Boiling Water Reactor
GE BWR/4 Technology
Technology Manual

Chapter 4.0

Containment Systems

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4.0 CONTAINMENT SYSTEMS

The containment systems, shown in Figure 4.0-1, provide a multibarrier pressure suppression type of containment. The containment systems provide two distinct fission product barriers (the primary containment and the secondary containment) in addition to the other fission product barriers that already exist (fuel cladding and reactor coolant pressure boundary). The term "pressure suppression" comes from the fact that steam generated as a result of a loss of coolant accident is channeled to a suppression pool, where it is condensed. This suppresses the peak pressure that otherwise would be realized in the primary containment.

4.0.1 Primary Containment System (Section 4.1)

The Primary Containment System contains fission products released from a loss of coolant accident (LOCA) so that off site radiation dose limits specified in 10 CFR 100 are not exceeded; provides a heat sink for certain safety related equipment; and provides a source of water for Emergency Core Cooling Systems and the Reactor Core Isolation Cooling System.

4.0.2 Secondary Containment System (Section 4.2)

The Secondary Containment System minimizes the ground level release of radioactive material following an accident and provides the containment boundary when the primary containment is not intact.

4.0.3 Reactor Building Standby Ventilation System (Section 4.3)

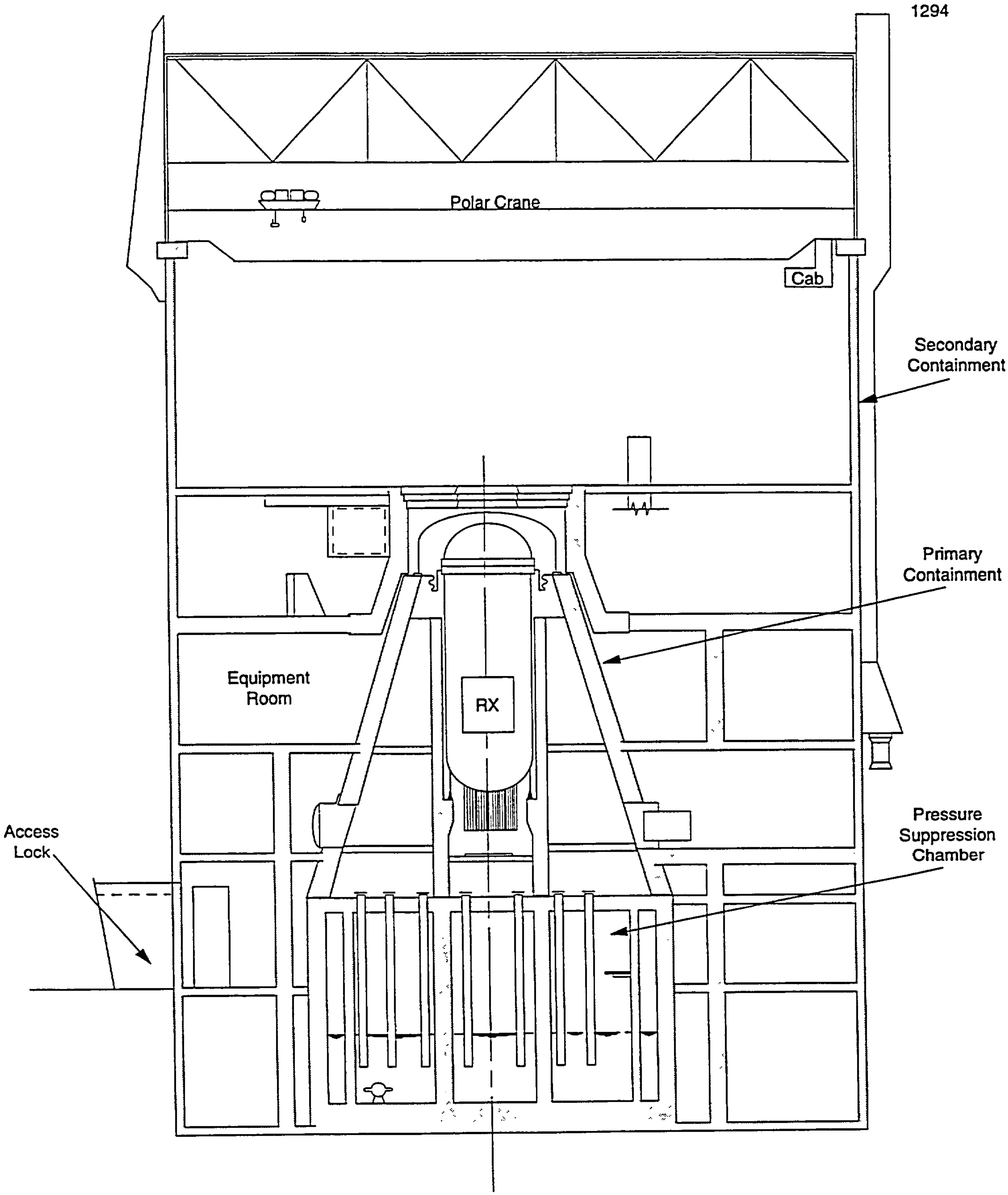
The Reactor Building Standby Ventilation System processes the secondary containment atmosphere prior to release under accident conditions; provides a means of venting the primary containment; and performs leak tests of the secondary containment.

4.0.4 Nuclear Steam Supply Shutoff System (Section 4.4)

The Nuclear Steam Supply Shutoff System (NSSSS) isolates the primary and secondary containments during accident conditions to limit the release of radioactive materials to within 10 CFR 100 limits.

4.0.5 Control Room Ventilation System (Section 4.5)

The Control Room Ventilation System ensures that control room operators can remain in the control room and take actions to operate and maintain the plant in a safe condition under all accident conditions.



Figrue 4.0-1 Mark II Containment System

Boiling Water Reactor
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Chapter 4.1

Primary Containment System

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4.1 PRIMARY CONTAINMENT SYSTEM

Lesson Objectives:

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the general arrangement and purpose of the following major system components:
   a. Drywell
   b. Vertical downcomers (vents)
   c. Suppression chamber
   d. Suppression pool
   e. Suppression chamber to drywell vacuum breakers
   f. Primary Containment Purge System
4. Explain the multibarrier, pressure suppression concept as applied to the Mark II Containment design.
5. Describe what constitutes the primary containment fission product barrier.
6. Explain the primary containment response to a major loss of coolant accident.
7. Explain how POST-LOCA hydrogen gas concentration is controlled.
8. Define primary containment integrity.
9. Describe the interfaces between this system and the following:
   a. Main Steam System
   b. Reactor Core Isolation Cooling System
   c. Secondary Containment System
   d. Reactor Building Standby Ventilation System
   e. Nuclear Steam Supply Shutoff System
   f. High Pressure Coolant Injection System
   g. Core Spray System
   h. Residual Heat Removal System

4.1.1 Introduction

The purposes of the Primary Containment System are to contain fission products released from a loss of coolant accident (LOCA) so that off site radiation dose limits specified in 10 CFR 100 are not exceeded, condense steam, provide a heat sink for certain safety related equipment, and to 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 is that of a safety related system. Its' regulatory classification is that of an engineered safety feature (ESF) system.

The Primary Containment System consists of several major components, many of which can be seen in Figure 4.1-1. These major components include the drywell, which surrounds the reactor vessel and recirculation loops; a suppression chamber, which stores a large body of water (the suppression pool); and an interconnecting vent network between the drywell and suppression chamber. Additionally, there are numerous auxiliary systems for the Primary Containment System. During plant operations, the primary containment is filled with nitrogen gas and pressurized to approximately one psig above atmospheric pressure.

In the event of a high energy process system piping failure within the drywell, reactor water and steam are released into the drywell. The resulting increase in drywell pressure forces a mixture of drywell atmosphere, steam, and water through the vents into the pool of water stored in the suppression chamber. The steam condenses in the suppression pool, resulting in a pressure reduction in the drywell. Drywell atmosphere (air or nitrogen) that is transferred to the suppression chamber pressurizes the chamber and is subsequently vented to the drywell to equalize the pressure between the two structures. Cooling systems are provided to remove heat from the drywell atmosphere and suppression pool under normal and accident conditions.

In addition to loss of coolant accident (LOCA) steam, the suppression pool also serves as a heat sink for steam discharged by the safety/relief valves, Reactor Core Isolation Cooling (RCIC) System turbine, and High Pressure Coolant Injection (HPCI) System turbine.

The suppression pool also serves as the primary source of water for the low pressure emergency core cooling systems (ECCS), and a backup source for the High Pressure Coolant Injection System and the Reactor Core Isolation Cooling System. The design bases of the primary containment are listed in Table 4.1-1 and some typical specifications are given in Table 4.1-2. During the course of an accident, containment isolation valves are automatically closed to ensure that radioactive materials, which might be released from the reactor, are kept within the primary containment boundary.

4.1.2 Component Description

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

4.1.2.1 Drywell

The purposes of the drywell are to contain the steam released from a loss of coolant accident (LOCA) and direct it to the suppression chamber, and to prevent radioactive materials from passing through its portion of the primary containment boundary.

The drywell, shown in Figure 4.1-1, is a steel lined pressure vessel shaped in the form of a truncated cone. The top head closure is made with a double tongue and a groove seal which permits periodic checks for tightness without pressurizing the entire vessel. Bolts hold the top head in position when primary containment integrity is required to be maintained. Reinforced concrete encloses the drywell, except where personnel airlocks,equipment hatches, and the drywell head are located, to provide additional shielding and resistance to deformation and buckling. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

Seal assemblies are installed between the reactor vessel and primary containment and between the primary containment and fuel pool (Figure 4.1-2). These bellows type seals form a water tight barrier which permits flooding the volume above the reactor vessel during refueling operations. Each seal assembly consists of a steel bellows assembly, backing plate, secondary seal, and a removable guard ring.

In addition to the drywell head, one 7 ft diameter double door personnel air lock and a 10 ft diameter bolted equipment hatch are provided for access to the drywell. Also, a smaller diameter personnel airlock is provided for emergency passage. The locking mechanisms on each air lock door (Figure 4.1-3) 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 one door may be operated only if the opposite door is closed and locked. Handwheels are provided inside and outside each end of the airlock which can be used to open or close either door. The door seals are designed to allow periodic testing for leakage. Access to the equipment hatch requires the removal of concrete plugs. The equipment hatch is bolted with a double seal arrangement.

To insure that primary containment integrity is maintained, process piping and electrical lines that pass through the containment wall are fitted with leak tight penetrations which are welded to the containment liner (Figure 4.1-4). Two types of process line penetrations are utilized. Hot process line penetrations are used for penetrations containing hot or variable temperature fluids that require thermal expansion capabilities. Cold process line penetrations are used for penetrations containing cold or relatively constant temperature fluids.

Primary containment isolation valves are provided on all process penetrations. The design function of those valves is to provide isolation of the containment in the event of an accident.

4.1.2.2 Suppression Chamber

The suppression chamber consists of a right circular cylinder shaped steel pressure vessel which contains a large body of water called the suppression pool. The purposes of the suppression chamber are to condense steam released from a LOCA and to prevent radioactive materials from passing through this portion of the primary containment boundary.

The purposes of the suppression pool are as follows:

- Serve as a heat sink for LOCA blow down steam.
- Serve as a heat sink for safety/relief valve discharge steam.
- Provide a source of water for the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System.
- Provide a source of water for the Core Spray (CS) System.
- Provide a source of water for the High Pressure Coolant Injection (HPCI) System.
- Provide a source of water for the Reactor Core Isolation Cooling (RCIC) System.
- Serve as a heat sink for HPCI and RCIC turbine exhaust steam.

The suppression chamber is located directly beneath the drywell. Vertical support and seismic loading is transmitted to the reinforced foundation slab of the reactor building. Design features of the suppression chamber are listed in Table 4.1-2. Access to the suppression chamber is provided through two 36 inch manways, each of which has a double gasketed bolted cover. Both manways are normally bolted shut and are opened only during plant shutdown when the suppression chamber is not required to be operational.

Providing a barrier between the drywell and the suppression chamber is the drywell floor. It is a circular reinforced concrete slab that is supported by the reactor pedestal and 14 concrete columns. Floor penetrations include 88 downcomer pipes and 11 safety/relief valve tail pipes. The floor is designed to withstand a 30 psid downward differential pressure and a 5.5 psid upward differential pressure. Two circumferential floor seals are provided between the primary containment and drywell floor. Each seal is pressurized and maintained with 60 psig nitrogen supplied from the plant nitrogen supply system.

4.1.2.3 Interconnecting Vent System

An interconnecting vent network is provided between the drywell and suppression chamber to channel the steam and water mixture from a LOCA, to below the surface of the suppression pool and allow noncondensable gases to be vented back to the drywell. 88 vent pipes (23.25" inside diameter) extend vertically downward from the upper surface of the drywell floor into the suppression chamber. The end of the vent pipes exhaust 8 ft below the suppression pool minimum water level. This method of steam condensing allows the primary containment to be designed to contain a LOCA within a relatively small volume.

Deflector plates are provided in the drywell at the entrance to each downcomer (vent) pipe to prevent a local coolant leak from overloading a single downcomer and creating a hot spot in the suppression pool.

4.1.2.4 Primary Containment Auxiliary Systems

There are numerous primary containment auxiliary systems, each with a specific function. These auxiliary systems can be seen in Figure 4.1-5 and are discussed in the paragraphs which follow.

Vacuum Relief System

During a LOCA, condensation of steam in the drywell creates a negative pressure in the drywell relative to the suppression chamber. Six pairs of vacuum breakers, installed on downcomer pipes, actuate to vent noncondensable gases from the suppression chamber to the drywell whenever suppression chamber pressure exceeds drywell pressure by 0.25 psid. This limits the upward force on the drywell floor to a maximum of 3 psid.

The suppression chamber to drywell vacuum breakers are remotely tested using air cylinder actuators. Pushbuttons, located on the vacuum breaker test panel, are used to operate the test system.

Drywell Air Cooling System

During normal plant operation there is a closed atmosphere within the drywell and suppression chamber. Since the reactor vessel is located within the drywell, heat must be continuously removed from the drywell atmosphere. Drywell average temperature is maintained less than 135°F by operating from one to eight drywell cooling units. Each cooling unit consists of a motor driven fan which forces the existing drywell atmosphere (either nitrogen gas or air) past a heat exchanger which is cooled by the Reactor Building Closed Loop Cooling Water (RBCLCW) System (Chapter 11.3). Limiting the maximum drywell atmospheric temperature ensures proper operation of motors, valves, sensors, instrument and electrical cables, and gasket materials or sealants used in containment penetrations. The Drywell Air Cooling System isolates during accident conditions to prevent compromising primary containment integrity.

Primary Containment Purge System

The purpose of the Primary Containment Purge System is to provide the means for supplying influent air to and for effluent atmosphere to be removed from the drywell and suppression chamber. The Reactor Building Normal Ventilation System (Chapter 4.2) supplies filtered and tempered fresh air to the Primary Containment Purge System for air purge and continuous ventilation which permits personnel access and occupancy during periods of reactor shutdown and refueling/maintenance operations.

The drywell/suppression chamber purge or vent exhaust air is removed by the Primary Containment Purge System and discharged to the station ventilation exhaust stack via the Reactor Building Normal Ventilation System; or to the elevated release point atop the reactor building via the Reactor Building Standby Ventilation System (Chapter 4.3). Normally, purge air is discharged at 10,000 scfm using the purge exhaust fan.

When excess radiation is detected, purge exhaust is accomplished by directing 1,000 scfm air flow through a purge filter and purge filter exhaust fan. The purge filter consists of an initial HEPA filter, a charcoal filter, and a final HEPA filter that are designed to reduce contamination and radioactive iodine in the exhaust to acceptable levels prior to release.

The purge exhaust filter and fan may also be used to vent excess pressure from the primary containment. Pressure may increase as heatup of the drywell atmosphere occurs during plant startup or by normal operation of pneumatically operated valves and/or minor instrument air/nitrogen leakage.

Containment Inerting System

The purpose of the Containment Inerting System is to create and maintain an inerted atmosphere of nitrogen gas inside the primary containment during normal power operation. Also, it supplies all inboard gas-operated valves in the primary containment thus precluding the addition of any oxygen to the containment atmosphere through valve operation or leakage. The inerting system is capable of reducing the oxygen concentration in the drywell and suppression chamber from a normal concentration of 21 percent to less than 4 percent (by volume) within 10 hours. An inerted atmosphere prevents an explosive mixture of hydrogen and oxygen from forming following a loss of coolant accident. Post LOCA hydrogen can be produced from radiolytic decomposition of water and/or the zircaloy-water reaction listed below:

$$Zr + 2H_2O \xrightarrow{\textit{high temp}} ZrO_2 + 2H_2 + \text{heat}$$

The Containment Inerting System consists of a nitrogen (N_2) purge supply and a N_2 makeup supply. The N_2 purge supply is used to create the initial inerted atmosphere in the primary containment. It consists of a 11,000 gallon liquid nitrogen storage tank, an electric vaporizer (to convert the liquid nitrogen to a gaseous state), and associated valving and piping to deliver nitrogen to the primary containment. Nitrogen gas is added to the primary containment through the purge supply at a rate of 1000 scfm while simultaneously discharging primary containment atmosphere to the Reactor Building Normal Ventilation System exhaust vent or to the Reactor Building Standby Ventilation System. The initial operation continues until the primary containment oxygen concentration is lowered to less than 4%.

After the containment atmosphere has been inerted, subsequent N_2 additions are accomplished using the nitrogen makeup supply to compensate for temperature changes or leakage and to maintain the primary containment at a slight positive pressure. The makeup supply shares the liquid nitrogen storage tank, but has its own vaporizer, valves and piping to deliver nitrogen gas at a rate of 100 scfm to the primary containment.

Containment Combustible Gas Control System

The Containment Combustible Gas Control (CCGC) System is designed to monitor and control the concentration of combustible gases in the primary containment subsequent to a LOCA with postulated high hydrogen generation rates. It has the capability for measuring the oxygen and hydrogen concentration in the primary containment, for mixing the atmospheres in the drywell and suppression chamber, and for controlling gas concentrations to less than 5 volume percent oxygen without reliance on purging of the primary containment. Subsystems of the CCGC System include the primary containment hydrogen and oxygen analyzers, containment atmosphere mixers, hydrogen and oxygen recombiners, and N_2 dilution.

Redundant hydrogen and oxygen sampling subsystems are available to measure the amounts of hydrogen and oxygen in the drywell, suppression chamber, and recombiner outlets during normal operation and following a LOCA. Representative samples are assured because of uniform mixing of the containment atmosphere by the Drywell Air Cooling System during normal plant operations and following an accident by the Containment Spray System and the fact that there are no enclosed subcompartments in the drywell or suppression chamber in which pockets of combustible gas could form.

A containment atmosphere mixing subsystem, consisting of the Containment Spray System, is initiated approximately 600 seconds after the occurrence of a postulated accident. Containment spray would be directed to the drywell and suppression chamber in an intermittent or continuous manner, to induce turbulence in those areas, thus ensuring a well mixed atmosphere. In addition to the spray systems, the spillout of steam and water through the broken pipe creates a large degree of turbulence and promotes mixing of the entrained hydrogen and oxygen with the containment atmosphere. The natural convection currents arising as a result of temperature differences between the containment atmosphere and walls will promote good mixing and prevent hydrogen and oxygen stratification.

The hydrogen and oxygen recombiner subsystem consists of two 100 percent capacity thermal recombiners that are designed to maintain hydrogen concentration below 5% following a postulated LOCA. Hydrogen is removed in a controlled manner through recombination with oxygen. Water vapor formed as a result of the reaction is directed to the suppression chamber. A flowrate of at least 60 scfm at an operating reaction chamber temperature of 1,300°F is maintained during system operation. The recombiners are located in the secondary containment (reactor building) and if needed, are aligned to the primary containment and placed in to service by a control room operator, approximately 48 hours following the accident.

As a backup to the recombiners, a nitrogen dilution subsystem is available to control the concentration of combustible gases in the primary containment. The subsystem receives N_2 from the Primary Containment N_2 Inerting System and when required, directs N_2 gas to the containment via supply valve (MOV-004) and a connection to Recombiner 1B return line to the suppression chamber. When the dilution system is adding nitrogen to the primary containment, the drywell and/or suppression chamber pressure will increase. Subsequently, when containment pressure approaches 24 psig, a controlled purge will be initiated to the secondary containment using the containment purge filter system. Venting continues until containment pressure has been reduced to atmospheric. Nitrogen is added during the venting period. Additions and releases are made at different times.

Following a LOCA, records are kept of hydrogen and oxygen concentrations and pressures in the drywell and suppression chamber. The nitrogen dilution system would be operated manually as necessary to keep the oxygen concentration <5% or the hydrogen concentration <4% in each volume.

4.1.3 System Features and Interfaces

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

4.1.3.1 Primary Containment Integrity

Primary containment integrity must be maintained at all times when the reactor is critical or moderator temperature is greater than 200°F and fuel is in the reactor vessel. Primary containment integrity means that the drywell and suppression chamber are intact and all the following conditions are met:

1. All automatic containment isolation valves are operable or deactivated in the closed position.
2. 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 may be opened to perform necessary operational activities.
3. At least one door in each personnel airlock is closed.
4. All equipment hatches are closed and sealed.
5. All blind flanges and manways are closed.
6. Primary containment leakage rates are within specified limits.
7. All containment penetration seal mechanisms are operable.

4.1.3.2 Normal Operation

Normal operation of the Primary Containment System is a condition in which primary containment integrity exists, the primary containment atmosphere has been inerted with nitrogen gas, the Containment N_2 Inerting System makeup supply is in service, the Drywell Cooling System is operating and removing heat from the drywell atmosphere, and all other influent and effluent lines to the primary containment atmosphere are isolated.

Additionally, numerous parameters are monitored to ensure proper performance of the primary containment, its supporting systems, and other plant systems. The parameters include containment pressure, temperature, radiation level, hydrogen concentration, oxygen concentration, and humidity; suppression pool water level and temperature; and drywell identified and unidentified leakage. The sensors for many of these parameters are illustrated on Figure 4.1-6.

4.1.3.3 Containment Heat Removal

There are two operational modes of the Residual Heat Removal (RHR) System (Chapter 10.4) which provide containment heat removal capability during abnormal or accident situations. The system modes are suppression pool cooling and containment spray. The former is used whenever suppression pool temperature is unusually high (an abnormal or accident situation) and the operator desires to reduce pool temperature. The latter is used when the operator desires to reduce primary containment pressure subsequent to a LOCA (accident situation only).

In the suppression pool cooling mode, the RHR System pumps suppression pool water through the RHR heat exchangers (with cooling water cut in) and then back to the suppression pool. Suppression pool temperature would slowly decrease over time. In the containment spray mode, the RHR System pumps suppression pool water through the RHR heat exchangers (with cooling water cut in) and delivers the cool water to containment spray rings mounted in the drywell and suppression chamber. The water spray condenses steam and reduces containment pressure and temperature. Chapter 10.4 provides a detailed description of the various operating modes of the RHR System.

4.1.3.4 Containment Response to a LOCA

The design basis loss of coolant accident (DBA) is a complete circumferential break (3.59 ft^2 area) of a Recirculation System 28 in. pump suction line. This accident results in worst case peak drywell pressure and temperature conditions. Table 4.1-3 illustrates the Primary Containment response to the DBA. Results are based on the assumption that the reactor and primary containment are at limiting operating conditions immediately preceding the accident. A brief explanation of the accident chronology is given in the following paragraphs.

t = 0 seconds

The postulated line break occurs and the drywell immediately pressurizes. A reactor scram is initiated by vessel low water level. An additional scram signal and containment isolation occurs when drywell pressure reaches 1.69 psig. The main steam isolation valves (MSIVs) receive closure signals from vessel low water level and main steam line high radiation and are fully closed 3.5 seconds later.

t = 0.53 seconds

The drywell pressurization is sufficient to cause the downcomers to be cleared of water and the drywell atmosphere (primarily nitrogen gas) and steam blow down through the vertical downcomers and into the suppression pool. The steam condenses in the suppression pool which suppresses the peak pressure realized in the drywell. Drywell to suppression chamber differential pressure reaches a maximum value of 22.6 psid at this time.

t = 9.26 seconds

Drywell pressure peaks at 46.0 psig (293.5°F saturation temperature), the time during which vessel blowdown changes from a liquid only to a two-phase mixture. This condition is assumed to occur when vessel level drops to the elevation of the recirculation suction line. Noncondensable gases discharged into the suppression pool during the blowdown period end up in the free air volume of the suppression chamber which results in an increase of suppression chamber pressure to approximately 34 psig. As LOCA steam is condensed in the suppression pool, drywell pressure decreases and stabilizes about 38 psig and suppression pool temperature reaches approximately 136°F.

t = 30.0 seconds

Low pressure emergency core cooling systems [Core Spray and Low Pressure Coolant Injection (LPCI) Mode of the RHR System] begin pumping water into the reactor vessel. The injected water removes decay heat and stored heat from the core and transports that heat out of the reactor vessel in the form of hot water. The hot water leaves the reactor via the broken recirculation loop, collects on the drywell floor, and then flows into the suppression chamber via the downcomer pipes. Thus, a closed loop is formed with low pressure ECCS pumping water from the suppression pool to the reactor vessel, water returns to the suppression pool from the broken loop, and the process is repeated.

t = 57.7 seconds

Drywell pressure equals reactor pressure which terminates blowdown from the reactor. Shortly thereafter, drywell pressure has decreased to the point that suppression chamber pressure exceeds it by 0.25 psid. This causes the suppression chamber-drywell vacuum breakers to open and vent noncondensable gases into the drywell which equalizes the drywell and suppression chamber pressures.

t = 179.5 seconds

The reactor vessel is reflooded to the level of the recirculation loops.

t = 600 seconds

It is assumed that the RHR System is realigned from the LPCI Mode to the Containment Spray Cooling Mode. Suppression pool water is pumped by the RHR pump, through the RHR heat exchanger, and then delivered to the containment spray headers. Suppression pool heat is rejected via the RHR heat exchangers thus causing primary containment temperature and pressure to decrease. The Containment Spray System delivers approximately five percent of its flow to the suppression chamber for cooling and steam condensation. If necessary to control primary containment pressure, the containment spray mode of the RHR System can be aligned to spray cooled suppression pool water into the drywell and/or suppression chamber atmospheres.

4.1.3.5 Suppression Pool Temperature Monitoring

Suppression pool temperature monitoring is provided to supply the control room operator with accurate indication of local water temperatures. The temperature monitoring logic consists of two divisions of 12 temperature elements each. All detectors in each division are divided equally between four quadrants of the pool. Additionally, 16 of 24 elements are located one foot below pool surface and the remaining 8 are located two feet below the surface. An accurate bulk temperature can be calculated with this type of arrangement.

The signals from all individual temperature elements are displayed on one of four recorders located on Control Room Panel 1H11*PNL-PCM. When the temperature as measured at the two foot level reaches 90°F, an alarm is annunciated in the control room. If water temperature as measured at the one foot level increases to 110°F, an additional alarm will actuate to alert the operator of a continuing problem.

4.1.3.6 System Isolation

Certain Primary Containment Auxiliary Systems valves are closed automatically as part of the Group 9 isolation logic of the Nuclear Steam Supply Shutoff System (Chapter 4.4). The automatic closure signals for that group include high drywell pressure (>1.69 psig), reactor vessel level 2 (<-38 inches), reactor building refueling floor ventilation exhaust radiation high (>35 mR/hr), or reactor building differential pressure low (< 0.35 in. H_2O below atmosphere). This group can also be initiated manually from the control room. The Group 9 valves affected by the Nuclear Steam Supply

Shutoff System (NSSSS) isolation signals are listed below.

AOV-001A,B	Drywell Inerting (4" valve)
AOV-004A,B	Suppression Chamber Inerting (4" valve)
AOV-038A,B	Purge Air to Drywell (18" valve)
AOV-038C,D	Purge Air to Suppression Chamber (18" valve)
AOV-039A,B	Purge Air from Drywell (18" valve)
AOV-039C,D	Purge Air from Suppression Chamber (18" valve)
AOV-078A,B	Vent Line - Drywell (6" valve)
AOV-079A,B	Vent Line - Suppression (6" valve)

4.1.3.7 System Interfaces

Interfaces between the Primary Containment System and other plant systems are discussed in the following paragraphs.

Main Steam System (Chapter 2.5)

The suppression pool serves as a heat sink for safety/relief valve discharge steam.

Reactor Core Isolation Cooling System (Chapter 2.7)

The suppression pool serves as a heat sink for the RCIC turbine exhaust steam and as an alternate water source for the RCIC pump.

Secondary Containment System (Chapter 4.2)

The Reactor Building HVAC System can be used to supply fresh air to the primary containment and is capable of receiving any atmosphere vented from the primary containment via the Primary Containment Purge System. Additionally, the secondary containment serves as the containment when primary containment integrity is not intact, such as during refueling.

Reactor Building Standby Ventilation System (Chapter 4.3)

The RBSVS can be aligned to vent the drywell and/or suppression chamber.

Nuclear Steam Supply Shutoff System (Chapter 4.4)

Several primary containment isolation valves are automatically closed as part of the NSSSS Group 9 isolation logic.

High Pressure Coolant Injection System (Chapter 10.1)

The suppression pool serves as a heat sink for the HPCI turbine exhaust steam and as an alternate water source for the HPCI pump.

Core Spray System (Chapter 10.3)

The suppression pool serves as the water source for the Core Spray (CS) System pumps and receives CS pumps test line return water.

Residual Heat Removal System (Chapter 10.4)

The suppression pool serves as the water source for the RHR pumps in low pressure coolant injection, suppression pool cooling, and containment spray modes. In addition, the suppression pool cooling mode can be used to reject unwanted suppression pool heat whereas the containment spray mode can be used to lower primary containment pressure and temperature following an accident. The suppression pool also receives RHR test line return water.

4.1.4 BWR Differences

Major differences exist between product lines BWR/2 through BWR/6 facilities because of the different containment designs. The discussion in this chapter is typical of product lines beginning with later BWR/4s and all BWR/5 facilities which use the Mark II Containment design.

Product lines BWR/2, BWR/3, and earlier model BWR/4 product lines use the Mark I Containment design which consists of a drywell shaped like an inverted light bulb; a suppression chamber located below the drywell that is shaped like a torus or doughnut; and an interconnecting network of eight 81" diameter vents that extend radially outward and downward from the drywell into the suppression chamber. Both Mark I and II Containments maintain a N_2 inerted atmosphere during normal operating periods.

The BWR/6 product line uses the Mark III Containment design which, comparably speaking, consists of an extremely large containment that completely surrounds the drywell and suppression pool; a drywell shaped like a right circular cylinder; a suppression pool which only a small fraction thereof is located inside the drywell; a series of horizontal interconnecting vents between the drywell and containment vessel suppression pool volumes; and a non-N_2 inerted containment atmosphere that is accessible during all modes of plant operation.

Long term control of post LOCA hydrogen gas concentration in the Mark I and II Containments is accomplished primarily through the addition of N_2 gas and by venting to the Reactor Building Standby Ventilation System. Additionally, some BWR/4s and all BWR/5 product line containments also use hydrogen recombiners. The Mark III design incorporates the use of drywell to containment venting with subsequent H_2 reduction accomplished by hydrogen igniters and/or hydrogen recombiners.

4.1.5 PRA Insights

Parts of the Primary Containment System are a major contributor to core damage frequency for the "LOCA" and the "Transients With Core Damage After Containment Failure" cut set sequences. Those scenarios involve failure of: 1) the suppression chamber structure which would result in a complete loss of ECCS equipment, and 2) failure of the downcomers and/or their vacuum breakers during a LOCA which would allow steam flow to bypass the suppression pool thus defeating the pressure suppression function of the Mark II Containment.Those events when combined with other failures could lead to core damage.

4.1.6 Summary

Classification - Safety related system; engineered safety feature system.

Purpose - To contain fission products released from a loss of coolant accident (LOCA) so that off-site radiation doses specified in 10 CFR 100 are not exceeded.

To provide a heat sink for certain safety related equipment.

To provide a source of water for emergency core cooling systems and the Reactor Core Isolation Cooling System.

Components - Drywell; Suppression Chamber; interconnecting vent system; Vacuum Relief System; Drywell Air Cooling System; Primary Containment Ventilation System;

Containment Inerting System; and Containment Combustible Gas Control System.

System Interfaces - Main Steam System; RCIC System; Secondary Containment System; RBSVS; NSSSS; HPCI System; Core Spray System; RHR System.

TABLE 4.1-3 LOCA CHRONOLOGY

EVENT	TIME (SEC.)
Downcomers Cleared	0.53
Max. Drywell Floor ΔP (22.6 psid)	0.53
MSIVs Closed	3.5
Blowdown Changes to Two Phase	9.26
Peak Drywell Pressure (46 psig)	9.26
Initiation of ECCS	30.0
End of Blowdown (DW press. = R_X press.)	57.7
Vessel Reflooded (to level of recirc. line)	179.5
Containment Spray & RHR Heat Exchanger Initiated	600

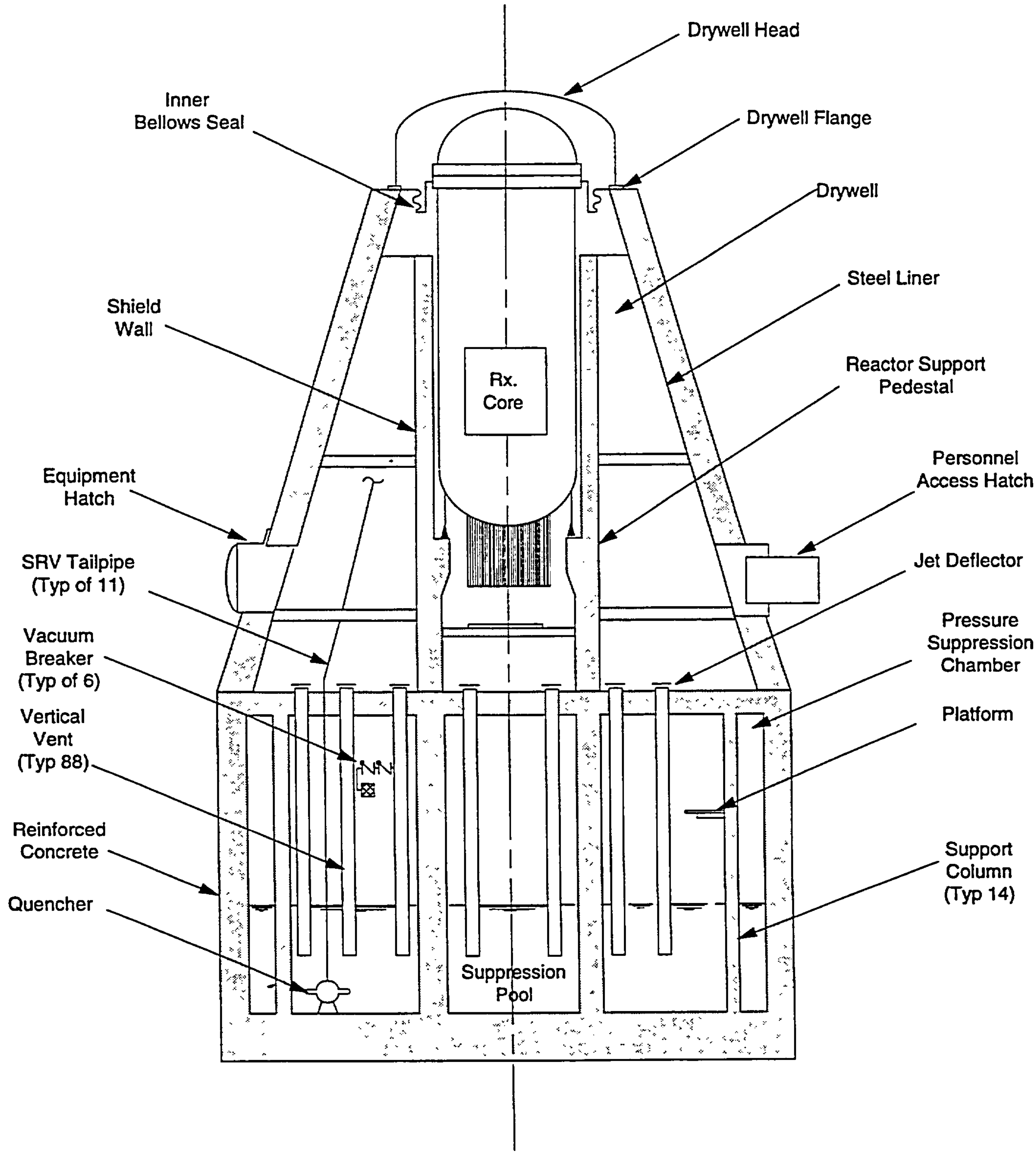


Figure 4.1-1 Primary Containment (Drywell)

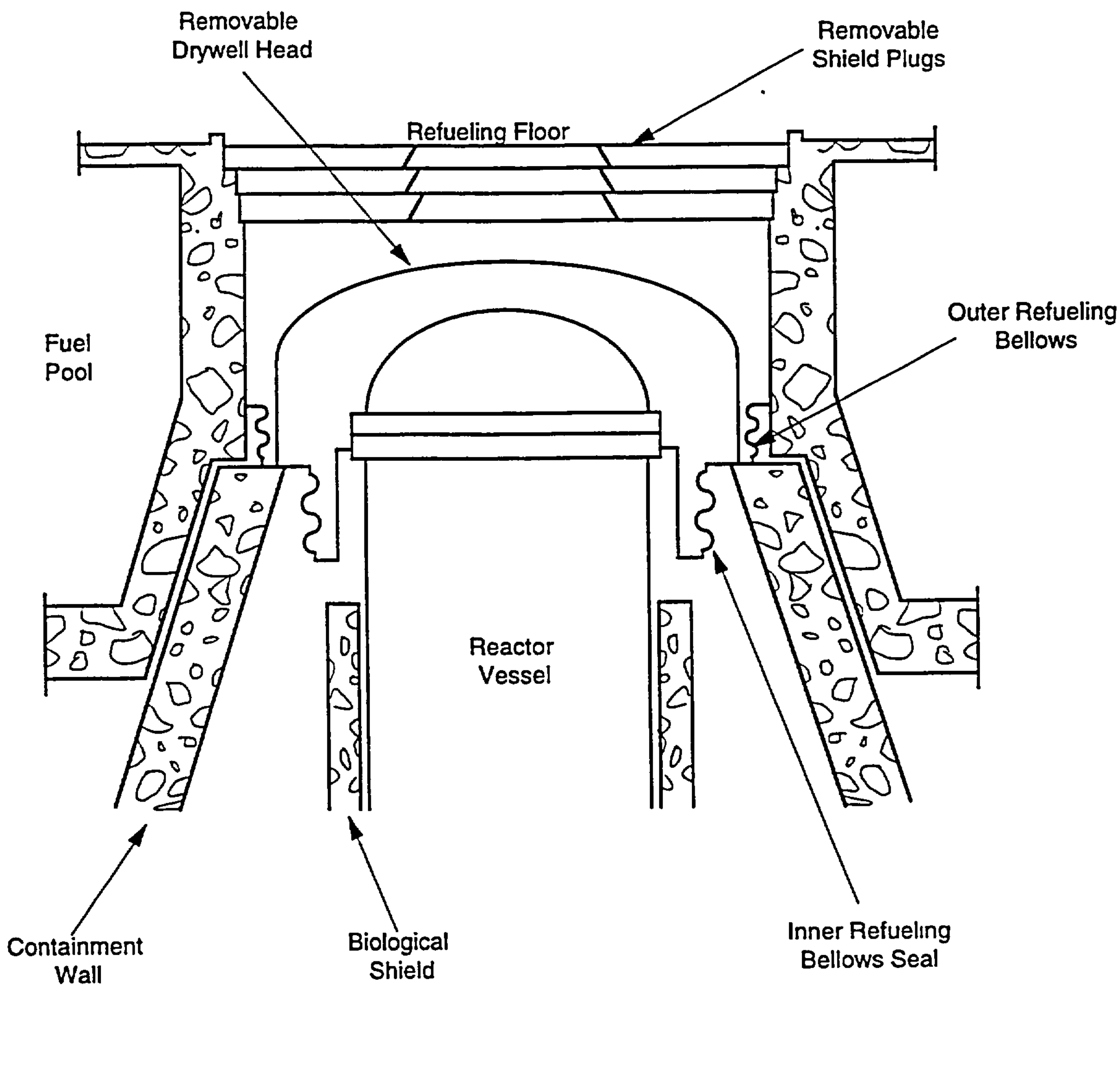
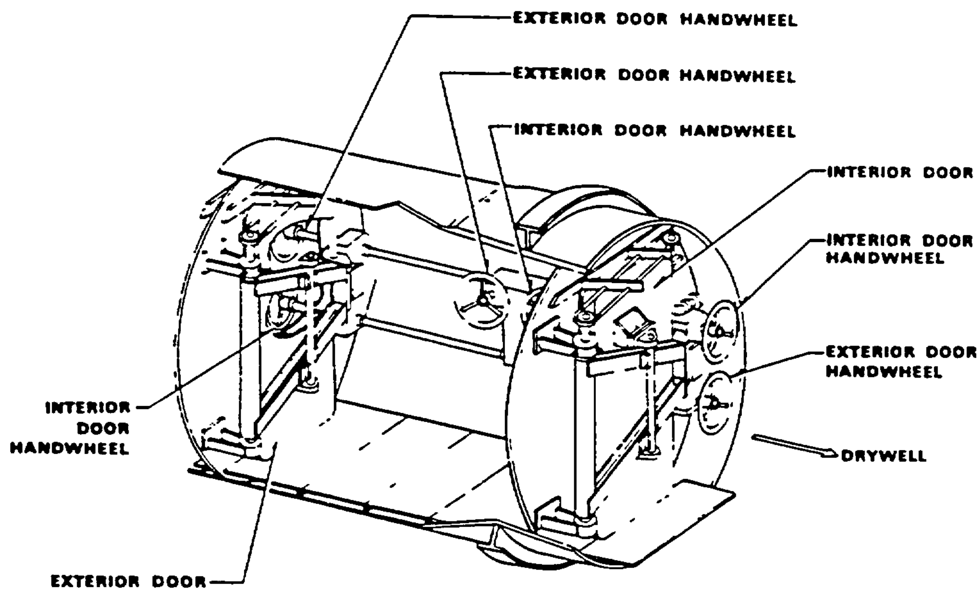
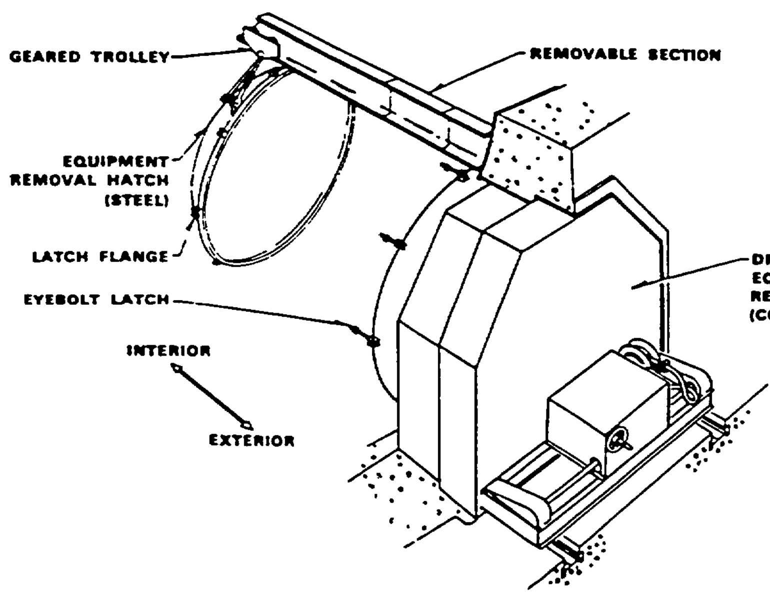


Figure 4.1-2 Drywell Head

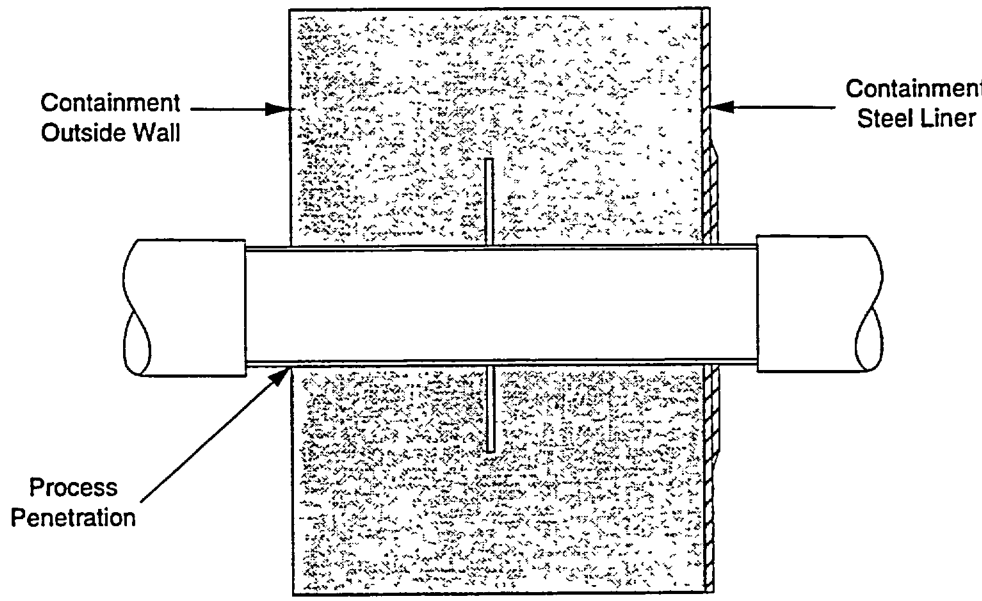


DRYWELL PERSONNEL AIR LOCK

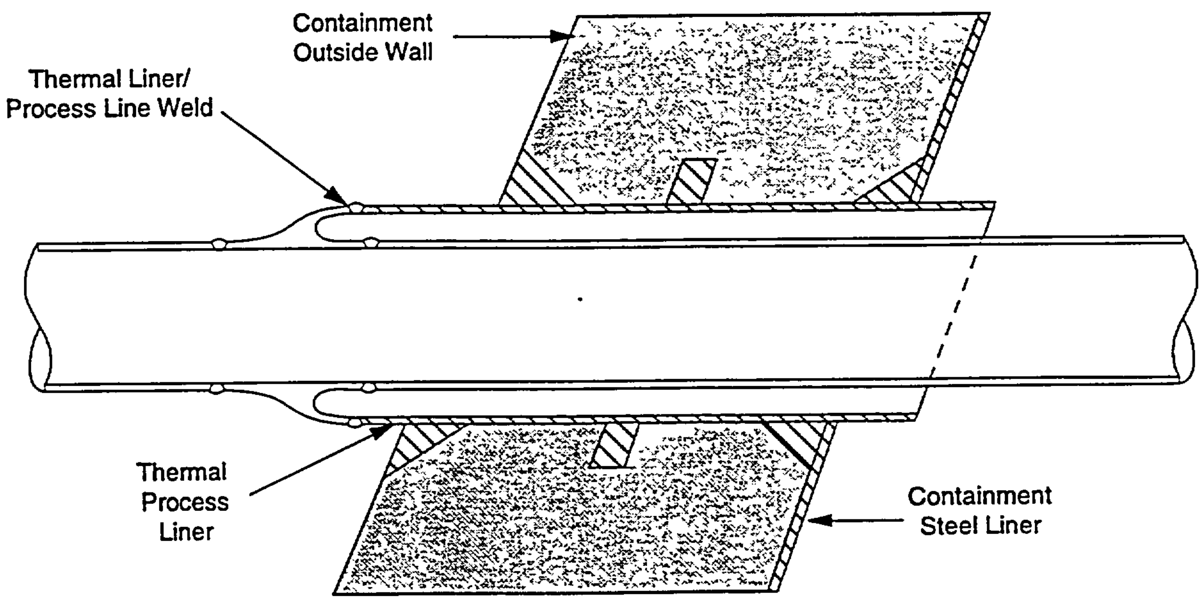


DRYWELL EQUIPMENT HATCH

Figure 4.1-3 Drywell Access Penetrations



Cold Process Line Penetration



Hot Process Line Penetration

Figure 4.1-4 Drywell Pipe Penetrations

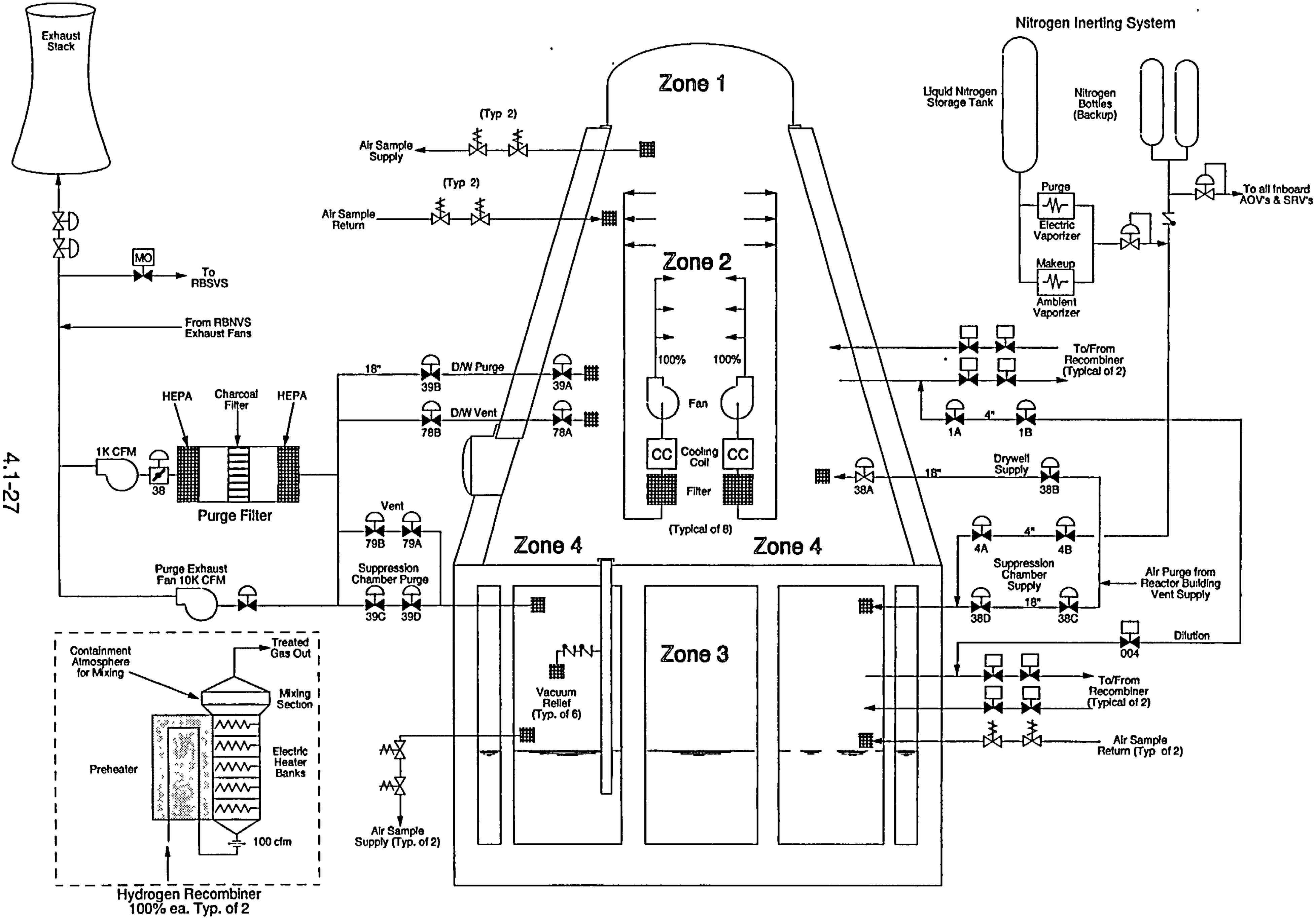


Figure 4.1-5 Primary Containment Auxiliary Systems

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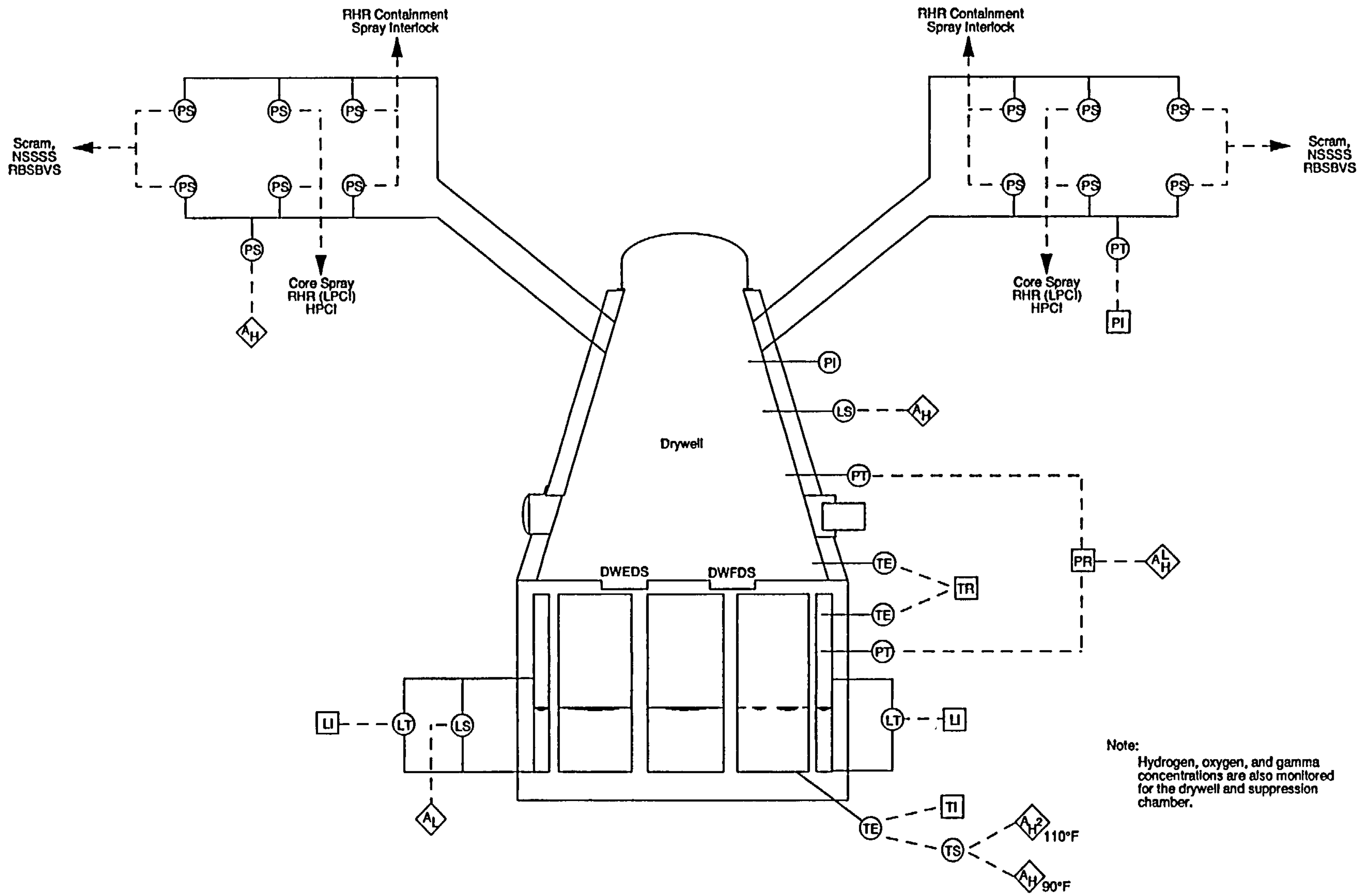

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Chapter 4.2

Secondary Containment System

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4.2 SECONDARY CONTAINMENT

Lesson Objectives:

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the general arrangement and purpose of the following major system components:
   a. Normal Ventilation System Supply and Exhaust Isolation Valves
   b. Normal Ventilation Supply Fans Discharge Dampers
   c. Area Unit Coolers
4. Describe what constitutes the secondary containment volume.
5. Define secondary containment integrity.
6. Describe the interfaces between this system and the following:
   a. Primary Containment System
   b. Reactor Building Standby Ventilation System
   c. Nuclear Steam Supply Shutoff System

4.2.1 Introduction

The purposes of the Secondary Containment System are to minimize the ground level release of radioactive material following an accident and to provide primary containment when the primary containment is open.

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. The Secondary Containment System consists of the physical boundary, the reactor building, and the Reactor Building Standby Ventilation System (RBSVS). The reactor building and its normal ventilation system is described below, whereas the RBSVS is covered in Chapter 4.3.

The reactor building encloses the entire primary containment boundary (drywell and suppression chamber) and provides a second containment barrier to fission product release. Additionally, the reactor building houses the refueling and reactor service areas, new and spent fuel storage facilities, and other reactor auxiliary and service equipment such as the Reactor Core Isolation Cooling System, Reactor Water Cleanup System, Standby Liquid Control System, Control Rod Drive System, Emergency Core Cooling Systems, Recirculation System MG sets, and supporting electrical components.

The Reactor Building Normal Ventilation System (RBNVS) supplies filtered air to and exhausts air from the secondary containment. It maintains the reactor building internal pressure at -1.5 inch water gage pressure to ensure that any leakage through the secondary containment boundary is from outside to inside. Access to the reactor building is provided by double door air locks and an equipment access hatch.

4.2.2 Component Description

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

4.2.2.1 Reactor Building

The reactor building is a cylindrical concrete structure designed to withstand tornado loads and missile hazards from the foundation mat to the top of the polar crane rail above the refueling floor. The reinforced concrete structure protects all of the safety systems in the reactor building (including the primary containment) necessary for the safe shutdown of the plant. The superstructure of the reactor building, above the polar crane rail, consists of a structural steel frame covered with metal siding and a reinforced concrete roof. The metal siding is not required for tornado protection and may be lost under this loading. In addition, relief panels (vents) are installed to prevent the buildup of excessive pressure differentials between the reactor building and surrounding atmosphere.

4.2.2.2 Relief Panels

Excessive reactor building pressure differentials caused by steam line ruptures and tornadoes are prevented by venting the secondary containment to atmosphere. The vents consist of the turbine building blowout panels, main steam tunnel relief vents, and the reactor building exterior metal siding. Steam ruptures occurring in the main steam pipe tunnel are relieved to the turbine building through the main steam tunnel relief vent. The interconnecting network of floor gratings in the reactor building enable pressurization effects of main steam and other high energy pipe ruptures to migrate to the refueling floor. The exterior siding panels vent the refueling floor to the outside atmosphere.

4.2.2.3 Air Locks and Penetrations

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

4.2.2.4 Reactor Building Normal Ventilation System

The reactor building is heated, cooled, and ventilated during normal operating and shutdown conditions by the Reactor Building Normal Ventilation System (RBNVS). Also, it maintains the secondary containment at a negative pressure to prevent unmonitored outleakage of air to the environment. During periods when access is required to Primary Containment, the RBNVS replaces the Primary Containment atmosphere with fresh air prior to personnel entry and then continues to provide fresh air during maintenance via the Primary Containment Purge System.

The Reactor Building Normal Ventilation System is automatically shutdown and isolated whenever the secondary containment is isolated and connected to the Reactor Building Standby Ventilation System. Even though the RBNVS is not an engineered safety feature, certain components do perform engineered safety feature functions. Those components include system supply and exhaust isolation valves, the Reactor Building Exhaust Fans, and the safety related equipment area unit coolers.

The Reactor Building Normal Ventilation System (Figure 4.2.2) consists of heating and air conditioning supply units with two supply fans, three exhaust fans, distribution ducts, manually positioned ventilation louvers, differential pressure controlled dampers, unit coolers, controls and instrumentation. Outdoor air is drawn from a louvered intake in the reactor building south wall through two air operated isolation valves (AOV-35A/B) and supplied to the suction of both supply fans. One of the two 100% capacity fans is normally operating and supplying 89,000 scfm to the reactor building. This flow rate equates to approximately 2.5 air changes per hour. Fresh air then passes from the supply fan, through its discharge pressure control damper (PCD), and on to a glycol heating coil which operates as necessary to maintain supply air temperature at 60°F. The supply fan discharge PCD of the non-running fan remains closed in order to prevent reverse flow. Should the outside temperature drop to 35°F, supply valves AOV-35A/B will close to prevent downstream coolers from freeze up. This will also result in an automatic initiation of the Reactor Building Standby Ventilation System and simultaneous isolation of the RBNVS.

Air leaving the heating coil is filtered and then distributed via three ducts to all levels of the Reactor Building (Secondary Containment). One duct supplies air to the Primary Containment when primary containment purge is in progress, a second duct directs air to the refueling level, and the third duct supplies air to all other levels. Air supplied to the refuel and reactor building areas passes through chilled water coolers and individual area heaters prior to being distributed to the various levels.

Exhaust air is drawn from each floor level, the potentially contaminated areas, the steam pipe tunnel, and the refuel floor into the reactor building exhaust plenum. [It should be noted that the refuel floor exhaust duct is designed so that the time required for airborne radiation to travel from the duct inlet to exhaust isolation valves AOV-40A/B (10 seconds) is greater than the time lag from refuel area ventilation exhaust radiation monitor actuation to isolation valve closure.]

Three 50% capacity Reactor Building Exhaust Fans (45,000 scfm each) draw air from the exhaust plenum and discharge through RBNVS discharge isolation valves AOV-37A/B, to the Station Ventilation Booster Exhaust Fans and the Station Ventilation Exhaust Stack, to atmosphere. Two of the three exhaust fans are usually in operation providing 90,000 scfm exhaust flow. Normal ventilation air exhaust is not filtered prior to release but is continuously monitored for radioactivity. High activity will isolate the secondary containment and start the Reactor Building Standby Ventilation System.

Air supplied to the refueling zone is distributed to one side of the refueling room and then flows directionally from the area of least towards the area of most potentially contaminated sections of the room. It is collected around the periphery of the fuel storage pool (including the dryer and separator pool and refueling well when primary containment is open) and other areas of high potential for contamination.

The flow path of fresh air through each room of a specific floor level of the reactor building begins at the outlet of the air supply ducts to that floor. Air is swept through open areas and rooms and is then collected for exhaust by the ventilation exhaust ducts located within those areas/rooms. Rooms below grade level are also ventilated by air moving down open stairwells and equipment floor grating to each of the corner rooms and then collected for exhaust by the room(s) exhaust ducts.

The RBNVS maintains a negative pressure of 1.5 inch H_2O within the Secondary Containment boundary during normal operating conditions. Subatmospheric pressure ensures that containment leakage is into rather that out of the containment boundary. This minimizes the possibility of an uncontrolled and unmonitored release of radioactive contaminants to the environment. Negative pressure is developed by automatic positioning of the reactor building supply fans discharge pressure control dampers (AOV-PCD-11A/B) so that reactor building ventilation exhaust flow is greater than supply flow. The parameter used to modulate the dampers is a differential pressure signal that is developed by comparing reactor building internal pressure with outdoor atmospheric pressure.

4.2.2.5 Area Unit Coolers

Area unit coolers are installed in various areas of the reactor building to supplement the RBNVS. They consist of a filter, cooler and fan. During a loss of the Normal AC Power System or a loss of coolant accident (LOCA) conditions, the normal ventilation system shuts down and isolates and the Reactor Building Standby Ventilation System is initiated. Certain equipment room unit coolers are supplied power from the plant Emergency AC Power System. This ensures that safety related equipment, needed to place the plant in a safe shutdown condition or to provide adequate core reflood and cooling following a loss of coolant accident (LOCA), is sufficiently ventilated and cooled so as to enable them to continuously perform their design functions. When the Reactor Building Standby Ventilation System (RBSVS) starts, RHR/Core Spray Area and Refueling Area unit coolers operation is initiated. Additional units, such as those located in vital Motor Control Center (MCC) areas and MG Set Rooms, are started and stopped automatically as controlled by area temperature sensor switches.

Cooling water is supplied to the area unit coolers by the Reactor Building Standby Ventilation System/Control Room Air Conditioning Chilled Water System which is also available during accident or loss of normal power situations. Supplemental cooling is provided in the Steam Pipe Tunnel, Reactor Water Cleanup (RWCU) Pump Rooms, and the RWCU Heat Exchanger area by unit coolers located within those rooms.

All areas are cooled or heated as necessary to maintain maximum exhaust temperatures of 130°F in the main steam pipe tunnel and 110°F on the refueling level. Temperature is controlled at approximately 104°F in the remainder of the secondary containment.

4.2.2.6 Primary Containment Purge System

The Primary Containment Purge System (PCPS) consists of connections to the RBNVS supply and exhaust ducting and a dedicated filter train unit for processing primary containment effluent atmosphere prior to release. The drywell and suppression chamber is supplied with fresh air from the purge system whereas exhaust air is processed by the PCPS filter train, directed to the RBNVS exhaust ducting, and discharged through the station exhaust stack.

The PCPS filter train consists of a high efficiency particulate absolute (HEPA) filter, a charcoal adsorber, and a fan. The HEPA filter can remove radioactive particulate matter of 0.3 micron size and larger with an efficiency of 99.95%. The charcoal adsorber is an activated charcoal bed. The adsorber unit is a standard tray type capable of removing >99.95% of radioactive iodine in the form of elemental iodine and 85% of radioactive iodine in the form of methyl iodide with the flow at 90% relative humidity. The fan at the end of the train provides system flow and delivers 1000 scfm at 8.5 inches water gage pressure.

If the Primary Containment Purge System is in service and the Reactor Building Standby Ventilation System is initiated, the purge system will automatically isolate and its running fan will stop.

4.2.3 System Features and Interfaces

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

4.2.3.1 Secondary Containment Integrity

Secondary containment integrity must be maintained at all times that primary containment integrity is required (the reactor is critical or moderator temperature is >212°F and fuel is in the reactor vessel), when fuel handling operations are in progress within the secondary containment, or when activities are being performed that have a potential for draining the reactor vessel.

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

- The Reactor Building Normal Ventilation System automatic isolation valves are operable or deactivated in the closed position.
- The Reactor Building Standby Ventilation System is operable.
- Secondary containment pressure is less than or equal to -1.5 inch H_2O gauge.
- Secondary containment leakage rates are within specified limits.
- All containment penetration seal mechanisms are operable.
- At least one door in each access opening is closed.
- All equipment hatches are closed and sealed.

4.2.3.2 Normal Operation

Normal operation of the Secondary Containment System is a condition in which secondary containment integrity exists. The Reactor Building Normal Ventilation System is in service maintaining a slight vacuum (-1.5" w.g. pressure) throughout the secondary containment volume. Reactor building ventilation exhaust is monitored for radioactivity but is not filtered.

4.2.3.3 Infrequent Operation

The primary containment is inerted with nitrogen gas during normal operation. When the plant is shutdown for an outage, it is necessary to deinert the primary containment to allow for personnel entry. This is accomplished by supplying fresh air to the primary containment from the RBNVS and using the Primary Containment Purge System (PCPS) to exhaust primary containment atmosphere (nitrogen gas and air) until the oxygen concentration >19.5%. The PCPS receives, filters (when unacceptable radiation levels exist), and then discharges primary containment effluent to the RBNVS exhaust ducting.

4.2.3.4 System Isolation

Certain Reactor Building Normal Ventilation System valves are closed automatically as part of the isolation logic of the Nuclear Steam Supply Shutoff System (NSSSS), Radiation Monitoring Instrumentation, and Reactor Building Standby Ventilation System. The isolation signals provided by those systems are listed below:

1. Drywell pressure high (1.69 psig)
2. Reactor vessel level 2 (-38 inches)
3. Refuel area exhaust radiation high (35 mR/hr)
4. Vent/purge exhaust duct rad high(>5.4 x 10^5 cpm)
5. Reactor Building ΔP low(<-0.30 in. H_2O)
6. RBSVS initiation signal multiple inputs

4.2.3.5 Secondary Containment Accident Response

If a LOCA occurs, as indicated by drywell pressure high and reactor water level (Level #2), or a fuel handling accident happens resulting in a refueling level exhaust duct radiation high (>35 mr/hr), then automatically the Reactor Building Normal Ventilation System isolates, e.g., the reactor building supply and exhaust valves (AOV-35A&B and AOV-37A&B) close. Closure of those valves will cause the Reactor Building Standby Ventilation System (RBSVS) to start, if not already running.

Initiation of the RBSVS results in additional ventilation system actions including trip of the running reactor building supply fan, refueling area exhaust dampers close, contaminated area exhaust dampers close, primary containment purge supply and exhaust valves close if open, the running reactor building exhaust fans continue operating to provide building recirculation air flow, and the area unit coolers start and/or cycle as described in paragraph 4.2.2.5 above. These system realignments isolate the secondary containment and places RBSVS in operation which provides for preservation of a slight negative pressure in the secondary containment, internal recirculation and cooling of the reactor building atmosphere, and a controlled, filtered, and elevated release of the building atmosphere to the environment.

4.2.3.6 System Interfaces

Interfaces between the Secondary Containment System and other plant systems are discussed in the following paragraphs.

Primary Containment System (Chapter 4.1)

The Reactor Building Normal Ventilation System can supply fresh air to and receive atmosphere vented from the primary containment via the Primary Containment Purge System. Additionally, the secondary containment serves as the primary containment when primary containment integrity is not intact, such as during refueling.

Reactor Building Standby Ventilation System (Chapter 4.3)

The Reactor Building Standby Ventilation System starts automatically during accident conditions to maintain the secondary containment volume at a slight negative pressure and provide a controlled, filtered, and elevated release of reactor building atmosphere.

Nuclear Steam Supply Shutoff System (Chapter 4.4)

Several Reactor Building Normal Ventilation System isolation valves are automatically closed by the Nuclear Steam Supply Shutoff System (NSSSS).

4.2.4 BWR Differences

Major differences exist between product lines BWR/2 through BWR/6 facilities because of the different containment designs. The discussion in this chapter is typical of product lines BWR/2 through BWR/5 facilities which use the Mark I or Mark II Containment designs. The secondary containment (reactor building) consists of a single building that is either rectangular or cylindrical in shape. It supplies all of the functions provided by the multiple buildings concept described below. The Reactor Building Standby Ventilation System discussed in this and the following chapter is called the Standby Gas Treatment System at most BWRs.

The BWR/6 product line uses the Mark III Containment design. The secondary containment consists of the Shield Building, Fuel Building and portions of the Auxiliary Building. The Shield Building encloses and protects the primary containment and provides biological shielding for plant personnel and the public. The Auxiliary Building houses the emergency core cooling systems, which would contain reactor coolant following an accident. The Fuel Building houses equipment and facilities concerned with fuel handling and storage operations. In addition, the Fuel Building provides an important barrier for the potential release of radioactive material associated with a postulated fuel handling accident.

4.2.5 PRA Insights

The Secondary Containment System is a major contributor to core damage frequency for the "Internal Flooding" cut set sequence. Flooding the lowest elevation of the Secondary Containment would disable the Low Pressure ECCS, RCIC, and HPCI. Those equipment losses would require a manual shutdown of the reactor. Additional losses of equipment could then lead to core damage. Event analyses indicates flooding caused by leaks or broken pipes from the Suppression Pool, Condensate Storage Tank and System, Reactor Coolant System, Service Water System, or the Fire Protection System. This sequence contributes to 7% of the core damage frequency at Shoreham.

4.2-6 Summary

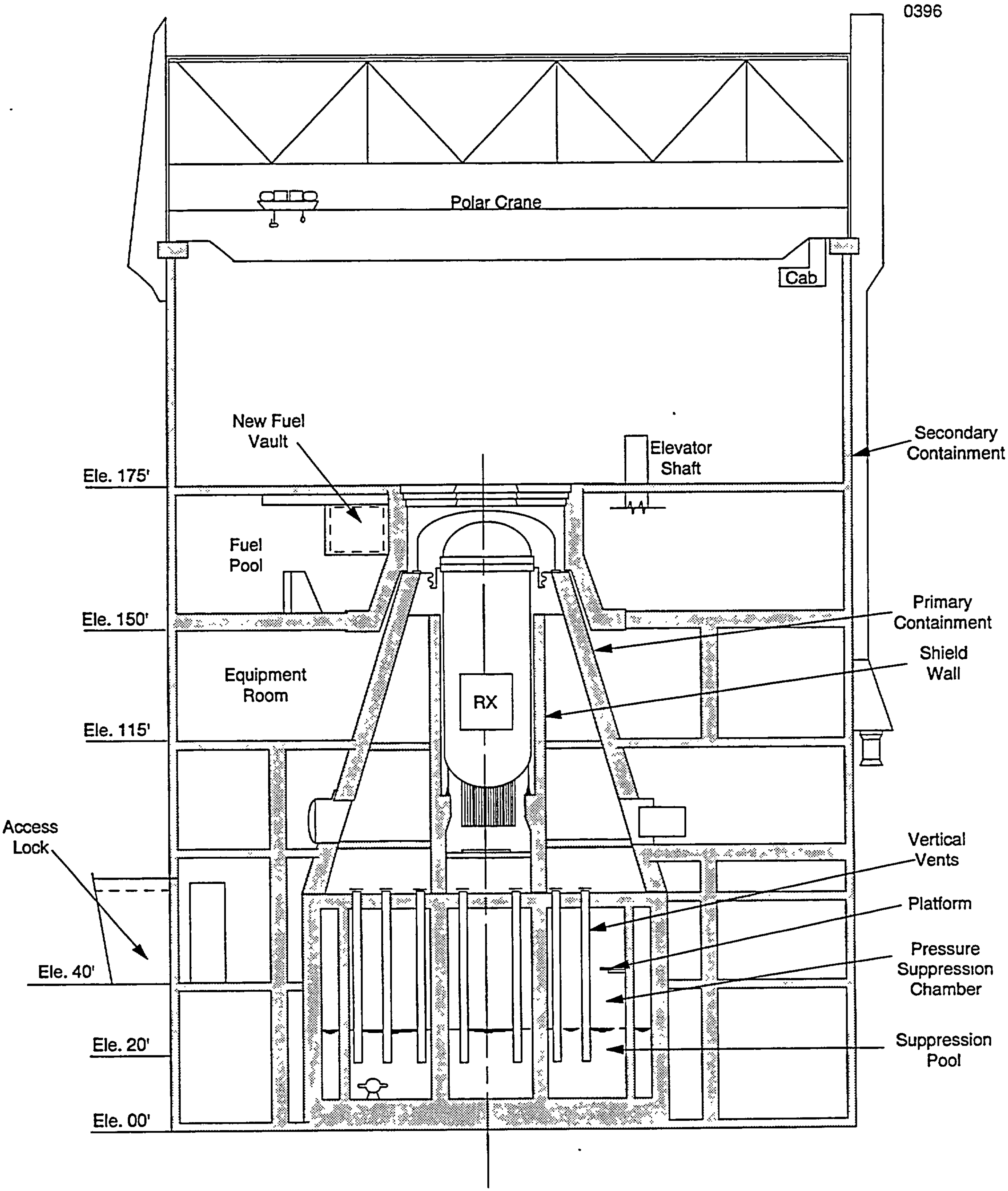
Classification - Safety related system; engineered safety feature system.

Purpose - To minimize the ground level release of radioactive material following an accident and to serve as the containment when the primary containment is not intact.
Components - Reactor building, relief panels; Reactor Building Normal Ventilation System; area unit coolers.

System Interfaces - Primary Containment System; Reactor Building Standby Ventilation System; NSSSS.

TABLE 4.2-1 SECONDARY CONTAINMENT DESIGN SUMMARY

Free Air Volume	2,000,000 ft^3
Design Inleakage (at -0.5 in. H_2O)	50 % of Volume/day (≈ 700 cfm)
Containment Pressure [accident conditions - Double-ended rupture (DER) of a recirculation loop or a refueling accident]	The Reactor Building Standby Ventilation System is designed to maintain the reactor building subatmospheric relative to the outside environment



Figrue 4.2-1 Mark II Containment

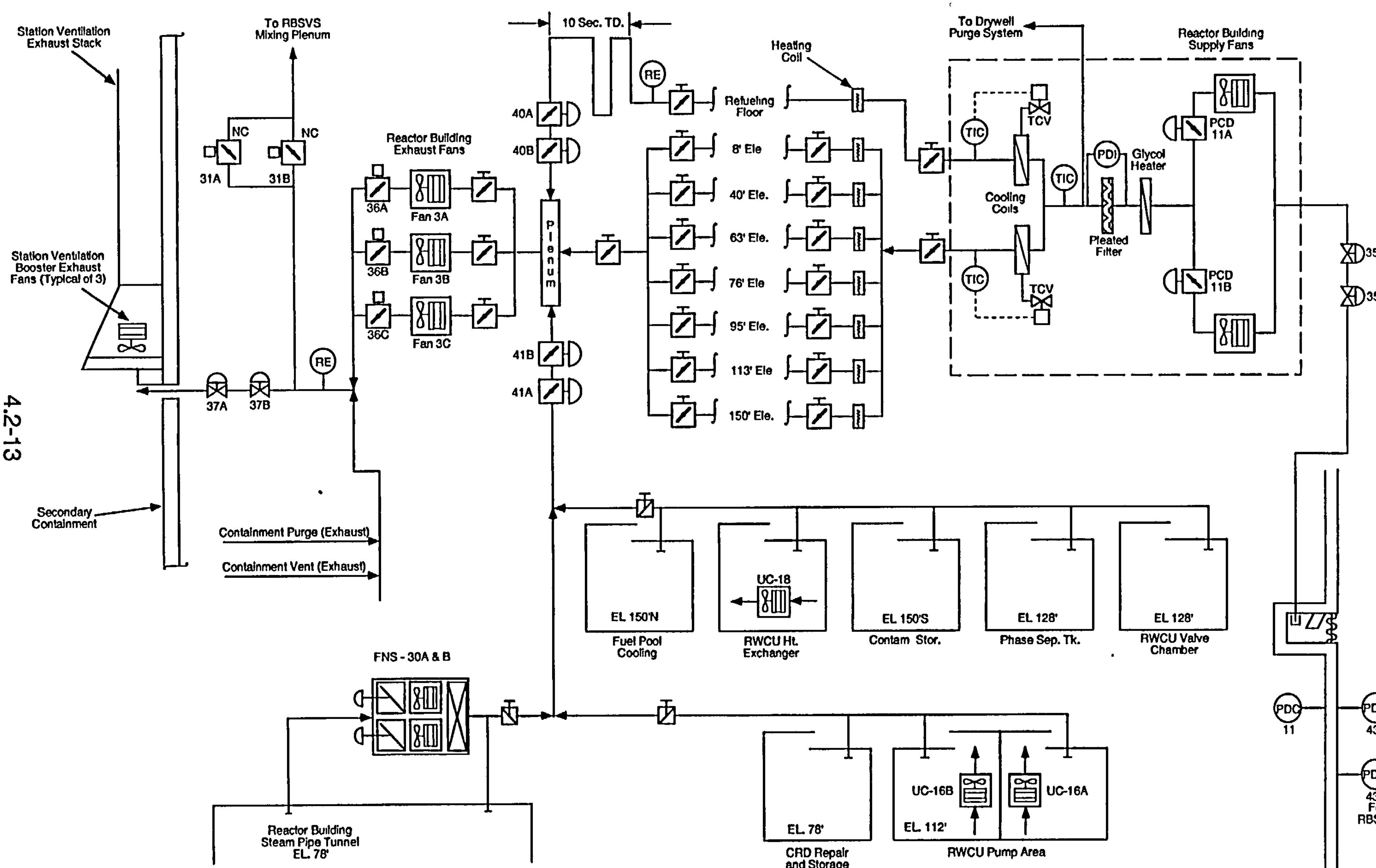


Figure 4.2.2 Reactor Building Normal Ventilation System

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Learning Objectives:

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. Explain the general arrangement and purpose of the following major system components:
   a. Reactor Bldg. Exhaust Fans
   b. Mixing plenum
   c. Cooling coils
   d. Filter Train Booster Fans
   e. Electric heater
   f. HEPA filters
   g. Charcoal filter
   h. Flow control dampers
4. State the system initiation signals and reason for each.
5. Describe the interfaces between this system and the following:
   a. Primary Containment System
   b. Secondary Containment System
   c. Process & Area Radiation Monitoring System
   d. High Pressure Coolant Injection System

4.3.1 Introduction

The purposes of the Reactor Building Standby Ventilation System are to process exhaust air from the secondary containment boundary under accident conditions in order to limit radiation dose rates to less than 10 CFR 100 guidelines, to maintain a negative pressure in the secondary containment upon loss of the normal ventilation system, to perform secondary containment leak tests, and to purge the primary containment. The functional classification of the Reactor Building Standby Ventilation System is that of a safety related system. Its regulatory classification is that of an engineered safety feature (ESF) system.

The Reactor Building Standby Ventilation System (Figure 4.3-1) operates independent of the Reactor Building Normal Ventilation System (RBNVS) described in Chapter 4.2. It consists of a suction duct network, two filter trains with fans, and a discharge vent. The suction duct removes air from the RBNVS exhaust duct, upstream of the secondary containment air exhaust isolation valves. It also draws air from the High Pressure Coolant Injection (HPCI) System gland exhaust line when the HPCI is running. Each filter train contains a heater to provide humidity control, and banks of particulate and charcoal filters to remove particulate matter and halogens. Both trains share a pair of booster fans which move containment air through the processing system and out the elevated release point for discharge. The fans and filter trains are located inside the Reactor Building.

Each Reactor Building Standby Ventilation System (RBSVS) train must be operable whenever reactor coolant temperature is >200°F, during a refueling outage, when handling irradiated fuel, during core alterations, or when operations are in progress having a potential to drain the reactor vessel. Either train is rated for 100% capacity and is capable of producing and maintaining the secondary containment at a pressure of -0.5 inch H_2O.

4.3.2 Component Description

The major components of the Reactor Building Standby Ventilation System are discussed in the paragraphs which follow.

4.3.2.1 Reactor Building Exhaust Fans

The three Reactor Building Exhaust Fans are 45,000 cfm, motor driven vane axial fans which are powered from 480Vac Emergency Busses. They normally exhaust the Secondary Containment to the Station Ventilation Exhaust Stack.

During the RBSVS mode, at least one exhaust fan remains running to draw air from the refueling level downward and through equipment hatch openings and deck grating, to the ventilation exhaust duct inlets at each floor level. Exhaust fan discharge is redirected from the stack to a mixing plenum.

4.3.2.2 Mixing Plenum

The Mixing Plenum acts as the collection point for all containment air that is picked up by the Reactor Building Exhaust Fans from the exhaust ducts located on all levels of the Reactor Building. Within the plenum, all air is mixed by means of internal baffles. The majority of air leaves the mixing plenum and is directed to a ring header located above the refueling floor, where it is discharged through diffusers. Some air is removed from the mixing plenum by a Filter Train Booster Fan and discharged to one of two 100% capacity filter trains.

4.3.2.3 Cooling Coils

Two 100% capacity cooling coils, located in series, cool the large volume of air in the Secondary Containment. Most of containment air entering the mixing plenum is directed through cooling coils and then to distribution ducts located on the refueling level. Each cooling coil is cooled by the Reactor Building Standby Ventilation System/Control Room Air Conditioning (RBSVS/CRAC) Chilled Water System. The combination of cooling coils and the area unit coolers (para. 4.3.2.11) help to maintain the reactor building at a negative pressure by cooling the containment atmosphere.

4.3.2.4 Filter Train Booster Fans

Two Filter Train Booster Fans are 1235 cfm, motor driven centrifugal fans that are powered from 480Vac Emergency Busses. Air is drawn from the mixing plenum and discharged through a common header to the filter trains. Only one fan and filter is required for RBSVS operation.

Flow divides at the inlet of each filter train. Air at 1190 scfm enters the filter train for processing while 45 scfm is discharged to the secondary containment atmosphere via a fixed orfice.

4.3.2.5 Electric Heater

The air stream entering the filtration system is heated by an electric heater that is regulated by a temperature controller to maintain prefilter influent flow at < 70 percent relative humidity. Charcoal adsorption efficiency is reduced if the relative humidity exceeds 70 percent. Each heater is powered from the same Emergency AC power source as its associated Filter Train Booster Fan.

4.3.2.6 Prefilter

The prefilter is located downstream of the electric heating coil. It is constructed of a replaceable, dry type, extended fiberglass media, The prefilter is designed to remove atmospheric dust and particulate matter with at least an 85 percent efficiency, thereby extending the life of the downstream high efficiency particulate air filters.

4.3.2.7 HEPA Filter

Immediately following the prefilter and upstream of the charcoal filter is the high efficiency particulate air (HEPA) filter bank. The HEPA filter is 99.97 percent efficient at removing particles 0.3 microns or larger in diameter. It removes particulate which could enter the filter train from the mixing plenum thus ensuring charcoal filter adsorber efficiency and protection from fouling. The HEPA filter bank is constructed of fiberglass media.

4.3.2.8 Charcoal Filter

Located downstream of the HEPA filter bank is the Charcoal Filter. The charcoal adsorber is impregnated with iodine and has a minimum design adsorption efficiency of 99.9 percent removal of elemental iodines [5 percent of which is in the form of methyl iodide (CH_3 I)] when operating at 90 percent humidity and 77°F.

The adsorption phenomena occurring in the charcoal adsorber involves adsorption of a gas phase (iodine and methyl iodine) on a solid phase (carbon). The adsorption of a gas on a solid surface may be classified in one of two categories: physical adsorption and chemisorption.

In physical adsorption, the gas molecules (adsorbate) adhere to the solid surface for a period of time because of intermolecular forces known as Van der Waal forces. The energy level of the adsorbate determines the molecular interactions with both the solid and gas phase. When the energy level of the adsorbate becomes sufficiently high, the molecule overcomes the Van der Waal forces and again enters the gas phase.

A given gas molecule becomes adsorbed and desorbed many times before leaving the charcoal filter.

In contrast to physical adsorption, chemisorption involves a chemical change in the adsorbate. Chemisorption may be accomplished by a sharing of electrons between the charcoal (adsorbent) and adsorbate, or the formation of ions from the adsorbate with subsequent chemical reactions. The adsorbate in a chemisorption process is relatively immobile because the chemical bond that is established is generally much stronger than Van der Waal forces.

Untreated carbon is effective at removing radioactive elemental iodine from air. However, a significant amount of iodine may be in the form of an organic compound methyl iodide (CH_3 I). Organic iodide is produced when elemental iodine reacts with organic compounds (i.e. lubricants). Physical adsorption of methyl iodide is not long enough to permit substantial decay of radioactive iodine. Fortunately, by impregnating activated carbon with stable iodine, it becomes very effective for removing methyl iodide. Radioactive I^{131} ions from methyl iodide molecules, are removed through a combination of adsorption and ion exchange as described in the equation below:

$$KI^{127} + CH_3 I^{131} = KI^{131} + CH_3 I^{127}$$

While in the adsorbed state, ion exchange occurs. The removal efficiency depends upon the relative amounts of radioactive iodine to stable iodine on the adsorbent.

4.3.2.9 Downstream HEPA Filter

A second HEPA filter bank is downstream of the charcoal adsorber bed. It has the same design parameters as the upstream HEPA filter. This filter provides redundancy in the event of any damage to the first HEPA filter. Also, it prevents any charcoal fines originating in the charcoal adsorber from exiting the discharge plenum and being carried out the elevated release point.

Of the 1190 scfm of air leaving the filter, 1160 scfm travels to the elevated release point and 30 scfm flows backward through the inactive filter train to remove decay heat from its charcoal filter. This cooling air exits through the fixed office located at the standby filters inlet and returns to the secondary containment atmosphere.

4.3.2.10 Flow Control Dampers

Flow control dampers MOV-34A&B modulate as necessary to maintain system design exhaust flow at 1160 scfm to the elevated release point irregardless of how many fans and filters are in operation. This flow rate is equivalent to the maximum design in-leakage created with the secondary containment internal pressure at -0.5 in. H_2O in combination with the external containment surface pressure created by a 30 mph wind.

4.3.2.11 Area Unit Coolers

Two redundant sets of four unit coolers each are provided in areas of high heat gain such as RHR/Core Spray Area and Refueling Level Area. These area unit coolers start when the RBSVS is initiated and the associated temperature control valves (TCVs) modulate as necessary with temperature. Unit coolers are discussed in greater detail in Chapter 4.2.

4.3.3 System Features and Interfaces

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

4.3.3.1 Normal Operation

During normal plant operation, the RBSVS is shutdown and in a standby mode. It is capable of being started manually or automatically. The RBSVS can process air flow from the following areas or components:

- Refueling Area
- All Reactor Building elevations (except designated Contaminated Areas)
- Drywell
- Suppression Chamber
- High Pressure Coolant Injection System gland exhaust

4.3.3.2 Automatic Initiation

Control logic for the RBSVS automatically starts all filter trains upon receipt of an accident signal. Once started, all trains continue running for the duration of the accident. Should one train fail, the remaining train will continue to provide design flow.The RBSVS automatically starts if *any* of the following conditions exist:

- Drywell pressure high (1.69 psig)
- Reactor water level 2 (-38 inches)
- Refuel area exhaust radiation high (35 mR/hr)
- Vent/purge exhaust duct rad high (>5.4 x 10^5 cpm)
- Reactor Building ΔP low (-0.30 in. H_2O and a 30 sec. time delay)
- Loss of power to Normal 480V Busses

- RBNVS supply or exhaust iso valve failure (valve not full open)
- RBSVS Manual initiation signal

When initiated, the RBSVS performs the following functions:

1. Isolates the Secondary Containment by closing the RBNVS supply and exhaust isolation valves (AOV-35A&B and AOV-37A&B) and trips the operating RBNVS supply fan. The Primary Containment Purge/Vent Supply and Exhaust Isolation Valves also close, if open. The Refueling Area and Contaminated Area Exhaust Dampers close. The two normally running Reactor Building Exhaust Fans continue operating and act as recirculation fans for the entire Secondary Containment volume. One exhaust fan is not required and is soon secured. The RBSVS Mixing Plenum inlet valves open and 43,765 cfm of air is circulated through the RBSVS air cooling coils and discharged to the refueling level air discharge header. The remaining 1235 scfm is redirected to the suction of the filter train booster fans.

2. Filter Train Booster Fans start and their associated discharge dampers open, filter train inlet dampers open, and filter train flow control dampers modulate to maintain flow at 1160 cfm to the elevated release point. Also, all RBSVS area unit coolers start.

With the RBSVS aligned and operating, the Secondary Containment is maintained at a - 0.5 inch H_2O pressure by modulation of the filter train flow control dampers. The Reactor Building atmosphere is cooled as it recirculates through the RBSVS Cooling Coils and the Area Unit Coolers in the RHR/Core Spray Pump rooms and refueling level areas.

The RBSVS Filter Trains, in conjunction with the large amount of containment internal recirculation air flow, provides holdup, mixing, and dilution of airborne radioactive contaminants and minimizes the release of radioactivity to the environment by filtering the exhaust air prior to release.

4.3.3.3 Inspection and Testing

The Secondary Containment inleakage rate is determined by operating the RBSVS while the Reactor Building Normal Ventilation System is isolated. RBSVS flow to the elevated release point is adjusted to 1160 scfm after which the secondary containment is verified to be at a pressure greater than 0.5 inches of water gauge less than the pressure outside the building.

The RBSVS trains and fans are arranged so that one train and its associated fan can be serviced or tested while the redundant train is ready to operate. Upon receipt of an accident signal secondary containment isolates, the RBSVS train being tested will shut down and isolate, and the redundant train will start automatically.

4.3.3.4 System Interfaces

Interfaces between the Reactor Building Standby Ventilation System and other plant systems are discussed in the following paragraphs.

Primary Containment System (Chapter 4.1)

The RBSVS can be used to purge atmosphere from either the drywell or suppression chamber.

Secondary Containment System (Chapter 4.2)

The RBSVS automatically starts and takes suction from the secondary containment under accident conditions in order to maintain the secondary containment at a negative pressure. The RBSVS is also used to test for secondary containment integrity.

Process and Area Radiation Monitoring System (Chapter 8.4)

The Process and Area Radiation Monitoring System provides signals which automatically start the RBSVS under conditions of high radiation in the Refueling Level Exhaust Duct or Primary Containment Vent/Purge Duct. RBSVS filtration effluent is monitored for activity by the Area Radiation Monitoring System prior to discharge to the environment.

High Pressure Coolant Injection System (Chapter 10.1)

The High Pressure Coolant Injection (HPCI) System gland exhaust blower discharges to the RBSVS for processing and release of noncondensible gases.

4.3.4 BWR Differences

Most all BWRs designate the RBSVS as the Standby Gas Treatment System (SGTS). Some differences exist between these systems but the system purpose and design are generic. Many filter trains locate the RBSVS Train Booster Fans downstream of the last HEPA filter.

4.3.5 Summary

Classification - Safety related system; engineered safety feature system.

Purpose - To process secondary containment atmosphere prior to release under accident conditions; to maintain a subatmospheric pressure in the secondary containment relative to the environment outside the building; to provide a means for purging the primary containment; to perform leak tests of the secondary containment.

Components - Reactor building exhaust fans; mixing plenum; booster fans; electric heater; prefilter; HEPA filters; charcoal filter; flow control dampers; area unit coolers.

System Interfaces - Primary Containment System; Secondary Containment System; Process and Area Radiation Monitoring System; High Pressure Coolant Injection System.

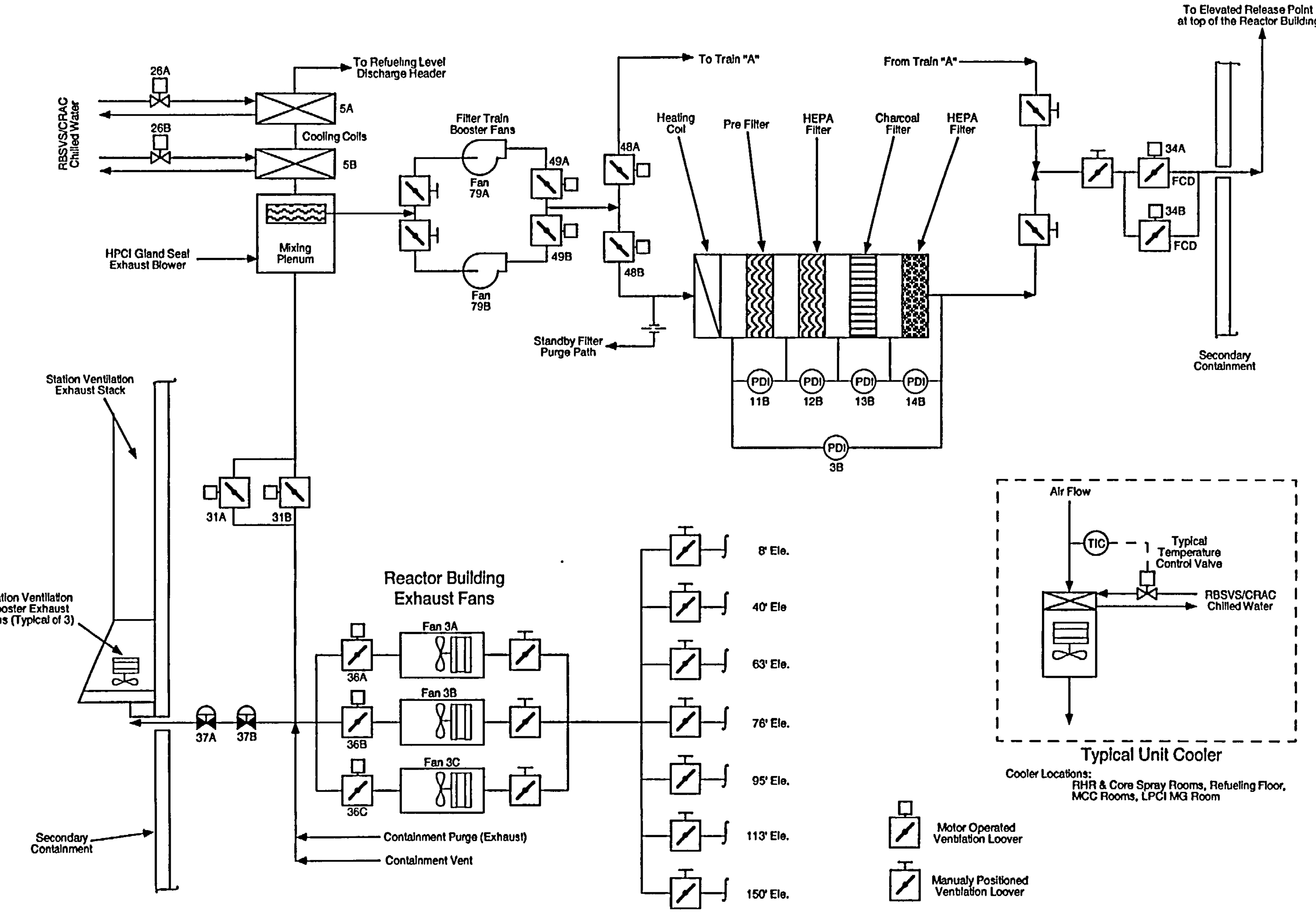

Figure 4.3-1 Reactor Building Standby Ventilation System

Boiling Water Reactor
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Technology Manual

Chapter 4.4

Nuclear Steam Supply Shutoff System

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4.4 NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM

Learning Objectives:

1. State the system's purpose.
2. Explain the logic arrangement within the Nuclear Steam Supply Shutoff System (NSSSS).
3. Explain the concept of "Group" Isolation.
4. List the parameters which initiate Group 1 isolation, conditions under which the isolation are or can be bypassed, and the reason for having each isolation signal.
5. Explain how isolation can be manually initiated.
6. Describe the interfaces between this system and the following:
   a. Main Steam System
   b. Reactor Water Cleanup System
   c. Recirculation System
   d. Residual Heat Removal System
   e. Liquid Radwaste System
   f. Reactor Protection System
   g. Primary Containment System
   h. Secondary Containment System
   i. Reactor Core Isolation Cooling System
   j. High Pressure Coolant Injection System

4.4.1 Introduction

The purpose of the Nuclear Steam Supply Shutoff System (NSSSS or NS4) is to isolate the primary and secondary containments during accident conditions in order to prevent the release of radioactive materials to the environment in excess of 10 CFR 100 limits.

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

The NSSSS includes sensors, power supplies, trip systems, logic channels, switches, transmitters and remotely operated valve closing mechanisms associated with those valves which affect isolation of primary containment and/or the reactor coolant system. Piping systems that penetrate primary containment and interface directly with the primary containment atmosphere or the reactor coolant system contain two isolation valves. One isolation valve is located inside primary containment and the other isolation valve is located outside primary containment, as close to the containment wall as practical.

The NSSSS determines, from information provided by reactor plant process instrumentation, which system(s) should be isolated and provides appropriate isolation signals to that system(s). This concept results in isolation of "Groups" that appear to be the problem rather than completely isolating both containments, the reactor coolant system, and unnecessarily challenging safety related or engineered safety feature systems. Local sensor elements provide input signals to the NSSSS from the containment systems, reactor vessel instrumentation system, reactor protection system, main steam system, condensate and feedwater system, standby liquid control system, process and area radiation monitoring system, and individual system leak detection parameters.

The NSSSS is divided into 17 basic isolation groups. Group 1 consists of Main Steam Isolation Valves (MSIVs). Main steam line isolation logic is provided to control the loss of coolant from the reactor vessel and the release of radioactive materials to the environment. The NSSSS logic responds to signals which indicate a breach of the reactor coolant pressure boundary, fuel cladding failure, loss of heat sink (main condenser), or a failure of the reactor pressure control system (Electro Hydraulic Control System).

Groups 2 through 17 include systems such as primary and secondary containments, reactor water cleanup, high pressure coolant injection, reactor core isolation cooling, residual heat removal shutdown cooling, reactor building closed loop cooling water, and Post Accident Sampling System (PASS).

4.4.2 Component Description

The major components of the nuclear steam supply shutoff system are discussed in the paragraphs which follow.

4.4.2.1 Sensors

Sensors utilized by the NSSSS are instruments belonging to other plant systems such as the systems leak detection processors and the reactor vessel instrumentation system. The sensors measure reactor water level; main steam line radiation, flow, pressure, and area temperature; drywell pressure; reactor water cleanup system flow and temperature; and the initiation of the standby liquid control system.

4.4.2.2 Trip System

A trip system is an arrangement of various sensors and associated components which are used to evaluate plant parameters and produce discrete trip outputs when the logic is satisfied. The components within a trip system must maintain electrical and physical separation from the components in the other trip system.

The logic trains that make up a trip system are divided into four separate channels. The NSSSS logic utilized for MSIV (Group 1) isolation is arranged in a one-out-of-two-taken-twice manner, similar to the reactor protection system (Chapter 7.3). The logic is composed of two trip systems (A and B). Trip system A consists of channels A1 & A2 whereas trip system B consists of channels B1 & B2. A full NSSSS group 1 isolation is defined as concurrent trips of trip systems A & B. This scheme requires at least one channel in each trip system to trip. This will result in closure of all valves in this group. A 1/2 group isolation is a trip of only one trip system. This satisfies part of the isolation logic for a particular valve group but will not cause valve motion.

Groups 2 - 17 isolation logic, Figure 4.4-6, unlike the MSIV logic, is arranged with channels A1 & B1 in trip system A and channels A2 & B2 in trip system B. Channels A1 & B1 are associated with inboard isolation valves and channels A2 & B2 are associated with the outboard isolation valves. These groups are referred to as dual trip systems consisting of a two-out-of-two logic (or an inboard-outboard logic). Inboard-outboard logic schemes require both channels of a trip system to trip before initiating a closure of the corresponding inboard or outboard isolation valve. This group logic portion of the NSSSS responds to signals that could indicate a breach of a specific system.

4.4.2.3 Power Supplies

Power for the trip systems of the NSSSS is supplied from the Reactor Protection System (RPS) motor-generator sets via 120Vac RPS busses A & B (Figure 4.4-1). All sensors and trip contacts essential to safety are closed when energized. The trip logic for each isolation group de-energize to trip.

The isolation valves receive power from emergency power sources. Power for the operation of two valves in the same process line is fed from different power sources. In most cases, one valve will be powered from an ac bus of the appropriate voltage, and the other by dc from the station batteries. For example, each MSIV utilizes a pair of ac and dc solenoid operated valves, in series, to control pneumatic pressure to the valve operators.

4.4.2.4 Isolation Valves

There are three classes of isolation valves, which are dependent on the degree to which their associated process lines penetrate the reactor coolant boundary or communicate with primary containment. Regulatory requirements for each valve class are located in 10 CFR 50, App A, Criterions 55,56, and 57.

Class A valves (Criterion 55) are in process lines which penetrate the primary containment and communicate with the reactor vessel or coolant boundary. Two isolation valves are required on these lines, one inside and one outside the primary containment.

Class B valves (Criterion 56) are in process lines which penetrate the primary containment and communicate with its atmosphere. The general design criterion states that one valve will be located on each side of the containment penetration.

Class C valves (Criterion 57) are in pipelines that penetrate the primary containment, but do not communicate with the reactor vessel or reactor coolant boundary and are not open to the primary containment. One valve outside the containment is provided for each line.

The classes of valves described are guidelines; there are exceptions to each. When the type or number of valves required would interfere with the operation of the system, alternative components are used. In order to limit radiological releases or maintain core coverage, class A and B valves are closed upon receipt of a trip signal.

Isolation valves for class A, B, and C process lines penetrating the primary containment boundary are listed in Table 4.4-1, along with pertinent information. This table provides information about each valve, e.g. the type of valve, the power required to operate the valve, the pipe class, the specific isolation signals which automatically close the valve, and the normal status of the valve.

4.4.2.5 NSSSS Isolation Groups

The NSSSS logic is divided into 17 discrete groups. The valve groups function individually or collectively to isolate the reactor coolant pressure boundary and primary containment atmosphere, or isolate closed systems that penetrate the primary containment but do not connect to the reactor coolant pressure boundary or the primary containment atmosphere.

Systems associated with isolation of the reactor coolant pressure boundary:

a. Main Steam System - Main steam lines (Grp 1) and main steam line drains (Grp 14).
b. Reactor Recirculation System - Reactor water sample line (Grp 10).
c. Reactor Water Cleanup System - Inboard and outboard suction isolation valves (Grps 3 & 4).
d. Residual Heat Removal System - LPCI injection valves (Grp 5 if in SDC mode),

head spray line to reactor (Grp 5), and Shutdown Cooling (SDC) suction isolation valves (Grp 5).

e. High Pressure Coolant Injection System- Inboard and outboard steam supply isolation valves (Grps 7 & 16), HPCI pump suction from the suppression pool (Grp 16), and HPCI exhaust line vacuum breaker isolation (Grp 13).

f. Reactor Core Isolation Cooling System - Inboard and outboard steam supply iso. valves (Grps 6 & 15) and exhaust line vacuum breaker isolation (Grp 12).

Systems associated with isolation of the primary containment atmosphere:

a. Primary Containment Inerting System (Grp 9).

b. Primary Containment Vent & Purge System (Grp 9).

c. Primary Containment Radiation Monitoring System (Grp 2).

d. Post Accident Sampling System (Grp 17) - Reactor water sample and primary containment atmosphere sample supply and return lines.

e. Residual Heat Removal System (Grp 2)- Containment spray (drywell or suppression pool), test line return to suppression pool, and steam condensing discharge to suppression pool.

f. Core Spray System (Grp 2) - Test line return to suppression pool.

g. Suppression Pool Cleanup/Pumpdown System (Grp 11) - Supply and return lines.

h. Drywell Floor & Equipment Drainage System (Grp 11) - Floor drain sump and equipment drain sump isolation valves to radwaste.

i. Traversing Incore Probe (TIP) System (Grp 11) - TIP withdrawal control, TIP ball valves closure, and N_2 purge for TIP drives.

j. Instrument Air to Suppression Chamber (Grp 2).

A closed system that penetrates primary containment but is not connected to the reactor coolant system nor is open to the primary containment atmosphere is:

a. Reactor Building Closed Loop Cooling Water System (Grp 8) - Drywell unit cooler supply and return valves and isolation from primary containment.

Table 4.4-2 provides a list of valve groups, the affected system or portion of system that is assigned to each group, and the specific valve(s) that are closed when its group receives an isolation signal. Table 4.4-3 specifies for each valve group, the isolation signal(s) and associated trip setpoint(s). The isolation groups are listed and described in the following paragraphs:

Groups 1 and 14 Main Steam System

Isolation Group 1 consists of Main Steam Isolation Valves (MSIVs) and Group 14 includes main steam line drain valves. These groups are provided to control the loss of coolant from the reactor vessel and the release of radioactive materials to the environment. The NSSSS Groups 1 & 14 logic respond to signals which indicate a breach of the Reactor Coolant Pressure Boundary (RCPB), a breach of the fuel cladding, a failure of the Electro Hydraulic Control (EHC) System or a loss of the primary heat sink (main condenser). When selected process parameters reach preset levels, isolation demand signals generated by a portion of the NSSSS cause the main steam isolation valves and main steam line drain valves to close.

Identical control inputs are provided to NSSSS Groups 1 & 14 logic that indicate a leak or break in the main steam system. In each main steam line, a flow element is used to develop a steam flow signal by measuring the differential pressure across the steam line flow restrictor and converting it to a signal proportional to flow. Excessive flow through a steam line is indicative of a break in that line. Steam line flow is sensed by four differential pressure indicating switches per line, one for each trip channel A - D. If either A or C and B or D d/p switches indicate a high flow in that steam line (>106 psid = 134%) all MSIVs will receive a closure signal.

Leakage sensing devices also provide control inputs from temperature elements that detect high steam line area space temperatures (>155°F) in the steam tunnel or turbine building and a main steam line tunnel differential temperature high (>50°F). By utilizing and placing temperature elements in the above spaces, a rapid detection of small steam line breaks inside and outside the secondary containment can be detected before the steam line high flow reaches a large break setpoint.

The reactor water level isolation signal (level 1) is high enough to ensure that enough water remains in the reactor vessel to cover the fuel and allow the emergency core cooling systems sufficient time to adequately cool the core in the event of a large main steam line break. Water level is sensed by level transmitters which send analog signals to trip units and relays of the NSSSS.

The low steam line pressure isolation signal (<825 psig in RUN mode) is sensed by four pressure transmitters, one located on each steam line just upstream of the turbine stop valves. The low pressure isolation is bypassed when the reactor mode switch is not in RUN mode to allow startup, heatup, and pressurization of the reactor with the MSIVs open. If steam line pressure decreases to 825 psig during power operation, the most probable cause is a malfunction in the Electro Hydraulic Control System (EHC, Chapter 3.2). The isolation on low steam line pressure prevents a rapid depressurization of the reactor vessel which otherwise would have resulted in an excessively rapid cooldown, an undesirable differential pressure across the fuel channels, and operation at a high power - low pressure condition (CPR limit consideration).

Main steam line high radiation also results in a main steam isolation. Four radiation detectors, located in the vicinity of the main steam lines within the steam tunnel, send signals to associated process radiation monitoring system trip units. The detectors are located in positions that ensure each can monitor radiation levels of all four steam lines simultaneously and not just one steam line. If the radiation levels reach 3 times normal full power background values, indicating a gross failure of the fuel barriers, or the trip unit becomes inoperable, a trip is sent to the RPS which in turn sends a signal to the NSSSS logic.

A loss of the reactor's normal heat sink occurs when main condenser vacuum is lost or reduced due to condenser tube ruptures or gross vacuum leaks. To preclude possible overpressurization of the condenser, the MSIVs are closed when vacuum decreases to <8.5 inches Hg vacuum.

Group 2 RHR, Core Spray, Instrument Air, Rad Monitoring

Isolation Group 2 consists of the drywell radiation monitoring subsystem; instrument air to suppression chamber outer isolation valve; core spray test line; and RHR system containment spray valves to the drywell and suppression chamber, test line return to suppression chamber, and steam condensing mode discharge. These system valves close in response to drywell pressure high (>1.69 psig) or reactor vessel water level 1 (>-132.5 inches).

Groups 3 & 4 Reactor Water Cleanup System

Isolation Groups 3 & 4 consists of Reactor Water Cleanup (RWCU) System suction inboard and outboard valves respectively. The RWCU system isolates on signals symptomatic of a reactor coolant pressure boundary leak, RWCU system leak, or a malfunction resulting in a high temperature upstream of the filter/demineralizers. In addition, a Standby Liquid Control (SLC) System initiation signal will cause the RWCU system to isolate.

Two reactor water level input signals are sent to each RWCU isolation logic channel. If both level signals in a single logic channel indicate a tripped condition (Level 2), an isolation demand signal is sent to the valve associated with that channel. This isolation eliminates a possible source of Reactor Coolant Pressure Boundary (RCPB) leakage that is indicated by the decreasing reactor water level condition.

Reactor water cleanup system differential flow is determined by comparing the difference between RWCU system inlet flow and the RWCU system outlet flow (which is the sum of the RWCU drain flow to the main condenser and/or liquid radwaste system and the RWCU return flow to the condensate and feedwater system). High RWCU differential flow indicates that water entering the system is not leaving by normal flow paths, i.e., it is leaking out of the system in an uncontrolled manner. A RWCU system isolation is initiated to eliminate this possible reactor coolant leakage path. The outlet flows are summed and compared to the inlet flow. If a high differential flow (>45 gpm) is sensed and is maintained for more than 45 seconds, an isolation demand signal is sent to the RWCU isolation logic channels in the NSSSS. The purpose of the 45 second time delay is to allow for changing RWCU system modes of operation and starting or stopping RWCU pumps without resulting in an unnecessary RWCU system isolation due to momentary system flow transients.

The RWCU system heat exchanger and pump room areas are monitored for high area temperatures (>155°F). A high temperature in either of these areas indicates that high temperature fluid may be leaking from the RWCU system and as a result, a system isolation is actuated. In addition, channel A alone will cause an outboard valve isolation because of high outlet temperature from the RWCU nonregenerative heat exchanger. This parameter is indicative of high drain flow or low cooling water flow to the nonregenerative heat exchanger and not a breach of system integrity, therefore only RWCU flow is stopped.

If the Standby Liquid Control (SLC) system is initiated, the RWCU system is isolated to prevent the RWCU system from removing the neutron absorbing material from the reactor vessel water, and thus defeating the function of the SLC system injection.
Only the RWCU suction outboard valve is closed under this condition.

Group 5 RHR Shutdown Cooling Mode

Isolation Group 5 consists of Residual Heat Removal (RHR) System valves including the shutdown cooling mode suction valves from the recirculation system, RHR injection line shutdown cooling return valves to the recirculation system, and head spray line valves to the reactor pressure vessel. Isolation of these valves is initiated by reactor vessel level 3 (<12.5 inches) or reactor recirculation system pump suction line pressure high (>125 psig - only if RHR is in shutdown cooling mode).

The RHR system shutdown cooling mode isolates in response to low reactor water level because leakage may be from the RHR system. If reactor pressure increases to such a value that the saturation temperature corresponding to that pressure is approaching the temperature rating of the RHR system pumps (360°F), the RHR system isolation logic closes the shutdown cooling suction and discharge valves and the head spray valve to protect the RHR pumps.

Groups 6 & 15 RCIC System

The Reactor Core Isolation Cooling (RCIC) system turbine steam inlet line MOVs 041 & 047 and MOVs 042 & 048 are actuated by Group 6 logic and Group 15 logic, respectively. Initiating signals consist of RCIC steam line flow high (>291 inches H_20), RCIC steam supply pressure low (<15 psig), RCIC turbine exhaust diaphragm pressure high (>10 psig), RCIC equipment area temperature high (>155°F), and RCIC elevation 63' area temperature high (>193°F).

Group 7 & 16 HPCI System

The High Pressure Coolant Injection (HPCI) System turbine steam inlet line MOVs 041 & 047 and MOVs 042 & 048 are actuated by Group 7 and 16 logic respectively. Also, Group 16 isolates suction from the suppression pool using valve MOV-032. Isolation signals include HPCI steam line flow high (>212 inches H_20), HPCI steam supply pressure low (<100 psig), HPCI turbine exhaust diaphragm pressure high (>10 psig), HPCI equipment area temperature high (>155°F), and HPCI elevation 63' area temperature high (>193°F).

Group 8 Reactor Building Closed Loop Cooling Water System

Reactor Building Closed Loop Cooling Water (RBCLCW) System valves closed by Group 8 initiation signals include the RBCLCW primary containment supply and return headers isolation valves and drywell unit coolers supply and return line valves. Isolation signals include drywell pressure high, reactor vessel water level 1, and RBCLCW head tank level low-low.

Group 9 Primary Containment Purge and N_2 Inerting Systems

Valves associated with the primary containment purge and N_2 inerting systems are normally maintained closed during power operation. Should conditions exist which would indicate a breach of the reactor coolant pressure boundary, a fuel handling accident inside the secondary containment, or the inability of the Reactor Building Normal Ventilation System (RBNVS) to maintain a negative reactor building pressure, all associated valves will be automatically closed, if open. Also, the RBNVS isolation valves will close, the Reactor Building Standby Ventilation

System (RBSVS) will actuate, and the control room emergency filtration system will start in the pressurization mode of operation. Initiation signals used to perform those functions include drywell pressure high, reactor vessel water level 2, refueling floor ventilation exhaust radiation high, reactor building differential pressure low, and manual initiation.

Group 10 Sample Coolant from Reactor Pressure Vessel

Reactor coolant water samples are normally obtained via the RWCU system. To provide sampling capability when the RWCU system is unavailable, samples are withdrawn from the recirculation loop B riser distribution header. In performing this function, the sample piping must penetrate the RCPB and, therefore, is a potential leakage path for reactor coolant. Because of this, when water level in the reactor vessel decreases to Level 2, reactor water sample line isolation occurs which automatically isolates this possible source of leakage.

Radiation levels in accessible plant areas could increase to very high levels if sampling were in progress using this line and a fuel cladding failure coincidently occurs. An excellent indication of gross fuel barrier failure is a sudden increase of main steam line radiation level. Therefore, the reactor water sample line valves isolate upon receipt of a main steam line high radiation signal.

Group 11 Drywell Equipment/Floor Drain System, Suppression Pool Cleanup System, and N_2 Purge for TIP System

To reduce the number of potential leakage paths of radioactive materials from the primary containment during accident conditions, the Group 11 isolation valves to/from the containment are closed if any of the following parameters are reached: reactor vessel water level 2, drywell pressure high, or manual initiation.

Groups 12 & 13 RCIC & HPCI Vacuum Breakers

The RCIC/HPCI vacuum breaker isolation valves MOVs 1E51-049/1E41-049 are closed by Groups 12 and 13 logic, respectively. Isolation signals include RCIC/HPCI steam supply pressure low (< 15/100 psig) coincident with drywell pressure high (>1.69 psig). This will effectively complete isolation of the entire RCIC/HPCI system from primary containment if a condition exists that could potentially threaten primary containment integrity.

Group 13 - see Groups 12 & 13 above.

Group 14 - see Groups 1 & 14 above.

Group 15 - see Groups 6 & 15 above.

Group 16 - see Groups 7 & 16 above.

Group 17 Post Accident Sampling System

The Post Accident Sampling System (PASS) reactor water and primary containment atmosphere sample supply and return lines are isolated by conditions indicative of a breach of the reactor coolant pressure boundary. These flow paths in themselves could be the location of the breach or become a potential uncontrolled release path of containment atmosphere radioactive particulates or gases following the breach. These lines automatically isolate upon receipt of reactor vessel water level 3, drywell pressure high, or manual initiation.

4.4.3 System Features And Interfaces

A short discussion of system operation, features, and interfaces with other plant systems is given in the paragraphs that follow.

4.4.3.1 Manual Isolation

There are four manual isolation pushbutton switches, one for each logic channel (Figures 4.4-4 and 4.4-5). Operation of a manual isolation pushbutton requires two separate actions by the operator. First, the switch is armed by rotating the switch collar located at the base of the switch. Secondly, the pushbutton is depressed to initiate the isolation function. This switch design prevents inadvertent operation of the switches during plant operation. Selective isolation is achieved by arming and depressing appropriate combinations of the manual pushbuttons as illustrated on Table 4.4-4.

The High Pressure Coolant Injection (HPCI) System and the Reactor Core Isolation Cooling (RCIC) System are each provided with its own manual isolation pushbutton switch. The pushbuttons are operated similar to those above but are active only if the HPCI (or RCIC) system has received an automatic initiation signal. Once activated, only the outboard steam supply valve closes for its respective system. Operation of these systems is addressed in greater detail in their respective manual chapters.

4.4.3.2 Automatic Isolation

Automatic isolation will be discussed for NSSSS Group I logic (MSIVs) followed by Groups 2 - 17 logic. The MSIV logic, Figures 4.4-2 and 4.4-3, is a contact and relay logic that is shown in the normal/energized condition. The logic utilized is divided into four individual logic channels (A1, A2, B1, and B2 which are also referred to as channels A, C, B, and D respectively). Channels A1 & A2 are assigned to trip system A and B1 & B2 to trip system B. A trip of either channel A1 or A2 will cause deenergization of the A pilot solenoids (120Vac) for inboard MSIVs and B pilot solenoids (125Vdc) for outboard MSIVs. A trip of either channel B1 or B2 will cause deenergization of the A solenoids for outboard MSIVs and B solenoids for inboard MSIVs. Therefore, both MSIV pilot solenoids must deenergize for an MSIV closure to occur. By following the information above and utilizing Figure 4.4-2, it is apparent that the trip logic is a one-out-of-two-taken-twice scheme.

The control of the K7 relay contacts can be seen on Figure 4.4-3. Reaching any of the isolation setpoints will open the appropriate contact in series with the K7 relay causing K7 to deenergize. If the correct two relays are deenergized (K7A or K7C and K7B or K7D) the MSIVs will receive a close signal.

NSSSS Groups 2 - 17 logic channels are assigned differently to the trip Systems than the MSIVs, e.g. channels A1 & B1 are assigned to trip system A and A2 & B2 to trip system B. This logic arrangement results in K7A and K7B deenergizing the valve control relays for and causing closure of inboard isolation valves only (K7C and K7D for outboard valves only). The trip logic are arranged in a two-out-of-two logic scheme which is also referred to as an inboard-outboard logic.

4.4.3.3 Isolation Reset

Once isolation is initiated, the valve(s) continue to close, even if the condition that caused the isolation is restored to normal. The operator must manually reposition valve control switches in the control room and manually reset the logic in order to reopen a valve that has been automatically closed. This involves placing the valve control switches in the closed position and then depressing the applicable isolation reset switch.

There are two isolation reset pushbutton switches provided in the control room. Pushbutton A resets the valve closure relay logic circuits for all inboard isolation valves (except MSIV and MSL drain isolation valves) and the pushbutton B is used to reset the outboard isolation valves (except MSIV and MSL drain isolation valves). In order to reset an MSIV isolation after the initiating signal has cleared, all MSIV control switches must be placed in the CLOSE position and both A and B reset pushbuttons must be depressed. This arrangement ensures the MSIVs logic will remain in the tripped condition unless both pushbuttons are depressed.

4.4.3.4 System Interrelations

The NSSSS interrelates with various plant systems as described in the paragraphs that follow.

Leak Detection System (No section in manual)

The LD System provides isolation demand signals to the NSSSS for automatic isolation of certain systems if leakage is detected.

Main Steam System (Section 2.5)

The NSSSS isolates the Main Steam System when Group 1 and 14 logic are satisfied.

Reactor Water Cleanup System (Section 2.8)

The NSSSS isolates the RWCU System when Groups 3 or 4 logic are satisfied.

Recirculation System (Section 2.4)

The NSSSS isolates the Recirculation System sample line when Group 10 logic is satisfied.

Residual Heat Removal System (Section 10.4)

The NSSSS isolates portions of the RHR System when Groups 2 or 5 logic are satisfied.

Liquid Radwaste System (Section 8.2)

The NSSSS isolates the Drywell Equipment and Floor Drain System lines that penetrate the Primary Containment when Group 11 logic is satisfied.

Reactor Protection System (7.3)

The NSSSS receives trip system and logic power from RPS Busses A & B.

Primary Containment Combustible Gas Control System (Section 4.1)

The NSSSS isolates the drywell purge lines, drywell pressure bleedoff vent lines, and primary containment inerting system when Group 9 logic is satisfied.

Reactor Building Closed Loop Cooling Water System (Section 11.3)

The NSSSS isolates the RBCLCW supply and return lines for the entire Primary Containment Structure and all individual drywell unit coolers when Group 8 logic is satisfied.

Service and Instrument Air System (Section 11.8)

The NSSSS isolates the SIA System line to the suppression chamber when Group 2 logic is satisfied.

4.4.4 BWR Differences

The NSSSS described in this section is similar to the NSSSS for some BWR/5 & 6 product line plants without solid state logic. This system is usually called the Primary Containment Isolation System (PCIS) for facilities of product lines BWR/2 through BWR/4. All BWR facilities have an NSSSS or PCIS.

4.4.5 Summary

Classification - Safety related system; Engineered safety feature system

Purpose - To isolate the primary and secondary containments during accident conditions in order to prevent the release of radioactive materials to the environment in excess of 10 CFR 100 limits.

Components - Sensors, power supplies, trip systems, logic channels, and isolation valves.

System Interrelations - LD System, Main Steam System, RWCU System, Recirculation System, RHR System, Liquid Radwaste System, Reactor Protection System, CCGC System, RBCLCW System, and SIA System.

TABLE 4.4-1 CONTAINMENT ISOLATION VALVE SUMMARY

(Numbers in parentheses are keyed to notes on pages 9 and 10 signal codes are listed on page 10)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
Main Steam	55	4	1	24	Inside	AO Globe	Air/AC/DC	Air/Spring	G,C,D,E,P,R,T,RM	5	Open	(1)
			1	24	Outside	AO Globe	Air/AC/DC	Air/Spring	G,C,D,E,P,R,T,RM	5	Open	(1)
Main Steam Line Drain and	55	4	1	2	Outside	MO Globe	AC	AC	G,C,D,E,P,R,T,RM,T1	10	Open	
MSIV-Leakage Control System	55	4	1	1 1/2	Outside	MO Globe	AC	AC	RM	N/A	Closed	(19)
Feedwater	55	1	1	18	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	18	Outside	VTC	Flow	Reverse Flow/ Air/Spring	Reverse Flow/F,G,RM	N/A	Open	(11)
Feedwater	55	1	1	18	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	18	Outside	VTC	Flow	Reverse Flow/ Air/Spring	Reverse Flow/F,G,RM	N/A	Open	(11)
Main Steam Line Drain	55	1	1	3	Inside	MO Gate	AC	AC	G,C,D,E,P,R,T,RM,T1	19	Open	
			1	3	Outside	MO Gate	DC	DC	G,C,D,E,P,R,T,RM,T1	19	Open	
RWCU Line from RPV	55	1	1	6	Inside	MO Gate	AC	AC	B,J,RM,T1	30	Open	
			1	6	Outside	MO Gate	DC	DC	B,J,W,Y,RM,T1	30	Open	
RHR Shutdown Cooling from RPV	55	1	1	20	Inside	MO Gate	AC	AC	A,U,RM	63	Closed	
			1	20	Outside	MO Gate	DC	DC	A,U,RM	56	Closed	
			1	3/4	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
RHR Injection Line to Re-Circulation System Return	55	2	1	24	Inside	VTC	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(3)
			1	2	Inside	MO Gate	AC	AC	RM	44	Closed	
			1	24	Outside	MO Gate	AC	AC	RM,A1	N/A	Closed	(12)
RHR - Containment Spray Dry Well	56	2	1	10	Outside	MO Gate	AC	AC	F,G,RM	92	Closed	(2)
			1	10	Outside	MO Angle	AC	AC	F,G,RM	16	Closed	(2)
RHR - Containment Spray Suppression Chamber	56	2	1	6	Outside	MO Globe	AC	AC	F,G,RM	80	Closed	(2)

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
RHR Pump Suction	56	4	1	20	Outside	MO Gate	AC	AC	RM	N/A	Open	(13)
RHR Test Line Return to	56	1	1	16	Outside	MO Globe	AC	AC	F,G,RM	146	Closed	(2)
Suppression Chamber,		1	1	16	Outside	MO Gate	AC	AC	F,G,RM	168	Closed	(2)
Suppression Pool		1	2	6	Outside	MO Gate	AC	AC	B,F,RM	56	Closed	
Cleanup Return												
RHR Steam Condensing Discharge		1	1	4	Outside	MO Gate	AC	AC	F,G,RM	48	Closed	
RHR Minimum Flow,		1	1	4	Outside	MO Gate	AC	AC	RM	N/A	Open	(16)
Core Spray Test Line,		1	1	10	Outside	MO Globe	AC	AC	F,G,RM	112	Closed	
Core Spray Minimum Flow,		1	1	3	Outside	MO Gate	AC	AC	RM	N/A	Open	(16)
Suppression Pool Pump Back,		1	1	3	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
and			1	3	Outside	MO Gate	AC	AC	A,F,RM	34	Closed	
Pass Sample Return		1	1	3/4	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
			1	3/4	Outside	SO Glove	AC	Reverse Flow Spring	A,F,RM	N/A	Closed	
RHR Test Line Return to	56	1	1	16	Outside	MO Globe	AC	AC	F,G,RM	146	Closed	(2)
Suppression Chamber,		1	1	16	Outside	MO Gate	AC	AC	F,G,RM	130	Closed	
RCIC Minimum Flow,		1	1	2	Outside	MO Globe	DC	DC	RM	N/A	Closed	(16,24)
HPCI Minimum Flow,		1	1	4	Outside	MO Globe	DC	DC	RM	N/A	Closed	(16,24)
RHR Steam Condensing		1	1	4	Outside	MO Gate	AC	AC	F,G,RM	48	Closed	
Discharge,												
RHR Minimum Flow,		1	1	4	Outside	MO Gate	AC	AC	RM	N/A	Open	(16)
Core Spray Test Line,		1	1	10	Outside	MO Globe	AC	AC	F,G,RM	112	Closed	
Core Spray Minimum Flow,		1	1	3	Outside	MO Gate	AC	AC	RM	N/A	Open	(16)
Relief Valve Discharge		1	1	2	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
from RHR												
Supply to RCIC Pump Suction,												
and PASS Sample Return		1	1	3/4	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
			1	3/4	Outside	SO Globe	AC	Reverse Flow Spring	A,F,RM	N/A	Closed	
RHR - Head Spray Line to RPV	55	1	1	4	Inside	MO Gate	AC	AC	A,U,RM	40	Closed	
			1	4	Outside	MO Globe	DC	DC	A,U,RM	27	Closed	
			1	3/4	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
HPCI Turbine Steam Inlet Line	55	1	1	10	Inside	MO Gate	AC	AC	K,RM	21	Open	(7)
			1	1	Inside	MO Globe	AC	AC	K,RM	21	Open	(7)
			1	10	Outside	MO Gate	DC	DC	K,RM	17	Closed	(7)
			1	1	Outside	MO Globe	DC	DC	K,RM	22	Open	(7)
HPCI Turbine Exhaust	56	1	1	18	Outside	MO Gate	DC	DC	RM	N/A	Open	
			2	18	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
Spare	-	-	-	-	-	-	-	-	-	-	-	(1) (1)
HPCI Pump Suction	56	1	1	16	Outside	MO Gate	DC	DC	K,RM	110	Closed	(19)
RCIC Turbine Steam Inlet Line	55	1	1	3	Inside	MO Gate	AC	AC	K,RM	26	Open	
			1	1	Inside	MO Globe	AC	AC	K,RM	26	Open	
			1	3	Outside	MO Gate	DC	DC	K,RM	26	Closed	(11)
			1	1	Outside	MO Globe	DC	DC	K,RM	22	Open	
RCIC Turbine Exhaust	56	1	1	8	Outside	MO Gate	DC	DC	RM	N/A	Open	
			2	8	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(11)
RCIC Vacuum Pump Discharge	56	1	1	2	Outside	MO Stop Check	Flow/DC	Reverse Flow//DC	Reverse Flow/RM	N/A	Closed	
			1	2	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
RCIC Pump Suction	56	1	1	6	Outside	MO Gate	DC	DC	RM	N/A	Closed	
Core Spray Pump Discharge to RPV	55	2	1	10	Inside	VTC	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
			1	2	Inside	MO Globe	AC	AC	RM	N/A	Closed	
			1	10	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Core Spray Pump Suction	56	2	1	14	Outside	MO Gate	AC	AC	RM	N/A	Open	
RBCLCW to Recirc Pump and Motor Coolers	57	2	1	4	Outside	MO Gate	AC	AC	RM	N/A	Open	(3) (12)
RBCLCW from Recirc Pump and Motor Coolers	57	2	1	4	Outside	MO Gate	AC	AC	RM	N/A	Open	(2) (2)
RBCLCW to Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	28	Open	(2)
RBCLCW to Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	32	Open	
RBCLCW to Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	28	Open	
RBCLCW to Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	28	Open	
RBCLCW to Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	42	Open	

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
RBCLCW from Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	24	Open	
RBCLCW from Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	24	Open	
RBCLCW from Dry Well Unit Coolers	56	1	1	3	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	3	Outside	MO Gate	AC	AC	F,G,Z,RM	22	Open	
RBCLCW from Dry Well Unit Coolers	56	1	1	4	Inside	MO Gate	AC	AC	F,G,Z,RM	36	Open	(23)
			1	4	Outside	MO Gate	AC	AC	F,G,Z,RM	38	Open	(23)
			1	3/4	Inside	RV	High Pressure	Spring	N/A	N/A	Closed	
RBCLCW from Dry Well Unit Coolers	56	1	1	4	Inside	MO Gate	AC	AC	F,G,Z,RM	30	Open	
			1	4	Outside	MO Gate	AC	AC	F,G,Z,RM	42	Open	
			1	3/4	Inside	RV	High Pressure	Spring	N/A	N/A	Closed	
Purge Air to Dry Well	56	1	1	18	Inside	AO Butterfly	AC/Air	Spring	L,RM	10	Closed	(17,25)
			1	18	Outside	AO Butterfly	AC/Air	Spring	L,RM	10	Closed	(17,25)
Purge Air from Dry Well	56	1	1	18	Inside	AO Butterfly	AC/Air	Spring	L,RM	10	Closed	(17,25)
			1	18	Outside	AO Butterfly	AC/Air	Spring	L,RM	10	Closed	(17,25)
Purge Air to Suppression Chamber	56	1	2	18	Outside	AO Butterfly	AC/Air	Spring	L,RM	10	Closed	(17,25)
Suppression Chamber Inerting		1	2	4	Outside	AO Globe	AC/Air	Spring	L,RM,S	5	Closed	(17)
Purge Air from Suppression Chamber	56	1	2	18	Outside	AO Butterfly	AC/Air	Spring	L,RM	10	Closed	(17,25)
Suppression Chamber - Vent/ Purge Line		1	2	6	Outside	AO Globe	AC/Air	Spring	L,RM,S	5	Closed	(17)
Sample Coolant from RPV	55	1	1	3/4	Inside	AO Globe	AC/Air	Spring	B,C,RM	10	Open	
			1	3/4	Outside	AO Globe	AC/Air	Spring	B,C,RM	10	Open	
Equipment Drains from Dry Well	56	1	2	3	Outside	MO Gate	AC	AC	B,F,RM	28	Open	
Floor Drains from Dry Well	56	1	1	4	Outside	MO Gate	AC	AC	B,F,RM	28	Open	
			1	4	Outside	MO Gate	AC	AC	B,F,RM	30	Open	
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Standby Liquid Coolant to RPV	55	1	1	1 1/2	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
			1	1 1/2	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
			2	1 1/2	Outside	Explosive	AC	N/A	RM	N/A	Closed	
TIP Drive Guide Tubes	57	4	1	3/8	Outside	Ball	AC	Spring	B,F,RM	N/A	Closed	(14)
			1	3/8	Outside	Explosive Shear	N/A	DC	RM	N/A	Open	(14)
Nitrogen Purge for TIP	57	1	1	1/2	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	1/2	Outside	SO Globe	AC	Reverse Flow/ Spring	A,F,RM	N/A	Open	
Instrument Air to Suppression Chamber	56	1	1	1	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	1	Outside	MO Globe	AC	AC	F,G,RM	14	Closed	
Instrument Air to Suppression Chamber	56	1	1	1	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
			1	1	Outside	MO Globe	AC	AC	F,G,RM	15	Closed	
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
HPCI Vacuum Breaker	56	1	1	2	Outside	MO Globe	DC	DC	F and X,RM	36	Open	
RCIC Vacuum Breaker	56	1	1	1 1/2	Outside	MO Globe	DC	DC	F and X,RM	32	Open	
RHR Relief Valve Discharge Vacuum Breaker	56	1	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
RHR Heat Exchanger Vent,		2	2	1	Outside	MO Globe	AC	AC	RM	N/A	Closed	
RHR Heat Exchanger,		2	1	1	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
HPCI Steam Supply to RHR Heat Exchanger, and		2	1	6	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
HPCI Steam Line Drain		1	2	1	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
Containment Atmospheric Control from Suppression Chamber, and	56	1	1	6	Outside	MO Gate	AC	AC	RM	N/A	Closed	
			1	4	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Dry Well Floor Seal Pressurization	57	1	1	1/2	Outside	MO Globe	AC	AC	RM	N/A	Open	
Containment Atmospheric Control from Suppression Chamber, and	56	1	1	6	Outside	MO Gate	AC	AC	RM	N/A	Closed	
			1	4	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Dry Well Floor Seal Pressurization	57	1	1	1/2	Outside	MO Globe	AC	AC	RM	N/A	Open	

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
Containment Atmospheric	56	1	1	6	Inside	MO Gate	AC	AC	RM	N/A	Closed	
Control from Dry Well			1	4	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Dry Well Inerting		1	2	4	Outside	AO Gate	AC/Air	Spring	L,RM,S	5	Closed	(17)
Containment Atmospheric	56	1	1	6	Inside	MO Gate	AC	AC	RM	N/A	Closed	
Control			1	4	Outside	MO Gate	AC	AC	RM	N/A	Closed	
CRD Insert and Withdraw	55	137	1	3/4	Outside	Gate	Manual	Manual	N/A	N/A	Open	(20)
Lines		137	1	1	Outside	Gate	Manual	Manual	N/A	N/A	Open	(20)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
RHR Relief Valve Dis-charge Vacuum Breaker	56	1	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	
RHR Heat Exchanger Vent,		2	2	1	Outside	MO Globe	AC	AC	RM	N/A	Closed	
RHR Heat Exchanger,		2	1	1	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
HPCI Steam Supply to RHR Heat Exchanger, and		2	1	6	Outside	RV	High Pressure	Spring	N/A	N/A	Closed	
HPCI Steam Line Drain		1	2	1	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
Suppression Pool Cleanup/ Pump Down	56	1	2	10	Outside	MO Gate	AC	AC	B,F,RM	94	Closed	(23)
Containment Atmospheric Control to Suppression Chamber	56	1	2	6	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Containment Atmospheric Control to Suppression Chamber	56	1	2	6	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Dry Well Service Air	56	1	1	1 1/2	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
			1	1 1/2	Outside	Gate	Manual	Manual	N/A	N/A	Locked/ Closed	
Containment Dry Well Radiation Monitoring Subsystem	56	1	1	1	Inside	MO Globe	AC	AC	F,G,RM	22	Open	
			1	1	Outside	MO Globe	AC	AC	F,G,RM	22	Open	
Containment Dry Well Radiation Monitoring Subsystem	56	1	1	1	Inside	MO Globe	AC	AC	F,G,RM	20	Open	
			1	1	Outside	MO Globe	AC	AC	F,G,RM	20	Open	
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Containment Atmospheric Control to Dry Well	56	1	1	6	Inside	MO Gate	AC	AC	RM	N/A	Closed	
			1	6	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Containment Atmospheric Control to Dry Well	56	1	1	6	Inside	MO Gate	AC	AC	RM	N/A	Closed	
			1	6	Outside	MO Gate	AC	AC	RM	N/A	Closed	
Dry Well Vent/Purge	56	1	1	6	Inside	AO Globe	AC/Air	Spring	L,RM,S	5	Closed	(17)
			1	6	Outside	AO Globe	AC/Air	Spring	L,RM,S	5	Closed	(17)
Spare (Reserved for RPV Internal Inspection)	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)

TABLE 4.4-1 (CONT)

Lines Isolated (22)	GDC	Number of Lines	Valves per Line	Nominal Pipe Size (in.)	Valve Location Relative to Primary Containment	Valve and/ or Operator Type (6, 22)	Power to open (5, 6)	Power to close (5, 6)	Isolation Signal	Closing Time (Sec) (10)	Normal Status (8, 9)	Remarks
Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
PASS Primary Containment Atmosphere Sample Return	56	1	1 1	3/4 3/4	Outside Outside	Check SO Globe	Flow AC	Reverse Flow Reverse Flow/ Spring	Reverse Flow A,F,RM	N/A N/A	Closed Closed	
PASS Dry Well Air Sample	56	1	2	3/4	Outside	SO Globe	AC	Reverse Flow/ Spring	A,F,RM	N/A	Closed	
Instrument Air to Dry Well	56	1	1 1	1 1/2 1 1/2	Inside Outside	Check MO Globe	Flow AC	Reverse Flow AC	Reverse Flow RM	N/A N/A	Open Open	
PASS Reactor Sample	55	1	2	3/4	Outside	SO Globe	AC	Reverse Flow/ Spring	A,F,RM	N/A	Closed	
Instrument Air to Dry Well	56	1	1 1	1 1/2 1 1/2	Inside Outside	Check MO Globe	Flow AC	Reverse Flow AC	Reverse Flow RM	N/A N/A	Open Open	
Recirc. Pump Seal Injection	55	1	1 1	3/4 3/4	Inside Outside	Check Check	Flow Flow	Reverse Flow Reverse Flow	Reverse Flow Reverse Flow	N/A N/A	Open Open	
Recirc Pump Seal Injection	55	1	1 1	3/4 3/4	Inside Outside	Check Check	Flow Flow	Reverse Flow Reverse Flow	Reverse Flow Reverse Flow	N/A N/A	Open Open	
PASS Primary Containment Atmosphere Sample Return	56	1	1 1	3/4 3/4	Outside Outside	Check SO Globe	Flow AC	Reverse Flow Reverse Flow/ Spring	Reverse Flow A,F,RM	N/A N/A	Closed Closed	
PASS Dry Well Air Sample	56	1	2	3/4	Outside	SO Globe	AC	Reverse Flow/ Spring	A,F,RM	N/A	Closed	
PASS Suppression Chamber Air Sample	56	1	2	3/4	Outside	SO Globe	AC	Reverse Flow/ Spring	A,F,RM	N/A	Closed	
PASS Suppression Chamber Air Sample	56	1	2	3/4	Outside	SO Globe	AC	Reverse Flow/ Spring	A,F,RM	N/A	Closed	

TABLE 4.4-1 (CONT)

NOTES

These Notes are keyed by number to correspond to numbers in parentheses

1 Main steam isolation valves require that both solenoid pilots be deenergized to close valves Accumulator air pressure plus spring set together close valves when both pilots are deenergized Voltage failure at only one pilot will not cause valve closure The valves are set fully close in less than 5 seconds

2 Containment spray to dry well and suppression chamber and RHR test line return to suppression chamber isolation valves will have the capability to be manually reopened after automatic closure This setup will permit containment spray, for high dry well pressure conditions, and/or suppression water cooling When automatic signals are not present, these valves may be opened for test or operating convenience

3 Testable check valves are designed for remote opening with zero differential pressure across the valve seat The valves will close on reverse flow even though the test switches may be positioned for open The valves will open when pump discharge pressure exceeds reactor pressure even though the test switch may be positioned for close

4 This line is only needed during maintenance Service air supply is disconnected during plant operation by administrative control

5 AC motor operated valves required for isolation functions are powered from the emergency AC power buses DC operated isolation valves are powered from the station batteries

6 All motor-operated isolation valves will remain in the last position upon failure of valve power All air operated isolation valves will close upon air failure

7 Signal B open, signal K overrides to close

8 Power operated valve can be opened or closed by remote manual switch for operating convenience during any mode of reactor operation except when automatic signal is present (see Note 2)

9 Normal status position of valve (open or closed) is the position during normal power operation of the reactor

10 The specified valve closure times incorporate isolation and/or system operability requirements, as appropriate

11. Special air testable check valves with a positive closing feature are designed for remote testing during normal operation to assure mechanical operability of the valve disc The remote testing feature will cause only a partial movement of the disc, into the flow stream, with only a minor effect on flow Upon receipt of an isolation signal, the actuator spring force will either cause a slight reduction in flow when the feedwater system is available or cause the valve to close, providing a positive closure differential pressure on the seated disc, when the feedwater flow is not available.

12 This valve will open when both a low reactor pressure vessel pressure and an accident signal are present

13 The motor operator of this valve is keylocked open during normal operating conditions

14 Traversing In-core Probe (TIP) System

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of four valves may be opened at any one time to conduct the calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, as indicated by a containment isolation signal, the cable is automatically retracted and the ball valve closes automatically after completion of cable withdrawal To ensure isolation capability, if a TIP cable fails to withdraw or a ball valve fails to close, an explosive, shear valve is installed in each line Upon receipt of a remote manual signal, this explosive valve will shear the TIP cable and seal the guide tube

15 All unused penetrations (designated "Spare") are capped and seal welded

16 Valve will close on system high flow

17 Isolation signals B, F, M, N, or S undervoltage will initiate the reactor building standby ventilation system (RBSVS) and simultaneously through a lockout relay isolate the purge air isolation valves

18 This valve will open when both a low differential pressure across the valve and an accident signal are present

19 Pressure sensors, sensing steam line pressure are used for interlock control to prevent inadvertent valve opening at high steam line pressures (above 35 psig)

20 Control Rod Drive (CRD) Insert and Withdraw Lines

Criteria 55 concerns these lines of the reactor coolant pressure boundary penetrating the primary reactor containment The CRD insert and withdraw lines are not part of the reactor coolant pressure boundary The classification of the insert and withdraw lines is Quality Group B, and therefore, designed in accordance with ASME Section III, Class 2 The basis to which the CRD lines are designed is commensurate with the safety importance of isolating these lines Since these lines are vital to the SCRAM function, their operability is of utmost concern

In the design of this system, it has been accepted practice to omit automatic valves for isolation purposes as this introduces a possible failure mechanism As a means of providing positive actuation, manual shutoff valves are used In the event of a break on these lines, the manual valves may be closed to ensure isolation In addition, a ball check valve located in the insert line inside the CRD is designed to automatically seal this line in the event of a break

21 This MO stop check valve is normally in a closed position due to its check valve feature, but its MO is in the open position The MO provides a backup to close the valve to provide additional high leak tight integrity.

22 Maximum isolation time = Nominal time plus 10%

23 Maximum isolation time = Nominal time plus 20%

24. Via limit switch adjustment

TABLE 4.4-1 (CONT)

NOTES (CONT)

25 Purge air valves operate only during shutdown and refueling modes

26 Abbreviations used in table

- AO - Air Operated
- MO - Motor-Operated
- VTC - Pneumatic Testable Check Valve
- RHR - Residual Heat Removal System
- RPV - Reactor Pressure Vessel
- RCIC - Reactor Core Isolation Cooling System
- RWCU - Reactor Water Cleanup
- HPCI - High Pressure Coolant Injection
- GDC - General Design Criterion
- RBCLCW - Reactor Building Closed Loop Cooling
- TIP - Transversing In core Probe
- CRD - Control Rod Drive
- MSIV - Main Steam Isolation Valve
- SO - Solenoid Operated
- PASS - Post Accident Sampling System
- RV - Relief Valve

ISOLATION SIGNAL CODES

Signal	Description
A*	Reactor vessel low water level 3 (A SCRAM will occur at this level)
B*	Reactor vessel low water level 2 (The reactor core isolation cooling system and the high pressure coolant injection system will be initiated at this level, and recirculation pumps are tripped)
C*	High radiation - main steam line
D*	Line break - main steam line (high steam flow)
E*	Line break - main steam line (steam line tunnel high temperature or high differential temperature)
F*	High dry well pressure
G	Reactor vessel low water level 1 (The core spray systems and the low pressure core injection mode of RHR systems will be initiated at this level)
J*	Line break in reactor water cleanup system - high space temperature, high differential flow
K*	Line break in steam line to/from turbine (high steam line space temperature, high steam flow, low steam line pressure or high turbine exhaust diaphragm pressure)
L	Reactor building standby ventilation system initiation (Note 17)
M	High radiation signal refueling level exhaust
N	High reactor building pressure
P*	Low main steam line pressure at inlet to turbine (RUN mode only)
R	Low condenser vacuum
S	High radiation signal in the 6 inch vent/purge duct
T	High temperature in turbine building
U	High reactor vessel pressure
W	High temperature at outlet of cleanup system nonregenerative heat exchanger
X	Low steam pressure
Y	Standby liquid control system actuated
Z	Low level in RBCLCW head tank
A1	Reactor vessel low water level 3 when in shutdown cooling mode only
RM*	Remote manual switch from main control room
T1	Exclusion area high temperature

* These are the isolation functions of the primary containment and reactor vessel isolation control system, other functions are given for information only

TABLE 4.4-2 NSSSS VALVE GROUPS

VALVE GROUPS	TYPE ISOLATION	VALVE NUMBER
1	Main Steam	1B21*AOV-081A,B,C,D
1		1B21*AOV-082A,B,C,D
2	Containment Drywell	1D11*MOV-032A,B
2	Radiation Monitoring Subsystem	1D11*MOV-033A,B
	RHR - Containment Spray Drywell	1E11 MOV-038A,B
		1E11 MOV-039A,B
	RHR - Containment	
2	Spray Suppression Chamber	1E11*MOV-041A,B
2	RHR Test Line Return to	1E11*MOV-042A,B
2	Suppression Chamber	1E11*MOV-040A,B
2	RHR Seam Condensing Discharge	1E11*MOV-044A,B
2	Core Spray Test Line	1E21*MOV-035A,B
2	Instrument Air to Suppression	1P50*MOV-104
2	Chamber Outer Isolation Valve	1P50*MOV-106
3	RWCU Line from RPV	1G33*MOV-033
4	RWCU Line from RPV	1G33*MOV-034
5	RHR Shutdown from RPV	1E11*MOV-047
5		1E11*MOV-048
5	RHR - Head Spray Line to RPV	1E11*MOV-053
5		1E11IMOV-054
5	RHR Injection Line to Recirculation System Return	1E11*MOV-081A,B
6	RCIC Turbine Steam Inlet Line	1E51*MOV-041
6		1E51*MOV-047
7	HPCI Turbine Steam Inlet Line	1E41*MOV-041
7		1E41*MOV-047

TABLE 4.4-2 (CONT)

VALVE GROUPS	TYPE ISOLATION	VALVE NUMBER
8	RBCLCW Primary Containment Isolation	1P42*MOV-147
8		1P42*MOV-148
8	RBCLCW from Drywell Unit Coolers	1P42*MOV-231
8	RBCLCW to Drywell Unit Coolers	1P42*MOV-232
8		1P42*MOV-233
8		1P42*MOV-234
8		1P42*MOV-235
8	RBCLCW from Drywell Unit Cooler	1P42*MOV-236
8	RBCLCW to Drywell Unit Coolers	1P42*MOV-237
8		1P42*MOV-238
8		1P42*MOV-239
8		1P42*MOV-240
9	Drywell Inerting (4" valve)	1T24*AOV-001A,B
9	Suppression Chamber Inerting (4" valve)	1T24*AOV-004A,B
9	Purge Air to Drywell (18" valve)	1T46*AOV-038A,B
9	Purge Air to Suppression Chamber (18" valve)	1T46*AOV-038C,D
9	Purge Air from Drywell (18" valve)	1T46*AOV-039A,B
9	Purge Air from Suppression Chamber (18" valve)	1T46*AOV-039C,D
9	Vent Line - Drywell (6" valve)	1T46*AOV-078A,B
9	Vent Line - Suppression Chamber (6" valve)	1T46*AOV-079A,B
10	Sample Coolant from RPV	1B31*AOV-081
10		1B31*AOV-082
11	N_2 Purge for TIP	1C51*SOV-028
11	Floor Drains from Drywell	1G11*MOV-246
11		1G11*MOV-247
11	Equipment Drains from Drywell	1G11*MOV-248
11		1G11*MOV-249
11	Suppression Pool Pumpback	1G11*MOV-639C
11	Suppression Pool Cleanup Return	1G41*MOV-033A,B
11	Suppression Pool Cleanup Pumpdown	1G41*MOV-034A,B

TABLE 4.4-2 (CONT)

VALVE GROUPS	TYPE ISOLATION	VALVE NUMBER
12	RCIC Vacuum Breaker	1E51*MOV-049
13	HPCI Vacuum Breaker	1E41*MOV-049
14	Main Steam Line Drains	1B21*MOV-031
14		1B21*MOV-032
14		1B21*MOV-061
14		1B21*MOV-062
14		1B21*MOV-063
14		1B21*MOV-064
15	RCIC Turbine Steam Inlet Line	1E51*MOV-042
15		1E51*MOV-048
16	HPCI Pump Suction	1E41*MOV-032
16	HPCI Turbine Steam Inlet Line	1E41*MOV-042
16		1E41*MOV-048
17	PASS Reactor Sample	1B21*SOV-313A,B
17	PASS Sample Return	1E11*SOV-168
17		1E11*SOV-169
17	PASS Drywell Atm Sample	1T48*SOV-126A,B
17	PASS Suppression Chamber Atm Sample	1T48*SOV-127A,B
17	PASS Drywell Atm Sample	1T48*SOV-128A,B
17	PASS Suppression Chamber Atm Sample	1T48*SOV-129A,B
17	PASS Atm Sample Return	1T48*SOV-130
17	PASS Primary Containment Sample Return	1T48*SOV-131

TABLE 4.4-3 (CONT)

		VALVE GROUP(S)	TRIP SETPOINT
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION		
a.	Flow - High		
b.	Heat Exchanger/Pump	3, 4	≤ 44 gpm 45 sec TD
	Area Temperature - High		
c.	SLCS Initiation	3, 4	≤ 155°F
d.	Reactor Vessel Water	4	N/A
	Level - Low Low, Level 2		
e.	Manual Initiation	3, 4	≥ -38 inches
		3, 4	N/A
4.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a.	RCIC Steam Line Flow - High	6, 15	< 291 inches H_2O 3 sec TD
b.	RCIC Steam Supply Pressure - Low	6, 12(e), 15	≥ 15 psig
c.	RCIC Turbine Exhaust Diaphragm Pressure - High	6, 15	≤ 10 psig
d.	RCIC Equipment Area Temperature - High	6, 15	≤ 155°F
e.	RCIC Elevation 63' Area Temperature - High	6, 15	≤ 193°F
f.	Drywell Pressure - High	12(e)	≤ 1.69 psig
g.	Manual Initiation	15(f)	N/A
5.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a.	HPCI Steam Line Flow - High	7, 16	≤ 212 inches H_2O 3 sec TD
b.	HPCI Steam Supply Pressure - Low	7, 13(e), 16	≥ 100 psig
c.	HPCI Turbine Exhaust Diaphragm Pressure - High	7, 16	≤ 10 psig
d.	HPCI Equipment Area Temperature - High	7, 16	≤ 155°F**
e.	HPCI Elevation 63' Area Temperature - High	7, 16	≤ 193°F**
f.	Drywell Pressure - High	13(e)	≤ 1.69 psig
g.	Manual Initiation	16	N/A

TABLE 4.4-3 (CONT)

		VALVE GROUP(S)	TRIP SETPOINT
6.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
	a. Reactor Vessel Water Level - Low, Level	5	≥ 12.5 inches
	b. Reactor Recirculation Suction Pressure - High (RHR Cut-in)	5	≤ 125 psig
	c. Manual Initiation	5	N/A
7.	REACTOR BUILDING CLOSED LOOP COOLING WATER SYSTEM/RHR CONTAINMENT SPRAY ISOLATION		
	a. Drywell Pressure - High	2, 8	≥ 1.69 psig
	b. Reactor Vessel Water Level - Low Low Low Level 1	2, 8	≥ 132.5 inches
	c. RBCLCW Head Tank Level - Low Low	8	≥ 5 ft. 9 in.
	d. Manual Initiation	2(g), 8(g)	N/A
8.	POST ACCIDENT SAMPLING SYSTEM ISOLATION		
	a. Reactor Vessel Water Level - Low Level 3	17	≥ 12.5 inches
	b. Drywell Pressure - High	17(b)(c)(d)	≤ 1.69 psig
	c. Manual Initiation	17	N/A

NOTES

(b) Also actuates the RBSVS.
(c) Also actuates the control room emergency filtration system in the pressurization mode of operation.
(d) Also actuates reactor building normal ventilation system isolation valves.
(e) Requires RCIC/HPCI system steam supply pressure-low coincident with drywell pressure-high.
(f) Manual initiation isolates outboard steam supply valve and bypass valve only and only with a coincident reactor vessel water level-low low, level 2.
(g) Individual valve controls.

TABLE 4.4-3 (CONT)

	VALVE GROUP(S)	TRIP SETPOINT
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low , Level	5	≥ 12.5 inches
b. Reactor Recirculation Suction Pressure - High (RHR Cut-in)	5	≤ 125 psig
c. Manual Initiation	5	N/A

		TRIP SETPOINT
7. REACTOR BUILDING CLOSED LOOP COOLING WATER SYSTEM/RHR CONTAINMENT SPRAY ISOLATION		
a. Drywell Pressure - High	2, 8	≥ 1.69 psig
b. Reactor Vessel Water Level - Low Low Low Level 1	2, 8	≥ 132.5 inches
c. RBCLCW Head Tank Level - Low Low	8	≥ 5 ft. 9 in.
d. Manual Initiation	2(g), 8(g)	N/A

8. POST ACCIDENT SAMPLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low Level 3	17	≥ 12.5 inches
b. Drywell Pressure - High	17(b)(c)(d)	≤ 1.69 psig
c. Manual Initiation	17	N/A

NOTES

(b) Also actuates the RBSVS.
(c) Also actuates the control room emergency filtration system in the pressurization mode of operation.
(d) Also actuates reactor building normal ventilation system isolation valves.
(e) Requires RCIC/HPCI system steam supply pressure-low coincident with drywell pressure-high.
(f) Manual initiation isolates outboard steam supply valve and bypass valve only and only with a coincident reactor vessel water level-low low, level 2.
(g) Individual valve controls.

TABLE 4.4-4 NSSSS MANUAL ISOLATION

* MANUAL PUSHBUTTON COMBINATIONS	RESULTANT ISOLATIONS
A	Nothing
B	All inboard valves close (except MSIVs)
C	Nothing
D	All outboard valves close (except MSIVs)
A or C and B or D	- All MSIVs close - All inboard or outboard valves close (depending whether PB 'B' or 'D' was depressed)
A and C	Nothing
B and D	All inboard and outboard valves (except MSIVs)

* Assumes the isolation reset pushbuttons are depressed prior to proceeding to the next pushbutton or combination of pushbuttons.

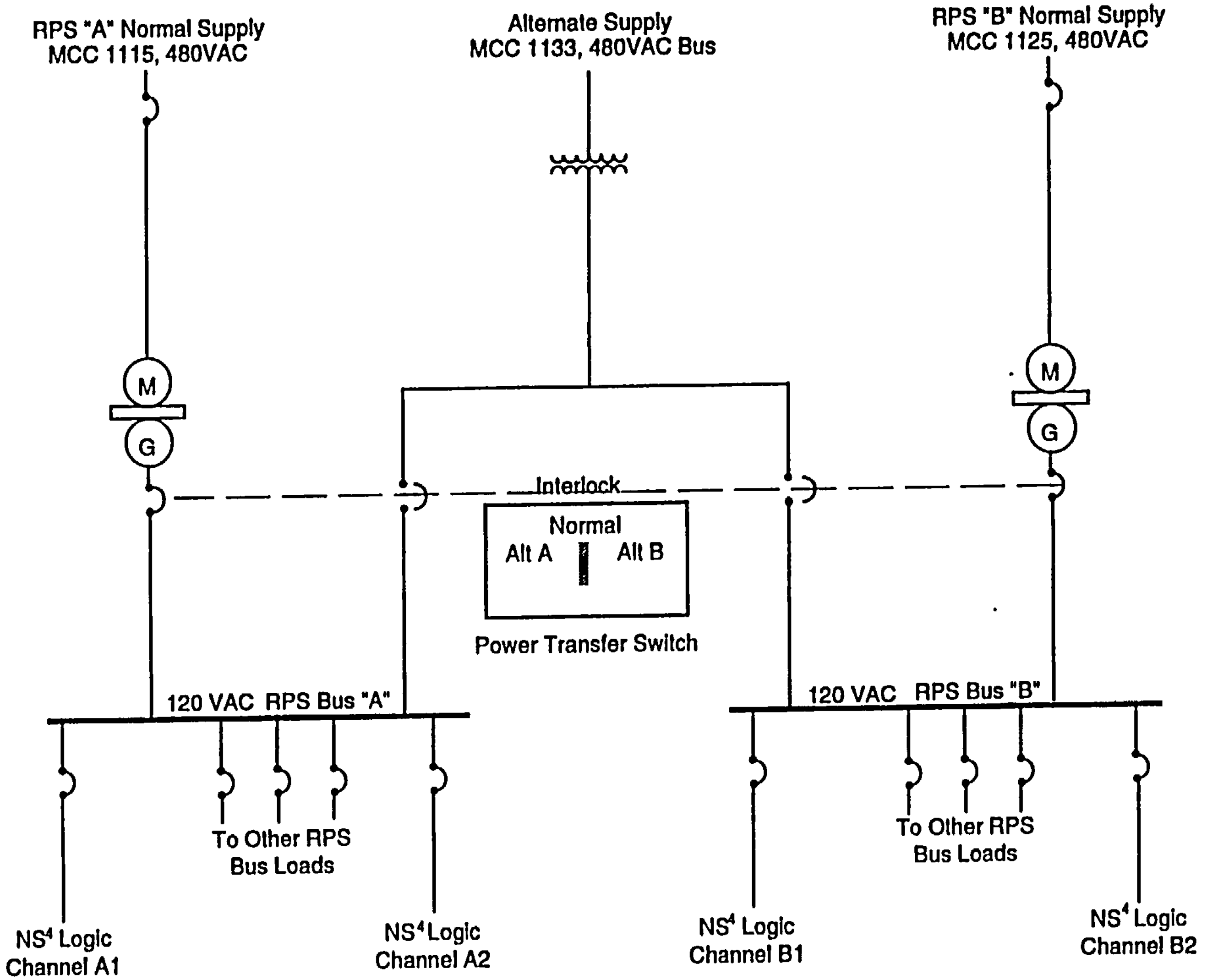

Figure 4.4-1 Nuclear Steam Supply Shutoff System Power Supply

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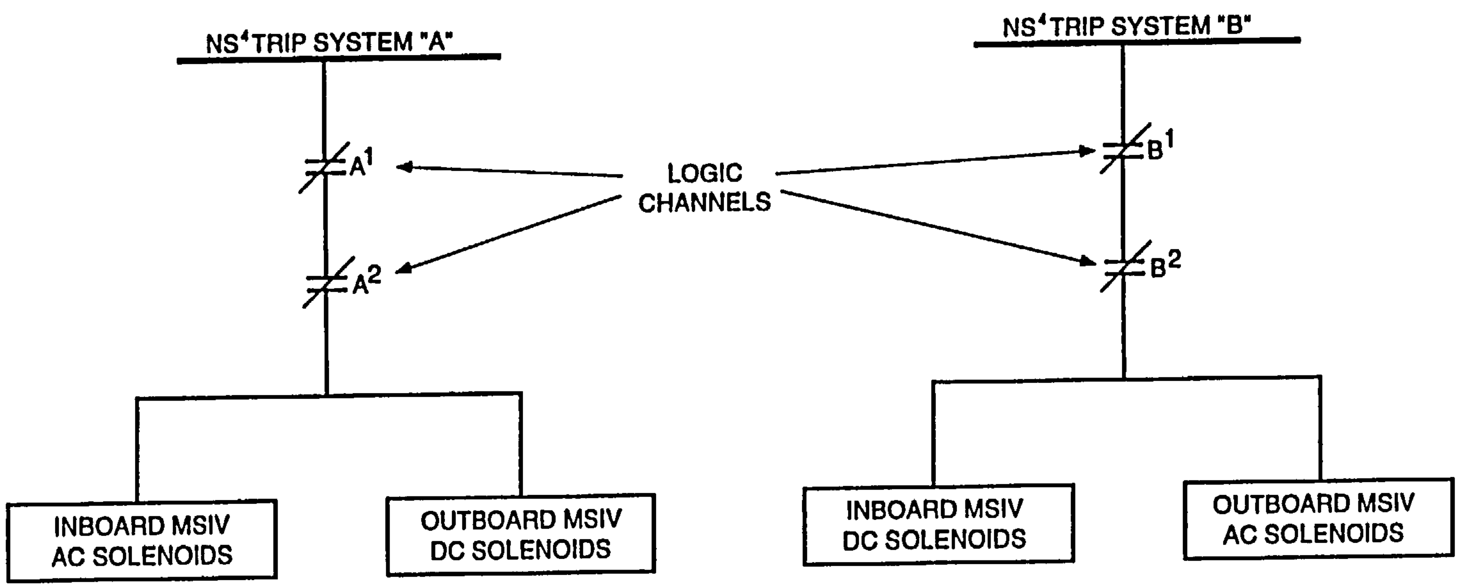

Figure 4.4-2 MSIV Isolation Logic

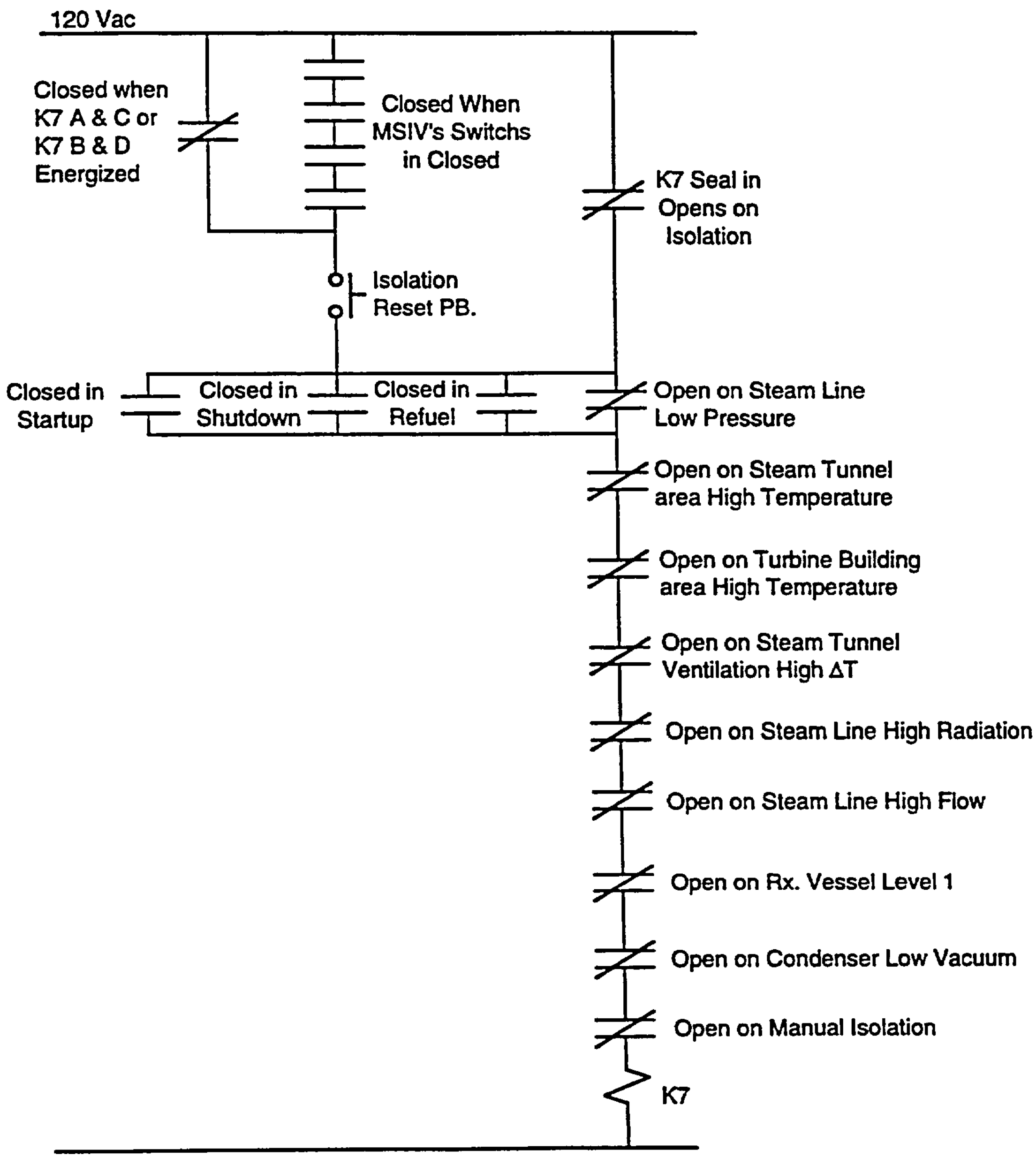

Figure 4.4-3 Typical MSIV Isolation Logic Channels A through D

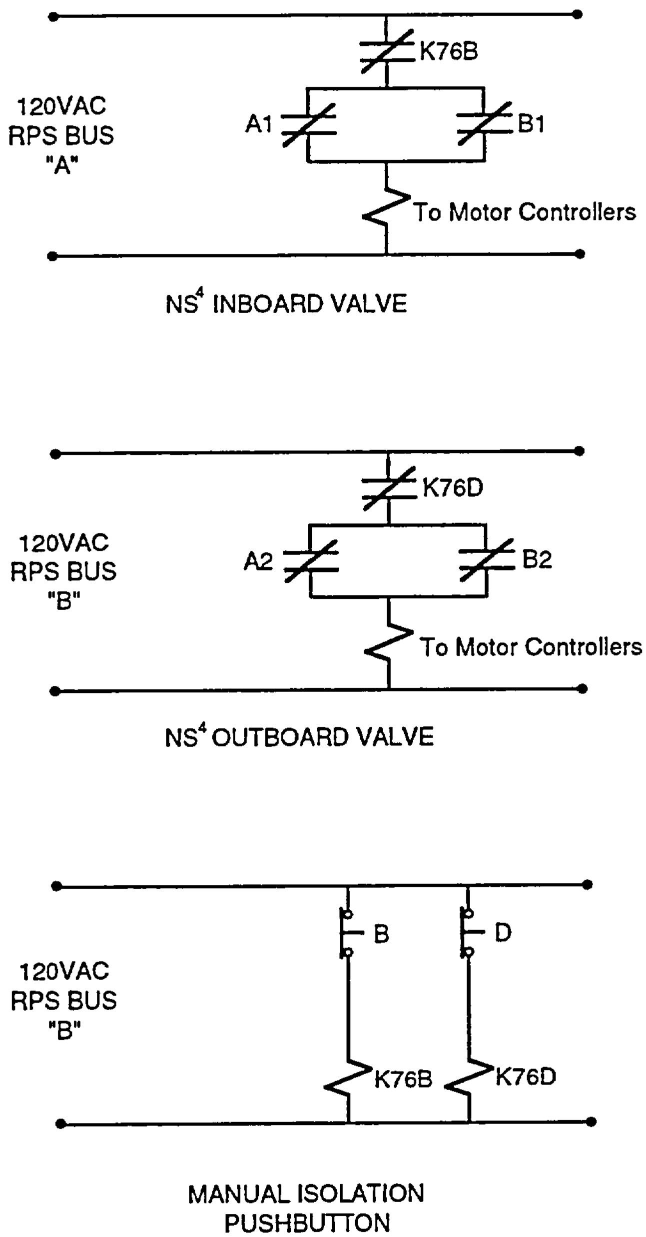

Figure 4.4-4 NS^4 Isolation Logic (Except MSIV's)

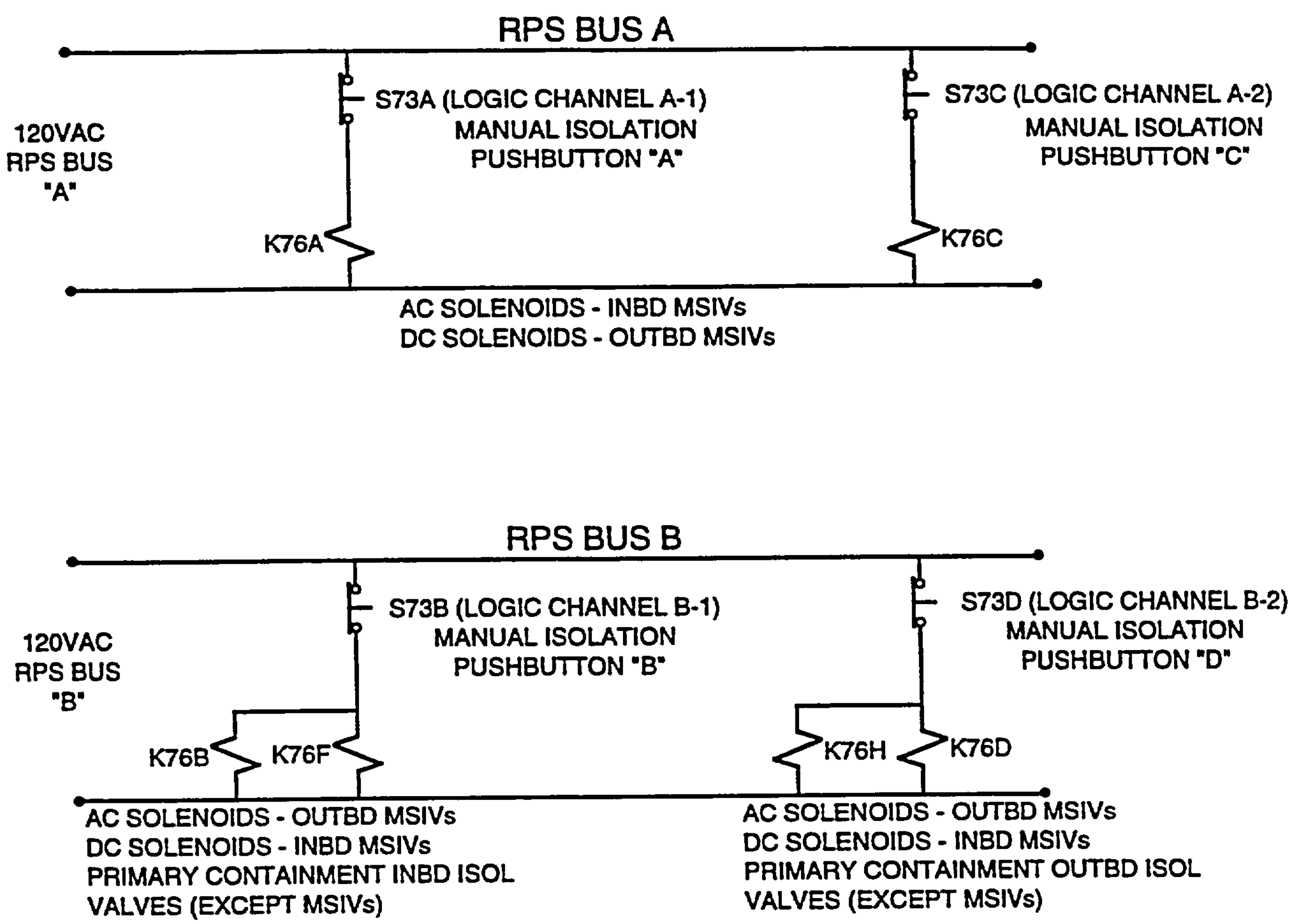

Figure 4.4-5 NS^4 Manual Isolation Logic.

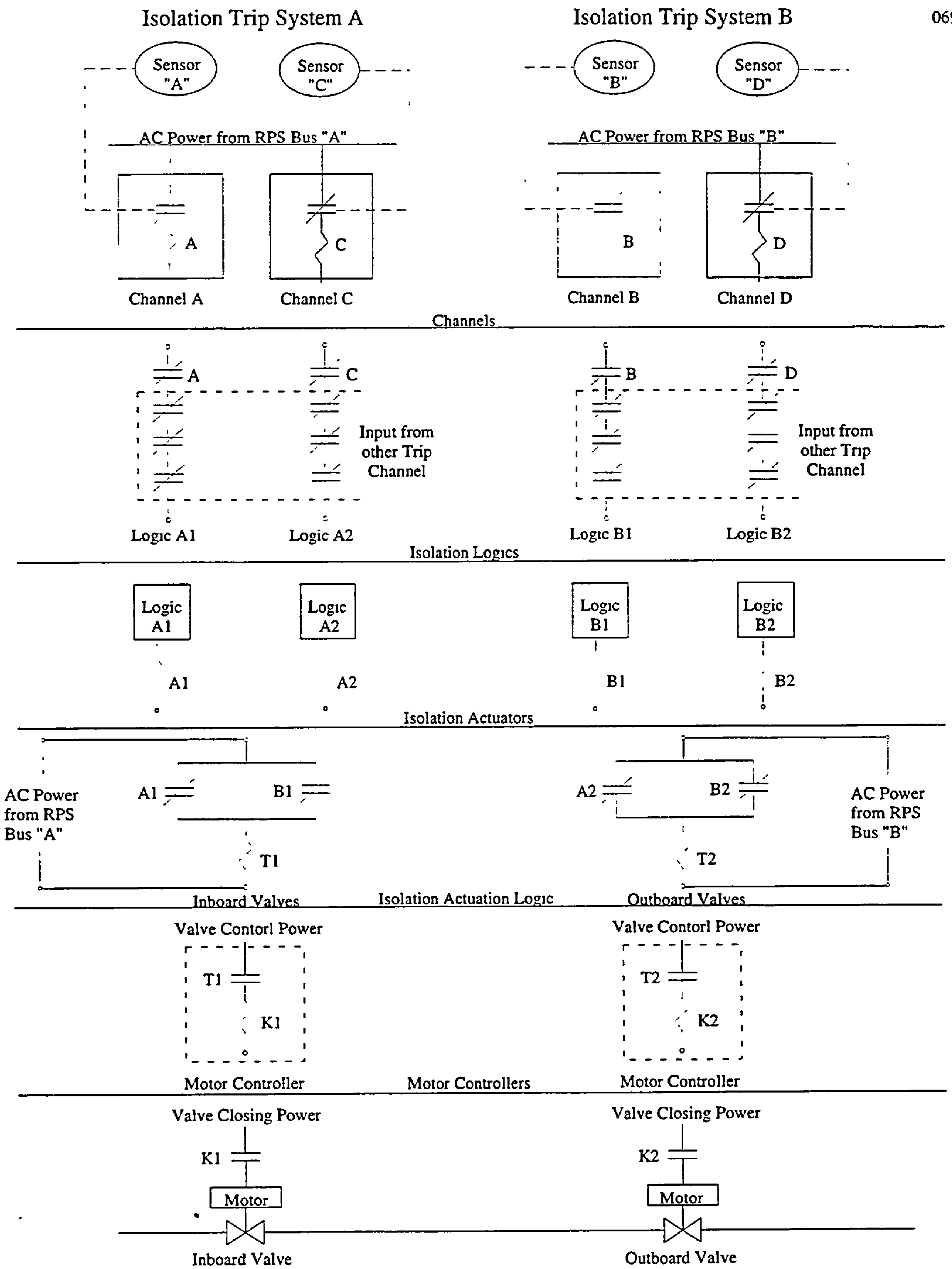

Figure 4.4-6 Isolation Control for Group 2-17

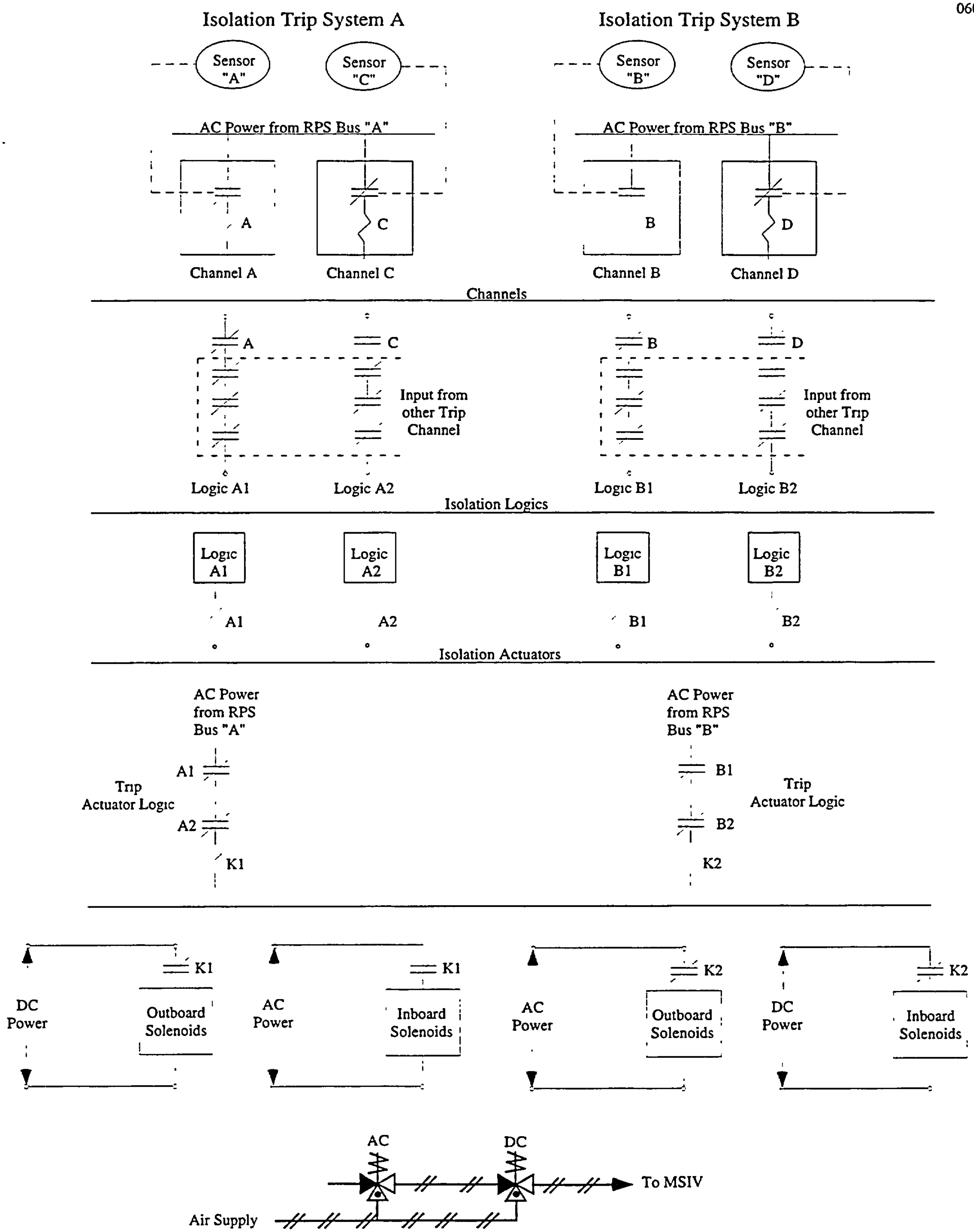

Figure 4.4-7 MSIV Group 1 Logic

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Chapter 5.0

Neutron Monitoring System

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5.0 NEUTRON MONITORING SYSTEM

Lesson Objectives:

1. State the system's purposes.
2. List the neutron monitoring systems and:
   a. State the operational condition (shutdown, startup, heatup, or power operation) during which each system is used.
   b. Describe the arrangement of the neutron monitoring systems' detectors within the reactor core.
   c. Explain the method used by the neutron monitoring systems' detectors to monitor neutron flux, and the measures employed to prolong useful detector life (information in later sections).
   d. Explain the interfaces among the systems used for power range operation.

The purposes of the Neutron Monitoring System (NMS) are to monitor reactor core neutron flux and provide indication during all modes of reactor operation, and to provide trip signals to the Reactor Protection System (RPS) and the Reactor Manual Control System (RMCS).

The safety objective of the Neutron Monitoring System 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 System 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 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 NMS individual 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 and 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 calculates neutron flux for display and initiation of rod withdraw blocks from shutdown conditions to where the neutron flux overlaps the intermediate range.

The SRM System consists of four separate and independent channels located at different radial incore locations, as shown in Figure 5.0-2. Each channel consists of a miniature fission chamber detector, a detector drive assembly and the necessary electronic signal conditioning equipment to process the fission detectors output for display and trip functions.

5.0.2 Intermediate Range Monitoring System (Section 5.2)

The Intermediate Range Monitoring (IRM) System provides neutron flux information from the upper portion of the source range to the lower portion of the power range. In addition to monitoring neutron flux, the IRM System provides trip signals to preserve the integrity of the fuel cladding.

The IRM System consists of eight separate and independent channels. Each channel consists of a miniature fission chamber detector, a detector drive assembly and the necessary electronic signal conditioning equipment to process the fission detector output for display and trip functions. The eight IRM detectors are located in the reactor core region at different radial locations as shown in Figure 5.0-2.

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 Process Computer, Average Power Range Monitoring and Rod Block Monitoring Systems.

The LPRM System (Figure 5.0-5) consists of 124 stationary incore fission chamber detectors and electronic signal conditioning equipment. The LPRM detectors are arranged in 31 radially located assemblies, with each assembly containing four detectors spaced at three foot intervals. This type of detector arrangement provides uniform coverage of both radial and axial core flux distribution.

5.0.4 Average Power Range Monitoring System (Section 5.4)

The Average Power Range Monitoring (APRM) System calculates core average neutron flux for various display and functions to preserve the integrity of the fuel cladding.

The APRM System is composed of six APRM channels (A through F). Each channel receives signals from two systems for calculations: the LPRM detectors provide local neutron flux signals for averaging, and the recirculation system provides recirculation loop flows for flux bias scram and rod block settings (Figure 5.0-5).

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 percent 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 six APRM channels with 17 or 14 LPRM detector inputs.

5.0.5 Rod Block Monitoring (Section 5.5)

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

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

5.0.6 Traversing Incore Probe System (Section 5.6)

The Traversing Incore Probe (TIP) System provides a means of measuring axial neutron flux over 31 fixed locations in the core. The measured axial neutron flux is used to calibrate the Local Power Range Monitor detectors and to update the process computer.

The TIP System, Figure 5.0-5, 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 operated drive mechanism. Operation of the drive mechanism causes the fission chamber to be inserted or retracted from the reactor core, within a TIP guide tube. The four TIP machines are divided between the 31 LPRM assemblies with one common LPRM assembly connected to all four TIP's for cross calibration. The output signal from the TIP channel may be used to plot an axial flux profile on an X-Y recorder and/or provide flux distribution signal into the process computer for LPRM calibration data.

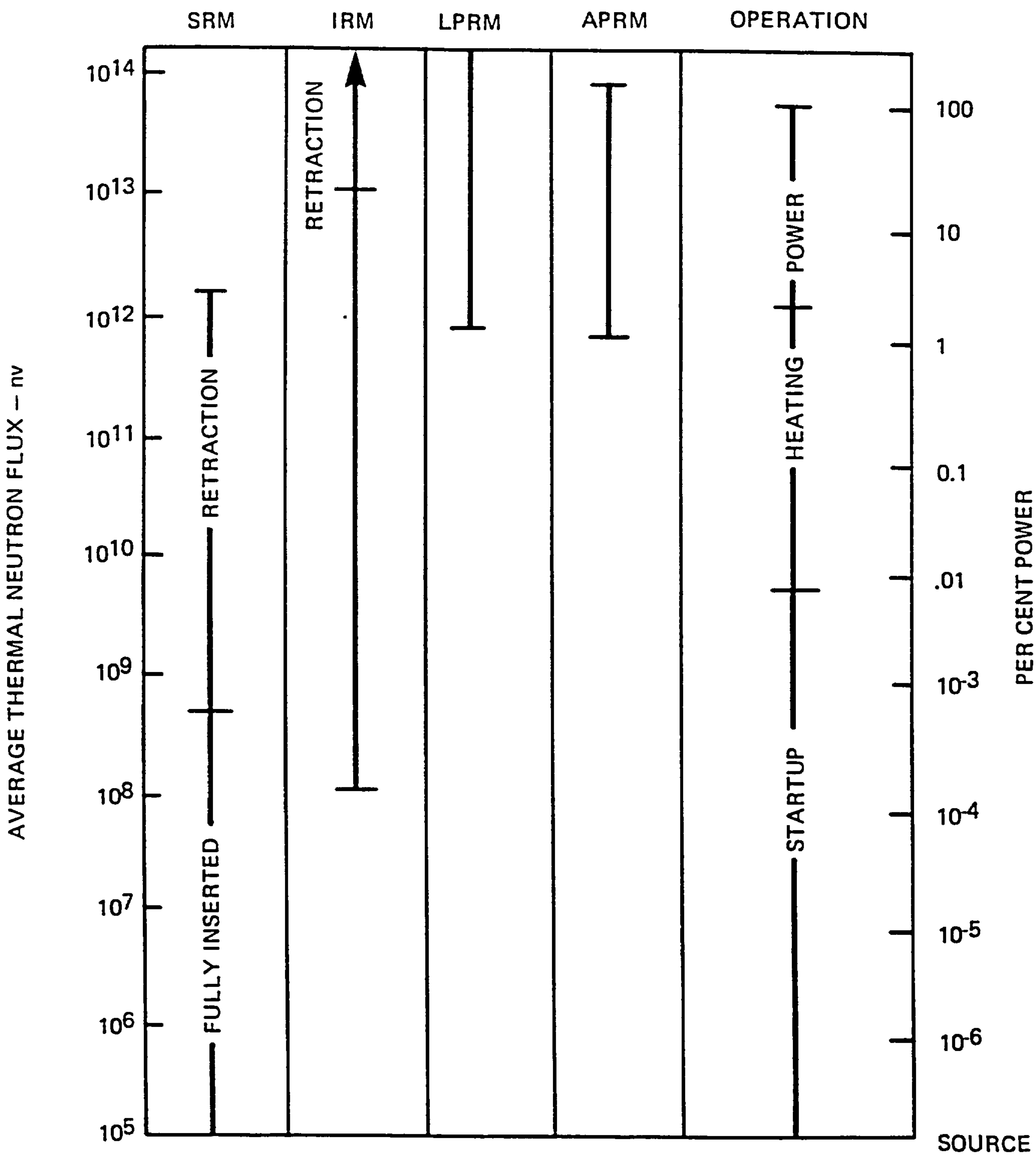


FIGURE 5.0-1 NEUTRON MONITORING SYSTEM RANGES

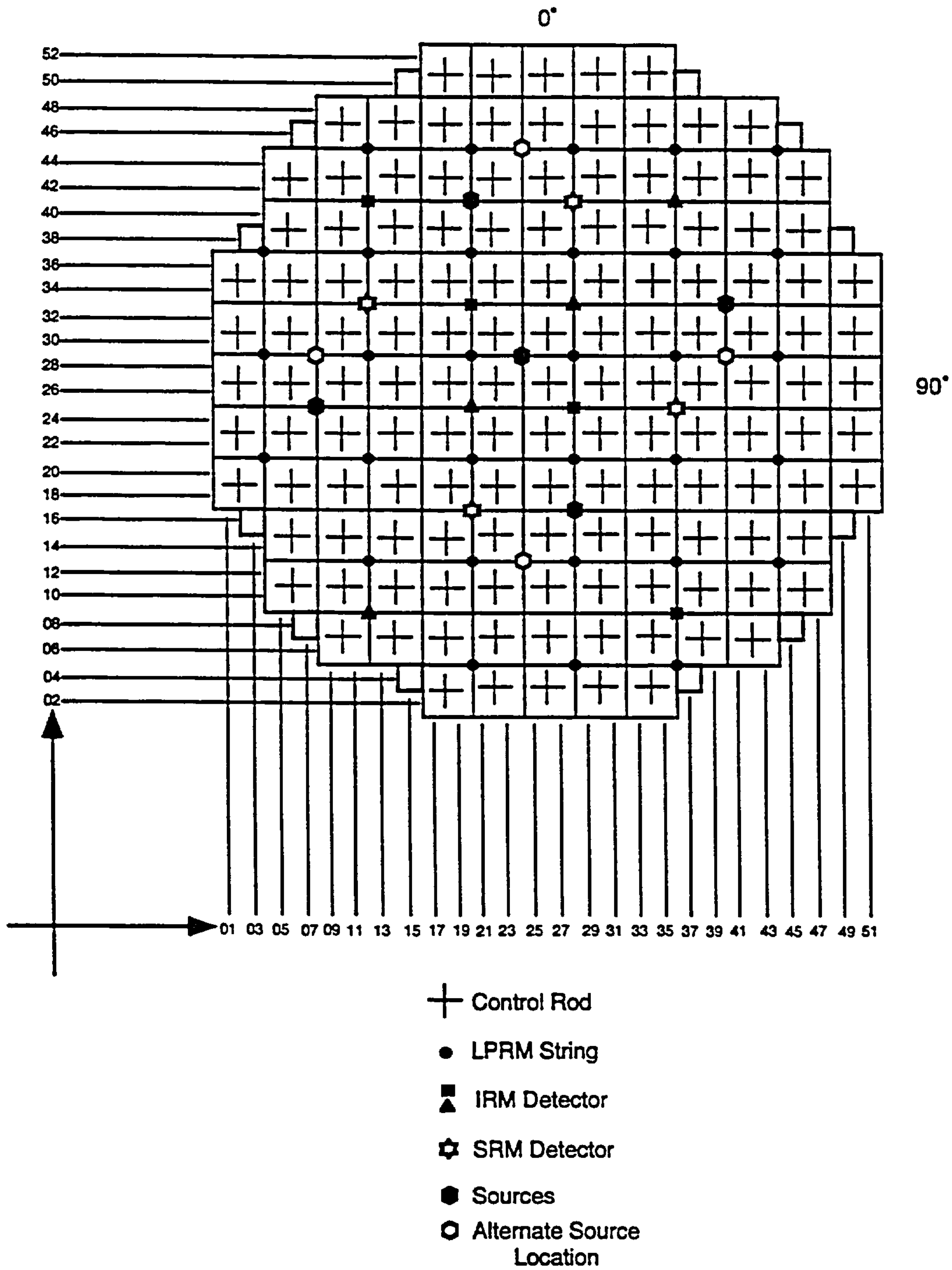


Figure 5.0-2 Detector and Control Element Arrangement

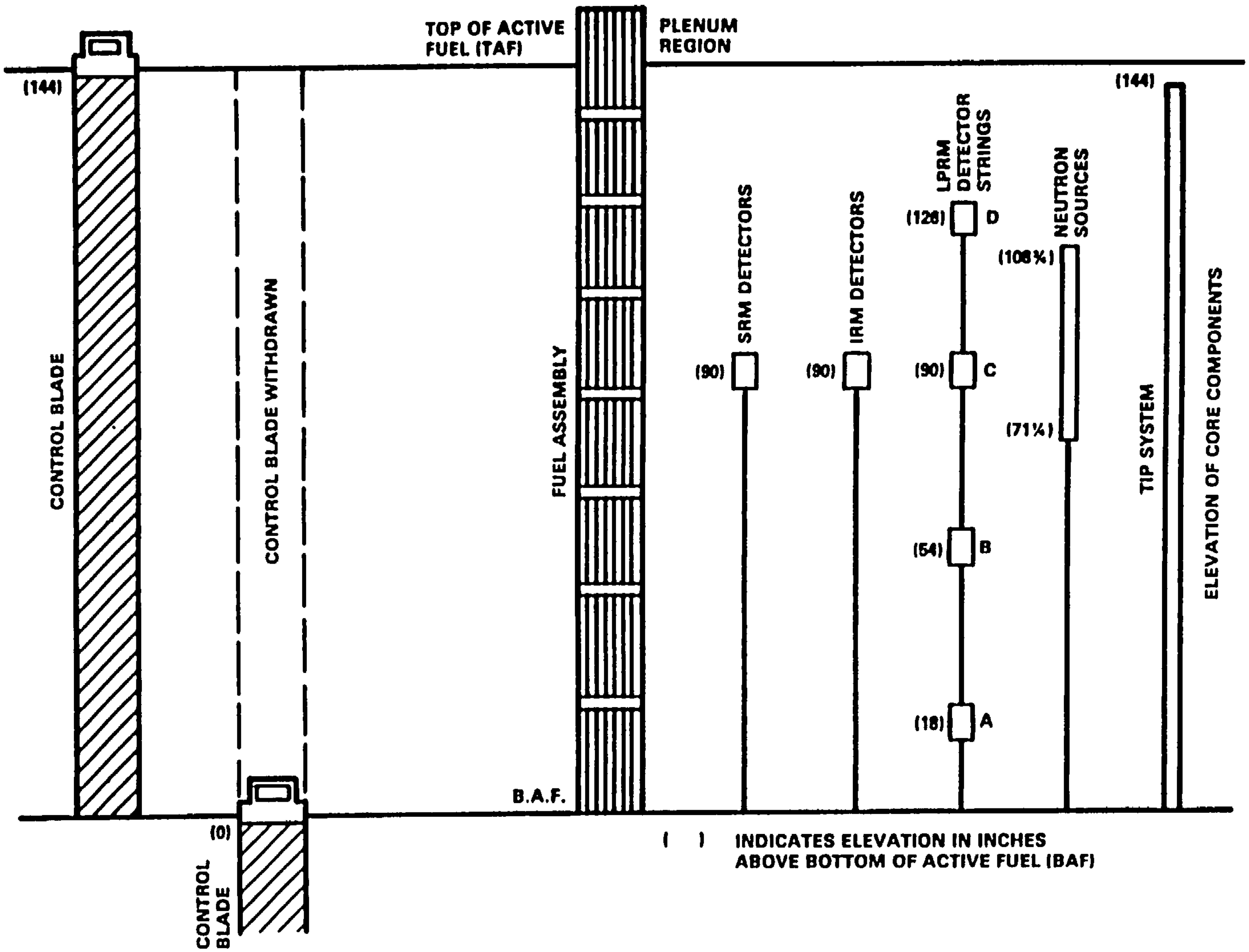
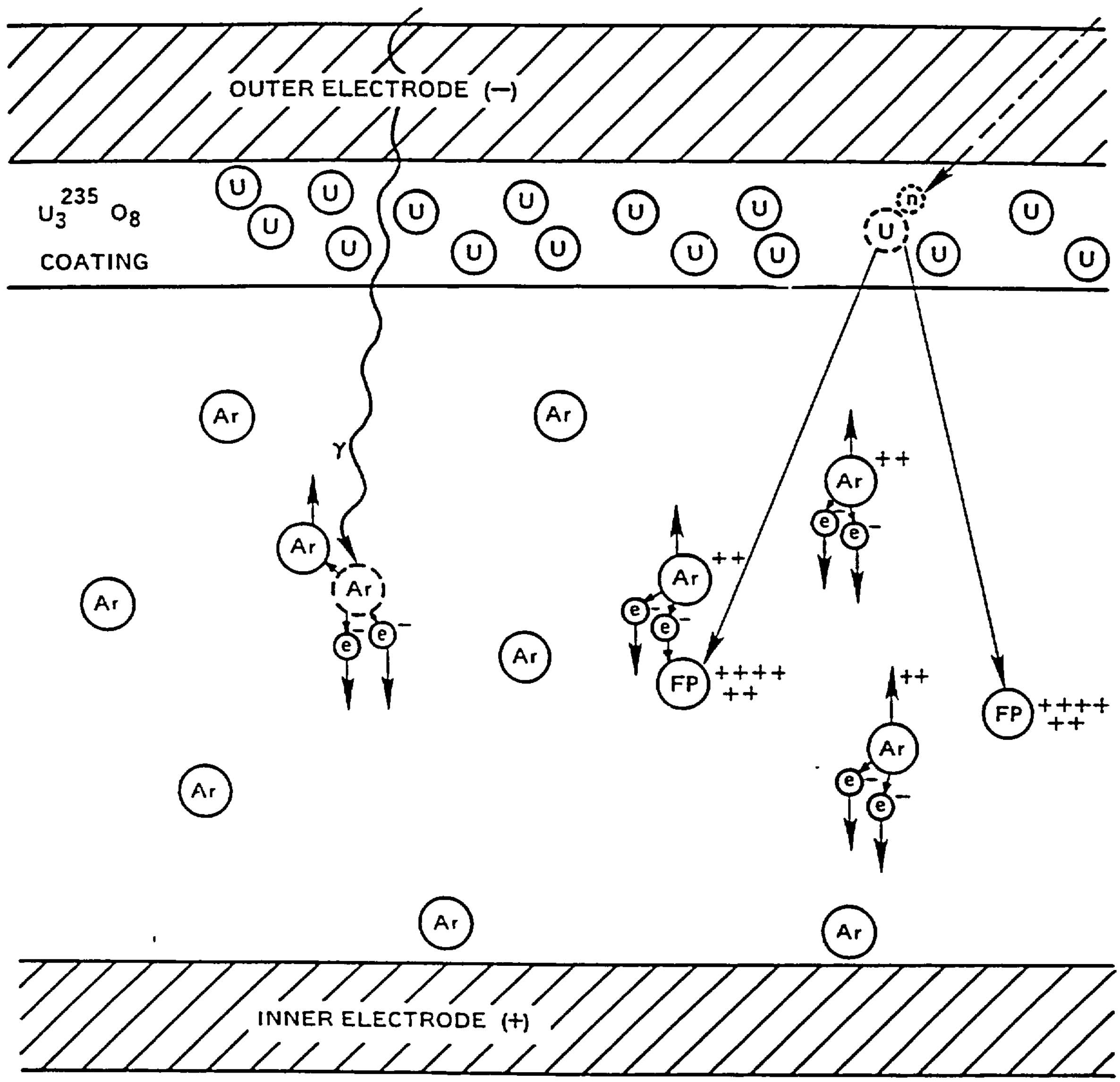


FIGURE 5.0-3 AXIAL LOCATION OF NEUTRON MONITORING SYSTEM COMPONENTS



DETECTOR DATA

90% ENRICHED IN U—235

INTERNAL PRESSURE 215 psi

LENGTH 1.6 INCHES

WIDTH 0.16 INCHES

FIGURE 5.0—4 FISSION CHAMBER OPERATION

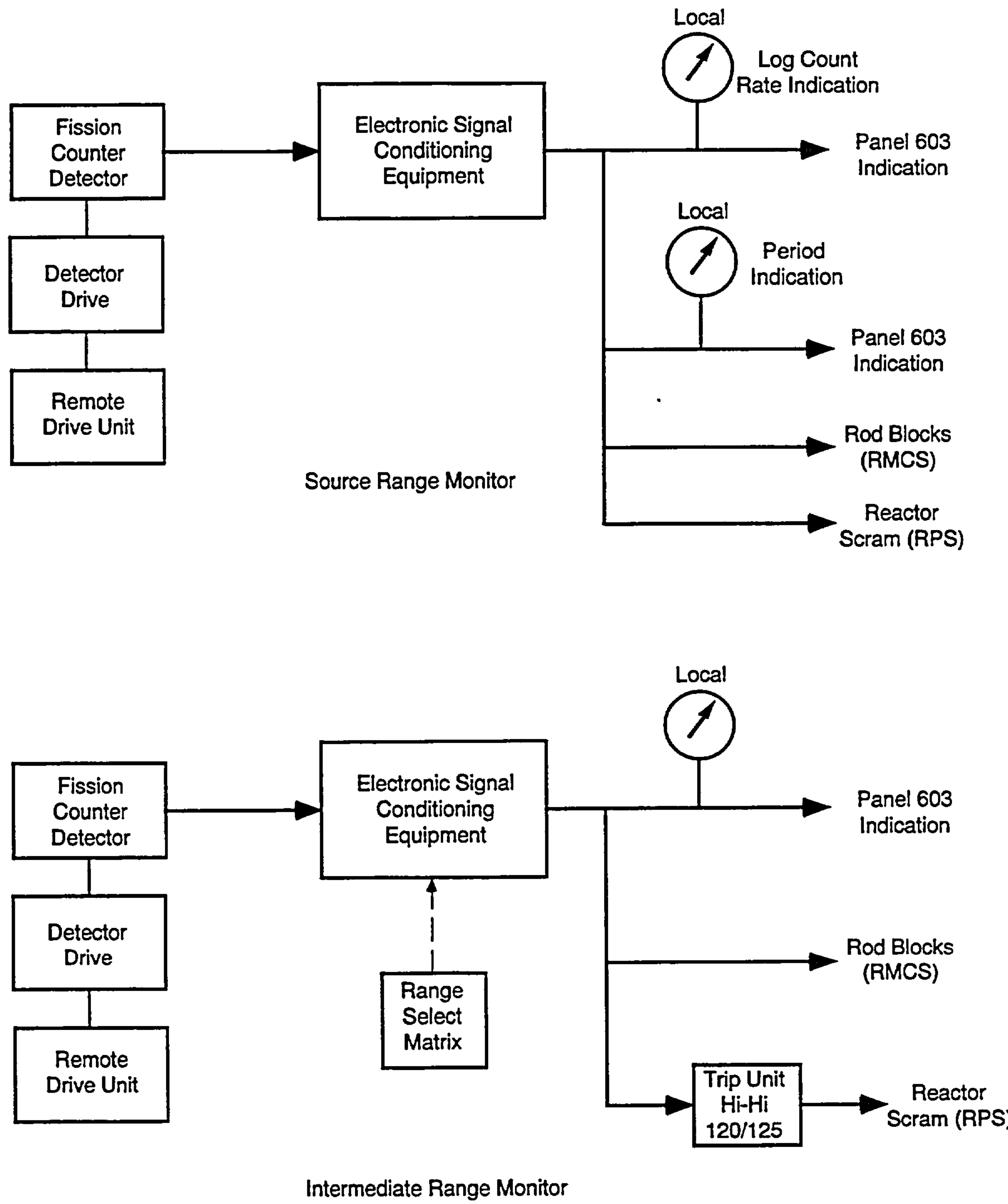


Figure 5.0-5 Shutdown, Startup and Heatup Operations

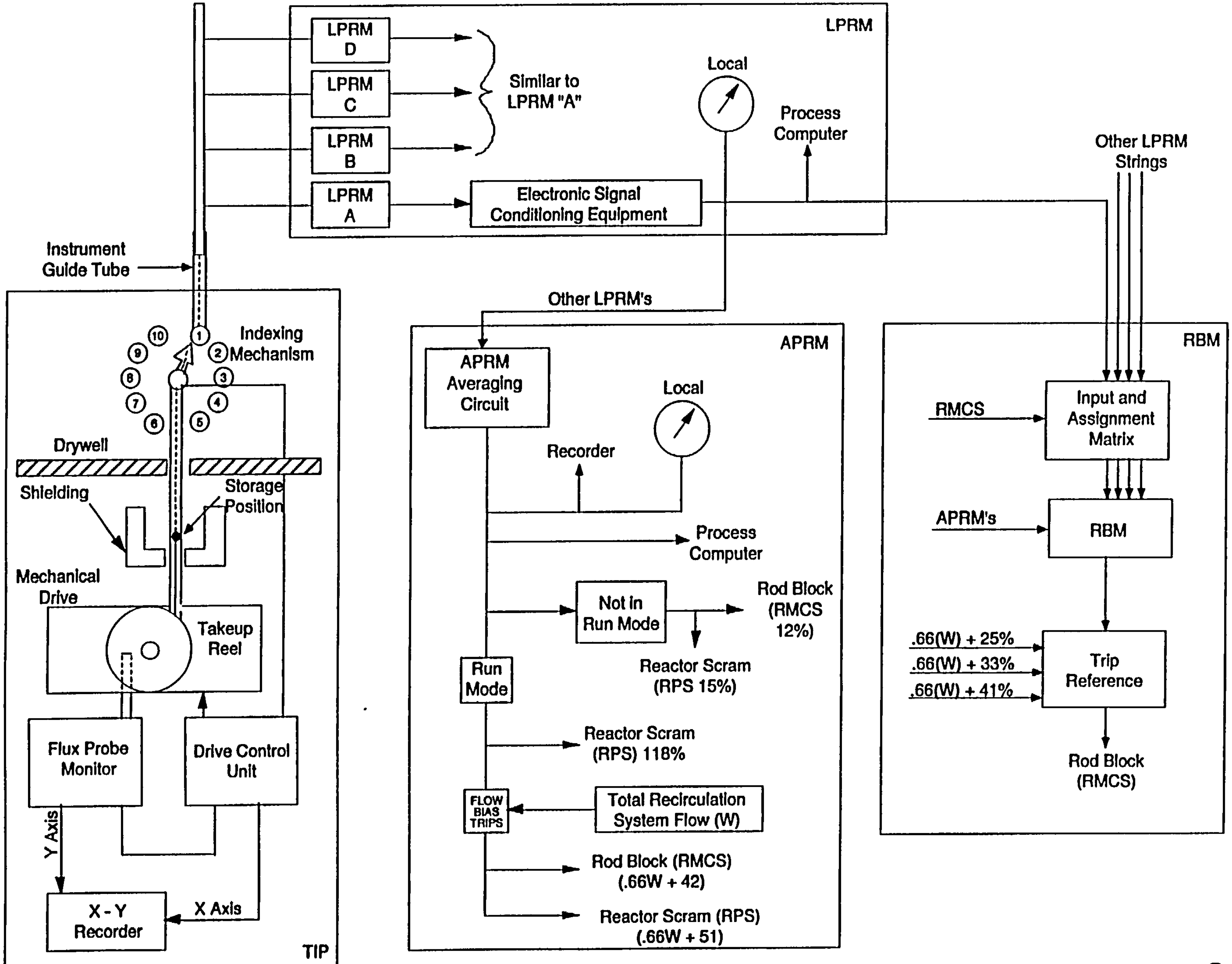


Figure 5.0-6 Power Range Operation

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Chapter 5.1

Source Range Monitor System

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5.1 SOURCE RANGE MONITORING SYSTEM

Learning Objectives:

1. State the system's purpose
2. Describe the indication provided by the system.
3. List and explain each rod block imposed by this system.
4. Explain how this system interfaces with the following plant systems:

   a. Reactor Manual Control System
   b. Reactor Protection System

5.1.1 Introduction

The purpose of the Source Range Monitoring (SRM) System is to monitor neutron flux for display and initiation of rod blocks from shutdown conditions to overlap with the intermediate range. The functional classification of the SRM system is that of a power generation system.

The SRM system consists of four independent channels monitoring and calculating local flux levels within the core region. Each channel consists of a miniature fission chamber detector, a detector drive assembly and the necessary electronic signal conditioning equipment to process the fission detector output for display and trip functions.

The signal conditioning equipment consists of pulse preamplifiers located in the reactor building and gamma discriminators, amplifiers, meters, recorders and trip units located in the control room. The control room meters display reactor power in count rate and reactor period in seconds. The trip units provide rod withdraw block signals to prevent rod out movement unless the proper SRM detector position and count rate signals are present.

5.1.2 Component Description

The major components of the SRM system are shown in Figure 5.1-1 and are discussed in the following paragraphs.

5.1.2.1 Fission Chamber Detector

The neutron flux is detected by four fission chambers (detectors) arranged radially and axially as shown in Figures 5.0-2 and 5.0-3. The detectors provide an output that consists of energy pulses. The output pulse rate is proportional to the neutron flux level, and the pulse energy is determined by the event that causes the pulse.

The SRM fission chambers are approximately one inch in length and 0.16 inches in diameter. The case and collector are fabricated from titanium and insulated from one another by a nonconducting material called forsterite. The inner surface of the case is coated with highly enriched (90% U^{235}) uranium oxide (U_3O_8) to a thickness of several mils. Total weight of U_3O_8 in the tube is 3.31 milligrams of which 2.72 mg. is U^{235}. The inner volume is pressurized with argon, an inert noble gas, to about 220 psia.

5.1.2.2 Detector Insert and Retract Mechanism

The detector insert and retract mechanism is used to position each SRM detector in its dry tube which extends from the bottom of the reactor vessel up into the reactor core. The mechanism allows the detector to be withdrawn to a position

30 inches below the bottom of the core when the neutron flux level has risen above the range covered by the detector. When the detector is being used during the initial stages of reactor startup, it is fully inserted to a position 18 inches above the core midplane. As the reactor power level 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 reactor operation in the power range. 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 determine the position and control the insertion and retraction of each detector. The detector is movable through 114 inches of travel, at a speed of 36 inches per minute.

The driving components of the insert and retract mechanism, shown in Figure 5.1-5, consist of a motor module, a flexible drive shaft, and a detector drive assembly. The driving power is supplied by an electric motor located in the motor module. The motor module output shaft is coupled to the flexible drive shaft which transmits the output power from the motor module to the gear box of the detector drive assembly. A sprocket gear, mounted on the gear box output shaft, drives a machined tube called the drive tube.

The insert and retract mechanism is controlled from the reactor control benchboard located in the control room. The detector can be stopped at any position within the full in and full out position, but only full in and full out positions are indicated by lamps at the control panel, Figure 5.1-6.

5.1.2.3 SRM Circuitry

The SRM circuitry (Figure 5.1-1) amplifies the energy pulse signal for transmission to the control room with a high signal to noise ratio, eliminates the gamma pulse while passing the pulse caused by neutron events, conditions the signal to display seven decades of power, and develops rod withdraw block signals to ensure a safe, controlled reactor startup.

Pulse Preamplifier

The output energy pulses of each SRM detector are applied to a pulse preamplifier which provides current amplification and impedance matching to the cable used to connect the preamplifier to the remainder of the SRM circuitry.

The pulse preamplifiers are located immediately outside of the drywell and as close to the detectors as possible. Placement of the pulse preamplifier ensures amplication of the detector energy pulse signal and not cable noise. Without the amplification of the very small detector signal, the cable noise would make the detector signal undetectable. Even with the amplification, arc welding in the vicinity of the cable will cause erroneous displays.

Pulse Height Discriminator

The pulse height discriminator is provided to eliminate the signal pulses from the gamma events while passing the pulses caused by the neutron events. By eliminating the gamma pulses, the pulses to be counted will reflect only the thermal neutron flux in the core. The pulse height discriminator accomplishes this function by means of an adjustable threshold pulse height.

Most fission fragment produced pulses exceed the threshold setting and are passed, while most gamma produced pulses are lower in amplitude and are blocked. The positive pulse output signals from the pulse height discriminator are then applied to the logarithmic integrator.

Logarithmic Integrator

The logarithmic integrator takes the output signal from the pulse height discriminator and produces a DC current output signal which is directly proportional to the common logarithm (Base 10) of the input count rate.

Because of the wide range of neutron flux values encountered during reactor startup operation, it is more convenient for purposes of indication to display the neutron count rate level on a logarithmic scale. Displaying the 7 decades of the source range (10^{-1} cps to 10^{6} cps) on a conventional linear meter would be impractical.

The logarithmic scaling provides indication with an equal amount of deflection for each decade. This indication enables the operator to observe trends in count rate with a fair amount of accuracy. A linear scale with seven decades of information on it would be impossible to read in the lower regions unless the meter indicating count rate was extremely large in size.

Log Count Rate Amplifier

The log count rate amplifier (LCR) provides a DC voltage output suitable for driving local and remote meters. The LCR amplifier output is also applied to the period circuit and to four of the six trip circuits.

Period 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^{3} cps to 1×10^{4} cps. It is reasonable to assume that reactor power has increased by a factor of 10.

The instantaneous time rate of change of power is called the reactor period and is defined as the amount of time required for power to change by a factor of "e" (where "e" is the natural number 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 circuit in the SRM channel which provides a voltage analog for reactor period, is a differentiating period amplifier that receives input from the LCR amp. The output of the period circuit is inversely proportional to the rate of change of the input count rate. The faster the rate of change, the smaller the reactor period will be. The output from the period circuit is directed to local and remote meters which are calibrated in seconds.

If the pulse rate starts increasing, the meter deflects in the clockwise direction indicating a shorter reactor period. Conversely, if the pulse rate decreases, the period meter deflects in the counter clockwise direction indicating a negative period.

Trip Units

The trip units provide alarm and protective functions in the form of rod block and/or scrams when a signal exceeds a preset reference limit. There are three dual trip units in a source range monitor circuit. Each dual trip unit contains two independent trip circuits. One of the circuits trips when the drawer is inoperative, two are downscale log count rate level trips, two are upscale log count rate level trips, and the sixth circuit is a negative input upscale period trip circuit.

Readout Equipment

Each SRM channel is equipped with log count rate and period meters which provide information to the operator at both the reactor control panel (panel 603) and the SRM instrument drawer. In addition, two of the four SRM channels (either A or C and B or D) are recorded as shown in Figure 5.1-1.

Bypass Switch

A four position bypass switch, located on the reactor control panel 603, allows the operator to bypass the trip functions of one of the four SRM channels during operation for maintenance or in case of a failure. The SRM trips are all bypassed at higher ranges of the IRM's and when the mode switch is in run.

5.1.3 System Features and Interfaces

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

5.1.3.1 Detector Operation

Under normal operating conditions, a high voltage (potential) of about 350 VDC is applied across the center electrode (collector) and the case. This potential causes the center electrode (collector) to be positive with respect to the case (Figure 5.1-3). When ionizing radiation enters the cylinder, a finite number of ion pairs are formed in the argon gas. Ions are formed as a result of high energy particles stripping away electrons from the argon atoms. Under the influence of the applied high voltage on the collector and case, the positive ions migrate toward the case (negative electrode) and the negative ions (primarily electrons) migrate toward the center electrode (positive electrode). The result is that some of the electrons and positive ions are collected and are seen in the detector circuitry as small pulses of electrical charges.

The amount of charge and thus the magnitude of the charge pulse is dependent on the voltage applied across the detector. A larger potential across the electrodes results in a large force applied to the ionized gas atoms. This causes them to accelerate toward the electrodes faster than they would at lower applied voltage. If the voltage were very low there would be only a small quantity of charge collected at the electrode. Many of the ion pairs would recombine before they could reach the electrodes. If the detector is exposed to a constant amount of ionizing radiation and the applied voltage is increased, a graph of voltage versus pulse rate can be made (Figure 5.1-4).

The low voltage region in which a large portion of the ion pairs recombine is called the recombination region. With increasing voltage, the ions move more rapidly causing a decrease in the amount of recombination effect and resulting in more charge collected at the center electrode. This region is of little use in nuclear detection instruments.

In the ionization region, if the applied voltage were continually increased, no additional charge would be collected. This is due to the fact that all of the ion pairs formed are being collected. The amount of charge collected is a function of the number of ion pairs formed in the gas by the nuclear radiation. Different types of nuclear radiation have different specific ionization properties. The number of ion pairs formed by a particular type of radiation passing through a medium may be many times greater than those formed by another type of radiation passing through the same medium. This is the reason for the upper and lower curves in Figure 5.1-4. The upper curve is for a certain type of radiation, possessing a higher specific ionization than that which formed the lower curve. This region of the curve is termed the ionization region or the ion chamber region, and has been selected for operation of the source range instrumentation.

Recall that in order to produce the ion pairs, ionizing radiation is required. Neutrons are of primary interest in the SRM; however, since they have a neutral charge (neither + nor -) they have an extremely low probability of ionizing the argon gas directly. In order to detect neutrons, the fission chamber is coated on the inner surface with a uranium oxide compound in the form of U_3O_8. Thermal neutrons entering the detector have finite probability of being absorbed by the U^{235} in the U_3O_8 coating.

Of these neutrons that are absorbed, a percentage cause U^{235} atoms to fission. The result is that two or more high energy, charged fission fragments moving in the argon gas within the detector causing ionization of the gas (Figure 5.1-3). The ions are collected, and a small electrical charge pulse can be observed. Ionization of the argon gas may also be caused by gamma radiation present in the reactor core. The amplitude of the charge-pulse is proportional to the number of ion pairs produced, and is a function of the energy of the ionizing radiation producing the ion pairs. The fission fragments resulting from interaction of neutrons with the U_3O_8 coating cause a significantly larger amount of ionization within the fission chamber than gamma radiation incident on the detector. This results in generated neutron charge pulses being significantly larger than the gamma charge pulses. This characteristic allows for the detection of neutron pulses alone.

When the neutron flux reaches the saturation point of the monitors, the detectors are retracted. The detector retraction increases the detector life and extends the range of the monitor. With the detectors retracted and the reactor operating at normal power levels, the SRM count rate indicates full scale. The period indication shows neutron activity, and any sudden increase or decrease of neutron activity in the reactor is reflected by the SRM period even though the SRM's are not measuring the full neutron activity of the core.

5.1.3.2 Rod Blocks and Scrams

During startup, 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. In addition, the SRM circuitry (when shorting links are removed) provides a high-high and inoperative trip which inputs a scram signal into the Reactor Protection System (RPS).

Any SRM detector not fully inserted into the core with a count level below 100 cps (retract permit level) and any of the range switches of the Intermediate Range Monitoring System on range one or two will produce a rod withdraw block. This assures no control rod is withdrawn unless the SRM detectors are properly inserted in the core to indicate a sufficient neutron flux level for startup.

The downscale rod block of 3 cps assures no control rod is withdrawn unless the count rate is above the minimum prescribed for low neutron flux level monitoring.The upscale rod block of 1 x 10^5 cps assures no control rod is withdrawn without proper neutron flux indications.

The SRM inoperative rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available in that all SRM channels are in service or properly bypassed.

5.1.3.3 SRM Drawers

The SRM drawers, shown in Figure 5.1-7, are located in four segregated metal panels in the control room. Each drawer contains a pulse height discriminator, logarithmic integrator, log count rate amplifier, period circuit, trip units, meters for indicating SRM count rate and reactor period and test switches. The SRM drawer contains three switches which allow for testing and resetting the local trip unit indicating lights. They are the mode switch, the ramp switch, and the reset switch.

The mode switch has nine positions which allow for operating mode changes in an SRM channel for maintenance or calibration. The ramp switch supplies signals for channel period calibration. The reset switch is used to reset the seal in light circuits for the individual trip units and to reset the ramp generator during period calibration. The alarm lights mounted on the front of each SRM drawer illuminate when a preset signal condition is reached and seal in. Even if the condition clears, the light remains illuminated until it is manually reset.

5.1.3.4 Fuel Loading

To ensure adequate monitoring of the neutron flux level during fuel loading operation, a neutron detector device which can be placed at various radial locations in the core is required. As the fuel is loaded into the reactor, the core size increases radially. The incore SRM detectors may not be sufficiently sensitive nor in the proper location to adequately monitor the neutron flux levels. Instead, four movable detectors called dunking chambers are substituted for the normal SRM detectors. These detectors are mounted inside dummy fuel assemblies to facilitate movement with standard fuel handling equipment. The output connectors from the dunking chambers are connected into the SRM preamplifiers in place of the normal SRM detector input.

The dunking chamber or fuel loading chamber (FLC) is different from the fission chamber used in the standard SRM detector. Since neutron levels are very low, more sensitivity is required. More sensitivity is designed into the FLC by providing a larger detector surface area, utilizing a boron-10 ($_5B^{10}$) coating and operating the FLC in the proportional region of radiation detector operation.

As fuel is loaded into the reactor, neutron flux level increases because of subcritical multiplication. When the normal SRM detectors are reading at least 3 cps, the fuel loading chambers can be removed and normal SRM detectors connected to the channels and placed in service.

5.1.3.5 Neutron Sources

Five neutron sources are installed in the reactor core to provide a background level of indication for the SRM channels. Without the background neutron level provided by the installed sources, the actual count level would be below the indicating range of the SRMs. In this condition, the reactor could be taken critical without any indication of neutron level. Should positive reactivity continue to be added in this situation, excessively short reactor periods could result without the operator's knowledge. To preclude the unsafe conditions arising from a reactor startup below source range indication, the neutron sources are installed in the core. These sources are mounted in tubes, shown in Figure 5.1-8, between the core plate and top guide.

The operating source, shown in Figure 5.1-9, is two antimony pins encapsulated in stainless steel within a beryllium sleeve. The neutron source reactions are shown in Figure 5.1-10. This source has a 60 day half life and is kept irradiated by core flux during normal operations. Core source locations are shown on Figure 5.0-2.

After a reactor has been operated for an entire cycle, then the startup sources are no longer needed. In addition, the source holders are not designed for multiple cycles at high exposures. Therefore, the installed sources are usually removed at the end of first cycle operation. Normally there is sufficient neutron population from the exposed fuel assemblies to maintain SRM count rates above 3 counts per second.

5.1.3.6 System Interfaces

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

Reactor Protection System (Section 7.3)

The Reactor Protection System (RPS) receives SRM detector signals (when the shorting links are removed) in order to generate a scram during refueling operations.

Reactor Manual Control System (Section 7.1)

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

Intermediate Range Monitoring System (Section 5.2)

The Intermediate Range Monitoring (IRM) System provides automatic bypassing of SRM rod blocks based on IRM range switch position.

Emergency Distribution System (Section 9.1)

The Emergency Distribution System supplies power to operate the SRM detector drive mechanism.

DC Power System (Section 9.4)

The 24 volt DC Power System provides power to the SRM System.

5.1.4 BWR Differences

The SRM System used on BWR/2 through BWR/6 product lines is virtually the same. Some large reactors use six SRM detectors but differences are usually slight.

5.1.5 Summary

Classification -Power generation system.

Purpose - To monitor neutron flux for display and initiation of rod withdraw blocks from shutdown conditions to overlap with the intermediate range.

Components - Detectors; detector drive mechanisms; pulse pre-amplifiers; pulse height discriminators; log integrator; log count rate amplifier; period circuit; trip units; readout equipment.

System Interfaces - Reactor Manual Control System; Reactor Protection System; Intermediate Range Monitoring System; Standby Auxiliary Power System.

TABLE 5.1-1 SRM INTERLOCKS AND TRIPS

ALARM OR TRIP (1)	SETPOINT	SRM CHASSIS INDICATION (2)	Panel 603 INDICATION	ANNUNCIA-TOR (2)	ACTION	AUTO BYPASS
SRM Upscale (High-High)	$\leq 2 \times 10^5$ cps	UPSCALE Trip (Red Light) (A-D)	UPSCALE Trip (A-D)		Scram when Shorting Links Removed	Bypassed When Shorting Links Installed
SRM Upscale (High)	$\leq 1 \times 10^5$ cps	UPSCALE Alarm or INOP (A-D)	UPSCALE Alarm or INOP (A-D)	SRM Upscale or INOP Rod Withdrawl Block	Rod Withdrawal Block	Both associated(3) IRMs above range 7; or SRM Bypassed
SRM Downscale	≥ 3 cps	DOWNSCALE (White Light) (A-D)	DOWNSCALE (A-D)	SRM DOWNSCALE Rod Withdrawal Block	Rod Withdrawal Block	Both associated IRMs above range 2; or SRM Bypassed
SRM INOP	(4)	INOP (White Light) (A-D)	UPSCALE Alarm or INOP (A-D)	SRM Upscale or INOP Rod Withdrawal Block	Rod Withdrawal Block Scram when shorting links removed	Both associated(3) IRMs above range 7; or SRM Bypassed Bypassed when shorting links installed

TABLE 5.1-1 SRM INTERLOCKS AND TRIPS (Continued)

ALARM OR TRIP (1)	SETPOINT	SRM CHASSIS INDICATION (2)	Panel 603 INDICATION	ANNUNCIA-TOR(2) (P680)	ACTION	AUTO BYPASS
SRM Period	≥50 sec	PERIOD (Amber Light) (A-D)	PERIOD (Amber Light) (A-D)	SRM PERIOD		
Retract Not Permitted	≥100 cps	RETRACT PERMIT (White Light) (A-D)	RETRACT PERMIT (A-D)	SRM DETECTOR RETRACT NOT PERMITTED Rod Withdrawal Block	Rod Withdrawal Block	Both associated IRMs above range 2; or SRM Bypassed or Mode SW in RUN with associated APRM not downscale
SRM Bypassed	Bypass Switch (5)	BYPASSED (White Light)	BYPASS (A-D)			Bypasses all trip functions of SRM when bypassed.

1. All trips automatically reset when the trip condition is cleared. Trip indicators on the SRM chassis must be manually cleared.
2. Pnl 603 trip status lights and annunciators are bypassed when the reactor mode switch is in the run position.

3. Associated IRMs and APRMs:

SRM CH	IRM CH	APRM CH.
A	A,E	A,E
B	B,F	B,F
C	C,G	C,E
D	D,H	D,F

4. Produced by: (a) SRM mode switch not in operate, (b) High voltage low (<95% normal), (c) Module unplugged

5. Only one SRM channel can be bypassed using the bypass switch.

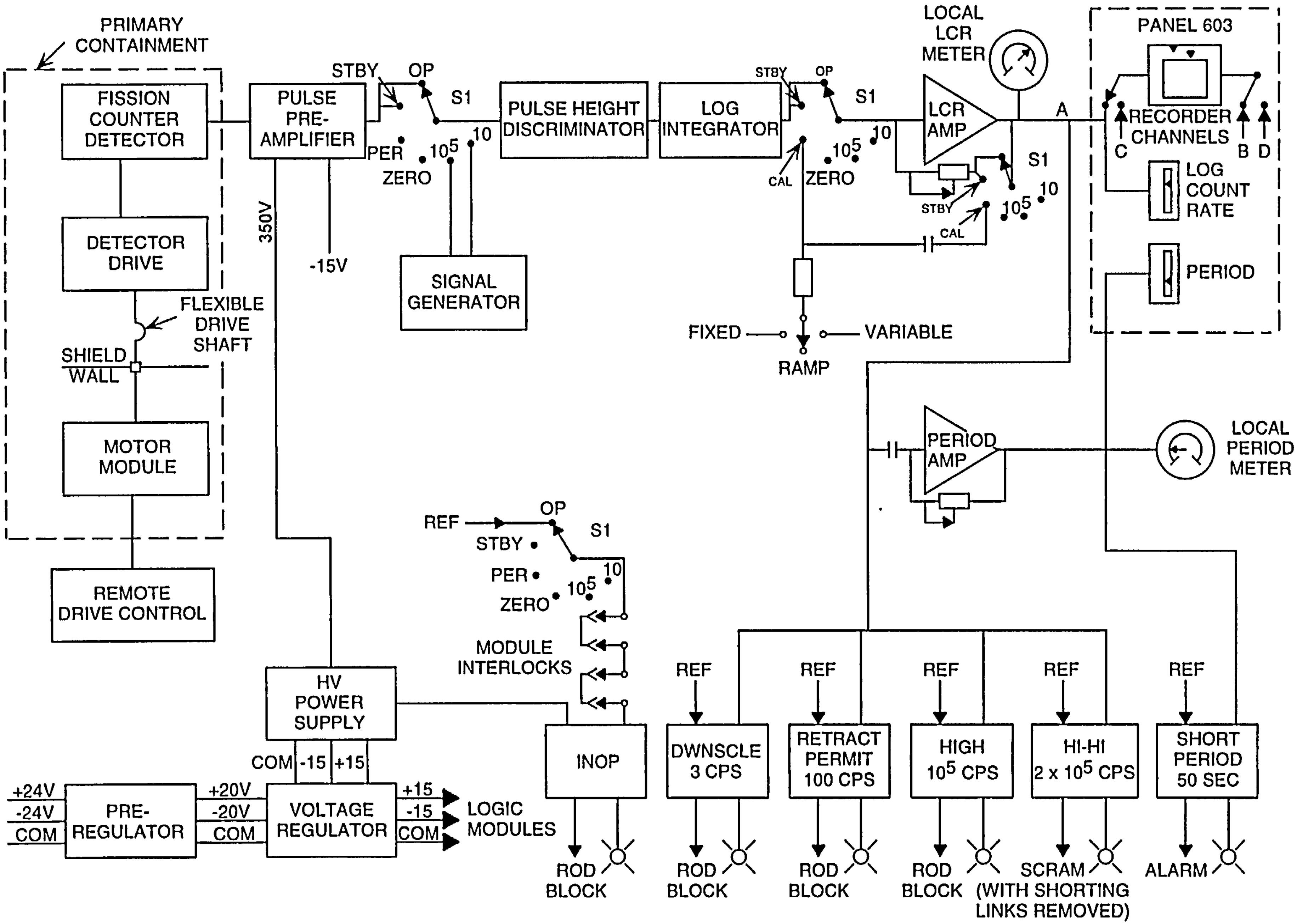


Figure 5.1-1 Source Range Monitoring Channel Functional Block Diagram

0601

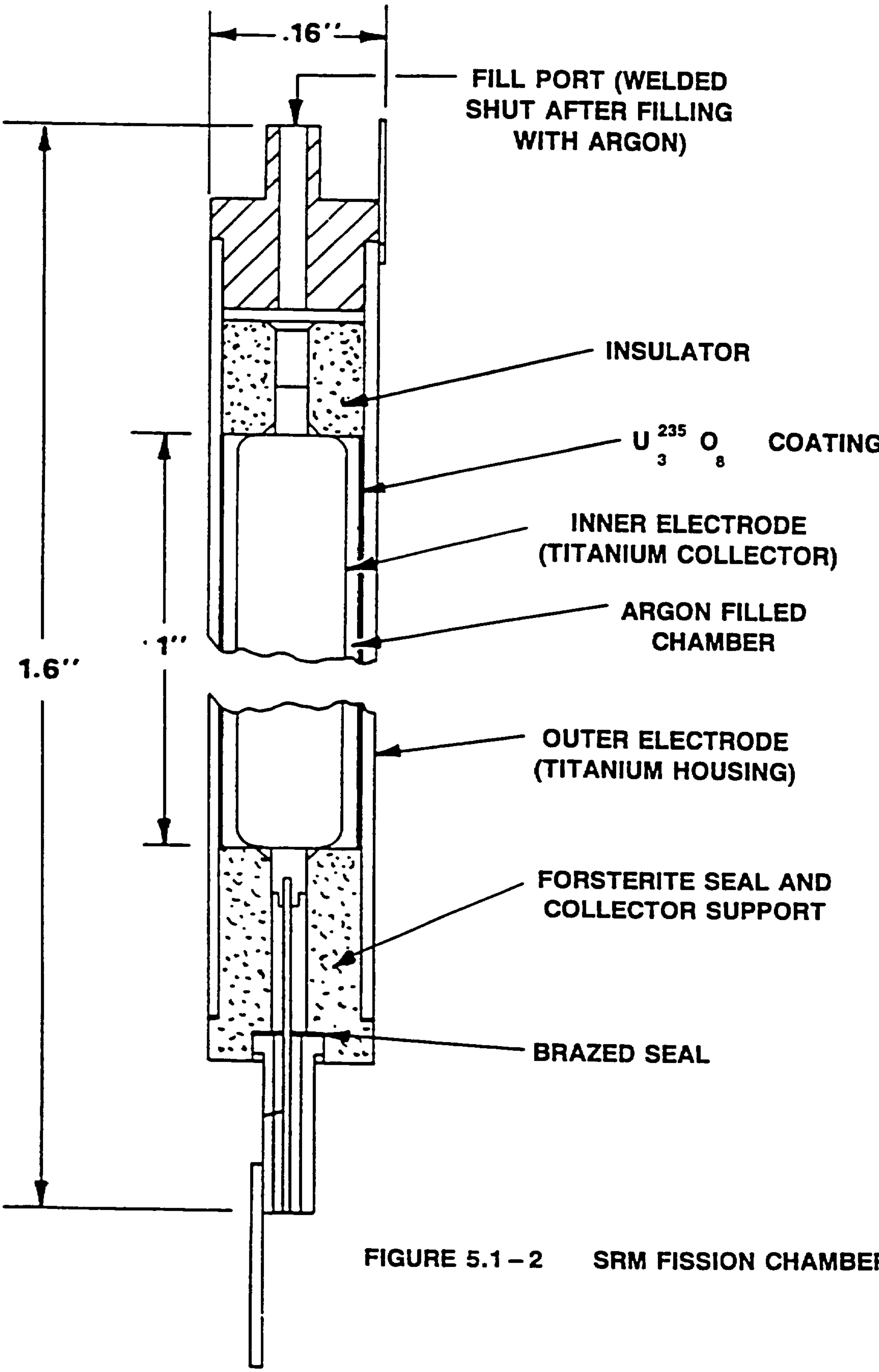
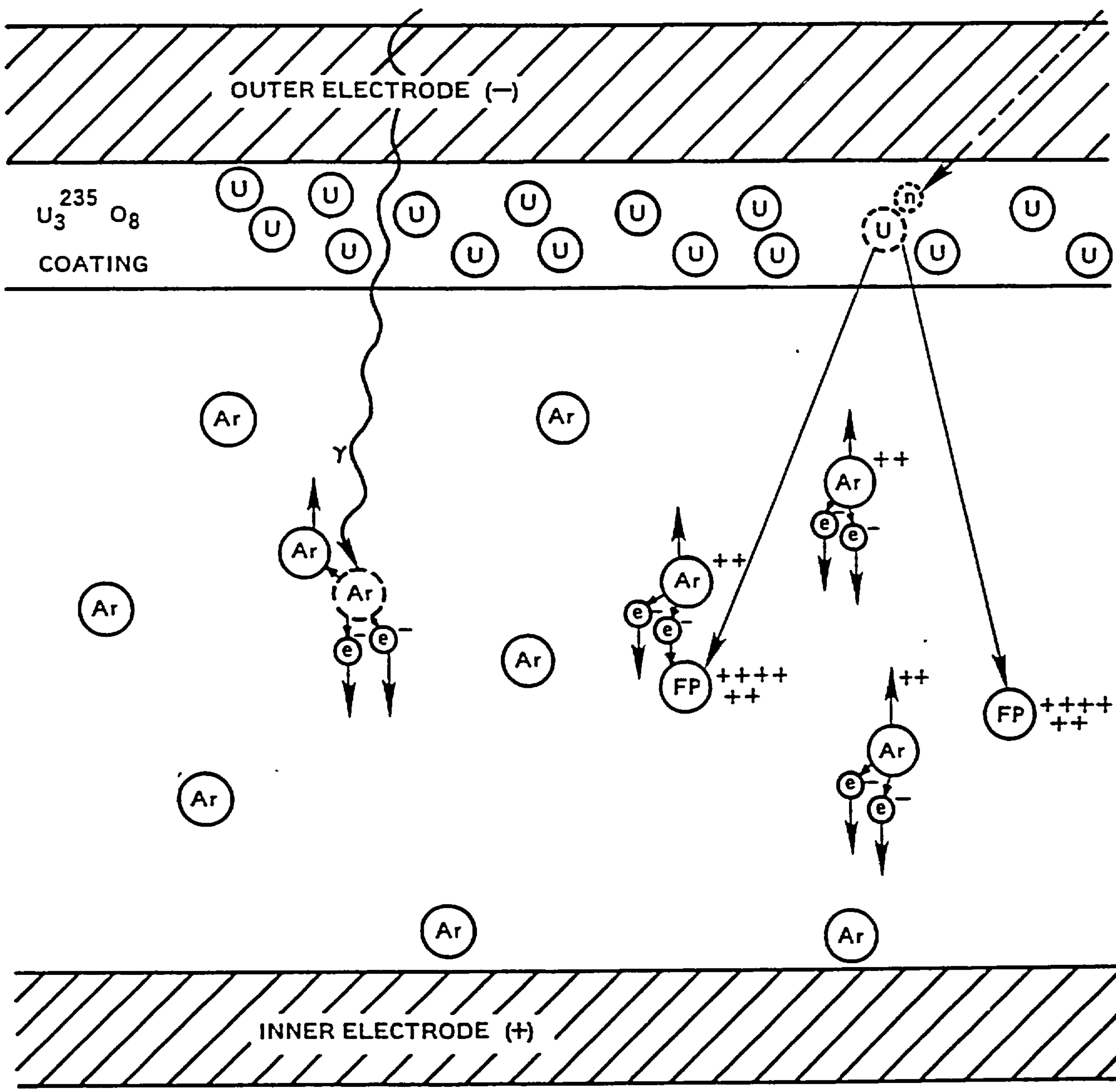


FIGURE 5.1 – 2 SRM FISSION CHAMBER



DETECTOR DATA

90% ENRICHED IN U—235

INTERNAL PRESSURE 215 psi

LENGTH 1.6 INCHES

WIDTH 0.16 INCHES

FIGURE 5.1 – 3 FISSION CHAMBER OPERATION

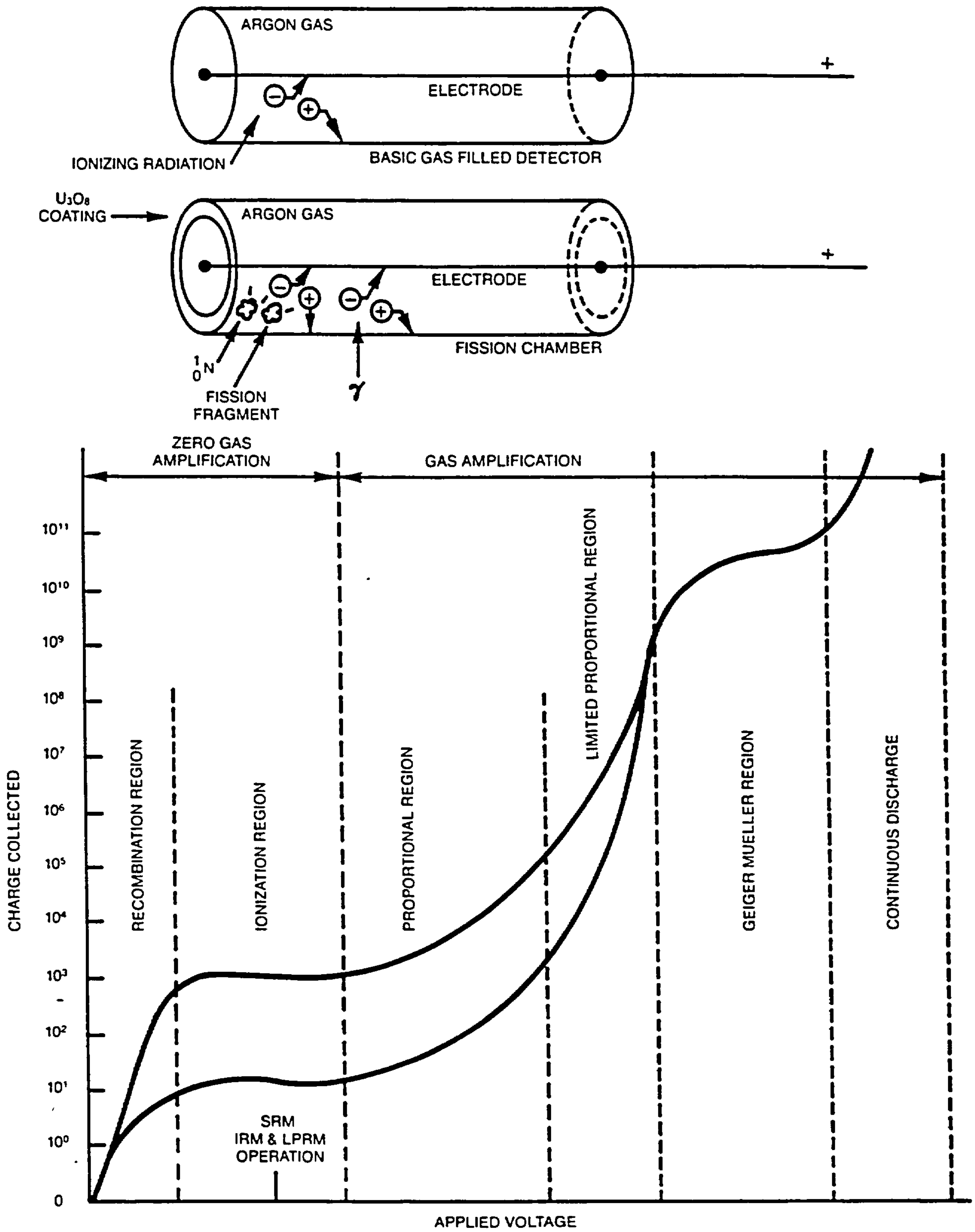


FIGURE 5.1–4 PULSE HEIGHT VS APPLIED VOLTAGE FOR GAS FILLED DETECTORS

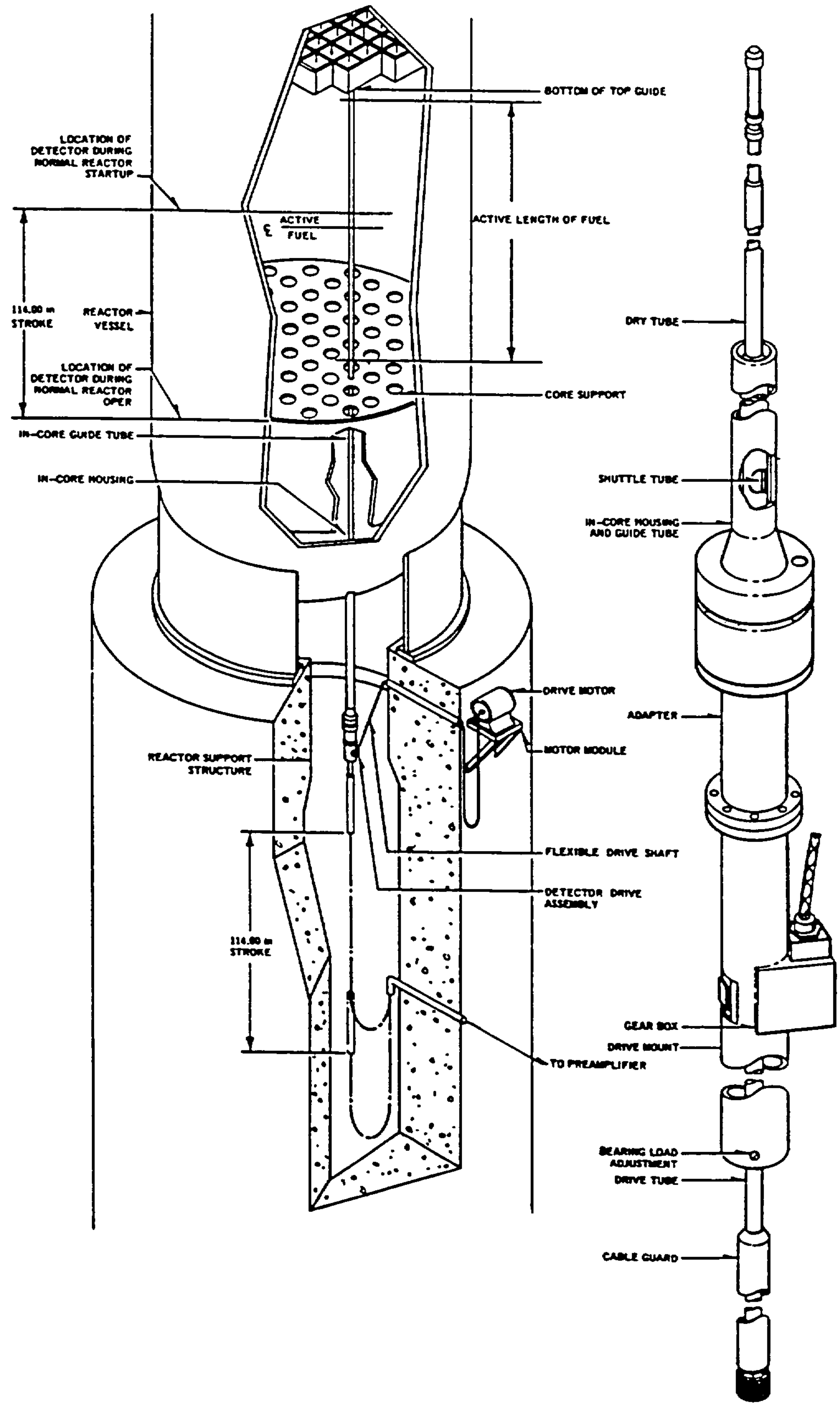


FIGURE 5.1-5 SOURCE RANGE AND INTERMEDIATE RANGE DETECTOR

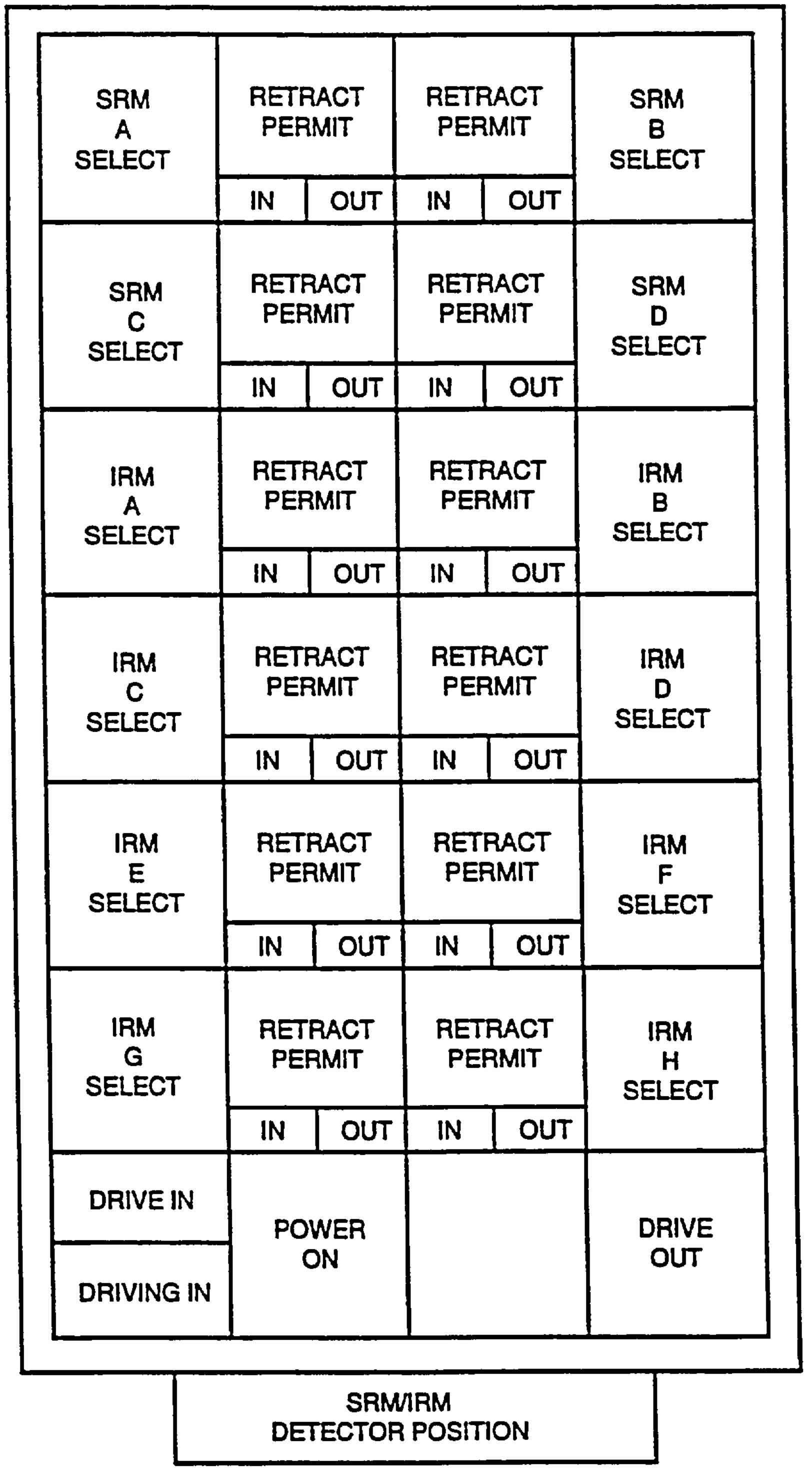


Figure 5.1-6 Detector Drive Control Switch Arrangement

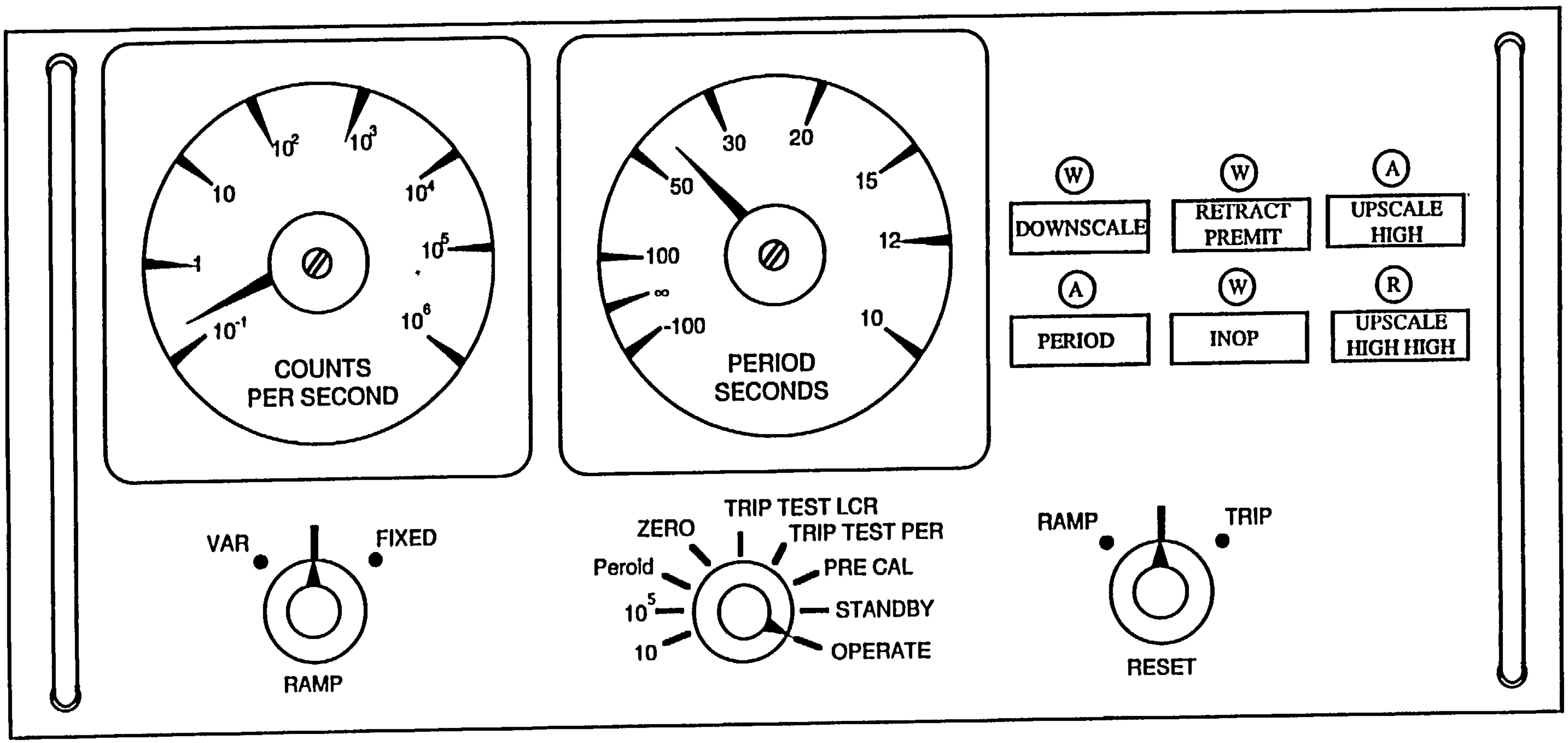


Figure 5.1-7 SRM Drawer Front Panel Layout

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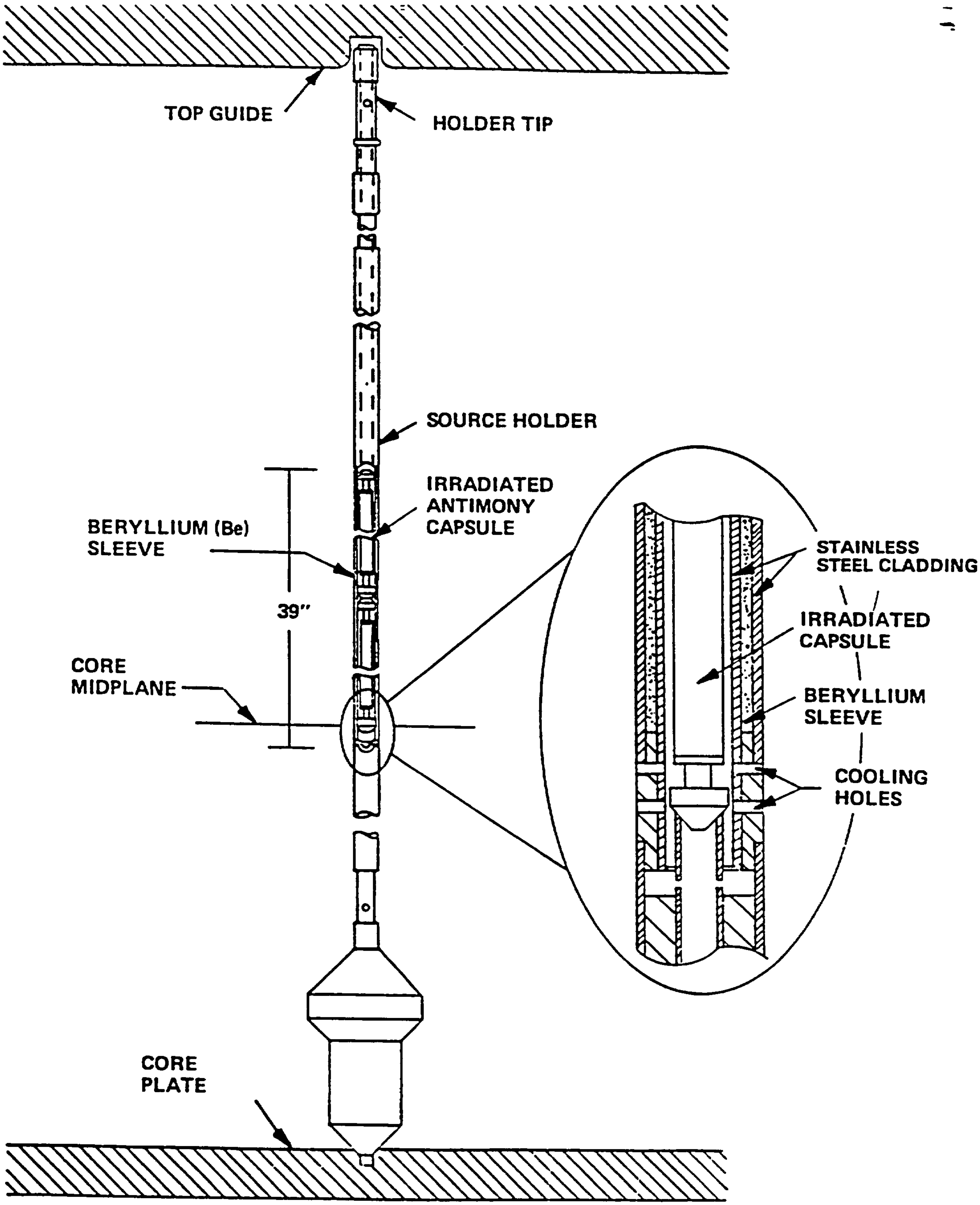

FIGURE 5.1–8 NEUTRON SOURCE TUBE

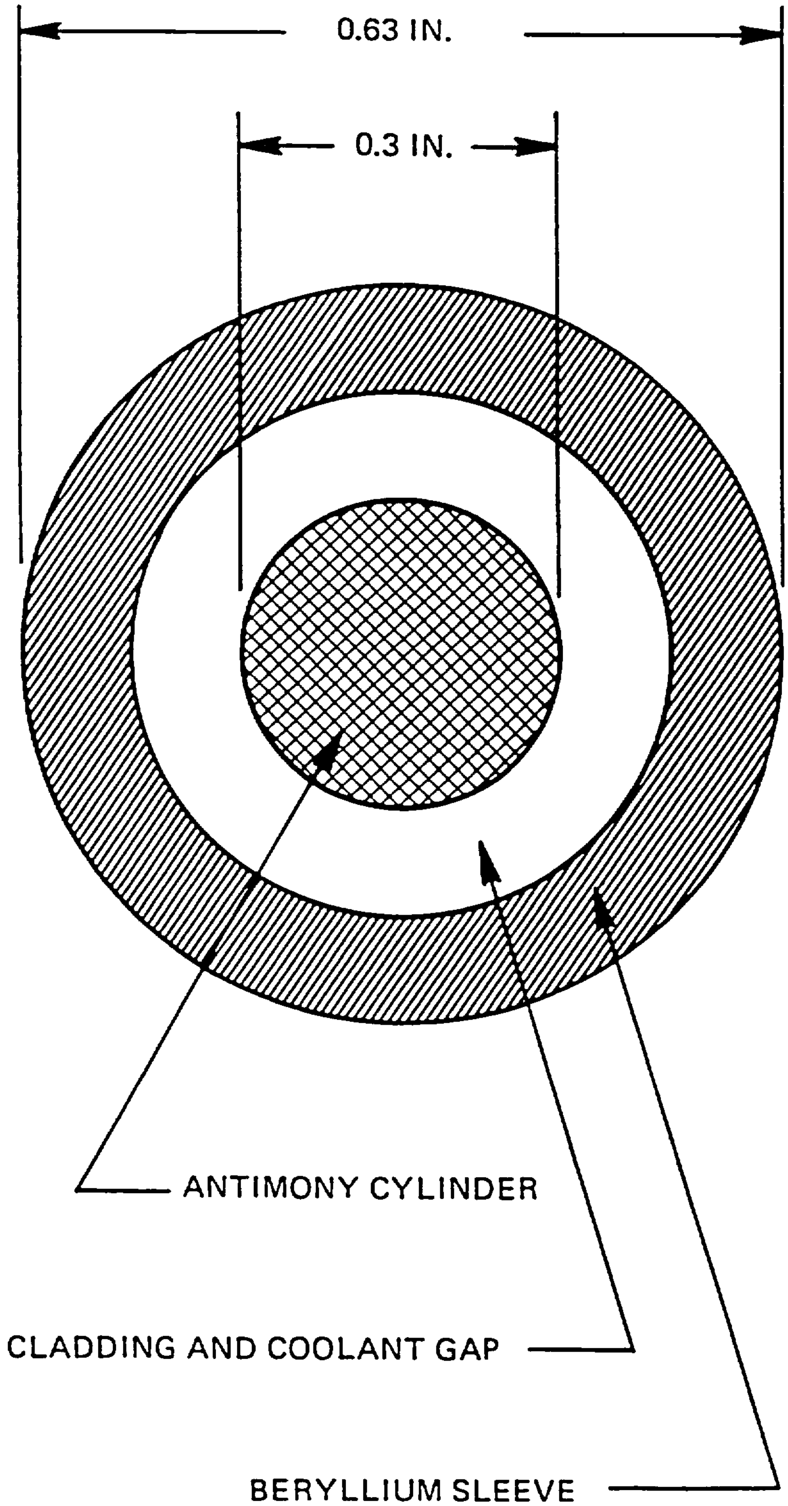

Figure 5.1-9 Neutron Source Configuration.

$$_{51}SB^{123} + {}_{0}N^{1} \rightarrow {}_{51}SB^{124}$$

$$_{51}SB^{123} \text{ (60 DAYS)} \rightarrow {}_{52}TE^{124} + \beta + \gamma$$

$$\gamma + {}_{4}BE^{9} \rightarrow {}_{0}N^{1} + 2_{2}\alpha^{4}$$

$$_{2}\alpha^{4} + {}_{4}BE^{9} \rightarrow {}_{6}C^{12} + {}_{0}N^{1}$$

(LOW PROBABILITY - 10^{-4} $_{0}N^{1}$ / $_{2}\alpha^{4}$

FIGURE 5.1-10 NEUTRON SOURCE REACTIONS

Boiling Water Reactor
GE BWR/4 Technology
Technology Manual

Chapter 5.2

Intermediate Range Monitor System

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5.2 INTERMEDIATE RANGE MONITORING SYSTEM

Lesson Objectives:

1. State the systems's purposes.
2. Explain how the method of ranging provides protection against rapid power increases.
3. List the protective trips generated by this system, the action caused by the trips, and the reason for the trips.
4. Explain the interfaces this system has with the following plant systems:
   a. Reactor Protection System
   b. Reactor Manual Control System

5.2.1 Introduction

The purposes of the Intermediate Range Monitoring (IRM) System are to provide neutron flux information from the upper portion of the source range to the lower portion of the power range and to provide trip signals to preserve the integrity of the fuel cladding.

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

The IRM System consists of eight independent channels, each of which has a miniature fission detector, electronic signal conditioning equipment, readout equipment, trip units, and a mechanical retraction mechanism. The eight fission chamber detectors are located at different radial core locations as shown in Figure 5.2-1.

The raw fission chamber detector signal is received by the voltage preamplifier which amplifies the low level detector signal and selects the correct frequency prior to sending the signal on to the amplifier and attenuator located in the control room. The amplifier and attenuator along with the mean square analog unit condition the detector signal for use by the trip units and indicators.

5.2.2 Component Description

The major components which make up the IRM system are shown in Figure 5.2-2 and are discussed in the following paragraphs. Several components in the IRM System are similar to components in the SRM System. Reference to Section 5.1 is made in these instances.

5.2.2.1 Detector

The detectors used for the IRM channels are basically the same as the SRM detectors discussed in Section 5.1. However, the IRM detectors use a lower argon pressure (17.7 psia), a lower uranium content, and closer spacing between the cathode and anode to permit operation at higher levels of thermal neutron flux. As with the SRM detectors, the IRM detectors are retractable, and are withdrawn when sufficient Average Power Range Monitoring (APRM) System overlap is achieved to preserve detector life.

5.2.2.2 Detector Insert and Retract Mechanism

The IRM detector insert and retract mechanisms are identical to and operate in the same manner as the SRM detector insert and retract mechanisms discussed in Section 5.1.

5.2.2.3 Voltage Preamplifier

The voltage preamplifier conditions the signal from the detector for input to the IRM electronics drawer by converting the varying DC signal to an AC voltage signal and selectively amplifying the signal. High voltage is supplied to the fission chamber detector through the voltage preamplifier from a high voltage power supply in the IRM electronics drawer in the control room. A filter is in series with the high voltage supply to reduce signal loss and isolate the detector output signal from the power supply.

In order to optimize amplifier response and ensure that the output of the preamplifier is uniformly representative of actual reactor power over a wide power range, two essentially independent channels of amplification selected to operate one at a time are employed. The gain and bandwidth response of the preamplifier are controlled by the range select matrix discussed in section 5.2.2.5.

5.2.2.4 Amplifier and Attenuator Unit

The AC signal from the voltage preamplifier is applied to the amplifier and attenuator unit in the IRM drawer. This unit provides six discrete gain ranges by switching different amounts of attenuation into the signal path. The amount of attenuation is selected so that in the output of the IRM channel there will be a factor of 10 between alternate IRM ranges. For example, if the IRM channel is reading 40 on range 1 it will read 4 on range 3 and 0.4 on range 5; or if the IRM channel is reading 100 on range 6 it will read 10 on range 8 and 1.0 on range 10. For each setting of the range switch, one of the contacts in the amplifier and attenuator module will be closed by outputs from the diode logic matrix and the input signal will be routed through one of six different attenuation paths to the amplifier stages. For example, when the range switch is set to range 1, the input signal from the low frequency section of the preamplifier is applied to the amplifier stages with no attenuation. With the range switch set to range 2, the input signal is attenuated by a factor of 1.78. Successively larger values of attenuation are switched into the signal path as the range switch is set to ranges 3 through 6. For ranges 7 through 10, the input signal from the high frequency section of the preamplifier is applied to the amplifier stages with attenuation ratios of 1.0, 1.78, 3.16, and 5.62 respectively. The combination of six attenuation ratios in the amplifier and attenuator module and two gain ranges in the preamplifier makes possible a total of 12 ranges of measurement; however, the range switch has a mechanical stop which prevents it from being set to ranges 11 or 12.

5.2.2.5 Diode Logic Matrix

The range switch position determines the output of the diode logic matrix. The diode logic matrix has two outputs, one to control the contacts in the preamplifier frequency section and the other to control the amount of attenuation in the amplifier attenuator. Thus, the diode logic matrix controls how the detector signal is conditioned and what attenuation will be applied before the signal reaches the operator's indication.

5.2.2.6 Inverter

The inverter is located between the amplifier and attenuator unit output and the mean square analog unit input. The purpose of the inverter is to condition the amplifier and attenuator output signal for use by the mean square analog unit. The inverter (Figure 5.2-3) module is an amplifier that takes the signal from the amplifier and attenuator unit and produces two outputs equal in amplitude but opposite in polarity. The inverter ensures the mean square analog circuit receives a positive signal at any one time.

5.2.2.7 Mean Square Analog Unit

The mean square analog unit (Figure 5.2-3) produces an analog signal directly proportional to the core neutron flux. The mean square analog unit receives two input signals, one from the inverter and one from the amplifier and attenuator unit. The squaring circuit, squares the two input signals to produce an output signal that is positive in polarity with the largest amplification performed on the larger neutron signal. The averaging circuit receives the positive signal from the squaring circuit and produces the mean of the squared signal. By squaring both the neutron and gamma signals produced in the detector the gamma signal's relative size becomes so small that its pulse size can be disregarded. The mean value is then sent to the output amplifier.

5.2.2.8 Operational Amplifier

The operational amplifier provides the required output to drive the displays and trip units. Individual IRM displays are provided on the control room front panel recorders shared with the APRM's (Figure 5.2-6) and their associated drawers. Both types of displays have a scale range of 0-125%. The even IRM ranges (2, 4, 6, 8, and 10) utilize the 0-125% scale, while the odd ranges use the 0-40 scale.

5.2.2.9 Trip Units

The trip units used by the IRM channels are identical to those used by the SRM's, with the exception of the trip points, which are listed in Table 5.2-1.

5.2.2.10 Bypass Switch

The IRM inputs to the Reactor Protection System (RPS, Section 7.3), are bypassed whenever the associated bypass switch for an RPS division is placed in that detector position. The arrangement of the switches ensures only one IRM channel for each RPS channel can be bypassed at a time. IRM channels A, C, E, and G provide trip inputs to RPS channel A. IRM channels B, D, F, and H provide trip inputs to RPS channel B.

In addition, the bypass switches also provide bypass for the rod withdrawal block inputs to the Reactor Manual Control System (Section 7.1).

5.2.3 System Features and Interfaces

A short discussion of system features and interfaces this system has with other plant systems 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 trips, along with the appropriate bypasses, are listed in Table 5.2-1.

5.2.3.2 IRM Recorders and Display Lights

The eight (8) IRM channels are displayed on four (4) two pen recorders on panel 603. These same recorders are shared with the APRM and RBM channels through the use of select switches as shown in Figure 5.2-6. In addition, each IRM channel is displayed on an IRM instrument drawer shown in Figure 5.2-5.

Each IRM channel has a group of lights associated with various trip functions listed in Table 5.2-1. They are located on panel 603 above the IRM range switches and on the IRM instrument drawer.

5.2.3.3 Operation During Startup

Initially during reactor startup, the approach to criticality and monitoring of low reactor power is monitored 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 (1). As soon as the IRM system is on scale, it is used to monitor reactor power, and the SRM detectors are incrementlly withdrawn to maintain a count rate between 10^2 and 10^5 cps. As reactor power approaches 75/125 of scale, the operator places the range switch on the next higher level. While ranging IRM channels, the IRM reading should be maintained betwen 25/125 and 75/125 of scale for each IRM channel to avoid rod withdraw blocks (upscale or downscale) from being imposed. In the intermediate range during supercritical reactor operation, the reactor power, although displayed on a linear scale (0-125), is increasing exponentially. This results in the indication appearing to accelerate on the upper end of the scale even though the reactor period remains constant. On range 9 of the intermediate range operation, the Average Power Range Monitoring (APRM) System indication begins to come on scale.

There should be sufficient overlap between the Average Power Range Monitoring 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 Monitoring (APRM) System is well on scale (5-10% power) the reactor mode switch is placed in the run position, and the IRM detectors are withdrawn to their fully withdrawn position.

5.2.3.4 System Interfaces

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

Reactor Protection System (Section 7.3)

The RPS receives various scram signals from the IRM System.

Reactor Manual Control System (Section 7.1)

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

Emergency Distribution System (Section 9.2)

The Emergency Distribution System supplies power to operate the IRM detector drive mechanism.

Source Range Monitoring System (Section 5.1)

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

Average Power Range Monitoring (Section 5.4)

The Average Power Range Monitoring (APRM) System shares recorders with the IRM's on the reactor control panel 603.

Rod Block Monitoring System (Section 5.5)

The Rod Block Monitoring System shares recorders with the IRM's on the reactor control panel 603.

5.2.4 BWR Differences

The IRM System used for all BWR/2 through BWR/6 reactors is virtually the same. Some smaller reactors use only six IRM channels; the method of bypassing channels and the amount of overlap between the IRM System and the APRM System may vary slightly.

5.2.5 Summary

Classification - Safety related system.

Purpose - To provide neutron flux information from the upper portion of the source range to the lower portion of the power range. To provide trip signals to preserve the integrity of the fuel cladding.

Components - Detector; insert and retract mechanism; voltage preamplifier; amplifier and attenuator unit; diode logic matrix; inverter; mean square analog unit; operational amplifier; trip units; bypass switches.

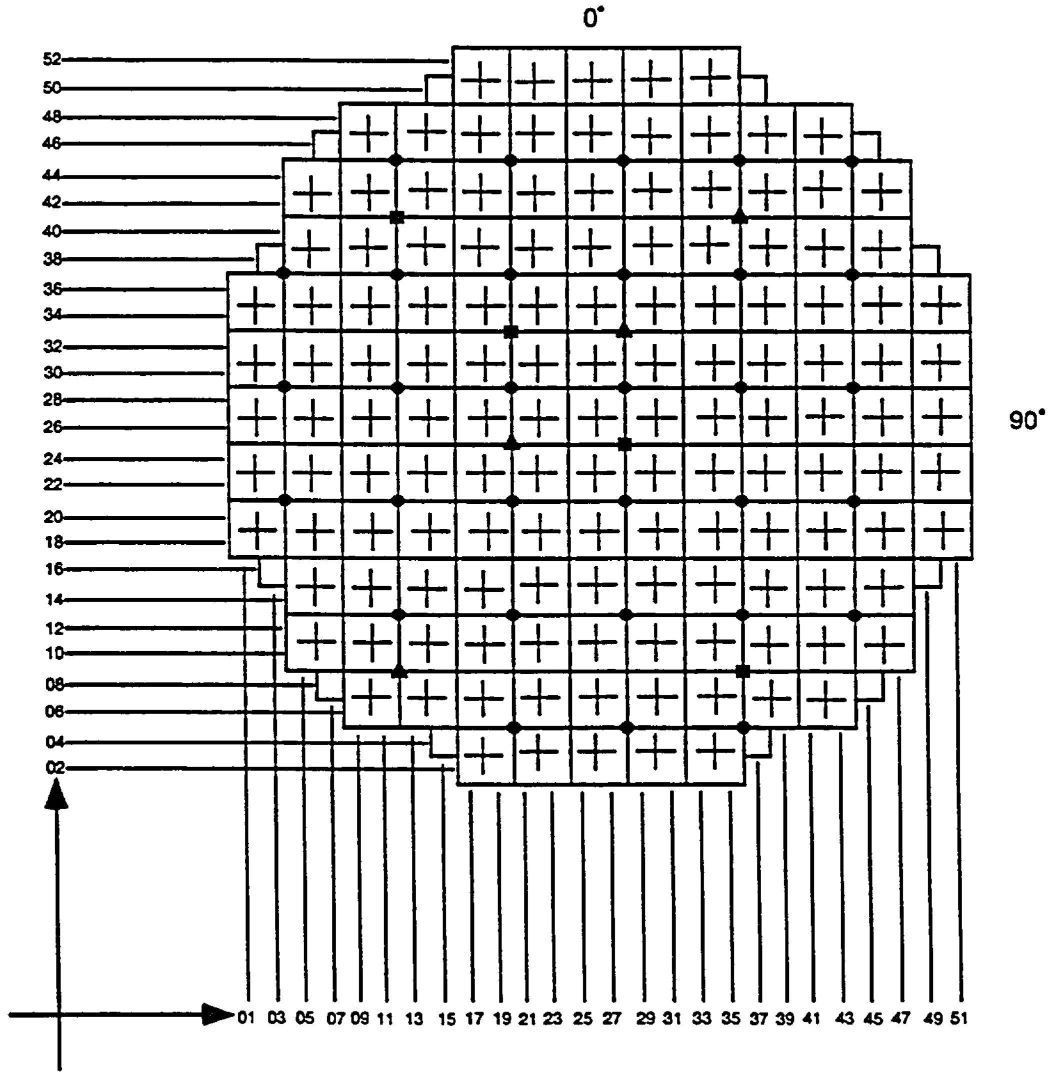
System Interfaces - Reactor Protection System; Reactor Manual Control System; Standby Auxiliary Power System; Source Range Monitoring System; Average Power Range Monitoring System; Rod Block Monitoring System.

TABLE 5.2-1 IRM INTERLOCKS AND TRIPS

ALARM or TRIP(1)	SETOINT	IRM CHASSIS INDICATION (2)	Panel 603 INDICATION (3)	ANNUNCIA-TOR(4)	ACTION (5)	AUTO BYPASS
IRM Upscale Trip (High High)	≤120 of 125 Scale	UPSCALE Trip (Red Light) (A-H)	UPSC TR or INOP(A-H)	IRM UPSCALE Trip or INOP (A or E, B or F, C or G, D or H)	Scram	Mode Sw. in RUN if associated APRM is not downscale
IRM Upscale Alarm (High)	≤ 108 of 125 Scale	UPSCALE Alarm (Amber Light) (A-H)	UPSC Alarm	IRM Upscale	Rod Withdraw Block	Mode Sw. in RUN
IRM Downscale	≤ 5 of 125 Scale	DOWNSCALE (White Light)	DNSC	IRM DOWNSCALE	Rod Withdraw Block	IRM on Range 1 or Mode Sw. in RUN
IRM INOP	(6)	INOP (White Light) (A-H)	UPSC TR or INOP (A-H)	IRM Upscale Trip or INOP (A/E, B/F, C/G, D or H)	Rod Withdraw Block ------------------ --Scram	Mode Sw. in RUN if associated APRM is not downscale
IRM Bypassed	Bypass Switch(7)	Bypassed (White Light)	Bypass (A-H)		Bypasses all trip functions of the IRM bypassed.	

NOTES FOR TABLE 5.2-1

1. All trips automatically reset when the trip condition is cleared. Trip indicators on the IRM chassis must be manually reset.
2. The trip status lights on the IRM drawer front panel operate regardless of mode switch position.
3. Operation of all Pnl 603 trip status lights is bypassed with the mode switch in the run position.
4. All IRM annunciators are bypassed with the mode switch in the run position.
5. IRM retraction produces a rod block except when the mode switch is in the run position or the channel is bypassed.
6. This is produced by the following:
   (a) IRM mode switch not in operate
   (b) High voltage low (<80 VDC)
   (c) Module unplugged
7. Only one IRM in each channel (A or B) may be bypassed.



+ Control Rod

● LPRM String

■ IRM Detector (RPS "A")

▲ IRM Detector (RPS "B")

Figure 5.2-1 Intermediate Range Monitor Detector in Core Location

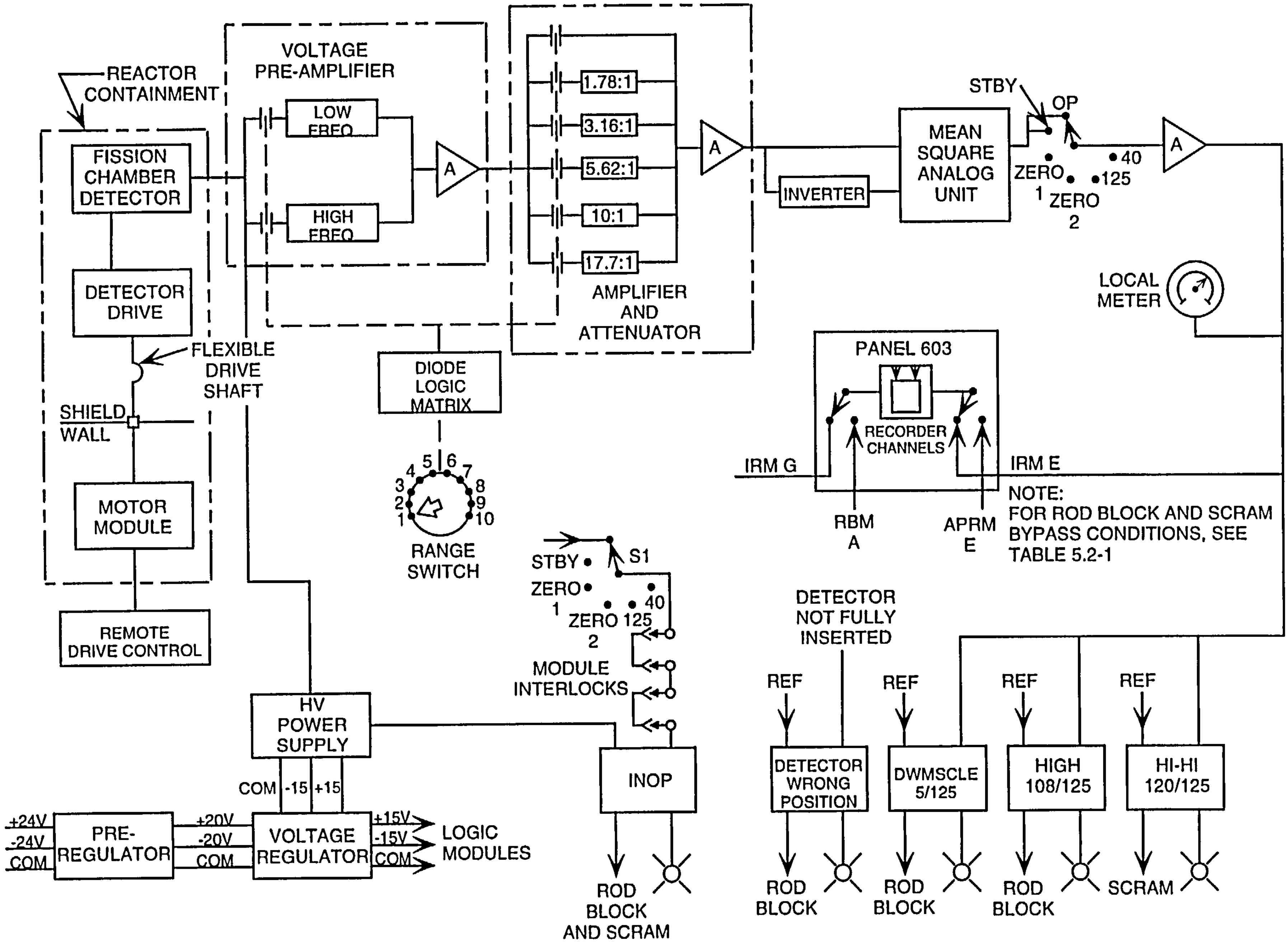


FIGURE 5.2-2 Intermediate Range Monitoring Channel Functional Block Diagram

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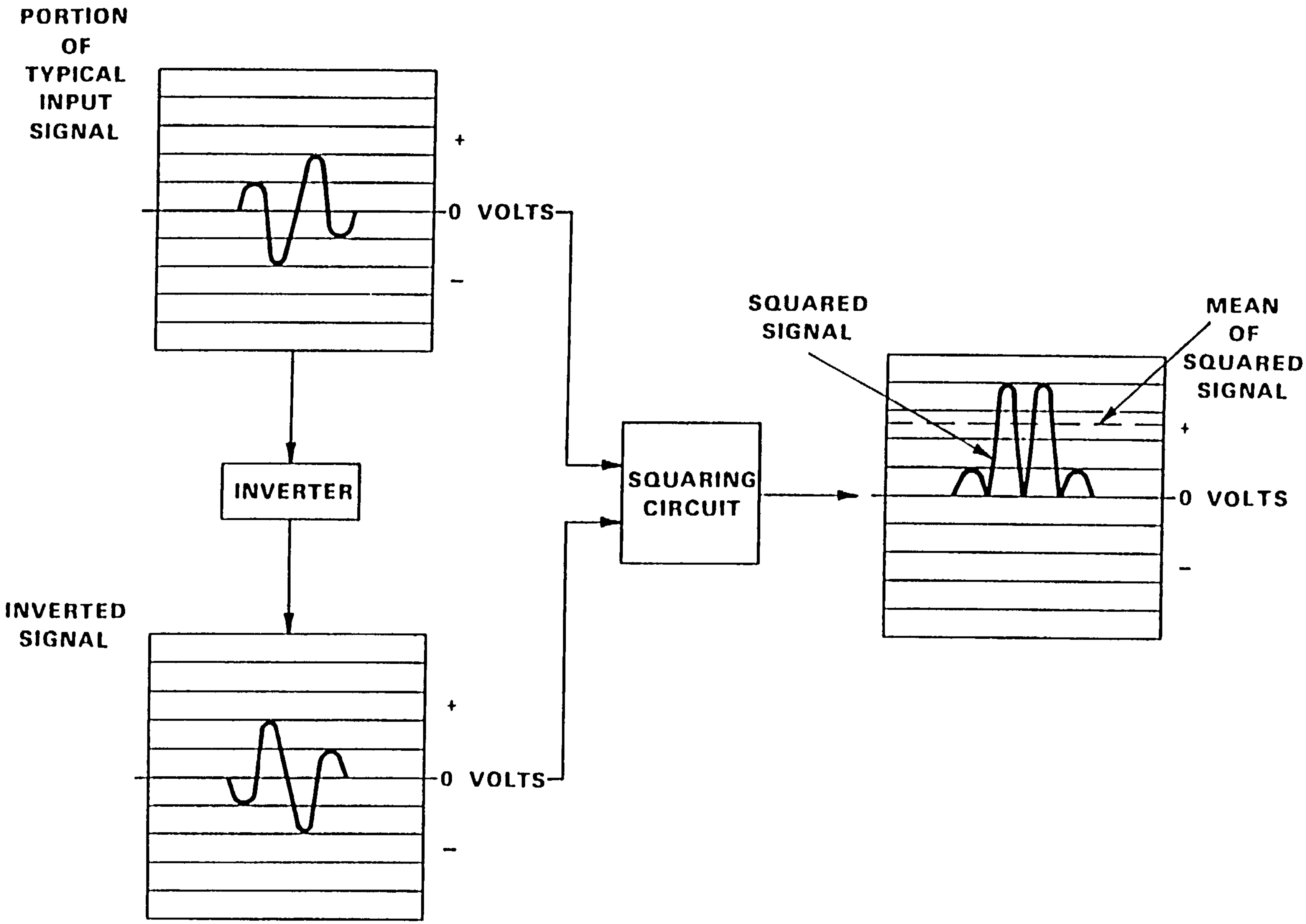

FIGURE 5.2 – 3 MEAN SQUARE ANALOG OPERATION

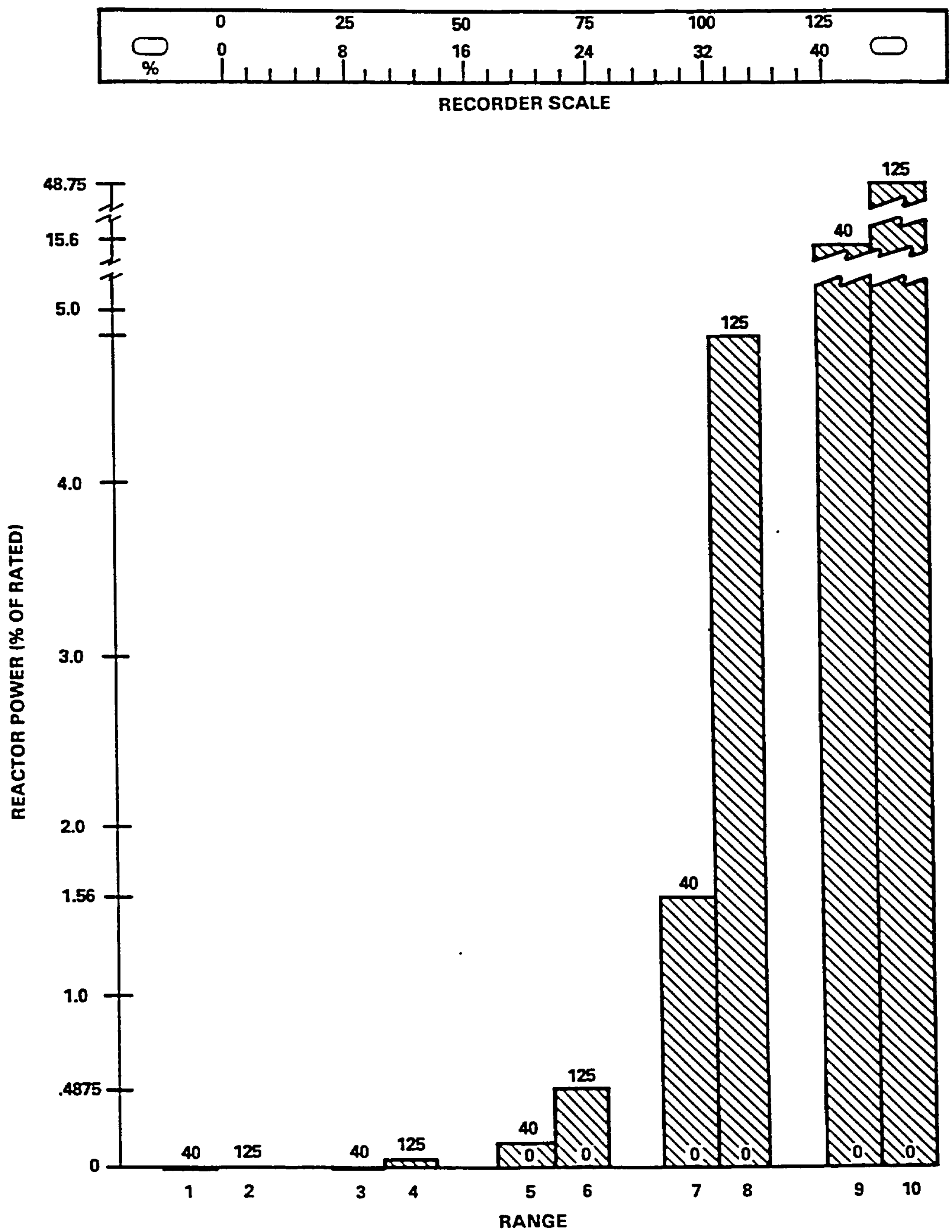

FIGURE 5.2-4 IRM RANGE AND SCALE RELATIONSHIPS

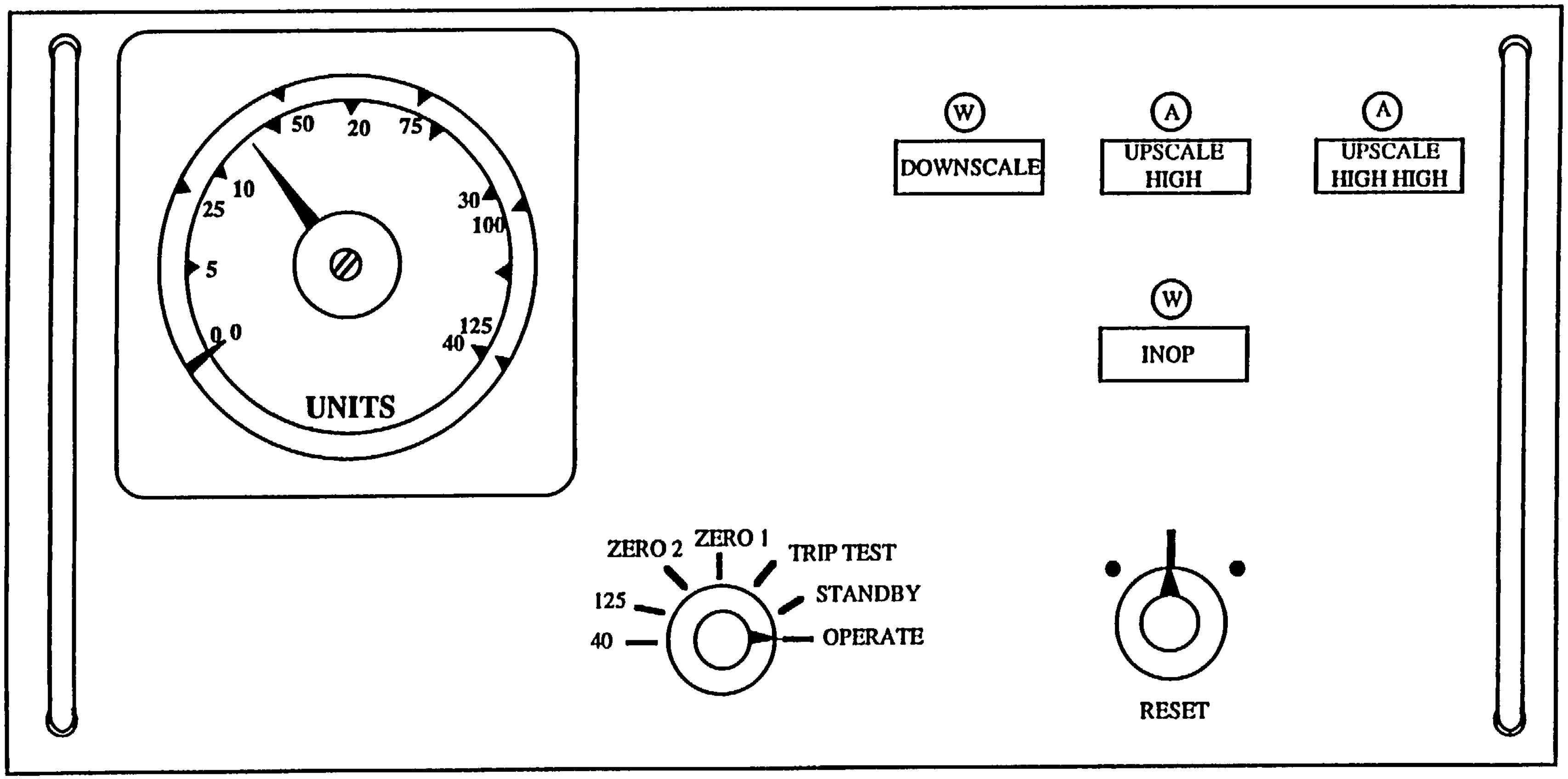

Figure 5.2-5 IRM Drawer Front Panel Layout

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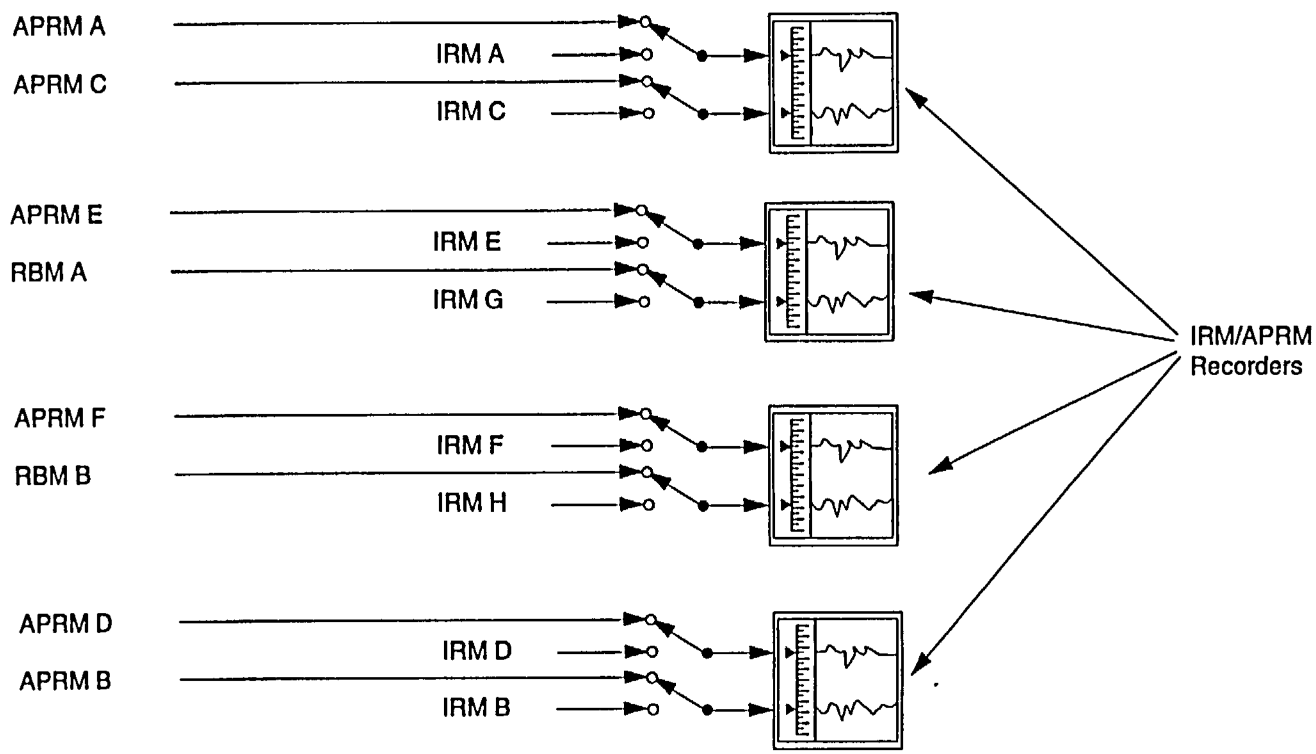

Figure 5.2-6 IRM/APRM Recorder Assignments

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Chapter 5.3

Local Power Range Monitor System

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TABLE 5.3-1 LPRM DESIGN PARAMETERS

Number	124
Electrode coating	1.21 mg of U_3O_8 (18% U^{235}, 78% U^{234}, 4% other U isotopes)
Neutron sensitivity	4.8 x 10^{-18} amps/nv (±20%)
Gamma sensitivity	Less than 2.0 x 10^{-14} amps/r/nv (≥ 1.5% full scale)
Collector operating voltage	100 VDC
Fill gas / pressure	Argon, 760 mm Hg @ 24^0 C
Diameter	0.213 inch
Sensitive length	0.987 inch
Case and collector material	Titanium

TABLE 5.3-1 LPRM DESIGN PARAMETERS

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Electrode coating	1.21 mg of U_3O_8 (18% U^{235}, 78% U^{234}, 4% other U isotopes)
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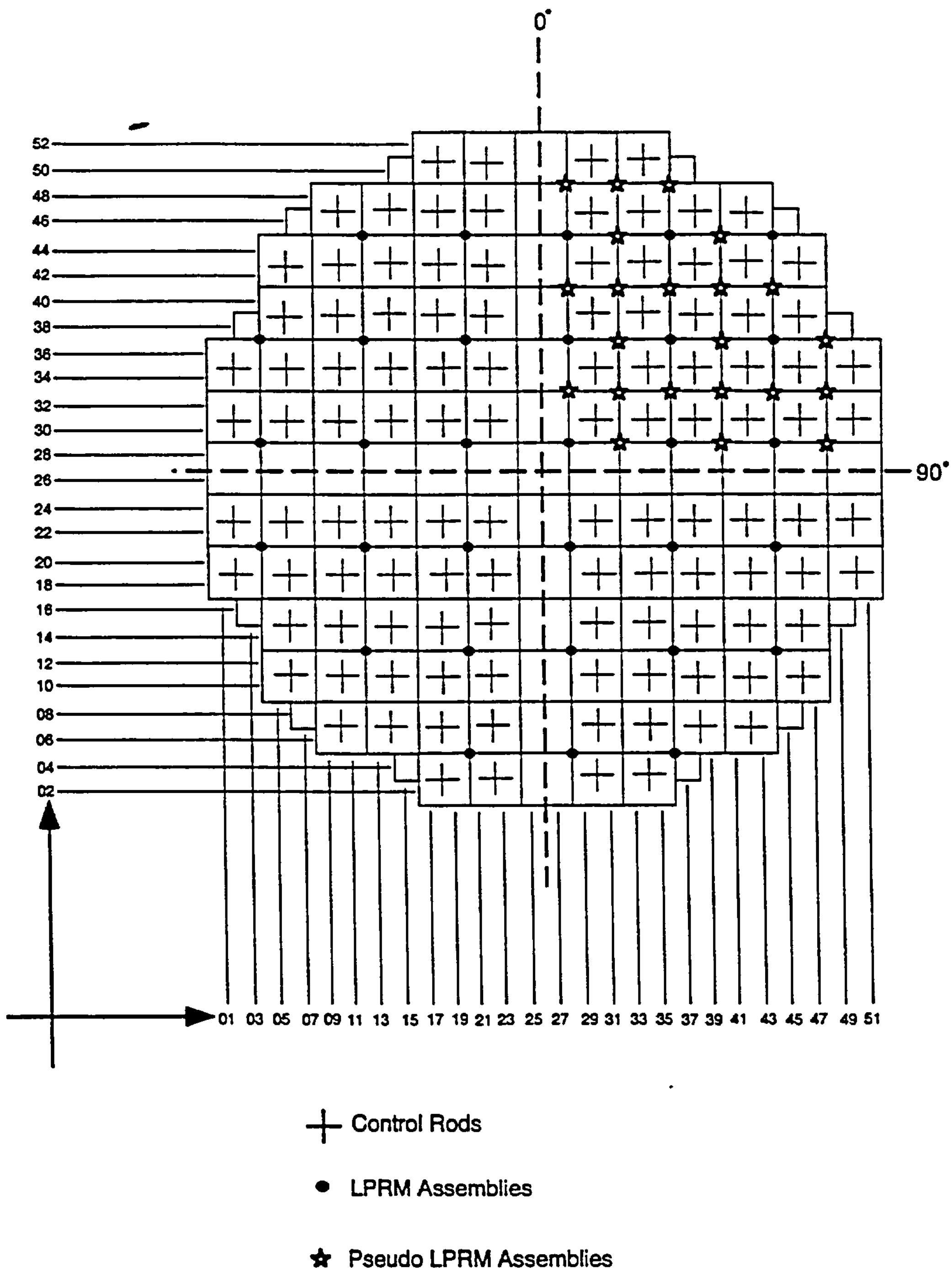

Figure 5.3-1 Local Power Range Monitor Detector Locations

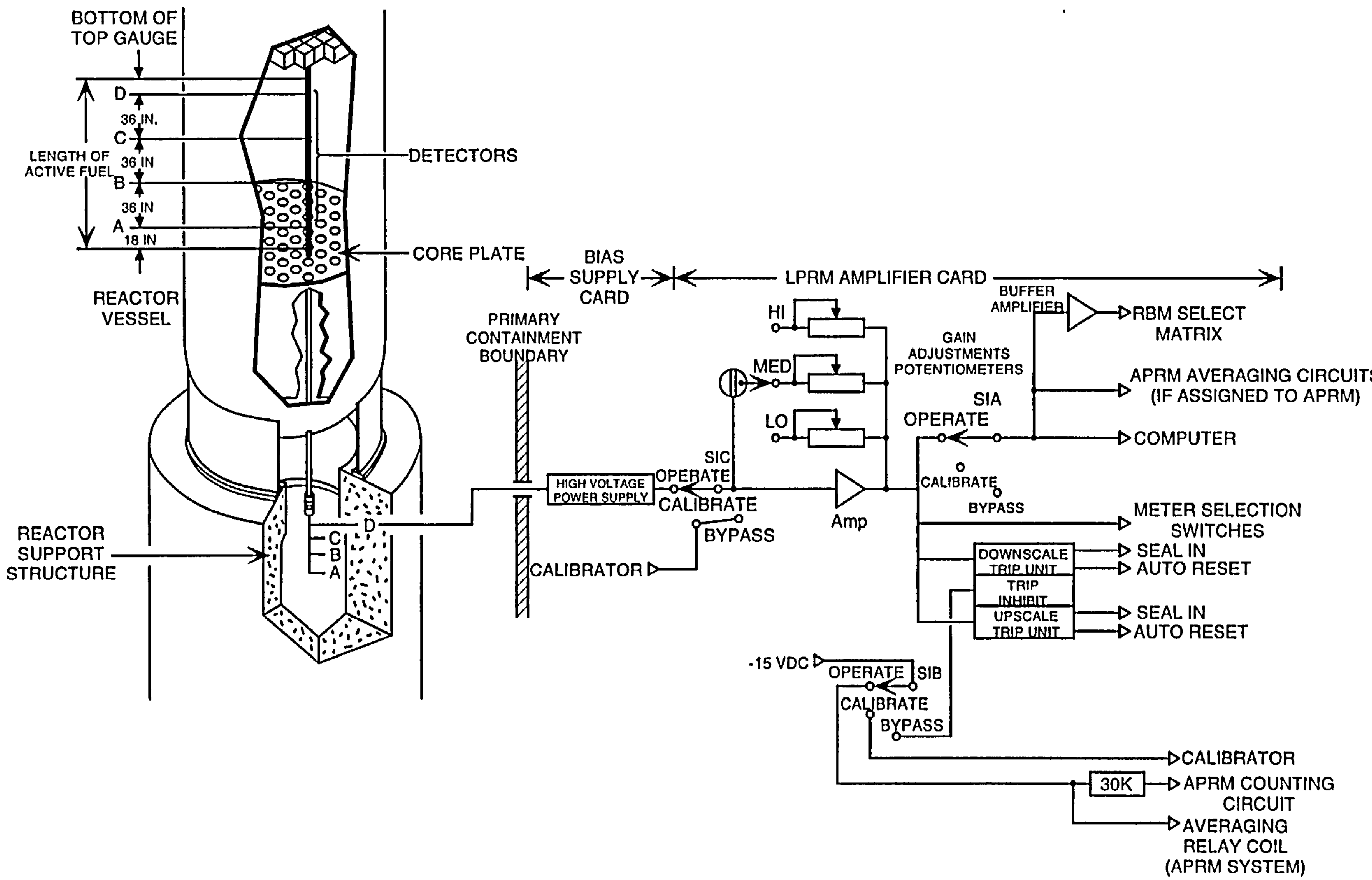

Figure 5.3-2 LPRM Simplified Block Diagram

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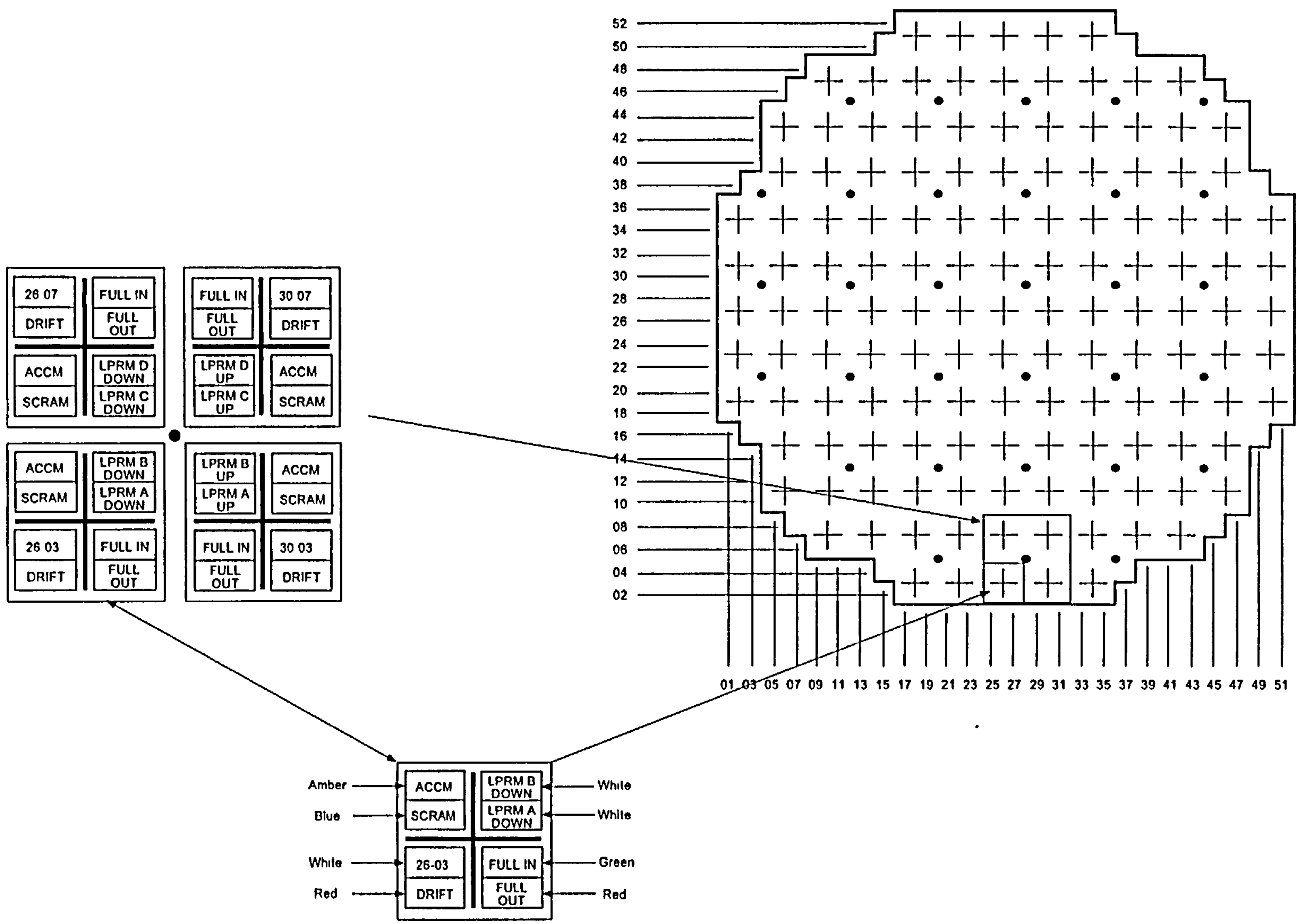

Figure 5.3-3 Full Core Display

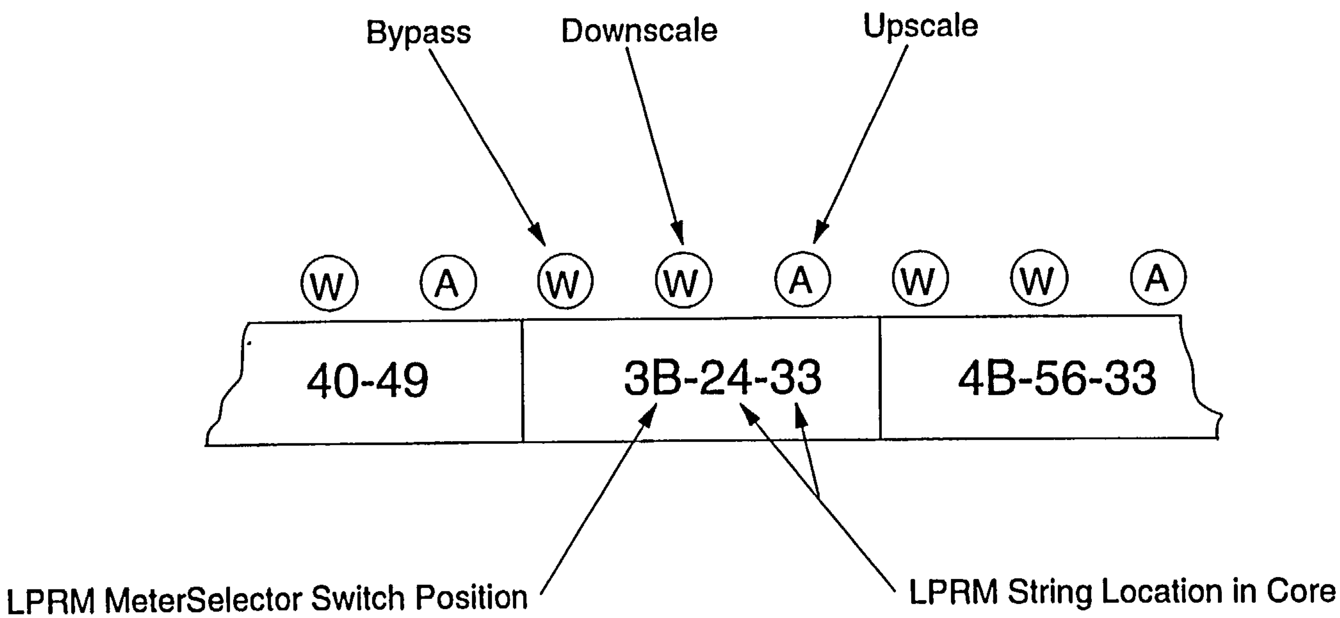

Figure 5.3-4 LPRM Alarm Indication - Top of Panel 608

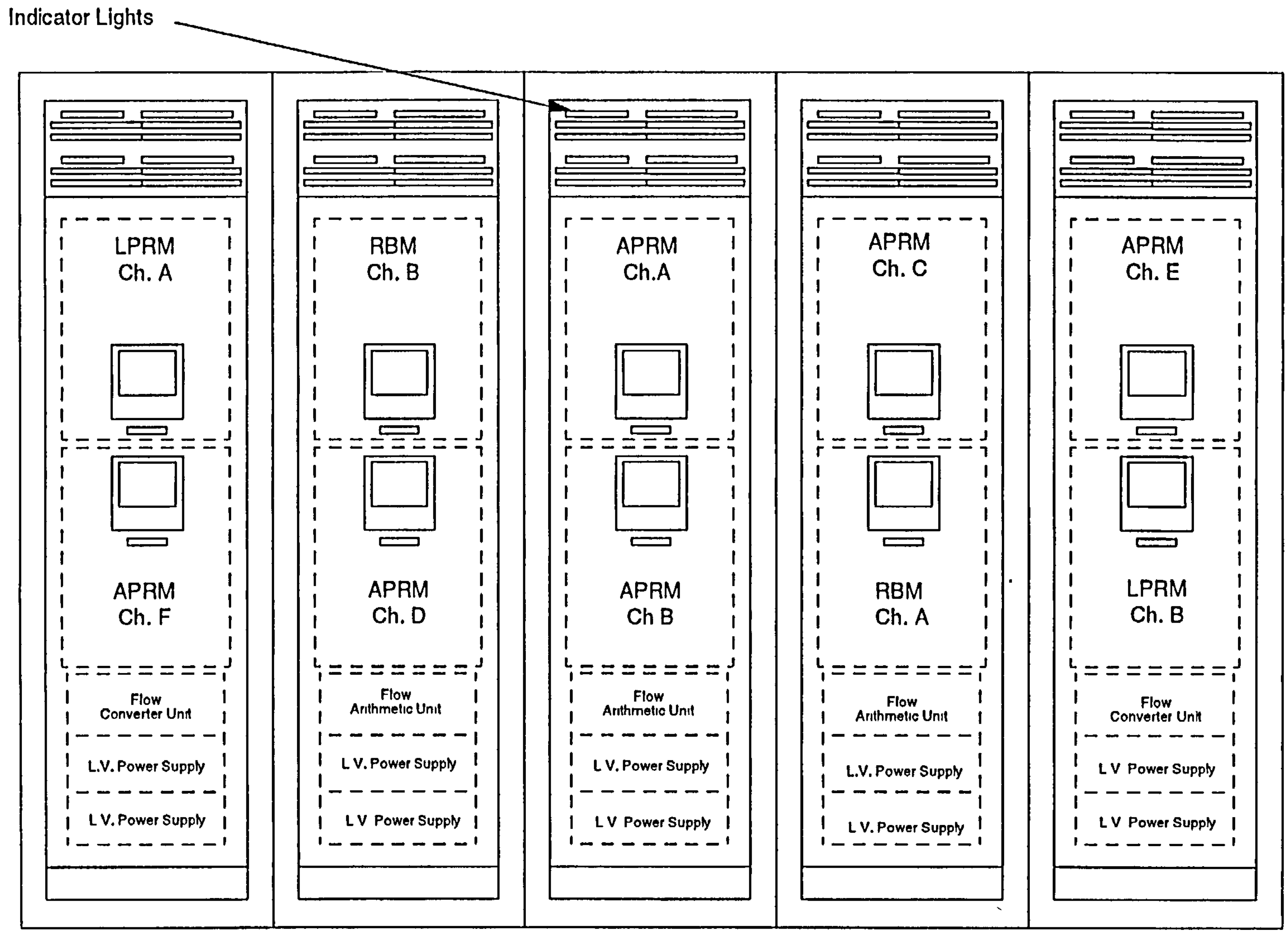

Figure 5.3-5 Power Range Cabinets

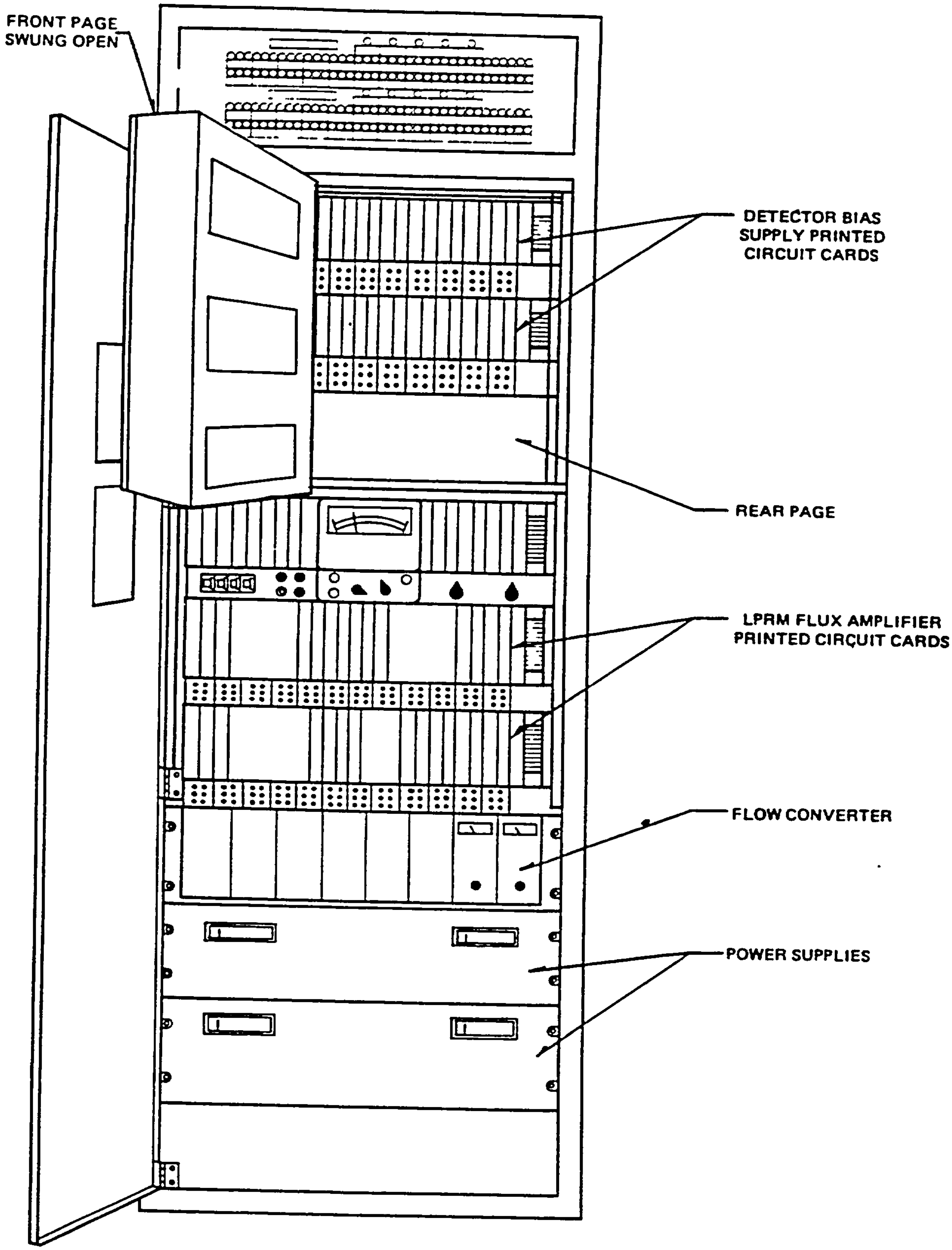

FIGURE 5.3-6 NEUTRON MONITORING SYSTEM CABINET LAYOUT.

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Chapter 5.4

Average Power Range Monitor System

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5.4 AVERAGE POWER RANGE MONITORING SYSTEM

Lesson Objectives:

1. State the system's purposes.
2. Explain how the system accomplishes its purposes.
3. List the protective trips generated by this system, the action caused by the trip, and the reason for the trips.
4. Describe the assignment of local power range monitor detectors assigned to the average power range monitors.
5. Explain the interfaces this system has with the following plant systems:
   a. Reactor Manual Control System
   b. Recirculation Flow Control System
   c. Local Power Range Monitoring System
   d. Reactor Protection System
   e. Rod Block Monitoring System

5.4.1 Introduction

The purposes of the Average Power Range Monitoring (APRM) System are to monitor core average thermal power and to provide trip signals to preserve the integrity of the fuel cladding.

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

The APRM System (Figure 5.4-1) consists of six channels designated A through F. Each APRM channel averages (electrically) the inputs from its assigned LPRM's, and provides an output signal proportional to the average of the LPRM flux signals. Because of the LPRM assignment to specific APRM channels, each APRM channel produces a signal proportional to the average neutron flux of the entire core.

The Recirculation System (Section 2.4) loop flow monitoring equipment provides APRM system trip units with a signal proportional to the total recirculation driving flow. Total recirculation driving flow consists of the sum of the recirculation pump (driving) flow in both recirculation loops.

The APRM System provides continuous indication of average reactor power from a few percent to 125% of rated reactor thermal power. It also provides interlock signals to the Reactor Manual Control System (Section 7.1) for blocking further control rod withdrawal at power and flow conditions below the reactor scram setpoint in order to avoid unnecessary reactor scrams. The APRM System also generates trip signals for the Reactor Protection System (RPS, Section 7.3) to scram the reactor in response to average neutron flux increases in time to prevent fuel damage even with the minimum number of LPRM inputs to satisfy operability requirements. The APRM System also provides visual indication to the operator of core thermal power through recorders and meters.

5.4.2 Component Description

The major components of the APRM System are illustrated in Figure 5.4-3 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. Specific LPRM assignments to the APRM's can be found in Figure 5.4-2.

5.4.2.2 APRM Averaging Circuit

Normally, APRM channels A, C and E each receive 17 LPRM inputs and APRM channels B, D, and F each receive 14 LPRM inputs. The averaging circuit averages only those LPRM signals that are operational. This averaged signal is calibrated to read percent of rated core thermal (heat balance) power. The percent core thermal power signal is then used for operator display, process computer information, Rod Block Monitor (Section 5.6) reference power and for APRM supplied trips.

5.4.2.3 Count Circuit

A minimum number of LPRM input signals are 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 LPRM inputs is less than 11 the count circuit generates an APRM inoperative trip signal.

The count circuit receives an input from each LPRM channel module as long as the LPRM module mode switch is in the operate position. With the LPRM mode switch in the operate position, a 15 volt source is connected to the LPRM count circuit which contains an input resistor. This provides a small DC current signal to an amplifier contained within the count circuit. Every LPRM assigned to an APRM channel is similarly wired. The output voltage from the count circuit is proportional to the input current signal. The trip reference level is set such that, for an analog signal input to the trip unit representing less than 11 of LPRM's assigned to the APRM channel, the trip unit causes a trip signal to be sent to the RMCS, RPS, and an annunciator.

5.4.2.4 Slope and Bias Circuit

The slope and bias circuit provides a reference for APRM rod block and scram trip points, based on total recirculation flow. Total recirculation flow is the summation of both recirculation loop flows which determines the ability to remove heat from the core. Without recirculation loop flow the rod block and scram setpoints are conservatively set at 42% and 51% power respectively. As indicated on Figure 5.4-4, from 42% to 108% and from 51% to 118% setpoints the reference 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 signal would be .66% W + 42 (rod block) and .66 W + 51 for the scram (where W = total recirculation flow). The maximum flow bias scram setting is clamped at 113.5% of rated power even if recirculation flow is in excess of 100%. The rod block flow bias trip is clamped at a maximum value of 108%.

5.4.2.5 Trip Units

Various trip units and their functions are provided by the APRM's as shown in Figure 5.4-3. Recirculation System total flow is used to bias the trip reference for the high and high-high (thermal) trip units so that it varies with total recirculation flow. The flow biased high trip reference is also referred to as the APRM alarm level. The high and high-high trip units have reduced trip references when the mode switch is not in run. The high-high (thermal) trip unit incorporates a time delay circuit of six seconds, to approximate the thermal time constant of the fuel. This time delay feature allows the plant to avoid spurious trips due to minor neutron flux overshoots.

5.4.3 System Features and Interfaces

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

5.4.3.1 Rod Blocks and Scram Functions

Rod withdrawal blocks are provided via the RMCS by the inoperable, downscale and high trip units, and by the recirculation flow converter upscale trip. These blocks are in effect in all modes of operation except for the downscale trip which is only in effect in the run mode. Note that the upscale high trip setpoints change with mode switch position. Reactor scrams are provided via the RPS by the inoperable, high-high (thermal) and high-high trip units. The high-high (thermal) trip is effective only in the run mode; the others are effective in all modes. The high-high trip unit setpoints change with mode switch position. These trips, along with the associated bypasses are listed in Table 5.4-1.

5.4.3.2 System Interfaces

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

Intermediate Range Monitoring System (Section 5.2)

The Intermediate Range Monitoring System contributes to the APRM downscale scram signal and shares recorders.

Local Power Range Monitoring System (Section 5.3)

The LPRM detectors provide the radial and axial flux distribution used to determine percent core average thermal power.

Reactor Protection System (Section 7.3)

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

Rod Block Monitoring System (Section 5.5)

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

Reactor Manual Control System (Section 7.1)

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

Recirculation System (Section 2.4)

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

5.4.4 BWR Differences

A number of design differences exist between the BWR/2 through BWR/6 reactors even within the respective classes. Some of the major differences in APRM Systems are listed below.

5.4.4.1 Number of LPRM Inputs

The number of LPRM inputs to each APRM channel is a variable and is core size dependent. For example:

Plant	No. of LPRM Strings
Browns Ferry	43
Hatch	31
Quad Cities	41
Grand Gulf	44

5.4.4.2 Number of APRM Channels

The number of APRM channels may vary from four to eight.

5.4.4.3 Recirculation Flow Comparators

Some BWR reactors are equipped with two flow units rather than four. Some facilities compare the output of the flow units to one another, in flow comparators, and if an error of more than 10% is detected the flow converter trip units and cause a rod block.

5.4.4.4 Flow Bias Setting

The formula used by the slope and bias circuit to determine trip points may vary slightly. Also, some APRMs may not have a fixed 120% scram setpoint. These units would have only the flow biased trips.

5.4.4.5 Method of APRM Bypass

BWR plants that have a solid state logic Reactor Protection System have sensor bypass switches that bypass all sensors within a given division (normally four divisions).

5.4.5 Summary

Classification - Safety related system.

Purpose - To compute the core average thermal power. To provide trip signals to preserve the integrity of the fuel cladding integrity.

Components - LPRM inputs; averaging circuits; count circuit; slope and bias circuit; trip units.

System Interfaces - Local Power Range Monitoring System; Intermediate Range Monitoring System; Rod Block Monitoring System; Reactor Protection System; Reactor Manual Control System; Recirculation System.

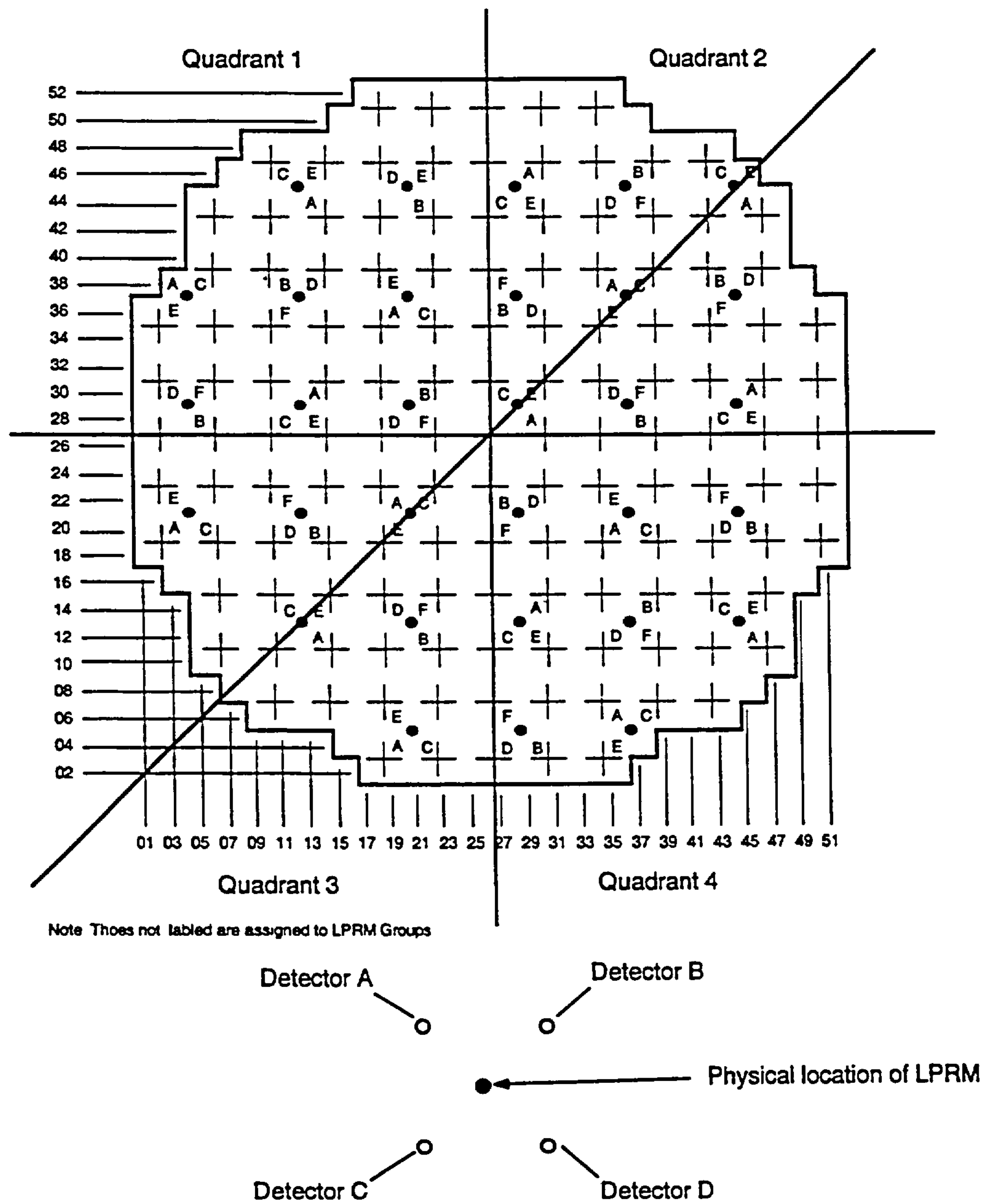

Figure 5.4-2 LPRM Detector Assignment for Division One APRM's

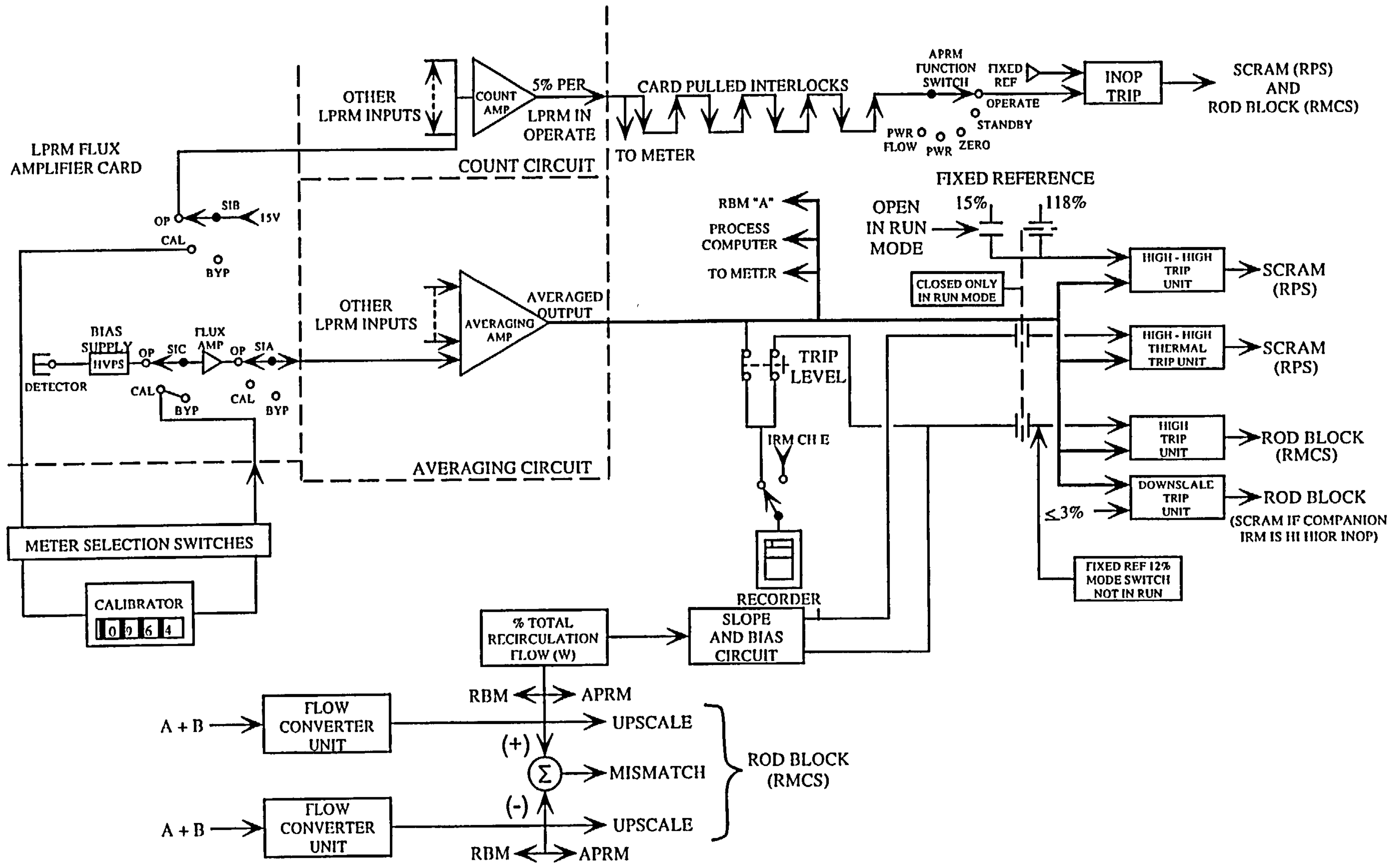

Figure 5.4-3 APRM Channel Block Diagram

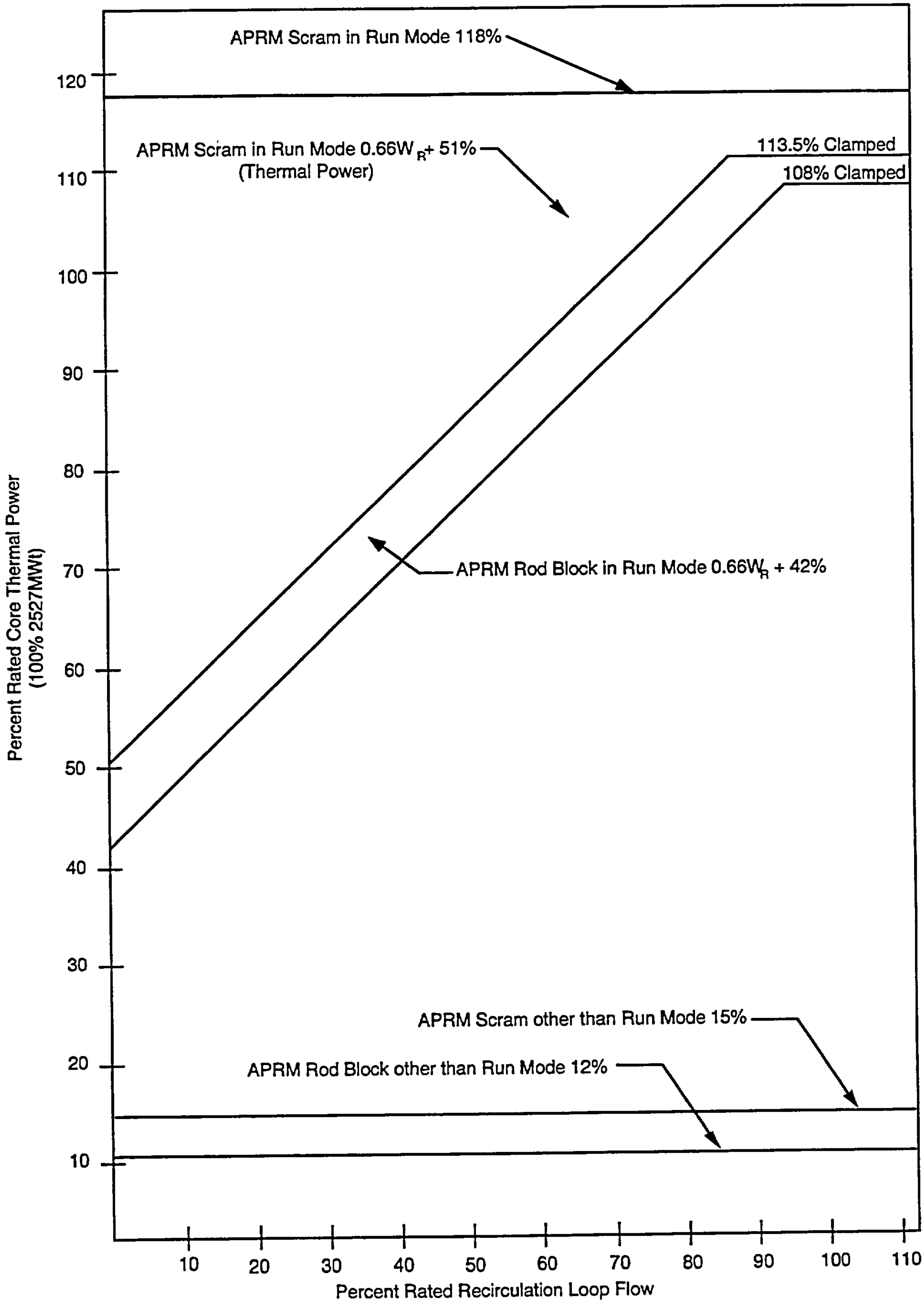

Figure 5.4-4 APRM Trip Levels

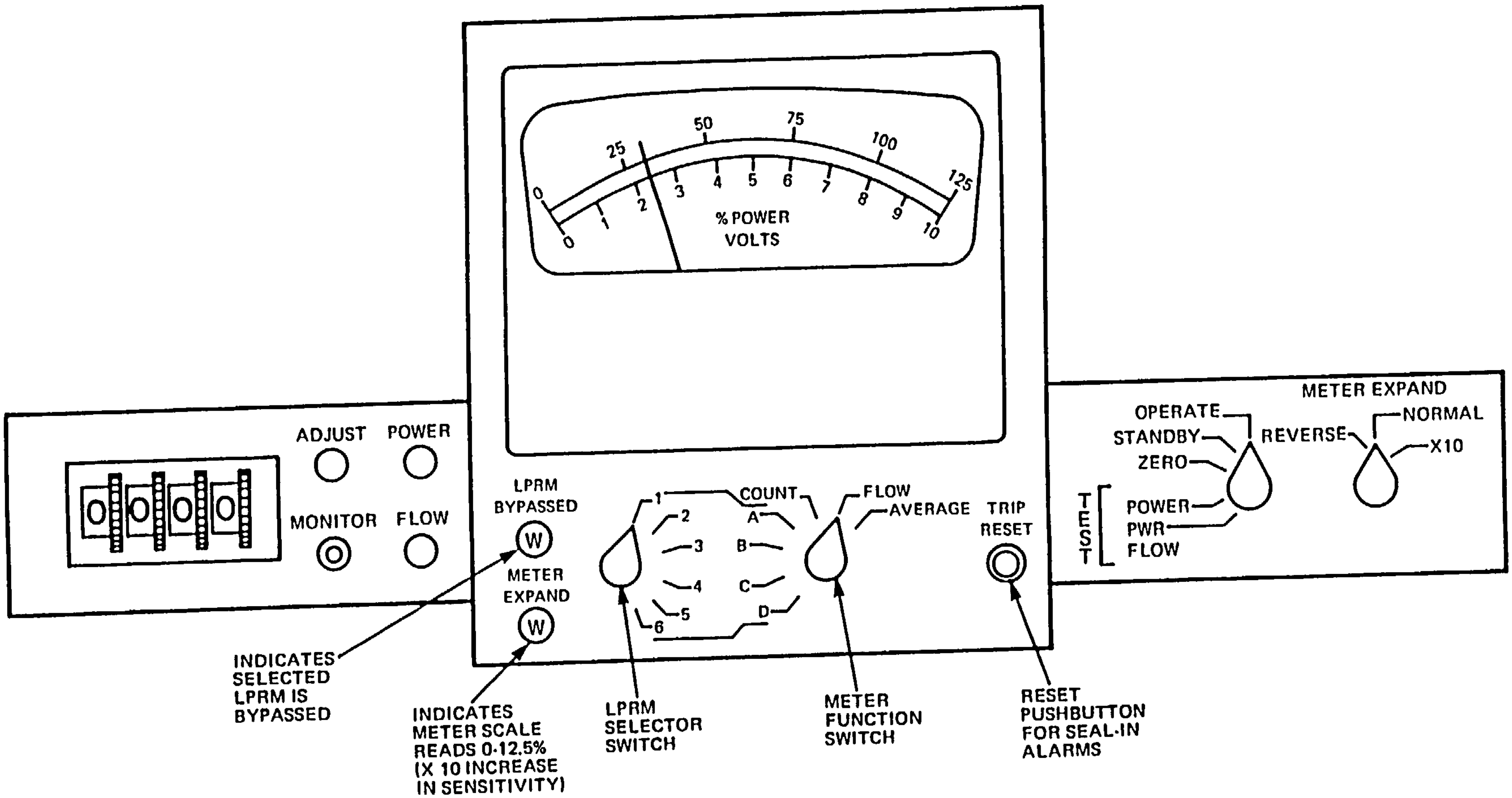

Figure 5.4-5 APRM Functional and Test Switches