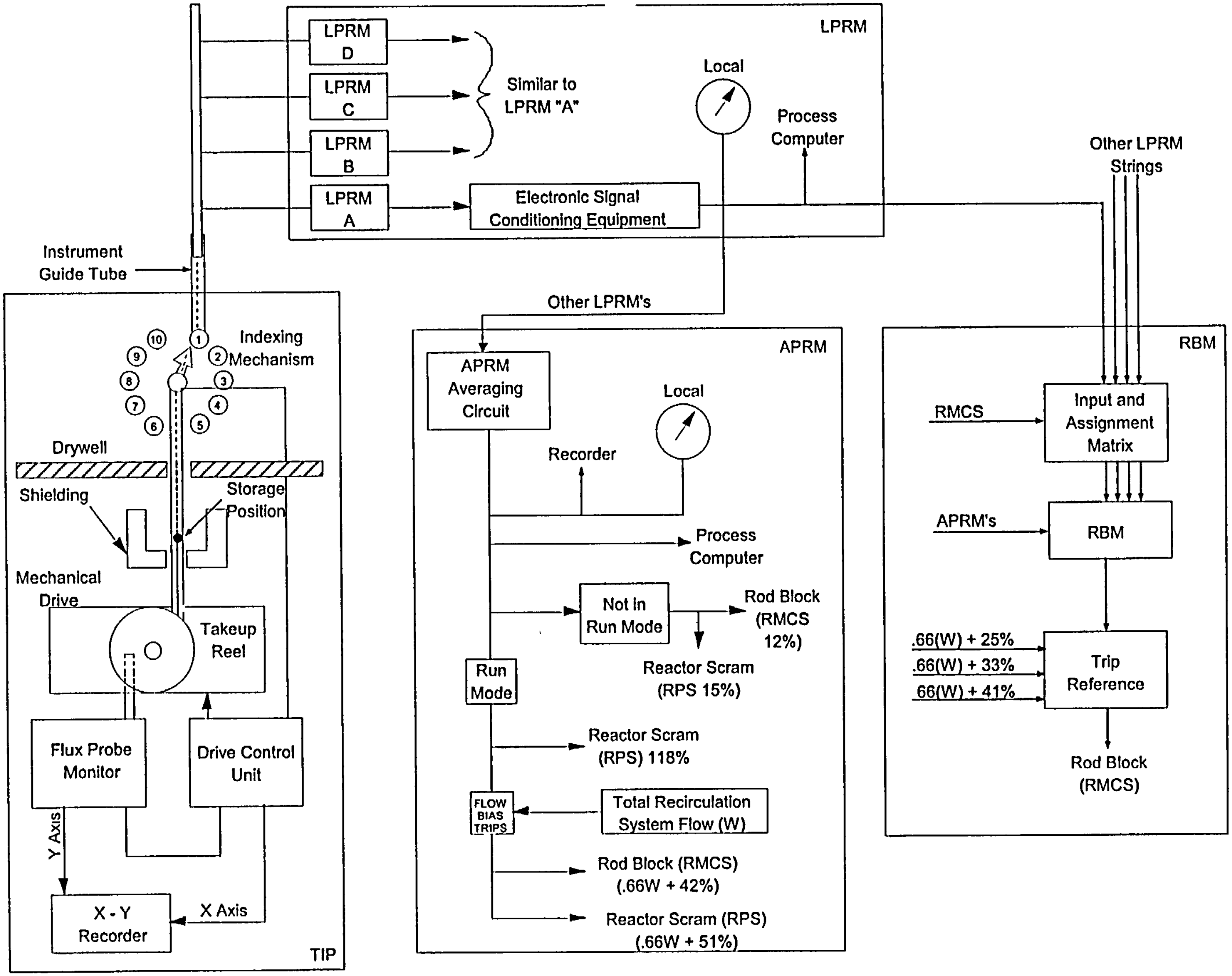


Figure 5.0-5 Power Range Operation

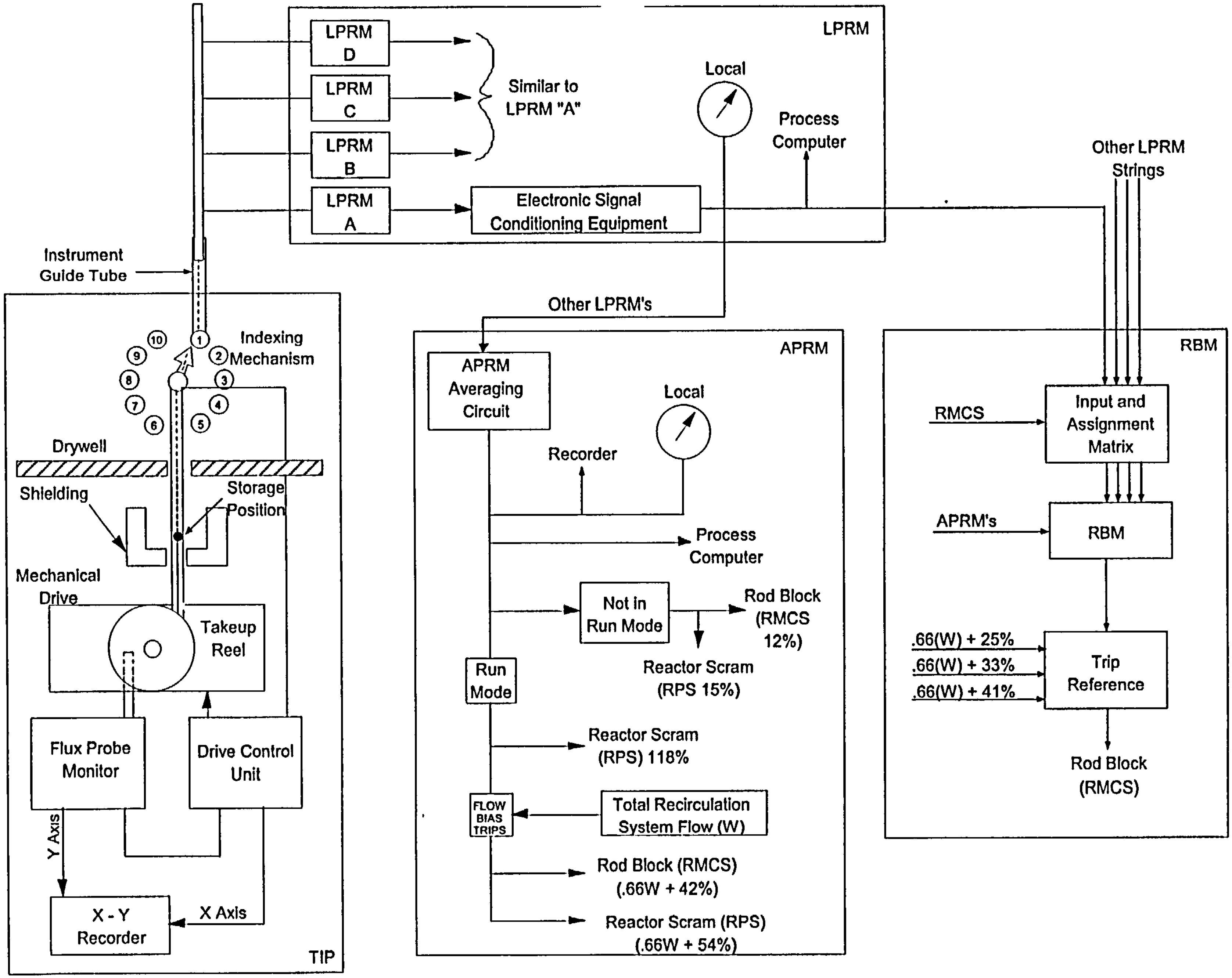


Figure 5.0-5 Power Range Operation

Table of Contents

List of Tables

List of Figures

5.1 SOURCE RANGE MONITORING SYSTEM

The purpose of the Source Range Monitoring (SRM) System is to monitor neutron flux for displays and initiation of rod withdraw blocks, from shutdown source conditions to that point where neutron flux overlaps the intermediate range monitors.

The functional classification of the SRM system is that of a power generation system.

5.1.1 System Description

The SRM system consists of: four miniature fission chambers located at different radial core locations, detector drive assemblies to insert and retract the detectors, and the required electronic equipment to condition the fission detector output for displays and rod block functions.

5.1.2 Component Description

The major components which makeup the SRM system are discussed in the following paragraphs and illustrated in block diagram form in Figure 5.1-1.

5.1.2.1 Fission Chamber Detector

The fission chamber detector illustrated in Figure 5.1-2 provides an output signal that consists of energy pulses. The output pulse rate is proportional to the neutron flux level. The energy of the pulse is dependent on the event that caused the pulse (i.e.: neutron or gamma). Pulses caused by gamma radiation are considerably less energetic than those caused by neutrons.

The SRM fission chambers are approximately one inch in length and 0.16 inches in diameter.

The detector is a titanium chamber coated on the inner surface with highly enriched (90% U-235) uranium oxide (U_3O_8). The inner volume of the titanium chamber is pressurized with argon gas to a pressure of 220 psia.

Thermal neutrons entering the detector interact with the uranium oxide coating causing the U-235 atoms to fission. The high energy fission fragments moving within the argon gas cause ionization of the gas. The ions produced are collected on the inner electrode producing an energy pulse.

Ionization of the argon gas may also be caused by gamma radiation present in the reactor core. The amplitude of the energy pulse is proportional to the number of ion pairs produced per interaction. The gamma interaction produces fewer ion pairs than a fission event, thus the energy pulse is smaller. This characteristic allows for the elimination of the gamma pulse in the electronic signal conditioning network, Section 5.1.2.3.

5.1.2.2 Detector Drive Assembly

The detector drive assemblies are used to position SRM detectors axially within the core. When the detector is being used to monitor core shutdown conditions during the initial stages of reactor startup, the detector is fully inserted to a position approximately 18 inches above the core midplane. As the reactor power is increased, the detector is retracted to prolong the life of the detector by decreasing the level of neutron flux to which the detector is exposed. During power range operation, the detector is in a fully withdrawn position, 18 inches below the core. The complete insert and retract system consists of the mechanical components required to drive each of the SRM detectors and the switching circuits which allow the operator to request detector movement.

5.1.2.3 Electronic Signal Conditioning Equipment

The electronic signal conditioning equipment amplifies the energy pulse signal for transmission to the control room with a high signal to noise ratio, discriminates the gamma pulse while passing the pulses caused by neutron events, conditions the signal to display the seven decades of the source range (10^{-1}cps to 10^{-6}cps), and develops rod block signals to ensure a safe controlled reactor startup.

5.1.2.4 Rate Change Circuit

During startup of the reactor, the rate at which the neutron count rate is increasing is of importance to the operator. Since neutron flux is proportional to reactor power, the rate at which neutron flux is increasing is essentially the same as the rate of reactor power increase. For example, suppose the SRM count rate has increased from 1×10^2 cps to 1×10^3 cps. It is reasonable to assume that reactor power has increased by a factor of 10.

The reactor period (T) is defined as the amount of time required for power to change by a factor of "e" (where e is the natural log base, 2.718). Knowledge of the reactor period enables the reactor operator to monitor and effectively control the rate of change of reactor power.

The rate change circuit determines the rate of change of the input count rate and produces an output signal displayed as reactor period.

5.1.3 System Features

A short discussion of system features is given in the paragraphs which follow.

5.1.3.1 Operation During Startup

Operation of the source range monitoring system during a reactor startup is covered in Section 5.2.3.3 of the intermediate range monitoring chapter.

5.1.3.2 Rod Blocks

During reactor startup from the source region the SRM's provide alarms and rod blocks to prevent rod withdrawal during certain conditions. These conditions along with the appropriate bypasses are listed on Table 5.1-1.

5.1.4 System Interfaces

The interfaces this system has with plant systems are discussed in the paragraphs which follow.

5.1.4.1 Reactor Manual Control System (Section 7.1)

The Reactor Manual Control System receives input signals from the SRM system to impose the rod blocks listed in Table 5.1-1.

5.1.4.2 Reactor Protection System (Section 7.3)

The Reactor Protection System receives SRM signals and will generate a high flux level reactor scram when the shorting links are removed. The shorting links are removed during initial fuel loading.

5.1.4.3 Intermediate Range Monitoring System (Section 5.2)

The IRM system provides automatic bypassing of SRM rod blocks based on IRM range switch position.

5.1.5 Summary

Classification:

Power generation system.

Purpose:

To monitor neutron flux for display and initiation of rod withdraw blocks, from shutdown source conditions to that point where neutron flux overlaps the intermediate range monitors.

Components:

Detectors; detector drive assemblies; electronic signal conditioning equipment; and rate change circuit.

System Interfaces:

Reactor Manual Control System; Reactor Protection System; Intermediate Range Monitoring System.

Table 5.1-1 SRM Interlocks and Trips

Alarm or Trip[1]	Setpoint	Panel Indication	Action	Auto Bypass
SRM Upscale (High-High)	5 X 10^5 CPS	SRM High-High (A-D)	Scram when shorting links removed.	Bypassed when shorting links installed.
SRM Upscale (High)	1 X 10^5 CPS	SRM High or Inop (A-D)	Row withdrawal block.	IRM's above range 7, or bypassed, or Mode switch in "RUN".
SRM Downscale	3 CPS	SRM Downscale (A-D)	Rod withdrawal block.	IRM's above range 2, or bypassed, or Mode switch in "RUN"
SRM Inop[2]		SRM High or Inop (A-D)	Rod withdrawal block.	IRM's above range 7, or bypassed, or Mode switch in "RUN".
SRM Period	<30 seconds	SRM Period (Amber A-D)		
Retract not Permitted	100 CPS	Retract Permit (A-D)	Rod withdrawal block.	IRM's above range 2, or bypassed, or Mode switch in "RUN"
SRM Bypassed	[3]Bypass switch	SRM Bypassed (A-D)	Bypasses all Trip function of the SRM when Bypassed.	Bypasses all Trip function of the SRM when Bypassed.

[1]All trips auto reset when the trip condition is cleared. Trip indicators on the SRM chassis must be manually cleared.
[2]Produced by: 1. Mode switch not in operate, 2. Power supply voltage low, 3. Circuit boards not in circuit.
[3]Only one SRM channel may be bypassed using the bypass switch.

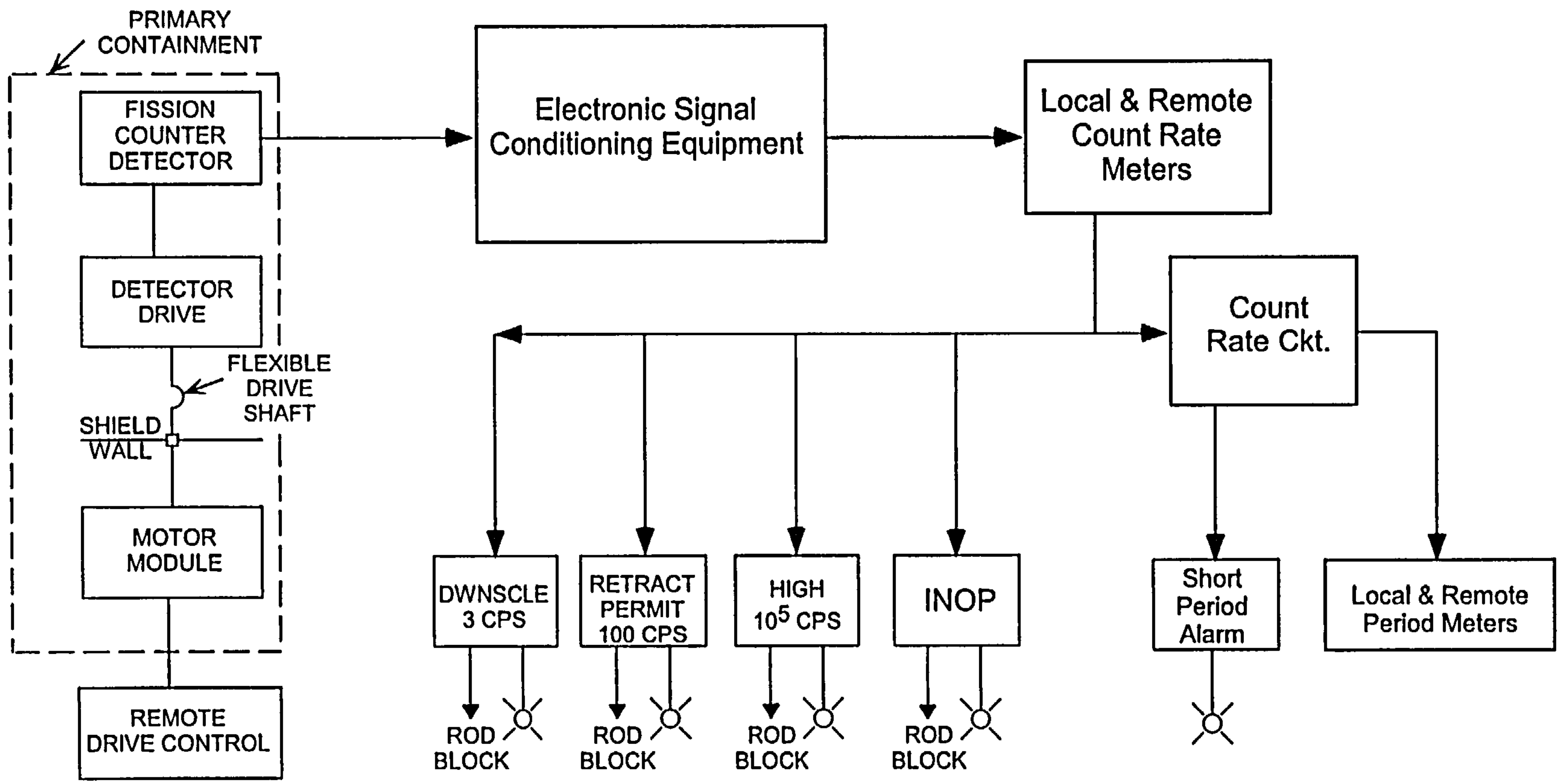
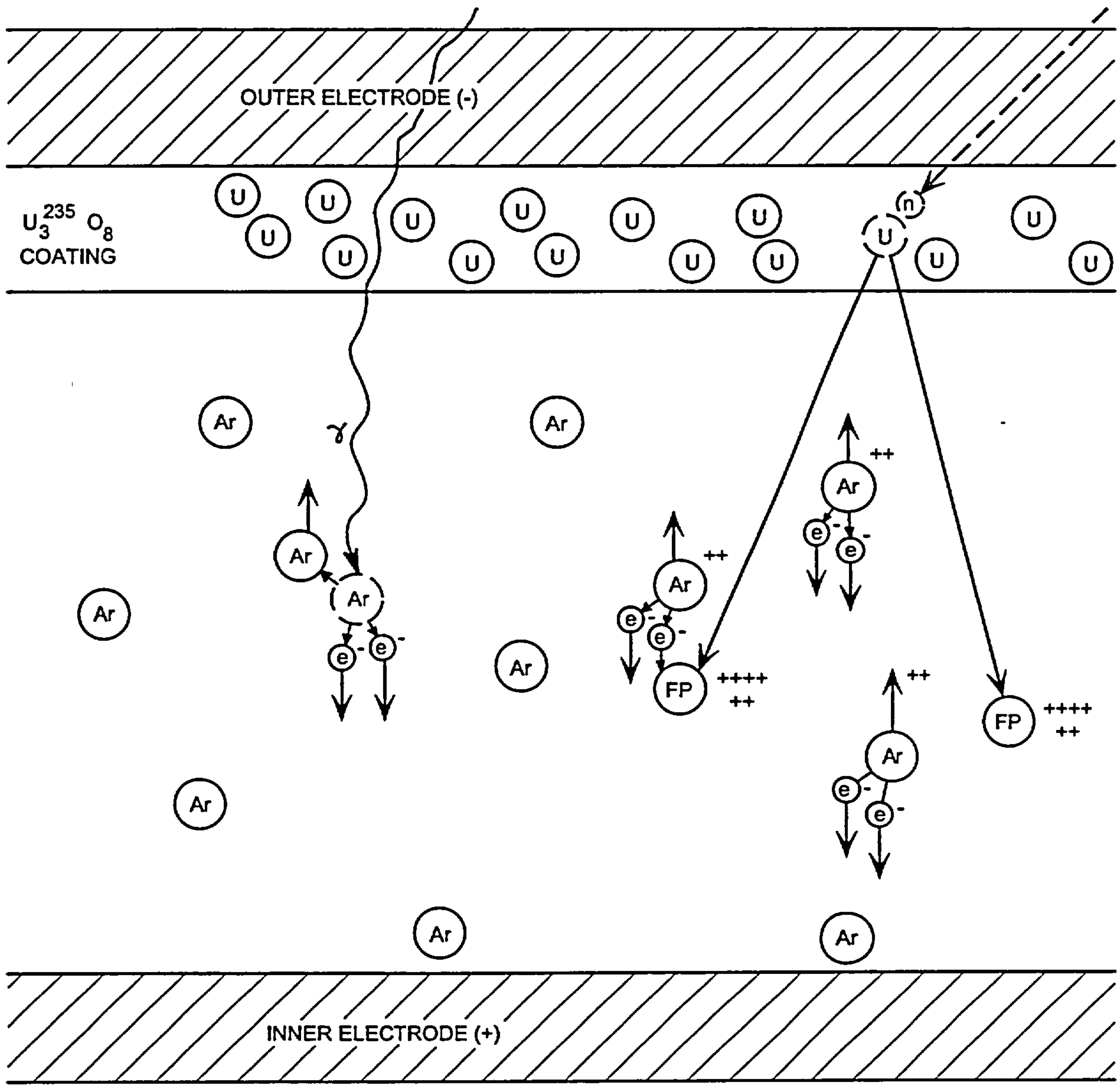


Figure 5.1-1 Source Range Monitoring Channel Block Diagram



DETECTOR DATA

90% ENRICHED IN U-235

INTERNAL PRESSURE 215 psi

LENGTH 1.6 INCHES

WIDTH 0.16 INCHES

Figure 5.1-2 Fission Chamber Operation

Table of Contents

List of Tables

List of Tables

5.2 INTERMEDIATE RANGE MONITORING SYSTEM

The purposes of the Intermediate Range Monitoring (IRM) System are to monitor neutron flux from the upper portion of the source range to the lower portion of the power range and to provide protective trip signals to prevent fuel damage.

The functional classification of the IRM System is that of a safety related system.

5.2.1 System Description

The IRM System consists of eight channels each having miniature fission chambers located at different radial core locations, detector drive assemblies to insert and retract the detectors, and the required electronic equipment to condition the fission detector output for display and trip functions.

5.2.2 Component Description

The major components which make up the IRM system are discussed in the following paragraphs and illustrated in block diagram form in Figure 5.2-1.

5.2.2.1 IRM Detector

The detectors used for the IRM channels are basically the same as the SRM detectors described in Section 5.1.2.1. However, the IRM detectors use lower argon pressure (17.7 psia), lower uranium dioxide content and closer spacing between the electrodes to permit operation at higher levels of neutron flux. As with the SRM detectors, the IRM detectors are moveable, and are retracted when sufficient overlap is achieved with the Average Power Range Monitors, to increase detector life.

5.2.2.2 Electronic Signal Conditioning Equipment

The electronic signal conditioning equipment amplifies the detector output signal to produce a high signal to noise ratio, discriminates the gamma signal, conditions the neutron flux signal for display on a linear scale with the use of range switches, and provides trip signals to ensure safe reactor operation in the intermediate range of neutron flux.

Each IRM channel contains a separate range switch with ten ranges to cover five decades of neutron flux. The range switches are provided to protect against rapid power excursions not under operator control. Automatic scram protection for excessively short periods is provided by the high flux scram level on each and every range, at 120/125 of scale. The high flux level scram is faster than a period scram for periods shorter than 200 millisecond.

5.2.3 System Features

A short discussion of system features is given in the paragraphs which follow.

5.2.3.1 Rod Blocks and Scrams

The IRM system provides rod blocks and scrams to prevent fuel damage. These along with the appropriate bypasses are listed in Table 5.2-1

5.2.3.2 IRM Recorders

The eight IRM channels are displayed on four two pen recorders on a control room panel. These same recorders are shared with the Average Power Range Monitoring(APRM) System (Section 5.4) and the Rod Block Monitoring(RBM) System (Section 5.5) through selector switches as shown in Figure 5.2-1.

5.2.3.3 Operation During Startup

Initially during reactor startup, the approach to criticality and monitoring of reactor power is performed by the SRM system. When the SRM count rate is between 10^4 and 10^5 cps, the IRM channels should be on scale in range one. As soon as the IRM channels are on scale, they are used to monitor reactor power, and the SRM detectors are incremently withdrawn to maintain SRMcount rate between 10^2 and 10^5 cps. As reactor power increases and IRM channels reach 75/125 of scale, the operator upranges the IRM channel. While ranging IRM channels, the IRM reading should be maintained between 25/125 and 75/125 of scale for each IRM channel to avoid rod withdraw blocks from being imposed.

As reactor power continues to increase during startup, the IRM's are up ranged via the range switches. On range nine of the intermediate range, the average power range monitors begin to indicate percent reactor power.

There should be sufficient overlap between the Average Power Range Monitor System and the IRM's. Typically, 100/125 on range 10 roughly corresponds to 40% of rated core thermal power. Once the Average Power Range Monitor System is well on scale and the reactor mode switch is placed in the RUN position, the IRM detectors are withdrawn to their fully withdrawn position.

5.2.4 System Interfaces

This system has interfaces with other plant systems as discussed in the paragraphs which follow.

5.2.4.1 Reactor Protection System (Section 7.3)

The RPS receives various scram signals from the IRM System.

5.2.4.2 Reactor Manual Control System (Section 7.1)

The RMC System receives various rod block signals from the IRM System.

5.2.4.3 Source Range Monitoring System (Section 5.1)

The SRM System receives signals from IRM range switch positions. These signals automatically bypass the SRM rod blocks.

5.2.5 Summary

Classification:

Safety related system.

Purpose:

Monitor neutron flux from the upper portion of the source range to the lower portion of the power range and provide protective trip signals to

Table 5.2-1 IRM Interlocks and Trips

Alarm or Trip[1]	Setpoint	Panel Indication	Action[2]	Auto Bypass
IRM Upscale (High-High)	120/125 of Scale	IRM High-High or Inop (A-H)	Scram	Mode switch in "RUN" and APRM on scale.
IRM Upscale (High)	108/125 of Scale	IRM High (A-H)	Rod withdrawal block.	Mode switch in "RUN"
IRM Downscale	5/125 of Scale	IRM Downscale (A-H)	Rod withdrawal block.	IRM on range 1 or Mode switch in "RUN"
IRM Inop[3]		IRM High-High or Inop (A-H)	Rod withdrawal block.	Mode switch in "RUN".
			Scram	Mode switch in "RUN" and APRM on Scale
IRM Bypassed	[4]Bypass switch	IRM Bypassed (A-H)	Bypasses all Trip functions of the IRM bypassed.	Bypasses all Trip functions of the IRM bypassed.

[1]All front panel trips auto reset when the trip condition is cleared. Trip indicators on the IRM chassis must be manually cleared. Scram or half scram must be manually reset.

[2]IRM retraction will produce a rod block except when the Reactor mode switch is in "RUN".

[3]Produced by: 1.Switch not in operate, 2. Power supply voltage low, 3. Circuit boards not in circuit.

[4]Only one IRM in each RPS channel (A or B) may be bypassed, using the bypass switch.

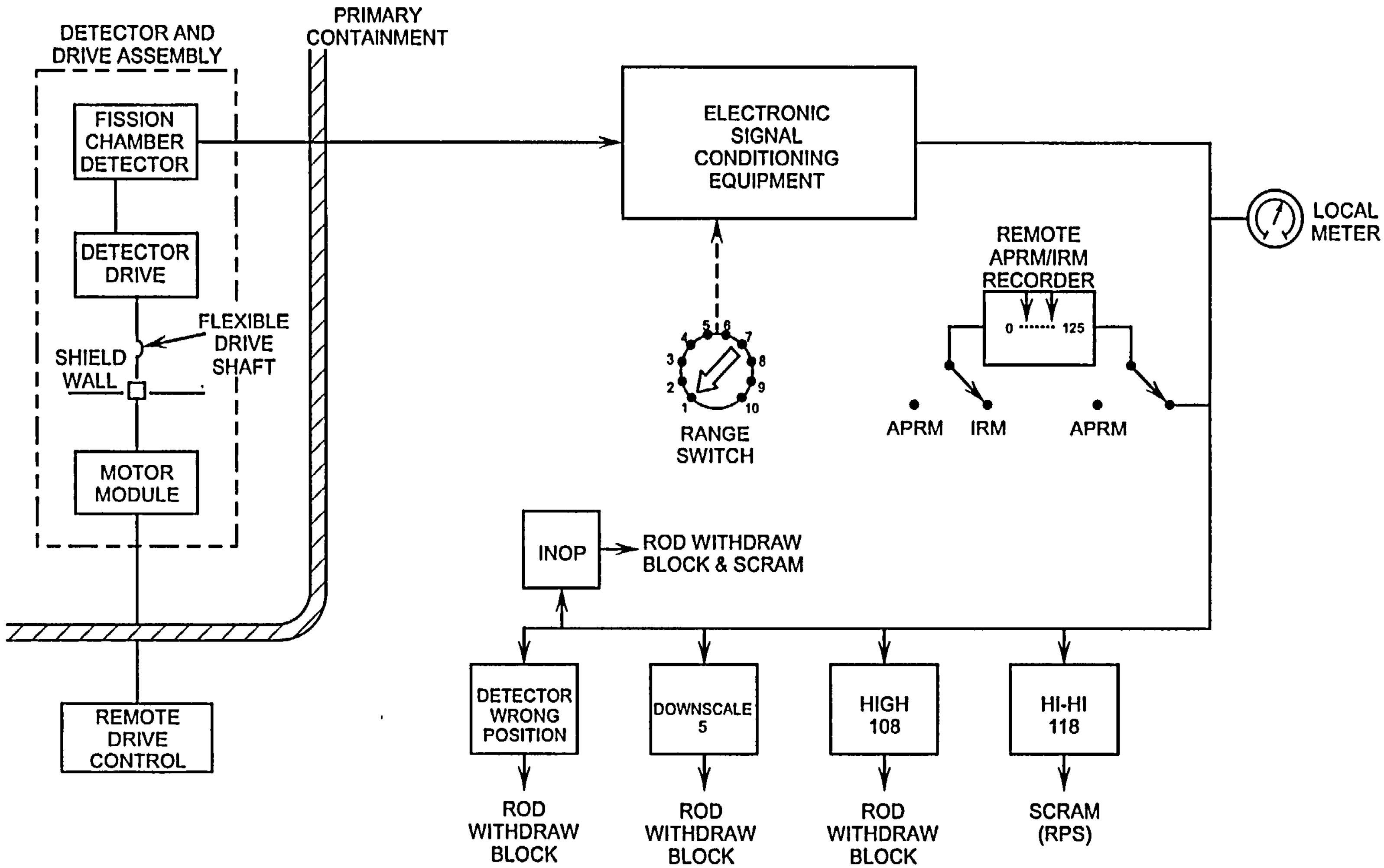


Figure 5.2-1 Intermediate Range Monitor Block Diagram

0102

Table of Contents

List of Figures

5.3 LOCAL POWER RANGE MONITORING SYSTEM

The Purpose of the local power range monitors (LPRM's) is to provide local neutron flux signals at various incore locations to power range monitoring systems and the process computer.

The functional classification of the LPRM System is that of a power generation system with the exception of the APRM System inputs which are functionally classified as safety related.

5.3.1 System Description

The LPRM System consists of stationary incore detectors and electronic signal conditioning equipment. The LPRM detectors are arranged in 31 radially located assemblies, with each assembly containing four detectors spaced axially at three foot intervals. The arrangement of the LPRM detectors provides uniform coverage of the radial and axial core flux distribution. The LPRM assemblies also contain a hollow dry tube for the Traversing Incore Probe (TIP) System (Section 5.5).

The LPRM detectors provide signal inputs to the Average Power Range Monitoring (APRM) System for computing core average flux, the Rod Block Monitoring (RBM) System for local average flux, and the process computer for core thermal limit evaluations.

5.3.2 Component Description

The major components which make up the LPRM System are described in the following paragraphs and illustrated in simplified block diagram form in Figure 5.3-1.

5.3.2.1 LPRM Detector

The LPRM detectors are fixed incore fission chambers, less sensitive than the source range and intermediate range detectors but of similar construction. A total of 124 LPRM detectors are arranged in 31 LPRM assemblies that are radially distributed throughout the core as shown in Figure 5.0-2.

The LPRM detector assemblies are approximately 43 feet in overall length and house the four LPRM detectors plus a traversing incore probe tube (Section 5.6). Within each LPRM assembly the detectors are arranged axially at three foot intervals, beginning 18 inches above the bottom of the active fuel. The LPRM detectors are designated from the bottom to top as A, B, C, and D. As illustrated in Figure 5.3-1, each detector cable runs from below the reactor vessel to individual signal conditioning equipment.

5.3.2.2 Electronic Signal Conditioning Equipment

The electronic signal conditioning equipment consists of LPRM cards where flux amplifiers convert detector current signals to analog voltage signals for use by the process computer, the APRM averaging circuit, and the RBM. Also located on each LPRM card is a function switch that allows calibration of the LPRM amplifier and bypassing an inoperative LPRM detector. The LPRM cards are located in the APRM cabinet that uses the LPRM signal.

The LPRM cards are assigned to the APRM's as follows: (Refer to 5.4 for APRM details)

Channel Used	Number of LPRM's
APRM-A	17
APRM-B	14
APRM-C	17
APRM-D	14

APRM-E	17
APRM-F	14
LPRM-A	17
LPRM-B	14
	124

LPRM cabinets A & B contain the detectors not used in the APRM averaging circuit.

The Rod Block Monitoring (RBM) System selectively uses outputs from all LPRM's through the Reactor Manual Control System (Section 7.1).

5.3.3 System Features

A short discussion of system features is given in the paragraphs which follow.

5.3.3.1 LPRM Display

On the vertical portion of a control room panel there are 16 meters in groups of four, each group consisting of the LPRM detectors in one assembly. The four groups of LPRM detectors read the power in watts per centimeter squared around the selected control rod. This information is used to observe flux changes due to control rod movement and as an input to the Rod Block Monitor for local flux determination (Section 5.5).

5.3.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

5.3.4.1 Average Power Range Monitoring System (Section 5.4)

The Average Power Range Monitoring System receives LPRM detector inputs for averaging, to determine average core flux.

5.3.4.2 Rod Block Monitoring System (Section 5.5)

The Rod Block Monitor receives LPRM inputs from the assemblies surrounding the control rod selected for movement, to calculate average local flux.

5.3.4.3 Traversing Incore Probe System (Section 5.6)

The Traversing Incore Probe System is used to calibrate the LPRM detectors.

5.3.4.4 Process Computer

The process computer receives LPRM detector outputs for determination of core and LPRM performance.

5.3.5 Summary

Classification:
Power generation system except that APRM inputs are safety related.

Purpose:
Provide local neutron flux signals at various incore locations to power range monitoring systems and the process computer.

System Interfaces:
Average Power Range Monitoring System; Rod Block Monitoring System; Traversing Incore Probe System.

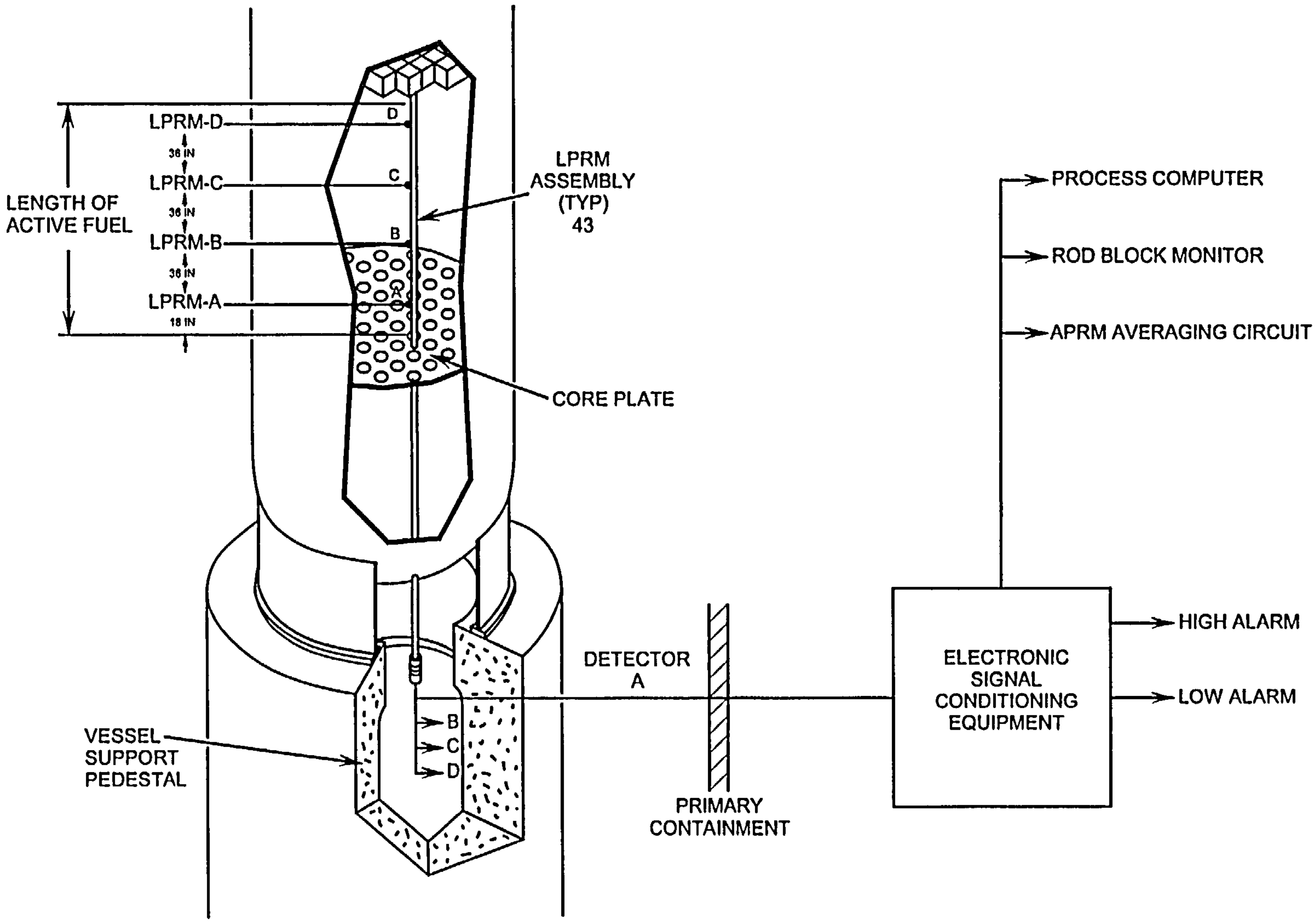


Figure 5.3-1 LPRM Simplified Block Diagram

Table of Contents

List of Tables

List of Figures

5.4 AVERAGE POWER RANGE MONITORING SYSTEM

The purpose of the Average Power Range Monitoring (APRM) System is to monitor core average power by computing the core average thermal flux, output this information for display, and provide protective trips to preserve the fuel clad integrity.

The functional classification of the APRM System is that of a safety related system.

5.4.1 System Description

The Average Power Range Monitoring System is comprised of six APRM channels (A through F). Each channel receives signals from two systems for calculations; i.e., the LPRM System detectors provide local neutron flux signals for averaging, and the Recirculation System provides recirculation loop flows for flux biased scram and rod block settings (Figure 5.4-1)

Each APRM channel averages selected LPRM signals to produce an average core thermal flux signal. The average core thermal flux signal is calibrated to read in per cent core thermal power. The selected LPRM signals ensure a good radial and axial power distribution sampling for an accurate percent core thermal power calculation. Typically a plant of the size and type discussed in this manual has each APRM channel supplied with 17 or 14 LPRM detector inputs.

5.4.2 Component Description

The major components of the APRM System are illustrated in Figure 5.4-1 and discussed in the paragraphs which follow.

5.4.2.1 LPRM Inputs

For information concerning LPRM detectors and instrumentation refer to Section 5.3.

5.4.2.2 APRM Averaging Circuit

The APRM averaging circuit calculates the percent of rated core thermal power by summing the assigned LPRM signals, dividing by the number of operable LPRM input signals, and calibrating the resulting average to read in per cent rated core thermal power. The percent core thermal power signal is used for operator display, process computer information, Rod Block Monitor reference power, and for APRM trips.

5.4.2.3 Count Circuit

A minimum number of 11 LPRM input signals is required to provide an adequate averaged representation of core power. The count circuit counts the number of operable LPRM signals to a given APRM. If the number of operable LPRM inputs is less than 11 the count circuit generates an inoperative trip signal.

Normally, APRM channels A, C, and E each receive 17 LPRM inputs and APRM channels B, D, and F each receive 14 LPRM inputs. Should any of the LPRMs be bypassed, the count circuit automatically reduces the output signal accordingly and provides the averaging circuit with a new number of operable LPRM detectors.

5.4.2.4 Flow Bias Trip Circuit

The flow bias trip circuit provides a variable reference for APRM rod block and scram trip points based on total recirculation flow. Total recirculation flow is the summation of both loop flows which determines the ability to remove heat from the core. Without recirculation loop flow the reference scram is conservatively set at 51%

power (See Figure 5.4-2). From 51% to 118% power , the reference scram signal rises linearly with the flow signal until a flow of 100% is achieved. Hence, from zero to 100% the line rise would be 66%, so the reference scram signal would be .66 x recirculation flow (w) + 51. The flow bias scram signal is limited to 113.5% power even if recirculation flow is in excess of 100%.

5.4.3 System Features

A short discussion of system features is given in the paragraphs which follow.

5.4.3.1 Rod Blocks and Scrams

The APRM System provides rod blocks and scrams to prevent fuel damage. The APRM HIGH-HIGH thermal power scram represents the fuel dynamics which will approximate the reactor thermal power during a transient or steady state condition. The intent of the time delayed trip is to avoid spurious scrams caused by momentary anomalous neutron flux spikes. These trips along with the appropriate bypasses are listed in Table 5.4-1.

5.4.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

5.4.4.1 Reactor Protection System (Section 7.3)

The Reactor Protection System receives scram signals from the APRM system.

5.4.4.2 Local Power Range Monitoring System (Section 5.3)

The Local Power Range Monitoring System detectors provide the radial and axial flux distribution signals used to determine percent core average thermal power.

5.4.4.3 Rod Block Monitoring System (Section 5.5)

The Rod Block Monitoring System uses selected APRM channel output signals for a reference core thermal power.

5.4.4.4 Reactor Manual Control System (Section 7.1)

The Reactor Manual Control System receives rod block signals from the APRM System.

5.4.4.5 Recirculation System (Section 2.4)

The APRMs use recirculation loop flows for flow bias rod blocks and scrams in the RUN mode.

5.4.4.6 Intermediate Range Monitoring System (Section 5.2)

The Intermediate Range Monitoring System provides inputs to the APRM system for downscale scram and rod blocks.

5.4.5 Summary

Classification:

Safety related system

Purpose:

To compute the core average thermal power and output this information for display and provide protective trips to preserve the fuel clad integrity.

Components:

LPRM inputs; averaging circuits; count circuit; total recirculation flow input; display; trips.

System Interfaces:

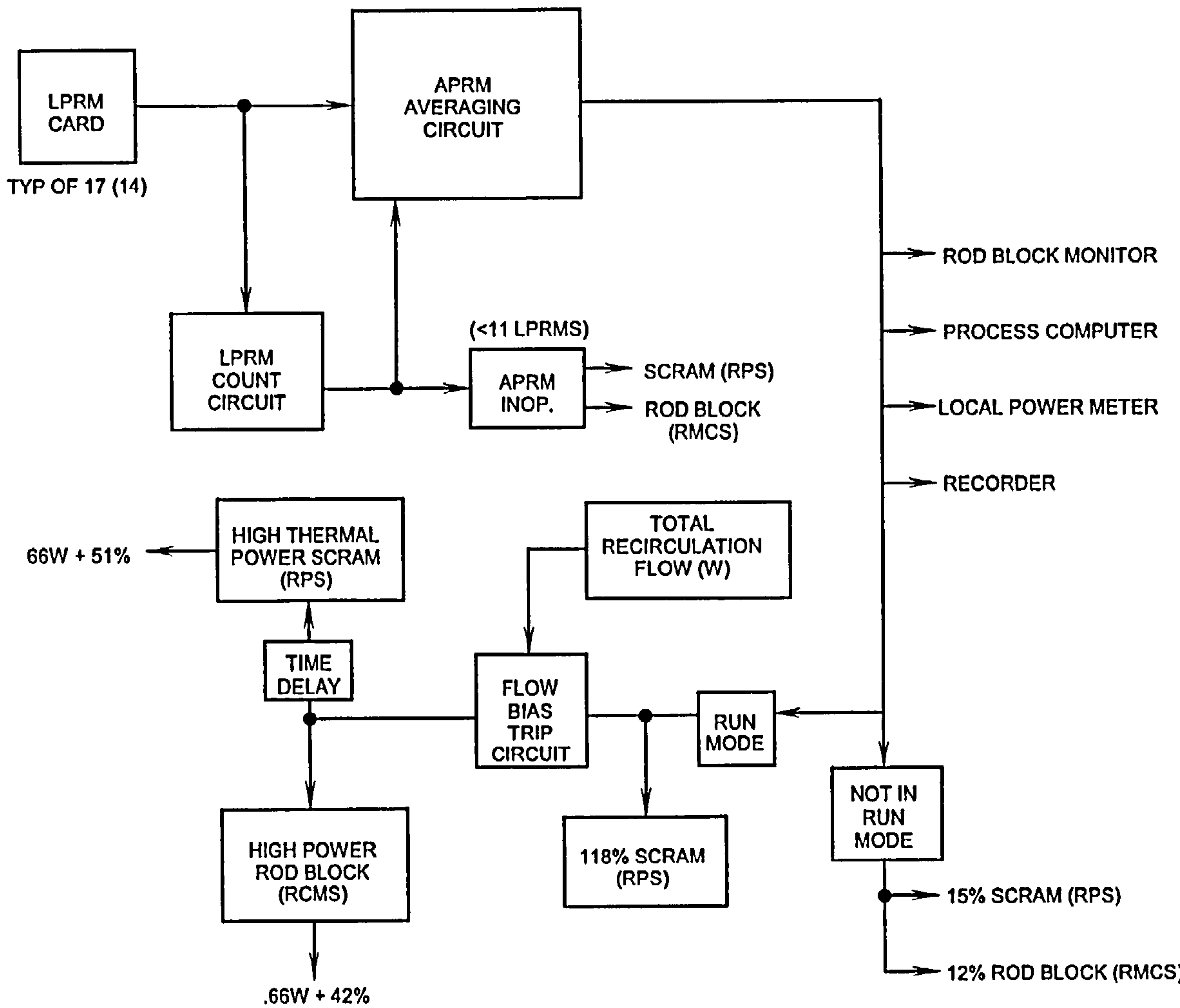
Local Power Range Monitoring System; Intermediate Range Monitoring System; Rod Block Monitoring System; Reactor Protection System; Reactor Manual Control System; Process Computer System; Recirculation System.

Table 5.4-1 APRM Scram and Rod Block Settings

Alarm Trip	Setpoint	Action	Auto Bypass
APRM Downscale	3%	Rod Block	Other than "Run"
APRM High	12%	Rod Block	In "RUN"
APRM High	.66(W) + 42% 108% Max.	Rod Block	Other than "RUN"
APRM Flow Bias not normal.	110% or Δ Flow 10%	Rod Block	No auto Bypass
APRM High-High Thermal Power	.66(W) + 51% 113.5% Max.	Scram	Other than "RUN"
APRM High-High Fixed Power	118%	Scram	Other than "RUN"
APRM Inop[1]		Rod Block & Scram	No auto Bypass
APRM High High	15%	Scram	In "RUN"
APRM downscale with companion IRM Hi Hi or inop.	3%	Scram	Other than "RUN"
APRM[2] Bypassed		Bypasses all trip functions if an APRM is by passed.	Bypasses all trip functions if an APRM is by passed.

1 <11 LPRM inputs, module unplugged, not in operate.

2 One channel per RPS group (A,C,E or B, D, F).



0102

Figure 5.4-1 APRM Simplified Block Diagram

Table of Contents

List of Tables

List of Figures

5.5 ROD BLOCK MONITORING SYSTEM

The purpose of the Rod Block Monitoring (RBM) System is to prevent exceeding thermal hydraulic limits in a local region of the core due to a single rod withdrawal error from a limiting control rod pattern. A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (Section 1.13).

The functional classification of the RBM System is that of a power generation system.

5.5.1 System Description

The RBM System consists of two separate and independent channels, A and B. Each channel monitors the local neutron flux during selection and movement of a control rod, and generates trip signals to actuate rod withdraw blocks when the monitored neutron flux exceeds preset limits.

The RBM accomplishes this function by averaging the selected LPRM inputs (local neutron flux) and applying the resultant signal to trip circuits for comparison with flow bias trip points. As long as the selected LPRM average is less than the flow bias trip points, no rod withdrawal blocks will be applied.

5.5.2 Component Description

The major components which make up the RBM system are discussed in the following paragraphs and in block diagram form in Figure 5. 5-1.

5.5.2.1 Input and Assignment Matrix

The RBM input and assignment matrix receives a rod select signal from the Reactor Manual Control System (RMCS). Once the selected control rod signal is received the input and assignment matrix automatically selects the LPRM detectors adjacent to the control rod selected for movement. The A and C level LPRM detectors are assigned to A RBM channel while the B and D level LPRM detectors are assigned to B RBM channel. The number of LPRM's providing inputs to the rod block monitor channels is dependent on the location of the selected control rod. The maximum number of LPRM inputs for any one RBM channel is eight.

5.5.2.2 Rod Block Monitor Channels

By monitoring the LPRM detectors around a selected control rod, a determination of local power is made. The two levels of LPRM inputs are averaged and then compared to the core average power signal provided by the appropriate Average Power Range Monitor channel.

When local power in the selected rod's location is lower than the average power of the core, rod withdrawal may cause a large rate of change and percent increase of local core power thus causing possible fuel damage. The RBM compensates for this high rod worth by automatically adjusting the Rod Block Monitor signal to equal its reference APRM, thus closer to the rod block trip points.

5.5.2.3 Flow Bias Rod Block

The flow bias rod block provides a reference for RBM rod block trip points based on total recirculation flow. Total recirculation flow is a measure of core flow which determines the ability to remove heat from the core. For further explanation of flow biasing, refer to the APRM System (Section 5.4.2.4).

5.5.3 System Features

A short discussion of system features is given in the paragraphs which follow.

5.5.3.1 Rod Blocks

During reactor operation above 30% power the RBM system provides alarm and rod withdraw blocks to prevent rod withdraw during certain conditions. These conditions along with the appropriate bypasses are listed in Table 5.5-1.

5.5.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

5.5.4.1 Reactor Manual Control System (Section 7.1)

The Reactor Manual Control System receives input signals from the RBM system to generate the rod blocks listed in Table 5.5-1. The RMCS also provides a rod select signal used for the LPRM selection.

5.5.4.2 Local Power Range Monitoring System (Section 5.3)

The RBM system receives Local Power Range Monitoring System inputs from the LPRM assemblies surrounding the control rod selected for movement.

5.5.4.3 Average Power Range Monitoring System (Section 5.4)

The RBM system uses selected Average Power Range Monitoring System channel output signals for a reference core thermal power.

5.5.4.4 Recirculation System (Section 2.4)

The recirculation loop flows provide the total recirculation flow signals used for the flow bias rod block settings.

5.5.5 Summary

Classification:

Power Generation System

Purpose:

Prevent exceeding a thermal hydraulic limit in a local region of the core by a single rod withdraw error from a limiting control rod pattern.

Components:

Input and assignment matrix; RBM channels A and B; trip reference.

System Interfaces:

Local Power Range Monitoring System; Average Power Range Monitoring System; Reactor Manual Control System.

Table 5.5-1 RBM Interlocks and Trips

Alarm or Trip	Setpoint	Panel Annunciator	Action	Auto Bypass
RBM Upscale	.66W+41% Pwr. .66W+33% Pwr. .66W+25% Pwr.	RBM Upscale/Inop	Rod Withdrawal Block	Reference APRM <30% Pwr. or Edge Rod Selected
RBM Downscale	2.5% Pwr.	RBM Downscale	Rod Withdrawal Block	
RBM Inop	*	RBM Upscale/Inop	Rod Withdrawal Block	
APRM Reference Downscale	30% Pwr.			
RBM Bypassed	**Bypass switch			

*Produced by:
1. Local Panel mode switch not in operate
2. Module unplugged.
3. Less than required number of LPRM inputs.
4. RBM fails to null.

** Only one RBM may be bypassed.

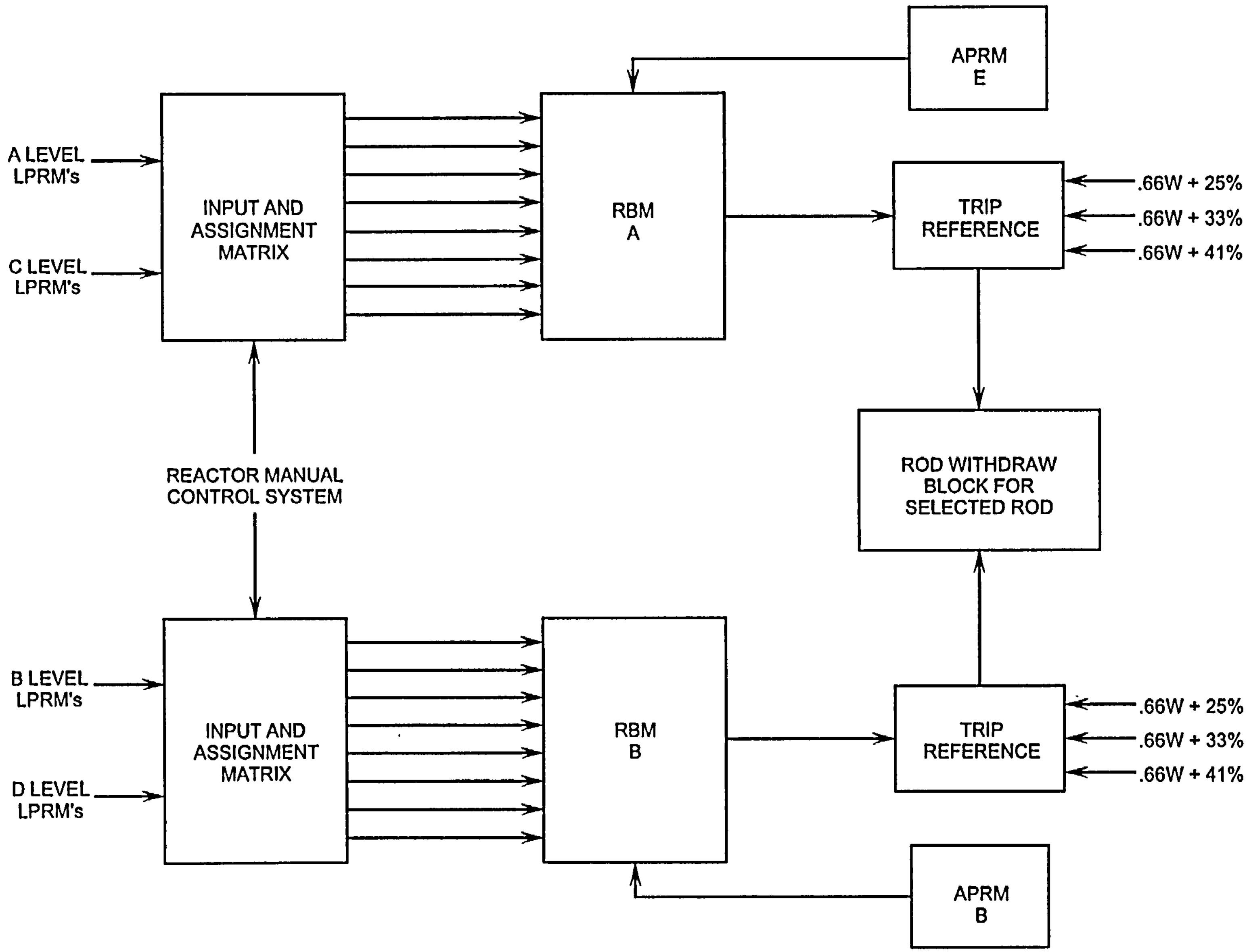


Figure 5.5-1Block Diagram of Rod Block Monitoring System

0102

Table of Contents

List of Figures

5.6 TRAVERSING INCORE PROBE SYSTEM

The purpose of the Traversing Incore Probe (TIP) System is to provide a means of measuring axial thermal neutron flux in 31 fixed core locations for calibration of local power range monitor detectors and updating computer data.

The functional classification of the TIP System is that of a power generation system.

5.6.1 System Description

The Traversing Incore Probe (TIP) System, illustrated in Figure 5.6-1, consists of four independent neutron detection units. Each unit contains a miniature fission chamber (probe) connected to a flexible drive cable that is driven by a motor drive mechanism. Operation of the drive mechanism causes the fission chamber to be inserted or retracted from the reactor core within individual TIP guide tubes. Each TIP unit uses an indexing device to route the detector to the desired LPRM assembly which houses the TIP guide tube.

The 31 LPRM assemblies are divided between the four TIP machines with one common LPRM assembly connected to all four TIP's for cross calibration of the TIP's. The output signal from a TIP channel may be used to plot an axial flux profile on an X-Y recorder and/or provide a flux distribution signal into the process computer for LPRM calibration data.

5.6.2 Component Description

The major components of the TIP System are discussed in the paragraphs that follow.

5.6.2.1 TIP Detector Assembly

The traversing incore probe assembly consists of a fission chamber and the signal drive cable. The TIP detector is a miniature fission chamber having similar characteristics of the LPRM detectors (Section 5.3)

The signal drive cable consists of a signal cable enclosed in a mechanical drive cable. The signal cable, a single coaxial cable, is used to transmit the detector current signal to amplifier and readout equipment. The drive cable provides protection for the signal cable and a means to move the TIP detector. The drive cable is constructed of carbon steel in a helix array. The drive mechanism engages this helix to move the TIP detector into and out of the reactor core.

5.6.2.2 Drive Mechanism

The drive mechanism consists of a metal enclosure housing a drive motor, cable reel assembly, and various electrical equipment needed for proper operation of the TIP drive mechanism.

The drive motor acts on commands from the control console to provide forward or reverse cable drive. The motor control circuits are electrically and mechanically interlocked to prevent invalid cable drive commands.

The take-up reel provides an electrical connection between a fixed cable and one that moves without the necessity of slip rings, brushes, or other coupling methods of relatively short life or high signal loss.

5.6.2.3 Chamber Shield

The chamber shield provides shielding for the detector, which is radioactive from core exposure.

The chamber shield includes limit switches which actuate status lights on the drive control console.

5.6.2.4 TIP Isolation Valves

Each TIP is equipped with two independent means of isolating the TIP tubes from the primary containment in the event of a loss of coolant accident. A ball valve and a cable shearing valve are mounted outside of the primary containment as shown in Figure 5.6-1. The ball valve is open only when the TIP is in use. The shear valve, an explosive valve, is normally open, and is used when the TIP detector is beyond the ball valve and will not retract or the ball valve fails to close when the TIP is retracted.

5.6.2.5 Index Mechanism

The index mechanism is a circular transfer machine with ten possible indexing points. Position number ten of each indexer is connected to 4-way connectors, shown in Figure 5.6-1, which permits all four TIP detectors to scan (one at a time) one common centrally located LPRM assembly. This arrangement provides a means to cross calibrate all four TIP machines during any phase of power operation.

5.6.2.6 Flux Probing Monitor

The flux probing monitor houses the flux amplifiers, power supplies, calibration unit, and electrical equipment needed to transmit the detector signal to the process computer and/or the X-Y recorder.

5.6.2.7 X-Y Recorder

The X-Y recorder operates during the core traverse and plots a graph of flux density versus detector position.

5.6.2.8 TIP Purge System

To prevent corrosion of the drive cable and deterioration of the lubricant used in the guide tubes a constant purge is applied by dry instrument air or nitrogen.

5.6.3 System Features

A short discussion of system features is given in the paragraphs which follow.

5.6.3.1 System Operation

The TIP system serves as an analytical device and is not normally in continuous operation. When the need for a core traverse is determined, the TIP machine can be operated in automatic or manual modes. The TIP's are normally used for the following situations: to verify a particular LPRM reading, to obtain an axial flux profile, and for calibration of LPRM detectors.

5.6.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

5.6.4.1 Local Power Range Monitoring System (Section 5.3)

The TIP System is used to calibrate LPRM System detectors and uses dry tubes mounted in the LPRM assemblies.

5.6.4.2 Process Computer (Section 6.3)

The TIP System provides signals to the process computer for core performance calculations.

5.6.5 Summary

Classification:

Power generation system.

Purpose:

To provide a means of measuring axial thermal neutron flux in 31 fixed core locations, for calibration of LPRM detectors and updating computer data.

Interfaces:

Local Power Range Monitoring System; Process Computer.

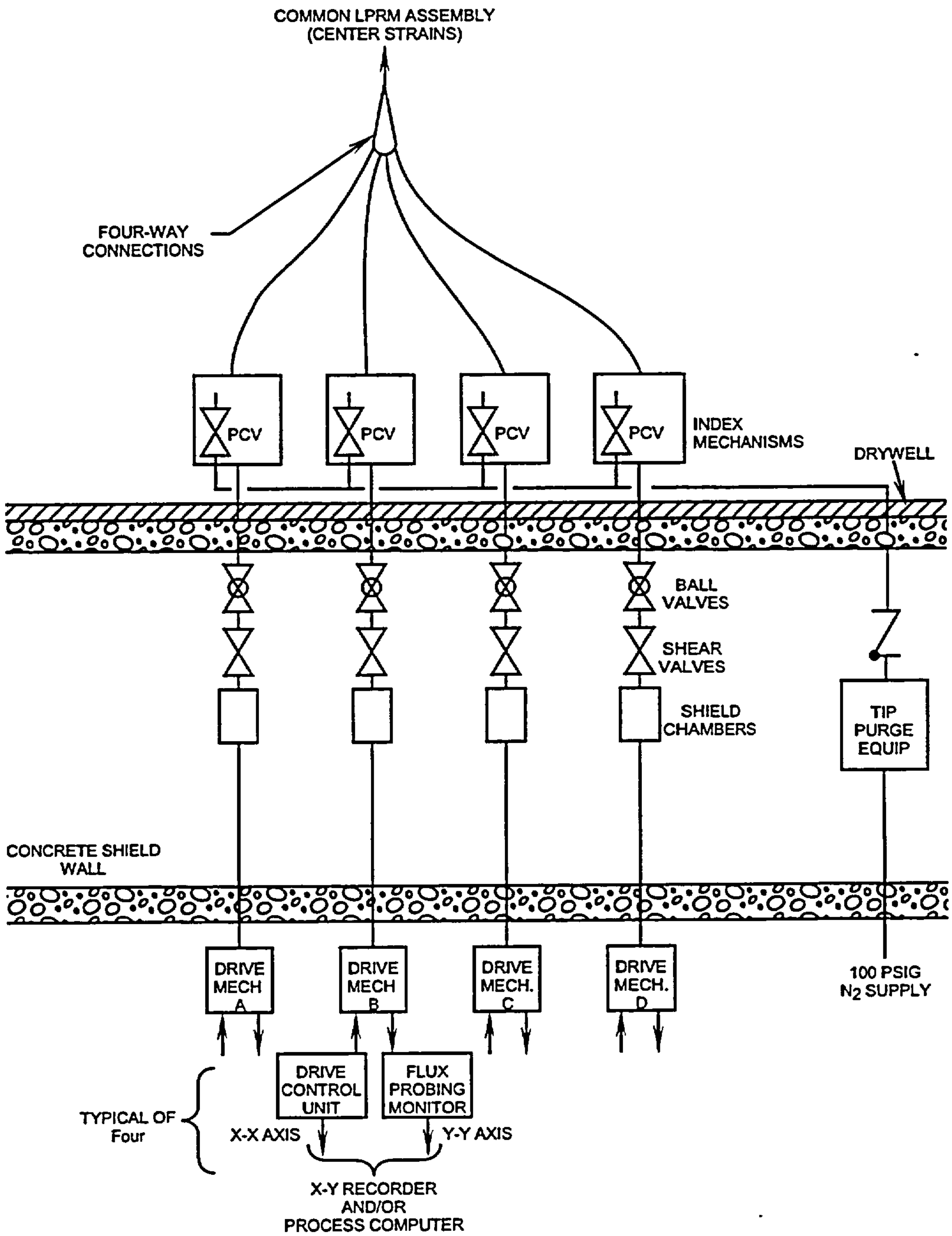


Figure 5.6-1 Traversing Incore Probe System

6.0 COMPUTER SYSTEMS

This chapter contains information on the Rod Worth Minimizer (RWM), the Rod Sequence Control System (RSCS), and the Process Computer. The RWM and the Process computer are computer systems, the RSCS is not. The RSCS is included in this chapter because it is a backup to the RWM.

6.0.1 Rod Worth Minimizer (Section 6.1)

The Rod Worth Minimizer (RWM) serves as a backup to procedural controls to limit control rod worth during startup and low power operation. This helps limit the reactivity addition rate in the event of a control rod drop accident.

6.0.2 Rod Sequence Control System (Section 6.2)

The Rod Sequence Control System (RSCS) is a backup system to the Rod Worth Minimizer. It independently imposes restrictions on control rod movement to mitigate the effects of a control rod drop accident.

6.0.3 Process Computer (Section 6.3)

The Process Computer's primary function is to perform reactor core calculations and provide the plant operating staff with current core operating data. In addition, it is the function of the computer to monitor, calculate, store, log, and alarm information which has been collected by the plant instrumentation.

Table 6.0-1 Funtional Comparisons of RWM, RSCS, and RBM

Restraints for 110% Rod Density to 50% Rod Density

RWM

Sequencing of groups 1-4 (1 before 2, 2 before 3, 3 before 4)

Protection provided by insert and withdraw blocks

RSCS

Allows selection of only A1/2 until fully withdrawn then A3/4 until fully withdrawn or vice versa (same for sequence B).

Protection provided by selection blocks.

Restraints for 50% Rod Density to 20% Power

RWM

Sequencing of groups 5-71.
Withdraw and insert limit restraints to enforce banking.
Group Notch Control.

RSCS

Allows selection of any remaining rod groups but enforces Group Notch Control.

Greater Than 20% Power

RBM

Rod movement is restricted by rod withdraw blocks being applied at three flow bias setpoints (greater than 30% power).

Table of Contents

List of Figures

6.1 ROD WORTH MINIMIZER

The purpose of the Rod Worth Minimizer (RWM) is to serve as a backup to procedural controls to limit control rod worth during low power operation so that the postulated Rod Drop Accident will not exceed the design limit.

The functional classification of the RWM System is that of a safety related system.

6.1.1 System Description

The Rod Worth Minimizer is a computer monitoring system which minimizes control rod reactivity worth by blocking rod movement if the existing control rod pattern deviates from a specific sequence. The sequences are developed by the plant nuclear engineers and loaded into the RWM memory. Actual rod positions are obtained for comparison to the sequence from the Rod Position Information System (RPIS).

Rod movement sequences are developed to limit rod worth to a level below which, if a Rod Drop Accident were to occur at a free-fall rate limited by the velocity limiter, the fuel enthalpy from the transient would be less than 280 calories/gram. Figure 6.1-1 shows rod worth curves relative to the danger level (that which would result in >280 cal/gm) for unrestrained rod movement (curve A) and Rod Worth Minimizer restrained movement (curve B). Due to the lower rod worth at power, the Rod Worth Minimizer is not needed to limit rod worth above 20% power.

6.1.2 Component Description

The major components of the Rod Worth Minimizer System are the computer program and the Operator's Display Panel.

6.1.2.1 Rod Worth Minimizer Program

To understand the program, definitions of terms used in the program are given below.

6.1.2.1.1 Operating Sequence

The Rod Worth Minimizer program contains an operating sequence which is loaded into the computer memory. The operating sequence is a schedule to be followed by the plant operator when withdrawing or inserting control rods. The sequence identifies the control rod by XX-YY coordinates and the positions to which each rod should be withdrawn in going from shutdown to full power. When reducing power, the rods are inserted in the reverse order of their withdrawal. The operating sequence is sequentially sub-divided into rod groups.

Each rod group consists of a number of specified control rods and a set of insert and withdraw position limits that apply to each rod in the group. The groups are numbered in the order in which they are to be withdrawn when raising power. Each sequence generally begins by withdrawing approximately half the rods in the core full out. Under cold conditions, this brings the reactor to the point of criticality and to heating power. The fully withdrawn control rods are distributed in a checker board (black and white) pattern shown in Figure 6.1-2. The remaining rods are then withdrawn to either full out or intermediate positions in the order specified by the sequence.

6.1.2.1.2 Notch Limits

All rods in groups higher than that in which the black and white pattern is achieved have notch control restraints superimposed on the normal group limits. This means that in addition to remaining within the group limits, any rod contained in one of these notch control groups must also remain within one notch position of every other rod in the same group.

6.1.2.1.3 Low Power Set Point

The low power setpoint is the core average power level below which the Rod Worth Minimizer program is actively enforcing adherence to the operating sequence of rod withdrawals or insertions. When the core power level is above the low power setpoint, the program does not impose any rod blocks as a result of rod movement by the operator. The low power setpoint is set above the level of required enforcement (20% power) and is sensed by both total steam flow and total feedwater flow being greater than 30% of rated power.

6.1.2.1.4 Withdraw Error

A withdraw error can occur either when a rod contained in the current group or any lower group is withdrawn past the withdraw limit for the group, or if a rod contained in a group higher than the current group is withdrawn past the insert limit for the higher group.

6.1.2.1.5 Insert Error

An insert error occurs when a rod contained in the current group is inserted past the insert limit for this group, or if a rod contained in a group lower than the current group is inserted past the withdraw limit for the lower group.

6.1.2.1.6 Select Error

A select error occurs whenever the operator selects a rod other than one contained in the current rod group. The select error provides the operator with warning that he has selected a rod, which if moved, will create an insert or withdrawal error.

6.1.2.1.7 Notch Error

A notch error occurs whenever the reactor is operating in a rod group higher than that in which a black and white pattern is achieved and notch limits are violated (rods in a group are more than one notch apart).

6.1.2.1.8 Withdraw Block

The RWM program will impose a control rod withdrawal block whenever:

1. A withdraw error has been made. This block applies to all control rods.
2. An insert block has been imposed by the Rod Worth Minimizer. This block applies to any control rod selected for movement except an existing insert error rod.
3. A notch error is made. This block applies to any control rod selected for movement except those which could be withdrawn to clear the notch error.

The purpose of these blocks is to force correction of existing errors before allowing further rod movement.

6.1.2.1.9 Insert Block

The RWM program will impose a control rod insert block whenever:

1. A third insert error is made. This block applies to all control rods.
2. A withdraw block has been imposed by the RWM. This block applies to any control rod selected for movement except an existing withdraw error rod.

3. A notch error is made. This block applies to any control rod selected for movement except those which could be inserted to clear the notch error.

The purpose of this block is to force correction of existing errors before allowing further rod movement.

6.1.2.2 Rod Worth Minimizer Operator's Panel

All the operating controls and indicators for the Rod Worth Minimizer are located on the operator's panel which is illustrated in Figure 6.1-3.

6.1.3 System Features

A short discussion of system features is given in the paragraphs which follow.

6.1.3.1 Normal Operation

Control rods are withdrawn according to the operating sequence. The Rod Worth Minimizer sequence restraints require that the rod groups be pulled in sequential order to specific group limits. The control rods within a group may be pulled in any order. Some flexibility is permitted by allowing two insert errors before rod blocks are applied. One withdrawal error will cause a rod withdrawal block. The rod blocks are normally applied so that only rod movements to correct errors are allowed. Thus, an operator must correct the errors before the RWM will permit further rod movements. Once the black and white pattern is obtained notch limits must be observed in addition to group limits. The rods within the group must remain within one notch position of every other rod in the group. A notch error exists if notch limits are violated resulting in rod blocks being applied forcing the correction of the notch error before rod movements can continue.

6.1.3.2 Manual and Auto Bypass

There are two ways in which the Rod Worth Minimizer function can be bypassed. It is automatically bypassed when reactor power is above the low power set point as determined by both total steam flow and total feed flow greater than 30% of rated. It can be manually bypassed at any power level by turning a keylock switch on the operator's panel from NORMAL to BYPASS.

6.1.3.3 Rod Drop Accident

For a Rod Drop Accident to occur, the following sequence of events must take place: (1) failure of the control rod to control rod drive mechanism coupling, (2) sticking of the control rod blade in the inserted position as the control rod drive mechanism is withdrawn (3) full withdrawal of the control rod drive mechanism, and (4) the control rod blade becomes unstuck and falls. The result is a very rapid power spike in the fuel which is such short duration that virtually all of the energy generated is deposited in the fuel pellets causing the fuel temperature to increase. As the fuel temperature increases the fuel will melt, vaporize, and begin to pressurize the fuel rod eventually causing fuel failure. The duration of the power spike and the energy deposited in the fuel determines the extent of fuel damage. 280 calories/gram fuel enthalpy is considered the design limit, beyond which rapid fuel rod failure is expected.

6.1.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

6.1.4.1 Reactor Manual Control System (Section 7.1)

The Reactor Manual Control System provides rod selected data and receives the rod block signals.

6.1.4.2 Control Rod Drive System (Section 2.3)

The RWM system receives control rod position indication of each and every control rod drive mechanism.

6.1.4.3 Feedwater Control System (Section 3.3)

The Feedwater Control System provides total steam flow and total feedwater flow signals for the automatic bypass.

6.1.4.4 Process Computer (Section 6.3)

The Process Computer is the central component of the Rod Worth Minimizer.

6.1.4.5 Rod Sequence Control System (Section 6.2)

The Rod Sequence Control System provides the backup to the Rod Worth Minimizer.

6.1.5 Summary

Classification:

Safety related system.

Purpose:

Serves as a backup to procedural controls to limit control rod worth during low power operation so that the postulated Rod Drop Accident will not exceed the design limit.

Components:

Process Computer, RWM Program, RWM Operator Panel.

System Interfaces:

Process Computer; Rod Sequence Control System; Reactor Manual Control System; Control Rod Drive System.

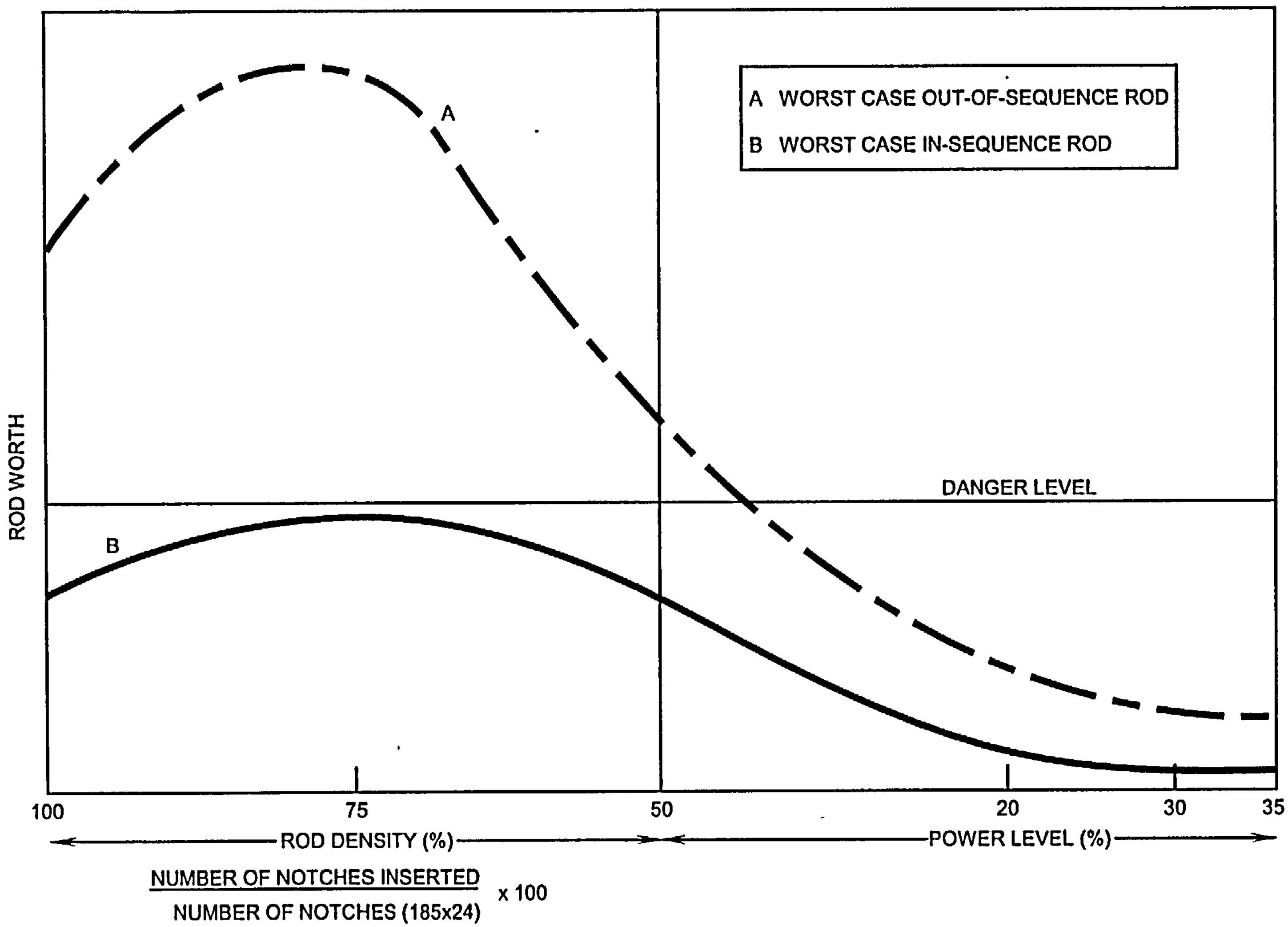


Figure 6.1-1 Rod Worth for Sequences of Rod Withdrawal or Insertion

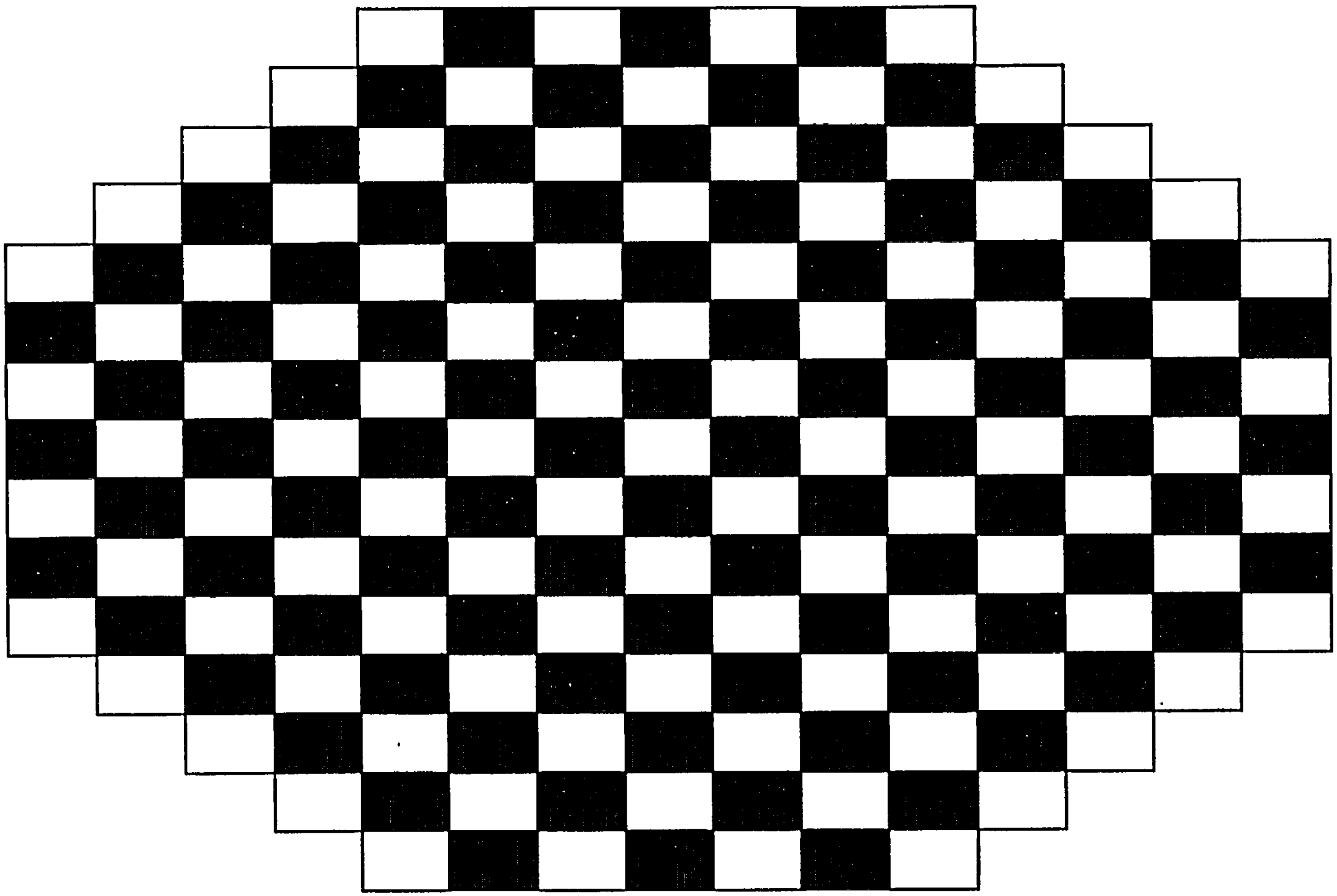


Figure 6.1-2 BLACK AND WHITE PATTERN

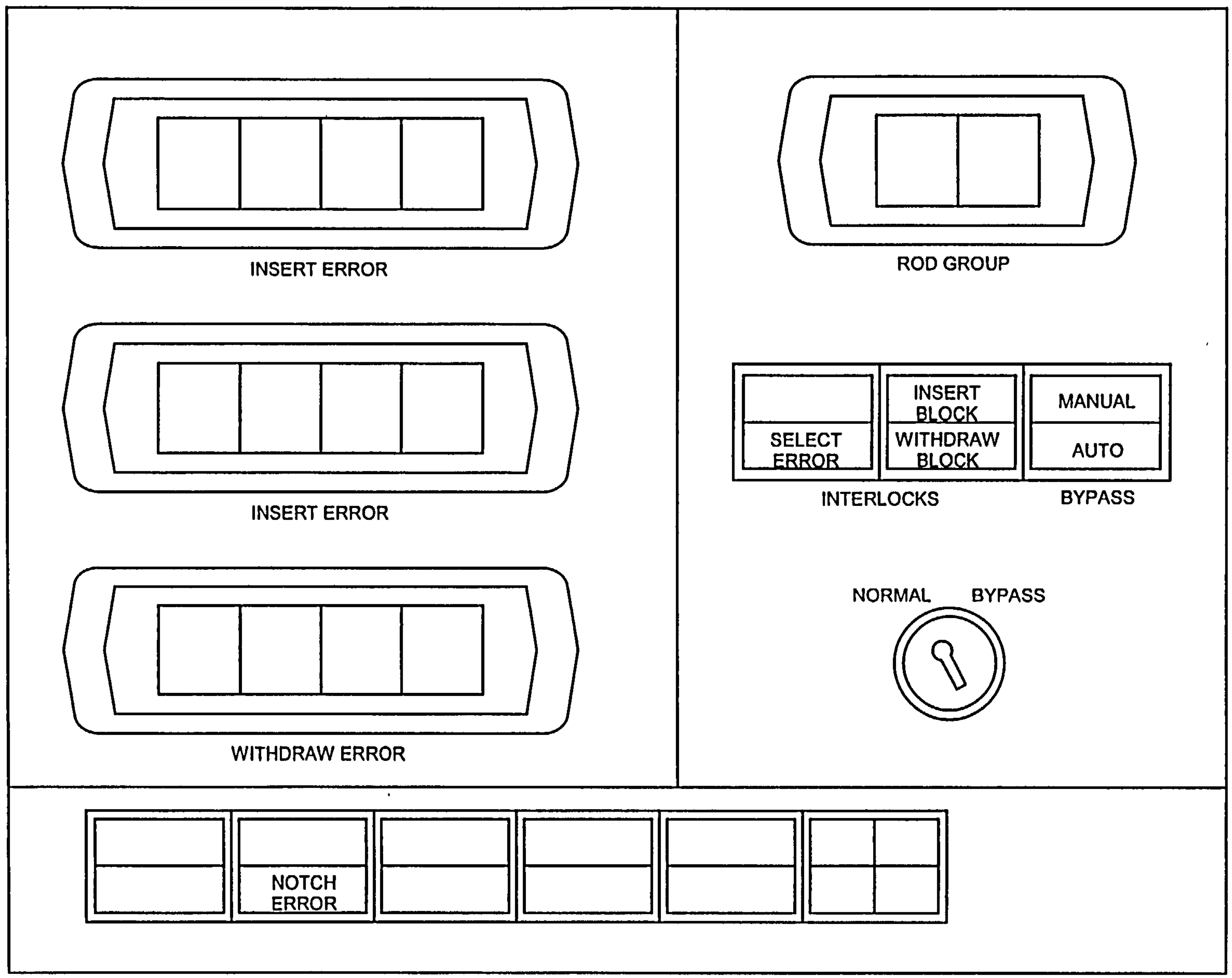


Figure 6.1-3 RWM Operator's Display Panel

Table of Contents

List of Figures

6.2 ROD SEQUENCE CONTROL SYSTEM

The purpose of the Rod Sequence Control System (RSCS) is to act as a backup to the Rod Worth Minimizer to ensure minimum rod worth to reduce the consequences of a postulated rod drop accident.

The functional classification of the RSCS is that of a safety related system.

6.2.1 System Description

The Rod Sequence Control System restricts rod movement to minimize the individual worth of control rods to lessen the consequences of a Rod Drop Accident. Control rod movement is restricted through the use of rod select, insert, and withdrawal blocks. The Rod Sequence Control System is a hardwired, redundant backup to the Rod Worth Minimizer. It is independent of the Rod Worth Minimizer in terms of inputs and outputs but the two systems are compatible. It is not required above 20% power.

The Rod Sequence Control System was designed as a backup to the Rod Worth Minimizer for the following reasons; 1) The Rod Worth Minimizer has had a poor reliability record, 2) The Rod Worth Minimizer can fail in an unsafe manner, and 3) The Rod Worth Minimizer is easily bypassed.

The Rod Sequence Control System operation is divided into two modes shown in Figure 6.2-1 with the black and white rod pattern being the division point. At less than a black and white rod pattern, the Sequence Control Mode controls rod movement from rods full-in to the black and white rod pattern by imposing select blocks. The Group Notch Control Mode controls rod movement from the black and white rod pattern to 30% power by imposing rod withdrawal and insert blocks.

6.2.2 Component Description

The components of the Rod Sequence Control System are grouped as belonging to the Sequence Control Mode or the Group Notch Control Mode of operation.

6.2.2.1 Sequence Control Mode

The Sequence Control Mode controls rod movement from rods full in to the black and white rod pattern by imposing rod select blocks. These rods are divided into two rod groups which are compatible with Rod Worth Minimizer rod groups. From an all rods full in condition, the operator may choose either of the two groups to begin movement. Once he begins to withdraw the first rod in a group, the logic will not allow selection of any rods but those in the chosen group, until all rods in that group are moved to the full out position. When all rods in the second group are moved full out, the Rod Sequence Control System will switch to the Group Notch Control Mode.

The sequence control logic makes decisions on the basis of inputs from the Control Rod Drive System. Each control rod drive mechanism provides full-in and full out position information to the Rod Sequence Control System. This information is derived from redundant switches in the control rod drive mechanism position indicating probe, and is not used for digital display or by the Rod Worth Minimizer.

The sequence control logic will not allow selection of out of sequence control rods for movement.

6.2.2.2 Group Notch Control Mode

The Group Notch Control Mode controls rod movement from the black and white pattern to the 30% power bypass, by imposing rod withdrawal and insert blocks. All control rods are assigned to notch control groups which are compatible with Rod Worth Minimizer rod groups.

Group notch control logic requires that all rods within a notch control group must remain within one notch. Once a rod is moved in either direction in a notch group, rod blocks are imposed on; (l) the initially moved rod to prevent further movement in the same direction, and (2) all other rods in that group to prevent movement in the opposite direction. After the initial movement, the logic is reset whenever all rods in the notch group are again at the same position. The logic consists of a set of memory units, one for each notch group. The memory units track the relative position of the rods in each group by sensing the rod selected, direction of movement requested, and the occurrence of the Reactor Manual Control System timer settle function (Section 7.1). The logic output signals are applied to the Reactor Manual Control System as withdraw or insert blocks.

6.2.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

6.2.3.1 Comparison to Rod Worth Minimizer

The Rod Sequence Control System restraints are designed to be compatible with those of the Rod Worth Minimizer. Several differences in philosophy between the two systems are outlined below:

1. The Rod Worth Minimizer is more restrictive in the sequencing of rod movements.
2. The Rod Worth Minimizer applies re straints only after the operator has deviated from the operating sequence. The Rod Sequence Control System restraints are applied so that the operator is not allowed to deviate.
3. The Rod Worth Minimizer can be entirely bypassed manually by the operator. The Rod Sequence Control System has only limited manual bypass capability.
4. The Rod Worth Minimizer is computer software and can be changed by a programmer. The Rod Sequence Control System is completely hardwired.

6.2.3.2 Bypass Capability

The Rod Sequence Control System is not required to limit rod worth at greater than 20% reactor power. A system bypass signal is generated at a conservative level of 30%, as measured by a pair of pressure sensors which monitor the main turbine first stage pressure.

The circuitry does allow certain bypass capability. In the Sequence Control logic, the full in <u>or</u> full out position for each rod can be bypassed. This is necessary for certain surveillance tests. In the Group Notch control logic, each notch group memory has a reset button which will reset the memory regardless of the previous latch states.

6.2.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

6.2.4.1 Control Rod Drive System (Section 2.3)

The RSCS receives full-in and full-out control rod drive position, for every drive mechanism, from the Control Rod Drive System.

6.2.4.2 Reactor Manual Control System (Section 7.1)

The Reactor Manual Control System provides the signals for rod selected, direction of rod movement, and the timer settle function. The Rod Sequence Control System restrains rod movement by imposing rod select, insert, and withdraw blocks through the Reactor Manual Control System.

6.2.4.3 Main Steam System (Section 2.5)

The Main Steam System provides power level bypass signals from the main turbine first stage pressure.

6.2.5 Summary

Classification:

Safety related system

Purpose:

To act as a backup to the Rod Worth Minimizer to ensure minimum rod worth to reduce the consequences of a postulated rod drop accident.

Components:

Sequence Control; Group Notch Control.

System Interfaces:

Control Rod Drive System; Reactor Manual Control System; Main Steam System.

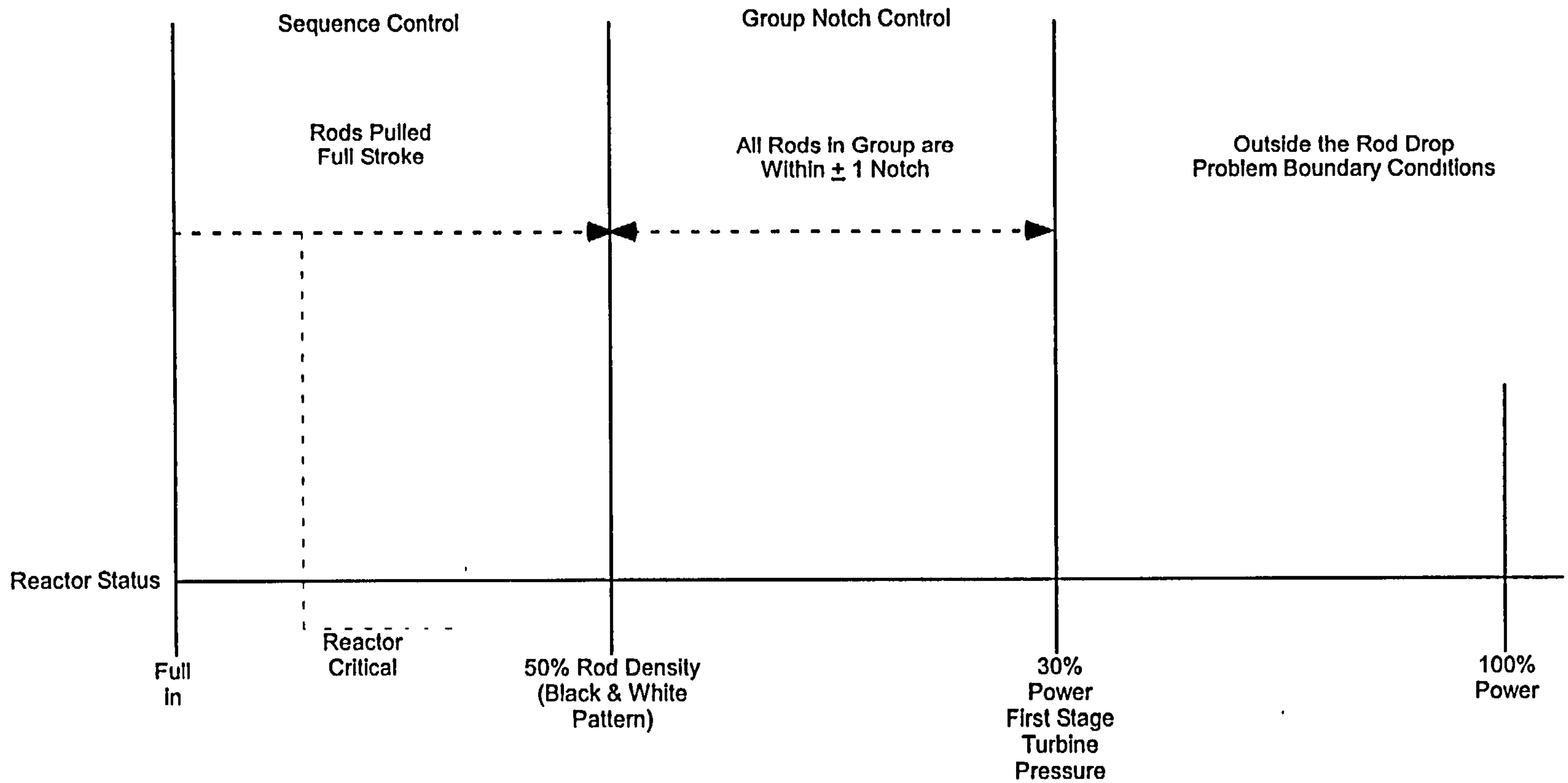


Figure 6.2-1 Rod Sequence Control System

0102

Table of Contents

List of Figures

6.3 PROCESS COMPUTER

The purpose of the Process Computer is to provide the plant operating staff with current plant performance data.

The functional classification of the Process Computer is that of a power generation system.

6.3.1 System Description

The Process Computer provides on-line monitoring of several hundred input points representing significant plant process variables. The system scans process inputs at specified intervals and issues appropriate alarm indications and messages if monitored values exceed predefined limits or digital trip signals occur. It performs calculations with selected input data to provide the operator with essential plant performance information through a variety of logs, trends, summaries and other typewritten data arrays.

The primary function is to perform reactor core calculations and provide the plant operating staff with current core performance information.

6.3.2 Component Description

The major components of the Process Computer are discussed in the paragraphs which follow.

6.3.2.1 Computer Hardware

A simplified block diagram of the process computer is shown in Figure 6.3-1. Analog voltage and current inputs representing reactor flux levels, flows, pressures, temperatures and power levels are applied to the analog input scanner. Digital (contact closure) inputs, which include various trips and alarms, control rod positions, and Rod Worth Minimizer inputs, are applied to the digital input scanner. The central processor performs the calculations required for the program being run, assigns priorities to the various programs and computer functions, and contains a core memory which, together with external memory provides for data storage. Program messages, logs, etc. are routed to the appropriate output typer or output device by the peripheral buffer, which also interfaces inputs from the input/output typer with the computer. Other program outputs from the central processor are distributed by the output controller.

6.3.2.2 Computer Software

Software is a term used to designate all the programs, subroutines, and functions used with a particular computer system. The software can be divided into the general categories shown in Figure 6.3-2.

6.3.2.2.1 Nuclear Steam Supply Programs

The Nuclear Steam Supply Programs perform the calculations and data logging required to provide current reactor core performance information. The nuclear steam supply software includes periodic programs which run automatically at specified intervals, and on-demand programs which run only when demanded from the operator control panel or called for by another program. The periodic programs calculate and update data required for the periodic, daily, and monthly core performance logs, through which operating personnel are furnished the current status of significant nuclear system operating parameters. The on-demand programs calculate and/or print out on demand a variety of nuclear system data arrays.

6.3.2.2.2 Scan, Log, and Alarm Programs

The Scan, Log and Alarm Programs perform continuous monitoring, testing, and alarming of computer input points and provide for automatic logging and trending of certain specified plant process variables. In addition, the programs provide a number of demandable console functions by which the operator can perform certain hardware and software manipulations and can demand printouts of various logs, trends, summaries, and other data arrays.

6.3.2.2.3 Balance of Plant Programs

The Balance of Plant Programs perform the calculations and data logging required to provide current balance of plant performance information. Selected balance of plant process parameters are provided on hourly, daily, and monthly logs.

6.3.2.3 Operator Control Panel

The Operator Control Panel shown in Figure 6.3-3 is the interface between the computer and the operator. Programs can be called up, variables can be monitored or trend recorded, and alarms received at the panel.

6.3.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

6.3.3.1 Computer Calculations

Several significant operations are accomplished by the computer. Total reactor core thermal power is computed. The core power distribution and thermal limits within the core are determined by neutron flux measurements and analytical procedures. Fuel isotopic composition, local power range monitor sensitivities and correction factors also are calculated, stored, and made avail-able for printout and utilization by other programs. The exposures of reactor fuel, control rods, and in-core detectors are accumulated and stored.

6.3.3.2 Computer Outputs

The computer outputs information to the plant staff through typers and CRT displays. Information resulting from the running of a program is stored in the computer's memory. It is typed out either in accordance with one of the controlling programs or upon request by an operator. The historical logs are programmed by the computer to be printed out periodically, daily and monthly.

6.3.3.3 Special Logging Functions

Among the logs printed out by the Scan, Log, and Alarm Programs are the Post Trip and Sequence of Events logs.

The balance of plant and nuclear steam supply Post Trip Logs are intended as analysis tools for use in determining the values of selected parameters immediately prior to and following a major plant trip event. The balance of plant Post Trip Log contains values for balance of plant inputs and the nuclear steam supply Post Trip Log contains selected nuclear system inputs. Both logs are automatically initiated upon occurrence of predefined plant trips, and either can also be demanded from the operator control panel.

The Sequence of Event Logs are intended to aid in determining the cause of a major plant disturbance such as a scram or turbine trip. They list the sequence in which changes of state occur among specified groups of digital inputs.

6.3.4 System Interfaces

The Process Computer interfaces with most plant systems.

6.3.5 Summary

Classification:

Power Generation System

Purpose:

To provide the plant operating staff with current plant performance data.

Components:

Computer Hardware; Nuclear Steam System Programs ; Scan, Log, and Alarm Programs; Balance of Plant Programs; Operator Control Panel.

System Interfaces:

Most plant Systems.

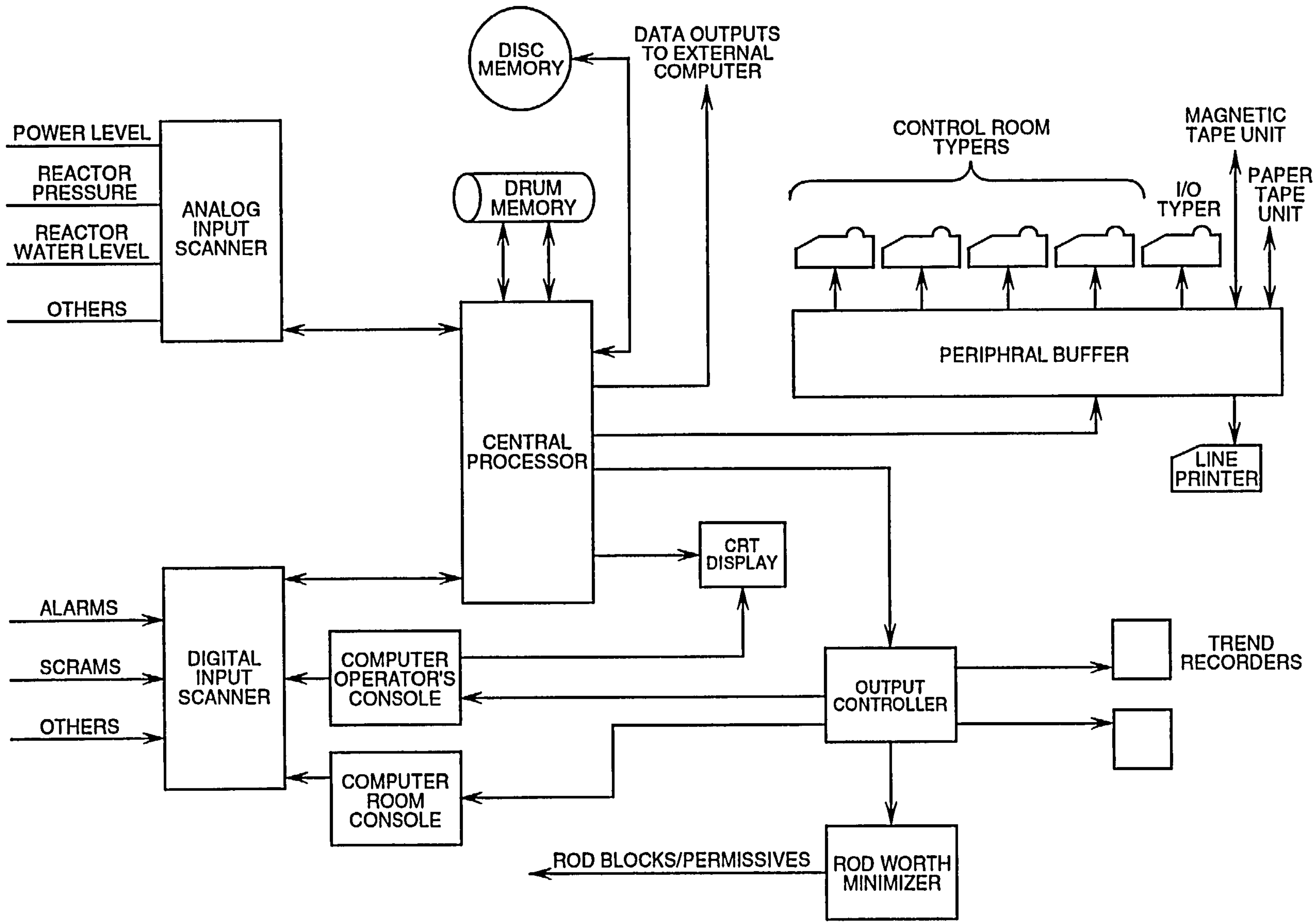


Figure 6.3-1 Process Computer Simplified Block Diagram

1297-6

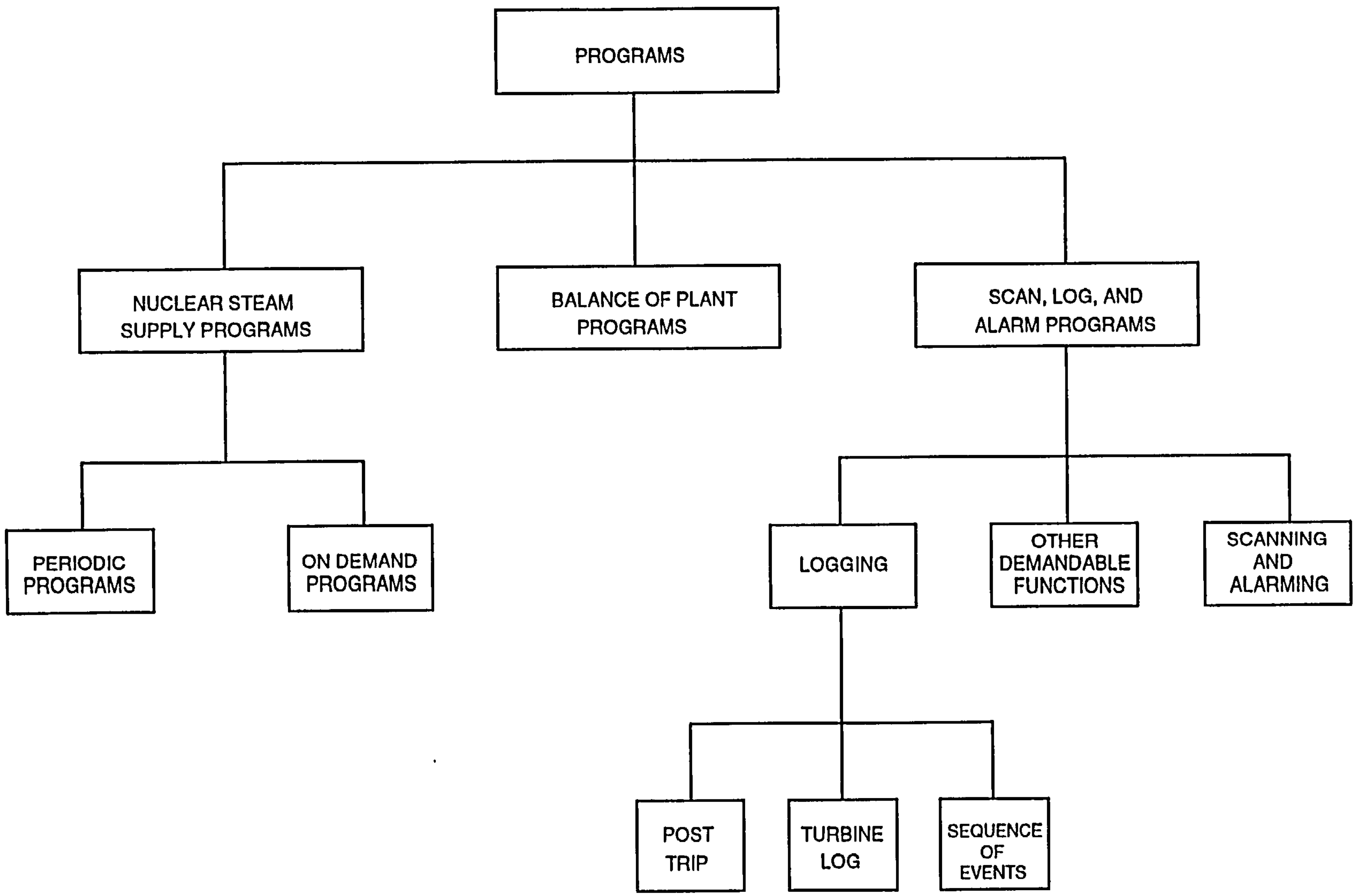


Figure 6.3-2 PROCESS COMPUTER SOFTWARE

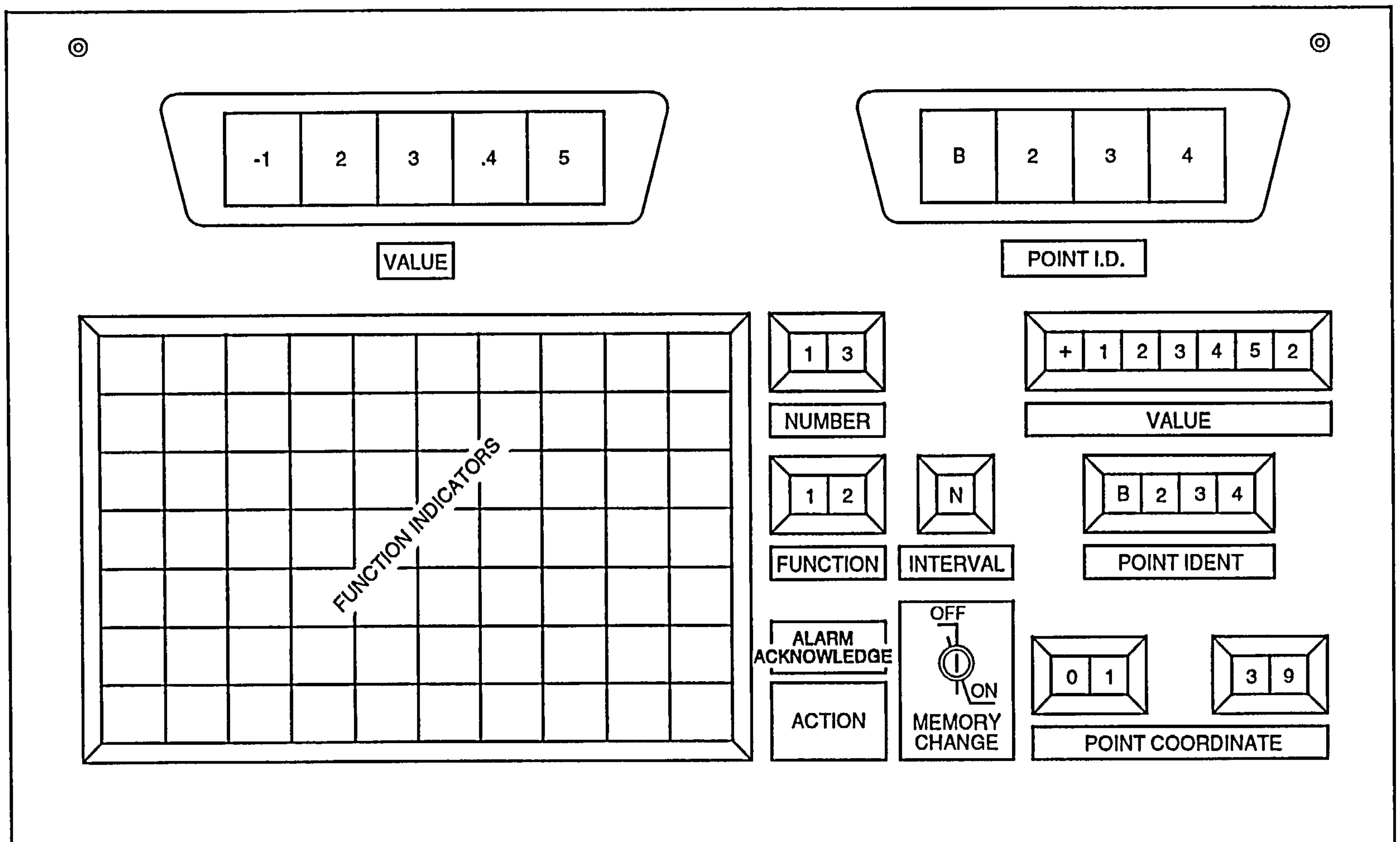


Figure 6.3-3 Operator Control Panel

1297-6