CHAPTER 7 - INSTRUMENTATION AND CONTROLS

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7.1 INTRODUCTION

7.1.1 Identification of Safety Related Systems

Instrumentation and controls for the following safety related systems are described in Chapter 7.

Reactor Protection System - (Sections 7.2 and 7.3)

The Reactor Protection System furnishes the signals to trip the reactor (Section 7.2) and to initiate certain Engineered Safety Feature Systems (Section 7.3). Reactor Protection System action prevents fuel damage, limits steam pressure, and prevents or restricts the release of radioactive materials.

Alternate Rod Injection System - (Section 7.3)

This system provides a means diverse from the Reactor Protection System for depressurizing the scram header in the unlikely event the Reactor Protection System does not cause a reactor scram in response to an operational transient.

Reactor Vessel (Main Steam/Isolation) - (Section 7.3)

The reactor is isolated by signals from the Reactor Protection System to limit the loss of reactor coolant and the release of radioactive materials in the event of a steam line break outside containment.

Primary Containment (Drywell) Isolation - (Section 7.3)

The drywell isolates under certain conditions to contain any activity which might escape the reactor and to limit the release of radioactive materials in the event of a line break. Radioactivity will be maintained below the limits allowed by 10CFR100.

Isolation Condenser System - (Section 7.3)

This high pressure passive system is used to remove reactor core decay heat in the event the main condenser is not available as a heat sink following a reactor scram.

Core Spray System - (Section 7.3)

This low pressure system provides reactor core cooling following a Loss-of-Coolant Accident to prevent fuel cladding failure from high temperature.

Standby Gas Treatment System - (Section 7.3)

This system is initiated by the Reactor Protection System (RPS), and also by non-RPS instrumentation during abnormal conditions, to limit radioactive material release.

Containment Spray System - (Section 7.3)

This system reduces the containment pressure and temperature following a Loss-of-Coolant Accident and serves to reduce the release of radioactive materials by reducing the pressure differential between the containment and the external environment.

Emergency Service Water System - (Section 7.3)

This system provides the cooling water to remove heat from the Containment Spray System.

Automatic Depressurization System - (Section 7.3)

This system provides for a controlled blowdown of the reactor pressure vessel to rapidly reduce pressure during a small pipe break. This is to permit core spray actuation prior to uncovering fuel.

Standby Liquid Control System - (Section 7.4)

This system shuts down and holds the reactor subcritical in the unlikely event of a control rod system failure.

Safety Related Display Instrumentation - (Section 7.5)

Section 7.5 is devoted to a discussion of the control panels in the Control Room which contain the indicators, recorders, annunciators, switches, lights, and other instruments required to run the plant safely during all phases of operation.

Neutron Monitoring System - (Section 7.5)

This system provides indication of reactor power during all conditions of plant operation, and provides trip inputs to the Reactor Protection System.

Reactor Vessel Instrumentation - (Section 7.6)

This system provides indication of reactor temperature, pressure, level, flow core differential pressure and core spray pipe break, and provides trip inputs to the Reactor Protection System. A subsystem provides position indication for the relief and safety valves.

Recirculation Pump Trip System - (Section 7.6)

The Reactor Protection System trips the reactor recirculation pumps on low low reactor water level. On a high reactor pressure signal, recirculation pumps A, B, and E are tripped, while recirculation pumps C and D will trip within 12 seconds on a persistent high pressure signal. (Time delay setpoints will provide margin for calibration and accuracy of the time delay relays). This permits proper operation of the ICS and minimizes the effects of an Anticipated Transient Without Scram (ATWS). Tripping the reactor recirculation pumps reduces reactor power levels thereby decreasing pressure and temperature transient during an ATWS event. Other trips are provided for equipment protection.

Leak Detection System - (Section 7.6)

This system provides the means to detect small leaks in the Reactor Coolant Pressure Boundary.

Containment Pressure, Water Level, and H_2/O_2 Monitoring System - (Section 7.6)

This system provides safety grade instrumentation for measurement of containment parameters during and following a Loss-of-Coolant Accident.

All instrumentation for those systems noted above was supplied by the General Electric Company. The Relief Valve/Safety Valve Position Indicating System was supplied by Babcock and Wilcox, and the Containment Pressure, Water Level and H_2/O_2 Monitoring System was supplied by Comsip. The airborne particulate monitoring portion of the Leak Detection System was supplied by Comsip.

Some transmitters, monitoring equipment, and Control Room recorders associated with various systems have been changed from those originally supplied.

Suppression Pool Temperature Monitoring System (SPTMS) - (Section 7.6)

The SPTMS provides safety grade instrumentation for indication of the Torus bulk water temperature during both normal and postaccident plant operating conditions. Suppression Pool Water temperature measurement provides plant operators with the necessary information for initiating appropriate actions to assure the integrity of the primary containment system.

7.1.2 Identification of Safety Criteria

7.1.2.1 Design Bases

The design basis for each system as required by Section 3 of IEEE 279 is discussed in each individual Section.

7.1.2.2 Title 10, Code of Federal Regulations, Part 50

Table 7.1-1 lists the General Design Criteria which were reviewed for applicability to the systems listed in Subsection 7.1.1. A discussion of the General Design Criteria is provided in Section 3.1. A discussion of design environmental conditions is provided in Subsection 3.11.

7.1.2.3 NRC Regulatory Guides

The regulatory guides examined for applicability to the safety related systems are listed in Table 7.1-2. A discussion of applicable regulatory guides is contained in each individual section. However, the discussions regarding Regulatory Guides 1.11 and 1.63 are contained in Section 1.8

7.1.2.4 Industry Standards

OCNGS was built and operational prior to the issuance of most IEEE Standards, including IEEE 279. Each Section of Chapter 7 gives a comparison of Oyster Creek design relative to IEEE 279. The discussion of IEEE 308 is provided in Chapter 8.

7.1.3 References

(1) GPUN Letter, "Topical Report 028, Rev 2, OC Response to US NRC Regulatory Guide 1.97, dated April 5, 1991.

TABLE 7.1-1
(Sheet 1 of 1)

GENERAL DESIGN CRITERIA REVIEWED FOR
APPLICABILITY TO INSTRUMENTATION SYSTEMS

System	GENERAL DESIGN CRITERIA																					
	1	2	3	4	10	12	13	19	20	21	22	23	24*	25	26	27	28	29	34	40	46	55	64
Reactor Protection System	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X				X	
Reactor Vessel Isolation System	X	X	X	X			X		X	X		X	X					X					
Primary Containment Isolation System	X	X	X	X			X		X	X		X	X					X					
Isolation Condenser System	X	X	X	X			X		X	X		X	X					X					
Core Spray System	X	X	X	X					X	X		X	X					X	X	X			
Standby Gas Treatment System	X	X	X	X					X	X	zx	X	X					X					
Alternate Rod Injection System	X	X	X	X					X	X		X	X					X					
Containment Spray System	X	X	X	X															X	X			
Emergency Service Water System	X	X	X	X															X	X	X		
Automatic Depressurization System	X	X	X	X						X													
Standby Liquid Control System	X	X	X	X											X	X							
Remote Shutdown System	X	X	X	X				X															
Safety Related Display Instrumentation	X	X	X	X				X															
Neutron Monitoring System	X	X	X	X	X	X	X																
Reactor Vessel Instrumentation	X	X	X	X			X															X	
Recirculation Pump Trip System	X	X	X	X	X	X	X																X
Leak Detection System	X	X	X	X																			
Containment Pressure, Water Level, and $H_2/0_2$ Monitoring System	X	X	X	X																		X	X

* Certain exceptions in Control Room panels

X indicates Applicable

TABLE 7.1-2
(Sheet 1 of 1)

REGULATORY GUIDES REVIEWED FOR
APPLICABILITY TO INSTRUMENTATION

System	Regulatory Guides											
	1.22	1.29	1.45	1.47	1.53	1.62	1.75	1.89	1.97*	1.100	1.105	1.118
Reactor Protection System	X	X		X	X	X	X	X		X	X	X
Reactor Vessel Isolation System	X	X		X	X	X	X	X		X	X	X
Primary Containment Isolation System	X	X		X	X	X	X	X	X	X	X	X
Isolation Condenser System	X	X		X	X	X	X	X	X	X	X	X
Core Spray System	X	X		X	X	X	X	X	X	X	X	X
Standby Gas Treatment System	X	X		X	X	X	X	X	X	X	X	X
Alternate Rod Injection System	X	X		X	X	X	X	X		X	X	X
Containment Spray System	X	X		X	X	X	X	X	X	X	X	X
Emergency Service Water System	X	X		X	X	X	X	X	X	X	X	X
Automatic Depressurization System	X	X		X	X	X	X	X		X	X	X
Standby Liquid Control System	X					X			X			
Safety Related Display Instrumentation								X		X		
Neutron Monitoring System	X	X		X	X	X	X	X	X	X	X	X
Reactor Vessel Instrumentation		X						X	X	X		
Recirculation Pump Trip System	X	X		X	X	X	X	X		X	X	X
Leak Detection System			X				X					
Containment Pressure, Water Level, and H_2/O_2 Monitoring System		X						X	X	X		

* See Reference 7.1.3(1)

7.2 REACTOR TRIP SYSTEM

The Reactor Protection System provides reactor trip signals, Engineered Safety Feature Systems actuation signals and other safety related signals. The reactor trip function is discussed in this section. Engineered Safety Feature Systems actuation is discussed in Section 7.3. Other safety related functions are discussed in Section 7.6.

7.2.1 Description

7.2.1.1 System Description

The Reactor Protection System (RPS) is a dual logic channel system. Each channel has an independent source of ac power, fail safe design and high reliability in initiating protective actions and preventing spurious trips. Each independent logic channel has two subchannels of tripping devices. Thus the system has a total of four independent subchannels. Each subchannel typically has an input from an independent sensor monitoring each of the critical parameters.

Exception to the above logic is discussed in Section 7.2.1.1.2.

Fail safe means that RPS relays are normally energized and loss of power or component failure results in trip initiation.

The RPS monitors plant parameters and automatically initiates protective action(s) if established limits are exceeded. The RPS acts to protect the core against fuel rod cladding damage and to protect the reactor vessel from overpressure. The System Block Diagram is shown in Figure 7.2-1. RPS protection actions are:

a. Reactor Trip
b. Rod Withdrawal Block (see Section 7.7)
c. Core Spray Cooling Initiation (see Section 7.3)
d. Reactor Vessel Isolation (Main Steam Isolation) (see Section 7.3)
e. Primary Containment Isolation (Drywell Isolation) (see Section 7.3)
f. Turbine Trip (see Section 7.7)
g. Recirculation Pump Trip (see Section 7.6)
h. Offgas System Isolation (see Section 11.5.2.4)

7.2.1.1.1 Initiating Circuits

The conditions that initiate a reactor trip are described below and are listed in Table 7.2-1.

a. Neutron Monitoring System - This trip signal limits the heat flux to a level well below that which could cause fuel damage. Four Average Power Range Monitors (APRM) channels are connected in each of the two RPS channels. The Neutron Monitoring System discussed in Section 7.5.

b. High Reactor Pressure - Reactor vessel pressure is monitored by electronic transmitters and analog electronic trip systems.

c. Low Reactor Water Level - Low water level is monitored by electronic transmitters and analog electronic trip systems.

d. Scram Discharge Volume High Water Level - This trip signal shuts down the reactor while there is still sufficient free volume in the scram discharge system to receive the control rod drives discharge upon trip. Adequate allowance is made for the closed gas volume above the water.

e. High Primary Containment Pressure - The primary containment (drywell) pressure is monitored by pressure switches to detect Loss-of-Coolant Accident (LOCA) conditions inside containment in order to minimize the energy that must be accommodated by the emergency cooling systems.

f. Condenser Low Vacuum/Turbine Trip - These trip signals are provided by limit and pressure switches on turbine equipment to anticipate loss of the main heat sink. (Refer to UFSAR Section 7.7.1.5)

g. Closure of Main Steam Isolation Valves - This trip signal limits the release of fission products from the Reactor Coolant System. The Main Steam Isolation Valves (MSIVs) automatically close when conditions indicating a steam line break occur. (Low steam pressure, high steam line tunnel temperature, high steam flow.) The MSIVs also close on reactor vessel low-low level.

h. Recirculation Flow Monitoring Inoperative – Trip signal initiated on recirculation flow being downscale. One inoperative signal provided for each RPS channel.

i. Loss of all ac power to the Reactor Protection System - All trip sensor relays, logic relays and solenoid valves will trip due to loss of power, as protection system Motor-Generator (M-G) sets coast down. This is a fail safe design discussed in Subsection 7.2.1.1.

j. Reactor Mode Switch in shutdown position - If the keylock mode switch is turned to "SHUTDOWN" the reactor will trip. This reactor scram signal can be bypassed as a result of the presence of temporary jumpers. The controlled bypassing of the reactor scram can only be initiated with the approval of the Plant Operations Director, or his designee. The scram signal will be jumpered for approximately 60 seconds while the reactor mode switch is transferred from the refuel position to the shutdown position.

k. Manual - Both channels are tripped by two manual push buttons. Separation of the manual trips in the two channels is to provide isolation and to allow individual testing.

l. Deleted.

7.2.1.1.2 Logic

The system is made up of two independent logic channels, each logic channel having two subchannels of tripping devices. Thus, the system has a total of four independent subchannels. Each subchannel has an input from an independent sensor monitoring each of the critical parameters. The outputs of two of the subchannels are combined in a one out of two logic, that is, an input signal on either one or both of the independent subchannels will produce a logic channel trip. The outputs of the remaining two subchannels are likewise combined in a one out

of two logic, independent of the other logic channel. The outputs of the two logic channels are combined so that they must be in agreement to initiate a trip. In this way, an out-of-limits signal initiated by either one or both of the independent sensors in the two subchannels paired into one logic channel must be confirmed by an out-of-limits signal initiated by either one or both of the independent sensors in the two subchannels paired into the other logic channel to provide a trip.

The condenser low vacuum scram logic differs from the logic of the other RPS initiating circuits. Each of the three condenser sections has two bellows vacuum instrument channels. Two vacuum trip systems are formed by using one vacuum channel from each condenser as an input in a 1 out of 3 logic. The output of each vacuum trip system provides actuation to both RPS channels to provide for a full scram. A single failure would not prevent a scram on low vacuum. However, if any 1 of the 6 vacuum channels was to inadvertently actuate, a full scram would result.

The Oyster Creek Technical Specifications Table 3.1.1 Protective Instrumentation Requirements item A.6 only requires 1 vacuum trip system with the 3 respective vacuum channels to be operable. Operation in this mode does not meet single failure criteria since the failure of a single vacuum channel could prevent a scram if vacuum was lost in that condenser section. The justification for the reduced availability requirement is that the condenser low vacuum trip is not required for safety or for reactor protection and, therefore, the reduced availability will not adversely affect safety. During normal operation, all sensor and trip contacts are closed, and all vital relays are operated energized. All scram pilot valve solenoids are energized, and instrument air pressure is applied to all the scram valves. There is redundancy in the system, such that the failure of any one device to perform its function will not prevent a trip. Nor will the spurious operation of any one device initiate a false trip. The Reactor Protection System is in operation at any time when nuclear fuel is in the vessel.

In addition, the following are exceptions to the typical RPS initiation logic:

Loss of all AC Power to the RPS – This trip does not involve monitoring of critical parameters with sensors. This trip is just the fail safe nature of the system design where loss of power causes a rely to deenergize which also occurs as the result of a trip signal.

Reactor Mode Switch in “Shutdown” Position – This single switch has contacts in both RPS channels. There are no subchannels or sensors involved in this trip function.

Manual Trip – This trip function requires pressing two independent switches. Each switch trips one channel of RPS. As was the case with the reactor mode switch, there are no subchannels or sensors involved in this trip function.

Recirculation Flow Monitoring Inoperative – This function detects a total recirculation flow signal being downscale. There are two independent inoperative trips (one for each RPS channel). Each inoperative trip results in a trip of both subchannels of an RPS channel, so instead of one-out-of-two twice, this function is best defined as one-out-of-one twice, or two-out-of-two. This scram function is not relied upon to perform an accident mitigation function.

Some functions have two independent sensors per subchannel providing an extra degree of redundancy. These include APRMS and IRMs for the neutron monitoring system (eight channels each), SDIV high level (one-out-of-two twice for each of two

SDIVs), and the MSIV closure limit switches (two scram limit switches for each of four MSIVs).

7.2.1.1.3 Bypassing

Bypassing of trip functions occurs automatically under certain plant conditions. However, certain trip functions can be manually bypassed. Trip functions and permitted bypasses are listed in Table 7.2-2.

7.2.1.1.4 Interlocks

Interlocks are provided in the Neutron Monitoring System (NMS) trip circuitry to trip an RPS subchannel whenever an NMS module is INOP, i.e., removed from its mounting or not in operate mode, or when the required minimum number of inputs to an APRM channel are not present.

Bypassing of NMS channels (where permitted) is accomplished by joysticks which mechanically and electrically prevent bypassing more than one channel at a time per logic channel or core quadrant.

7.2.1.1.5 Redundancy

All functions of the RPS are implemented by redundant sensors, logic, and actuation devices.

7.2.1.1.6 Diversity

The RPS trip circuits have been designed with diverse trip functions to protect the integrity of the fuel cladding and Reactor Coolant System barriers.

7.2.1.1.7 Actuated Devices

Reactor trip is accomplished through two scram valves on each control rod drive. (See Figure 7.2-2.) The valves are air to close, spring to open. The instrument air header normally supplies air to the valves through the energized scram pilot solenoid valves to hold the valves shut. During a trip the scram pilot solenoid valves deenergize and vent the air lines causing the scram valves to open. One valve opens faster, venting the over piston area to the discharge volume. A second valve opens later, applying accumulator pressure to the under piston area, inserting the control rod.

Two backup scram solenoid pilot valves are provided. Upon receipt of a trip signal these valves block the air supply and go to the vent position. Because of the long bleeddown, these valves cannot adequately cause a trip by themselves. However, if a scram pilot or discharge volume pilot failed to vent, the backup valves would insure that all scram valves open and the vent and drain valves close.

Two discharge volume pilot solenoid valves are deenergized simultaneously with the backup scram solenoid pilot valves. These valves vent the actuators on the drain and vent valves, closing them.

7.2.1.1.8 Mode Selector Switch

There are four principal modes of operation of the Mode Selector Switch: RUN, STARTUP, REFUEL and SHUTDOWN. Each operating mode has its own individual restrictions for safe operation. The Mode Selector Switch is provided to ensure that all these restrictions are imposed at the proper time, and to ensure that the transition from one mode to another is also safe. Table 7.2-3 lists the modes and the permissible operations in each mode if all RPS parameters are within established limits.

7.2.1.1.9 Information Display

There are no analog displays in the RPS since it is a relay logic system.

Displays associated with the Neutron Monitoring System are discussed in Section 7.5.

Local indication associated with reactor vessel instrumentation is discussed in Section 7.6. Reactor Protection System annunciators are provided in the Control Room as shown in Figure 7.2-3. These annunciators keep the operator informed of the status of the Reactor Protection System at all times.

7.2.1.1.10 Power Supplies

Redundant Class IE protection provided for the RPS power supply. The RPS power is supplied through two independent buses (Protection System Panels No. 1 and 2). Each panel supplies power to one logic channel and its respective pilot and backup scram valve solenoids, one half of the incore flux amplifiers, one half of the steam line radiation monitors, and one half of the flux amplifier summers.

The normal power supply to Protection System Panel No. 1 (No. 2) is from 4160 volt bus 1A(1B) to 4160 volt emergency bus 1C(1D), then through a transformer to 460 volt substation 1A2(1B2) to VMCC 1A2(1B2) to M-G set 1-1 (1-2) which supplies 120 volt single phase power to Protection System Panel No. 1 (2). See Table 7.2-4 for M-G set specifications. The motor generator set is equipped with a flywheel to provide inertial smoothing of switching transients upstream of the motor.

An alternate source of power is available to either channel of the RPS when a motor generator set is out of service. This source takes power from either vital motor control center 1A2(1B2) at 460 volts ac, steps the voltage down to 115 volts, and supplies power to either bus 1 or 2 by means of selector switches in the Control Room. If normal ac power to the RPS M-G set is lost, emergency power is supplied from the Emergency Diesel Generators, which feed vital motor control centers 1A2 and 1B2. Shifting of RPS power sources must be a dead bus transfer. Caution must be taken during this transfer, since a half trip (and isolation) condition will exist and any single trip signal on the operating channel will result in a reactor trip.

7.2.1.1.11 Reset Circuitry

Once a logic channel trip or a reactor trip is initiated, contacts in the scram relays 1K51(2K51) and 1K52(2K52) open and keep the circuit deenergized until it is reset manually. Reset is accomplished by push button 3S1, which energizes reset relays 1K61(2K61) and 1K62(2K62). The contacts of the reset relays momentarily close and bypass the seal-out contacts of the scram relays, allowing the relays to energize provided all RPS scram inputs have cleared.

Scram reset due to MSIV closure and or low condenser vacuum signals can be initiated without reducing the reactor pressure below 600 psig provided the Reactor mode switch is in the SHUTDOWN position only.

7.2.1.2 Design Bases Information

Conformance with Section 3 of IEEE 279

a. The generating station conditions that require protective action are:

1. Preservation of Fuel Cladding Integrity

The fuel cladding represents one of the primary physical barriers which separates radioactive material from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative, continuously measurable and tolerable. Fuel cladding perforations, however, could result from thermal effects if reactor operation is significantly above design conditions and the associated protection system setpoint. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal conditions may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which may result in cladding perforation.

A critical heat flux occurrence results in a decrease in heat transferred from the clad and, therefore, high clad temperatures and the possibility of clad failure.

However, the existence of a critical heat flux occurrence is not a directly observable parameter in an operating reactor. Furthermore, the critical heat flux correlation data which relates observable parameters to the critical heat flux magnitude is statistical in nature.

The margin to boiling transition is calculated from plant operating parameters such as core pressure, core flow, feedwater temperature, core power, and core power distribution. The margin for each fuel assembly is characterized by the Critical Power Ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR).

2. Reactor Overpressurization

The Reactor Coolant System represents an important barrier in the prevention of the uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1375 psig was derived from the design pressures of the reactor pressure vessel, coolant piping, and Isolation Condensers. The respective design pressures are 1250 psig at 575°F, 1200 psig at 570°F and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section I for the pressure vessel; ASME Boiler and Pressure Code, Section III, for the Isolation Condensers; and the ANSI Piping Code, Section B31.1, for the Reactor Coolant System piping. The ASME Code permits pressure transients up to 10% over the design pressure (110% x 1250 = 1375 psig) and the ANSI Code permits pressure transients up to 15% over the design pressure (115% x 1200 = 1380 psig).

Trips and trip setpoints are established to provide the necessary protection so that MCPR and pressure limits are not exceeded. Subsection 7.2.1.1.1 provides a description of each RPS trip.

b. The generating station variables that provide reactor trips are shown in Table 7.2-1.

c. The requirements for the minimum number and location of sensors required to monitor adequately, for protective function purposes, those variables that have spatial dependence are:

   1. Failure of four chambers assigned to any one APRM channel shall make the APRM channel inoperable.

   2. Failure of two chambers from one radial core location in any one APRM channel shall make that APRM channel inoperable.

   3. Any two Local Power Range Monitor (LPRM) assemblies which are input to the APRM System and are separated in distance by less than three times the control rod pitch may not contain a combination of more than three inoperable detectors (i.e., APRM channel failed or bypassed, or LPRM detectors failed or bypassed) out of the four detectors located in either the A and B, or the C and D levels.

   4. A Traversing Incore Probe (TIP) chamber may be used as an APRM detector input to meet the criteria of c.1, c.2 or c.3 provided the TIP is positioned in close proximity to one of the failed LPRM's. If the criteria of c.2 or c.3 cannot be met, power operation may continue at up to rated power level provided a control rod withdrawal block is operating, or at power levels less than 61% of rated power until the TIP can be connected, positioned and satisfactorily tested, as long as Technical Specifications are satisfied.

d. For the prudent operational limits for each variable in each operation, refer to the Technical Specifications.

d. For the margin between each operational limit and level marking onset of unsafe conditions, refer to the Technical Specifications.

e. For the level that, when reached, will require protective action, refer to Table 7.2-1 (approximate values) and to Standing Order No. 1 (exact values).

f. The range of transient and steady state conditions of the energy supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform has been determined, as discussed in Section 3.11.

g. The malfunctions, accidents, or other unusual events that could physically damage protection system components, for which provisions must be incorporated to retain necessary protection system action are as follows:

    1. All instruments used in the protection system and engineered safeguards are designed to operate under the most unfavorable environmental conditions that can reasonably be associated with an abnormal or accident condition. For conventional devices, the manufacturer's quality control, initial calibration, plant preoperational testing and plant maintenance procedures are relied upon to obtain a high degree of reliability.

    2. All sensing elements, except neutron detectors, are located outside the drywell, where they are not exposed to unfavorable environment. They are equipped with weatherproof enclosures which can withstand, at least on a temporary basis, such conditions as might result from a steam or water line break outside the drywell. In addition, the sensors are located in such a way that no single event is likely to affect both sensors in one protection logic system. The scram discharge volumes high level switches and transmitters may be considered an exception to this general separation statement. As these detectors are called upon to operate following all scrams, operations are individually recorded. These switches and transmitters are built to withstand full system pressure and temperature, and are intended primarily to prevent startup of reactor prior to draining the scram discharge system for the control rod drive. Since the discharge system is normally open to drain, and position switches provided on valves, isolation of this system requires multiple failures and complete disregard by the operator to create any possible condition which could be construed to be dangerous. The reference column with the auxiliary reservoir has been subjected to carefully controlled and documented tests to prove that the reference column remains full under blowdown conditions similar to those postulated for a Loss-of-Coolant Accident in the reactor vessel.

i. The minimum performance requirements, including system response times, ranges of the magnitudes, and rates of change of sensed variables to be accommodated until proper conclusion of the protection system action, are presented in the Technical Specifications.

7.2.1.3 Final System Drawings

The RPS logic is shown in Drawing GE237E566

7.2.2 Analysis

The RPS is made up of two independent logic channels, each having two subchannels of tripping devices. Thus, the system has four independent subchannels. Each subchannel has inputs from independent sensors monitoring each of the critical parameters. One of the parameters monitored by each subchannel sensor, any one off-standard or out-of-limits condition which initiates a sensor trip will de-energize the subchannel scram relay (i.e., relays 1K51, 1K52, 2K51, or 2K52 on Drawing GE237E566), which will in turn de-energize one half of the pilot scram valve solenoids (one on each rod).

This condition, with one half of the pilot scram valve solenoids de-energized, is known as a 1/2 scram. Two logic channel trips are required to produce a reactor trip; therefore, the minimum requirement for a trip is two subchannel trips (one per logic channel). With the one-out-of-two-twice logic of the Reactor Protection System, a single component failure, or the spurious trip of a single sensor, will cause a logic channel trip but will not cause a reactor trip. In addition this logic will not prevent a reactor trip when a second sensor trip occurs in the remaining logic channel. Thus, one can tolerate a single component failure in the RPS and still operate the reactor safely.

Exceptions to the above logic are discussed in Section 7.2.1.1.2

The entire system, except for the main condenser low vacuum scram can be tested during plant operation. Theoretically, the one-out-of-two-twice logic of the RPS is slightly more reliable than a two-out-of-three and slightly less reliable than a one-out-of-two system. The dual-logic-channel protection system facilitates more testing during full power operation in comparison to the one-out-of-two system. The thorough and frequent testing significantly increases the reliability of the system.

The RPS is designed to "fail safe" for the most probable failure mode. In every case except sensor failure, a safe failure is annunciated so that the location of the failure can be readily ascertained.

The term "fail-safe" as applied to the RPS refers to the condition of the system component in its failure mode. For example, the RPS relays are normally energized when the plant is operating within the established safe limits; if a relay coil were to fail or a subchannel were to lose power, the relay or relays would be deenergized and their open contacts would cause a logic-channel trip. Therefore, when an RPS component fails, it is designed to fail into a safe, or trip mode.

7.2.2.1 Conformance with Section 4 of IEEE-279

OCNGS was built and operational prior to the issuance of IEEE 279. The following discussions are keyed to Section 4 of IEEE 279 and provide a comparison of Oyster Creek RPS design with this standard:

(4.1) General Functional Requirements

The RPS automatically performs its protective function of tripping the reactor whenever plant conditions exceed preset levels under the design conditions discussed in Subsection 7.2.1.1.

(4.2) Single Failure Criteria

No single failure can prevent the RPS from performing its protective function. However, the condenser low vacuum scram does not meet this criteria if only 1 trip system is operable.

This is permitted by the tech specs since the low condenser vacuum trip is not required for safety or for reactor protection. See Section 7.2.1.1.2 for further discussion of this logic.

(4.3) Quality of Components and Modules

Materials used in the control and instrumentation for the Reactor Protection System, and other engineered safeguards (with the exception of insulating material, protective coatings, etc.), are noncombustible or highly fire resistant. The panel wiring and phenolic moulded parts of relays and other components are fire resistant.

Some information readout devices have combustible plastic cases and fronts (e.g., GE type 180 meters) but these devices are not active in protective functions and are considered to be sufficiently protected from ignition sources that they do not constitute a safety hazard.

All instrumentation and control components and/or assemblies important to safety have been designed, manufactured and installed to criteria that require the facility to withstand an earthquake giving a floor acceleration of 0.22g horizontally without loss of capability to fulfill essential functions. Panel and rack structures were examined and judged to be rigid over the frequency range of the assumed earthquake time history acceleration peaks. The "fail-safe" nature of the Reactor Protection System assures initiation of a trip under design basis earthquake conditions.

Information presented above describes past compliance with Section 4 of IEEE 279. However, beginning in September 1995, instrumentation and control components are analyzed, evaluated and designed using the new EQE response spectra and the analytical methodology described in Section 3.7.3.4.

(4.4) Equipment Qualification

Equipment qualification is discussed in Sections 3.10 and 3.11.

(4.5) Channel Integrity

Each RPS channel is designed and fabricated so that the channel integrity will be maintained under the conditions specified in the design bases for the system.

(4.6) Channel Independence

All wiring from scram trip sensors is run in rigid metallic conduit which is used exclusively for protection system wiring. The two logic channels (except for anticipatory trips discussed below) are in separate conduits. Subchannels of the two protection system logic channels are in separate jacketed cables and connected in such a way that shorting of wires within one or both cables will not disable the scram sensors. Insulation breakdown of a cable through its jacket should cause grounding to the conduit which will produce a channel trip. Breakage of wires from any cause will result in a channel trip.

Wiring associated with the anticipatory trip functions (Low Condenser Vacuum, Turbine Trip, and Generator Load Rejection) is primarily non-jacketed cables. In some locations, the two logic channels are in the same conduit.

Lack of Channel Independence was inherent in the original design of the Low Condenser Vacuum circuit. The Turbine Trip and Generator Load Rejection Scram functions were later additions to the Anticipatory Trip Circuits as part of the thermal power uprate from 1600 to 1690 MW. By incorporation into the Low Condenser Vacuum scram circuits, these additional anticipatory trips are subject to the same limitations of physical separation inherent in the original design of the Low Condenser Vacuum scram circuit. As discussed in 7.7.1.5, if a scram was not initiated by a Turbine Trip or Generator Load Rejection, the resultant increase in reactor power would result in a trip from the pressure increase directly.

Sensors of reactor level, reactor pressure and drywell pressure are located outside the drywell approximately 20 feet apart in pairs such that failure of any one pair in one location will not prevent ability to scram or to initiate automatic safeguards.

The scram solenoid pilot valves are powered from eight circuits, four circuits from each of the two scram contactors going to the four A and four B groups.

These eight circuits are kept isolated. The final logic decision to scram individual rods is made in the instrument air circuit controlled by the scram pilot valves and the decision is made independently for each and every control rod. Any malfunction in the individual rod scram valves or pilots will not affect any other control rod.

(4.7) Control and Protection System Interaction

The Neutron Monitoring System is considered part of the RPS and meets all the requirements of the RPS.

Reactor vessel instrumentation (described in Section 7.5) provides analog indication and recording in addition to RPS trip functions. The analog display instrumentation is separate from the trip instrumentation.

(4.8) Derivation of System Inputs

The measured variables listed in Table 7.2-1 are direct measurements of the required parameters.

(4.9) Capability for Sensor Checks

The protection system is tested at rated power in four separate tests. The first of these is the relay logic test which is accomplished by operating the keylocked test switches provided in the logic subchannel strings and located on the two protection system relay panels. This switch deenergizes the scram contactors and verifies the ability to de-energize each of the four groups of scram solenoid pilot valves as shown by the group indicator lights on the protection panels.

The second test includes calibration of neutron monitors and radiation monitors by means of simulated inputs from calibration signal units. The neutron monitoring average power range monitors are bypassed one at a time for calibration. The LPRMs are individually calibrated using information from the Traversing In-Core Probe and their readings normalized so that the operator can tell at a glance how the readings compare with those expected at rated power.

The third test is the single rod scram test, which verifies the capability of each rod to scram. It is accomplished by operation of toggle switches on the protection system operations panel. Timing can be determined for each rod scrammed. This type test requires physics review to assure that the rod pattern during scram testing does not create a high worth rod condition.

The fourth test involves applying a test signal to each process sensor in turn and observing that it produces a protection logic channel trip at the proper value. The test signals are applied to the process type sensing instruments (pressure and differential pressure) through calibration taps.

(4.10) Capability for Test and Calibration

The use of one-out-of-two-twice logic between channels permits an RPS channel to be tested on-line without initiating a reactor trip. A test signal is routinely applied to one input at a time. Maintenance, to the extent of removing and replacing any component within a channel, may be accomplished in the on-line state without a reactor trip.

(4.11) Channel Bypass or Removal From Operation

The RPS channel bypass is described in Subsection 7.2.1.1.3. This feature permits the testing and maintenance of a single channel during power operation. With the bypass in effect, the three remaining channels will provide the necessary protection.

Since only two channel trips (one in each channel) are required to cause a reactor trip, a single failure will not prevent the RPS from fulfilling its protective function.

The RPS is a de-energize to trip system. Therefore, if power is lost to a channel, that channel will trip, reducing the system trip coincidence to one-out-of-two in the other channel. In the event that a module that performs a protective function is removed from its rack, that RPS channel will trip (unless the channel is bypassed).

(4.12) Operating Bypass

Operating bypasses are discussed in Subsection 7.2.1.1.3.

(4.13) Indication of Bypasses

The shutdown mode trip is bypassed 20 seconds after placing the Reactor Mode Switch in the shutdown position. The Scram Discharge Volume trip is bypassed when the Reactor Mode Switch is in the shutdown or refuel positions.

When the mode switch is not in RUN and reactor pressure is less than 600 psig, or reactor mode switch is in "SHUTDOWN", the following reactor trips are bypassed:

1. Turbine Stop Valves less than 90% open
2. Turbine Trip/Generator Load Rejection
3. Condenser Vacuum
4. MSIVs less than 90% open

When the Reactor mode switch is in the SHUTDOWN position only, scram reset due to MSIV closure and/or low condenser vacuum signals can be initiated without reducing the reactor pressure below 600 psig.

The Turbine Trip/Generator Load Rejection reactor trip is also bypassed when reactor thermal power is below 40%. This bypass is accomplished using pressure switches off of the Turbine Third Stage Extraction Steam Line.

Control Room indication is provided for the following scram bypass functions:

1. Scram Discharge Volume Trip
2. Condenser Low Vacuum
3. Main Steam Isolation Valve Closed
4. Turbine Trip/Generator Load Rejection (Turbine Low Power)

(4.14) Access to Means for Bypassing

RPS bypasses are generally in effect only when the plant is not operating at power. The bypasses are effected by the Reactor Mode Switch.

(4.15) Multiple Setpoints

All setpoints in the RPS are fixed, except the setpoint for the APRM trip.

The APRM trip setting is varied automatically with recirculation flow as prescribed in the Technical Specifications.

(4.16) Completion of Protective Action Once Initiated

All RPS trips are sealed types, so that a tripped channel will remain in that state until deliberately reset by the operator. Trip reset by the operator is prevented by a time delay for 10 seconds after the reactor trip signals have initiated.

(4.17) Manual Initiation

This criterion is not met literally in that protective actions are not initiated at a "system level" using a "minimum of equipment." It is believed that these two requirements are contradictory and practically unattainable because equipment added to obtain an initiation at the system level would clearly be in addition to the minimum needed to obtain operation manually.

The trip system which uses two manual initiation buttons in order to obtain separation and testability is clearly more reliable than it would be if a single button were utilized.

(4.18) Access to Setpoint Adjustments, Calibration and Test Points

Setpoint adjustment and calibration of the Neutron Monitoring System is described in Section 7.5.

Adjustment and calibration of the pressure, differential pressure and limit switches is accomplished locally.

(4.19) Identification of Protective Action

The plant computer and/or sequence of alarms recorder will indicate that a subchannel has tripped and which trip function relay has tripped. Each trip function is also annunciated in the Control Room.

(4.20) Information Readout

Readouts for the Neutron Monitoring System are discussed in Section 7.5. The sequence of alarms recorder indicates subchannel and trip function trips.

(4.21) System Repair

The design permits but does not necessarily "facilitate" the recognition, location, replacement and repair or adjustment of malfunctioning components or modules. In some cases the location of a specific faulty element of a circuit requires checking more than one device or location. Failed elements ("opens") on an energize to trip circuit may not be evident except during periodic test.

Conversely wiring shorts in de-energize to trip circuits ("fail safe circuits") may not be evident except during periodic tests.

(4.22) Identification

RPS components and wiring can be identified only after reviewing plant documents. A visual inspection does not suffice to identify RPS components.

7.2.2.2 Conformance to NRC General Design Criteria

The applicable NRC General Design Criteria are listed in Section 7.1 and a discussion of each is presented in Section 3.1.

7.2.2.3 Conformance with Regulatory Guides

Regulatory Guide 1.22

The RPS incorporates a scheme of testing that complies with Regulatory Guide 1.22. Utilizing logic trip switches in the Control Room the operator can trip the output from any RPS channel so that the associated solenoid valve will de-energize. This operation can be done without disrupting station operation since two solenoid valves in series must trip in order to cause control rod insertion. Test circuits allow the operator to completely test the RPS channels at any time during reactor operation. See Subsection 7.2.2.1 for additional information.

Regulatory Guide 1.29

A discussion relating to equipment qualification is contained in Section 3.10.

Regulatory Guide 1.47

The RPS design includes provisions for automatic indication of a discharge volume trip bypass or inoperative condition of neutron monitors in conformance with this regulatory guide.

Regulatory Guide 1.53

The RPS is designed to conform with the single failure criteria as applied in IEEE 379. Condenser low vacuum scram does not meet this criteria if only 1 trip system is operable. See Section 7.2.1.1.2 for further discussion of this logic.

Regulatory Guide 1.62

Manual initiation of the reactor trip by the RPS complies with Regulatory Guide 1.62 and is discussed in Subsections 7.2.1.1.8 and 7.2.2.1.

Regulatory Guide 1.75

Redundant components of the RPS are located to minimize the possibility of a single event affecting more than one redundant subsystem. RPS sensors and sensor-to-process connections are separated to the maximum practicable extent. Subsections 7.1.2.2 and 8.3.3 discuss the separation of redundant signal and power cabling, respectively, to and between RPS systems and components.

Regulatory Guide 1.89

See Sections 3.10 and 3.11 for a discussion on equipment qualification.

Regulatory Guide 1.100

See Section 3.10 for a discussion of qualification testing.

Regulatory Guide 1.105

OCNGS began commercial operation in 1969. As such, OCNGS established its instrument setpoints prior to the issuance of Regulatory Guide 1.105.

To establish new instrument setpoints (post 1988), GPUN follows ES-002, "Instrument Error Calculation and Setpoint Determination". ES-002 addresses factors which influence instrument setpoints. These factors include the relationships of the instrument setpoint to the various limits of normal operation, design basis environmental conditions, calibration and surveillance testing.

During the preparation of ES-002, GPUN reviewed various industry documents, such as R.G.1.105, and incorporated some of their guidelines into the standard. Those guidelines which were incorporated are identified by references within the standard.

Regulatory Guide 1.118

See Section 7.1 for conformance with this guide.

TABLE 7.2-1
(Sheet 1 of 4)
REACTOR PROTECTION SYSTEM DATA SHEET

	Parameter	Sensor Number	Actuated Relays (See Footnote 4) I	II	III	V	Trip Pt. (Footnote 3)	Automatic Action
1.	High Neutron Flux or Inoperative Trip	IRM Chan. 11 & 12 10K35 & 10K37	1K1	1K51			120/125% of scale, 38/40 of scale, or Inoperative	Trip Subchannel 1A which trips Channel I Pilot Scram Valve Solenoids when the Mode Switch is Not in "Run".
		IRM Chan. 13 & 14 10K39 & 10K41	1K2	1K52			120/125% of scale, 38/40 of scale, or Inoperative	Trip Subchannel 1B which trips Channel I Pilot Scram Valve Solenoids when the Mode Switch is Not in "Run".
		IRM Chan. 15 & 16 10K43 & 10K45	2K1	2K51			120/125% of scale, 38/40 of scale, or Inoperative	Trip Subchannel 2A which trips Channel II Pilot Scram Valve Solenoids when the Mode Switch is Not in "Run".
		IRM Chan. 17 & 18 10K47 & 10K49	2K2	2K52			120/125% of scale, 38/40 of scale, or Inoperative	Trip Subchannel 2B which trips Channel II Pilot Scram Valve Solenoids when the Mode Switch is Not in "Run".
		APRM Chan. 1 & 2	1K1	1K51			For all APRMs, one set of contacts at 120% of full power; second set when flow monitoring electronics pro duce inoperative trip.	Trip Subchannel 1A When Mode Switch is in "Run". Trip Subchannel 1B When Mode Switch is in "Run". Trip Subchannel 2A When Mode Switch is in "Run". Trip Subchannel 2B When Mode Switch is in "Run
		APRM Chan. 3 & 4	1K2	1K52				
		APRM Chan. 5 & 6	2K1	2K51				
		APRM Chan. 7 & 8	2K2	2K52				
2.	APRM Downscale, and IRM Upscale or Inop	APRM Chan. 1 & 2 IRM Chan. 11 & 12	1K1	1K51				Trip Subchannel 1A when Mode Switch is in "Run". Trip Subchannel 1B when Mode Switch is in "Run".
		APRM Chan. 3 & 4 IRM Chan. 13 & 14	1K2	1K52				
		APRM Chan. 5 & 6 IRM Chan. 15 & 16	2K1	2K51			APRM 2/150 and IRM 120/125, 38/40 or INOP	Trip Subchannel 2A when Mode Switch is in "Run". Trip Subchannel 2B when Mode Switch is in "Run".
		APRM Chan. 7 & 8 IRM Chan. 17 & 18	2K2	2K52				

TABLE 7.2-1
(Sheet 2 of 4)

REACTOR PROTECTION SYSTEM DATA SHEET

	Parameter	Sensor Number	Actuated Relays (See Footnote 4) I	II	III	IV	V	Trip Pt. (Footnote 3)	Automatic Action	
3.	Recirculation Flow Monitoring	FS-622-38A	1K1 1K2	1K51 1K52				Inoperative	Trip Channel I Pilot Scram Valve Solenoids	
		FS-622-38B	2K1 2K2	2K51 2K52				Inoperative	Trips Channel II Pilot Scram Valve Solenoids	
4.	Reactor High	REO3A	1K3	1K51				1060 psig	Trips Subchannel 1A	
	Pressure	REO3C	1K4	1K52				1060 psig	Trips Subchannel 1B	
		REO3B	2K3	2K51				1060 psig	Trips Subchannel 2A	
		REO3D	2K4	2K52				1060 psig	Trips Subchannel 2B	
5.	Reactor Low	REO5/19A	1K5	1K51				Decreasing	Trips Subchannel 1A	
	Water Level	REO5B	1K6	1K52				Reactor Water	Trips Subchannel 1B	
		REO5A	2K5	2K51				Level at 137-11/16" above the top	Trips Subchannel 2A	
		REO5/19B	2K6	2K52				Of the active fuel	Trips Subchannel 2B	
6.	Dump Volume	RD91A, RD87A	1K7	1K51				Rising Water Level at 18.36	Trips Subchannel 1A	
	High Water	RD91C, RD87C	1K8	1K52				Gallons	Trips Subchannel 1B	SEE
	Level	RD92B, RD88B	2K7	2K51					Trips Subchannel 2A	FOOTNOTE 1
		RD92D, RD88D	2K8	2K52					Trips Subchannel 2B	
7.	Drywell High	REO4A	1K9	1K51	1K71	1K75		2.4 psig	Trips Subchannel 1A	
	Pressure	REO4C	1K10	1K52	1K72	1K76			Trips Subchannel 1B	
		REO4B	2K9	2K51	2K71	2K75			Trips Subchannel 2A	
		REO4D	2K10	2K52	2K72	2K76			Trips Subchannel 2B	
8.	Condenser	RSCS-11	1K11	1K51				20 in. Hg Vacuum	Trips Subchannel 1A	
	Low Vacuum	RSCS-21	1K12	1K52					Trips Subchannel 1B	SEE
		RSCS-12	2K11	2K51					Trips Subchannel 2A	FOOTNOTE 2 and 5
		RSCS-22	2K12	2K52					Trips Subchannel 2B	

FOOTNOTES:
1. All trips bypassed when the mode switch is in "refuel" or “shutdown” and the Instrument Volumes bypass selector switch (3S4-1) is in "bypass".
2. All trips bypassed by contacts of relays 1K112 A&B and 2K112 A&B when mode switch is in "refuel" or "startup" and reactor pressure less than 600 psi.
5. All trips bypassed when Mode Switch is in “Shutdown” following a time delay of 20 seconds.

TABLE 7.2-1
(Sheet 3 of 4)

REACTOR PROTECTION SYSTEM DATA SHEET

	Parameter	Sensor Number	Actuated Relays (See Footnote 4) I	II	III	IV	V	Trip Pt. (Footnote 3)	Automatic Action
9.	Turbine Trip-Generator Load Rejection or Turbine Stop Valve Closure		Refer to Table 7.7-2						
10.	Main Steam Line Isolation Valves Closure	NS03A, NS04A NS03A, NS04A NS03B, NS04B NS03B, NS04B	1K17 1K18 2K17 2K18	1K51 1K52 2K51 2K52				Limit switches set To trip with valve Closing at 10% off Full open position.	Trips Subchannel 1A. Trips Subchannel 1B. Trips Subchannel 2A. Trips Subchannel 2B. Scram is bypassed by contacts of 1K112 A&B & 2K112 A&B (see condenser low vacuum)
11.	Reactor Manual Scram	Button 1S2	1K21A						Trips Channel I Pilot Scram Valve Solenoids
			1K21B					When both buttons are Pushed simultaneously.	Seal out relay. Reset with reset button 3S1.
			1K21C						Trips Channel I Scram discharge Pilot Valve Solenoids
		Button 2S2	2K21A					When both buttons are Pushed simultaneously.	Trips Channel II Pilot Scram Valve Solenoids.
			2K21B						Seal out relay. Reset with Reset button 3S1.
			2K21C						Trips Channel II Scram discharge Pilot Valve Solenoids
12.	Loss of A.C. Power to Reactor Protection System								All relays and scram solenoid valves will trip due to loss of power, as MG sets coast down
									Power to Reactor protection system

TABLE 7.2-1
(Sheet 4 of 4

REACTOR PROTECTION SYSTEM DATA SHEET

	Parameter	Sensor Number	Actuated Relays (See Footnote 4) I	II	III	IV	V	Trip Pt. (Footnote 3)	Automatic Action
13.	Reactor Mode Switch	Switch 1S1	1K21 A&B					"Shutdown"	Trips Channel I Pilot Scram Valve Solenoids Position

FOOTNOTES:
1. All trips bypassed when the mode switch is in "refuel" or "shutdown" and the Instrument Volumes bypass selector switch (3S4-1) is in "bypass".
2. All trips bypassed by contacts of relays 1K112 A&B and 2K112 A&B when mode switch is in "refuel" or "startup" and reactor pressure less than 600 psi.
5. All trips bypassed when Mode Switch is in "Shutdown" following a time delay of 20 seconds.

TABLE 7.2-2
(Sheet 1 of 2)

REACTOR TRIP FUNCTION BYPASSES

	FUNCTION	BYPASS CONDITIONS AUTOMATIC	BYPASS CONDITIONS MANUAL
	NEUTRON MONITORING SYSTEM TRIP		
a)	Recirculation Flow Monitoring INOP	Never	Never
b)	IRM INOP or Upscale	Mode Switch in RUN and APRM on scale	One sensor channel per logic channel for testing. Mechanical interlock prevents two channels per logic channel from being bypassed.
c)	IRM Upscale or INOP and companion APRM Downscale	Never	One sensor channel per logic channel for testing. Mechanical interlock prevents two channels per logic channel from being bypassed.
d)	APRM Upscale or INOP	Never	One sensor in each channel via joystick. Two joysticks permit one sensor in each of two channels to be bypassed. An electrical interlock and administrative procedure prevent two sensors in the same quadrant from being bypassed at same time.
	Reactor High Pressure Trip	Never	Never
	Reactor Low-Low Level Trip	Never	Never

TABLE 7.2-2
(Sheet 2 of 2)

REACTOR TRIP FUCTION BYPASSES

FUNCTION	BYPASS CONDITIONS AUTOMATIC	BYPASS CONDITIONS MANUAL
Discharge Volume High Water Level trip	No	Manual Switch 354 in Bypass and Mode Switch in Shutdown or Refuel to permit draining after a trip.
High Drywell Pressure Trip	No	No
Condenser Vacuum and Turbine Trip (Turbine Stop Valve Closure and Turbine Control Valve Fast Closure/Load Rejection)	Mode Switch in refuel or startup and Reactor Pressure less than 600 psig. Mode switch in “shutdown.”	Mode switch in Refuel to allow rod scram time testing.
Turbine Trip (Turbine Stop Valve Closure and Turbine Control Valve Fast Closure/Load Rejection)	Reactor thermal power less than 40%	No
MSIV Closure Trip	Mode Switch in refuel or startup and Reactor Pressure less than 600 psig. Mode switch in “shutdown.”	No
Mode Switch in Shutdown	Twenty seconds after the Mode switch is placed in SHUTDOWN, time delay relays bypass the trip, trip may then be manually reset.	The scram signal can be jumpered (for approximately 60 seconds) while the reactor mode switch is transferred from the refuel position to the shutdown position in accordance with plant procedures.
Manual Trip	No	N/A

TABLE 7.2-3
(Sheet 1 of 1)

REACTOR MODE SWITCH 1S1

Mode		Permissives or Limitations
Shutdown	a.	RPS logic channels I and II de-energized.
	b.	Permissible to open all isolation valves.
	c.	No control-rod-drive movement possible.
	d.	Possible to bypass dump-volume-high-level scram.
	e.	Possible to reset a scram due to low condenser vacuum and/or MSIV closure signals without reducing reactor pressure below 600 psig.
Refuel	a.	One rod free-movement interlock in service.
	b.	Possible to bypass dump-volume-high-level scram.
	c.	Condenser-low-vacuum and main-steam-line-isolation trips in the RPS are bypassed.
	d.	Refueling platform interlocks in service.
	e.	When leads in the manual scram circuit are lifted, the neutron monitoring system trips are combined in a one-out-of-twenty scram logic.
	f.	RPS Energized.
	g.	Permissible to open all isolation valves.
Startup	a.	Condenser low vacuum and main-steam-line-isolation trips in the RPS are bypassed when reactor pressure is less than 600 psi.
	b.	RPS energized with IRM protection in neutron-monitoring- system trips.
	c.	Normal control-rod-withdrawal interlocks in service with IRM protection.
Run	a.	RPS energized with APRM protection.
	b.	Normal control-rod-withdrawal interlocks in service with APRM rod-block protection.
	c.	Absorption-pool vent valves and drywell ventilation and purge isolation valves must be closed.

TABLE 7.2-4
(Sheet 1 of 1)

MOTOR-GENERATOR SET SPECIFICATIONS

Motor Input:	440 volts (±10%), 3 phase, 60 Hz.
Generator Output:	15 kw, 18.75 kva continuous 115 volts (rated), 1 phase regulated, 60 cps
Performance:	a. With a 2 second supply voltage interruption, output voltage and frequency do not drop more than 5 percent. Recovery to steady state regulation after restoration of rated voltage does not exceed 5 seconds. b. With 50 percent (kva) step load change, output voltage does not drop more than 15 percent and output frequency does not drop more than 5 percent. Recovery to steady state does not exceed 1 second.

7.3 ENGINEERED SAFETY FEATURE SYSTEMS

This section describes the instrumentation provided to initiate the following Engineered Safety Feature (ESF) Systems:

a. Reactor Vessel (Main Steam) Isolation System

b. Primary Containment (Drywell) Isolation System

c. Isolation Condenser System

d. Standby Gas Treatment System

e. Core Spray System

f. Containment Spray System

g. Emergency Service Water System

h. Standby Gas Treatment System (non RPS Initiation)

i. Automatic Depressurization System

j. Secondary Containment Isolation

Engineered Safety Feature Systems and their associated instrumentation are described in Chapter 6. This section describes the safety related initiating instrumentation for each system.

7.3.1 Description

7.3.1.1 System Description

7.3.1.1.1 Reactor Protection System

The Reactor Protection System (RPS) provides reactor trip signals, Engineered Safety Feature (ESF) systems actuation signals and other safety related signals. The reactor trip function is discussed in Section 7.2. ESF systems that receive either all or some initiating signals from the RPS are identified below. RPS signals for the ESF systems identified below are shown in Table 7.3-1.

Reactor Vessel (Main Steam) Isolation

Reactor vessel isolation can be initiated by any one or any combination of the following conditions:

a. Deleted

b. Steam Line High Flow - Eight differential pressure switches, two switches per subchannel, detect a steam line break.

c. Trunnion Room High Temperature - sixteen temperature switches, four switches per subchannel, detect a steam line break.

d. Main Steam Line Low Pressure - Four pressure transmitters, one transmitter per subchannel, detect low steam pressure.

e. Reactor Low-Low Water Level - Four level transmitters, one transmitter per subchannel, detect low-low water level in the reactor vessel.

Reactor vessel isolation takes place when at least one subchannel in each RPS channel has been deenergized (one-out-of-two twice). See Drawing GE 237E566 for circuit details.

Primary Containment (Drywell) Isolation

Except for Reactor Building Closed Cooling Water (RBCCW) drywell isolation, the signals that initiate primary containment (drywell) isolation and originate from the RPS are:

a. Reactor Low-Low Level - Same sensors as for Main Steam Isolation, Item e. above.

b. High Drywell Pressure - Four pressure switches, one switch per subchannel, detect high drywell pressure.

c. Some drywell isolation valves also isolate on containment high radiation signals.

Reactor Building Closed Cooling Water (RBCCW) Isolation

RBCCW isolation will occur when coincident reactor low low level and high drywell pressure signals are present or when a reactor triple low level signal is present. See Drawing NU 5060E6003.

a. Reactor Low-Low-Low Level: Four level switches, two per logic channel, actuate when reactor water level decreases below triple low setpoint. A time delay is provided to prevent isolation should a spurious triple low signal occur.

or

b. High Drywell Pressure - Two pressure switches, one per logic channel, actuates on high drywell pressure,

and

c. Reactor Low-Low Level - Two level sensors, one per logic channel, actuates on reactor water level decreases below double low setpoint.

Isolation Condenser System

The automatic signal to begin the operation of the Isolation Condenser System is initiated by the detection of a high pressure condition in the reactor vessel or upon the detection of low-low water level in the reactor vessel.

There are four high pressure sensors and four low-low water level sensors. These sensors are combined in a one-out-of-two twice configuration to initiate operation of the Isolation Condenser System. See Drawing BR 3029 for details.

Coincident with the low-low reactor water level signal for ICS initiation, is a signal to trip all five reactor recirculation pumps. Coincident with the high reactor pressure signal for ICS initiation is a signal to trip recirculation pumps A, B, and E. Recirculation pumps C and D will trip within 12 seconds on a persistent high pressure signal. See Drawing BR 3029. (Time delay setpoints will provide margin for calibration and accuracy of the time delay relays).

Standby Gas Treatment System

The conditions sensed by the RPS that initiate the Standby Gas Treatment System (SGTS) are described below and shown on Drawing GE237E566 and GU 3D-822-17-1002.

a. High Drywell Pressure - Same sensors as for primary containment isolation given above.

b. Low-Low Reactor Water Level - Same sensors as for primary containment isolation given above.

Non-RPS initiation of the SGTS is listed under Subsection 7.3.1.1.4.

Secondary Containment Isolation

Secondary containment isolation is initiated by the same sensors that initiate the Standby Gas Treatment system.

Core Spray System

The Core Spray System is initiated by either of the conditions described below and shown on Drawing NU5060E6003.

a. High Drywell Pressure - Four pressure switches, one switch per subchannel, detect high drywell pressure.

b. Reactor Low-Low Water Level - Same sensors as for primary containment isolation given above.

Both of the redundant Core Spray Loops will be initiated if one or more of the eight sensors for a. and b. above are tripped.

7.3.1.1.2 Containment Spray System

The Containment Spray System is initiated manually by procedure. An interlock is provided to prevent pump starting during the normal diesel-generator loading sequence.

The containment spray system will trip on decreasing drywell pressure if operating in the drywell spray mode. There are four pressure switches, two per containment spray system, which will actuate on low drywell pressure.

Reactor Low Low Level and High Drywell Pressure signals initiate relays in the containment spray that provide inputs to the normal emergency power interlocks and subsequently the control logics of the Reactor Building Closed Cooling Water Pumps, service water pumps, and drywell recirculation fans. However, these low low level and high drywell pressure signals are independent of containment spray system operation and do not provide any controlling function to the containment spray system. The drywell pressure switches used for this function are the same as the ones that trip the containment spray system on low drywell pressure.

7.3.1.1.3 Emergency Service Water System

The Emergency Service Water System is manually initiated by procedure. An interlock is provided to prevent ESW pump starting during the normal diesel-generator loading sequence.

7.3.1.1.4 Standby Gas Treatment System (non-RPS)

In addition to the RPS initiation discussed in Subsection 7.3.1.1.1, the SGTS is initiated by high radiation detected in the Reactor Building vent, or the Spent Fuel Storage Pool area.

7.3.1.1.5 Automatic Depressurization System

The Automatic Depressurization System is initiated if the following conditions occur simultaneously:

a. Low-Low-Low Reactor Water Level - four level switches, two per logic channel, detect triple low water level in the reactor vessel.

b. High Drywell Pressure - same sensors as used for Core Spray System initiation.

c. Core Spray System Operating - eight relay contacts, four per channel which are activated when core spray booster pump differential pressure is high enough to indicate that the Core Spray System pumps are operating to supply coolant makeup that will be required when depressurization occurs.

If all of the conditions listed in a., b., and c. are satisfied in either channel, ADS operation will be initiated after a two minute time delay. ADS Timer Switches are provided for resetting the two minute timers to delay actuation. Annunciate alarms continuously inform the operator when ADS Switches are placed in the BYPASS position, thereby bypassing ADS actuation.

7.3.1.1.6 Alternate Rod Injection (ARI) System

This system provides a method diverse from the RPS for depressurizing the scram header in the unlikely event the RPS does not cause a reactor scram in response to an operational transient. The ARI System employs five normally deenergized solenoid valves on the scram header. Depressurizing of the air header will result in control rod insertion. ARI is initiated automatically if either of the following conditions occur:

a. Hi-Hi Reactor Pressure

or

b. Reactor Low-Low Level

ARI level and pressure initiation sensor loops are arranged in a single channel, two-out-of-two logic scheme and as such does not meet single failure criteria in that both sensor loops must actuate in order for ARI to initiate automatically. Other discussions herein pertaining to redundancy and single failure criteria as they apply to Engineered Safeguard circuits do not apply to ARI.

7.3.1.2 Logic

7.3.1.2.1 Reactor Protection System

The Reactor Protection System logic is discussed in Section 7.2.

7.3.1.2.2 Containment Spray System

The Containment Spray System logic has two independent systems. The containment spray pumps are started manually; however, an interlock is provided to prevent start-up until the diesel-generator normal loading sequence is complete.

7.3.1.2.3 Emergency Service Water System

The ESW pumps are started manually; however, an interlock is provided to prevent start-up until the diesel-generator normal loading sequence is complete.

7.3.1.2.4 Standby Gas Treatment System

To minimize spurious actuation of the system, the exhaust duct radiation monitor trips are set at a fairly high level above normal expected radiation levels. The spent fuel storage pool and reactor cavity radiation monitor trips are set at a low level, however there is a time delay (set between 1 and 5 minutes) to prevent spurious initiation of the SGTS. If the high radiaiton condition has not cleared in this interval, the Reactor Building normal ventilation is automatically isolated and the SGTS is automatically initiated.

7.3.1.2.5 Automatic Depressurization System (ADS)

The control logic for the ADS contains two channels. Either channel can initiate automatic depressurization.

A two minute timer will start (and seal in) if three signals are sensed:

a. High drywell pressure

b. Triple low water level

c. Core Spray System operating (differential pressure across the core spray booster pump).

The timer allows the operator to prevent ADS operation if safe operation can be achieved without ADS. Safe operation means that reactor water level is recovering. Alarms will inform the operator when ADS Timer Reset Switches have been placed in the BYPASS position.

7.3.1.2.6 Alternate Rod Injection System

The ARI System utilizes the Fuel Zone Pressure loops PT-622-1018 (Channel C) and PT-622-1019 (Channel D) which are part of the Reactor Plant Instrumentation System. The ARI System utilizes these loops for the RV high pressure inputs to the ARI logic. ARI automatic level and pressure initiation sensor loops are arranged in a single channel, two-out-of-two logic. Since both sensor loops must trip in order to automatically initiate the ARI System, the failure of one sensor loop will not cause inadvertent ARI actuation.

7.3.1.3 Bypassing

7.3.1.3.1 Reactor Protection System

Reactor Protection System bypassing is discussed in Section 7.2.

7.3.1.3.2 Containment Spray System

There are no automatic bypasses.

7.3.1.3.3 Emergency Service Water System

There are no automatic bypasses.

7.3.1.3.4 Standby Gas Treatment System (SGTS)

There are no automatic bypasses in the SGTS. There is a time delay before initiation of SGTS when spent fuel storage pool and reactor cavity area radiation monitors exceed preset levels. This time delay prevents spurious SGTS initiation. If the high radiation condition has not cleared in this interval, the Reactor Building normal ventilation is automatically isolated and the SGTS is automatically initiated.

7.3.1.3.5 Automatic Depressurization System

There are no automatic bypasses in the ADS. The operator may choose to delay initiation during the timed delay start by manually turning the key operated reset switches, one per channel, in the Control Room. Annunciator alarms will inform the operator when ADS Timer Reset Switches are placed in the BYPASS position, thereby bypassing ADS actuation. Each time the switch is turned to reset and returned to auto, the time delay cycle will be repeated until the initiation signal clears. A spuriously opened EMRV may be closed by use of a Keylocked Control Switch in Control Panel 1F/2F (Refer to 6.3.1.2.4).

7.3.1.4 Interlocks

7.3.1.4.1 Reactor Protection System

RPS interlocks are discussed in Section 7.2.

7.3.1.4.2 Containment Spray System

Containment Spray pumps are stopped automatically when containment pressure falls to 0.6 psig or less and the mode selector switch is in the "Drywell Spray" mode of operation.

Header test block valves V-21-5 and V-21-11 are normally closed and can be opened from local keylock switches and mode select switches. The valves will open automatically when the mode select switch is placed in the "Drywell Spray" mode.

Emergency service water discharge valves V-3-87 and V-3-88 are normally open and locked in the proper throttled position. Control Room indication of the valve position is maintained.

Suppression Chamber spray block valves V-21-15 and V-21-18 are normally closed and will open automatically if containment spray is required. The valves can be operated from local keylock switches. They are normally closed, but they will open automatically if they are required for spray service.Torus cooling test valves V-21-13 and V-21-17 are normally open and are used for full flow testing of the Containment Spray System without wetting the drywell. The flow from these valves is directed to the suppression chamber through connections into a vacuum breaker line. The valves can be closed from local keylock switches and from the mode select switches in the Control Room. They will close automatically upon placing the mode select switch in the "Drywell Spray" mode.

The Containment Spray System is interlocked with the normal emergency power interlocks as follows: Reactor Low Low Level and High Drywell pressure signals initiate relays in the Containment Spray System that provide inputs to the normal emergency power interlocks and subsequently the control logics of the Reactor Building Closed Cooling Water pumps, Service Water pumps, and drywell recirculation fans. However, these low low level and high drywell pressure signals are independent of Containment Spray System operation and do not provide any controlling function to the Containment Spray System. Nor are the emergency power interlocks and associated systems dependent upon containment spray pump operation or mode of operation.

7.3.1.4.3 Emergency Service Water System

The Emergency Service Water (ESW) pumps are manually operated. There are no interlocks except to prevent start-up during normal diesel-generator load sequencing.

7.3.1.4.4 Standby Gas Treatment System

There are no interlocks in the SGTS.

7.3.1.4.5 Automatic Depressurization System

There are no interlocks in the ADS. However, timing circuits are included which stagger the valve openings so that the relief valves do not all open at once.

7.3.1.4.6 Isolation Condenser System

Line break sensors are provided in the design in order to automatically isolate the affected Isolation Condenser if a persistent high steam or condensate flow condition occurs.

7.3.1.4.7 Reactor Vessel (Main Steam) Isolation

The MSIV low steam line pressure trip is bypassed unless the reactor mode switch is in the run position or the IRM range switches are in the range 10 position. This bypass is required to allow plant startup.

7.3.1.5 Redundancy

All functions of ESF System actuation are implemented by redundant sensors, logic, and actuated devices except that SGTS logic is common to both SGTS filter trains.

7.3.1.6 Diversity

Diverse initiating functions have been provided to mitigate the consequences of an accident.

7.3.1.7 Actuated Devices

The pumps and valves actuated by the ESF Systems initiation circuits are partially listed in Tables 7.3-1 and 7.3-2. Complete lists and description of components are given in the respective system discussion or in the figures in Chapter 6.

7.3.1.8 Information Display

There are no analog displays associated with the actuation logic of the Engineered Safety Features Systems.

Annunciators in the Control Room provide the operator with status information.

Certain ESF System actuation signals, in addition to reactor trip signals, are connected to a sequence of alarms recorder.

7.3.1.9 Power Supplies

The RPS sensors and relays receive ac power as described in Section 7.2. The ESF actuation relays and switches receive power from the 125 Volt dc system. The 125 Volt dc system is described in Section 8.3.

7.3.2 Design Bases Information

Conformance with Section 3 of IEEE 279

a. The generating station conditions that required protective action are:

   1. Preservation of fuel cladding integrity.

      The Core Spray System, with the associated Automatic Depressurization System, constitutes the low pressure Emergency Core Cooling System, which provides ECCS water after the improbable occurrence of a pipe break accident in the reactor primary system. The ECCS thus prevents fuel cladding melting by removing fission product decay heat from the

reactor following postulated accidents which result in loss of coolant, and which would lower reactor water level to uncover the fuel.

The Isolation Condenser System is a passive standby, high pressure system for removal of fission product decay heat after reactor isolation when the main condenser is not available as a heat sink. Removal of decay heat by the Isolation Condenser System helps limit pressure rise in the reactor vessel, thereby minimizing the loss of reactor coolant through operation of the reactor vessel safety valves or Electromatic Relief Valves.

2. Prevention of Release of Radioactive Materials to the environment.

The Containment Spray System with the associated Emergency Service Water System removes heat from the primary containment after a postulated loss of coolant accident. The system is designed to:

a. Remove post LOCA heat from the primary containment, and in conjunction with the Core Spray System to assure continuing of core cooling.

b. Minimize the long term post LOCA pressure inside containment to reduce fission product leakage.

The Standby Gas Treatment System functions as a barrier between the radiation source and the environs during an emergency condition. Gas treatment minimizes the effects to the environs of any fission products which might be released within the Reactor Building during a postulated accident.

The Drywell Isolation and Main Steam Isolation Systems serve to close isolation valves in lines which connect to the reactor vessel or which are open to primary containment before significant amounts of fission products are released from the reactor core under design basis accident conditions.

b. Generating Station variables that are required to be monitored in order to provide protective actions are shown in Tables 7.3-1 and 7.3-2.

c. The minimum number and location of sensors required to monitor adequately, for protective function purposes, those variables that have spatial dependence are given in the Technical Specifications.

d. Limiting conditions for operation for Technical Specification variables are presented in Section 3.0 of the Technical Specifications.

e. Limiting safety system settings and the basis for these settings are presented in Section 2.0 of the Technical Specifications.

f. The levels that, when reached, will require protective action are shown in Tables 7.3-1 and 7.3-2. See Note #2 on Tables 7.3-1 and 7.3-2 with regard to nominal vs. actual setpoints.

g. The range of transient and steady state conditions of the energy supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform has been assessed and equipment qualification is discussed in Section 3.11.

h. The malfunctions, accidents, or other unusual events that could physically damage protection system components, for which provisions must be incorporated to retain necessary protection system action are as follows:

   All instruments used in the RPS and ESF Systems that are required to function under accident conditions are designed to operate under the most unfavorable environmental conditions for which they are required under design basis accident conditions. For conventional devices, manufacturer's quality control, initial calibration, plant preoperational testing and plant maintenance procedures are relied upon to obtain a high degree of reliability.

   All sensing elements for the variables identified in Tables 7.3-1 and 7.3-2 are located outside the Drywell. Environmental qualification of safety related components is discussed in Section 3.11. Discussions regarding HELB requirements and an assessment of compliance to these requirements are presented in Amendment 75 of the FDSAR.

i. The determination of Technical Specification limits for ESF System variables is performed in accordance with approved engineering standards. The Technical Specification limits ensure that ESF Systems perform their respective safety functions as required during design basis accidents. When determining Technical Specification limits and associated margins of safety (allowances between Technical Specification limits and safety limits) , consideration is given to, but not limited to the following:

   - Effects of potential transient overshoot as determined by the design basis event analyses
   - Effects of time response characteristics of the total instrument channel
   - Environmental effects on instrument accuracy
   - Instrument drift.

7.3.3 Deleted

7.3.4 Analysis

7.3.4.1 Reactor Protection System

The analysis of the RPS is given in Section 7.2.

7.3.4.2 Non-RPS Systems

The Non-RPS systems are dual logic channels with an independent source of power and high reliability from the aspect of issuing initiating signals as well as preventing spurious initiating signals. The systems (except for SGTS initiation logic) are designed to perform their protection functions even if a single failure should occur.

7.3.5 Conformance with Section 4 of IEEE 279

OCNGS was built and operational prior to the issuance of IEEE 279. The following discussions are keyed to the paragraphs of Section 4 of IEEE 279 and provide a comparison of Oyster Creek design with this standard:

7.3.5.1 Reactor Protection System

Conformance with Section 4 of IEEE 279 is discussed in Section 7.2.

7.3.5.2 Non-RPS Systems

(4.1) General Functional Requirements

Except for the containment Spray and Emergency Service Water Systems, the ESF initiating circuits automatically perform their protective functions of initiating engineered safety features whenever plant conditions exceed preset levels.

(4.2) Single Failure Criteria

No single failure can prevent the initiating circuits from performing their protective function except as described below for the Standby Gas Treatment System.

The Automatic Depressurization System has full redundancy of dc control power supplied from the two station batteries but achieves its single failure tolerance through the application of loss-of-power transfer relays on each valve solenoid circuit and a loss of power transfer relay on each of the two control logic circuits. The circuit protection has been verified to have sufficient degree of coordination to tolerate shorting of any single circuit or total shorting of any single transfer relay within the control logic cabinet. However, certain combinations of short circuits in the Control Room panels could disable the ADS function. This vulnerability is due to the presence of both (+) and (-) sides of the manual control indicator lights. A single short applied in the most disadvantageous way could disable one valve. Additional shorts similarly applied could disable one valve each.

The location of the Electromatic Relief Valves (and their solenoid circuits) inside the drywell does not clearly make them invulnerable to pipe whip damage. Such damage could conceivably disable more than one EMRV.

Main steam isolation can be initiated by main steam line high flow, low pressure or high temperature in the vicinity of the main steam lines. The high flow switches are vulnerable to a single sensing line shutoff but the high temperature trip or low pressure trip will cause isolation. Thus, the system can tolerate single failure.

The automatic initiation logic downstream of the monitoring systems associated with the Standby Gas Treatment System is not designed to function in the event of a single failure. However, the capability exists to sense applicable parameters (each of the two RPS conditions, and, the four non-RPS radiation monitors when combined) assuming a single failure. Manual actuation of the SGTS is not vulnerable to single failures and provides an acceptable backup to automatic initiation.

See additional discussion in paragraph 4.6.

(4.3) Quality of Components and Modules

A quality control program and a component classification program are in place to ensure that the quality of RPS and ESF System components and equipment meet system performance and design requirements. The seismic qualification and environmental qualification programs are discussed in Sections 3.10 and 3.11 respectively.

All instrumentation and control components and/or assemblies important to safety have been designed, manufactured and installed to criteria that require the facility to withstand an earthquake giving a floor acceleration of 0.22g horizontally without loss of capability to fulfill essential functions. Panel and rack structures were seismically examined and judged to be rigid over the frequency range of the assumed earthquake time history acceleration peaks. Seismic qualifications of many of the types of instruments used have been completed but no seismic test qualification data is available on the instrument racks, control panels or relay panels. Therefore, by currently accepted seismic criteria and test methods seismic qualification is not demonstrated.

Similarly the Containment Isolation System and the Isolation Condenser System control can be shown to be initiated in the safe direction in the event of reasonably postulated seismically induced failure.

The Core Spray System loops cannot be assured to be free from inadvertent operation during a seismic event. Seismic test data available gives assurance that they will continue to operate once initiated because the initiating relays seal in and have a high seismic tolerance when energized.

The Automatic Depressurization System can be assured to be free from erratic operation during a design basis seismic event because the time delay circuit would prevent inadvertent seal in on momentary closure of a sensor contact. However, the level switches used at the OCNGS have snap acting contacts which have been seismically qualified in later designs, and for this reason erratic sensing is highly improbable.

Information presented above describes past compliance with Section 4 of IEEE 279. However, beginning in September 1995, instrumentation and control components are analyzed, evaluated and designed using the new EQE response spectra and the analytical methodology described in Section 3.7.3.4

(4.4) Equipment Qualification

The qualification of the equipment is discussed in Sections 3.10 and 3.11, and paragraph (4.3) above.

(4.5) Channel Integrity

Each channel is designed and fabricated so that channel integrity will be maintained under the conditions specified in the design bases for the system, except for the Automatic Depressurization System as discussed in paragraph 4.2.

(4.6) Channel Independence

This criterion is met within the limits set forth by design bases - physical separation of redundant channels of control are strictly maintained outside the main control panels which contain the manual controls for the Core Spray and Automatic Depressurization Systems and isolation valves. The Control Room panel wiring is protected from overheating by fuses and/or circuit breakers having ratings low enough to give assurance against conductor temperatures that could cause failure of the panel wiring. However, wiring for redundant manual control circuits are not necessarily separated physically within the Control Room panels. The physical separation for pressure sensing lines and sensors in the RPS is in accordance with the following general rule: sensing lines and sensors providing redundant protection system functions essential to reactor safety are physically separated such that no physical event considered credible could disable

RPS function in an unsafe direction. In general, this assumes a physical separation of several feet between redundant functions, sensors, and sensing lines. Most functions are much more widely dispersed but no specific distance is established as minimum.

Single failures and events designed against are:

a. Single instrument channel failure in any mode.

b. Any single logic subchannel failure.

c. Any single power supply failure.

d. Any mechanical damage to a single conduit or section of cable tray (approximately 12 ft in length), or that might result from a loss of coolant accident.

e. Any cable tray overheating from electrical sources.

f. Any single device or circuit failure in a relay panel or control panel (but not necessarily to include an extensive fire or mechanical damage shorting out more than one circuit) in an area under constant operator surveillance such as the Control Room control panel.

   The physical separation of wiring for the ECCS is in accordance with the following general rules:

a. Cables to sensors piped to a common process tap may run in the same wireway.

b. Cables to sensors of more than one variable for a given ECCS subsystem may share a common wireway tray.

c. Wiring for ECCS including control and power wiring to all devices required to be actuated during a Loss of Coolant Accident are separated by redundant systems or by redundant functions by either a distance of at least three feet between trays (except in cable spreading area and in limited cases in the 480V Emergency Switchgear Room), or by a steel barrier equivalent to a rigid metallic conduit wall and are sufficiently separated that no single event is reasonably likely to disable both systems. This requirement is considered satisfied by having the cables of at least one of the two systems in rigid conduit.

d. The Control Room panels have not been required to have special wiring or equipment separation but must demonstrate freedom from objectionable heating, which results in any temperature above the continuous temperature rating of cable insulation, during normal steady state operation at highest expected ambient temperatures.

e. Openings in the floor under the Control Room panels are closed to provide a "fire stop" function.

The separation practices employed in original design and construction are contained in "Separations Practices for Safeguard Systems" by APED Engineering, General Electric Co., (Revised November 26, 1968) from which the above general rules were extracted.

(4.7) <u>Control and Protection System Interaction</u>

Equipment classification is discussed in Paragraph (4.3). Adequate redundancy is provided to ensure redundant system availability in the event of a single failure. See Paragraph (4.2) for the discussion on single failures.

(4.8) <u>Derivation of System Inputs</u>

The measured variables listed in Tables 7.3-1 and 7.3-2 are direct measurements of the required parameters.

(4.9) <u>Capability for Sensor Checks</u>

All sensors can be checked during plant operation, except that testing of the main steam tunnel high temperature sensors requires access to the main steam lines area. Thus testing is accomplished during plant shutdown. The number of switches is sufficient to allow for multiple failures between tests without impairing capability to isolate, therefore testing during plant operation is not considered necessary.

(4.10) <u>Capability for Test and Calibration</u>

Sufficient testing capability is provided to test the systems during power operation without disturbing power operation. The Automatic Depressurization System is actually

tested only during refueling outages because of the transient that would occur if the valves opened.

(4.11) Channel Bypass or Removal from Operation

There are no channel, or operational, bypasses provided in the control systems for the Containment Isolation System or the Isolation Condenser System.

Cross channel inhibit manually operated switches are provided in each of the four (4) logic sub channels for Core Spray System logic trains to facilitate surveillance testing.

(4.12) Operating Bypass

The ADS logic is designed to allow the operator to reset or inhibit the two minute timers in order to delay or prevent ADS actuation.

The core spray logic is designed to allow the operator to override the ECCS signal and turn off the Core Spray System from the Control Room. The override signal automatically clears if the initiating signal clears, thus resetting the control logic of the Core Spray System.

(4.13) Indication of Bypasses

Bypassing or inhibiting the ADS actuation is annunciated in the Control Room.

Indication of the inhibit condition of the Core Spray System inhibit switches, as described under Item (4.11) is provided on the Control Room Annunciator.

(4.14) Access to Means for Bypassing

Administrative controls are in place that allow the operator the means for manually bypassing or overriding channels or protective functions as described in Paragraph (4.12).

(4.15) Multiple Setpoints

All setpoints are fixed.

(4.16) Completion of Protective Action Once Initiated

The Engineered Safety Features Systems comply with this paragraph.

(4.17) Manual Initiation

This criterion is not met literally in that protective actions are not initiated at a "system level" using a "minimum of equipment." It is believed that these two requirements are contradictory and practically unattainable because equipment added to obtain an initiation at the system level would clearly be in addition to the minimum needed to obtain operation manually.

The Automatic Depressurization System utilizes one manual switch for each of the five relief valves. A single device to control all five valves would raise a question of whether a single failure in this control circuit allowing all valves to open would not be an unacceptable alternative.

The isolation valves' manual controls are grouped and designed to give the operator necessary information regarding status and to reduce the number of manual operations required to complete containment isolation. This is considered as fulfilling the intent of 4.17.

The core spray manual control can be made operable under normal and emergency conditions. However, multiple steps may be required depending upon plant conditions. For example if a high drywell pressure or low low reactor vessel water level signal has initiated core spray, the operator may decide the trip is spurious. He will first override the sensor or sensors that are tripped and then he will be able to trip the pumps.

(4.18) Access to Setpoint Adjustments, Calibration and Test Points

Adjustment and calibration of the pressure, differential pressure and limit switches is accomplished locally.

(4.19) Identification of Protective Action

Actuation of any of the systems is indicated via annunciators in the Control Room.

(4.20) Information Readout

Information available to the Control Room operator is presented in such a manner to minimize operator confusion.

(4.21) System Repair

The design permits but does not necessarily "facilitate" the recognition, location, replacement and repair or adjustment of malfunctioning components or modules. In some cases the location of a specific faulty element of a circuit requires checking more than one device or location. Failed elements ("opens") on an energize to trip circuit may not be evident except during periodic test.

(4.22) Identification

Both RPS and non-RPS circuits comply with the identification requirement of this subparagraph.

7.3.6 Conformance to NRC General Design Criteria

The applicable NRC General Design Criteria are listed in Section 7.1 and a discussion of each is presented in Section 3.1.

7.3.7 Conformance with Regulatory Guides

Regulatory 1.22

The Automatic Depressurization System is not tested during power operation. It is tested during refueling outages. The other ESF actuation systems are tested during power operation as described in the Technical Specifications.

<u>Regulatory Guide 1.29</u>

A discussion relating to qualification is contained in Section 3.10.

<u>Regulatory Guide 1.47</u>

See paragraph (4.11) in Subsection 7.3.5.2.

<u>Regulatory Guide 1.53</u>

Regulatory Guide 1.53 provides guidance in applying the single failure criterion of Section 4.2 of IEEE-279. Conformance with Section 4.2 of IEEE 279 is discussed in Subsection 7.3.5.2.

<u>Regulatory Guide 1.62</u>

Manual initiation of ESF protective actions complies with the intent of Sections 4.16 and 4.17 of IEEE 279 as required by Regulatory Guide 1.62 as discussed in Subsection 7.3.5.2, except that the Core Spray System and Containment Isolation System initiation circuits do not comply.

<u>Regulatory Guide 1.75</u>

Although built and operational prior to the issuance of IEEE 279, the ESF Actuation Systems comply to a large extent with IEEE 279 as discussed above in this section, and with General Design Criteria 3, 17, and 21 of Appendix A to 10CFR Part 50 as required by Regulatory Guide 1.75. Conformance with Criterion 3 is further discussed in Sections 3.1.3 and 9.5.1.1. Conformance with Criterion 17 is further discussed in Sections 3.1.13 and 8.3. Conformance with Criterion 21 is further discussed in Section 3.1.17.

<u>Regulatory Guide 1.89</u>

See Sections 3.10 and 3.11 for a discussion on equipment qualification.

<u>Regulatory Guide 1.100</u>

See Section 3.10 for a discussion of qualification testing.

<u>Regulatory Guide 1.105</u>

See Section **7.2.2.3** for conformance with this guide.

<u>Regulatory Guide 1.118</u>

Instrumentation is checked, tested and calibrated as discussed in the Technical Specifications.

TABLE 7.3-1
(Sheet 1 of 4)

REACTOR PROTECTION SYSTEM ENGINEERED SAFETY FEATURE SYSTEMS ACTUATION

	Parameter	Sensor Number	Actuated Relays (Footnote 1) I	II	III	IV	Trip Pt. (Footnote 2)	Action
	Reactor Vessel (Main Steam) Isolation							
1.	DELETED							
2.	Steam Line High Flow	RE22A, E	1K15	1K73			120% rated flow	Also closed main steam line drain isolation valve, off-gas exhaust valve V7-31 and holdup drain Valve V7-29.
		RE22B, F	1K16	1K74				
		RE22C, G	2K15	2K73				
		RE22D, H	2K16	2K74				
3.	Trunnion Room High Temperature	1B10A, E, J, N	1K15	1K73			178°F	
		1B10B, F, K, P	1K16	1K74				
		1B10C, G, L, Q	2K15	2K73				
		1B10D, H, M, R	2K16	2K74				
4.	Main Steam Line Low Pressure	RE23A	1K117	1K73			825 psig	Same action as Steam Line High Flow, Item 2.
		RE23C	1K118	1K74				
		RE23B	2K117	2K73				
		RE23D	2K118	2K74				

Footnote 1: The actuated relays subcolumns give the order of relay actuation, i.e. sensor operates the relay in Subcolumn I, relay contacts in Subcolumn I actuate the relay in Subcolumn II, etc.

Footnote 2: Nominal setpoints shown. Refer to Standing Order #1 for actual setpoints.

TABLE 7.3-1
(Sheet 2 of 4)

REACTOR PROTECTION SYSTEM
ENGINEERED SAFETY FEATURE SYSTEMS ACTUATION

	Parameter	Sensor Number	I	II	III	IV	Trip Pt.(Footnote 2)	Action
	Reactor Vessel (Main Steam) Isolation (Continued)							
5.	Reactor Low-Low Water Level	RE02A RE02C RE02B RE02D	1K19 1K20 2K19 2K20	1K71 1K72 2K71 2K72	1K73 1K74 2K73 2K74	1K75 and 1K77, 1K76 and 1K77, 2K75 and 2K77, 2K76 and 2K77	Decreasing Reactor water level at 86" above the top of the active fuel.	a. 1K71, 1K72, 2K71 and 2K72-for action, see drywell high pressure, Item 7. b. 1K73, 1K74, 2K73 and 2K74 - for action, see main steam line flow. c. 1K75, 1K76, 2K75, 2K76 - cause cleanup and shutdown system trip and isolation from the reactor. d. 1K77 and 2K77 cause recirculation pumps to trip.
	Primary Containment (Drywell) Isolation							
6.	Reactor Low-Low Water Level	See Item 5, above						
7.	Drywell High Pressure	RE04A RE04C RE04B RE04D	1K9 1K10 2K9 2K10		1K71 1K72 2K71 2K72		≤3.5 psig	Trips Subchannel 1A. Trips Subchannel 1B Trips Subchannel 2A. Trips Subchannel 2B.

NOTES: 1) Some valves will isolate on containment high radiation (See text)
2) See Sheet 3 for RBCCW System Isolation as isolating circuits are different than what is shown above

Footnote 2: Either nominal setpoints or process limits, or Technical Specification limits are shown. Refer to Standing Order #1 for actual setpoints.

TABLE 7.3-1
(Sheet 3 of 4)

REACTOR PROTECTION SYSTEM
ENGINEERED SAFETY FEATURE SYSTEMS ACTUATION

Parameter	Sensor Number	I	II	III	IV	Trip Pt. (Footnote 2)	Action
Isolation Condenser System Initiation							
8. Reactor Vessel Overpressure	RE15A RE15B RE15C RE15D	6K9 6K11 6K10 6K12				1060 psig	Open condensate return valves V-14-34, V-14-35 after 1.5 second delay. Close vent valves V-14-1, V-14-5, V-14-19, V-14-20.
9. Reactor Low-Low Water Level	RE02A RE02B RE02C RE02D	16K110A 16K110B 16K110C 16K110D	6K9 6K11 6K10 6K12			Decreasing reactor water level at 86" above the top of the active fuel	
Standby Gas Treatment System Initiation							
10. Drywell High Pressure		See Item 7, above					Start SGTS
11. Reactor Low-Low Water Level		See Item 6, above					Start SGTS
Core Spray System Initiation							
12. Reactor Low-Low Water Level	RE02A RE02C RE02B RE02D	16K110A 16K110C 16K110B 16K110D				Decreasing Reactor water level at 86" above the top of the active fuel	Start core spray system
13. Drywell High Pressure	RV46A RV46B RV46C RV46D	16K115AX 16K115CX 16K115BX 16K115DX	16K115A 16K115C **16K115B** 16K115D	16K101A/102A 16K101C/102C 16K101B/102B 16K101D/102D		$\leq$3.5 psig	

Footnote 2: Nominal Setpoints shown. Refer to Standing Order #1 for actual setpoints.

TABLE 7.3-1
(Sheet 4 of 4)

REACTOR PROTECTION SYSTEM
ENGINEERED SAFETY FEATURE SYSTEMS ACTUATION

	Parameter	Sensor Number	I	II	III	IV	Trip Pt. (Footnote 2)	Action
Reactor Building Closed Cooling Water System (Drywell) Isolation								
14.	Reactor Low-Low Water Level	RE02C RE02B	16K110C 16K110B	16K110CC 16K110BB	6K76 6K77 and 6K78		See 12 Above	RBCCW Isolation will occur either on a sustained reactor triple low water level (2 channels per division) signals or coincident signals of drywell high pressure and reactor low low water level
15.	Drywell High Pressure	RV46B RV46C	16K125C 16K125B		6K76 6K77 and 6K78		See 12 Above	
16	Reactor Low-Low-Low Water Level	RE18A RE18C RE18B RE18D	16K217A 16K217C 16K217B 16K217D	16K217AX 16K217CX 16K217BX 16K217DX	6K76 6K76 6K77 and 6K78 6K77 and 6K78		= 4' 8" above the top of the active fuel	

TABLE 7.3-2
(Sheet 1 of 2)

NON-REACTOR PROTECTION SYSTEM
ENGINEERING SAFETY FEATURES ACTUATION

Parameter	Sensor Number	Actuated Relays (Footnote 1) I	II	III	IV	Trip Pt. (Footnote 2)	Action
Containment Spray System The Containment Spray System is initiated manually by procedure.							
1. Drywell High/**Low** Pressure	IP15A IP15C IP15B IP15D	16K6A 16K8A 16K6B 16K8B	16K25A 16K26A 16K25B 16K26B			2.9 psig (High) 0.6 psig (low)	• Automatic trip on low drywell pressure by actuating 16K25A & 16K26A (Sys. I) or 16K25B & 16K26B (Sys. II). (Drywell spray made only)
And							
2. Reactor Vessel Low-Low Water Level	RE02A RE02B RE02C RE02D	16K110A 16K110B 16K110C 16K110D					• Drywell high pressure & low low level provide input signals to the normal emergency power interlock (see text). No automatic initiation of containment spray.
Emergency Service Water System							
3. The Emergency Service Water System is initiated manually by procedure							
Standby Gas Treatment System		(See also Item 9, Table 7.3-1)					
4. Reactor Building Vent High Radiation	RN04A1 RN04A2					9 Mr/Hr includes background (no time delay)	Start SGTS, isolate secondary containment.
5. Fuel Pool Area High Radiation	RO14C9					50 Mr/Hr includes background - 2 minute time delay	Start SGTS, isolate secondary containment.

Footnote 1: The actuated relays subcolumns give the order of relay actuation, i.e. sensor operates the relay in Subcolumn I, relay contacts in Subcolumn I actuate the relay in Subcolumn II, etc.

Footnote 2: Nominal setpoints shown. Refer to Standing Order #1 for actual setpoints.

TABLE 7.3-2
(Sheet 2 of 2)

NON-REACTOR PROTECTION SYSTEM
ENGINEERING SAFETY FEATURES ACTUATION

	Parameter	Sensor Number	Actuated Relays (Footnote 1) I	II	III	IV	Trip Pt. (Footnote 2)	Action
6.	Reactor Operating Floor High Radiation	RO14B9					50 Mr/Hr includes background - 2 minute time delay	Start SGTS, isolate secondary containment
Automatic Depressurization System								
7.	Low-Low-Low Reactor Water Level	RE18A RE18C RE18B RE18D	16K217A 16K217C 16K217B 16K217D				4'8" above top of active fuel	Open Relief Valves
	and							
	High Drywell Pressure	RV46A RV46C RV46B RV46D	16K115A 16K115B 16K115C 16K115D				$\leq$3.5 psig	
	and							
	Core Spray Booster Pump Differential Pressure High	RV40A RV40C RV40B RV40D	16K114A1 6K114C16 K114B16K 114D				30.5 PSID 28.5 PSID 47.0 PSID 25.0 PSID	

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

Of the systems required for safe shutdown of the OCNGS, only the Standby Liquid Control System and the Remote Shutdown System are discussed in this section. Other systems required for safe shutdown at OCNGS are discussed in other sections, as follows:

	System	Reference Section(s)
a.	Reactor Trip System	7.2
b.	Isolation Condenser System	6.3 and 7.3
c.	Core Spray System	7.3
d.	Automatic Depressurization System	7.3
e.	Condensate Transfer System	10.4
f.	Emergency Service Water System	7.3
g.	Containment Spray System	6.2 and 7.3
h.	Emergency Power System (Emergency Diesel Generators and 125 Vdc)	8.3
i.	Safe Shutdown Instrumentation and Control System	7.4
j.	Reactor Recirculation Trip System	7.7
k.	Reactor Control and Protection System	7.2 and 7.5
l.	V Substations (1A2, 1B2)	8.3
m.	Standby Liquid Control System	9.3

7.4.1 Standby Liquid Control System (Liquid Poison System)

7.4.1.1 System Description

The Standby Liquid Control System (SLCS), described in detail in Subsection 9.3.5, is a standby, redundant, independent control system for use in the unlikely event that the control rod system is inoperable, to shut down and hold the reactor subcritical as the reactor cools and xenon decays.

The SLCS is not intended as a backup for reactor trip functions; most transient conditions that require reactor trip occur too rapidly to be controlled by this system. The SLCS is actuated only by remote manual action from the Control Room and will be used only during extreme conditions, such as jamming of several control rods in the withdrawn position.

Since inadvertent actuation of the SLCS System is very undesirable, the circuitry is arranged so that the operator must take positive action to inject poison into the reactor.

One three position rotary switch is provided on the console in the Control Room for poison system actuation. This switch has center position OFF; clockwise position SYSTEM II for poison pump NP 02B and poison valve NP 05B; and counterclockwise position SYSTEM I for poison pump NP 02A and poison valve NP 05A. Rotating this switch fires one explosive valve and starts one pump. Returning the switch to the OFF position stops the pump, but the valve remains open. Rotating the switch in the opposite direction fires the other explosive valve and starts the other pump. There are local switches at the pumps for test starting the pumps. The pump motor starters are interlocked to prevent simultaneous operation of both pumps.

The flow switch at the injection valve discharge is a positive indication of poison solution flow to the reactor. This flow switch interlocks with the Reactor Cleanup System to prevent poison removal until the operator decides to initiate the Reactor Cleanup System (Section 5.4), as required by plant conditions.

7.4.1.2 Initiating Circuits

There are no automatic initiations of the SLCS.

7.4.1.3 Logic

There is no instrumentation logic associated with the SLCS.

7.4.1.4 Bypasses

There are no bypasses in the SLCS instrumentation.

7.4.1.5 Interlocks

The pump motor starters are interlocked to prevent simultaneous operation of both pumps. When poison flow starts, ac and dc isolation valves on the Reactor Cleanup System (RWCU) are closed and the cleanup pumps stopped to prevent removal of poison by the cleanup system.

Complete redundancy exists in the full capacity pumps and full flow valves in the system, with common piping connections between pumps and valves for interchangeability.

7.4.1.6 Actuated Devices

Poison System Injection Valves

Each of the two Conax explosive valves is a double-squib-actuated shear plug. When either squib is fired, the shear plug is forced across a welded cap allowing flow through the valve body. The shear plug insert is replaceable and removable spool pieces are installed upstream of each valve body to facilitate servicing. This type of valve gives zero leakage. Each valve is designed for 30 gpm flow.

Two firing squibs are installed for redundancy in each valve. A low current is passed through each of these primers to indicate firing readiness by checking circuit continuity. The approximate firing current is 2 amperes, and the valve operating time is 0.002 seconds.

SLCS Pumps

One pump is started via the manual switch. The same switch fires the corresponding valve as described in Subsection 7.4.1.1. The pumps are interlocked to prevent both pumps running simultaneously. If a pump fails to start, the other pump is started manually.

7.4.1.7 Information Display

The following analog displays are provided on Panel 4F, in the Control Room:

Pump Discharge Pressure Gage
Pump "On" Light
"Squib-Fired" Light
Tank Level Gages
V-19-19 Position
Circuit Continuity Meter (Back of Panel 4F)

The following annunciator windows are provided on Panel 3F, in the Control Room:

Liquid Poison Tank Level High/Low
Liquid Poison Flow On
Liquid Poison Tank Temp High/Low
Liquid Poison Squib Open

7.4.2 Design Bases Information

Conformance of the Standby Liquid Control System with Section 3 of IEEE 279 is as follows:

1. The generating station conditions that require protective action:

   Failure of several control rods to insert fully. The actual number of control rods whose failure would require protective action varies with core life. Details are provided in the Technical Specifications.

2. Generating station variables that are required to be monitored in order to provide protective action:

3. Control Rod Position Indication

4. Source Range Neutron Monitor Operation

5. Minimum number and location of sensors required to monitor adequately those variables that have spatial dependence: Technical Specifications are formulated to prevent the control rods from being withdrawn unless three source range monitors (SRM) are operational. It is expected that these three SRMs would remain operational following a reactor trip which would precede the use of the SLCS.

6. The prudent operational limits for each variable in each operation are provided in the Technical Specifications.

7. The margin between each operational limit and level marking onset of unsafe conditions are provided in the Technical Specifications.

8. The level that, when reached, will require protective action is provided in the Technical Specifications.

7. The range of transient and steady-state conditions of the energy supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform has been assessed, a discussion on equipment qualification is presented in Section 3.11.

8. The malfunctions, accidents, or other unusual events that could physically damage protection system components, for which provisions must be incorporated to retain necessary protection system action are provided in Section 3.11.

9. Minimum performance requirements including system response times, system accuracies, ranges of the magnitudes, and rates of change of several variables to be accommodated until proper conclusion of the protection system action have been determined to be not applicable.

7.4.3 Final System Drawings

Refer to Section 1.7 for a list of drawings.

7.4.4 Analysis

7.4.4.1 Conformance with Section 4 of IEEE 279

Except for paragraph 4.17, Manual Initiation, paragraph 4 of IEEE 279 is not applicable to the SLCS.

7.4.4.2 Conformance to NRC General Design Criteria

The applicable NRC General Design Criteria are listed in Section 7.1 and a discussion of each is presented in Section 3.1.

7.4.4.3 Conformance with Regulatory Guides

Regulatory Guide 1.22

The explosive valves may be tested during plant shutdown. The explosive valve squib continuity is continuously monitored and annunciated in the Control Room. The remainder of the SLCS can be tested during normal plant operation.

Regulatory Guide 1.62

The SLCS is manually initiated at the system level from the Control Room by actuation of the system switch, which starts the pump and opens its corresponding valve.

7.5 SAFETY RELATED DISPLAY INSTRUMENTATION

This section describes instrumentation available to the operator in the Control Room to enable him to determine the status of the plant and the need to take immediate action. Instrumentation provided to support the systems described in Sections 7.2, 7.3 and 7.6 is included. The Neutron Monitoring System (NMS) is also described.

7.5.1 Description

Figure 7.5-1 shows the arrangement of panels in the Control Room. The safety related instrumentation is mounted as indicated in the figure.

7.5.1.1 Reactor and Drywell Cooling Panel

The Reactor and Drywell Cooling Panel (1F/2F) contains the instrumentation and controls required to monitor drywell and torus conditions. It includes the controls for the Core Spray, Containment Spray, Isolation Condenser, Reactor Building Closed Cooling Water and Shutdown Cooling Systems. The indicators and recorders are listed in Table 7.5-1.

7.5.1.2 Cleanup and Recirculation Panel

The Cleanup and Recirculation Panel (3F) contains the instrumentation and controls for the Reactor Water Cleanup System and the recirculation pumps. Panel 3F is included herein because the annunciator windows for the Standby Liquid Control System are located on this panel.

7.5.1.3 Reactor Control Panel

The Reactor Control Panel (4F) contains the Control Rod Drive System controls and the neutron monitoring indicators and recorders required to operate the reactor. The indicators and recorders are listed in Table 7.5-2.

7.5.1.4 Feedwater and Condensate Panel

The Feedwater and Condensate Panel (5F/6F) contains the control and instrumentation for the Feedwater and Condensate system. These systems are not safety related. Panel 5F/6F is included herein because the annunciators associated with the Reactor Protection System and indicators of certain reactor variables are located on this panel. The indicators and recorders are listed in Table 7.5-3.

7.5.1.5 Process Radiation Monitor Panel (Front)

The Process Radiation Monitor Panel (10F) contains the indicators, recorders and annunciators associated with radiation monitoring throughout the plant. The indicators and recorders are listed in Table 7.5-4.

7.5.1.6 Isolation Panel

The Isolation Panel (11F) contains the controls and indicating lights associated with the Containment Isolation System valves. Torus water level indicators with a range of 138 to 158 inches are also mounted on this panel.

7.5.1.7 Process Radiation Monitor Panel (Rear)

The Process Radiation Panel (1R) contains the indicators and electronics for process radiation monitors which require less frequent surveillance. The indicators are listed in Table 7.5-5. The Process Radiation Monitoring System is described in Section 11.5.

7.5.1.8 Neutron Monitoring

7.5.1.8.1 Neutron Monitoring Panel

The Neutron Monitoring Panels (3R and 5R) contain the indicators, and electronics for the Neutron Monitoring System (NMS). Indicators for Panel 3R are listed in Table 7.5-6 and indicators for panel 5R are listed in Table 7.5-7.

7.5.1.8.2 Neutron Monitoring System

Function

The purpose of the Neutron Monitoring System is to provide the capability to monitor neutron flux in the reactor core from the low intensity of the shutdown condition (approximately 10^3n/cm^2 sec) to the neutron flux anticipated in the case of overpower conditions requiring reactor scram (approximately 4x10^{13}n/cm^2 sec).

Ranges

Three basic types of chambers and signal conditioning equipment are used since no one type of detector can effectively monitor neutron flux over the entire range anticipated during reactor operation. The neutron flux is monitored over the entire range by the Source Range Monitoring (SRM) System, the Intermediate Range Monitoring (IRM) System, and the Local Power Range Monitoring (LPRM/APRM) System.

The minimum fission chambers used in the three nuclear instrumentation systems are essentially the same. However, the various systems operate at different voltages and utilize different gas (argon) pressures.

7.5.1.8.3 Source Range Monitoring (SRM) System

Four SRM channels (one per core quadrant) monitor neutron flux between 10^3 nv and 10^9 nv. Each channel indicates the magnitude of neutron flux and rate of change of neutron flux.

Major Components of Each Channel

a. Detector

Positioned axially by insert and retract mechanism.

Miniature fission chamber.

Electrode coated with U_3O_8. (The U is approximately 90% enriched).

Argon gas medium at higher pressure than detectors in other systems results in gas amplification.

Operated at a voltage of approximately 350 Vdc.

Sensitive length is 1.00 inch.

Detector output is current pulse. Current pulses are also produced by gamma radiation. Pulses produced by neutrons have greater amplitudes than pulses produced by gamma radiation.

b. Preamplifier

Located outside drywell.

Amplifies current pulses from detector.

c. Pulse Height Discriminator

Eliminates low amplitude pulses caused by gamma radiation or noise by discriminating at a preset current level.

Performs current to voltage transformation, resulting in output of rectangular voltage pulses.

d. Log Integrator

Produces a dc current output which is directly proportional to the common logarithm of the input pulse rate (from the pulse height discriminator).

e. Log Level Amplifier

Amplifies the integrator output to a level sufficient to drive the SRM level meters and recorders. (Meters and recorders have ranges of 10^{-1} to 10^{6} cps).

f. Period Amplifier

Takes output of log level amplifier and computes rate of change of signal. Output of period amplifier drives two period meters which are graduated from -100 seconds through infinity, to +10 seconds.

g. Trip Circuits

1. Each source range monitor is provided with three dual channel trips which provide the following six indications on the unit drawer (panel 3R or 5R):

(a) Inoperative - indicates that the mode switch on unit drawer is out of OPERATE position, or that there is a loss of high voltage to the detector, or that a module has been removed. This inoperative condition results in a rod withdrawal block if operating in the STARTUP or REFUEL modes and if IRMs are below range 8.

(b) Downscale - indicates that count rate is less than 0.5 cps. This is merely an alarm condition indicative of a channel failure.

(c) Retract Permit - indicates that the count rate is greater than 100 cps. The detector may be withdrawn from fully inserted position without generating rod withdrawal block. Note that this rod block is only effective when the mode switch is in STARTUP or REFUEL modes and when IRMs are below range 8.

(d) Hi-SRM Hi count rate of $5x10^5$ cps initiates a rod block so that the chamber can be relocated to a lower flux area to maintain SRM capability as power is increased to the IRM range. This rod block is bypassed in IRM Ranges 8 and higher since a level of $5x10^5$ cps is reached and the SRM chamber is at its fully withdrawn position.

(e) Hi-Hi - indicates that the count rate is greater than 5×10^5 cps. This condition generates a reactor trip only if the refueling non-coincidence jumpers are removed.

(f) Period - indicates that the reactor period is less than 30 seconds. This is merely an alarm condition and results in no additional automatic action.

2. The previously discussed conditions also result in the following additional Control Room indications:

(a) Rod Block Display - Panel 4F

SRM Hi Count
SRM Detector Position
SRM Inop

(b) Panel 4F indications (Apron)

Period (Amber light)
Retract Permit (White light)
Hi Hi or Bypass (Red light)
Hi or Bypass (Amber light)
Dwnscl, Inop or Bypass (White light)

(c) Panel 5F Annunciator Alarms

SRM Downscale
SRM High or Inop
SRM Period
SRM High High

h. Bypass Switch

The alarms listed previously and the associated rod block or scram signals may be bypassed for one (one at a time, that is) SRM channel by operating the SRM Bypass Switch on Panel 4F. This switch is a four way throw type that prohibits the bypassing of more than one channel at a time. The bypassed condition is indicated by a white light on the associated Trip Auxiliary Drawer plus three bypass indicators on the apron of Panel 4F.

SRM Rod Block Summary

The following three conditions result in rod withdrawal blocks. These blocks are only operative when the Mode Switch is in the startup or refuel modes and when the IRMs are below range 8.

a. SRM Inoperable

Reflects: loss of high voltage to the detector, module removed, or mode switch on channel drawer not in OPERATE position.

b. Detector Position

Indicates that the count rate is less than 500 cps and the detector is not fully inserted. This rod block prevents the instrument chamber from being withdrawn too far from the core during the period that it is required to monitor the neutron flux.

c. SRM High Count Rate

Indicates that the count rate is Hi at 5×10^5 cps; initiates a rod block so that the chamber will be relocated to a lower flux area to maintain SRM capability as power is increased through the IRM range.

SRM Trip Summary

An SRM High High count rate condition, greater than 5×10^5 cps results in a reactor trip if the refueling non coincidence jumpers have been removed. This function provides additional protection during certain unusual refueling operations.

With the jumpers installed (normal condition) only an alarm is received at a count rate of 5×10^5 cps and the protective function is bypassed.

7.5.1.8.4 Intermediate Range Nuclear Instrumentation

Eight intermediate range monitors (IRMs) measure core neutron flux in the approximate range of 10^8 nv to 5×10^{12} nv. In addition to providing the operator with power level (flux) indication, the IRMs generate rod block and scram signals, as discussed below.

Major Components of an IRM CHANNEL

a. Detector

A miniature fission chamber positioned axially by detector drive system.

Operated at a voltage of approximately 150 Vdc.

Argon fill gas (at lower pressure than for the SRM detectors).

Sensitive length is 1.00 inch.

Detector produces a significant number of pulses which overlap and "pile up" to form a gross signal level which is proportional to neutron flux.

Electrode coated with U_3O_8 (with highly enriched uranium).

b. Preamplifier

Located outside drywell.

Amplifies detector output.

c. Amplifier and Attenuator

As the neutron flux of the core increases, the output from the fission chamber becomes larger. The fission chamber output is attenuated to keep the inverter input signal within the required range.

Output controlled by the associated range switch on Panel 4F.

d. Inverter

Reverses the polarity of the negative part of its input signal so that the mean square analog unit always has a positive signal.

e. Mean Square Analog Unit

Develops an output current proportional to the instantaneous square of its input voltage. This output current is proportional to the power contained in the pulses received from the fission chamber.

Gamma discrimination accomplished in the IRM channels by a Campbelling Technique, using the mean square of the fission chamber signal rather than average values. This technique greatly enhances the relative size of the neutron-fission produced signal to the significantly smaller gamma ray dependent signal.

f. Operational Amplifier

Amplifies the mean square analog unit output sufficiently to drive the recorder on Panel 4F and the local meter. Meter and recorder have two ranges (0-125% and 0-40%). Local meter has a tick mark at 127% for readability above 125% during surveillance testing.

g. Trip Units

1. Each IRM channel has two dual channel trip units. These units are tripped by the following four conditions:

   (a) Inoperative - indicates that high voltage is lost to the detector, that the module is removed or that the mode switch on the channel drawer is not in the OPERATE position.

   (b) Downscale - indicates a flux level less than 5% on the 0-125% scale or less than 1.6% on the 0-40% scale.

   (c) Hi - indicates a flux level greater than 108% on the 0-125% scale or greater than 35% on the 0-40% scale.

   (d) Hi Hi - indicates a flux level greater than 120% on the 0-125% scale or greater than 38% on the 0-40% scale.

2. The conditions listed above are indicated by lights on the IRM drawer. In addition, each will generate alarm and rod withdrawal blocks and trips, as described in the summaries below.

h. Bypass Switches

The alarms, rod blocks, and scrams from the trip units may be bypassed for up to two (two channels at one time, that is) IRM channels by using the IRM bypass switches on panel 4F. These are four way throw type switches, one for channels 11, 12, 13, or 14 (Reactor Protection System 1) and one for channels 15, 16, 17, or 18 (Reactor Protection System 2). A review of the IRM detector locations shows that it is impossible to bypass more than one channel in any core quadrant.

IRM Rod Block Summary

The following four (4) conditions result in rod withdrawal blocks. These blocks are only operative in the startup and refuel modes:

a. Inoperative

   Indicates: high voltage to detector lost, mode switch on IRM drawer not in OPERATE position or module removed.

b. Downscale

   Indicates the flux level is less than 5% on the 0-125% scale or 1.6% on the 0-40% scale. This rod block is not operative in IRM range No. 1. The IRM downscale rod block, in conjunction with the chamber full-in position and range switch setting, provides a rod block to assure that the IRM is in its most sensitive condition before startup. If the two latter conditions are satisfied, control rod withdrawal may commence even if the IRM is not reading at least 5% on the 0-125% scale.

c. Detector Position

Indicates that the detector is not in the fully inserted position. This rod block prevents the IRM detector from being withdrawn to an insensitive area of the core during reactor startup.

d. APRM Downscale/IRM Upscale

Indicates a flux level greater than 108 on the 0-125% scale or 35% on the 0-40% scale, coincident with an APRM channel reading of less than 2%.

IRM Scram Summary

a. High High or Inop

Indicates that the flux level is greater than 120% on the 0-125% scale or 38% on the 0-40% scale; or channel inoperative. This function is only operative in the STARTUP and REFUEL modes. Channels 11, 12, 13 and 14 are associated with Reactor Protection System #1 and channels 15, 16, 17 and 18 are associated with Reactor Protection System #2. Trips of both Reactor Protection Systems are required to generate a full trip.

b. IRM High High or Inop/APRM Downscale

Indicates that the flux level is greater than 120% on the 0-125% scale or 38% on the 0-40% scale is, or that the IRM channel inoperative coincident with associated APRM channel reading less than 2%. This trip function is only operative in the RUN mode. Simultaneous trips of IRM and APRM channels must occur in the following combinations to generate a half scram:

	Combination	IRM Channels	APRM Channels
RPS Channel No. 1	1	11, 12	1, 2
	2	13, 14	3, 4
RPS Channel No.2	1	15, 16	5, 6
	2	17, 18	7, 8

Use of Ranges to Provide Rate Protection

A reasonable rate of power increase is maintained by requiring that each IRM be on scale (between downscale and high trip points) in order to obtain a control rod withdrawal permissive. If a second high level point (Hi-Hi) is reached, a half scram condition results. Hi-Hi trips of the channels in both Reactor Protection Systems result in a full trip.

7.5.1.8.5 Detector Drive System

The Detector Drive System is common to both SRMs and IRMs. The SRM and IRM detectors are positioned axially between a fully inserted position (2 feet above the core mid plane) and a fully withdrawn position (2 feet below the bottom of the core) by the Detector Drive System.

Although a detector may be positioned to any position between the limits of travel, only the positioning at the ends of travel is indicated (remotely on panel 4F).

Objective

To remove a detector from the core when it is no longer required to be there. Prevents detector failure during power operations and prolongs the life of the fission chamber by minimizing contamination of its internal gas and minimizing internal heat production due to high fission rates.

Major Components

The Detector Drive System consists of a motor module, a detector drive assembly and a dry tube. The following paragraphs provide a brief description of each of the system components.

a. Motor Module

The module consists of a 208 volt, 3 phase, 60 cycle fractional HP ac motor, a drive shaft, and a limit switch, all mounted on an aluminum base plate. The limit switch, which is chain linked to the motor output shaft, controls the insertion limits (FULL-IN and FULL-OUT) of the detector; the switch also provides outputs to auxiliary relays for indicators in the Control Room.

A flexible drive shaft, connected to the output shaft of the motor module, transmits power to the gear box of the detector drive assembly.

The motor module is located outside the reactor support structure for three reasons; to avoid the crowded conditions in the structure, to keep the electrical equipment out of an environment which is usually damp, and to reduce the radiation exposure to the equipment.

Detector Drive Assembly

This assembly consists of the gear box, the housing, the drive tube and the shuttle tube. The gear box is mounted on the housing and drives the drive tube. The drive tube is a round tube with holes along one side. The shuttle tube is a long, slender tube assembly mounted on the end of the drive tube. The shuttle tube and drive tube move as one.

The detector assembly consists of the detector and the detector cable. The detector assembly is equipped with one of two types of detectors to measure two different ranges of neutron flux activity. The lower range detector provides an input to the source range monitor (SRM) and the higher range detector provides an input to the intermediate range monitor (IRM). The detector assembly is installed inside the protective shuttle tube and drive tube. The drive tube extends from the back of the detector drive to mate with an external connector.

b. Dry Tube

This is the tube that mates with the reactor vessel, providing a dry channel within the reactor core. The detector assembly, shuttle tube, and drive tube are driven up and down in the dry tube.

Operation and Precautions

There are three selector switches and one drive control switch on Panel 4F. Either SRM or IRM detector movement can be selected but not simultaneously. Any one detector can be moved or all SRMs or IRMs detectors can be moved together. All movement is manual. There is no automatic movement nor are there any interlocks to prevent inadvertent movement.

7.5.1.8.6 Local Power Range Monitor (LPRM) System

Functions

Selected groups of LPRMs provide input signals to the Average Power Range Monitoring System for bulk power level monitoring and automatic core protection. The system supplies signals proportional to local neutron flux to drive the indicating meters to be used for manual evaluation of core performance.

The system generates signals to annunciators which indicate high local flux or low detector reading. The system also generates the rod withdrawal block.

Major Components

Detector Assemblies

Prior to the 15R Refueling Outage there were thirty-one LPRM detector strings each containing four fission chambers are distributed so as to form four horizontal planes throughout the core. In 15R, two LPRM strings were removed from service due to interferences in the instrument guide tube thimbles. In 19R, these two LPRM strings were reinstalled and returned to service. The detector assemblies are inserted into the core in spaces between the fuel bundles and through thimbles mounted permanently at the bottom of the core lattice which penetrate the bottom of the reactor vessel. These thimbles are welded to the reactor vessel at the penetration point. They extend down into the access area where they terminate in a flange which mates to the mounting flange on the incore detector assembly. The detector assemblies are locked, at the top end, to the top grid of the core by means of a spring loaded plunger. This type of assembly is referred to as top entry-bottom connect, since the assembly is inserted through the top of the core and penetrates the bottom of the reactor vessel. Special water sealing caps are placed over the connection end of the assembly and over the penetration at the bottom of the vessel during installation or removal of an assembly. This prevents the loss of reactor coolant water upon removal of an assembly and also prevents the connection end of the assembly from being immersed in the water during installation or removal.

Each in-core flux detector assembly contains four miniature fission chambers with an associated solid sheath cable. Each individual chamber of the assembly is a moisture proof, pressure sealed unit. The assembly also contains the calibration tube for the traversing probe, and an enclosing tube around the entire assembly. The enclosing tube around the entire assembly contains vent holes evenly spaced along its length. These holes allow circulation of the reactor coolant to cool the fission chambers. The miniature fission chambers are essentially the same as those utilized by the SRM and IRM systems except that they are operated at a lower high voltage (100 Vdc).

b. Flux Amplifier

The flux amplifier consists of two identical dc amplifiers and two identical trip units. Thus, each amplifier module processes signals from the two LPRM detectors. The trip units are tripped under the following two conditions:

LPRM Upscale - When flux level is greater than 97 watts/cm^2.

LPRM Downscale or Inoperative - When flux level is less than 2 watts/cm^2 or when high voltage to the detector lost, the mode switch on the amplifier drawer is not in OPERATE position, or module removed.

Power Supply and Monitor

The power supply and monitor furnishes three regulated voltages to operate the flux amplifiers and their associated detectors: +115 Vdc, +100 Vdc and +85 Vdc. Each power supply and monitor is associated with two flux amplifiers (four LPRM detectors).

The monitoring circuit on the power supply and monitor is comprised of front panel meter M1, selector switch S1, and three resistors (R4, R5 and R6). The monitoring circuit performs two functions: 1) when selector switch S1 is placed in one of the PERCENT POWER positions, it connects the meter between one of the flux amplifier outputs and the +100 volt dc bus (+100 volts dc is common in the flux amplifier), and 2) when selector switch S1 is placed in one of the SUPPLY VOLTS positions, the meter is connected between the power supply output voltage and common (in the case of the +100 volt dc output) or +100 volts dc (in the case of the ±15 volt dc outputs). Thus, the monitoring circuit provides the capability to monitor the power supply or flux amplifier outputs on the front panel meter.

LPRM Indications and Alarms

LPRM outputs are indicated on the vertical section of Panel 4F. The four meters associated with the detectors at a given radial core location are stacked together in sequence (i.e., for the A, B, C and D levels). Each meter has a range of 0-125 watts/cm^2 and is related to heat flux. There is an amber light next to each LPRM meter which illuminates if the associated channel is upscale (greater than 97 watts/cm^2) downscale (less than 2 watts/cm^2) or bypassed. An LPRM channel is bypassed by operating a toggle switch (one per channel) inside the associated trip auxiliary drawer.

Two annunciator alarms on Panel 5F are associated with the LPRMs: 1) LPRM High, 97 watts/cm^2, and 2) LPRM Downscale (less than 2 watts/cm^2) or Inop (loss of high voltage, mode switch not in OPERATE or module removed).

7.5.1.8.7 Average Power Range Monitor (APRM) System

Function

The APRM system consists of the electronic equipment which averages the output signals from selected LPRM amplifiers, the trip units which actuate automatic protective actions when APRM signals exceed preset values and the signal readout equipment. This system provides continuous indication of average reactor power from a few percent to 150% rated power.

There are eight APRM channels - two per core quadrant. Channels 1, 2, 3 and 4 are associated with Reactor Protection System No. 1 and Channels 5, 6, 7 and 8 are associated with Reactor Protection System No. 2. Thus, each core quadrant is monitored by two APRM channels - each associated with a different Reactor Protection System. The APRM channels in a given core quadrant utilize the same LPRM detector strings with the Reactor Protection System No. 1 APRM channels receiving inputs from the A and C level detectors and Reactor Protection System No. 2 APRM channels receiving inputs from the B and D level LPRM detectors. Each APRM channel normally averages the inputs of eight LPRM channels.

The APRM high power trip and rod block set points are flow biased. That is, they vary with recirculation flow. Consequently, the APRM system requires recirculation flow signals to compare with the APRM power level signals.

Processing of Recirculation Flow Signals

a. Recirculation Flow Transmitters

   Oyster Creek's five recirculation loops each have two flow transmitters (one per RPS division). The transmitters are physically located on instrument rack RK04. These transmitters monitor the differential pressure across the flow venturis. The differential pressure is proportional to the square of the flow rate. The transmitters perform a square root function internally and provide output current signals that are proportional to flow. These flow signals are sent to the Control Room panels 3R (division 1) and 5R (division 2) where the remaining electronics are located.

b. Current-to-Voltage (I-to-V) Converters

   These modules provide input isolation and convert the current signal from each transmitter to proportional voltage signals. The I-to-V converters also act as power supplies for the transmitters.

c. Summers

   Two summers are used in each division to develop total recirculation flow signals as follows. Since each summer has only four inputs, flow summing must be done in stages. The voltage outputs from three of the five I-to-V converters are input into the first summer. Its output, along with outputs from the remaining two I-to-V converters is input into the second summer. The output from the second summer is a voltage signal proportional to the total flow rate.

The individual loop and total flow voltage signals are used to provide output signals as described below.

d. Output Flow Signals

Individual Loop Flow Signals

The individual loop flow voltage signals from the I-to-Vs go through Voltage-to-Current (V-to-I) converters. These isolated current signals are provided to the plant computer (division 1), and to individual loop flow indicators on panel 3F (division 2).

Total Flow Indication

The total flow voltage signals in each division go through V-to-I converters. These isolated total flow current signals are provided to a recorder on panel 3F (division 1) and to an indicator on panel 4F (division 2).

Total Flow Signals to APRM Trip Bias Units

The total flow voltage signals in each division go through V-to-I converters that act as output isolators. The output current signals are sent through dropping resistors. The resulting total flow voltage signals are sent to the APRM trip bias units channels 1, 2, 3 and 4 (division 1), and channels 5, 6, 7 and 8 (division 2).

Total Flow Signal to Other Division

The total flow voltage signals in each division are provided to the other division through output and input isolators. These signals are used for the comparator mismatch function as described below.

e. Trip Functions

Upscale Trip

The total flow voltage signal in each division is sent to a respective absolute alarm module. Recirculation flow in excess of 120% rated flow results in an upscale trip which is provided as an input to the Reactor Manual Control System (RMCS). This input to the RMCS from each division of the recirculation flow monitoring electronics produces a rod block and a rod block display.

Inop Trip

Isolated total flow voltage signals (identical to the signals provided to the APRM trip bias units) are sent to respective absolute alarm modules. Loss of power to the signal processing modules would cause this signal to go downscale. The absolute alarm module detects this downscale condition and provides an inop trip. This input to the RPS from each division of the recirculation flow monitoring electronics produces a half scram. In addition, a rod block, rod block display, and alarm are produced by the inop trip.

Comparator Mismatch Trip

The total flow signal from each division is compared to the isolated total flow signal received from the other division. A deviation alarm module in each division produces a mismatch trip if the flow signals differ by more than 10% of rated flow. The mismatch trip is provided as an input to the Reactor Manual Control System to produce a rod withdrawal block. In addition, a rod block display and APRM FLO BIAS OFF NORMAL alarm are produced.

The status of the above trip conditions is indicated at the electronics nests located in panels 3R (division 1) and 5R (division 2).

Major Components of an APRM Channel

a. Input Bypass Switches

These eight switches (S1-S8) are located inside the drawer behind the front panel and are turned counterclockwise to bypass an associated input signal.

b. Input Counting Circuit

The input counting circuit: 1) generates a channel INOPERATIVE condition when more than three LPRM inputs are bypassed and 2) provides a signal proportional to the number of unbypassed LPRM inputs (to be read out on an APRM front panel meter).

c. Averaging Amplifier

This amplifier averages the LPRM input signals and develops an output signal related to percent of rated core thermal power. An APRM channel is calibrated to read percent of rated thermal power by using heat balance techniques to determine thermal power and making appropriate amplifier gain adjustments. The output signals from the individual LPRM amplifiers, which are averaged in the APRM channel, are not affected by APRM channel calibration. After an LPRM input has been bypassed or removed from bypass condition, the APRM gain may have to be adjusted.

d. Flow Control Trip Reference Card

Develops input signals which are used for the trip units.

e. Trip Units

Each APRM channel has six trip units which are tripped under the following conditions:

1. Inoperative

Indicates that the mode switch on the APRM drawer front panel is not in the OPERATE position, that the module is removed, or more than three LPRM inputs are bypassed.

2. Downscale

   Indicates reactor power level less than 2% of full scale.

3. Deleted

4. Hi

   Indicates that core power has exceeded a predetermined level for the existing recirculation flow condition. When this alarm point is reached it results in: a non-resettable rod block; a rod withdrawal block if APRM channels 1, 2, 3, or 4 are tripped; or a rod deselect block if APRM channels 5, 6, 7 or 8 are tripped.

5. Hi-Hi

   Indicates that core power has exceeded a predetermined level for the existing recirculation flow condition. When this alarm setpoint is reached it results in a Reactor trip signal to the RPS channel (one out of four, twice).

f. Bypass Switches

   APRM channels are bypassed by throwing the associated bypass switch located on the apron of Panel 4F. There are two bypass switches. One switch bypasses channel 1, 2, 3 or 4 and the other switch bypasses channel 5, 6, 7 or 8. These four way throw switches preclude the bypassing of more than one channel in each RPS channel. In addition, an electrical interlock prevents more than one channel in a given core quadrant from being bypassed.

Control Room Indications Related to APRMs

Indications on auxiliary drawer:

a. Input Bypassed
b. APRM Bypassed
c. Alarm (Rod Block)
d. Scram
e. Downscale
f. Inop

APRM Rod Block Summary

a. IRM High/APRM Downscale

   This coincident rod block is operative in the STARTUP and REFUEL modes only. Refer to the discussion under IRM rod blocks.

b. APRM Downscale

   This rod withdrawal block is operative in the RUN mode only and is generated when the downscale trip unit is tripped at less than 2% indicated. The Technical Specifications

require the APRM downscale block trip setpoint to be greater than or equal to 2/150 of fullscale.

c. <u>APRM High or Inop</u>

This rod block function is operative in all modes of operation and is generated by the tripping of the alarm or when units are inoperative as previously described.

The Technical Specifications specify the operating limits for the APRM flux rod block setpoint in equation form. For recirculation flow less than or equal to rated, the APRM high flux rod block trip point must be set at a calculated percent of rated neutron flux when total peaking factors in all types of fuel are less than or equal to those in the Technical Specifications. The core normally operates such that the total peaking factors listed in the Technical Specifications are not exceeded. However, if these specified peaking factors are exceeded, the APRM high flux rod block trip point must be reduced.

The APRM high flux rod block trip point of rated recirculation flow cannot exceed 115% of rated flux.

The Technical Specifications further require that both APRM downscale and APRM high flux rod withdrawal block functions be operable for at least three channels in each trip system when operating in the REFUEL, STARTUP or RUN modes. In the REFUEL mode with the reactor temperature less than 212°F and the vessel head removed or vented, the APRMs are no longer required to be operable. Two additional requirements are that Technical Specifications must not be violated by any channel bypassing and that two APRMs in the same quadrant may not be concurrently bypassed or inoperable with two exceptions. Refer to the APRM high flux scram discussion for the two exceptions.

<u>APRM Trip Summary</u>

a. <u>IRM High-High or Inoperative/APRM Downscale</u>

See discussion under IRM Scrams.

<u>APRM High-High or Inoperative</u>

This trip function is operative in all modes of operation and is generated by the tripping of the scram or inoperative trip units as previously described. At least one APRM channel in each RPS channel must have a high flux or inoperative trip condition to produce a full reactor trip.

The Technical Specifications specify the operating limits for the APRM high flux scram in equation form. For recirculation flow less than or equal to rated, the APRM high flux trip point must be set at a calculated percent of rated neutron flux when total peaking factors in all fuel types are less than or equal to those in the Technical Specifications. The core normally operates such that the total peaking factors specified in the Technical Specifications are not exceeded. However, if these specified peaking factors are exceeded, the APRM high flux trip point must be reduced. Refer to Technical Specifications for further detail.

The APRM high flux trip point at rated recirculation flow cannot exceed 120% of rated flux.

The Technical Specifications require the APRM high flux scram function to be operable for at least three channels in each trip system when operating in the REFUEL, STARTUP or RUN modes. In the REFUEL mode with the reactor temperature less than 212°F and the vessel head removed or vented, the APRMs are no longer required to be operable.

Two additional requirements are that Technical Specifications must not be violated by any channel bypassing and that two APRMs in the same quadrant may not be concurrently bypassed or inoperable with the following exceptions:

1) If one APRM in a quadrant cannot satisfy the Technical Specifications, the other APRM channel in that quadrant may be removed from service for up to six hours for test or calibration without inserting a trip signal into its trip system only if the first APRM is unbypassed and meets Technical Specification 3.1.B.1, and no control rod is moved outward during the calibration and/or test.

2) When in the REFUEL mode, with the reactor temperature less than 212°F and the vessel head removed or vented, the APRMs are no longer required to be operable.

7.5.1.8.8 Traveling In-Core Probe (TIP) System

Functions

a. Calibration of the LPRM detectors.

b. Determination of the axial neutron flux levels for power distribution measurements.

Major Components

a. Detector

A fission chamber which is identical to the LPRM fission chambers. Attached to the fission chamber is a drive cable which is a triaxial signal cable, helically wrapped in carbon steel.

b. Guide tubing

Commonly known as TIP tubing, provides a guide for the TIP detector throughout its traverse from the storage pig to the core top.

c. Drive Mechanism

Drives the TIP detector through the guide tubing. Two drive speeds are available, 60 ft./min and 15 ft./min. Features include electrical braking and a

double reel cable takeup arrangement which allows direct cable connection without brushes or slip rings. This significantly reduces noise. A digital position detector provides core top and bottom indication and control, and continuous digital indication of detector position.

d. Storage Pig

Provides radiation shielding when a detector is fully retracted and access is required to the TIP room for maintenance. It contains a limit switch to indicate "IN SHIELD." The IN SHIELD condition must be satisfied in order to close the ball isolation valve for the TIP guide tubing (interlocked).

e. Indexing Mechanism

Located inside the drywell, the indexer allows the selection of any of 10 core locations for detector insertion. The 10th position selects a TIP location which is an LPRM string common to all four TIP machines via a four way connector. Interlocks prevent indexer channel change when a detector is inserted beyond the indexer. Other interlocks prevent detector insertion until the proper channel is lined up.

f. Isolation Valve System

Consists of a ball valve and an explosive actuated shear valve. The purpose of the valves is to provide isolation should a coolant leak develop in a TIP guide tube.

The ball valve is normally closed except when a detector is inserted. Although tip insertion is a manual operator action, the ball valve is normally opened and closed automatically. An interlock automatically opens the ball valve when the tip is driven out of this shield. The ball valve may also be opened and closed by the operator. Another interlock will deenergize the drive mechanism should the ball valve not open after the forward (insert) direction is selected for a TIP.

The shear valve is used only to isolate a leak while a detector is inserted (out of shield) and either power is lost to the drive mechanism or some other fault has occurred which prevents retraction of the TIP. The shear valve is manually activated by a keylock switch on the valve control monitor drawer.

g. TIP Flux Probing Monitor

Consists of a dual channel flux amplifier and a power supply. The amplifier section amplifies the detector signal sufficiently to provide an input to the plant computer for determining flux level.

The power supply section supplies operating voltages to the flux amplifier and also supplies a regulated 100 Vdc for detector biasing.

h. Deleted

i. Valve Control Monitor

Controls and indicates the position of the ball and shear valves. As previously stated, interlocks prevent ball valve closure if the TIP is not in the IN SHIELD condition, and will stop detector TIP insertion within five seconds if the ball valve fails to open on an insert signal.

Indicating lights on the VCM drawer and the Drive Control Unit monitor ball valve position. On Panel 11F a common ball valve position is provided which indicates if all valves are closed or any one is open.

The shear valve keylock switch is located and powered from this drawer. When the shear valve is actuated, a guillotine will cut the TIP guide tube and the drive cable inside it and will seal the guide tube. When actuated "SHEAR VALVE MONITOR" and "SQUIB MONITOR" lights will illuminate.

j. Drive Control Unit

Provides selection of a reactor core location for detector insertion. It determines (via limit switches) and displays core top and bottom positions. It provides control of detector insertion and retraction. It also supplies the X-axis signal to the XY plotter and to the plant computer. The following indicator lights monitor the TIP throughout its operation.

1. READY - meaning that the indexer is properly aligned to the selected guide tube.

2. CORE TOP - which indicates that the TIP is at the top of core, i.e., the end of forward travel; it deenergizes forward motion.

3. IN CORE - indicates that the TIP is inserted past the core bottom position; the TIP must move at SLOW speed while in core and does so automatically.

4. IN SHIELD - indicates that the TIP is fully retracted and is in the shielded storage pig; the ball valve may now CLOSE.

5. SCAN - ensures that the XY plotter pen is down and making a trace.

6. LOW SPEED - selects slower speed; TIP will move at a slower speed throughout its travel distance and will not automatically shift to a faster speed between the indexer and the core bottom.

7. REV - indicates that the TIP is being retracted (reverse).

8. FWD - indicates that the TIP is being inserted (forward).

9. Valve - when lamp is dimly lit means ball valve OPEN
- when lamp is brightly lit means ball valve CLOSED.

7.5.1.9 Reactor Protection Panel

The Reactor Protection Panels (6R and 7R) contain the reactor trip relays, channel test switches, and other infrequently used RPS control switches. Panel 7R is identical to Panel 6R except that it is in the other electrical channel.

7.5.1.10 Reactor Protection System Operation Panel

The Reactor Protection System Operation Panel (6XR) contains the control rod pilot scram valve test switches and rod group test jacks.

7.5.1.11 Shutdown and Fuel Pool Cooling Panel

The Shutdown and Fuel Pool Cooling Panel (10R) contains power supplies and signal conditioning electronics for various systems, and monitors temperatures in various plant areas. See Table 7.5-8 for a list of indicators on this panel.

7.5.1.12 Valve Position Indication Panel

The Valve Position Indication Panel (15R) contains the indication and alarm electronics to indicate open or closed status of the nine safety valves and the five electromatic relief valves.

Each of the nine safety valves and the five electromatic relief valves have identical position monitoring systems as described herein: A piezoelectric accelerometer, which generates an electrical signal when it is vibrated, is clamped to the discharge pipe near each valve. The signal, which is a positive indication of flow through the valve (hence its position), is amplified by a pre-amp located inside the drywell. The signal is then transmitted to Panel 15R. This panel contains a separate module for each valve which conditions the signal for indication and alarm functions. Each module contains adjustments for the alarm set points, alarm lights, testing provisions, and an indication of valve position. Panel 15R also has the capability to provide an audio signal to a speaker or headset which can be used to verify an alarm. On Panel 1F/2F, which is in full view of the Control Room Operator, are located twenty-one individual position indicators and a common alarm window.

During 13R, the acoustic monitors removed for the safety valve reduction modification were relocated such that seven of the nine remaining safety valves now have two operational acoustic monitors.

Each of the five electromatic relief valves acoustic monitors is provided with a spare channel. These spares are located in the drywell, and are normally unpowered. In the event that a channel monitoring an EMRV should fail during normal plant operation, the installed spare will be connected near the drywell penetration #54 in the Reactor Building.

7.5.1.13 Safety Parameter Display System

The Safety Parameter Display System (SPDS) aids the Control Room operators in determining the overall plant safety status during power operation (except startup) and post reactor trips, and assists in the identification of abnormal plant conditions. Since the SPDS only augments the Control Room instrumentation for safe reactor operation, the SPDS components are not required to meet the single failure criteria, nor be qualified to Class 1E requirements.

The SPDS provides this overview of the plant status by displaying five Critical Safety Functions (CSFs). A CSF measures the safety status of a group of plant parameters which together convey a coherent meaning with regard to plant safety. The displayed CSFs are:

CSF1 - Reactivity/Power Distribution

CSF2 - Heat Removal/Core Cooling

CSF3 - Reactor Coolant System/Fuel Integrity

CSF4 - Containment Conditions

CSF5 - Radiation Control

The displayed CSFs are consistent with the NUREG 0737 (Supplement 1) safety categories and the Emergency Operating Procedures (EOPs). The SPDS also meets the intent of NUREG-1342 with the following clarifications:

The EOPs confirm reactor shutdown by control rod position instead of monitoring the SRMs.

SPDS provides an indication of containment isolation demand in lieu of isolation valve status.

The Plant Computer System (PCS) uses a maximum-minimum range check for validation.

The SPDS is used during power range operation (except startup) and reactor trips.

The SPDS as a subset of the PCS has a dedicated computer monitor located in the Control Room. A utility monitor and an alarm monitor are also available in the vicinity of the dedicated SPDS terminal. The utility monitor serves as an SPDS display terminal or to trend SPDS parameters. The alarm monitor provides a status of the plant computer system alarms which includes alarms associated with the SPDS. Each CSF has a warning condition (priority 2) and an alert condition (priority 1) for a total of five priority 2 alarms and five priority 1 alarms.

The Class 1E signals from the safety systems pass through Class 1E analog and digital isolator to the non-Class 1E PCS. These Class 1E isolators comply with the environmental and seismic qualifications for their application. The isolators operate in a mild environment, and their seismic qualification envelope exceeds the floor response spectra for the Reactor Building (El 95).

7.5.2 Analysis

7.5.2.1 Design Bases

The safety related displays are provided to aid the operator in achieving safe shutdown of the plant, and in maintaining the plant in a safe shutdown condition from either inside or outside the Control Room.

The Plant Computer System which includes the Safety Parameter Display System is designed to determine the safety status of the plant during normal plant operations as well as for retrieval of event data during an operational transient.

7.5.2.2 Conformance to NRC General Design Criteria

The applicable General Design Criteria are listed in Section 7.1 and discussed in Section 3.1.

7.5.2.3 Conformance to IEEE 279

The conformance of the protection systems to IEEE 279 has been discussed in Sections 7.2, 7.3 and 7.4. A discussion of the conformance of the Plant Computer System will be supplied later; also see Subsection 7.5.2.6.

7.5.2.4 Information Required for Operator to Perform Manual Functions

7.5.2.4.1 Normal Operation

During normal operation, the operator must have sufficient information to monitor the status of safety related equipment. Accordingly, the status of these variables monitor the status of the devices they actuate which are provided by means of indicators and annunciators. The important parameters are displayed so that the operator can evaluate the performance of control systems. Pressure, temperature level and the status of the various safeguards devices are provided to ensure the capability to mitigate the consequences of a LOCA. Control rod position indication allows the operator to assess operation of the CRD and to insure that safe rod patterns are maintained. Refer to Section 7.7 for details of control rod position indication and for listing of other non safety related display instrumentation available to the operator.

7.5.2.4.2 Maintaining Safe Shutdown

In compliance with General Design Criterion 19, safety related displays required for maintaining a safe shutdown are provided redundantly inside the Control Room and outside the Control Room at various locations. These displays are provided to aid the operator in placing and maintaining the unit in a safe hot shutdown condition and for going to a subsequent cold shutdown by providing for the monitoring of essential primary system parameters. The safety related displays provided outside the Control Room in compliance with Criterion 19 are considered adequate to achieve and maintain safe shutdown. Additionally, redundant safety related displays are provided inside the Control Room to monitor essential parameters for the safe shutdown of the unit.

7.5.2.4.3 Other Occurrences

The indications available to the operator provide sufficient information so that he can monitor the status of the plant during and after all operational occurrences. The indication includes that necessary to: monitor the status of all protection systems and systems actuated by protection systems; assess the operation of all control systems; monitor reactor and reactor plant status; and monitor conditions in the Containment. Tables 7.5-1 through 7.5-8 list the safety related display instrumentation available for operator use.

Refer to Section 7.7 for listing of non safety related display instrumentation available.

7.5.2.4.4 Postulated Accidents

The safety analysis in Chapter 15 enumerates the assumptions made in the study of postulated accidents.

7.5.2.5 Conformance to NRC General Design Criteria

See Table 7.1-1 for applicable General Design Criteria and Section 3.1 for General Design Criteria discussions.

7.5.2.6 Conformance to IEEE 279

The safety related display instrumentation complies with the following applicable portions of Section 4 of IEEE 279:

(4.2) Single Failure Criterion

No single failure within the safety related display instrumentation will prevent proper operation of two redundant displays monitoring the same parameter of a single component or two non redundant displays monitoring the same parameter of redundant components.

(4.3) Quality of Components and Modules

Equipment manufacturers are required to use high quality components and modules in equipment design and fabrication. Quality control procedures used during fabrication and testing verify compliance with this requirement.

(4.4) Equipment Qualification

The safety related display instrumentation is qualified as discussed in Section 3.10 and 3.11.

(4.5) Channel Integrity

The safety related display instrumentation is designed to operate under extremes of conditions relating to environment, energy supply, malfunctions and accidents. All of this equipment is designated Seismic Category I.

(4.6) Channel Independence

Instrument strings that provide signals for the same function are physically separated and have separate and independent vital battery backed power supplies.

(4.7) Separation of Protection and Control Systems

The safety related display instrumentation is kept separate from process controls.

(4.8) Derivation of System Inputs

Wherever practical, signals are derived from direct measurements of desired process variables and used directly for the intended function.

(4.9) Capability of Sensor Checks

The sensors of the safety related display instrumentation can be checked for operational availability by: (1) Cross checking between instrument strings that are measuring the same parameter and have readouts available, and/or (2) Cross checking between instrument strings that bear a known relationship to each other and that have readouts available.

(4.10) Capability of Test and Calibration

The safety related display instrumentation has provisions for string calibration at regular intervals and is tested by observing and cross checking between instrument strings.

(4.18) Access to Setpoint Adjustment, Calibration and Testpoints

All safety related display instrumentation equipment will be located in areas of controlled access.

(4.20) Information Readout

The safety related display instrumentation is designed to provide the operator with accurate, complete and timely information pertinent to the safety of the plant.

(4.21) System Repair

The safety related display instrumentation is designed to facilitate the recognition and location of equipment malfunctions.

7.5.3 References

(1) GPUN Letter, P.B. Fiedler to D.G. Eisenhut (NRC), "Supplement 1 to NUREG 0737, SPDS Parameter Selection/Safety Analysis Study", dated April 2, 1984.

(2) GPUN Letter, P.B. Fiedler to D.G. Eisenhut (NRC), "Supplement 1 to NUREG 0737, SPDS Implementation Plan", dated June 6, 1984.

(3) GPUN Letter, P.B. Fiedler to W.A. Paulson (NRC), "Request for Additional Information for Evaluation of SPDS", dated September 28, 1984.

(4) Safety Evaluation by the Office of NRR relating to the SPDS, dated March 5, 1986.

(5) OCNGS SPDS Certification, Topical Report 065.

TABLE 7.5-1
(Sheet 1 of 3)

REACTOR AND DRYWELL COOLING PANEL 1F/2F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Containment Spray	IND/REC	0-500°F	Indicate and record Water Temperature temperature of water entering and leaving containment.
Isolation Condenser Shell and Steam Inlet Temperatures/EMRV-Relief Valve Discharge Temperatures	IND/REC	0-500°F	
Shutdown Cooling System and Fuel Pool Water Temperature	REC	0-500°F	
Containment Spray System I, II Heat Exchanger, Differential Pressure	IND	0-20 psid	
Drywell Pressure	IND	0-75 psig	Indicates drywell condition and possible accident condition.
Drywell Pressure	IND	0-40 psig	Indicates drywell condition postaccident.
Suppression Chamber Pressure	IND	0-30" W.C.	
Suppression Pool Water Temperature	IND	40-240°F	
Core Spray Booster Pump Discharge Pressure	IND	0-300 psig	Indicates operating condition of core spray system.

* IND - INDICATOR; REC = RECORDER

TABLE 7.5-1
(Sheet 2 of 3)

REACTOR AND DRYWELL COOLING PANEL 1F/2F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Core Spray System Flow	IND	0-5000 gpm	Indicates operating condition of core spray system.
Isolation Condenser Shellside Water Level	IND	0-10 ft	Indicates need for makeup water.
Isolation Condenser Inlet Pressure	IND	0-1500 psig	
Containment Spray Pump Current	IND	0-100 amps	Indicates pump operating condition.
Containment Spray System Flow	IND	0-7500 gpm	Indicates whether sufficient flow exists in system (when operating).
Core Spray Pump Current	IND	0-100 amps	Indicates pump operating condition.
Reactor Head Cooling Flow	IND	0-80 gpm	Indicates whether sufficient flow exists in system.
Shutdown Cooling Pump Discharge Pressure	IND	0-300 psig	Indicates pump operating condition.
Core Spray Booster Pump Current	IND	0-500 amps	Indicates pump operating condition.
Safety Relief Valve Position	IND	Open Closed	Indicates whether valve is open or closed.

*IND = INDICATOR

TABLE 7.5-1
(Sheet 3 of 3)

REACTOR AND DRYWELL COOLING PANEL 1F/2F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
EMRV Position (A to E)	IND	Open Closed	Indicates whether valve is open or closed.
Emergency Service Water Pump Current	IND	0-100 amps	Indicates pump operation
Valve Positions V-5-106 V-17-55 V-17-56 V-17-57	IND	0-100%	Indicates valve position.

* IND = INDICATOR

TABLE 7.5-2
(Sheet 1 of 2)

REACTOR CONTROL PANEL 4F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Control Rod Drive Position	IND	00 to 48**	Used to select rods during power changes.
In-Core Flux Monitor (LPRM)	IND		Used to select rods during power changes.
Rod Worth Minimizer	IND		Used to select rods during startup.
Source Range Monitors Period	IND	-100 to ∞ to +10	
Source Range Monitors CPS	IND, REC	10^{-1} to 10^{6} CPS	
IRM-APRM Neutron Flux	REC	0-40% 0-150% power	Indicates reactor power during operation.
Total Recirculation Flow	IND	0-200,000 gpm	Used to vary reactor power over limited range.
Total Steam Flow	IND	0 to $8x10^{6}$ PPH	
Reactor Pressure	IND	970 to 1070 psig	

* IND = INDICATOR; REC = RECORDER

** 00 indicates normal full-in position.
48 indicates normal full-out position.
Even numbers indicate latched positions.

TABLE 7.5-2
(Sheet 2 of 2)

REACTOR CONTROL PANEL 4F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Control Rod Drive Water Differential Pressure	IND	0-400 psi	Used to determine status of control rod drive.
Reactor Core Heat Flux	IND	0-125%	
Liquid Poison System Parameters (Pump Discharge Pressure/Tank Level)	IND	0-1500 psig 0-5000 gallons	Monitor status of SLCS.
Control Rod Drive Cooling Water Differential Pressure	IND	0-75 psi	Monitor status of control rod drive.
Control Rod Drive Cooling Water Flow	IND	0-50 gpm	Monitor status of control rod drive.
Control Rod Drive Water Flow	IND	0-5 gpm	Monitor status of control rod drive.
Control Rod Drive Flow to Reactor	IND	0-50 gpm	Monitor status of control rod drive.
Control Rod Drive Charging Water Header Pressure	IND	0-2000 psig	Monitor status of control rod drive.
Squib Valves Electrical Continuity	IND	0-100V	Check circuits' electrical continuity.
Control Rod Drive Filter Flow	IND	0-100 gpm	Monitor status of control rod drive

*IND = INDICATOR; REC = RECORDER

TABLE 7.5-3
(Sheet 1 of 3)

FEEDWATER AND CONDENSATE PANEL 5F/6F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Reactor Pressure/Reactor Water Level (Narrow Range)	REC IND	0-1600 psig and 90-186 in. W.C.**	Required for normal plant operation
Reactor Water Level (Yarway)	IND REC	85-185 in.**	Required for normal plant operation and during abnormal conditions.
Reactor Fuel Zone Level	IND REC	(-) 144 to (+) 180 in.**	Required during abnormal conditions.
Reactor Wide Range Level	IND	100 - 700 in.**	Required for all plant conditions
Feedwater Temperature	IND REC	0-400°F	Required for normal plant operation
Reactor Pressure Fuel Zone	IND REC	0-1500 psig	Used in normal operation and during abnormal conditions
Total Feedwater Flow	REC	0-8x10^6 pph	Used in normal operation and during abnormal conditions
Total Steam Flow	REC	0-8x10^6 pph	Used in normal operation and during abnormal conditions

*IND = INDICATOR; REC = RECORDER
**Referenced to Top of Active Fuel, See Figure 7.6-3

TABLE 7.5-3
(Sheet 2 of 3)

FEEDWATER AND CONDENSATE PANEL 5F/6F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Steam Flow	IND	0-4x10^6 pph	Used in normal operation and during abnormal conditions
First Stage Turbine Steam Flow/Reactor Pressure (Narrow Range)	REC	0-8x10^6 pph 970-1170 psig	Required for normal plant operation.
Condenser Vacuum	IND	30-20 in. Hg	Required for normal plant operation.
Feedwater Pump Suction Pressure	IND	0-250 psig	Required for normal plant operation.
Feedwater Pump Discharge Pressure	IND	0-2000 psig	Required for normal plant operation.
Main Condenser Hotwell Level	IND	0-60 in.	Required for normal plant operation and postaccident monitoring.
Service Water Pressure	IND	0-100 psig	Required for normal plant operation.
Feedwater Flow	IND	0-3x10^6 pph	Required for normal plant operation.
Feedwater Pump/Hydrogen Injection Isolation	IND	closed-open	Required for normal plant operation

*IND = INDICATOR

TABLE 7.5-3
(Sheet 3 of 3)

FEEDWATER AND CONDENSATE PANEL 5F/6F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Feedpump Motor Current	IND	0-800 amps	Required for normal plant operation.
Condensate Pump Current	IND	0-200 amps	Required for amps normal plant operation.
Condensate Flow	IND	$0\text{-}8x10^6$	Required for normal plant operation.
Condensate Storage Tank Level	IND	0-45 ft.	Required for normal plant operation and postaccident monitoring.
Demineralized Water Storage Tank Level	IND	0-36 ft.	Required for normal plant operation
Condensate Header Discharge Pressure	IND	0-400 psig	Required for normal plant operation.
Circ. Water Pump Motor Current	IND	0-200 amps	Required for normal plant operation.
Circ. Water Pressure	IND	-30" Hg -0-30 psig	Required for normal plant operation.
Demin. Influent Conductivity	REC	0-.2 umhos 0-20 umhos 0-200 umhos	Required for normal plant operation.
Demin. Effluent Conductivity	REC	0-.2 umhos 0-20 umhos	Required for normal plant operation.
Condensate Pre-Filter Differential Pressure	IND	0-60 psid	Required for normal plant operation.

*IND = INDICATOR; REC = RECORDER

TABLE 7.5-4
(Sheet 1 of 1)

PROCESS RADIATION MONITOR PANEL 10F
DISPLAY INSTRUMENTATION

Displayed Variable	Type Of Display*	Range	Purpose
Reactor Building Closed Cooling Water System Activity	REC	10^{-1}-10^{6} cps	Required for normal plant operation
Steam Line Activity	REC	1-10^{6} mr/hr	Required for normal plant operation
Area Radiation	REC	0.1 to 1000 mr/hr	Required for operation and during abnormal conditions
Area and Vent Radiation	REC	0.1 to 1000 mr/hr	Required for normal plant operation and during abnormal conditions
Offgas Air Ejectors/Stack Gas Activity	REC	10^{0}-10^{6} cps/10-10^{4} cps	Required for normal plant operation and during abnormal conditions
Offgas Sample Flow/Offgas Line Flow	REC	0-1 scfm/0-180 scfm	Required for normal plant operation
Service Water Discharge Activity	REC	10^{1}-10^{6} cpm	Monitors for radioactive releases

* IND = INDICATOR; REC = RECORDER

TABLE 7.5-5
(Sheet 1 of 1)

PROCESS RADIATION PANEL 1R
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Stack Gas Radiation	IND REC	10-10^4 cps (Log) 0-50 cps (Linear)	Indicate radioactivity discharged to plant environs.
		10^{-13}-10^{-6} amps (High Range – Channel 1 only)	
	IND	0-50 cps (Linear)	
Off-gas Radiation	IND	0-10^6 cps	Indicate radioactivity discharged to plant environs.
Radioactive Waste Discharge Radiation	IND	10^{-1}-10^6 cps	Indicate radioactivity discharged to plant environs.
Steam Line Radiation	IND	0-10^6 cps	Provide indication of loss of fuel cladding integrity.
Closed Cooling Water Discharge Radiation	IND	10^{-1}-10^6 cps	Indicate radioactivity discharged to plant environs.

* IND = INDICATOR
* REC = RECORDER

TABLE 7.5-6
(Sheet 1 of 1)

NEUTRON MONITORING PANEL 3R
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
APRM 1, 4	IND	0-150%	Neutron Monitoring
APRM 2, 3	IND	0-150%	Neutron Monitoring
IRM Ch. 11, 12 **	IND	0-40% 0-125%	Neutron Monitoring
IRM Ch. 13, 14 **	IND	0-40% 0-125%	Neutron Monitoring
SRM Ch. 21, 22	IND	10^{-1}-10^{6} CPS	Neutron Monitoring
SRM Ch. 21, 22	IND	-100 to 10 Seconds	Neutron Monitoring
LPRM Calibrator	IND	0-1.0 0-125%	Calibration
LPRM Power Supply And Monitor	IND	0-125	Neutron Monitoring

*IND = INDICATOR
** IRM Meters have a tick mark at 127% for readability

TABLE 7.5-7
(Sheet 1 of 1)

NEUTRON MONITORING PANEL 5R
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
APRM 5,8	IND	0-150%	Neutron Monitoring
APRM 6,7	IND	0-150%	Neutron monitoring
IRM Ch. 15,16 **	IND	0-40% 0-125%	Neutron monitoring
IRM Ch. 17, 18 **	IND	0-40% 0-125%	Neutron Monitoring
SRM Ch. 23, 24	IND	10^{-1}-10^{6} CPS	Neutron Monitoring
SRM Ch. 23,24	IND	-100 to 10 Seconds	Neutron monitoring
LPRM Calibrator	IND	0-1.0 0-125%	Calibration
LPRM Power Supplies And Monitor	IND	0-125	Neutron monitoring

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* IND = INDICATOR
** IRM Meters have a tick mark at 127% for readability

TABLE 7.5-8
(Sheet 1 of 1)

SHUTDOWN AND FUEL POOL COOLING PANEL 10R
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Shutdown System/Isolation Condenser/Main Steam Tunnel Area Monitoring	IND	0-400°F	Area temperature monitoring
Cleanup System Area Temperature	IND	0-400°F	Area temperature monitoring

*IND = INDICATOR

7.6 ALL OTHER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

7.6.1 Description

This section describes the instrumentation and control systems required for safety which are not discussed in other sections. These systems include:

a. Reactor Vessel Instrumentation

b. Recirculation Pump Trip System

c. Leak Detection System

d. Containment Pressure, Water Level and H_2/O_2 Monitoring System

The Process Radiation Monitoring System, also required for safety, is discussed in Section 11.5.

7.6.1.1 Reactor Vessel Instrumentation

Instrumentation is provided to measure the following system parameters:

a. Level

b. Pressure

c. Temperature

d. Flow

e. Core Differential Pressure

f. Core Spray Pipe Break Occurrence

g. Relief Valve/Safety Valve Acoustical Monitoring Systems

7.6.1.1.1 Reactor Water Level Instruments

Reactor water level is measured by differential pressure transmitters. The instruments are compensated for temperature induced variations in density of the reference water column as shown in Table 7.6-1. Administrative procedures are used to effect reference leg compensation on those instruments that are not automatically compensated.

The level instruments at OCNGS are all referenced to the top of active fuel (TAF) which is defined as 353.3 inches above vessel zero. This definition is applicable to fuel bundles with active fuel lengths of 145.24 inches and 144.0 inches. Figure 7.6-3 shows the relationships that exist among the various sensors. Table 7.6-1 is a summary of reactor vessel level instruments.

7.6.1.1.2 Pressure Instruments

Pressure instruments are summarized in Table 7.6-2.

7.6.1.1.3 Temperature Instruments

The design and operation of the reactor pressure vessel (RPV) and internals are based on the logical sequence of maintaining proper heatup and cooldown temperature rates.

The thermal stresses and strains which can result from system operation are limited in order to prevent fatigue or distortion of the RPV. These conditions are controlled by providing limitations on allowed heatup and cooldown rates and limitations on cold inlet water temperatures.

Normal heatup and cooldown rates, cold water introduction rates, and other parameters result in strains far below the RPV design limit. Thermal strains below design limits do not constitute a safety concern. A thermal strain greater than the design limit requires mandatory shutdown. Analysis indicates that the RPV can be subjected to very severe heatup and cooldown rates, well past those allowed by operating procedures, without exceeding the design limit. Therefore, operating procedures are adequate to prevent the maximum vessel strain from approaching this limit.

The plant operating procedures specify limits of:

a. Heatup/cooldown rates of water or steam in the pressure vessel of 100°F/hr.

b. Maximum differential temperature between vessel and head flange of 200°F.

c. Maximum differential temperature between reactor vessel water and idle recirculation loop water for pump start of 50°F.

Several thermocouples are attached to the RPV at different sections to monitor the difference in temperature between the sections. Six thermocouples are provided for monitoring RPV insulation temperatures. See Table 7.6-3 and Figure 7.6-1 for the thermocouple details.

The rate of heating or cooling the RPV is controlled so that the stress set up between two sections of the RPV (computed from the temperature difference between the various points) is held within the allowable limits. The temperatures are recorded on a 0-600°F multipoint recorder in Reactor Building El 51' 3" near Rack Rk03. This recorder is also used to monitor the differential temperature of the RPV Head Metal vs. RPV Head Flange and RPV Vessel vs. RPV Vessel Flange (computed from the temperature difference between the various points).

The thermocouples are copper constantan and, glass insulated and clad with stainless steel. The thermocouples are clamped under pads welded to the RPV.

Also, recirculation loop water temperature detection is provided. Recirculation loop suction temperatures are indicated on two digital indicators on Panel 3F. An interlock is provided to prevent the operation of the Shutdown Cooling System if the water temperature in any one of the five recirculation loops exceeds 350°F.

7.6.1.1.4 Flow Instruments

Flow instruments are summarized in Table 7.6-4. Flow is measured by sensing the differential pressure produced across a flow element installed in the piping. This differential pressure is

proportional to the square of the flow. Differential pressure transmitters measuring flow normally produce a current signal proportional to the differential pressure generated by the flow element. Some flow transmitters perform a square root function internally, thus their output signal is proportional to the flow rate.

7.6.1.1.5 Water Quality Instrumentation

No water quality instrumentation is provided on the RPV. A chemical sample sink is located on RB El. 75' with a secondary station on RB El. 51'. A chemical sample sink is located on El. 51' of the Reactor Building. This sample station provides a line for sampling the recirculation loop and the Reactor Water Cleanup System (RWCUS).

Reactor water quality measurement is provided by three conductivity instruments provided in the RWCUS. A conductivity recorder and alarm are installed on Panel 3F. True reactor water conductivity is indicated by two pre-filter conductivity cells which have a range of 0-10 micromhos.

A third conductivity cell indicates the post-demineralizer water quality and has a range of 0-1 micromhos.

Alarm set points for "hi conductivity" and "hi-hi conductivity" are 1 micromho and 2 micromhos, respectively.

7.6.1.1.6 Core Differential Pressure Transmitter

Core differential pressure is the pressure drop across the core support plate (pressure drop across the active core is small compared to core plate differential). Differential pressure across the core plate utilizes nozzle N12 (see Figure 7.6-2) which has a pipe within a pipe. The outer pipe detects pressure above the core plate (low pressure leg) and the inner pipe, which is also the liquid poison line, detects pressure below the core plate (high pressure leg).

The differential pressure transmitter (PT-IA07) is located on RK04 and is connected to a recorder on panel 3F. The recorder has a range of 0-30 psid.

7.6.1.1.7 Core Spray Differential Instruments

Core spray differential pressure transmitter installation is shown in Figure 7.6-2. Core Spray differential pressure is measured to check the integrity of the core spray piping between the shroud and the vessel wall. It measures the differential pressure between the top of the core support plate and the Core Spray Sparger. This pressure should be independent of recirculation flow and should be essentially zero (pressure drop across the core from interchannel leakage only). Should a significant break develop in the core spray pipe between the shroud and top of the core support plate and the annulus outside the shroud. That differential pressure would be about 8-10 psid greater, due to the pressure drop across the steam separators and dryers.

7.6.1.1.8 Relief Valve/Safety Valve Acoustical Monitoring System (VMS)

The VMS monitors the closed/not closed status of the 9 safety valves and 5 Electromatic Relief Valves. Seven of the nine safety valves have two operational acoustic monitors. The system was supplied by Babcock & Wilcox. Each of the 14 channels consists of a piezoelectric accelerometer

attached to the valve discharge piping, a preamplifier located in the drywell, and signal conditioning electronics located on Panel 15R in the Control Room.

Sound in the frequency range above 2 khz is detected by the accelerometer acting as a microphone. This sound is produced by fluid flowing through the discharge piping which occurs if the valve is open or leaking. The rms (Root-Mean-Square) acceleration value of the acoustic signal is displayed on Panel 15R and the valve analog indicators on Panel 1F/2F. An alarm on Panel 1F/2F is actuated (after a time LAG with an approximate 2.2 second time constant. The time delay is dependent on the magnitude of the signal) indicating that a valve has opened. A loudspeaker and headphones are provided on Panel 15R so that the operator can listen to the sound.

7.6.1.2 Recirculation Pump Trip System

Any one of the following functions automatically trip the Reactor Recirculation Pumps:

a. Drive motor overcurrent

b. Drive motor ground overcurrent

c. Low-low water level in the reactor vessel

d. 4160V bus undervoltage

e Suction valve not fully open

f Main discharge valve not fully closed, discharge bypass valve not fully open and suction valve not fully open (interlocked with pump start)

g Loss of field to MG set generator

h Locked rotor current

i Generator neutral overvoltage

j Incomplete sequencing (initiated by drive motor control switch is a time delay relay. If pump starts within allotted time, the sequencing circuit will be satisfied bypassing motor trip)

k High reactor pressure will trip pumps A, B, and E. Pumps C and D will trip within 12 seconds on a persistent high pressure signal. (Time delay setpoints will provide margin for calibration and accuracy of the time delay relays).

l Pump generator differential current

m Exciter (MG set) overcurrent

n Discharge valve fully closed and discharge bypass valve not fully open

Turning the recirculation pump control switch to the STOP position will trip the generator field breaker and the drive motor breaker.

The low-low water level and high pressure trip signal is produced by the Reactor Protection System (Section 7.2).

7.6.1.3 Leak Detection System

The Leak Detection System is described in Section 5.2.

7.6.1.4 Containment Pressure, Torus Water Level, Drywell Hydrogen/Oxygen and Suppression Pool Temperature Monitoring

7.6.1.4.1 Containment Pressure

Two redundant containment pressure measuring loops, PT-53 and PT-54, are provided, each consisting of a pressure transmitter and recorder. The loop power supply is provided by the recorder. The transmitters are located in the Reactor Building at El. 51'-3". The indicators are located on Panel 1F/2F. The recorders are located on panel 16R in the Control Room. The measurement and indication covers a range of -15 psig to +245 psig.

7.6.1.4.2 Torus Water Level

a. Wide Range - Two redundant Torus water level measuring loops, LT-37 and LT-38, are provided, each consisting of a differential pressure transmitter, digital indicator and recorder. The loop power supply is provided by the recorder. The recorders are located on panel 16R in the Control Room. The digital indicators are located on panel 1F/2F in the Control Room. The transmitters are located in the Reactor Building at El.(-)19'-6".

   The wide range instrument covers a range of 10 to 360 inches from the bottom of the torus.

b. Narrow Range - Two torus water level measuring loops, LT-IP09A and LT-IP09B, are provided. Loop "A" consists of a differential pressure transmitter, power supply, recorder and indicator. Loop "B" consists of a differential pressure transmitter, power supply and indicator.

   Both transmitters are located in the Reactor Building at El.(-)15'-6". Both indicators are located on Panel 11F in the Control Room. The recorder is located on Panel 9XR in the Control Room. "A" loop power supply is located in Panel 10R, and "B" loop power supply is located in Panel 11F, both in the Control Room.

   The narrow range instrument covers a range of 138 to 158 inches from the bottom of the torus.

7.6.1.4.3 Drywell Hydrogen/Oxygen Monitoring

Two redundant drywell hydrogen and oxygen measuring loops are provided, each consisting of a hydrogen and oxygen analyzer, indicator, recorder, system trouble alarm and a control switch with indicating lights for the containment isolation valves. The analyzers are located outside containment at El. 75'-3", azimuth 25 degrees and 55 degrees respectively. The indicators, recorders and control switch are located on panel 16R in the Control Room. The measurement

capability exceeds the range of zero to ten percent hydrogen concentration for both positive and negative ambient pressure conditions. The oxygen measurement range is zero to ten percent.

The Hydrogen/Oxygen Monitoring system is normally in the standby mode except for calibration or maintenance. The system is used in the analyze mode for post accident monitoring of hydrogen and oxygen levels in the drywell.

7.6.1.4.4 Suppression Pool Temperature Monitoring

Two redundant suppression pool temperature monitoring channels are provided, each consisting of six dual element RTD temperature sensors within thermowells extending in through the Torus shell. The temperature sensors are connected to temperature transmitters located in Main Control Room cabinets 18R and 19R. In addition to the temperature transmitters, these cabinets also contain computational modules, alarm modules, signal isolators, and system surveillance test panel. These signal loops compute bulk water temperature via a computer derived algorithm for display and alarm on Main Control Room Panel 1F/2F. The temperature range of indication is 40°F to 240°F. All system alarms are transmitted to the Plant Sequence of Alarm Recorder (SAR) for logging purposes. This system provides the reliable monitoring capabilities necessary to determine the integrity of the Primary Containment and the ECCS Pumps NPSH margin. Also, two isolated outputs of the Suppression Pool Bulk Water Temperature signal (one per safety division) are recorded on the non safety plant computer for parameter trending purposes. The system contains integral diagnostics and test and calibration facilities for in-place surveillance testing.

7.6.1.5 Design Bases

7.6.1.5.1 Reactor Vessel Instrumentation

The reactor vessel instrumentation provides inputs to the Reactor Trip System and the Engineered Safety Features Systems actuation circuits. The analog displays and recordings provide information to the operator, but do not initiate protective action. The design bases for those instruments in the reactor vessel instrumentation system which provide protective action are given in Sections 7.2 and 7.3.

7.6.1.5.2 Recirculation Pump Trip System

Recirculation pump trip is required for ATWS mitigation and to permit the proper operation of the Isolation Condenser System. The Isolation Condensers are normally in standby readiness with all valve control switches in the "AUTO" position. All valves are open with the exception of the outside condensate return valve. Coincident with the low-low reactor water level signal for ICS initiation, is a signal to trip all five reactor recirculation pumps. Coincident with the high reactor pressure signal for ICS initiation, is a signal to trip recirculation pumps A, B, and E. Recirculation pumps C and D will trip within 12 seconds on a persistent high pressure signal. (Time delay setpoints will provide margin for calibration and accuracy of the time delay relays).

Having two recirculation pumps operational following a scram with a temporary pressure spike is desirable as it provides better core cooling with level and temperature indications representative of the core region. In addition, thermal overstress of the CRD stub tubes and incore housing welds will be prevented and there will be reduced thermal cycling of the plant. This trip is provided by the Reactor Protection System (RPS). Refer to Section 7.2 for a discussion of RPS design bases.

The other recirculation pump trips listed in Subsection 7.6.1.2 are for equipment protection and not for safety.

7.6.1.5.3 Leak Detection System

Refer to Section 5.2.

7.6.1.5.4 Containment Pressure, Torus Water Level and Drywell Hydrogen/Oxygen Monitoring

The design basis for this system is establish by NUREG-0578, "TMI-2 Lessons Learned Task Force Report and Short Term Recommendations." The requirement is to provide safety related instrumentation for these four parameters designed to meet LOCA and post-LOCA conditions. This instrumentation performs no protective functions, but provides valuable information in case of an accident. As of 2003, the NRC no longer requires drywell H_2 / O_2 monitoring instrumentation to be safety-related.

7.6.2 Analysis

7.6.2.1 Conformance with Section 4 of IEEE-279

7.6.2.1.1 Reactor Vessel Instrumentation

The protection functions of the reactor vessel instrumentation are performed by switches which are part of the Reactor Protection System. Refer to Sections 7.2 and 7.3 for a discussion of RPS conformance to IEEE 279.

The analog indicating instrumentation is separate from the protection instrumentation except for common impulse lines. As such, it is not subject to the requirements of IEEE 279.

7.6.2.1.2 Recirculation Pump Trip System

The safety related recirculation pump trip is provided by the Reactor Protection System. Refer to Section 7.2 for a discussion of RPS conformance to IEEE 279.

The non safety related recirculation pump trip signals are provided for equipment protection, and are not required to meet the criteria of IEEE 279.

7.6.2.1.3 Leak Detection System

The rate of change of liquid level, drywell temperature and drywell pressure instrumentation was not designed to meet the requirements of IEEE 279.

The Containment Atmosphere Particulate and Gaseous Radioactivity Monitoring System is capable of test and calibration per IEEE 279 with the plant in operation. Cable routing to each isolation valve meets the separation criteria of IEEE 279.
This system does not provide any protection signals. The Leak Detection Systems are described in Section 5.2.5.

7.6.2.1.4 Containment Pressure, Water Level, and Hydrogen/Oxygen Monitoring

The electrical portions of these instruments are Class 1E, and conform to the requirements of Regulatory Guide 1.29, 1.97 and 1.75 and IEEE 279, 323 and 344.

The instruments implement the requirements of NUREG 0578 and provide information mainly during accident and post accident conditions. It also improves the plant's safety response function which will allow operators to make proper decisions.

These instruments do not provide any protection signals.

7.6.2.1.5 Suppression Pool Temperature Monitoring System

The design basis for this system is established by NUREG-0661 and Regulatory Guide 1.97. The requirement is to provide safety related instrumentation to reliably indicate the bulk pool temperature to ensure that the suppression pool is within the allowable limits set forth in the plant Technical Specifications. This instrumentation implements the requirements of NUREG-0661 and Regulatory Guide 1.97 and provides reliable information during accident and postaccident conditions. The electrical portions of these instruments are Class 1E, and conform to the requirements of Regulatory Guide 1.75 and IEEE 279, 323 and 344. These instruments do not provide any protection signals.

7.6.2.2 Conformance to NRC General Design Criteria

The applicable NRC General Design Criteria are listed in Section 7.1 and a discussion of each is provided in Section 3.1.

7.6.2.3 Conformance with Regulatory Guides

Regulatory Guide 1.29

This regulatory guide concerns seismic design classification and Section C-1.k describes requirements for Seismic Category I systems or portions of systems required for monitoring of systems important to safety.

A discussion relating to environmental qualification is contained in Section 3.10 of the FSAR or the Regulatory Guide.

Regulatory Guide 1.45

This regulatory guide concerns reactor coolant pressure boundary leakage detection systems. Acceptable detection methods for unidentified leakages are described as well as identifiable sources of leakage. Detector sensitivity is outlined as well as response time.

The Leak Detection System is arranged so that all identified leakage drains to the Drywell Equipment Drain Tank and all unidentified leakage drains to the Drywell drain sump. The alarms on the Drywell drain sump are set at the normal identified leakage plus 80% of the Technical Specification limit.

Regulatory Guide 1.75

This regulatory guide deals with physical independence of Class 1E electric systems so as to perform their safety related functions in the event of outage of redundant circuits assuring single failure criteria. The applicable sections are generally covered in Sections A and B.

Redundant instruments are located to minimize the possibility of a single event affecting more than one instrument. The airborne particulate radioactivity monitor in the Leak Detection System is not redundant.

Regulatory Guide 1.97

This regulatory guide deals with instrumentation to assess plant and environs conditions during and following an accident and to provide Control Room personnel information to take preplanned action to shut the plant down safely.

The reactor water level (fuel zone and Yarway), containment pressure, torus water level, temperature, and drywell hydrogen/oxygen monitoring system meets the design provisions of Regulatory Guide 1.97 including redundancy, qualification and testability. In addition, the Suppression Pool Temperature Monitoring System meets the design requirements in Appendix A to NUREG-0661 as well as Regulatory Guide 1.97.

The applicable sections of this guide are 1.3.1, Design and Qualification Criteria - Category I and Table I, Type A, B, C, D and E Variables.

TABLE 7.6-1
(Sheet 1 of 2)

SUMMARY OF REACTOR VESSEL WATER LEVEL INSTRUMENTS

Qty	SENSOR Description	ID No.	INDICATION Local	INDICATION Remote	Recorder	Actuation or Control Functions	Density Compensation
2	"Fuel Zone" Level Indication 1 Sensor Per Channel (Channels A and B)	DPT-5-IA0091A DPT-6-IA0091B	No	Yes	Yes	Automatic On and Off Features - No Control Functions	Density compensated using reactor pressure. Instrument reference leg temp. in drywell and Reactor Building are also used for compensation.
4	"Low-Low" Level Transmitters	RE-02A	No	Yes	No	1. Core Spray Init. 2. Reactor Isol	Level is calibrated for operating conditions (1,000 psia saturated steam & water) and a fixed reference leg temperature (301^{o}F) Ref. Calc C-1302-622-5350-056, Rev. 0
		RE-02B	No	Yes	No	3. Containment Isol 4. Recirc. Pump Trip	
		RE-02C	No	Yes	No	5. Isol. Cond. Init. 6. SGTS Init.	
		RE-02D	No	Yes	No	7. Annunciators 8. Isolates Cleanup System 9. Isolates Shutdown System 10. Isolates RBCCW to Drywell 11. Isolates Air/N_1 to Drywell 12. Alternate Rod Injection (RE-02A&C)	
4	"Low" Level Transmitters	RE-05A; RE-05/19A;RE-05B; RE-05/19B	No	Yes		1. Low Level Scram 2. High Level Turbine Trip & Target Relay	Level is calibrated for operating conditions (1000 psia saturated steam & water) and a fixed reference leg temperature (301^{o}F) Ref. Calc C-1302-622-5350-056, Rev. 0.
			No	Yes		3. Annunciators (Low Level)	
			No	Yes		4. Reactor Overfill Protection (High Level)	
			No	Yes			
2	Remote Shutdown "Fuel Zone" Level Indication 1 sensor per channel (Channels C and 10)	DPT-622-1009 DPT-622-1011	No	Yes*	No	1. Alternate Rod Injection Trip (High Pressure from transmitters used for density compensation)	Density compensated using reactor pressure. Reactor Building temp. is also used for compensation. Reference leg density inside the DW is based on an assumption of 200^{o} F (Ref. SDD OC-615B, R.4, Para. 1.6.2.2).

*At remote Shutdown Panel and Control Room

TABLE 7.6-1
(Sheet 2 of 2)

SUMMARY OF REACTOR VESSEL WATER LEVEL INSTRUMENTS

Qty	SENSOR Description	ID No.	INDICATION Local	INDICATION Remote	Recorder	Actuations or Control Functions	Density Compensation
4	Low-Low-Low Level Indicating Switches	RE-18A	Yes	No	No	1.Auto Depressurization System Initiation	Calibrated for Cold Conditions (Ref. Calc C-1302-622-5350-036, R.1)
		RE-18B	Yes	No	No	2. Annunciators	
		RE-18C	Yes	No	No	3. Isolates RBCCW from Drywell	
		RE-18D	Yes	No	No		
2	Narrow Range Level	ID-13A	No	Yes	Selected A or B or C	1.Feedwater Control	Level is Density Comp throughout Range. Reference leg assumes a fixed density value at 160° F (Ref. Calc C-1302-622-5350-036, R.1)
		ID-13B	No	Yes		2. Annunciator (Hi/Low Level)	
		ID-13C	No	Yes			
1	Wide Range	ID-12	No	Yes	No	NONE	No Compensation Accurate at Cold Conditions Only.

Notes:

1. Variable legs of sensors listed above sense level in downcomer region (annulus) except the triple low sensors which sense above core region at core spray sparger and Fuel Zone Level Instrument.

2. Reference leg condensate pots tap off of upper downcomer region for all sensors above except wide range level which taps into top of upper head.

TABLE 7.6-2
(Sheet 1 of 1)

PRESSURE TRANSMITTERS

ID No.	Location	Description	Range	Power Supply	Readout		Controls/ Trips	Alarms
					Meter	Recorder		
ID46A	RX Bldg 23' El. N	Rx Pressure	0-1600	Continuous	5F	5F *	Note 1	
ID46B	RX Bldg 23' El. N	Rx Pressure	0-1600	Instr. Panel #3 (CIP3)	5F	5F *	Note 1	
ID45	RK02	Rx Pressure+	970-1070		4F	5F/6F	Note 2	1030 PSIG OFF RECORDER PR-FR-ID0077
IG04A	Local	ISOL Cond 1	0-1500	Panel IP-4	1F/2F	None	Note 3	
IG04B	Local	ISOL Cond 2	0-1500	CIP-3	1F/2F	None	Note 3	
IA52A	RK03	A RCP #1 Seal	0-1200	Panel IP-4B	3F	None	Note 4	
IA52B	RK03	B RCP #1 Seal	0-1200	Panel IP-4B	3F	None	Note 4	
IA52C	RK03	C RCP #1 Seal	0-1200	Panel IP-4B	3F	None	Note 4	
IA52D	RK03	D RCP #1 Seal	0-1200	Panel IP-4B	3F	None	Note 4	
IA52E	RK03	E RCP #1 Seal	0-1200	Panel IP-4B	3F	None	Note 4	
IA67A	RK03	A RCP #2 Seal	0-1200	Panel IP-4B	3F	None	Note 5	
IA67B	RK03	B RCP #2 Seal	0-1200	Panel IP-4B	3F	None	Note 5	
IA67C	RK03	C RCP #2 Seal	0-1200	Panel IP-4B	3F	None	Note 5	
IA67D	RK03	D RCP #2 Seal	0-1200	Panel IP-4B	3F	None	Note 5	
IA67E	RK03	E RCP #2 Seal	0-1200	Panel IP-4B	3F	None	Note 5	
IJ01	RK05	Cleanup System	0-1250		3F	None		

* Selected of these two signals
\+ Referenced for pressure control
NOTE 1 Used to compensate reactor level and steam flow for density.
NOTE 2 Referenced by Operators for controlling reactor pressure.
NOTE 3 Indicates reactor pressure if isolation condensers are not fully isolated.
NOTE 4 Normally indicates reactor pressure if seals are not damaged.
NOTE 5 Reads interseal pressure, normally 500 psig while operating. Will indicate reactor pressure if No. 1 seal fails.

TABLE 7.6-3
(Sheet 1 of 2)

REACTOR VESSEL THERMOCOUPLE LOCATIONS
THERMOCOUPLE PAD LOCATIONS
(REF. GE DWG. 885D731)

T.E. No.	Location	Azimuth	Elevation	
FA54 (IA01-1)	Vessel Flange	70^o	54 ft. 7 in.	Note 1
FC54 (IA01-2)	Vessel Flange	230^o	54 ft. 7 in.	Note 3
FD54 (IA01-3)	Vessel Flange	354^o	54 ft. 7 in.	Note **3**
VA50 (IA01-4)	Vessel Seam	70^o	50 ft. 8 in.	Note **3**
VC50 (IA01-5)	Vessel Seam	230°	50 ft. 8 in.	Note 3
VD50 (IA01-6)	Vessel Seam	354^o	50 ft. 8 in.	Note 1
VA37 (IA01-7)	Vessel Below Low Water	70^o	37 ft. 6 in.	Note 1
VC37 (IA01-8)	Vessel Below Low Water	230^o	37 ft. 6 in.	Note 3
VD37 (IA01-9)	Vessel Below Low Water	354^o	37 ft. 6 in.	Note 3
VA23 (IA01-14)	Vessel Core	90^o	23 ft. 6 in.	Note 1
VC23 (IA01-15)	Vessel Core	230^o	23 ft. 6 in.	Note 4
VD23 (IA01-16)	Vessel Core	354^o	23 ft. 6 in.	Note 4
VA12 (IA01-17)	Vessel Downcomer	90^o	12 ft. 4 in.	Note 1
VC12 (IA01-18)	Vessel Downcomer	230^o	12 ft. 4 in.	Note 3
VD12 (IA01-19)	Vessel Downcomer	354^o	12 ft. 4 in.	Note 4
BA04 (IA01-20)	Vessel Bott. Above Skirt Unit	90^o	4 ft. 5 in.	Note 3
BC04 (IA01-21)	Vessel Bott. Above Skirt Unit	230^o	4 ft. 5 in.	Note 4
BD04 (IA01-22)	Vessel Bott. Above Skirt Unit	354^o	4 ft. 5 in.	Note 1
BA03 (IA01-23)	Vessel Bott. Below Skirt Unit	90^o	3 ft. 1 1/4 in.	Note 3
BC03 (IA01-24)	Vessel Bott. Below Skirt Unit	230^o	3 ft. 1 1/2 in.	Note 3
BD03 (IA01-25)	Vessel Bott. Below Skirt Unit	354^o	3 ft. 1 1/2 in.	Note 3
SA02 (IA01-26)	Skirt Insulation	90^o	2 ft. 7 1/2 in.	Note 1
SC02 (IA01-27)	Skirt Insulation	230^o	2 ft. 7 1/2 in.	Note 4
SD02 (IA01-28)	Skirt Insulation	354^o	2 ft. 7 1/2 in.	Note 4
SA00 (IA01-29)	Skirt Insulation	90^o	0 ft. 4 in.	Note 1
SC00 (IA01-30)	Skirt Insulation	230°	0 ft. 4 in.	Note 3
SD00 (IA01-31)	Skirt Insulation	354^o	0 ft. 4 in.	Note 4
BA01 (IA01-32)	Vessel Bottom	85^o	1 ft. 0 in.	Note 4
BC01 (IA01-33)	Vessel Bottom	232^o	1 ft. 1 1/2 in.	Note 1
BD01 (IA01-34)	Vessel Bottom	355^o	1 ft. 0 in.	Note 4

TABLE 7.6-3
(Sheet 2 of 2)

REACTOR VESSEL THERMOCOUPLE LOCATIONS
THERMOCOUPLE PAD LOCATIONS
(REF. GE DWG. 885D731)

T.E. No.	Location	Azimuth	Elevation	
LC50 (IA01-35)	Lagging (Insulation) TCs		230° 50 ft. 8 in.	Note 1
LD50 (IA01-36)	Lagging (Insulation) TCs		354° 50 ft. 8 in.	Note 3
LC37 (IA01-37)	Lagging (Insulation) TCs		230° 37 ft. 6 in.	Note 4
LD37 (IA01-38)	Lagging (Insulation) TCs		354° 37 ft. 6 in.	Note 1
LC12 (IA01-39)	Lagging (Insulation) TCs		230° 12 ft. 4 in.	Note 4
LD12 (IA01-40)	Lagging (Insulation) TCs		354° 12 ft. 4 in.	Note 3
FA56 (IA01-47)	Vessel Head Flange	70°	56 ft. 1 in.	Note 3
FC56 (IA01-48)	Vessel Head Flange	230°	56 ft. 1 in.	Note 1
FD56 (IA01-49)	Vessel Head Flange	354°	56 ft. 1 in.	Note 3
HA58 (IA01-50)	Vessel Head	70°	58 ft.	Note 3
HC58 (IA01-51)	Vessel Head	230°	58 ft.	Note 3
HD58 (IA01-52)	Vessel Head	354°	58 ft.	Note 1

Notes:

1. Provides input to Recorder TR-IA02 in Reactor Building EL 51' 3"
2. Deleted
3. Spare or abandoned in place.
4. Abandoned in Place

TABLE 7.6-4
(Sheet 1 of 1)

FLOW TRANSMITTERS

ID No.	Location	Description	Range – GPM	Power Supply	Readout	Controls/ Trips	Alarms
FTIA60A	RK04	Recirc. Loop Flow	0-40,000	FY-622-29A to Instr.	Note 1	Note 1	None
B	RK04	Recirc. Loop Flow	0-40,000	FY-622-29A to Instr.	Note 1	Note 1	None
C	RK04	Recirc. Loop Flow	0-40,000	FY-622-30A to Instr.	Note 1	Note 1	None
D	RK04	Recirc. Loop Flow	0-40,000	FY-622-30A to Instr.	Note 1	Note 1	None
E	RK04	Recirc. Loop Flow	0-40,000	FY-622-31A to Instr.	Note 1	Note 1	None
A1	RK04	Recirc. Loop Flow	0-40,000	FY-622-29B to Instr.	Note 1	Note 1	None
B1	RK04	Recirc. Loop Flow	0-40,000	FY-622-29B to Instr.	Note 1	Note 1	None
C1	RK04	Recirc. Loop Flow	0-40,000	FY-622-30B to Instr.	Note 1	Note 1	None
D1	RK04	Recirc. Loop Flow	0-40,000	FY-622-30B to Instr.	Note 1	Note 1	None
E1	RK04	Recirc. Loop Flow	0-40,000	FY-622-31B to Instr.	Note 1	Note 1	None
DPTIA50A	RK03	RCP DP	0-50 psi	IA82A to Instr.	3F	Note 2	Low DP 5 psid
B	RK03	RCP DP	0-50 psi	IA82A to Instr.	3F	Note 2	Low DP 5 psid
C	RK03	RCP DP	0-50 psi	IA82A to Instr.	3F	Note 2	Low DP 5 psid
D	RK03	RCP DP	0-50 psi	IA82A to Instr.	3F	Note 2	Low DP 5 psid
E	RK03	RCP DP	0-50 psi	IA82A to Instr.	3F	Note 2	Low DP 5 psid
ID33A	23 elev.	RX Steam Flow	$0\text{-}4x10^6$ #/hr	Note 5	Note 3	Note 4	Note 4
ID33B	23 elev.	RX Steam Flow	$0\text{-}4x10^6$ #/hr	Note 5	Note 3	Note 4	Note 4

NOTE 1 There are 2 Flow Transmitters for each recirculation loop. Each transmitter performs a square root function internally to produce a signal proportional to the flow rate. The individual recirculation flows are summed and used for APRM flow biasing and indication to the operator. Each loop has individual readouts on Panel 3F. The total recirculation flow is provided on 3F (recorder) and 4F (meter).

NOTE 2 Provides input to recirculation pump starting logic. Will cause incomplete sequence trip of pump if sufficient DP (10 psid) is not obtained during pump start sequence.

NOTE 3 Individual steam line meters on Panel 5F. Total steam flow recorder on Panel 5F.

NOTE 4
a. Part of main steam flow mismatch alarm (7% error for 10 seconds).
b. Input to Rod Worth Minimizer (Reactor power signal).
c. Input to Feedwater Control

NOTE 5 Powered by Feedwater Control System Transmitter Power Supplies

7.7 CONTROL SYSTEMS

The system discussed in this section include those control systems used for normal operation that are not relied upon to perform safety functions following anticipated operational occurrences or accidents, but which control plant process having a significant impact on plant safety. These control systems for OCNGS are as follows:

a. The Reactor Manual Control System, including the rod withdrawal block and refueling interlocks

b. The Recirculation Flow Control System

c. The Rod Worth Minimizer

d. The Reactor Level and Feedwater Control System

e. The Turbine Generator Control System

f. The Reactor Overfill Protection System (ROPS)

7.7.1 Description

7.7.1.1 Reactor Manual Control System

The Reactor Manual Control System consists of the circuits, switches, and relays that control and program the electrical power to the solenoid operated valves in the Control Rod Drive Hydraulic System. Control rod drive operation is initiated and monitored in the Control Room with switches and devices mounted on Panel 4F.

The sequence of operation of the individual solenoids of the direction control valves in the hydraulic control unit for the individual drive to be moved is determined by the operation of relays in the Reactor Manual Control System. Proper relay sequencing for control rod drive movement is governed by a multicircuit automatic sequence timer. The timer and the relays controlling rod selection are installed in the rod selection relay rack (ER-1) located in the old Cable Spreading Room.

The principal features provided by this system include:

a. Only one control rod may be positioned at a time.

b. Control rod positioning is independent of the protective functions of the Reactor Protection System. These protective functions are described in Section 7.2.

c. A single control switch movement is required to notch a rod out or to insert a rod any number of notches.

d. Two switches must be held to continuously withdraw a drive.

e. The movement of rods is controlled by a time switch. This requires proper speed adjustment of the rod drive.

f. Control rod movement inhibit signals are provided when unsafe or potentially unsafe conditions exist.

7.7.1.1.1 Control Rod Positioning

Control rods are selected for movement by use of the rod select pushbutton matrix (4S2) on Panel 4F. Each control rod is represented by a pushbutton switch which is momentarily depressed to select a rod for movement. The interruption monitor circuit (shown on Drawing GE237E912) in the Manual Reactor Control System ensures that the previous rod selection is cancelled. In this manner, only one rod may be selected and moved at a time. Each selected pushbutton illuminates, and remains illuminated until another selection is made, or until power is turned off through the rod-power-switch (4S4).

The mode and direction of movement of the selected control rod is made by momentarily positioning the rod control switch (4S1) to either "ROD IN" or "ROD OUT-NOTCH". When switch 4S1 is momentarily moved to "ROD IN", the contacts of the timer in the control system will automatically program single notch drive-in and settle control rod drive movement. When switch 4S1 is momentarily moved to "ROD OUT-NOTCH" the timer will automatically program unlatch, single notch drive-out, and settle control rod drive movement. Following either of these sequences, the timer will automatically reset in preparation for another command. (Drawing GE237E912).

Continuous drive-in movement is obtained by switch 4S1 in the "ROD IN" position. The control rod will drive in until switch 4S1 is released and returned to the "OFF" position after which the timer will automatically program the control rod drive settle function, then reset. To move a control rod in the continuous out mode, two distinct operations must take place: (1) the notch override switch (4S3) must be positioned and held on "NOTCH OVERRIDE" and (2) switch 4S1 is simultaneously positioned and held on "ROD OUT-NOTCH" (Drawing GE237E912). The control rod will continuously drive out until either switch 4S1 or switch 4S3 is released and spring returned to the "OFF" position, following which the timer will automatically program the control rod drive settle function, then reset. This dual switch provision precludes the possibility of the reactor operator inadvertently initiating continuous withdrawal of a selected control rod when notch out withdrawal is intended. Normally, the continuous-in or continuous-out modes are used only in timing rod inserting and withdrawal speeds for test purposes and during startup and shutdown, as specified in reactor operation procedures. (Drawing GE237E912)

A provision in the Manual Reactor Control System permits control rod insertion in the event of a malfunction of the timer. This is accomplished by positioning switch 4S3 to "EMERGENCY ROD IN", which bypasses the timer and its associated relays and applies power directly to the drive-in circuit until switch 4S3 is spring-returned to "OFF".

The (4S4) on Panel 4F is used to remove power from the rod select and positioning circuits when the system is not required for rod movement.

7.7.1.1.2 Control Rod Position Indication

Each control rod drive is provided with a position indicator probe fitted with glass enclosed read switches. These switches are closed by the magnet installed in the drive piston during control rod drive movement to provide readouts in the form of units and ten digits on the individual rod position displays in the Control Room. Even numbered readouts, from "00" to "48" are provided

at each latched drive position and odd numbered readouts ("01" to "47") at the midpoints between latched positions.

The 137 rod position displays are installed on the vertical portion of Panel 4F. In addition to providing a digital readout of control rod position, each display bears its numerical control rod identification and a red light, which is illuminated when the inlet and outlet scram valves (305-126, 205-127) serving the particular control rod are open. These displays serve as a ready means for determining that all 137 sets of scram valves are open upon initiation of a reactor trip. Rod selection relay action energizes and illuminates the control rod identification on the rod position display at any time that the particular control rod is selected. The illumination of the identification occurs simultaneously with the illumination of the particular rod select pushbutton switch.

Figure 7.7-5 shows the numbers of the switches in the drive position indicator probe, the installed position of each switch relative to the fully inserted drive position, the display provided in the Control Room by each switch, and the drive position corresponding with that indication.

The 53 reed switches in each control rod drive position indicator probe energize the various Control Room indications and annunciations through relay cards and rod position relays installed in rod position relay panel ER-2. This panel is located in the old Cable Spreading Room between the Reactor and Turbine Building. The relays serve as connectors between the 137 control rod position indicator probes and the various rod position indicator probes, and the various rod position monitors and alarm instruments. There are 49 reed switches in an individual probe to provide the rod position indication numerical display. One switch is provided to energize the red backlighting on the rod position display simultaneously with the digital readout of "48". Two switches are used to provide for green backlighting wherever the rod is fully inserted, one switch operates together with the "00" digital display, and the other provides for green backlighting should the drive overtravel beyond full-in. This green backlighting signal is also used to provide a signal for the one-rod-free-movement permissive interlock during refueling. One additional switch is installed in the probe to provide for a rod overtravel alarm should the drive travel two inches below the normal fully withdrawn position.

7.7.1.1.3 Rod Position Alarms

a. Rod Drift Alarm (5F "H")

The odd numbered switches (S01 through S47) in each drive position indicator probe, which transmit the odd numbered readouts ("01" to "47") to the rod position indicator on Panel 4F, are also utilized to provide the rod drift indication and annunciation on Panel 5F; the contacts of the rod position relays associated with these switches are connected in series with the contacts of the rod drift alarm relay. The contacts for a specific rod are bypassed, however, by contact action of the specific rod selection relay for that rod, so that the rod drift circuit for a particular rod is bypassed at any time that a particular control rod is selected. This bypass remains in effect as long as the particular rod is selected. The drift alarm circuit, therefore, is energized any time that an unselected rod drifts to an odd numbered position. The drift alarm circuit, therefore, is energized any time that an unselected rod drifts to an odd numbered position. When energized, the circuit actuates the "ROD DRIFT" annunciator on Control Room Panel 5F. The rod drift alarm relays also provide a rod drift signal to the Rod Worth Minimizer. Each relay module on rack ER-2 is provided with a neon indicating lamp that

energizes at odd numbered rod positions to assist in locating which rod is drifting should a rod drift annunciation occur.

b. Rod Overtravel Alarm 5F/6F (H-5-A)

Switch S50 in each control rod drive position indicator probe is installed two inches below the normal fully withdrawn position, and is closed only when the control rod and drive are uncoupled. When closed, switch S50 energizes the rod overtravel alarm relay which actuates the "ROD OVERTRAVEL" annunciator on Panel 5F.

c. CRD High Temperature 5F/6F (H-5-C)

The thermocouple leads of each of the 137 control rod drive position indicator probes terminate at a data acquisition system (DAS) located in control room panel 8R. The DAS monitors and displays all 137 points. Should the operating temperature of one or more control rod drives rise to 250°F, a set of contacts on the DAS change state energizing the “CRD HIGH TEMPERATURE” annunciator on panel 5F.

d. Control Rod Drive Accumulator Low Pressure/High Level 5F/6F (H-8-C)

Should the nitrogen pressure in any Control Rod Drive accumulator drop to below 1025 ±15 psig, or if 37 cc of water leak past the accumulator piston, this alarm will sound.

7.7.1.1.4 <u>Rod Blocks</u>

Control rod movement is blocked by interlock action in the Reactor Manual Control System when any one of several potentially unsafe conditions exist relative to the reactor or associated systems or components. These interlocks function primarily to prevent control rod withdrawal, however, the interlocks in the Rod Worth Minimizer (RWM) system can prevent rod selection*, notch-in rod insertion, and emergency rod insertion, as well as rod withdrawal, if certain preconditions are not met.

* Only if the RWM fails - not a programmed block

An elementary diagram of the reactor manual control system interlock circuitry appears on Drawing GE237E912. All contacts shown in this circuitry must be closed as a precondition to control rod withdrawal. When one or more of these contacts are open, the "ROD BLOCK" annunciator on Panel 5F is energized, and the particular interlock(s) in effect when such an annunciation occurs is (are) displayed by the illumination of one or more indications on the rod block display, located on the lower right side of the vertical portion of Panel 4F. Figure 7.7-7 shows the rod block display and the interlocks indicated.

Rod blocks are summarized in Table 7.7-1. Rod withdraw blocks are accomplished by interrupting power to the rod out relay 4K1 (Drawing GE237E912). Rod deselect blocks are accomplished by removing power to the rod selection circuit during "rod out", if a trip signal is in. The circuit is shown on Figure 7.7-8.

7.7.1.2 Recirculation Flow Control

Reactor power can be controlled over an approximate 50 percent range through recirculation flow control. The lower limit of flow control is set by the lowest pump speed attainable (about 20% of rated flow) and the natural recirculation flow established by the existing power level of the reactor. A reactor power change is accomplished by utilizing the large void negative power coefficient characteristic of the core design. To increase reactor power, recirculation flow is increased, which reduces the void accumulation in the core, thus increasing reactivity. As reactor power increases, a new power level is established where the transient excess reactivity is balanced by increased void formation. Conversely, when recirculation flow is reduced a power reduction is achieved.

The rotational fluid speed of the reactor recirculation pumps is varied to change the recirculation flow. Motor-Generator sets with adjustable speed couplings vary the frequency of the power supply to the pump motors to give the desired pump speed. To change reactor power, the operator adjusts the master speed controller set point. The speed set point is transmitted to the individual control logic for each generator. The measured generator speed signal for each generator is transmitted to the individual control logic. When an individual controller is in automatic, a proportional and integral action is applied to the difference between speed set point and measured speed. The resulting automatic signal is added to the speed setpoint (feed-forward signal) to generate a demand signal to increase or decrease generator speed. The recirculating pump motor adjusts its speed in accordance with the frequency of the generator output voltage. Manual speed control of each generator is available to the operator. High and low limits as well as rate limits are applied to the signal from the master controller as well as to the signals from the individual control logic for each generator. An independent, diverse generator speed rate limit is applied by the pneumatic positioners which change the torque transmitted from the M-G set motor to the M-G set generator.

The control system is designed to be fault tolerant. Loss of the primary digital control computer will result in bumpless transfer to the backup digital control computer. Loss of both computers results in bumpless transfer to the diverse controllers on the control room front panels. If the signal from the control system to any of the pneumatic positioners is lost the positioner locks up. When the signal to the positioner returns the lock up may be reset by the operator thereby allowing the control system to control the positioner again.

Control rod withdrawal from a low recirculation flow, low power operating state could result in a reduction of the minimum critical power ratio (MCPR) below established limits before a high flux

scram would occur. Consequently, a backup electrical interlock system will prevent rod withdrawal outside the acceptable power and flow range.

7.7.1.3 Rod Worth Minimizer (RWM)

The design basis of the Rod Worth Minimizer is that it should serve as a backup to procedural control in limiting control rod worths so that, in the event of a control rod drop from the reactor core at a more rapid rate than can be achieved by the use of the control rod drive mechanism, the reactivity addition rate would not lead to damage of the Reactor Coolant System nor to significant fuel damage. For the design basis, a fully inserted control rod is assumed to drop out of the core after becoming disconnected from its drive and after the drive has been removed to the fully withdrawn position, this is known as the Control Rod Drop Accident. There are two devices that are designed to limit the rate at which reactivity can be inserted in a Control Rod Drop Accident. One is the rod velocity limiter that limits the maximum free fall velocity of a control rod. The other is the Rod Worth Minimizer that is used as a backup to procedural controls to limit the maximum worth of each control rod. The Rod Worth Minimizer is used up to 10% of rated power. Because of the effect of steam voids at higher power, the Control Rod Drop Accident ceases at power levels above 10%. For this reason, the initial RWM design only blocked rod movement below 10%. Above this Low Power Set Point (LPSP), the RWM would alarm only to warn the operator that his rod pattern does not comply with procedures. At a slightly higher power level of 20%, the RWM ceases to alarm. This second power level is referred to as the Lower Power Alarm Point (LPAP). On the redesigned RWM introduced in Cycle 12R, the LPSP and LPAP setpoints are adjustable, and may be set at 20% and 30% power, respectively.

Good operating procedures are the primary defense against high worth control rod patterns. Normal plant operation results in control rod patterns which have sufficiently low individual rod worths. The RWM is not intended to replace operator selection of control patterns, but is intended simply to monitor and reinforce good procedures. In performing this function, it causes minimum interference with desired plant operation.

One of the principal reasons that the RWM is not required above 10% power is the presence of coolant voids. A rod drop reactivity addition would be less severe when a significant amount of voids are present. The voids would increase during the transient, and greatly limit the power excursion.

The RWM System consists of the following components interconnected as shown on the block diagram of Figure 7.7-9:

a. Relay Input Buffer

b. Patch Panel

c. Redundant Data Acquisition System

d. Control Room CRT and Touch Screen

e. Redundant Digital Computers (PDP 11/84)

f. Keylock Switch

g. Video Generator

The RWM system continuously monitors control rod positions, compares the operator selected rod movements and positions against a predetermined rod pattern, and prevents rod movements that are not in accordance with this pattern.

The desired control rod movements are stored in the computer memory together with the actual rod positions. The pre-established control rod pattern is entered into the computer by means of keyboard entry with editor; the actual rod position data is received from the control rod position indicating system and interfaced via plugs to the "patch panel." Rod selection and rod drive motion are evaluated by the computer with reference to permissible and existing control rod patterns. If rod operation is in accordance with the selected withdrawal sequence, the RWM output is permissive. If the operator attempts a rod selection or movement that deviates significantly from the selected program, the RWM either alarms or blocks such action.

Normal plant operation is at power levels that range from 10 percent of rated power at the low end to 100 percent of rated power. Within this range, rod pattern control is not required to maintain the reactor operating parameters. The low-power end of the normal operating range is established as the low power set point. To allow maximum operator flexibility over the normal operating range, an indication of steam flow is provided as an input to the RWM. When the steam flow indicates that the low power setpoint has been exceeded, the RWM does not inhibit rod selection or movement.

The Rod Worth Minimizer (RWM) supplements the operator procedural controls. Because most plant operation occurs above the low power setpoint, the RWM is effectively out of service a majority of the time. Below the low power set point it provides a backup service to the operating controls to maintain individual control rod worths at the desired low values.

The data acquisition system is housed in a computer cabinet in the lower cable spreading room (LCSR). The relay input buffer is housed in the rod position relay rack ER2. The Control Room display CRT and touch screen is mounted on the vertical section of panel 4F on the left side. A keylock bypass switch enables the operator to bypass the RWM and regain control of rod selection and movement in the event of an RWM failure or as plant operation requires. The bypass switch is located on the reactor control panel (Panel 4F) to the left of the control rod selection switches.

a. Relay Input Buffer

The Relay Input Buffer interfaces plant data and internal system controls to the Patch Panel and thereby to the Data Acquisition System (DAS). The DAS implements all RWM input and output (I/O) to and from the plant. In normal operation, the Relay Input Buffer monitors the position of each rod as it is selected. During a core scan, the Relay Input Buffer sequentially applies the rod position and select data for each rod to the DAS. The relay input buffer consists of a nodal relay assembly for each rod, and a scan control unit. The scan control unit is comprised of a stepping switch that energizes one nodal assembly at a time on command, and the relay logic that accepts the scan commands from the computer and systematically applies them to the stepping switch. When a rod is manually selected, the corresponding nodal assembly is energized by the rod select relays in rack ER1.

b. Patch Panel

The Patch Panel serves as an interface point between the Plant and the new Rod Worth Minimizer DAS. Originally, because it employs plug-in connectors, it allowed for the rapid transition from the old to the new Rod Worth Minimizer (RWM) when that transition took place. The Patch Panel contains the signals required by the old and the new RWMs.

c. Redundant Data Acquisition System (DAS)

The redundant DAS performs the input and output (I/O) functions between the plant and the RWM. The input signals provide Rod data and status information to the RWM. The outputs perform the rod block or permissive functions. Scan demands and any plant enabling functions are also generated here.

d. Control Room Cathode Ray Tube (CRT) Monitor and Touch Screen

This CRT is the primary point of operator interface and allows control of the RWM via a touch screen display on the CRT. An assortment of displays to observe core rod configuration and sequence of data may be presented on the CRT by manipulating the touch screen. System visual alarms are also presented on the CRT screen.

e. Redundant Digital Computers

The actual calculations and implementation of various algorithms takes place in the Site Emergency Building (SEB), computer room. This redundant pair of PDP 11/84 computers is linked to the Data Acquisition System (DAS). The sequence monitoring and alarming functions are implemented here.

The archiving and associated magnetic media are here. Communications to the Plant Computer System are also situated here. Terminals and printers to support RWM computer system maintenance and control as well as Sequence Editing are supported here.

Throwover to the backup system is manually accomplished at the SEB computer room location.

f. Video Generator

The video generator is driven by the Digital Computer and generates the signals that cause the various displays to appear on the Control Room CRT. This is temporarily located in the Control Room, but will eventually be installed in the PCG Multiplexer Termination cabinet in the Multiplexer Room.

g. Power Sources

The RWM receives 120 volts, 6O Hz, from the Plant Computer System Uninterruptable Power Source (UPS). This source is routed to the Digital, Computer, the DAS, the Video Generator (VG) and the CRT when the CRT and VG are finally mounted in panel 4F.

The indicator lamps on the display panel and output buffer are powered by a 60 volt dc power supply located in the output buffer.

The dc power for energizing the relays in the input and output buffers is provided by a peripheral +28 volts dc supply digital computer cabinet.

7.7.1.4 Reactor Water Level and Feedwater Control System

The Feedwater Control System maintains the proper water level in the reactor vessel. Reactor vessel water level too high results in water carryover to the steam turbine and turbine blade erosion. Reactor vessel water level too low uncovers the bottom of the separators, resulting in the introduction of steam to the downcomer region and thus possible cavitation of the recirculation pump.

Reactor Water Level Control During Normal Operation

Measured reactor water level, steam flow, and feedwater flow signals are transmitted to the control system. A "mass inventory" signal is developed by subtracting the feedwater flow signal from the steam flow signal. An "effective power" signal is developed by the control logic based on steam flow. The mass inventory and effective power signals are used to anticipate level changes and improve transient response. The mass inventory and effective power signals are not used at low power due to the inaccuracy of the steam flow measurements at low flows. Steam flow is maintained relatively constant by a separate control system (turbine control system) and as feedwater flow increases reactor water level will increase. The feedwater control system processes the signals and adjusts the regulating valves in the feedwater system to maintain desired reactor water level.

A low flow regulating valve (LFRV) dedicated controller is provided for each LFRV A and C. At low power, when controlling on an LFRV, the operator adjusts the respective LFRV controller automatic level setpoint to set the desired reactor water level. A proportional, integral, and derivative action is applied to the difference between level setpoint and measured level to form a demand signal. The demand signal is transmitted to the respective LFRV. The operator can bumplessly select manual control on the LFRV controller as required. Transfer from an LFRV to the respective main feedwater regulating valve (MFRV) and vice versa is performed manually.

A master level controller (MLC) is provided for automatic or manual reactor water level control when controlling on the MFRV A, B, and/or C. At high power, when controlling the MFRV(s), the operator adjusts the MLC automatic level setpoint to set the desired reactor water level. A proportional and integral action is applied to the difference between level setpoint and measured level minus the mass inventory signal. The resulting automatic signal is added to the effective power signal to form a demand signal. The demand signal is transmitted to the MFRV controllers which have been selected to "auto".

The MFRV controllers which are in "auto" pass the signal to the respective MFRV. The operator can bumplessly select manual control on the MFRV controller as required. The operator can bumplessly select ganged manual control of MFRV(s) whose controllers are in "auto" by selecting "manual" on the MLC.

Reactor Water Level Control Post Scram

An abnormally low water level condition followed by a level swell is expected as a result of post reactor trip void collapse and subsequent reactor depressurization. The post scram level control logic, initiated on a validated full scram, takes appropriate control action to avoid the expected high water level excursion following a reactor trip. The post scram level control logic applies to automatic main feed regulating valve control only.

Feedwater Pump Runout Protection Flow Control

Automatic feedwater pump runout protection is provided when controlling on the main feed regulating valves to prevent pump motor overload and subsequent pump trip during high feedwater flow operating conditions. Runout protection also protects the condensate demineralizers from excess flow. If the measured flow in any of the three feedwater lines exceeds the runout protection flow setpoint for that line, demand to the affected regulating valve(s) is transferred from reactor water level control logic to runout protection flow control logic which maintains feedwater flow at the runout protection flow setpoint. The flow demand from runout flow control logic is selected back to level control logic when level control flow demand becomes less than runout protection control logic flow demand.

Reliability Considerations

The control system is designed to be fault tolerant. Loss of the primary digital control computer will result in bumpless transfer to the backup digital control computer. Loss of both computers results in transfer to the diverse controllers on the control room front panels. In the unlikely event of transfer to the front panel controllers a bump in level is expected. Also, the front panel controllers do not have the following: mass inventory control, post scram level control, feedwater pump runout protection and automatic lock up of main feed regulator valves on loss of electrical signal.

If the signal from the control system to a main feedwater regulating valve is lost the valve locks up. When the signal to the valve returns the lock up logic will automatically reset allowing the control system to control the valve again.

7.7.1.5 Turbine Generator Controls

The Turbine Generator is provided with a complete control system for startup, shutdown, and changes in load. This system is discussed in Section 10.2. This section discusses only the Turbine-Generator interaction with the Reactor Protection System.

If the water level in the reactor vessel exceeds the height of the top of the steam separators, excessive moisture carryover occurs, resulting in turbine blade erosion. Level sensors RE 05/19A, RE05B (RPS Channel 1) RE05A and RE05/19B (RPS Channel 2) trip the turbine on high level to protect against blade damage. Note that these are safety related switches which also provide the reactor low water level trip.

If the turbine trips while operating at power levels above approximately 200 MW the reactor will be tripped.

Load rejection within the Turbine Bypass System (Section 10.4) capacity will cause the control valves to close and the bypass valves to open and dump steam to condenser. The design mismatch of five percent rated flow under these conditions should not be enough to cause a high flux scram. Load rejections beyond the bypass system capacity will cause a high flux

scram, but the bypass system will normally limit the pressure rise to keep the safety valves from opening.

Three separate turbine trip sensors, listed in Table 7.7-2, anticipate the reactor power increase and start rod motion (scram) before the pressure excursion begins in order to minimize the flux peak. Although the resultant increase in reactor power would result in a trip from the pressure increase directly, the anticipatory trip feature lessons the reactor power and pressure excursion. The anticipatory trips are bypassed when the Reactor Mode Switch is not in RUN and the reactor pressure is less than 600 psig, to permit reactor startup. The Turbine Trip and Generator Trip anticipate the need for a reactor trip if a turbine trip or generator trip occurs over 40% reactor thermal power. Below this power level, a scram is not required since the bypass system is capable of passing this flow rate.

7.7.1.6 Reactor Overfill Protection System (ROPS)

If the water level in the reactor vessel exceeds the height of the main steam lines, a potential for Main Steam Line Break (MSLB) exists. The Reactor Overfill Protection System (ROPS) is designed to minimize the potential for such conditions. Existing level sensors (RE05A, RE05/19A, RE05B, RE05/19B) used in Reactor Protection System (RPS) are utilized in ROPS to trip all three (3) feedwater pumps on reactor high level provided the total feedwater flow is not low and the "normal/bypass" switch located in control room panel 4F is not in bypass position.

The "Normal/Bypass" selector switch allows system testing. The low feedwater flow interlock automatically bypasses ROPS when feedwater flow is already reduced by automatic or operator action. This flow interlock is provided by the total feedwater flow loop FT-422-0001. The ROPS initiating logic is arranged in a one out of two taken twice configuration to ensure plant availability, and is designed to prevent inadvertent actuation due to a malfunction of one level sensing line. The one-out-of-two (1/2) taken twice logic arrangement is such that a common mode failure of the level sensors associated with a level sensing line due to any malfunction of that line such as improper line isolation or line break will not initiate the ROPS. In that event, the two out of two (2/2) level sensors in the failed division may actuate but at least one more level sensor from the redundant division will be required for the ROPS initiation on high water level. Alarms are provided at the control room panel 5F/6F for ROPS Actuate A, ROPS Actuate B and ROPS Bypassed. The ROPS is a backup to the Reactor Level and Feedwater Control System discussed in Section 7.7.1.4 and is totally independent from the Feedwater Control System as required by USNRC Generic Letter GL 89-19.

TABLE 7.7-1
(Sheet 1 of 3)

CONTROL ROD BLOCK INTERLOCKS

Display	Trip Device(s)	Interlock Description
SCRAM DUMP VOLUME	CRD hydraulic system level switch RD86 & RD90	Blocks rod withdrawal if water level in scram discharge volumes exceeds 30 inches during normal reactor operation; ensures adequate discharge volume in event of reactor scram.
ACCUMULATOR LEVEL/PRESS	Hydraulic control unit pressure switches 305-130, level detectors 305-129; control system relays 4K5-	Blocks rod withdrawal if: a. low gas pressure exists in any two scram accumulators, b. water exists on the gas side of any two scram accumulators series and 4K7; or c. one of either condition exits simultaneously on any two accumulators.
REFUEL INTERLOCK	Refueling platform position switches K2, K3, K5; control system relay 4K17	Blocks rod withdrawal in all modes except RUN when refueling platform hoist or fuel grapple is fuel-loaded and over the reactor, or if the service platform hoist is fuel-loaded.
SRM DETECTOR POSITION	Source range monitor trip auxiliary units; NMS	Blocks rod withdrawal if a. any source range monitor channel is not in "Bypass" b. the downscale level trip is not cleared c. the associated detector is not full-in, and d. the reactor mode switch (1S1) is not in the "RUN" position.
IRM DETECTOR POSITION	Intermediate range monitor trip auxiliary unit; NMS	Blocks rod withdrawal if any IRM channel is not in "Bypass" the associated detector is not full-in and the reactor mode switch (1S1) is not in the "RUN" position.

TABLE 7.7-1
(Sheet 2 of 3)

CONTROL ROD BLOCK INTERLOCKS

Display	Trip Device(s)	Interlock Description
APRM HIGH FLUX	Average power range monitor trip auxiliary unit; NMS	Blocks rod withdrawal if any APRM channel is not in "Bypass" and there is an APRM upscale rod block trip.
TIMER MALFUNCTION	Control system TD relay 4TD1	Blocks rod withdrawal if automatic timer in control system fails during a rod-out sequence.
SRM HI COUNT	Source range monitor trip	Blocks rod withdrawal when any SRM channel is not in "Bypass" and there is an SRM upscale rod block trip.
IRM DOWNSCALE	Intermediate range monitor trip auxiliary units; NMS	Blocks rod withdrawal when any IRM range switch is not in the 125 x 10^{-5}% power position, not in "Bypass", and a downscale (rod block) trip exists.
APRM DOWNSCALE	Average power range monitor	Blocks rod withdrawal monitor if APRM is not in "Bypass" an APRM downscale trip is present, and the reactor mode switch (1S1) is in the "RUN" position.
IRM/APRM	Average power range monitor and intermediate range monitor trip auxiliary units; NMS	Blocks rod withdrawal if any APRM is not in "Bypass", and APRM downscale trip is present, the reactor mode switch (1S1) is not in the "RUN" position, and IRM is not in "Bypass", and there is an upscale rod block trip.
SRM INOP	Source range monitor trip auxiliary units; NMS	Blocks rod withdrawal when any SRM that is not "Bypass" is inoperative.

TABLE 7.7-1
(Sheet 3 of 3)

CONTROL ROD BLOCK INTERLOCKS

Display	Trip Device(s)	Interlock Description
IRM INOP	Intermediate range monitor	Blocks rod withdrawal when trip auxiliary units; NMS any IRM that is not in "Bypass" is inoperative.
APRM INOP OR FLOW BIAS	Recirculation Flow Monitoring Electronics, average power range monitor trip auxiliary units; NMS	Blocks rod withdrawal if a. an upscale, inoperative or comparator mismatch trip exists on recirculation flow monitoring systems, or b. any APRM not in "Bypass" is inoperative.
RWM	Rod Worth Minimizer relays K1, K2, K3, K4, K5, K6, K7, K8, K9	Blocks rod insertion, and/or withdrawal if programmed pre-conditions are not met.
None	Key Locked Switch 4S15	Administrative block used after an APRM status change to provide protection against a withdrawal error.

TABLE 7.7-2
(Sheet 1 of 1)

ANTICIPATORY REACTOR TRIPS
FOLLOWING TURBINE TRIP

Sensor	Sensor Number	Location	Reactor Protection System Channel
Low Condenser Vacuum	RSCS 11	Cam operated limit switches part of	I
	RSCS 21	vacuum trip units on turbine front	I
	RSCS 12	standard	II
	RSCS 22		II
Turbine Trip	SVS1A	Limit switches* on turbine stop valves	I
	SVS2A		I
	SVS3A		I
	SVS4A		I
	SVS1		II
	SVS2		II
	SVS3		II
	SVS4		II
Load reject	PSLA	Pressure switches on the	I
	PSLC	emergency trip oil line in the	I
	PSLB	acceleration relay in turbine front	II
	PSLD	standard	II

* One valve closed - no action
Two valves closed - uncertain
Three or four valves closed - reactor trip if greater than 40% power.