CHAPTER 7

INSTRUMENTATION AND CONTROL SYSTEMS

7.1 INTRODUCTION

Chapter 7 presents specific detailed design and performance information for instrumentation and control of safety-related and major plant control systems. The design and performance considerations of these systems, their safety functions, and their mechanical aspects are described in other chapters. See Section 1.2 for plant layout drawings.

7.1.1 Identification of Safety-Related Systems

The systems discussed in Chapter 7 are categorized as reactor protection (trip) system (RPS), engineered safety feature (ESF) systems, safe shutdown systems, safety-related display instrumentation (SRDI), other systems required for safety, and control systems not required for safety. Table 7.1-1 lists safety-related systems and identifies the designer and/or the supplier. Table 7.1-2 identifies plant instrumentation and control systems that are similar or identical to those of nuclear power plants of similar design that have recently received Nuclear Regulatory Commission (NRC) design or operation approval through the issuance of either a construction permit or an operating license. The comparisons shown in Table 7.1-2 were considered valid at the time the operating license was issued.

Unit 2 safety-related systems are designed to conform with the requirements of IEEE Standard 279-1971. The inherent redundancy designed in the control of the reactor is such that all trip functions are redundant at the trip system level, trip function level, or both.

> Example 1: Redundant instrument channels, each causing the closure of a valve, are redundant at the trip system level.
>
> Example 2: Two completely independent trip functions, each causing the closure of one isolation valve (one inboard, the other outboard), are redundant at the trip function level.

The following is an identification of RPS, ESF systems, safe shutdown systems, SRDI, and other systems required for safety. Detailed descriptions of the controls and instrumentation for these systems are provided in Sections 7.2 through 7.6.

7.1.1.1 Reactor Protection (Trip) System

Instrumentation and controls initiate reactor shutdown via automatic control rod insertion (scram) if selected variables exceed preestablished limits. This action prevents fuel damage, limits nuclear system pressure, and restricts the release of radioactive material.

7.1.1.2 Engineered Safety Features Systems

The following is a list of ESF systems and a cross-reference to the sections where design and performance characteristics are discussed:

	Section
Emergency core cooling systems (ECCS, HPCS, ADS, LPCS, and LPCI)	6.3, 7.3.1.1.1
Primary containment and reactor vessel isolation control system (PCRVICS)	7.3.1.1.2
RHR/containment spray cooling mode (RCSCM)	7.3.1.1.3
RHR/suppression pool cooling mode (RSPCM)	7.3.1.1.4
Standby gas treatment system (SGTS)	6.5.1
Combustible gas control system (CGCS)	6.2.5
Reactor building heating, ventilating, and air conditioning (HVAC) system	9.4.2
Service water system (SWP)	9.2.1
Service water pump bays ventilation system	9.4.7.2.2
Control building heating, ventilating, and air conditioning (HVAC) system	9.4.1
Control building chilled water system	9.4.10
Standby power system	9.5.4-9.5.8
Diesel generator building, heating,	

ventilating, and air conditioning (HVAC) system	9.4.6

7.1.1.3 Systems Required For Safe Shutdown

The following is a list of systems required for safe shutdown and a cross-reference to the sections where the design and performance characteristics are discussed:

	Section
Reactor core isolation cooling (RCIC) system	7.4.1.1
Standby liquid control system (SLCS)	7.4.1.2
RHR reactor shutdown cooling mode (RSCM)	7.4.1.3
Remote shutdown system (RSS)	7.4.1.4

7.1.1.4 Safety-Related Display Instrumentation

The SRDI is provided to inform the Reactor Operator when a manual safety action should be taken or is required and allows assessment of safety system status.

7.1.1.5 All Other Instrumentation Systems Required for Safety

Following is a list of instrumentation systems required for safety and a cross-reference to sections where design and performance characteristics are discussed:

	Section
Process radiation monitoring system (PRMS)	7.6.1.1
High-pressure/low-pressure interlocks	7.6.1.2
Leak detection system (LDS)	7.6.1.3
Neutron monitoring system (NMS)	7.6.1.4
Recirculation pump trip (RPT)	7.6.1.5
Safety relief valves (SRVs) - relief function	7.6.1.6

7.1.2 Identification of Safety Criteria

The following sections discuss design bases that are generally applicable to safety systems identified in this chapter. These generally applicable design bases include instrument error considerations and requirements of 10CFR50 Appendix A, regulatory guides, and industry codes and standards. Table 7.1-3 provides a listing of regulatory requirements and IEEE standards, and the safety systems to which they pertain. Sections 7.2 through 7.6 provide discussions of specific design bases for each safety-related system.

7.1.2.1 Conformance to 10CFR50 Appendix A

A discussion of conformance to the general design criteria of 10CFR50 Appendix A is provided in Section 3.1.

7.1.2.2 Conformance to IEEE Standards

The following conformance discussions apply generically to the safety-related systems. Refer to the analysis portions of Sections 7.2 through 7.6 for conformance discussions for IEEE-279, IEEE-338, and IEEE-379 as they apply specifically to individual systems. Refer to Section 8.3 for a discussion of conformance to IEEE-308, IEEE-317, and IEEE-336. Refer to Sections 3.10 and 3.11 for a discussion of conformance to IEEE-323 and IEEE-344.

IEEE-384-1974 The safety-related systems described in Sections 7.2 through 7.6 meet the independence and separation criteria for redundant systems in accordance with IEEE-279, Paragraph 4.6.

The electrical power supply, instrumentation, and control wiring for redundant safety-related circuits are physically separated to preserve redundancy and ensure that no single credible event will prevent completion of the protective function. Credible events include, but are not limited to, the effects of short circuits, pipe rupture, pipe whip, high-pressure jets, missiles, fire, and earthquake, and are considered in the basic plant design.

The independence of tubing, piping, and control devices for safety-related controls and instrumentation is achieved by physical space or barriers between separation groups of the same protective function.

The criteria and bases for the independence of safety-related instrumentation and controls, electrical equipment, cable, cable routing, marking and cable derating are discussed in Sections 8.3.1.3 and 8.3.1.4. Fire detection and protection in the areas where wiring is installed are described in Section 9.5.1.

IEEE-387-1977 (Divisions I and II) and IEEE-387-1972 (Division III HPCS) Design and qualification testing of the standby power system used to furnish electrical power to safety loads conforms to IEEE-387 to ensure that system requirements for redundancy, single-failure criteria, adequate capacity, capability, and reliability are adequately met. The standby power source as an integrated system component satisfies the requirements of IEEE-308 as discussed in Section 8.3.

7.1.2.3 Conformance to Regulatory Guides

The degree of compliance with all applicable regulatory guides is provided in Section 1.8. The following paragraphs provide either a reference to or additional comments on the method of conformance to the regulatory guides as generally applicable to the safety-related systems. Refer to the analysis portions of Sections 7.2 through 7.6 for conformance discussions applying specifically to individual systems. Table 7.1-4 lists examples of regulatory guides applicable to plant systems.

Regulatory Guide 1.6 Refer to Section 8.3.1.

Regulatory Guide 1.7 Refer to Section 6.2.5.5.

Regulatory Guide 1.11 Refer to Section 6.2.4.

Regulatory Guide 1.22 Refer to the analysis portion of Sections 7.2, 7.3, 7.4, and 7.6 for applicable systems.

Regulatory Guide 1.29 All safety-related instrumentation and control equipment is classified as Category I, designed to withstand the effects of the safe shutdown earthquake (SSE) and remain functional during normal and accident conditions. Qualification and documentation procedures used for Category I equipment and systems are identified in Section 3.10 and Table 3.2-1.

Regulatory Guide 1.30 Refer to Section 3.11 and Chapter 17.

Regulatory Guide 1.40 There are no continuous duty motors installed inside the containment that are part of the instrumentation and control system.

Regulatory Guide 1.45 Refer to Section 5.2.5.

Regulatory Guide 1.47 The Nine Mile Point Nuclear Station - Unit 2 (Unit 2) design incorporates a bypass/inoperability design which incorporates both a component and system implementation of Regulatory Guide (RG) 1.47. This design ensures that should a component be disabled, thereby causing an inoperable system, this system inoperability will be alarmed and indicated to the Control Room Operator. The component inoperability is also indicated to the Control Room Operator. Each safety-related system described in Sections 7.2, 7.3, 7.4, and 7.6 has an automatically- or operator-initiated system level bypass or inoperability annunciator. In addition to system level annunciation, component level indicators are provided in the main control room near the system controls to indicate the cause of the system bypass or inoperability. A switch is provided for manual actuation of each system level annunciator to allow display of those bypassed or inoperable conditions that are not automatically indicated.

Automatic bypassed and inoperable status indication is provided for all systems that affect plant safety. This indication system accompanies any operational procedure for all safety-related systems.

Any deliberate action which makes a support system inoperable and also inhibits the proper function of a dependent safety system will also be automatically indicated.

Equipment within the supporting/safety-related systems that is Class 1E is annunciated (component level). In this group, the equipment that, when bypassed, causes a system to be defeated will contribute to its respective system level annunciator and indicate a bypassed system condition. If a system is declared inoperable, whether it be a redundant portion of a system or a total system, and also supports other systems, the bypassed and inoperable indication will cascade into the dependent systems.

The design of the bypassed and inoperable status indication system complies with Branch Technical Position (BTP) ICSB-21, as discussed below:

a. The component level bypass indicators are arranged within their respective system confines on the control panels. The system level annunciator is located immediately above the component level indicator matrix within the standard plant annunciator system. All operability annunciation is amber in color.

b. The design of the bypassed and inoperable status indication system will indicate which particular system and/or systems are affected whenever a component or function is rendered inoperable. This is provided by the integrated system/cascaded system approach.

c. At no time can the Operator cancel an automatic bypass condition that has been annunciated. Only the manual inoperability function, which is provided for equipment and/or functions that do not have an automatic indication feature, can be either activated or deactivated.

d. The design intent of the bypassed and inoperable status indication system is to aid the Operator in determining the operability of a particular system. The indication system does not perform any functions that may impede a protective system's performance.

e. All of the equipment used in the bypassed and inoperable status indication system is Category I. Also, no adverse effect on the interdependence between safety systems exists.

f. The bypassed and inoperable status indication system has push-to-test indicators for component levels and the manually inoperable feature for system level annunciator illumination.

Following is a list of systems and their respective bypass inoperable statuses.

System Mnemonic	Systems Containing Bypassed/Inoperable Requirements

ADS	Automatic Depressurization System (24 V dc System)
BYS	Battery System
CCP	Reactor Building Closed Loop Cooling Water (Component Level Only)
CMS	Containment Atmosphere Monitoring
CPS	Primary Containment Purge (Component Level Only)
CSH	High-Pressure Core Spray - Power Supply
CSL	Low-Pressure Core Spray
DER	Reactor Building Equipment Drains (Component Level Only)
EGA	Standby Diesel Generator Air Startup
EGF	Standby Diesel Generator Fuel
EGP	Standby Diesel Generator Protection (Breaker)
EGS	Standby Diesel Generator Protection (Generator)
EJS	Standby Station Service Substation
ENS	Standby Station Service Supply Breakers
FPW	Fire Protection - Water (Component Level Only)
FWS	Feedwater System (Component Level Only)
GTS	Standby Gas Treatment
HCS	Hydrogen Recombiner
HVC	Control Building Air Conditioning
HVK	Control Building Chilled Water
HVP	Standby Diesel Generator Building Ventilation
HVR	Reactor Building Ventilation (Component Level Only)

HVY	Yard Structure Ventilation (Component Level Only)
IAS	Instrument Air (Component Level Only)
MSS	Main Steam (Component Level Only)
RHS	Residual Heat Removal
SFC	Spent Fuel Pool Cooling and Cleanup
SWP	Service Water
WCS	Reactor Water Cleanup (Component Level Only)

It is not a requirement of RG 1.47 that the bypassed and inoperable status indication system be Class 1E. However, the control circuit wiring is associated with the Class 1E components and is designed in accordance with Class 1E requirements. Optical isolation will be employed to separate the annunciator, which is a non-Class 1E circuit, from the bypassed and inoperable logic circuits.

The component level inoperability will be displayed using master specialty switch-light units within their respective operating area on the control panel.

The system level indicators used are the standard plant annunciator windows separated into divisions if applicable. At the system level, the indicators will be accompanied by an audible alarm. All bypassed and inoperable status indication has unique amber illumination.

Typically, the following bypass or inoperable conditions cause actuation of system level and/or component level annunciation for the affected system:

1. Pump motor breaker in racked out position.

2. Loss of pump motor control power.

3. Loss of motor-operated valve (MOV) control power/motive power when the valve is in the nonsafety position.

4. Logic power failure.

5. Logic in test.

6. Bypass or test switches actuated.

7. Pump control switch in pull-to-lock position.

Auxiliary supporting system inoperability or bypass resulting in the loss of other safety-related systems will cause actuation of system level annunciators for the auxiliary supporting system as well as those safety-related systems affected.

Reviews and testing provide assurance that divisional separation is maintained in design. Format of procedures, which requires removal of safety systems or components from service, requires the prior demonstration of operability of redundant components or systems. Additionally, procedures utilized for safety-related components or systems require the approval of the on-duty Senior Reactor Operator (SRO) prior to the performance of the procedure. Verification of operability of components or systems is completed prior to this approval.

<u>Regulatory Guide 1.53</u> Refer to the analysis portions of Sections 7.2, 7.3, 7.4, 7.5, and 7.6 for applicable system.

<u>Regulatory Guide 1.62</u> Refer to the analysis portions of Sections 7.2, 7.3, 7.4, 7.5, and 7.6 for applicable system.

<u>Regulatory Guide 1.63</u> Refer to Section 8.3.1.1.5.

<u>Regulatory Guide 1.68</u> Refer to Section 14.2.

<u>Regulatory Guide 1.73</u> Refer to Sections 3.10 and 3.11.

<u>Regulatory Guide 1.75</u> A complete description of physical and electrical separation criteria is contained in Section 8.3. The extent of implementation for the requirements of RG 1.75 is as follows:

1. Isolators or physical separation are provided without affecting building or control/relay room arrangements.

2. Physical separation between divisions of essential systems and between essential systems and essential circuits is maintained for all essential nuclear steam supply systems (NSSS) except the NMS, the RPS, the PRM system, and the control rod drive (CRD) hydraulic system.

3. Design criteria for fire protection is discussed in Section 9.5.1.

Regulatory Guide 1.89 Refer to Sections 3.10 and 3.11.

Regulatory Guide 1.97 Refer to Section 1.10.

Regulatory Guide 1.100 Refer to Section 3.10.

Regulatory Guide 1.105 The trip setpoint (instrument setpoint) and allowable value (Technical Specification limit) are contained in Technical Specifications. These parameters are all appropriately separated from each other and their selection is based on instrument accuracy, calibration capability, and design drift (estimated) allowance data. The setpoints are within the instrument accuracy range. The established setpoints provide margin to satisfy both safety requirements and plant availability objectives.

Unit 2 provided a detailed technical assessment of the current Technical Specification setpoints in relation to NEDC-31336, General Electric Setpoint Methodology (Unit 2 letter NMP2L 1459 dated December 28, 1993). The results indicated that the existing NMP2 protection system setpoints are equal or conservative when compared to the setpoints calculated using the methods delineated in NEDC-31336. Therefore, the existing Technical Specification setpoints ensure that the reactor core and reactor coolant system are prevented from exceeding the licensing safety limits for the transients and accidents analyzed. Accordingly, no Technical Specification changes were required.

Regulatory Guide 1.118 This guide endorses/modifies IEEE-338-1975. Discussion of IEEE-338 is presented on a system-by-system basis in the analysis portions of Sections 7.2, 7.3, 7.4, and 7.6 with the following clarification of the regulatory guide requirements.

Position C.6b Trip of an associated protective channel or actuation of an associated Class 1E load group is required on removal of fuses or opening of a breaker only for the purpose of deactivating instrumentation and control circuits.

Periodic surveillance testing procedures for the RPS have been developed. Lifting of leads or the use of jumpers is not normally utilized in procedure steps, unless the procedure would require this action to accomplish the purpose of the procedure and no other method, such as opening a breaker, is possible. A list of cases where it is required to lift leads for at-power testing has been provided to the NRC. Whenever lifted leads or the use of jumpers are utilized, specific instructions are given to identify, lift, and replace the leads or jumpers. Such actions are performed under strict administrative control with independent verification for safety-related systems or components.

Additionally, Unit 2 commits to implementing the recommendation of IE Information Notice 84-37 for all safety systems, as discussed below:

1. The Unit 2 Administrative Procedures governing the use of lifted leads and temporary jumpers during testing are kept in the control room and are utilized by personnel performing and authorizing testing.

2. Controls included in the procedures provide for a sign-off by the test personnel and an independent verification of system/component status before returning the system/component to service.

3. Any instrumentation channel undergoing testing need not be placed in the trip condition immediately but is declared inoperable and subject to the Limiting Condition for Operations (LCO) criteria of the Technical Specifications.

4. Any instrumentation channel declared inoperable has been considered in the Technical Specifications regarding the minimum number of operable channels analyses (for trip functions), and that the system with a channel under test and declared inoperable meets the single failure criteria.

Unit 2 will endeavor to perform actions to functionally check safety-related equipment after lifting leads (to accommodate surveillance or maintenance) whenever practical.

7.1.2.4 Instrument Errors

The determination of setpoints requires that during the design of safety-related systems, instrument drift, setability, and repeatability be considered when selecting instruments and controls.

Adequate margin between safety limits and instrument setpoints is provided to allow for instrument error. The appropriate allowable values are listed in Technical Specifications. The amount of instrument error is determined by test and experience. The setpoint is selected based on these known errors. The surveillance frequency is increased on instrumentation that demonstrates a tendency to drift.

TABLE 7.1-1

DESIGN AND SUPPLY RESPONSIBILITY
OF SAFETY-RELATED SYSTEMS

	GE Design	GE Supply	SWEC Design	Others Supply
Reactor Protection System				
Reactor protection (trip) system (RPS)	X	X		
Engineered Safety Feature Systems				
Emergency core cooling systems (ECCS) High-pressure core spray (HPCS) Automatic depressurization system (ADS) Low-pressure core spray system (LPCS) RHR low-pressure coolant injection (LPCI)	X	X		
Primary containment and reactor vessel isolation control system (PCRVICS)	X	X	X	X
Process radiation monitoring system (PRM) (portion used for PCRVICS)	X	X		
RHR containment spray cooling mode (RCSCM)	X	X		
RHR suppression pool cooling mode (RSPCM)	X	X		
Standby gas treatment system (SGTS)			X	X
Combustible gas control system (CGCS)			X	X
Reactor building HVAC system			X	X
Service water (SW) system			X	X

TABLE 7.1-1 (Cont'd.)

	GE Design	GE Supply	SWEC Design	Others Supply
SW pump bays ventilation system			X	X
Control building HVAC system			X	X
Control building chilled water system			X	X
Standby power system	X	X	X	X
Diesel generator building HVAC system			X	X
Systems Required for Safe Shutdown				
Reactor core isolation cooling system (RCIC)	X	X		
Standby liquid control system (SLCS)	X	X		
RHR reactor shutdown cooling mode (RSCM)	X	X		
Remote shutdown system (RSS)			X	X
Safety-Related Display Instrumentation	X	X	X	X
All Other Safety Systems				
Process radiation monitoring system (BOP)			X	X
High-pressure/low-pressure interlocks	X	X		
Leak detection system (LDS) (portion in ESF)	X	X	X	X

TABLE 7.1-1 (Cont'd.)

	GE Design	GE Supply	SWEC Design	Others Supply
Neutron monitoring systems Intermediate range monitor (IRM) Average power range monitor (APRM) Local power range monitor (LPRM) Oscillation power range monitor (OPRM)	X	X		
Recirculation pump trip (RPT)	X	X		
Safety relief valves (SRV) relief function	X	X		
Spent fuel pool cooling and cleanup (SFC) system			X	X
Redundant reactivity control system (RRCS)	X	X		

TABLE 7.1-2

SAFETY-RELATED SYSTEMS
SIMILARITY TO LICENSED REACTORS

Instrumentation and Controls System	Plants Applying For or Having Construction Permit or Operating License	Similarity of Design
Reactor protection system	Zimmer-1	Identical[1,3,8]
Primary containment and reactor vessel isolation control system	Zimmer-1	Identical[4,8]
Emergency core cooling systems	Zimmer-1	Identical[1,5,8]
Neutron monitoring system	LaSalle	Identical[1]
Process radiation monitoring system (main steam lines)	Zimmer-1	Identical[1]
Reactor core isolation cooling system	Zimmer-1	Identical[1,6,8]
Standby liquid control system	Zimmer-1	Identical[1,9]
Leak detection system	Zimmer-1	[1,2,8]
RHR/reactor shutdown cooling mode	Zimmer-1	Identical[1,8]
Safety-related display instrumentation	Zimmer-1	Identical[1]
RHR/containment spray cooling mode	Zimmer-1	Identical[1,7,8]
Recirculation pump trip	Zimmer-1	Identical[1,8]
RHR/suppression pool cooling mode	Zimmer-1	Identical[1]
High pressure/low pressure Interlocks	Zimmer-1	Identical[1,8]

TABLE 7.1-2 (Cont'd.)

Instrumentation and Controls System	Plants Applying For or Having Construction Permit or Operating License	Similarity of Design
Redundant reactivity control system (RRCS)	Perry, Zimmer-1	Identical[1,9]
Standby gas treatment system	None	--
Combustible gas control system	None	--
Reactor building HVAC system	None	--
Service water system	None	--
Service water pump bays ventilation system	None	--
Control building HVAC system	None	--
Control building chilled water system	None	--
Standby power system	None	--
Diesel generator building HVAC system	None	--
Remote shutdown system	None	--
Spent fuel pool cooling and cleanup system	None	--

(1) System function and hardware are identical; size or capacity might differ.

(2) Unit 2 uses ambient temperature sensors in turbine building. Zimmer does not. Due to different sump arrangements, the sump monitoring subsystems of the two plants are not identical.

(3) Unit 2 uses both an uninterruptible power supply (UPS) and a motor generator (MG) set for the RPS power supply. Zimmer uses only the MG set. In addition, the Unit 2

TABLE 7.1-2 (Cont'd.)

design uses improved instrumentation in the scram discharge volume high water level scram circuitry.

(4) The Zimmer nuclear steam shutoff supply system power supply is from the RPS MG set. Unit 2 uses the RPS UPS.

(5) Zimmer uses two-out-of-two logic for the HPCS high water level trip. Unit 2 uses one-out-of-two-twice logic. The low pressure permissive used at Unit 2 for the LPCI injection valve is a differential pressure interlock across the valve for both the automatic and manual opening of the injection valve. Zimmer uses a differential pressure interlock for only the manual mode and a direct reactor pressure interlock for the automatic mode.

(6) Zimmer uses two-out-of-two logic for the RCIC high water level trip. Unit 2 uses one-out-of-two-twice logic. Leak detection isolation logic and RCIC pump suction transfer logic are also different between the two plants.

(7) For Zimmer, a LOCA signal and a high drywell pressure signal must always be present before valves F016 and F017 are permitted to open. For Unit 2, valves F016 and F017 can be opened without the existence of a LOCA signal and a high drywell pressure signal only if valve F027 is opened first.

(8) The Unit 2 design provides improved testability through the use of transmitters and trip units.

(9) Unit 2 uses a signal from RRCS for automatic initiation of the SLCS. Zimmer and Perry do not.

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY INDEX

Codes and Standards	Safety System																								
	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	Y[1]	Z	AA
GDC 1	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X
GDC 2	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X
GDC 3	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X
GDC 4	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X
GDC 5	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X
GDC 10	X		X			X	X	X	X	X	X	X	X	X	X			X				X	X	X	
GDC 12																						(5)			
GDC 13	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	
GDC 15	X		X												X		X			X					
GDC 17																									
GDC 18																									
GDC 19	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X
GDC 20	X	X	X												X				X		X	X	X		X
GDC 21	X	X	X	X	X										X				X		X	X	X		X
GDC 22	X	X	X	X	X										X				X		X	X	X		X
GDC 23	X	X	X	X	X										X				X		X	X	X		X
GDC 24	X	X	X	X	X										X				X		X	X	X		X
GDC 25	X																					X			X
GDC 26																(2)									
GDC 27																									
GDC 28																									
GDC 29	X	X	X	X	X										X	2	X		X	X	X	X	X		X
GDC 30		X	X												X	X	X			X	X	X			

TABLE 7.1-3 (Cont'd.)

Codes and Standards	Safety System																							
	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	Y[(1)]	Z	AA
GDC 33		X													X										
GDC 34		X	X		X												X								
GDC 35		X																		X					
GDC 37		X																		X					
GDC 38				X	X			X	X	X															
GDC 40				X	X			X	X	X															
GDC 41						X	X	X																	
GDC 43						X	X	X																	
GDC 44									X	X															
GDC 46									X	X															
GDC 54			X			X	X	X	X						X	X	X				X			X	
GDC 55			X												X	X	X								
GDC 56			X				X	X							X	X	X							X	
GDC 57			X			X			X																
GDC 60						X					X								X						
GDC 61																								X	
GDC 63																									
GDC 64											X								X						
IEEE-279-1971	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-308-1971	X		X	X	X										X	X	X			X					
IEEE-308-1974						X	X	X	X	X	X	X	X	X				X						X	X
IEEE-317-1976	X	X	X	X	X		X	X	X				X		X	X	X	X	X		X	X	X		
IEEE-323-1971	X	X	X	X	X								X		X	X	X		X	X	X	X	X		
IEEE-323-1974			X				X	X	X	X	X	X	X	X				X	X			(5)			X

TABLE 7.1-3 (Cont'd.)

Codes and Standards	Safety System																								
	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	Y[(1)]	Z	AA
IEEE-336-1971	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-338-1971	X	X	X	X	X								X		X	X	X		X	X	X	X	X		
IEEE-338-1977						X	X	X	X	X	X	X	X	X				X	X					X	
IEEE-338-1975																									X
IEEE-344-1971	X	X	X	X	X								X		X	X	X		X	X	X	X	X		
IEEE-344-1975																									X
IEEE-379-1972	A	A	A	A	A										A	A	A		A	A	A	A	A		
IEEE-379-1977						X	X	X	X	X	X	X	X	X				X						X	X
IEEE-384-1974	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	A	X	X	X
IEEE-387-1972		X											X												
IEEE-387-1977									X	X			X	X											
RG 1.6 REV. 0		X		X	X										X		X							X	X
RG 1.7 REV. 2							X																		
RG 1.11 REV. 0		X	X	X	X										X		X								
RG 1.22 REV. 0	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A		A	X	A	A	A	A	A	X	X
RG 1.29 REV. 3	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A	A	A	X	A	A	A	A	A	X	X
RG 1.30 REV. 0						X	X	X	X	X	X	X	X	X				X						X	X
RG 1.40 REV. 0																									
RG 1.45 REV. 0																					X				
RG 1.47 REV. 0	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X
RG 1.53 REV. 0	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A	A	A	X	A	A	A	A	A	X	X
RG 1.62 REV. 0	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X						X	X	X
RG 1.63 REV. 2	X	X	X		X		X	X	X				X		X		X	X	X		X	X	X	X	X
RG 1.68 REV. 2	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A	A	A	X	A	A	A	A	A	X	X

TABLE 7.1-3 (Cont'd.)

Codes and Standards	Safety System																								
	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	Y[1]	Z	AA
RG 1.70 REV. 3	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.73 REV. 0		A	A	A	A	X	X		X						A	A	A							X	
RG 1.75 REV. 2[3]	A	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	A	X	X	A	X	X	X
RG 1.89 REV. 0	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A	A	A	X	A	A	A	A	A	X	X
RG 1.97 REV. 3[4]																									
RG 1.100 REV. 1	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A	A	A	X	A	A	A	A	A	X	X
RG 1.105 REV. 2	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A		A		A	A	A	A	A	X	X
RG 1.118 REV. 2	A	A	A	A	A	X	X	X	X	X	X	X	X	X	A	A	A	X		A	A	A	A	X	X
RG 1.68.2 REV. 1																		X							
RG 1.183 REV. 0				A												A									

(1) The RPT is related to the RPS trip. It is not assessed as the ATWS trip.
(2) Alternate reactivity control systems do not include SLCS for BWRs, only reactor manual control and recirculation flow control.
(3) The extent of implementation for the requirements of RG 1.75 for NSSS are as follows:

a. Electrical isolation and/or physical separation may be accomplished with isolators and barriers (such as metallic conduits, enclosures and plates) used in accordance with RG 1.75. Such means are not intended to substitute for structural barriers. They will not affect any arrangement of control room, control panels, or any other mechanical structure so designed to meet their safety function.
b. Physical separation between divisions of essential systems and between essential systems and essential circuits must be maintained for all essential NSSS systems except the NMS, the RPS, the PRMS, and the CRD hydraulic system. Physical separation between Class 1E divisional circuits for these specific NSSS systems is impractical to maintain due to system configuration, channel grouping, space limitation, and/or operational considerations. These systems, however, are designed to achieve single-failure protection so that no credible event can disable sufficient equipment to prevent reactor shutdown, removal of decay heat from the core, or to prevent isolation of the primary containment in the event of a DBA.

(4) RG 1.97 is discussed in Section 1.10.
(5) Applicable to PRNM system only.

NOTES:

1. The letter X on the table indicates a system requirement and the letter A indicates that the code or standard is not a design basis, but the text provides a description of the extent of design agreement.

2. All general design criteria, selected IEEE standards, and RG 1.1 through 1.118 are included as indicated for each NSSS system.

TABLE 7.1-3 (Cont'd.)

<u>KEY</u>:

B	=	RPS (Reactor Protection System)
C	=	ECCS (Emergency Core Cooling System)
D	=	PCRVICS (Primary Containment Reactor Vessel Isolation Cooling System)
E	=	RCSCM (RHR Containment Spray Cooling Mode)
F	=	RSPCM (RHR Suppression Pool Cooling Mode)
G	=	SGTS (Standby Gas Treatment System)
H	=	CGCS (Combustible Gas Control System)
I	=	Reactor Building HVAC
J	=	Service Water
K	=	SW Pump Bays Ventilation
L	=	Control Building HVAC
M	=	Control Building Chilled Water
N	=	Standby Power
O	=	DG Building Ventilation
P	=	RCIC (Reactor Core Isolation Cooling)
Q	=	SLCS (Standby Liquid Control System)
R	=	RSCM (RHR Reactor Shutdown Cooling Mode)
S	=	RSS
T	=	PRMS (Process Radiation Monitoring System)
U	=	High/Low Pressure Interlocks
V	=	LDS (Leak Detection System)
W	=	NMS (Neutron Monitoring System)
Y	=	RPT(1) (Recirculation Pump Trip)
Z	=	Spent Fuel Pool Cooling and Cleanup
AA	=	RRCS (Redundant Reactivity Control System)

TABLE 7.1-4

EXAMPLES OF REGULATORY GUIDES APPLICABLE TO PLANT SYSTEMS

Reg. Guide	Description	SWEC System (Typical Example)	FSAR Sec. No.	FSAR Figure No. (P&ID)	FSAR Figure No. (LSK)
1.21	Liquid radioactive release Gaseous radioactive release Solid radioactive release	LWS, SWP RMS, OFG WSS	11.2.3 11.3.3 11.4.4	11.2-1 A-J 11.3-1 A-C 11.4-1 A-H	N/A
1.23	Onsite meteorological program	MMS	2.3.3	2.3-1 - 2.3-46	N/A
1.45	Reactor coolant pressure boundary	LDS, LMS (GE-E31)	5.2	5.2-1 - 5.2-9	N/A
1.47	Bypassed/inoperable status	SFC	7.1.2.3	Table 7.1-1 Table 7.1-2 Table 7.1-3	N/A
1.53	Single failure criteria for protection system	SFC	7.6.2.7.3	N/A	N/A
1.62	Manual initiation of protective actions	CMS	7.5.2	Table 7.5-1	N/A
1.68	Initial test program	FWS, CNM	14.2	Table 14.2-30 Table 14.2-27	N/A
1.75	Physical independence of electrical systems	ISC	7.1.2.3 8.3	Table 7.1-3	N/A
1.89	Qualification of Class 1E equipment	SWEC 1E Specs	3.10 3.11	Table 3.10A-1 Table 3.10B-1	N/A
1.97	Post-accident parameter set	ERF	1.8 1.10	Table 1.8-1 Table 1.10-1	II.B.3-1
1.105	Instrument setpoints	All	7.1.2.3	N/A	N/A
1.106	Thermal overload protection of motors on motor-operated valves	E;RHS N1E:LWS	8.3.1.1.6	N/A	N/A
1.120	Fire protection guidelines	FPW	9.5.1 Appendix 9A & 9B	N/A	N/A
1.133	Loose parts detection program	LPM	4.4.6	N/A	4.4-7 4.4-8 4.4-9
8.8	ALARA (Occupational radiation exposure as low as reasonably achievable)	RMS	12.1	N/A	N/A

7.2 REACTOR PROTECTION (TRIP) SYSTEM INSTRUMENTATION AND CONTROLS

7.2.1 Reactor Protection System Description

7.2.1.1 Reactor Protection System Function

The RPS is a dual-trip electrical alarm and actuating system designed to prevent the reactor from operating under unsafe or potentially unsafe conditions. The RPS is designed to provide a signal to cause rapid insertion of control rods (scram) and shut down the reactor when specific variables exceed predetermined limits.

7.2.1.2 Reactor Protection System Operation

Arrangements of RPS logic, instrumentation, equipment, and information displayed to the Operator are shown on Figure 7.2-1. The RPS instrumentation is shown in Table 7.2-1. Sensor channel arrangements are shown on Figure 7.2-1. The RPS power supply is discussed in Chapter 8.

The RPS instrumentation is divided into trip channels, trip logics, and trip actuator logics. During normal operation all trip channels, trip logics, and trip actuator logics essential to safety are energized.

The RPS design is based on two separate (A and B) trip systems. Each trip system has at least two independent trip channels (A1, A2, and B1, B2). Each trip channel is associated with trip logics of the same designation.

Trip logics A1 and A2 (Trip System A) outputs are combined in a one-out-of-two logic arrangement to control the A pilot scram valve solenoid in each of the four rod groups (a rod group consists of approximately 25 percent of the total of control rods). Trip logics B1 and B2 (Trip System B) outputs control the B pilot scram valve solenoid in each of the four rod groups.

When a trip channel contact opens, the trip logic de-energizes the trip actuator logic which de-energizes the pilot scram valves associated with that trip actuator logic. However, the other pilot scram valves for each rod must also be de-energized before the scram valves provide a reactor scram.

There is one dual-coil pilot scram valve and two scram valves for each control rod. The pilot scram valve is solenoid operated, with both solenoids normally energized. The pilot scram valves control the air supply to the scram valves for each control rod. With either pilot scram valve solenoid energized, air pressure holds the scram valves closed. The scram valves control the supply and discharge paths for CRD water.

When trip logics A1 or A2 and B1 or B2 are tripped, air is vented from the scram valves and allows CRD water to act on the CRD piston. This logic is one-out-of-two twice logic. Thus, all control rods are scrammed. The water displaced by the movement of each rod piston is exhausted into a scram discharge volume (SDV).

To restore the RPS to normal operation following any single actuator logic trip or a scram, the trip actuators must be reset manually. After a 10-sec delay, reset is possible only if the conditions that caused the scram have been cleared. The trip actuators are reset by operating switches in the control room. Four reset switches (one per trip channel) are provided.

There are two 125-V dc solenoid-operated backup scram valves that provide a second means of controlling the air supply to the scram valves for all control rods. When the solenoid for either backup scram valve is energized, the associated backup scram valve vents the air supply for the scram valves. This action initiates insertion of any withdrawn control rods regardless of the action of the scram pilot valves. The backup scram valves solenoids are energized (initiate scram) when the trip logics A1 or A2 and B1 or B2 are both tripped. They are energized by two separate Class 1E 125-V dc buses (Div. I, Div. II) making them electrically independent from the ac-operated pilot scram solenoid valve. Verification of backup pilot scram solenoid valve operation will be determined by postscram analysis as required by procedures.

The RPS power system is discussed in Section 8.3.1.

Sensor trip channel inputs to the RPS causing reactor scram are discussed in the following paragraphs.

7.2.1.2.1 Neutron Monitoring System

Neutron flux is monitored to initiate a reactor scram when predetermined limits are exceeded.

NMS instrumentation is described in Section 7.6.

The NMS sensor channels are part of the NMS and not the RPS; however, the NMS logics are part of the RPS. Each NMS-IRM logic receives its signal from one intermediate range monitor (IRM) channel, each average power range monitor (APRM) logic receives its signal from one APRM channel, and each oscillation power range monitor (OPRM) logic receives its signal from one OPRM channel. The output logics of the APRM, OPRM and IRM are combined to initiate the RPS trip circuit.

The NMS logics are arranged so that failure of any one logic cannot prevent the initiation of a high neutron flux, thermal power, or oscillation scram. As shown on Figure 7.6-8, there are eight NMS logics associated with the RPS. Each RPS trip channel receives inputs from two NMS logics. See Sections 7.6.1.4.1 and 7.6.1.4.3, respectively, for further discussions concerning the IRM and APRM/OPRM trip functions to RPS.

For the initial fuel load, high-high flux trip inputs from each source range monitor (SRM) produce a noncoincident reactor NMS trip. Following the initial fuel loading this noncoincident trip is removed. For subsequent refueling operations, the ability of the SRMs to produce a noncoincident NMS trip signal is governed by the method of refueling and the associated Technical Specifications and Technical Requirements Manual (TRM) implications.

The NMS scram logic trip contacts for IRM and APRM/OPRM can be bypassed by selector switches located in the main control room. IRM channels A, C, E, and G bypasses are controlled by one selector switch, and channels B, D, F, and H bypasses are controlled by a second selector switch. Each selector switch will bypass only one IRM channel at any time.

A single selector switch allows bypass of one of the four APRM/oscillation power range monitor (OPRM) channels. None of the four two-out-of-four voter channels may be bypassed.

Bypassing either an APRM/OPRM or an IRM channel will not inhibit the NMS from providing protective action where required.

<u>Intermediate Range Monitor</u> The IRMs monitor neutron flux between the upper portion of the SRM range to the lower portion of the APRM range. The IRM detectors are positioned in the core by remote control from the control room.

The IRM is divided into two groups of four IRM channels in each group. Two IRM channels are associated with each of the trip channels of the RPS. The arrangement of IRM channels allows one IRM channel in each group to be bypassed.

Each IRM channel includes four trip circuits. One trip circuit is used as an instrument trouble trip. It operates on three conditions:

1. When the high voltage drops below a present level,

2. When one of the modules is not plugged in, or

3. When the OPERATE-CALIBRATE switch is not in the OPERATE position.

Each of the other trip circuits is specified to trip when preset downscale or upscale levels are reached.

The trip functions actuated by the IRM trips are indicated in Table 7.6-4. The reactor mode switch determines whether IRM trips are effective in initiating a reactor scram. With the reactor mode switch in REFUEL, STARTUP, or SHUTDOWN, an IRM upscale or inoperative trip signal actuates a NMS trip of the RPS. Only one of the IRM channels must trip to initiate a NMS trip of the associated RPS trip channel.

Average Power Range Monitors The APRM channels receive and average input signals from the local power range monitoring (LPRM) channels and can provide a continuous indication of average reactor power from a few percent to greater than rated reactor power.

The APRMs supply trip signals to the RPS via two-out-of-four voter channels. Table 7.6-6 lists the APRM trip functions. The APRM upscale thermal power scram trip setpoints vary as a function of reactor recirculation loop flow. Each APRM channel receives a flow signal representative of total recirculation flow. This signal is provided by summing the flow signals from the two recirculation loops. These flow signals are sensed from four pairs of elbow taps, two in each recirculation loop. The APRM signal for the thermal power scram trip is passed through a 6-sec time constant circuit to simulate thermal power. A faster response (approximately 0.09 sec) APRM upscale trip has a fixed setpoint, not variable with recirculation flow. Any APRM upscale or inoperative trip initiates a NMS trip in the RPS. Only the trip logic associated with that APRM is affected. At least two APRM channels must trip to cause a scram. The Operator can bypass only one APRM channel.

In addition to the IRM upscale trip, an APRM trip function with a setpoint of 15-percent power is active when the reactor mode switch is in the STARTUP position, or while in the refueling operational condition and performing shutdown margin demonstrations.

Diversity of trip initiation for excursions in reactor power is provided by the NMS trip signals and reactor vessel high-pressure trip signals. An increase in reactor power will initiate protective action from the NMS as discussed in the above paragraphs.

This increase in power results in a reactor pressure increase due to a higher rate of steam generation. The turbine control valve will stay open until the load limit of the turbine generator occurs. Once the pressure control limits are reached, reactor pressure will increase until the reactor vessel high-pressure trip results. These variables are independent of one another and provide diverse protective action for this condition.

<u>Oscillation Power Range Monitor</u> Each APRM chassis includes an OPRM function. The OPRM monitors LPRM detector signals to detect thermal-hydraulic instabilities in the reactor and initiates a reactor scram if the oscillations exceed predefined levels. An OPRM channel processes LPRM detector data and total recirculation flow data associated with the APRM channel in which the OPRM is located. The OPRMs supply trip signals to the RPS via two-out-of-four voter channels. In addition, the OPRM trips are voted separately from the APRM trips in the two-out-of-four voter channels. The OPRM trip is enabled only when the reactor core flow, as measured by the recirculation, is below approximately 75 percent and the reactor power is above approximately 23 percent. The OPRM is discussed in Section 7.6.1.4.3 and the OPRM functions are listed in Table 7.6-6.

7.2.1.2.2 Other Trip Signals

1. <u>Reactor Vessel High Pressure</u> A reactor vessel pressure increase during reactor operation compresses the steam voids and results in increased reactivity; this causes increased core heat generation that could lead to fuel barrier failure and reactor overpressurization. A scram counteracts a pressure increase by quickly reducing core fission heat generation. The reactor vessel high-pressure scram

works in conjunction with the pressure relief system to prevent reactor vessel pressure from exceeding the maximum allowable pressure. The reactor vessel high-pressure scram setting also protects the core from exceeding thermal-hydraulic limits that result from pressure increases during events that occur when the reactor is operating below rated power and flow.

Reactor pressure is monitored by four redundant pressure transmitters, each of which provides a reactor high pressure signal input to one of the four RPS trip logics.

The reactor level instruments reference legs that also act as pressure transmitter sensing lines are connected to backfill injection lines and receive injection water from the CRD system. This injection flow eliminates the buildup of noncondensable gases in the reference leg. The reference leg isolation valve located between the point of backfill injection and the condensing chamber is locked open. This prevents reference leg overpressurization due to CRD injection flow driving head.

2. Reactor Vessel Low Water Level Decreasing water level while the reactor is operating at power decreases the reactor coolant inventory. Should water level decrease too far, fuel damage could result as steam voids form around fuel rods. A reactor scram reduces the fission heat generation within the core. Reactor vessel water level is monitored by four redundant differential pressure transmitters, each of which provides a reactor vessel low water level signal input to one of the four RPS trip channels.

   Backfill injection lines are installed on each reactor water level instrument reference leg. The injection lines provide a metered flow of water from the CRD system to each leg. The backfill is designed to prevent the accumulation of dissolved noncondensable gases in the reference legs. These gases can produce erroneous level measurements during reactor pressure vessel (RPV) depressurization. The effects due to the backfill injection flow rate are negligible. These effects are within the boundaries of the low water level trip setpoints.

Diversity of trip initiation for breaks in the reactor coolant pressure boundary (RCPB) is provided by high drywell pressure trip signals.

3. Turbine Stop Valve Position A turbine trip will initiate closure of the turbine stop valve which can result in a significant addition of positive reactivity to the core as the reactor vessel pressure rise causes steam voids to collapse. The turbine stop valve closure scram initiates a scram earlier than either the NMS or reactor vessel high pressure to provide the required margin below core thermal-hydraulic limits for this category of abnormal operational transients. The scram counteracts the addition of positive reactivity caused by increasing pressure by inserting negative reactivity with control rods.

   Although the reactor vessel high-pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the nuclear system, the turbine stop valve closure scram provides additional margin to the reactor vessel pressure limit.

   Turbine stop valve closure inputs to the RPS originate from four redundant valve stem position switches mounted on the four turbine stop valves. Each switch opens before the valve is closed more than specified in Technical Specifications, and provides positive indication of closure. Each switch provides an input signal to two of the four RPS sensor trip logics. The logic of the whole system is arranged so that closure of three or more valves is required to initiate a scram. The switches are arranged so that no single failure can prevent a turbine stop valve closure scram.

   Diversity of trip initiation for increases in reactor vessel pressure due to termination of steam flow by turbine stop valve closure is provided by reactor vessel high pressure and high neutron flux trip signals.

   Turbine stop valve closure trip bypass is initiated by four pressure devices sensing turbine first-stage pressure. The turbine stop valve closure scram is automatically bypassed if the turbine first-stage pressure is less than that corresponding to 26 percent of rated reactor power. The bypass is automatically removed above 26 percent of reactor power.

4. Turbine Control Valve Position Generator load rejection with the turbine power above 26 percent or a turbine trip automatically initiates a fast closure of the turbine control valves, which results in a significant addition of positive reactivity to the core as nuclear steam pressure rises. The turbine control valve fast closure scram initiates a scram earlier than either the NMS or nuclear system high pressure to provide the required margin below core thermal-hydraulic limits for this category of abnormal operational transients. The scram counteracts the addition of positive reactivity resulting from increasing pressure by inserting negative reactivity with control rods. Although the reactor vessel high-pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the reactor vessel, the turbine control valve fast closure scram provides additional margin to the nuclear system pressure limit. The turbine control valve fast closure scram setting is selected to provide timely indication of control valve fast closure.

   Turbine control valve fast closure inputs to the RPS originate from oil line pressure switches on each of four fast-acting control valve hydraulic mechanisms. Each pressure switch provides an input signal to one of the four RPS sensor trip channels. If hydraulic oil line pressure is lost, a turbine control valve fast closure scram is initiated.

   Automatic turbine control valve fast closure scram bypass is provided as described above for the turbine stop valve.

5. Main Steam Isolation Valves Position Main steam isolation valve (MSIV) closure can result in a significant addition of positive reactivity to the core as nuclear system pressure rises. There are two position switches mounted on each of the eight MSIVs. One inboard MSIV position switch and one outboard MSIV position switch provide a closure signal to the RPS.

Each switch is arranged to open before the valve is closed more than the setpoint specified in the Technical Specifications to provide the earliest positive indication of closure. Either of the two channels sensing isolation valve position can signal valve closure.

Each RPS sensor trip channel logic receives signals from the valves associated with two steam lines. Closure of at least one valve in three or more steam lines is required to initiate a scram.

At plant shutdown and during initial plant startup, a bypass is required for the MSIV closure scram trip to properly reset the RPS.

This bypass is in effect when the mode switch is in the SHUTDOWN, REFUEL, or STARTUP position. The bypass allows plant operation when the MSIVs are closed during low power operation. The operating bypass is removed when the mode switch is placed in RUN. Diversity of trip initiation due to main steam isolation is provided by reactor vessel high pressure and reactor power trip signals.

6. Scram Discharge Volume Water Level Water displaced by the CRD pistons during a scram goes to the SDV. If the SDV fills with water so that insufficient capacity remains for the water displaced during a scram, control rod movement would be hindered during a scram. To prevent this situation, the reactor is scrammed when the water level in the discharge volume is high enough to verify that the volume is filling up, yet low enough to ensure that the remaining capacity in the discharge volume can accommodate a scram.

   Four nonindicating level switches (one for each channel) provide SDV high water level inputs to the four RPS channels. In addition, a trip unit, with transmitter in each channel, provides redundancy with the level switch in that channel. This arrangement provides diversity as well as redundancy to assure that no single event could prevent a scram due to SDV high water level.

   The SDV high water level trip bypass for each trip logic is controlled by the manual operation of five keylocked switches: four bypass switches and the reactor mode switch. The mode switch must be in the SHUTDOWN or REFUEL position to allow manual bypass of this trip by closure of a separate bypass switch for each channel. This bypass allows the Operator to reset the RPS scram relays so that the SDV may be drained. Resetting the trip actuators opens the SDV vent and drain valves. An annunciator in the main control room indicates the bypass condition.

7. Drywell Pressure High pressure inside the drywell may indicate a break in the RCPB. Scram is initiated to minimize the possibility of fuel damage. Drywell pressure is monitored by four pressure transmitters. Each transmitter provides an input to one of the four RPS sensor trip channels.

8. Deleted.

9. Manual Scram Scram can be initiated manually. There are four manual scram switches (A1, A2, B1, and B2), one for each of the four RPS trip channels. Activating manual scram switch A1 or A2 will de-energize the A scram pilot solenoid for all rods. Activating manual scram switch B1 or B2 will de-energize the B scram pilot solenoid for all rods. To manually initiate a full scram, manual scram switches A1 or A2 and B1 or B2 must be activated. By operating the manual scram switch for one logic at a time and then resetting that logic, each actuator logic can be tested for manual scram capability.

10. Reactor Mode Switch Manual Scram Even though the action is not a safety function, reactor scram can be initiated by placing the mode switch in the SHUTDOWN position. The mode switch consists of four independent banks of contacts. A SHUTDOWN position contact from each of the four contact blocks provides an input to one of the four RPS trip channels. The scram signal, initiated by placing the mode switch in SHUTDOWN, is automatically bypassed after 10 sec by a timer which allows the RPS logic to be reset.

7.2.1.3 Testability

The RPS can be tested during reactor operation by the following tests.

Manual Scram Test Depressing the manual scram button for one trip logic will de-energize the actuators, opening contacts in the actuator output logic. After the first trip logic is reset, the second trip logic is tripped manually, then reset, and so forth for the four manual scram buttons. The total test verifies the ability to de-energize all eight groups of scram pilot valve solenoids by using the manual scram push-button switches. In addition to control room and computer printout indications, scram group indicator lights verify that the actuator contacts have opened.

Single Rod Scram Test This test verifies the capability of each rod to scram. Timing traces can be made for each rod scrammed. Prior to the test, a physics review must be conducted to assure that the rod pattern during scram testing will not create a rod of excessive reactivity worth.

Sensor Test This test applies a test signal to each RPS sensor trip channel in turn to observe that a logic trip results. This test also verifies the electrical independence of the channel circuitry. Test signals can be applied to the process-type sensing instruments (pressure and differential pressure) through calibration taps. Calibration and test controls for pressure transmitters, level transmitters, and valve position switches are located in the turbine building and reactor building. A cover plate or sealing device must be removed to gain access to the calibration adjustments on each transmitter. Only properly qualified plant personnel are granted access for the purpose of testing or calibration adjustments.

Transmitter operation will be ascertained during plant operation by comparison of the four individual channel trip units. Any deviation of a reading from normal (other units) would indicate a malfunction. Transmitter testing and calibration requirements are discussed in Technical Specifications.

The printer provided with the process computer shows verification of the correct operation of many sensors during plant startup and shutdown. The verification provided on the printer is not considered in the selection of test and calibration frequencies and is not required for plant safety.

The overall RPS response time is verified during preoperational testing from sensor trip to sensor trip channel relay de-energization and actuator de-energization, and can be verified thereafter by similar testing.

The response time limits for RPS instrumentation are described in TRM Section 3.3.1.1.

7.2.1.4 Design Basis

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Chapter 15 and Appendix A identify and evaluate events that jeopardize the fuel barrier and RCPB. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are also presented in Chapter 15.

The following variables are monitored in order to provide protective actions to the RPS indicating the need for reactor scram:

1. Neutron flux.
2. Reactor vessel high pressure.
3. Reactor vessel low water level.
4. Turbine stop valve closure.
5. Turbine control valve fast closure.
6. Main steam line isolation.
7. Scram discharge volume high level.
8. Drywell high pressure.
9. Deleted.

Plant conditions that require protective action involving the RPS are described in Chapter 15 and Appendix 15A.

7 2.1.4.1 Location and Minimum Number of Sensors

Neutron flux is the only essential variable of significant spatial dependence that provides inputs to the RPS. The basis for the number and locations is discussed below. The other requirements are fulfilled through the combination of logic arrangement.

Two transient analyses are used to determine the minimum number and physical location of required LPRMs for each APRM:

1. The first analysis is performed with operating conditions of 100-percent reactor power and 100-percent recirculation flow using a continuous rod withdrawal of the maximum worth control rod. In the analysis, LPRM detectors are mathematically removed from the APRM channels. This process is continued until the minimum numbers and locations of detectors needed to provide protective action are determined for this condition.

2. The second analysis is performed with operating conditions of 100-percent reactor power and 100-percent recirculation flow using a reduction of recirculation flow at a fixed design rate. Again, LPRM detectors are mathematically removed from the APRM channels. This process is continued until the minimum numbers and locations of detectors needed to provide protective action are determined for this condition.

   The results of the two analyses are analyzed and compared to establish the actual minimum number and location of LPRMs needed for each APRM channel.

An OPRM is associated with an APRM channel. The LPRM detectors assigned to an APRM are also used by the OPRM associated with the APRM. The LPRMs are grouped into cells within the OPRM. Analyses were performed with different levels of random LPRM failures. The OPRM becomes more sensitive to oscillations as the number of LPRM detectors per cell is reduced.

7.2.1.4.2 Prudent Operational Limits

Prudent operational limits for each safety-related variable trip setting are selected with sufficient margin so that a spurious scram is avoided. It is then verified by analysis that the release of radioactive material, following postulated gross failures of the fuel or the RCPB, is kept within acceptable bounds. Design basis operational limits are based on operating experience and constrained by the safety design basis and the safety analysis (Technical Specifications).

7.2.1.4.3 Margin

The margin is the value between analytical limits and the nominal trip setpoints of operation (scram) for the RPS for those parameters listed in Technical Specifications. The margin includes the maximum allowable accuracy error, calibration error, and sensor setpoint drift. Annunciators are provided, at the setpoints for the functions, to alert the Reactor Operator (RO) to the cause of the unsafe condition.

7.2.1.4.4 Levels

Levels requiring protective action are provided in Technical Specifications.

7.2.1.4.5 Range of Transient, Steady-State, and Environmental Conditions

Environmental conditions for proper operation of the RPS components are discussed in Section 3.11. The RPS power supply range of steady-state and transient conditions are provided in Chapter 8.

7.2.1.4.6 Malfunctions, Accidents, and Other Unusual Events That Could Cause Damage to Safety Systems

Unusual events are defined as malfunctions, accidents, and others that could cause damage to safety systems. Chapter 15 and Appendix 15A describe the following credible accidents and events: floods, storms, tornadoes, earthquakes, fires, accidents, and missiles. Each of these events is discussed below for the RPS.

Floods

The buildings containing RPS components have been designed to withstand the probable maximum flood (PMF) at the site location as described in Section 3.4.

Storms and Tornadoes

All buildings containing RPS components, except the turbine building, are protected against storms and tornadoes as described in Section 3.3.

Earthquakes

All buildings containing RPS components, except the turbine building, have been seismically designed as described in Chapter 3. Although the sensors that monitor turbine stop valve position and turbine governor valve fast closure (scram function variables) are designed and purchased to Category I requirements, they are physically mounted on non-Category I equipment and located in the turbine building, which is not Category I or tornado or missile protected. For this reason other diverse variables may be relied upon for reactor scram if components in the turbine building fail.

Turbine scram inputs are not guaranteed to function during a seismic event since full compliance to IEEE-279 and associated standards is not possible because the turbine building is not a Category I structure. To ensure implementation of a system which is as reliable as reasonably achievable, the design of the turbine scram inputs, up to the trip solenoids, conforms to those sections of IEEE-279 concerning single failure (Section 4.2), quality (Section 4.3), channel integrity (Section 4.5 excluding seismic), channel independence (Section 4.6), and testability (Section 4.10). The following describes the degree of compliance of the turbine scram inputs to IEEE-279.

The design of the RPS, which incorporates the turbine stop valve closure and turbine control valve closure trips, conforms to IEEE-279 excluding the seismic requirements in Section 4.5.

The design of the RPS uses a highly-reliable power supply, as described in Section 8.3.1.1.3. The RPS is designed to be fail-safe and trip on loss of power.

All cables associated with these instruments are routed in enclosed raceways. Each channel is routed in separate enclosed raceways and is never mixed with another channel. See Sections 8.3.1.3.1 and 8.3.1.4.2 for details.

The routing of the cables to the trip sensors in the turbine enclosure for the turbine stop valve and the turbine control valve fast closure trips is such that the only credible failures that will challenge the RPS are a SSE, a turbine missile, and a high-energy line break (HELB). The expected failure mode caused by these events would be loss of the sensors due to an open circuit or loss of continuity. This failure mode would cause a reactor trip. If the trip sensors failed closed or shorted due to the failure, the reactor pressure and reactor power trips, which are diverse, will still function to prevent damage to the reactor. Shorting of a single sensor would not prevent protective action by the other sensors. Each of the inputs is isolated from the remainder of the RPS logic by the use of interposing relays. An open sensor circuit would signal a trip.

The following is a listing of other sensors which provide input signals to the RPS or perform a safety-related function and are located in or routed through nonseismically-qualified structures.

Sensor	Location
Turbine stop valve and turbine control valve trip bypass	Turbine enclosure
MSIV pressure trip input	Turbine enclosures
Condenser vacuum	Turbine enclosure

The degree of conformance to IEEE-279 and associated standards, as discussed in Section 7.2.2 for RPS and Section 7.3.2 for PCRVICS, includes the functions listed above.

The above all use the RPS power supply which is described in Section 8.3.1.1.3.

The cable routing information is shown in Sections 8.3.1.3.1 and 8.3.1.4.2.

The routing of the cables to the trip sensors in the turbine enclosure for each of the above is such that the only credible failures that will challenge the system are a SSE, a turbine missile, and a HELB. The expected failure mode caused by these events would be loss of the sensors due to opening or loss of continuity.

A turbine stop valve and control valve trip bypass failure that produced an open circuit or a short to ground of either lead wire would result in a bypass of turbine stop valve and control valve fast closure trip circuit and disabling of RPT circuit. Two or more trips in separate channels would result in bypass of the turbine stop valve and control valve fast closure trips to the RPS and disabling of the RPT function. Turbine stop valve and control valve trip failure would still leave reactor pressure and reactor power trips and protect the reactor. RPT is postulated on failure of normal scram; therefore, RPS is backup to RPT.

A failure that produced shorted lead wires would not enable the bypass function for the turbine stop valve and control valve trips, and would enable the RPT.

A MSIV pressure trip failure that produced an open circuit would result in a circuit trip. Two or more trips in separate channels would result in isolation signal for main steam lines. A short to ground of either lead wire would have the same result. A failure that produced shorted lead wires would disable that trip circuit. At least two such failures in one trip channel are needed to disable the main steam lines isolation signal for low steam line pressure. For analysis of low steam line pressure sensors in the event of turbine pressure regulator failure-open, refer to Section 15.1.3.

A condenser vacuum sensor failure that produced an open circuit would result in a circuit trip. Two or more trips in separate channels would result in isolation signal for main steam lines. A short to ground of either lead wire would have the same result. A failure that produced shorted lead wires would disable that trip circuit. At least two such failures in one trip channel are needed to disable the main steam line isolation signal for low condenser vacuum. For analysis of condenser vacuum sensors in the event of loss of condenser vacuum, refer to Section 15.2.5.

Fires

To protect the RPS in the event of a postulated fire, RPS trip logics have been divided into four separate sections within two separate RPS panels separated by fire barriers. If a fire occurred within one of the sections or in the area of one of the panels, the RPS functions would not be prevented. Functions of the RPS are protected from the effects of fire as described in Section 9.5.1.

Accidents

The following RPS system components are located inside the drywell and are designed for the environmental conditions described in Section 3.11.

1. MSIV (inboard) position switches.

2. Reactor vessel pressure and reactor vessel water level instrument taps and sensing lines, which terminate outside the drywell.

3. Drywell pressure instrument taps.

Pipe Breaks Outside the Primary Containment

RPS components are designed for these environmental conditions as described in Section 3.11.

Feedwater Line Breaks

RPS components are designed for these environmental conditions as described in Section 3.11.

Missiles

Buildings containing RPS components, except the turbine building, are protected against missiles as described in Section 3.5.

7.2.1.4.7 Minimum Performance Requirements

Minimum performance requirements for RPS instrumentation and controls are provided in Technical Specifications.

Table 7.2-2 provides a list of instruments used for protection that have a common instrument tap. In each case, a redundant instrument is utilized such that if a line break or blockage were to occur, protection is not compromised. Diversity is not used.

7.2.1.5 Final System Drawings

The instrument and electrical drawings (IED) have been provided for the RPS in this section. RPS elementary diagrams are listed in Final Safety Analysis Report (FSAR) Table 1.7-1. These drawings were submitted separate from the FSAR as a drawing package. Functional and architectural design differences between the Preliminary Safety Analysis Report (PSAR) and FSAR are listed in Table 1.3-8.

7.2.2 Analysis

The RPS is designed so that loss of plant instrument air, a plant load rejection, or a turbine trip will not prevent the completion of the safety function. A system level qualitative-type plant failure modes and effects analysis (FMEA) of the RPS is provided in Appendix 15A, Plant Nuclear Safety Operational Analysis (NSOA).

7.2.2.1 Conformance to 10CFR50, Appendix A, General Design Criteria

The general design criteria conformance discussions provided in Section 3.1 apply to the RPS as specified by Table 7.1-3.

The following paragraphs provide additional information on RPS system performance to the general design criteria.

General Design Criterion 22 The redundant portions of the RPS are separated in such a way that no single failure or credible natural disaster can prevent a scram, except the turbine scram inputs that originate from the non-Category I turbine building. In addition, drywell pressure and vessel water level are diverse variables.

General Design Criterion 23 The RPS is designed (including logic and actuated devices) to be fail-safe. A loss of electrical power or air supply will not prevent a reactor scram. Postulated adverse environments will not prevent a scram.

General Design Criterion 24 The RPS has no direct interaction with any plant control system. However, the RPS does receive inputs from the reactor mode switch and the NMS, which also provide inputs to plant control systems through isolation devices.

7.2.2.2 Conformance to IEEE Standards

The IEEE Standards that apply to the RPS are specified in Table 7.1-3. The following conformance discussions apply specifically to the RPS. Refer to Section 7.1.2.2 for conformance discussions that apply generically to the RPS.

7.2.2.2.1 Conformance to IEEE-279-1971

Paragraph 4.1 The RPS automatically initiates the appropriate protective actions, whenever the conditions described in Section 7.2.1 reach predetermined limits, with precision and reliability assuming the full range of conditions and performance discussed in Section 7.2.1.4.

Paragraph 4.2 Each of the conditions (variables) described in Section 7.2.1 is monitored by redundant sensors supplying input signals to redundant trip logics. Independence of redundant RPS equipment, cables, instrument tubing, etc., is maintained and single-failure criteria preserved through the application of the separation criteria, as described in Section 8.3, to assure that no single credible event can prevent the RPS from accomplishing its safety function.

Paragraph 4.3 For a discussion of the quality of RPS components and modules, refer to Sections 3.2 and 3.11.

Paragraph 4.4 All safety-related equipment, as defined in Tables 3.10A-1 and 3.10B-1, is designed to meet its performance requirements under the postulated range of operational and environmental constraints. Detailed discussion of qualification is contained in Sections 3.10 and 3.11.

Paragraph 4.5 For a discussion of RPS channel integrity under all extremes of conditions described in Section 7.2.1.4, refer to Sections 3.10, 3.11, 8.2.1, and 8.3.1.

Paragraph 4.6 RPS channel independence is maintained through the application of the separation criteria as described in Section 8.3.

Paragraph 4.7 The RPS has no direct interaction with any plant control system. However, the RPS receives inputs from the reactor mode switch and the NMS, which also provide inputs to plant control systems through isolation devices.

Paragraph 4.8 The RPS trip variables are direct measures of the following possible conditions: reactor overpressure, reactor overpower, gross fuel damage, or abnormal conditions within the RCPB except as follows:

> Due to the normal throttling action of the turbine control valves with changes in the plant power level, measurement of control valve position is not an appropriate variable from which to infer the desired variable, which is "rapid loss of the reactor heat sink." Consequently, a measurement of a control valve fast closure trip is used, as the trip signal (indicative of load reject).

Paragraph 4.9 Refer to RG 1.22 described in Section 7.2.2.3.

Paragraph 4.10 Refer to RG 1.22 described in Section 7.2.2.3.

Paragraph 4.11 The following RPS trip variables have no provision for sensor removal from service because of the use of valve position limit switches as the channel sensor:

1. MSIV closure trip.

2. Turbine stop valve closure trip.

During periodic test of any one trip channel, a sensor may be valved out of service and returned to service under administrative control procedures. Since only one sensor is valved out of service at any given time during the test interval, protection action capability for RPS automatic initiation is maintained through the remaining redundant instrument channels.

A sufficient number of IRM channels has been provided to permit any one IRM channel in a given trip system to be manually bypassed and still ensure that the remaining operable IRM channels comply with the IEEE-279 single-failure design requirements.

One IRM manual bypass switch has been provided for each RPS trip system. The mechanical characteristics of this switch permit only one of the four IRM channels of that trip system to be bypassed at any time. In order to accommodate a single failure of this bypass switch, electrical interlocks have also been incorporated into the bypass logic to prevent bypassing of more than one IRM in that trip system at any time. Consequently, with any IRM bypassed in a given trip system, three IRM channels remain in operation to satisfy the protection system requirements.

One APRM manual bypass switch has been provided to permit one of the four APRMs/OPRMs to be bypassed at any time. Mechanical interlocks have been provided in the bypass switch and electrical interlocks have been provided in the bypass circuitry to accommodate the possibility of switch failure. With an APRM/OPRM bypassed by the switch, sufficient APRM/OPRM channels remain in operation to provide the necessary protection for the reactor.

The mode switch produces operating bypasses which need not be annunciated because they are removed by normal reactor operating sequence.

Paragraph 4.12 For a discussion of RPS operating bypasses, refer to Section 7.2.1.2.

Paragraph 4.13 For a discussion of bypass and inoperability indication, refer to Section 7.1.2.3, RG 1.47.

Paragraph 4.14 All instrumentation valves associated with the periodic testing of individual RPS trip variable sensors are under administrative control.

Manual bypassing of any IRM or APRM channel is accomplished with control room selector switches under the administrative control of the Operator.

Manual controls for the SDV high water level trip operating bypass and the MSIV closure trip operating bypass are located in the control room and are under the direct administrative control of the Operator.

Under normal operating conditions, all four channels of the turbine stop valve closure trip and control valve fast closure trip operating bypass are in operation and are automatically removed from service as reactor power is increased above the switch setpoint, and are automatically reinstated as reactor power is reduced below this same setpoint. During periodic testing of each bypass channel, one sensor at a time is removed from service under administrative control.

Paragraph 4.15 The reactor mode switch implements more restrictive scram trip setpoints when it is shifted from RUN to STARTUP. As the mode switch is moved to STARTUP:

1. The APRM upscale neutron scram trip is replaced by the more restrictive APRM startup scram trip at 15 percent power.

2. The IRM range switch dependent scram trips are enabled.

In addition to the mode switch dependent multiple setpoints, the flow channels that supply control and reference signals for the APRM upscale thermal scram continually vary the scram setpoint as recirculation system flow changes. A sensed reduction in flow results in more restrictive scram trip setpoints.

The devices used to prevent improper use of the less restrictive setpoints (the mode switch, IRM range switches, the IRM and APRM signal conditioning equipment, and the flow channels) are designed in accordance with criteria regarding the performance and reliability of protection system equipment.

Paragraph 4.16 Once the RPS trip logic has been de-energized as a result of a sensor trip channel becoming tripped, or the depressing of a manual scram push button, the scram contactor seal-in contact opens and completion of protection action is achieved without regard to the state of the initiating sensor trip channel. After initial conditions (variable trip and logic de-energization) return to normal, deliberate Operator action is required to return (reset) the RPS logic to normal (energized).

Paragraph 4.17 Refer to the discussion of RG 1.62 in Section 7.2.2.3.

Paragraph 4.18 During reactor operation, access to setpoint or calibration controls is not possible for the following RPS trip variables:

1. SDV high water level trip (except for redundant level transmitters).

2. MSIV closure trip.

3. Turbine stop valve closure trip.

Access to setpoint adjustments, calibration controls, and test point for all other RPS trip variables is under the administrative control of the Control Room Operator.

Paragraph 4.19 When any one of the redundant RPS trip sensors exceeds its setpoint value, a control room annunciator is initiated to identify that variable and a typed record is available from the process computer.

Paragraph 4.20 The RPS is designed to provide the Operator with accurate and timely information pertinent to its status. It does not give anomalous indications confusing to the Operator.

Paragraph 4.21 During periodic testing of the RPS sensor channels, the Operator can determine defective components and (except as noted below) replace them during plant operation. During reactor operation, the Control Room Operator is able to determine failed sensors for the following RPS trip variables, but subsequent repair can only be accomplished during reactor shutdown:

1. MSIV trip.

2. Turbine stop valve closure trip.

3. Neutron monitoring (APRM) system trip.

4. Neutron monitoring (IRM) system trip.

Replacement of IRM and LPRM detectors must be accomplished during plant shutdown. Repair of the remaining portions of the NMS may be accomplished during plant operation by appropriate bypassing of the defective instrument channel (except the NMS two-out-of-four voter channels which may not be bypassed). The design of the systems facilitates rapid diagnosis and repair.

Paragraph 4.22 The identification scheme for the RPS system is discussed in Section 8.3.

7.2.2.2.2 Conformance to IEEE-338-1971

Periodic testing of protection systems is complied with by being able to test the RPS from sensors to final actuators at any time during plant operation. The test must be performed in overlapping portions.

7.2.2.2.3 Conformance to IEEE-344-1971

Seismic qualification of Class 1E electric equipment requirements are satisfied by all Class 1E RPS equipment as described in Sections 7.1.2.3 and 3.10.

7.2.2.2.4 Conformance to IEEE-379-1972

While not a design basis, the extent to which the RPS satisfies this criteria is as follows: requirements are satisfied by consideration of the different types of failure and carefully designing all potential violations of the single-failure criterion out of the system. A discussion of the different types of failures is provided in Chapter 15.

7.2.2.2.5 Conformance to IEEE-384-1974

For a discussion on the criteria for independence of Class 1E equipment and circuits, see Section 7.1.2.2.

7.2.2.3 Conformance to Regulatory Guides

The regulatory guides that apply to the RPS are specified in Table 7.1-3. The following conformance discussions apply specifically to the RPS. Refer to Section 7.1.2.3 for conformance discussions that apply generically to the RPS.

Regulatory Guide 1.22 While not a design basis, the extent of compliance is as follows: the system is designed so that it may be tested during plant operation from sensor device to final actuator device except as noted in Section 7.2.2.2.1. The test must be performed by bypassing no more than one trip logic channel at a time so that a reactor scram will not occur as a result of the testing.

Regulatory Guide 1.53 See IEEE-279-1971, Paragraph 4.2, Section 7.2.2.2.

Position C.1 See Section 7.2.2.2.

Position C.2 The RPS is a fail-safe, fully testable system.

Position C.3 The requirements of this position are met at the system level. The reactor mode switch that supplies signals to the four redundant channels of the RPS conforms to the separation criteria. This switch consists of four separate cam-driven units, each one serving only one channel. Each individual unit is enclosed in a metallic enclosure. The contacts of each switch unit are used by its dedicated channel only. Circuits leaving each unit are run in conduits separate from other circuit conduits.

Position C.4 Failure of a single logic channel or a single actuator using the same power source will not disable the functional capability of redundant channels. Thus, the system level protective action is preserved.

Regulatory Guide 1.62 Means are provided for manual initiation of the RPS at the system level through the use of four armed push-button switches located on the control room benchboard.

Operation of two switches (one in each trip system) accomplishes the initiation of all actions performed by the automatic initiation circuitry. Placing the reactor mode switch in the SHUTDOWN position will also cause a system level initiation.

TABLE 7.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SPECIFICATIONS

Function	Instrument	Instrument Range*
Scram		
Reactor vessel high pressure	Pressure transmitter	0-3,000 psig
Drywell high pressure	Pressure transmitter	0-55 in Hg abs
Reactor vessel low water level (Level 3)**	Level transmitter	0-150 in H_2O
SDV high water level	Level switch Level transmitters	NA 0-100 in H_2O
Turbine stop valve closure	Position switch	NA
Turbine control valve fast closure	Pressure switch	NA
MSIV closure	Position switch	NA
Neutron monitoring system	See Section 7.6.1.4	
Bypass		
Turbine stop valve and control valve fast closure trip bypass	Pressure transmitter	0-3,000 psig

* See Technical Specifications for Allowable Values. The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

** The Technical Specifications Allowable Value for this function is expressed in terms of "above instrument zero."

TABLE 7.2-2

CATEGORY I INSTRUMENTATION WITH COMMON SENSING LINES

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag. No. SWEC	Instrument Tag. No. GE	Location	Remarks (5)	Remarks (6)
001(2)	K-001	K-001A/K-001B		2MSS*EFV3A	1J	--	--	--	--	--	--
	K-001A	--	K-001	--	1J	AG	2MSS*FT12A(H)	E31-N086A(H)	2CES*RAK041	002	014
	--	--	--	--	1J	AB	2MSS*FT13A(H)	E31-N086D(H)	2CES*RAK041	002	014
	K-001B	--	K-001	--	1J	A-	2MSS*FT11A(H)	C33-N003A(H)	2CES*RAK015	002	--
002(2)	K-002	K-002A/K-002B	--	2MSS*EFV2A	1J	--	--	--	--	--	--
	K-002A	--	K-002	--	1J	AG	2MSS*FT12A(L)	E31-N086A(L)	2CES*RAK041	001	013
	--	--	--	--	1J	AB	2MSS*FT13A(L)	E31-N086D(L)	2CES*RAK041	001	013
	K-002B	--	K-002	--	1J	A-	2MSS*FT11A(L)	C33-N003A(L)	2CES*RAK015	001	--
003(2)	K-005	K-005A/K-005B	--	2MSS*EFV3B	1J	--	--	--	--	--	--
	K-005A	--	K-005	--	1J	BG	2MSS*FT12B(H)	E31-N087A(H)	2CES*RAK041	004	016
	--	--	--	--	1J	BB	2MSS*FT13B(H)	E31-N087D(H)	2CES*RAK041	004	016
	K-005B	--	K-005	--	1J	B-	2MSS*FT11B(H)	C33-N003B(H)	2CES*RAK015	004	--
004(2)	K-006	K-006A/K-006B	--	2MSS*EFV2B	1J	--	--	--	--	--	--
	K-006A	--	K-006	--	1J	BG	2MSS*FT12B(L)	E31-N087A(L)	2CES*RAK041	003	015
	--	--	--	--	1J	BB	2MSS*FT13B(L)	E31-N087D(L)	2CES*RAK041	003	015
	K-006B	--	K-006	--	1J	B-	2MSS*FT11B(L)	C33-N003B(L)	2CES*RAK015	003	--
005(2)	K-019	K-019A,B & C	--	2ISC*EFV5	28A	--	--	--	--	--	--
	K-019A	--	K-019	--	28C	AG	2ISC*LT13A(L)	B22-N044A(L)	2CES*RAK010	034	007
	K-019B	--	K-019	--	28A	AG	2ISC*PT5D	B22-N068E	2CES*RAK004	--	--
	--	--	--	--	28A	AB	2ISC*PT4D	B22-N078D	2CES*RAK004	--	007
	--	--	--	--	28A	A-	2ISC*PI3A	B22-R004A	2CES*RAK004	--	007
	--	--	--	--	28A	AG	2ISC*PT2A	B22-N403A	2CES*RAK004	--	007
	--	--	--	--	28A	AG	2ISC*PT2B	B22-N403E	2CES*RAK004	--	007
	--	--	--	--	28A	AG	2ISC*PT5A	B22-N068A	2CES*RAK004	--	--
	--	--	--	--	28A	AB	2ISC*LT7D(L)	B22-N080D(L)	2CES*RAK004	010	007
	--	--	--	--	28A	AG	2ISC*LT12A(L)	B22-N095A(L)	2CES*RAK004	010	007
	--	--	--	--	28A	AG	2ISC*PT6A	B22-N062A	2CES*RAK004	--	007
	--	--	--	--	28B	A-	2ISC*PDI31A(L)	B22-R009A(L)	2CES*RAK004	026	007
	--	--	--	--	28B	AB	2ISC*LT11D(L)	B22-N081D(L)	2CES*RAK004	026	007
	--	--	--	--	28B	AG	2ISC*LT8A(L)	B22-N402A(L)	2CES*RAK004	026	007
	--	--	--	--	28B	AG	2ISC*LT8B(L)	B22-N402E(L)	2CES*RAK004	026	007
	--	--	--	--	28B	AG	2ISC*LT9A(L)	B22-N091A(L)	2CES*RAK004	026	007
	--	--	--	--	28B	AG	2ISC*LT9C(L)	B22-N091E(L)	2CES*RAK004	026	007
	--	--	--	--	31A	AG	2RHS*PDT24A(L)	E12-N058A	2CES*RAK004	--	007
	K-019C	--	K-019	--	28B	AG	2RSS*LT101(L)	--	LOCAL	026	007
	--	--	--	--	28A	AG	2RSS*PT102	--	LOCAL	--	007

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
006(2)	K-025	K-025A/K-025B	--	2ISC*EFV6	28A	--	--	--	--	--	--
	K-025A	--	K-025	--	28A	AG	2RSS*LT115(L)	--	LOCAL	010	028
	K-025B	--	K-025	--	28A	A-	2ISC*PT108	C33-N005	2CES*RAK026	--	--
	--	--	--	--	28A	A-	2ISC*PT109	C33-N008A	2CES*RAK026	--	028
	--	--	--	--	28A	AG	2ISC*PT4A	B22-N078C	2CES*RAK026	--	028
	--	--	--	--	28A	AO	2ISC*LT7A(L)	B22-N080C(L)	2CES*RAK026	--	028
	--	--	--	--	28A	A-	2ISC*PDT14A(L)	C33-N004A(L)	2CES*RAK026	010	028
	--	--	--	--	28B	AP	2ISC*LT10A(L)	B22-N073L(L)	2CES*RAK026	025	028
	--	--	--	--	28B	AP	2ISC*LT10C(L)	B22-N073R(L)	2CES*RAK026	025	028
	--	--	--	--	28B	AO	2ISC*LT11C(L)	B22-N081C(L)	2CES*RAK026	025	028
007(2)	K-034	R-564,R-575 K-034A,B, R-576,577		2ISC*EFV2	28A	--	--	--	--	--	--
	K-034A	--	K-034	--	28C	BY	2ISC*LT13B(L)	B22-N044B(L)	2CES*RAK009	037	005
	K-034B	--	K-034	--	28A	B-	2ISC*PT115	B35-N040	2CES*RAK027	--	--
	--	--	--	--	28A	BY	2ISC*PT4B	B22-N078B	2CES*RAK027	--	005
	--	--	--	--	28A	B-	2ISC*PI3B	B22-R004B	2CES*RAK027	--	005
	--	--	--	--	28A	BY	2ISC*PT2C	B22-N403B	2CES*RAK027	--	005
	--	--	--	--	28A	BY	2ISC*PT2D	B22-N403F	2CES*RAK027	--	005
	--	--	--	--	28A	BY	2ISC*PT6B	B22-N062B	2CES*RAK027	--	005
	--	--	--	--	28A	BY	2ISC*LT12B(L)	B22-N095B(L)	2CES*RAK027	008	005
	--	--	--	--	28A	BY	2ISC*LT7B(L)	B22-N080B(L)	2CES*RAK027	008	005
	--	--	--	--	28A	B-	2ISC*PDT14B(L)	C33-N004B(L)	2CES*RAK027	008	--
	--	--	--	--	28B	BY	2ISC*LT8C(L)	B22-N402B(L)	2CES*RAK027	027	005
	--	--	--	--	28B	BY	2ISC*LT8D(L)	B22-N402F(L)	2CES*RAK027	027	005
	--	--	--	--	28B	BY	2ISC*LT9B(L)	B22-N091B(L)	2CES*RAK027	027	005
	--	--	--	--	28B	BY	2ISC*LT9D(L)	B22-N091F(L)	2CES*RAK027	027	005
	--	--	--	--	28B	BY	2ISC*LT11B(L)	B22-N081B(L)	2CES*RAK027	027	005
	--	--	--	--	28B	B-	2ISC*PDI31B(L)	B22-R009B(L)	2CES*RAK027	027	005
	R-575	--	R-564	--	31B	BY	2RHS*PDT24B(L)	E12-N058B	2CES*RAK027	--	005
	R-564	R-575	K-034	--	31B	ZY	2RHS*PDT24C(L)	E12-N058C	2CES*RAK027	--	--
	R-577	--	K-034A	--	28B	BY	2RSS*LT112(L)	--	LOCAL	027	005
	R-576	--	R-577	--	28A	BY	2RSS*PT113	--	LOCAL	--	005
008(2)	K-038	K-038A,B, R-579	--	2ISC*EFV4	28A	--	--	--	--	--	--
	K-038A	--	K-038	--	28A	BG	2ISC*LT7C(H)	B22-N080A(H)	2CES*RAK005	028	010
	--	--	--	--	28A	B-	2ISC*PDT14C(H)	C33-N004C(H)	2CES*RAK005	028	010
	K-038B	--	K-038	--	28A	BY	2ISC*LT12B(H)	B22-N095B(H)	2CES*RAK027	007	010
	--	--	--	--	28A	BY	2ISC*LT7B(H)	B22-N080B(H)	2CES*RAK027	007	010
	--	--	--	--	28A	B-	2ISC*PDT14B(H)	C33-N004B(H)	2CES*RAK027	007	--
	--	--	--	--	28A	Z-	2ISC*LT105(H)	B22-N027(H)	2CES*RAK027	030	--

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
008(2)	--	--	--	--	28A	Z-	2ISC*PDT110(H)	C33-N017(H)	2CES*RAK027	030	--
cont'd.	R-579	--	K-038A	--	28A	BY	2RSS*LT116(H)	--	LOCAL	028	010
009(2)	K-041	K-041A,B,C	--	2ISC*EFV21	28C	--	--	--	--	--	--
	K-041A	--	K-041	--	30B	--	2RDS-PDT114(L)	C12-N008(L)	2CES-RAK103	--	--
	--	--	--	--	30B	--	2RDS-PDT117(L)	C12-N011(L)	2CES-RAK103	--	--
	K-041B	K-041C	K-041	--	33A	ZP	2CSH*PDT109(L)	E22-N057(L)	2CES*RAK024	--	--
	K-041C	--	K-041B	--	28C	Z-	2ISC-PDT114(L)	B22-N032(L)	2CES*RAK009	031	--
010(2)	K-068	K-068A,B,C,D	--	2ISC*EFV7	28A	--	--	--	--	--	--
	K-068A	K-068D	K-068	--	28A	AO	2ISC*LT7A(H)	B22-N080C(H)	2CES*RAK026	006	008
	--	--	--	--	28A	A-	2ISC*PDT14A(H)	C33-N004A(H)	2CES*RAK026	006	008
	K-068B	K-068C	K-068	--	28A	AG	2ISC*LT12A(H)	B22-N095A(H)	2CES*RAK004	005	008
	--	--	--	--	28A	AB	2ISC*LT7D(H)	B22-N080D(H)	2CES*RAK004	005	008
	K-068C	--	K-068B	--	28C	Z-	2ISC*PDI103(L)	B22-R005(L)	2CES*RAK010	032	--
	K-068D	--	K-068A	--	28A	AG	2RSS*LT115(H)	--	LOCAL	006	008
011(2)	K-107	K-107A/K-107B	--	2WCS*EFV222	37A	--	--	--	--	--	--
	K-107A	--	K-107	--	37A	Z-	2WCS*PDIS115(L)	G33-N025(L)	2CES*RAK006	012	--
	K-107B	--	K-107	--	37A	ZG	2WCS*FT67X(L)	E31-N036A(L)	2CES*RAK010	012	--
012(2)	K-113	K-113A/K-113B	--	2WCS*EFV300	37A	--	--	--	--	--	--
	K-113A	--	K-113	--	37A	Z-	2WCS*PDIS115(H)	G33-N025(H)	2CES*RAK006	011	--
	K-113B	--	K-113	--	37A	ZG	2WCS*FT67X(H)	E31-N036A(H)	2CES*RAK010	011	--
013(3)	K-003	--	--	2MSS*EFV1A	1J	AY	2MSS*FT15A(L)	E31-N086B(L)	2CES*RAK015	014	002
	--	--	--	--	1J	AO	2MSS*FT14A(L)	E31-N086C(L)	2CES*RAK015	014	002
014(3)	K-004	--	--	2MSS*EFV4A	1J	AY	2MSS*FT15A(H)	E31-N086B(H)	2CES*RAK015	013	001
	--	--	--	--	1J	AO	2MSS*FT14A(H)	E31-N086C(H)	2CES*RAK015	013	001
015(3)	K-007	--	--	2MSS*EFV1B	1J	BY	2MSS*FT15B(L)	E31-N087B(L)	2CES*RAK015	016	004
	--	--	--	--	1J	BO	2MSS*FT14B(L)	E31-N087C(L)	2CES*RAK015	016	004
016(3)	K-008	--	--	2MSS*EFV4B	1J	BY	2MSS*FT15B(H)	E31-N087B(H)	2CES*RAK015	015	003
	--	--	--	--	1J	BO	2MSS*FT14B(H)	E31-N087C(H)	2CES*RAK015	015	003
017(3)	K-009	--	--	2MSS*EFV3C	1J	C-	2MSS*FT11C(H)	C33-N003C(H)	2CES*RAK025	018	--
	--	--	--	--	1J	CG	2MSS*FT12C(H)	E31-N088A(H)	2CES*RAK025	018	020
	--	--	--	--	1J	CB	2MSS*FT13C(H)	E31-N088D(H)	2CES*RAK025	018	020
018(3)	K-010	--	--	2MSS*EFV2C	1J	C-	2MSS*FT11C(L)	C33-N003C(L)	2CES*RAK025	017	--
	--	--	--	--	1J	CG	2MSS*FT12C(L)	E31-N088A(L)	2CES*RAK025	017	019
	--	--	--	--	1J	CB	2MSS*FT13C(L)	E31-N088D(L)	2CES*RAK025	017	019

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag. No. SWEC	Instrument Tag. No. GE	Location	Remarks (5)	Remarks (6)
019(3)	K-011	--	--	2MSS*EFV1C	1J	CO	2MSS*FT14C(L)	E31-N088C(L)	2CES*RAK022	020	018
	--	--	--	--	1J	CY	2MSS*FT15C(L)	E31-N088B(L)	2CES*RAK022	020	018
020(3)	K-012	--	--	2MSS*EFV4C	1J	CO	2MSS*FT14C(H)	E31-N088C(H)	2CES*RAK022	019	017
	--	--	--	--	1J	CY	2MSS*FT15C(H)	E31-N088B(H)	2CES*RAK022	019	017
021(3)	K-013	--	--	2MSS*EFV3D	1J	D-	2MSS*FT11D(H)	C33-N003D(H)	2CES*RAK025	022	--
	--	--	--	--	1J	DG	2MSS*FT12D(H)	E31-N089A(H)	2CES*RAK025	022	024
	--	--	--	--	1J	DB	2MSS*FT13D(H)	E31-N089D(H)	2CES*RAK025	022	024
022(3)	K-014	--	--	2MSS*EFV2D	1J	D-	2MSS*FT11D(L)	C33-N003D(L)	2CES*RAK025	021	--
	--	--	--	--	1J	DG	2MSS*FT12D(L)	E31-N089A(L)	2CES*RAK025	021	023
	--	--	--	--	1J	DB	2MSS*FT13D(L)	E31-N089D(L)	2CES*RAK025	021	023
023(3)	K-015	--	--	2MSS*EFV1D	1J	DO	2MSS*FT14D(L)	E31-N089C(L)	2CES*RAK022	024	022
	--	--	--	--	1J	DY	2MSS*FT15D(L)	E31-N089B(L)	2CES*RAK022	024	022
024(3)	K-016	--	--	2MSS*EFV4D	1J	DO	2MSS*FT14D(H)	E31-N089C(H)	2CES*RAK022	023	021
	--	--	--	--	1J	DY	2MSS*FT15D(H)	E31-N089B(H)	2CES*RAK022	023	021
025(3)	K-021	--	--	2ISC*EFV15	28B	AP	2ISC*LT10A(H)	B22-N073L(H)	2CES*RAK026	006	029
	--	--	--	--	28B	AP	2ISC*LT10C(H)	B22-N073R(H)	2CES*RAK026	006	029
	--	--	--	--	28B	AO	2ISC*LT11C(H)	B22-N081C(H)	2CES*RAK026	006	029
026(3)	K-027	K-027A	--	2ISC*EFV17	28B	A-	2ISC*PDI31A(H)	B22-R009A(H)	2CES*RAK004	005	027
	--	--	--	--	28B	AG	2ISC*LT8A(H)	B22-N402A(H)	2CES*RAK004	005	027
	--	--	--	--	28B	AB	2ISC*LT11D(H)	B22-N081D(H)	2CES*RAK004	005	027
	--	--	--	--	28B	AG	2ISC*LT8B(H)	B22-N402E(H)	2CES*RAK004	005	027
	--	--	--	--	28B	AG	2ISC*LT9A(H)	B22-N091A(H)	2CES*RAK004	005	027
	--	--	--	--	28B	AG	2ISC*LT9C(H)	B22-N091E(H)	2CES*RAK004	005	027
	K-027A	--	K-027	--	28B	AG	2RSS*LT101(H)	--	LOCAL	005	027
027(3)	K-031	R-578	--	2ISC*EFV10	28B	BY	2ISC*LT11B(H)	B22-N081B(H)	2CES*RAK027	007	026
	--	--	--	--	28B	BY	2ISC*LT9B(H)	B22-N091B(H)	2CES*RAK027	007	026
	--	--	--	--	28B	BY	2ISC*LT9D(H)	B22-N091F(H)	2CES*RAK027	007	026
	--	--	--	--	28B	BY	2ISC*LT8C(H)	B22-N402B(H)	2CES*RAK027	007	026
	--	--	--	--	28B	BY	2ISC*LT8D(H)	B22-N402F(H)	2CES*RAK027	007	026
	--	--	--	--	28B	B-	2ISC*PDI31B(H)	B22-R009B(H)	2CES*RAK027	007	026
	R-578	--	K-031	--	28B	BY	2RSS*LT112(H)	--	LOCAL	007	026
028(3)	K-032	R-580	--	2ISC*EFV3	28A	B-	2ISC*PDT14C(L)	C33-N004C(L)	2CES*RAK005	008	006
	--	--	--	--	28A	BG	2ISC*LT7C(L)	B22-N080A(L)	2CES*RAK005	008	006
	--	--	--	--	28A	BO	2ISC*PT4C	B22-N078A	2CES*RAK005	--	006

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag. No. SWEC	Instrument Tag. No. GE	Location	Remarks (5)	Remarks (6)
028(3) cont'd.	--	--	--	--	28B	BG	2ISC*LT11A(L)	B22-N081A(L)	2CES*RAK005	029	006
	--	--	--	--	28B	BP	2ISC*LT10B(L)	B22-N073C(L)	2CES*RAK005	029	006
	--	--	--	--	28B	BP	2ISC*LT10D(L)	B22-N073G(L)	2CES*RAK005	029	006
	--	--	--	--	28A	B-	2ISC*PT122	C33-N008C	2CES*RAK005	--	006
	R-580	--	K-032	--	28A	BY	2RSS*LT116(L)	--	LOCAL	008	006
029(3)	K-036	--	--	2ISC*EFV8	28B	BG	2ISC*LT11A(H)	B22-N081A(H)	2CES*RAK005	028	025
	--	--	--	--	28B	BP	2ISC*LT10B(H)	B22-N073C(H)	2CES*RAK005	028	025
	--	--	--	--	28B	BP	2ISC*LT10D(H)	B22-N073G(H)	2CES*RAK005	028	025
030(3)	K-040	--	--	2ISC*EFV1	28A	Z-	2ISC*PDT110(L)	C33-N017(L)	2CES*RAK027	008	--
	--	--	--	--	28A	Z-	2ISC*LT105(L)	B22-N027(L)	2CES*RAK027	008	--
031(3)	K-042	--	--	2ISC*EFV22	28C	Z-	2ISC-PDT114(H)	B22-N032(H)	2CES*RAK009	009	--
	--	--	--	--	28C	B-	2ISC-FT47K(H)	B22-N034K(H)	2CES*RAK009	036	032
	--	--	--	--	28C	B-	2ISC-FT47W(H)	B22-N034W(H)	2CES*RAK009	038	032
	--	--	--	--	28C	B-	2ISC-FT47B(H)	B22-N034B(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47D(H)	B22-N034D(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47F(H)	B22-N034F(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47H(H)	B22-N034H(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47M(H)	B22-N034M(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47P(H)	B22-N034P(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47S(H)	B22-N034S(H)	2CES*RAK009	--	032
	--	--	--	--	28C	B-	2ISC-FT47U(H)	B22-N034U(H)	2CES*RAK009	--	032
	--	--	--	--	37A	--	2WCS-FT134(H)	G33-N037(H)	2CES*RAK009	--	--
032(3)	K-043	--	--	2ISC*EFV14	28C	A-	2ISC-FT47A(H)	B22-N034A(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47C(H)	B22-N034C(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47E(H)	B22-N034E(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47G(H)	B22-N034G(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47J(H)	B22-N034J(H)	2CES*RAK010	033	031
	--	--	--	--	28C	A-	2ISC-FT47L(H)	B22-N034L(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47N(H)	B22-N034N(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47R(H)	B22-N034R(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47T(H)	B22-N034T(H)	2CES*RAK010	--	031
	--	--	--	--	28C	A-	2ISC-FT47V(H)	B22-N034V(H)	2CES*RAK010	035	031
	--	--	--	--	28C	Z-	2ISC*PDI103(H)	B22-R005(H)	2CES*RAK010	010	--
033(3)	K-048	--	--	2ISC*EFV18	28C	A-	2ISC-FT47J(L)	B22-N034J(L)	2CES*RAK010	032	036
	--	--	--	--	28C	A-	2ISC-FT48A(L)	B22-N033A(L)	2CES*RAK010	034	036
034(3)	K-049	--	--	2ISC*EFV31	28C	A-	2ISC-FT48A(H)	B22-N033A(H)	2CES*RAK010	033	037
	--	--	--	--	28C	AG	2ISC*LT13A(H)	B22-N044A(H)	2CES*RAK010	005	037
035(3)	K-054	--	--	2ISC*EFV33	28C	A-	2ISC-FT47V(L)	B22-N034V(L)	2CES*RAK010	032	038
	--	--	--	--	28C	A-	2ISC-FT48C(L)	B22-N033C(L)	2CES*RAK010	--	038

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag. No. SWEC	Instrument Tag. No. GE	Location	Remarks (5)	Remarks (6)
036(3)	K-060	--	--	2ISC*EFV11	28C	B-	2ISC-FT48B(L)	B22-N033B(L)	2CES*RAK009	037	033
	--	--	--	--	28C	B-	2ISC-FT47K(L)	B22-N034K(L)	2CES*RAK009	031	033
037(3)	K-061	--	--	2ISC*EFV40	28C	B-	2ISC-FT48B(H)	B22-N033B(H)	2CES*RAK009	036	034
	--	--	--	--	28C	BY	2ISC*LT13B(H)	B22-N044B(H)	2CES*RAK009	007	034
038(3)	K-066	--	--	2ISC*EFV42	28C	B-	2ISC-FT48D(L)	B22-N033D(L)	2CES*RAK009	--	033
	--	--	--	--	28C	B-	2ISC-FT47W(L)	B22-N034W(L)	2CES*RAK009	031	033
039(3)	K-075	--	--	2RCS*EFV47A	29B	AB	2RCS*FT8A(H)	B35-N014B(H)	2CES*RAK009	040	059
	--	--	--	--	29B	AG	2RCS*FT6A(H)	B35-N014A(H)	2CES*RAK009	040	059
	--	--	--	--	29B	A-	2RCS*FT83A(H)	B35-N011A(H)	2CES*RAK009	040	059
040(3)	K-077	--	--	2RCS*EFV48A	29B	AB	2RCS*FT8A(L)	B35-N014B(L)	2CES*RAK009	039	058
	--	--	--	--	29B	AG	2RCS*FT6A(L)	B35-N014A(L)	2CES*RAK009	039	058
	--	--	--	--	29B	A-	2RCS*FT83A(L)	B35-N011A(L)	2CES*RAK009	039	058
041(3)	K-080	--	--	2RCS*EFV45B	29C	BO	2RCS*FT7B(L)	B35-N024C(L)	2CES*RAK015	043	060
	--	--	--	--	29C	BO	2RCS*FT9B(L)	B35-N024D(L)	2CES*RAK015	043	060
042(3)	K-081	--	--	2IAS*EFV201	19E	ZG	2IAS*PT230	--	LOCAL	--	--
	--	--	--	--	19E	--	2RSS-PT108	--	LOCAL	--	--
043(3)	K-082	--	--	2RCS*EFV46B	29C	BO	2RCS*FT7B(H)	B35-N024C(H)	2CES*RAK015	041	061
	--	--	--	--	29C	BO	2RCS*FT9B(H)	B35-N024D(H)	2CES*RAK015	041	061
044(3)	K-083	--	--	2RCS*EFV62A	29B	--	2RCS-PI43A	B35-R002A	2CES*RAK006	--	--
	--	--	--	--	29B	--	2RCS-PT44A	B35-N006A	2CES*RAK006	--	--
045(3)	K-084	--	--	2RCS*EFV63A	29B	--	2RCS-PI41A	B35-R001A	2CES*RAK006	--	--
	--	--	--	--	29B	--	2RCS-PT42A	B35-N005A	2CES*RAK006	--	--
046(3)	K-085	--	--	2RCS*EFV62B	29C	--	2RCS-PI43B	B35-R002B	2CES*RAK022	--	--
	--	--	--	--	29C	--	2RCS-PT44B	B35-N006B	2CES*RAK022	--	--
047(3)	K-086	--	--	2RCS*EFV63B	29C	--	2RCS-PI41B	B35-R001B	2CES*RAK022	--	--
	--	--	--	--	29C	--	2RCS-PT42B	B35-N005B	2CES*RAK022	--	--
050(3)	K-099	--	--	2ICS*EFV1	35A	JG	2ICS*PDT167(L)	E31-N084A(L)	2CES*RAK017	051	--
	--	--	--	--	35A	JG	2ICS*PT167X	E31-N085E	2CES*RAK017	--	--
051(3)	K-100	--	--	2ICS*EFV2	35A	JG	2ICS*PDT167(H)	E31-N084A(H)	2CES*RAK017	050	--
	--	--	--	--	35A	JG	2ICS*PT167Y	E31-N085A	2CES*RAK017	--	--

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
052(3)	K-105	--	--	2ICS*EFV3	35A	JY	2ICS*PDT168(L)	E31-N084B(L)	2CES*RAK029	053	--
	--	--	--	--	35A	JY	2ICS*PT168X	E31-N085F	2CES*RAK029	--	--
053(3)	K-106	--	--	2ICS*EFV4	35A	JY	2ICS*PDT168(H)	E31-N084B(H)	2CES*RAK029	052	--
	--	--	--	--	35A	JY	2ICS*PT168Y	E31-N085B	2CES*RAK029	--	--
054(3)	K-109	--	--	2ISC*EFV9	28B	BP	2ISC*PT16B	B22-N067C	2CES*RAK005	--	056
	--	--	--	--	28B	BP	2ISC*PT16D	B22-N067G	2CES*RAK005	--	056
	--	--	--	--	28B	BO	2ISC*PT15C	C72-N050A	2CES*RAK005	--	056
055(3)	K-110	--	--	2ISC*EFV19	28C	AG	2ISC*PT17A	B22-N094A	2CES*RAK004	--	057
	--	--	--	--	28C	AG	2ISC*PT17C	B22-N094E	2CES*RAK004	--	057
	--	--	--	--	28C	AB	2ISC*PT15D	C72-N050D	2CES*RAK004	--	057
056(3)	K-111	--	--	2ISC*EFV16	28B	AG	2ISC*PT15A	C72-N050C	2CES*RAK026	--	054
	--	--	--	--	28B	AP	2ISC*PT16C	B22-N067R	2CES*RAK026	--	054
	--	--	--	--	28B	AP	2ISC*PT16A	B22-N067L	2CES*RAK026	--	054
057(3)	K-112	--	--	2ISC*EFV12	28C	BY	2ISC*PT17B	B22-N094B	2CES*RAK027	--	055
	--	--	--	--	28C	BY	2ISC*PT17D	B22-N094F	2CES*RAK027	--	055
	--	--	--	--	28C	BY	2ISC*PT15B	C72-N050B	2CES*RAK027	--	055
058(3)	K-115	--	--	2RCS*EFV47B	29C	B-	2RCS*FT83B(L)	B35-N011B(L)	2CES*RAK025	059	040
	--	--	--	--	29C	BG	2RCS*FT6B(L)	B35-N024A(L)	2CES*RAK025	059	040
	--	--	--	--	29C	BB	2RCS*FT8B(L)	B35-N024B(L)	2CES*RAK025	059	040
059(3)	K-117	--	--	2RCS*EFV48B	29C	B-	2RCS*FT83B(H)	B35-N011B(H)	2CES*RAK025	058	039
	--	--	--	--	29C	BG	2RCS*FT6B(H)	B35-N024A(H)	2CES*RAK025	058	039
	--	--	--	--	29C	BB	2RCS*FT8B(H)	B35-N024B(H)	2CES*RAK025	058	039
060(3)	K-119	--	--	2RCS*EFV45A	29B	AO	2RCS*FT7A(L)	B35-N014C(L)	2CES*RAK006	061	041
	--	--	--	--	29B	AO	2RCS*FT9A(L)	B35-N014D(L)	2CES*RAK006	061	041
061(3)	K-121	--	--	2RCS*EFV46A	29B	AO	2RCS*FT7A(H)	B35-N014C(H)	2CES*RAK006	060	043
	--	--	--	--	29B	AO	2RCS*FT9A(H)	B35-N014D(H)	2CES*RAK006	060	043
062(3)	K-122	K-122A	--	2CMS*EFV8B	82B	BY	2CMS*LT9B(L)	--	LOCAL	063	064
	--	--	--	--	82B	BY	2CMS*LT11B(L)	--	LOCAL	063	064
	--	--	--	--	82B	BY	2RSS*LT105(L)	--	LOCAL	063	064
---(2)	K-122A	K-125	K-122	--	--	--	--	--	--	--	--
063(3)	K-125	K-122A	--	2CMS*EFV9B	82B	BY	2CMS*LT9B(H)	--	LOCAL	062	065
	--	--	--	--	82B	BY	2CMS*LT11B(H)	--	LOCAL	062	065
	--	--	--	--	82B	BY	2RSS*LT105(H)	--	LOCAL	062	065

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
064(3)	K-129	K-129A	--	2CMS*EFV8A	82B	AG	2CMS*LT9A(L)	--	LOCAL	065	062
	--	--	--	--	82B	AG	2CMS*LT11A(L)	--	LOCAL	065	062
	--	--	--	--	82B	AG	2RSS*LT114(L)	--	LOCAL	065	062
---(2)	K-129A	K-130	K-129	--	--	--	--	--	--	--	--
065(3)	K-130	K-129A	--	2CMS*EFV9A	82B	AG	2CMS*LT9A(H)	--	LOCAL	064	063
	--	--	--	--	82B	AG	2CMS*LT11A(H)	--	LOCAL	064	063
	--	--	--	--	82B	AG	2RSS*LT114(H)	--	LOCAL	064	063
066(3)	K-134	--	--	2IAS*EFV206	19F	ZY	2IAS*PT236	--	LOCAL	--	--
	--	--	--	--	19F	--	2RSS-PT110	--	LOCAL	--	--
067(3)	K-135	--	--	2IAS*EFV205	19F	ZY	2IAS*PT233	--	LOCAL	--	--
	--	--	--	--	19F	--	2RSS-PT111	--	LOCAL	--	--
068(3)	K-138	--	--	2IAS*EFV200	19E	ZG	2IAS*PT231	--	LOCAL	--	--
	--	--	--	--	19E	--	2RSS-PT109	--	LOCAL	--	--
069(2)	R-011	R-285	--	2SWP*V143B	11P	BY	2SWP*FT13B(H)	E12-N007B(H)	2CES*RAK021	070	--
	R-285	--	R-011	--	11P	BY	2SWP*FT201B(H)	--	LOCAL	070	--
070(2)	R-012	R-284	--	2SWP*V144B	11P	BY	2SWP*FT13B(L)	E12-N007B(L)	2CES*RAK021	069	--
	R-284	--	R-012	--	11P	BY	2SWP*FT201B(L)	--	LOCAL	069	--
071(2)	R-015	R-147	--	2CSH*V27	33B	ZP	2CSH*PT105	E22-N051	2CES*RAK024	--	--
	R-147	--	R-015	--	33B	--	2CSH-PI128	--	LOCAL	--	--
072(2)	R-072	--	R-074	--	37B	ZY	2WCS*FT68Y(H)	E31-N035B(H)	2CES*RAK003	073	--
	R-074	R-072	--	2WCS*V210	37B	ZG	2WCS*FT68X(H)	E31-N035A(H)	2CES*RAK002	073	--
073(2)	R-073	--	R-075	--	37B	ZY	2WCS*FT68Y(L)	E31-N035B(L)	2CES*RAK003	072	--
	R-075	R-073	--	2WCS*V211	37B	ZG	2WCS*FT68X(L)	E31-N035A(L)	2CES*RAK002	072	--
074(2)	R-102	R-363	--	2SFC*V110A	38B	AG	2SFC*PT30A	--	2CES*RAK105	--	--
	R-363	--	R-102	--	38B	--	2SFC-PI60A	--	LOCAL	--	--
075(2)	R-177	--	R-259	--	31F	AG	2RHS*PT3A	E12-N063A	LOCAL	--	082
	R-259	R-177	--	2RHS*V107	31F	A-	2RHS*PI50A	E12-R002A	2CES*RAK018	--	082
076(2)	R-183	R-412	--	2ICS*V73	35C	JG	2ICS*FT102(H)	E51-N051(H)	2CES*RAK017	077	--
	--	--	--	--	35C	JG	2ICS*FT101(H)	E51-N003(H)	2CES*RAK017	077	--
	R-412	--	R-183	--	35C	JG	2RSS*FT106(H)	--	LOCAL	077	--
077(2)	R-184	R-413	--	2ICS*V74	35C	JG	2ICS*FT102(L)	E51-N051(L)	2CES*RAK017	076	--
	--	--	--	--	35C	JG	2ICS*FT101(L)	E51-N003(L)	2CES*RAK017	076	--
	R-413	--	R-184	--	35C	JG	2RSS*FT106(L)	--	LOCAL	076	--

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
078(2)	R-254	R-410	--	2SWP*V143A	11C	AG	2SWP*FT13A(H)	E12-N007A(H)	2CES*RAK018	079	--
	R-410	--	R-254	--	11C	AG	2SWP*FT201A(H)	--	LOCAL	079	--
079(2)	R-255	R-411	--	2SWP*V144A	11C	AG	2SWP*FT13A(L)	E12-N007A(L)	2CES*RAK018	078	--
	R-411	--	R-255	--	11C	AG	2SWP*FT201A(L)	--	LOCAL	078	--
080(2)	R-256	R-492	--	2RHS*V56	31C	AG	2RHS*FT14A(H)	E12-N015A(H)	2CES*RAK018	081	084
	--	--	--	--	31C	AG	2RHS*FT86A(H)	E12-N052A(H)	2CES*RAK018	081	084
	R-492	--	R-256	--	31C	AG	2RHS*FT60A(H)	--	LOCAL	081	084
081(2)	R-257	R-493	--	2RHS*V55	31C	AG	2RHS*FT14A(L)	E12-N015A(L)	2CES*RAK018	080	085
	--	--	--	--	31C	AG	2RHS*FT86A(L)	E12-N052A(L)	2CES*RAK018	080	085
	R-493	--	R-257	--	31C	AG	2RHS*FT60A(L)	--	LOCAL	080	085
082(2)	R-270	R-288	--	2RHS*V127	31E	B-	2RHS*PI50B	E12-R002B	2CES*RAK021	--	075
	R-288	--	R-270	--	31E	BY	2RHS*PT3B	E12-N063B	LOCAL	--	075
083(2)	R-271	R-289	--	2RHS*V135	31G	Z-	2RHS*PI50C	E12-R002C	2CES*RAK021	--	--
	R-289	--	R-271	--	31G	ZY	2RHS*PT3C	E12-N095	LOCAL	--	--
084(2)	R-276	R-287	--	2RHS*V83	31B	BY	2RHS*FT14B(H)	E12-N015B(H)	2CES*RAK021	085	080
	--	--	--	--	31B	BY	2RHS*FT86B(H)	E12-N052B(H)	2CES*RAK021	085	080
	R-287	--	R-276	--	31B	BY	2RHS*FT60B(H)	--	LOCAL	085	080
085(2)	R-277	R-286	--	2RHS*V84	31B	BY	2RHS*FT14B(L)	E12-N015B(L)	2CES*RAK021	084	081
	--	--	--	--	31B	BY	2RHS*FT86B(L)	E12-N052B(L)	2CES*RAK021	084	081
	R-286	--	R-277	--	31B	BY	2RHS*FT60B(L)	--	LOCAL	084	081
086(3)	R-017	--	--	2CSH*V18	33B	ZP	2CSH*PT102	E22-N052	2CES*RAK024	--	--
	--	--	--	--	33B	Z-	2CSH*PI103	E22-R001	2CES*RAK024	--	--
087(3)	R-019	--	--	2CSH*V25	33B	ZP	2CSH*FT104(L)	E22-N005(L)	2CES*RAK024	088	--
	--	--	--	--	33B	ZP	2CSH*FT105(L)	E22-N056(L)	2CES*RAK024	088	--
088(3)	R-020	--	--	2CSH*V26	33B	ZP	2CSH*FT104(H)	E22-N005(H)	2CES*RAK024	087	--
	--	--	--	--	33B	ZP	2CSH*FT105(H)	E22-N056(H)	2CES*RAK024	087	--
089(3)	R-171	--	--	2ICS*V90	35B	JG	2ICS*PT2A	E51-N055A	2CES*RAK017	--	090
	--	--	--	--	35B	JG	2ICS*PT2C	E51-N055E	2CES*RAK017	--	090
090(3)	R-172	--	--	2ICS*V91	35B	JY	2ICS*PT2B	E51-N055B	2CES*RAK029	--	089
	--	--	--	--	35B	JY	2ICS*PT2D	E51-N055F	2CES*RAK029	--	089

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
091(3)	R-176	--	--	2ICS*V71	35C	JG	2ICS*PT104	E51-N050	2CES*RAK017	--	--
	--	--	--	--	35C	J-	2ICS*PI138	E51-R001	2CES*RAK017	--	--
092(3)	R-178	--	--	2ICS*V85	35D	JG	2ICS*PT105	E51-N053	2CES*RAK017	--	--
	--	--	--	--	35D	JG	2ICS*PT106	E51-N052	2CES*RAK017	--	--
	--	--	--	--	35D	J-	2ICS*PI141	E51-R002	2CES*RAK017	--	--
095(3)	R-249	--	--	2CSL*V20	32A	J-	2CSL*PI103	E21-R002	2CES*RAK001	--	--
	--	--	--	--	32A	JG	2CSL*PT109	E21-N052	2CES*RAK001	--	--
	--	--	--	--	32A	JG	2CSL*PT110	E21-N053	2CES*RAK001	--	--
096(3)	R-251	--	--	2CSL*V23	32A	JG	2CSL*FT126(H)	E21-N003(H)	2CES*RAK001	097	--
	--	--	--	--	32A	JG	2CSL*FT107(H)	E21-N051(H)	2CES*RAK001	097	--
097(3)	R-252	--	--	2CSL*V22	32A	JG	2CSL*FT126(L)	E21-N003(L)	2CES*RAK001	096	--
	--	--	--	--	32A	JG	2CSL*FT107(L)	E21-N051(L)	2CES*RAK001	096	--
098(3)	R-272	--	--	2RHS*V128	31E	BY	2RHS*PT6B	E12-N056B	2CES*RAK021	--	131
	--	--	--	--	31E	B-	2RHS*PI87B	E12-R008B	2CES*RAK021	--	131
	--	--	--	--	31E	BY	2RHS*PT5B	E12-N055B	2CES*RAK021	--	131
099(3)	R-273	--	--	2RHS*V136	31G	ZY	2RHS*PT6C	E12-N056C	2CES*RAK021	--	--
	--	--	--	--	31G	Z-	2RHS*PI87C	E12-R008C	2CES*RAK021	--	--
	--	--	--	--	31G	ZY	2RHS*PT5C	E12-N055C	2CES*RAK021	--	--
100(3)	R-278	--	--	2RHS*V86	31B	ZY	2RHS*FT14C(H)	E12-N015C(H)	2CES*RAK021	101	--
	--	--	--	--	31B	ZY	2RHS*FT86C(H)	E12-N052C(H)	2CES*RAK021	101	--
101(3)	R-279	--	--	2RHS*V85	31B	ZY	2RHS*FT14C(L)	E12-N015C(L)	2CES*RAK021	100	--
	--	--	--	--	31B	ZY	2RHS*FT86C(L)	E12-N052C(L)	2CES*RAK021	100	--
102(3)	R-269	--	--	2RHS*V260	31G	XY	2RHS*PT133	E12-N062	LOCAL	--	--
	--	--	--	--	31G	X-	2RHS-PI132	E12-R696	LOCAL	--	--
103(3)	2CNM*PT46A		--	2CNM*V2A	3A	AG	2CNM*PT46A	B22-N075A	LOCAL (T)	--	--
	*PT46B										
	--	--	--	--	3A	BY	2CNM*PT46B	B22-N075B	LOCAL (T)	--	--
104(3)	2CNM*PT46C		--	2CNM*V2B	3A	CO	2CNM*PT46C	B22-N075C	LOCAL (T)	--	--
	*PT46D										
	--	--	--	--	3A	DB	2CNM*PT46D	B22-N075D	LOCAL (T)	--	--
105(3)	2MSS-PT96B			2MSS*V93							
		2MSS*PT16C,D		--	1G	CO	2MSS*PT16C	C72-N052C	2CES*RAK232(T)	--	106
				--	1G	DB	2MSS*PT16D	C72-N052D	2CES*RAK232(T)	--	106
		2MSS-PT96B		--	1G	--	2MSS-PT96B	C12-N054B	LOCAL (T)	--	--

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
106(3)	2MSS-PT96A			2MSS*V154							
		2MSS*PT16A,B		--	1G	AG	2MSS*PT16A	C72-N052A	2CES*RAK231(T)	--	105
				--	1G	BY	2MSS*PT16B	C72-N052B	2CES*RAK231(T)	--	105
		2MSS-PT96A		--	1G	--	2MSS-PT96A	C12-N054A	LOCAL (T)	--	--
107	2SWP*FT96A(H)			2SWP*V936A	11B	AG	2SWP*FT96A(H)	--	LOCAL (SW)	108	--
	*FT200A(H)			--	11B	AG	2SWP*FT200A(H)	--	LOCAL (SW)	108	--
108	2SWP*FT96A(L)			2SWP*V937A	11B	AG	2SWP*FT96A(L)	--	LOCAL (SW)	107	--
	*FT200A(L)			--	11B	AG	2SWP*FT200A(L)	--	LOCAL (SW)	107	--
109	2SWP*FT96B(H)			2SWP*V936B	11A	BY	2SWP*FT96B(H)	--	LOCAL (SW)	110	--
	*FT200B(H)			--	11A	BY	2SWP*FT200B(H)	--	LOCAL (SW)	110	--
110	2SWP*FT96B(L)			2SWP*V937B	11A	BY	2SWP*FT96B(L)	--	LOCAL (SW)	109	--
	*FT200B(L)			--	11A	BY	2SWP*FT200B(L)	--	LOCAL (SW)	109	--
111	2SWP*FT96C(H)			2SWP*V936C	11A	AG	2SWP*FT96C(H)	--	LOCAL (SW)	112	--
	*FT200C(H)			--	11A	AG	2SWP*FT200C(H)	--	LOCAL (SW)	112	--
112	2SWP*FT96C(L)			2SWP*V937C	11A	AG	2SWP*FT96C(L)	--	LOCAL (SW)	111	--
	*FT200C(L)			--	11A	AG	2SWP*FT200C(L)	--	LOCAL (SW)	111	--
113	2SWP*FT96D(H)			2SWP*V936D	11A	BY	2SWP*FT96D(H)	--	LOCAL (SW)	114	--
	*FT200D(H)			--	11A	BY	2SWP*FT200D(H)	--	LOCAL (SW)	114	--
114	2SWP*FT96D(L)			2SWP*V937D	11A	BY	2SWP*FT96D(L)	--	LOCAL (SW)	113	--
	*FT200D(L)			--	11A	BY	2SWP*FT200D(L)	--	LOCAL (SW)	113	--
115	2SWP*FT96E(H)			2SWP*V936E	11B	AG	2SWP*FT96E(H)	--	LOCAL (SW)	116	--
	*FT200E(H)			--	11B	AG	2SWP*FT200E(H)	--	LOCAL (SW)	116	--
116	2SWP*FT96E(L)			2SWP*V937E	11B	AG	2SWP*FT96E(L)	--	LOCAL (SW)	115	--
	*FT200E(L)			--	11B	AG	2SWP*FT200E(L)	--	LOCAL (SW)	115	--
117	2SWP*FT96F(H)			2SWP*V936F	11A	BY	2SWP*FT96F(H)	--	LOCAL (SW)	118	--
	*FT200F(H)			--	11A	BY	2SWP*FT200F(H)	--	LOCAL (SW)	118	--
118	2SWP*FT96F(L)			2SWP*V937F	11A	BY	2SWP*FT96F(L)	--	LOCAL (SW)	117	--
	*FT200F(L)			--	11A	BY	2SWP*FT200F(L)	--	LOCAL (SW)	117	--

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees To	Line Tees From	Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No. SWEC	Instrument Tag No. GE	Location	Remarks (5)	Remarks (6)
119	2EGA*PS20A			2EGA*V53A	104A	A-	2EGA*PS20A	--	LOCAL (DG)	--	--
	*PS22A			--	104A	A-	2EGA*PS22A	--	LOCAL (DG)	--	--
	PI12A			--	104A	--	2EGA-PI12A	--	LOCAL (DG)	--	--
120	2EGA*PS19A			2EGA*V53B	104A	C-	2EGA*PS19A	--	LOCAL (DG)	--	--
	*PS21A			--	104A	C-	2EGA*PS21A	--	LOCAL (DG)	--	--
	-PI11A			--	104A	--	2EGA-PI11A	--	LOCAL (DG)	--	--
121	2EGA*PS20B			2EGA*V54A	104A	B-	2EGA*PS20B	--	LOCAL (DG)	--	--
	*PS22B			--	104A	B-	2EGA*PS22B	--	LOCAL (DG)	--	--
	PI12B			--	104A	--	2EGA-PI12B	--	LOCAL (DG)	--	--
122	2EGA*PS19B			2EGA*V54B	104A	D-	2EGA*PS19B	--	LOCAL (DG)	--	--
	*PS21B			--	104A	D-	2EGA*PS21B	--	LOCAL (DG)	--	--
	-PI11B			--	104A	--	2EGA-PI11B	--	LOCAL (DG)	--	--
123	2EGA*PS110			2EGA*V36A	104A	J-	2EGA*PS110	--	LOCAL (DG)	--	--
	*PS109			--	104A	J-	2EGA*PS109	--	LOCAL (DG)	--	--
124	2EGA*PS106			2EGA*V36B	104A	J-	2EGA*PS106	--	LOCAL (DG)	--	--
	*PS117			--	104A	J-	2EGA*PS117	--	LOCAL (DG)	--	--
125	2EGA*PS120			2EGA*V55A	104A	J-	2EGA*PS120	--	LOCAL (DG)	--	--
	*PS122			--	104A	J-	2EGA*PS122	--	LOCAL (DG)	--	--
	-PI124			--	104A	--	2EGA-PI124	--	LOCAL (DG)	--	--
126	2EGA*PS119			2EGA*V55B	104A	J-	2EGA*PS119	--	LOCAL (DG)	--	--
	*PS121			--	104A	J-	2EGA*PS121	--	LOCAL (DG)	--	--
	-PI119			--	104A	--	2EGA-PI119	--	LOCAL (DG)	--	--
127	R-122	R-293	--	2WCS*V165	37C	ZG	2WCS*FT69X(H)	E31-N015A	2CES*RAK002	128	--
	R-293	--	R-122	--	37C	ZY	2WCS*FT69Y(H)	E31-N015B	2CES*RAK002	128	--
128	R-123	R-294	--	2WCS*V166	37C	ZG	2WCS*FT69X(L)	E31-N015A	2CES*RAK002	127	--
	R-294	--	R-123	--	37C	ZY	2WCS*FT69Y(L)	E31-N015B	2CES*RAK002	127	--
129	R-250	R-565	--	2CSL*V26	32C	J-	2CSL*PI111	E21-R001	2CES*RAK001	--	--
	R-565	--	R-250	--	32C	JG	2CSL*PT130	E21-N057	LOCAL	--	--
130	K-097	R-562	--	2CSL*EFV1	31A	AG	2RHS*PDT18A(L)	E12-N060A	2CES*RAK018	--	--
	R-562	--	K-097	--	32A	JG	2CSL*PDT132(L)	E21-N050	2CES*RAK001	--	--

TABLE 7.2-2 (Cont'd.)

Group # (4)	Line Ref. (1)	Line Tees		Source Valve (S/V)	PID[7]	Line Code	Instrument Tag No.		Location	Remarks	
		To	From				SWEC	GE		(5)	(6)
131	R-258	--		2RHS*V109 -- --	31F 31F 31F	AG AG A-	2RHS*PT5A 2RHS*PT6A 2RHS*PI87A	E12-N055A E12-N056A E12-R008A	2CES*RAK018 2CES*RAK018 2CES*RAK018	-- -- --	098 098 098
132	R-110	--	--	2SFC*V110B --	38A 38A	BY --	2SFC*PT30B 2SFC-PI60B	-- --	LOCAL LOCAL	-- --	-- --
133(3)	R-182	--	--	2ICS*V78 --	35C 35C	JG J-	2ICS*PT103 2ICS*PI139	E51-N007 E51-R003	2CES*RAK017 2CES*RAK017	-- --	-- --
134(3)	K-142	--	--	2ICS*EFV5 --	35C 35C	J- J-	2ICS*PT142 2ICS*PT143	-- --	LOCAL LOCAL	-- --	-- --

GENERAL NOTES:

A. All instruments are located in the reactor building unless otherwise noted.
B. Only lines in the reactor building are assigned a line reference number.
C. (T) - Turbine Building Location
D. (DG) - Diesel Generator Building
E. (SW) - Service Water Pump Room
F. In some instances only an instrument's high or low side may be indicated. The instrument's other leg does not tie into another instrument and so is not indicated here.

(1) Line reference numbers with a "K" prefix represent lines that originate in the primary containment. Line reference numbers with an "R" prefix represent lines that originate in the secondary containment. In the absence of a "K" or "R" prefix the line is tagged with the instrument numbers being supplied.
(2) Those lines which tee somewhere between source and instrument to service instruments (local or rack mounted) in different areas of the building with a common source.
(3) Those lines which tee inside the rack (close couple in case of locally mounted instruments) to service more than one instrument.
(4) The Group Reference number was generated for this table only; it is used to allow cross-referencing of related lines (see 5 and 6).
(5) A number in this column refers you to the high press side or low press side by group number for differential type instruments.
(6) A number in this column refers you to the redundant instrument train by group number.
(7) Also see EK-401 series drawings.

7.3 ENGINEERED SAFETY FEATURE SYSTEMS

7.3.1 Description

Section 7.3 describes the instrumentation and controls of the following plant ESF systems:

1. Emergency core cooling systems (ECCS).
2. Primary containment and reactor vessel isolation control systems (PCRVICS).
3. RHR/containment spray cooling mode (RCSCM).
4. RHR/suppression pool cooling mode (RSPCM).
5. Standby gas treatment system (SGTS).
6. Combustible gas control system (CGCS).
7. Portion of reactor building HVAC system.
8. Portion of service water system.
9. Service water pump bays ventilation system.
10. Control building HVAC system.
11. Control building chilled water system.
12. Standby power system.
13. Diesel generator building HVAC system.

The sources that supply power to the ESF systems originate from onsite ac and/or dc safety-related buses or, as in the case of the PCRVICS fail-safe logic, from the nonsafety-related uninterruptible power supply (UPS) bus. Refer to Chapter 8 for a complete discussion of the ESF systems power sources.

7.3.1.1 System Descriptions

7.3.1.1.1 Emergency Core Cooling Systems - Instrumentation and Controls

The ECCS is a network of the following subsystems:

1. High-pressure core spray (HPCS) system.

2. Automatic depressurization system (ADS).

3. Low-pressure core spray (LPCS) system.

4. Low-pressure coolant injection (LPCI) mode of the residual heat removal (RHR) system.

The purpose of the ECCS network (Figure 7.3-1) is to protect the reactor core against fuel cladding damage in the unlikely event of a loss-of-coolant accident (LOCA). Protection is provided for any primary steam or liquid line break, up to and including the double-ended break of the largest line (Sections 6.3.1 and 6.3.2). The ECCS instrumentation detects a need for core cooling systems operation, and the trip systems initiate the appropriate response.

The response time limits for the ECCS systems are shown in TRM Section 3.5.1.

Included in this section is a discussion of protective considerations that are taken between the high-pressure reactor coolant system (RCS) and the low-pressure ECCS system. The high-pressure/low-pressure interlocks are discussed in Section 7.6.1.2.

The following plant variables are monitored and provide automatic initiation of the ECCS when these variables exceed predetermined limits with the exception of the ADS, which does not monitor and provide automatic initiation on high drywell pressure.

1. <u>Reactor Vessel Water Level</u> A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the RCPB, and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes. Refer to Figure 5.1-2 for a schematic arrangement of reactor vessel instrumentation.

2. <u>Drywell Pressure</u> High pressure in the drywell could indicate a breach of the RCPB inside the drywell and that the core is in danger of becoming overheated as reactor coolant inventory diminishes.

7.3.1.1.1.1 High-Pressure Core Spray System - Instrumentation and Controls

System Function

The HPCS system supplies makeup water to the reactor core in the event of a LOCA or reactor isolation and failure of the RCIC system. The HPCS system is capable of starting and delivering rated flow into the vessel within 27 sec following receipt of the initiation signal. Start time includes standby power source start time, and the time required to reach operating voltage.

System Operation

Schematic arrangements of system mechanical equipment are shown on Figure 6.3-1. HPCS system component control logic is shown on Figure 7.3-2 and 7.3-3. Instrument specifications are listed in Table 7.3-1 and Technical Specifications. Operator information displays are shown on Figures 7.3-2 and 7.3-3.

The HPCS is initiated automatically by reactor vessel low water level (Trip Level 2) and/or drywell high pressure. The system is designed to operate automatically with no action required by the Control Room Operator. Once initiated, the HPCS logic seals in and can be reset by the Operator if reactor water level has been restored. The HPCS pump can then be stopped and the injection valves closed. The HPCS actuation logic is further discussed in Section 1.10, Task II.K.3.21.

The reactor vessel water level (Trip Level 2) is monitored by four redundant level transmitters, each providing an input to a trip unit. The associated trip unit relay contacts are connected in a one-out-of-two-twice logic arrangement to assure that no single failure can prevent the initiation of the HPCS. Initiation diversity is provided by drywell pressure which is monitored by four redundant pressure transmitters. The associated trip unit relay contacts are electrically connected in a one-out-of-two-twice logic arrangement to assure that no single instrument failure can prevent the initiation of the HPCS.

The HPCS components respond to an automatic initiation signal as follows (actions are simultaneous unless stated otherwise):

1. The HPCS diesel generator is signaled to start and its protective relays are bypassed, with the exception of engine overspeed and generator differential. Once the diesel generator is started, it signals to open

service water valving, thus providing cooling water to the HPCS diesel generator heat exchanger (Section 6.3).

2. The HPCS pump motor is signaled to start.

3. The normally open pump suction valve from the condensate storage tank (CST) valve MO F001 (MOV101) is signaled to open.

4. Test return valves MO F010 (MOV110), F011 (MOV112), and F023 (MOV111) are signaled to close.

5. The HPCS injection valve MO F004 (MOV107) is signaled to open.

The HPCS pump discharge flow and pressure are monitored by a flow transmitter, a pressure transmitter, and associated trip units. If pump discharge pressure is normal but discharge flow is low enough that pump overheating may occur, the minimum flow return line valve MO F012 (MOV105) is signaled to open. The valve is automatically closed if flow is normal.

If the water level in the CSTs falls below a low level, as determined by low pressure at the pump suction, the suppression pool suction valve MO F015 (MOV118) automatically opens. When the suppression pool suction valve (MO F015) is fully open, the CST suction valve (MO F001) automatically closes. Two pressure transmitters are used to detect low pressure at the pump suction. A time delay is provided to prevent inadvertent switchover on startup or other transients. Either transmitter can cause automatic suction transfer. The suppression pool suction valve also automatically opens if high water level is detected in the suppression pool. Two level transmitters monitor suppression pool water level and either transmitter can initiate opening of the suppression pool suction valve. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other closes.

The HPCS provides makeup water to the reactor until the vessel water level reaches the high level trip (Trip Level 8) at which time the HPCS injection valve MO F004 (MOV107) is automatically closed. The pump will continue to run on minimum flow recirculation. The injection valve will automatically reopen if vessel level again drops to the low level (Trip Level 2) initiation point.

The HPCS pump motor and injection valve can be manually overridden to permit the Reactor Operator to manually control the system following automatic initiation.

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.1.2 Automatic Depressurization System - Instrumentation and Controls

System Function

The ADS is designed to provide automatic depressurization of the reactor vessel by activating seven SRVs. These valves vent steam to the suppression pool in the event that the HPCS cannot maintain the reactor water level following a LOCA. The ADS reduces the reactor pressure so that flow from the low-pressure ECCS, LPCI system, and LPCS can inject into the reactor vessel in time to cool the core and limit fuel cladding temperature.

System Operation

Schematic arrangements of system mechanical equipment are shown on Figure 10.1-3. ADS component control logic is shown on Figure 7.3-4. Instrumentation specifications are listed in Table 7.3-2 and Technical Specifications. Operator information displays are shown on Figures 5.1-2, 10.1-3, and 7.3-4.

To prevent inadvertent actuation of the ADS, two channels of logic for each ADS trip system (A and B) are used. Both channels must be activated to actuate an ADS trip system. Refer to Figure 7.3-4 for a schematic representation of the ADS initiation logic.

Each channel contains a single input from a reactor vessel low level (trip level 1) sensor. In addition, one channel includes a pressure sensor input monitoring reactor vessel low water level trip (trip level 3). The second low water level trip (trip level 3) provides confirmation of a reactor vessel low water level condition.

To assure that adequate makeup water is available after the vessel has been depressurized, each trip system includes a pump discharge pressure permissive signal indicating a LPCI or LPCS pump is available for vessel water makeup. Any one of the three LPCI pumps or the LPCS pump is sufficient to permit automatic depressurization.

After receipt of the initiation signals and after a delay provided by timers, each of the two solenoid pilot air valves are energized. This allows pneumatic pressure from the accumulator to act on the air cylinder operator. Each ADS trip system timer can be reset manually to delay system initiation, or each trip system timer can be disabled to prevent depressurization from occurring automatically. If reactor vessel water level is restored by the HPCS prior to the end of the time delay, ADS initiation will be prevented.

The ADS trip system A actuates the A solenoid pilot valve on each ADS relief valve. Similarly, the ADS trip system B actuates the B solenoid pilot valve on each ADS relief valve. Actuation of either solenoid pilot valve causes the associated ADS valves to open to provide depressurization.

Once initiated, the ADS logic seals in and can be reset by the Control Room Operator only when the vessel water level returns to normal. The ADS actuation logic is further discussed in Section 1.10, Task II.K.3.18. The ADS system automatic initiation disable switches (one for each trip system) are located in the main control room. Operability requirements for the disabled switches are described in TRM Section 3.3.5.

The control switches (one for each trip system solenoid) are located in the main control room for each SRV associated with the ADS. Each switch controls one of the two solenoid pilot valves.

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.1.3 Low-Pressure Core Spray - Instrumentation and Controls

System Function

The purpose of the LPCS is to provide low-pressure reactor vessel core spray following a LOCA when the vessel has been depressurized and vessel water level has not been restored by the HPCS. The LPCS is functionally diverse from the LPCI mode of the RHR system.

System Operation

Schematic arrangements of system mechanical equipment are shown on Figure 6.3-7a. LPCS components control logic is shown on Figure 7.3-5.

The LPCS is initiated automatically by reactor vessel low water level and/or high drywell pressure. The system is designed to operate automatically for at least 10 min without any action required by the Control Room Operator. Once initiated, the LPCS logic seals in and can be reset by the Control Room Operator only when the water level and drywell pressure return to normal. Refer to Figure 7.3-5 for a schematic representation of the LPCS system initiation logic.

Reactor vessel water level (Trip Level 1) is monitored by two redundant level transmitters. Drywell pressure is monitored by two redundant pressure transmitters. The vessel level trip unit relay contacts and the drywell pressure trip unit relay contacts are connected in a one-out-of-two-twice logic arrangement so that no single instrument failure can prevent initiation of the LPCS.

The LPCS components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows:

1. The Division I diesel generator is signaled to start.

2. The normally closed test return line to the suppression pool valve MO F055 (FV114) is signaled closed.

3. When power (offsite or onsite) is available at the LPCS pump motor bus, the LPCS pump is signaled to start. If offsite power is available, the LPCS pump starts after a 10-sec delay. If offsite power is not available and the Division I diesel generator is providing power, the LPCS pump starts after a 6-sec delay.

4. A differential pressure transmitter senses the pressure differential between the low pressure side of LPCS injection valve MO F005 (MOV104) and reactor vessel pressure. When the pressure differential is

low enough to protect the LPCS from overpressure and power is available to the pump motor bus, the injection valve is signaled to open.

The LPCS pump discharge flow is monitored by a differential pressure transmitter. When the pump is running and discharge flow is low enough to cause pump overheating, the minimum flow return line valve MO F011 (MOV107) is opened. The valve is automatically closed if flow is normal.

The LPCS pump suction from the suppression pool valve MO F074 (MOV112) is normally open, and the control switch is keylocked in the open position and thus requires no automatic open signal for system initiation.

The LPCS pump and injection valve have manual override controls that permit the Operator to manually control the system subsequent to automatic initiation.

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.1.4 RHR Low-Pressure Coolant Injection Mode - Instrumentation and Controls

System Function

The LPCI is an operating mode of the RHR system. The purpose of the LPCI mode is to provide low-pressure reactor vessel coolant makeup following a LOCA when the vessel has been depressurized and vessel water level is not maintained by the HPCS.

System Operation

Schematic arrangements of system mechanical equipment and instrument locations are shown on Figure 5.4-13. LPCI component control logic is shown on Figure 7.3-6. Instrument specifications are listed in Table 7.3-4 and Technical Specifications. Operator information displays are shown on Figures 5.4-13 and 7.3-6.

The LPCI system is initiated automatically by reactor vessel low water level and/or by high drywell pressure. The system is designed to operate automatically for at least 10 min without any action required by the Control Room Operator. Once initiated, the LPCI logic seals in and can be reset by the Control Room Operator when initial conditions return to normal. Refer to Figures 5.4-13 and 7.3-6 for a schematic representation of the LPCI A and the LPCI B/C initiation logic, respectively.

Reactor vessel water level (Trip Level 1) is monitored by two redundant differential pressure transmitters. To provide diversity, drywell pressure is monitored by two redundant pressure transmitters.

To initiate the Division II LPCI (Loops B and C), the vessel level transmitter contacts and the two drywell pressure transmitter contacts are connected in a one-out-of-two-twice arrangement so that no single instrument failure can prevent initiation of LPCI.

The Division I LPCI (Loop A) receives its initiation signal from the LPCS logic. The LPCI system components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows (the Loop A components are controlled from the Division I logic; the Loop B and C components are controlled from the Division II logic):

1. The Division I and II diesel generators are signaled to start.

2. If offsite power is available at the pump motor buses, the LPCI pumps A and B start after a 5-sec time delay; LPCI pump C and the LPCS pump start after a 10-sec time delay. If offsite power is not available and diesel generators are providing power to the pump motor buses, sequential loading of the diesel generators is required. This is accomplished by starting LPCI pumps A and B after a 1-sec time delay; LPCI pump C and the LPCS pump start after a 6-sec time delay.

3. Differential pressure transmitters monitor the pressure difference between the low pressure side of each LPCI injection valve MO F042A (MOV24A), F042B (MOV24B), F042C (MOV24C) and reactor pressure. When the differential is low enough and power is available at the associated pump motor bus, the injection valve is signaled to open.

4. The following normally closed valves are signaled closed to ensure proper system lineup:

a. Test return line to the suppression pool valves MO F024A (FV38A), F024B (FV38B) and F021 (FV38C).

b. Containment spray to suppression pool valves MO F027A (MOV33A), F027B (MOV33B).

5. The normally open heat exchanger bypass valves MO F048A (MOV8A), F048B (MOV8B) are signaled to open. The open signal is automatically removed 10 min after system initiation to allow the Operator to close the valve and initiate use of the heat exchanger.

In addition, RHR sample valves F060A, B (SOV36A, B) and F075A, B (SOV35A, B) receive an isolation signal from the nuclear steam supply shutdown system (NSSS).

Each LPCI pump discharge flow is monitored by a differential pressure transmitter which, when the pump is running and following an 8-sec time delay, opens the minimum flow return line valves MO F064A (MOV4A), F064B (MOV4B), F064C (MOV4C) if flow is low enough that pump overheating may occur. The valve is automatically closed if flow is normal.

The three RHR pump suction lines from the suppression pool valves MO F004A (MOV1A), F004B (MOV1B), F004C (MOV1C) and the RHR heat exchanger shell side inlet and outlet valves MO F047A (MOV9A), F047B (MOV9B) and F003A (MOV12A), F003B (MOV12B) have control switches keylocked in the open position, and thus require no automatic open signal for system initiation.

The two series service water crosstie valves MO F093 (MOV115) and MO F094 (MOV116) have control switches keylocked in the closed position and thus require no automatic close signal for system initiation.

The two series containment spray valves MO F016A (MOV15A), F016B (MOV15B), and F017A (MOV25A); F017B (MOV25B); the two series RHR heat exchanger vent valves MO F073A (MOV27A), F073B (MOV27B) and F074A (MOV26A), F074B (MOV26B); and the RHR shutdown cooling mode suction valves MO F006A (MOV2A), F006B (MOV2B) are all normally closed and thus require no automatic close signal for system initiation.

The LPCI pump motors and injection valves have manual override controls that permit the Operator to manually control the system subsequent to automatic initiation.

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.2 Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) - Instrumentation and Controls

System Function

The PCRVICS provides the means to automatically isolate the primary containment and/or reactor vessel by closing the inboard and outboard isolation valves of the main steam lines and of the process lines of other systems. Isolation of these systems prevents uncovering of the reactor core and limits the release of radioactive materials from other systems that may incur leaks or breaks. Leaks are detected by monitoring and sensing high temperatures, abnormal pressures, abnormal flow rates, and low water levels.

The PCRVICS encompasses sensors, instrumentation, and trip logic from the LDS, PRMS, and systems that provide sensing or require isolation. See Section 6.2 for a complete description of primary containment and reactor vessel process lines and isolation signals applied to each. The PCRVICS logic is further discussed in Section 1.10, Task II.E.4.2.

System Operation

Schematic mechanical arrangements of containment isolation valves and other components initiated by the PCRVICS are shown on Figures 5.4-2, 5.4-16, 10.1-3, 10.1-6, 5.4-13, and other figures referenced in Table 6.2-56. PCRVICS component control logic is shown on Figures 7.3-4, 7.3-6, and 7.3-7. Instrument specifications are listed in Table 7.3-5. NSSS isolation valve signals are listed in Table 6.2-56. Operator information displays are shown on Figures 7.3-4, 10.1-3, 10.1-6, and 5.1-2.

During normal plant operation, the isolation control system sensors and trip logic relays that are essential to safety are energized. When abnormal conditions are sensed, instrument contacts open and de-energize the trip logic relays and thereby initiate isolation. Once initiated, the PCRVICS trip logics seal in and may be reset by the Operator only when the initial conditions return to normal.

Each MSIV has two control solenoids, each of which receives inputs from two redundant logics. A signal from either logic will de-energize one solenoid. For any one valve to close automatically, both of its solenoids must be de-energized.

The MSIV logic uses four redundant instrument channels for each measured variable. One channel of each variable is connected to one trip logic. One group of redundant logics (A,C) is used to control one solenoid of both inside and outside valves of all four main steam lines, and the other group of redundant logics (B,D) is used to control the other solenoid of both inside and outside valves. The four PCRVICS trip logics are arranged in a one-out-of-two-twice logic combination (Trip Logic A or C and B or D) (Figure 7.3-8).

The main steam line drain valves, reactor water sample valves, reactor water cleanup (RWCU) isolation valves, and RHR system isolation valves also operate in pairs. The valves close if both isolation logics C and B are tripped, and the outside valves close if both logics A and D are tripped. Dual logic trip is required for monitored variables such as water level. However, only single logic trip is required for other monitored variables such as high temperature or reactor high pressure (Figure 7.3-8). Control logic for balance-of-plant (BOP) ESF actuation is shown on Figure 7.3-10.

The PCRVICS also provides signals to start the SGTS, to remove nonessential loads from essential buses, and to isolate the reactor building ventilation system and the primary containment purge and vent system.

The following variables provide inputs to the PCRVICS logics for initiation of reactor vessel and containment isolation, as well as the initiation or trip of other plant functions when predetermined limits are exceeded. Combinations of these variables, as necessary, provide initiation of various isolating and initiating functions as described in Section 6.2 and below.

<u>Reactor Vessel Low Water Level</u> A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the RCPB and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes.

Reactor vessel low water level initiates closure of various valves. The closure of these valves is intended to isolate a breach of the RCPB pipes, conserve reactor coolant by closing off process lines, and limit the escape of radioactive materials from the primary containment through process lines that communicate with the primary coolant boundary or primary containment.

Three reactor vessel low water level isolation trip settings are used to complete the isolation of the primary containment and the reactor vessel. The first (and higher) reactor vessel low water level isolation trip (Trip Level 3) initiates closure of all RHR system isolation valves. The main steam lines are left open to allow the removal of heat from the reactor core. The second (middle) reactor vessel low water level isolation trip (Trip Level 2) initiates closure of the two RWCU isolation (suction) valves and all other isolation valves not closed by Level 1 or Level 3, and also provides inputs to logic for other plant equipment. The third (lowest) reactor vessel low water level isolation trip (Trip Level 1) initiates closure of the MSIVs and the main steam line drain valves.

In each of the four logic divisions, reactor vessel low water levels are monitored by two level transmitters, and associated trip units and logic. One transmitter, with one master trip unit and one slave trip unit, monitors Levels 1 and 2 and is located in the nuclear boiler system. The other transmitter, with one trip unit, monitors Level 3 and is located in the RPS.

Diversity of trip initiation for pipe breaks inside the primary containment is provided by monitoring drywell high pressure.

<u>Drywell High Pressure</u> High pressure in the drywell could indicate breach of the RCPB inside the drywell and that the core is in danger of becoming overheated as reactor coolant inventory diminishes. High drywell pressure is monitored by one set of four redundant pressure transmitters and associated trip units and logic. This set of instruments is located in the RPS and distributed among the four divisions. Relay contact interlocks for high drywell pressure are provided from the RPS. The drywell high-pressure isolation trip initiates closure of isolation valves as shown in Table 6.2-56.

<u>Main Steam Line - Tunnel and Pipe Routing in Turbine Building (i.e., main steam line tunnel lead enclosure, MSLTLE) High Ambient Temperature and Differential Temperature</u> High ambient temperature in the tunnel and pipe routing areas in the turbine building, in which the main steam lines are located outside the primary containment, could indicate a leak in a main steam line. Such a leak might also be indicated by high differential

temperature between the outlet and inlet ventilation air for these areas. The automatic closure of valves prevents the excessive loss of reactor coolant and the release of a significant amount of radioactive material from the RCPB.

Four redundant main steam line high ambient temperature sensors are provided in the main steam tunnel and 12 in the steam line area of the turbine building. Four redundant differential temperature sensors monitor the outlet and inlet ventilation air ducts of the main steam line tunnel. Each main steam line trip isolation logic is de-energized by high ambient temperature in the main steam line tunnel or the turbine building, or by high differential temperature in the tunnel inlet/outlet ventilation air. When a predetermined increase in main steam line tunnel ambient or differential temperature is detected, trip signals initiate closure of all MSIVs and drain valves. Diversity of trip initiation signals for main steam line tunnel (including lead enclosure) ambient temperature and high differential temperature is provided by main steam line high flow, and steam line low-pressure instrumentation.

<u>Main Steam Line - High Flow</u> Main steam line high flow could indicate a breach in a main steam line. Automatic closure of the isolation valve prevents excessive loss of reactor coolant and release of significant amounts of radiomaterial from the RCPB. Four redundant differential pressure transmitters and four associated trip units for each main steam line provide inputs to each of the four trip channels. When a significant increase in main steam line flow is detected, trip signals initiate closure of all MSIVs and drain valves.

<u>Main Turbine Inlet - Low Steam Pressure</u> Low steam pressure at the turbine inlet while the reactor is operating could indicate a malfunction of the nuclear system pressure regulator in which the turbine control valves or turbine bypass valves become fully open, and causes rapid depressurization of the reactor vessel. From reduced power, the rate of decrease of nuclear system saturation temperature could exceed the allowable rate of change of vessel temperature. A rapid depressurization of the reactor vessel while the reactor is near full power could result in undesirable differential pressure across the channels (around some fuel bundles) of sufficient magnitude to cause mechanical deformation of channel walls. Such depressurizations, without adequate preventive action, could require thorough vessel analysis or core inspection prior to returning the reactor to power operation.

Four pressure transmitters and their four respective trip units, one set for each main steam line, monitor main steam line low pressure and each provides an input to one of the four trip channels. When a predetermined decrease in main steam line pressure is detected, the PCRVICS initiates closure of all MSIVS and drain valves. The main steam line low-pressure trip is bypassed by the reactor mode switch in either the SHUTDOWN, REFUEL, or STARTUP modes of reactor operation. In the RUN mode, the low-pressure trip function is operative.

Reactor Water Cleanup System - High Differential Flow High differential flow in the RWCU system could indicate a breach of the RCPB in the cleanup system. The flow at the inlet to the system (suction from recirculation lines) is compared with the flow at the outlets of the system (flow return to feedwater or flow to the main condenser and/or radwaste).

Two redundant differential flow-sensing channels compare the RWCU system inlet-outlet flow. Each of the flow-monitoring sensing channels provides an input to one of the two (inside or outside) logic trip channels. When an increase in RWCU system differential flow is detected, the PCRVICS initiates closure of all RWCU system isolation valves.

Diversity of trip initiation signals for a RWCU system line break is provided by instrumentation for reactor water level, differential flow, and ambient temperature in RWCU equipment areas. The RWCU system high differential flow trip is bypassed by an automatic timing circuit during normal RWCU system surges. This time delay bypass prevents inadvertent system isolations during system operational changes.

Reactor Water Cleanup System - Area High Ambient Temperature High temperature in the equipment room areas of the RWCU system could indicate a breach in the reactor cleanup system. Six ambient temperature sensor/switches (TSS) monitor the RWCU system area (pump rooms and heat exchanger room) temperatures. Three of the six ambient TSSs are associated with each of the two (Division I and II) trip logics. One of each of the three ambient TSSs within a division is assigned to pump room 1, pump room 2, and the heat exchanger room. When a predetermined increase in RWCU system area ambient temperature is detected by any one or more of the three TSSs within Division I, the RWCU outside isolation valve is signaled to close. A similar predetermined temperature increase detected by any one or more of the three Division II TSSs will signal the inside RWCU isolation valve to close. Isolation signals for RWCU are also provided by eight ambient TSSs located in the reactor building pipe chase, utilizing a similar trip logic as the TSSs in the RWCU equipment area. The TSSs located in the reactor building pipe chase also provide an isolation signal to RCIC and RHR.

Reactor Water Cleanup Standby Liquid Control System Actuation
Actuation of the SLCS initiates isolation of the RWCU. The RWCU outside isolation valve closes when SLCS pump A is started; the RWCU inside isolation valve closes when SLCS pump B is started. Both valves also receive isolation signals from the RRCS (see Section 7.2.1.8). Use of both isolation valves in series provided redundancy and eliminates single failure problems.

Reactor Water Cleanup High Temperature at Outlet of Nonregenerative Heat Exchanger A predetermined increase in temperature at the outlet of the nonregenerative heat exchanger will initiate isolation of the RWCU by closing the RWCU outboard isolation valve.

RHR System - Area High Ambient Temperature High temperature in the equipment/pump room areas of the RHR system could indicate a breach in the RHR system. Four ambient TSSs monitor the RHR system area (pump rooms) temperatures. Two of the four ambient TSSs are associated with each of the two (Division I and II) divisional trip logics. Within each division, one ambient TSS is assigned to each of the two RHR pump rooms.

When a predetermined increase in RHR system area ambient is detected by any one of the two TSSs within Division I, the associated RHR Division I isolation valves are signaled to close. A similar predetermined temperature increase detected by any one of the two TSSs within Division II will signal the associated RHR Division II isolation valves to close. Isolation signals for RWCU are also provided by eight ambient TSSs located in the reactor building pipe chase, utilizing a similar trip logic as the TSSs in the RWCU equipment area. The TSSs located in the reactor building pipe chase also provide an isolation signal to RCIC and RHR. Valves are identified in Table 6.2-56.

Main Condenser Vacuum Trip The main condenser low vacuum signal could indicate a leak in the condenser. Four redundant vacuum switches monitor the main condenser vacuum. Each switch provides an input to one of the four trip logics. When a significant decrease in main condenser vacuum is detected, the PCRVICS initiates closure of all MSIVs and drain valves. Main condenser low vacuum trip can be bypassed manually when the turbine stop valve is not fully open and the reactor mode switch is in the SHUTDOWN, REFUEL, or STARTUP position. Placing the mode switch in the RUN position prevents or removes the bypass. The bypass is also prevented or removed when the turbine stop valve is open.

High Reactor Pressure High reactor pressure indicates that the reactor is in operation, calling for isolation of the RHR shutdown cooling subsystem injection and suction lines.

Four redundant pressure transmitters, two for each set of valves, monitor reactor vessel pressure. Each of the pressure-monitoring sensors provides a signal to one of the two (inside or outside) logic trip channels.

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

The response time limits of isolation actuation instrumentation are presented in TRM Section 3.3.6.1.

7.3.1.1.3 RHR Containment Spray Cooling Mode - Instrumentation and Controls

System Function

The RCSCM is an operating mode of the RHR system. It is designed to condense steam and remove airborne fission products in the suppression chamber air volume and/or the drywell atmosphere following a LOCA (Sections 6.2.2.3.1 and 15.6.5).

System Operation

Schematic arrangements of system mechanical equipment are shown on Figure 5.4-13. RHR system component control logic is shown on Figure 7.3-6. Instrument specifications are listed in Table 7.3-6. Operator information displays are shown on Figures 5.4-13 and 7.3-6.

The RCSCM is initiated by the Control Room Operator by diverting LPCI flow to the suppression pool via valves MO F027A (MOV33A) or F027B (MOV33B). The LPCI flow can also be diverted to the drywell via MO F016A (MOV15A) and F017A (MOV25A) or F016B (MOV15B) and F017B (MOV25B).

The following permissive conditions must exist before the Operator can initiate a containment spray cooling loop:

1. The automatic LOCA signal or manual push-button signal that initiated the LPCI must still exist.

2. Drywell high pressure is monitored by two redundant pressure transmitters. One of the two transmitters must indicate high pressure. (Applies only to valves F016A, B and F017A, B.)

3. The Operator must close the LPCI injection valves MO F042A (MOV24A), F042B (MOV24B).

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.4 RHR Suppression Pool Cooling Mode - Instrumentation and Controls

System Function

The RSPCM is an operating mode of the system. It is designed to prevent suppression pool temperature from exceeding predetermined limits following a reactor blowdown of the ADS or SRVs.

System Operation

Schematic arrangements of system mechanical equipment are shown on Figure 5.4-13. Component control logic is shown on Figure 7.3-6. Instrument specifications are listed in Table 7.3-7. Operator information displays are shown on Figures 5.4-13 and 7.3-6.

The RSPCM is initiated by the Control Room Operator either during normal plant operation or following a LOCA, when the suppression pool temperature monitoring system (Section 6.2.1.7) indicates that pool temperature may exceed a predetermined limit.

During normal plant operation the Operator initiates the RSPCM as follows:

1. The RHR pump (A or B) is started. The service water pump is started and the RHR heat exchanger service water discharge valves, MO F068A and F068B, are opened.

2. The RHR test return line valves, MO F024A and F024B, are opened.

3. The RHR heat exchanger inlet and outlet valves, MO F047A, F047B, F003A, and F003B, are keylocked open. The heat exchanger bypass valves, MO F048A and F048B, and the RHR test return line valves, F024A and F024B, are throttled as necessary.

Subsequent to a LOCA the Operator initiates the RSPCM as follows:

1. Once reactor vessel water level has been restored, the LPCI flow must be terminated by closing the LPCI injection valves, MO F042A and/or F042B. Closing the injection valve causes the LOCA initiation logic to be overridden and allows Operator control of the system.

2. The control logic for the RHR test return line valves, MO F024A and F024B, also has LOCA signal override which allows the Operator to open the valve.

3. The RHR heat exchanger inlet and outlet valves, MO F047A, F047B, F003A, and F003B, are keylocked open. The RHR test return line valves, MO F024A and F024B, and, after a 10-min delay, the heat exchanger bypass valves, MO F048A and F048B, are throttled as necessary. A 10-min timer keeps valves MO F048A and F048B open following a LOCA.

<u>Testability</u>

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.5 Standby Gas Treatment System

<u>System Function</u>

The SGTS processes potentially-radioactive exhaust air from the following sources prior to discharging the air to the main plant exhaust duct:

1. Reactor building auxiliary filtration system (reactor vessel head cavity exhaust).

2. Reactor building exhaust from the emergency recirculation air system.

3. Containment/drywell purge exhaust.

System Operation

Schematic arrangement of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 6.5.1. Instrumentation specifications are listed in Table 7.3-8, and the appropriate allowable values are listed in Technical Specifications. The control logic is shown on Figure 6.5-1. Instrument locations are identified on Figures 9.4-8k and 9.4-8L.

Detailed instrumentation requirements for the system are described in Section 6.5.1.5.

7.3.1.1.6 Combustible Gas Control System (Hydrogen Recombiners)

System Function

The CGCS is used to reduce the hydrogen concentration inside the containment to a safe, nonexplosive level. The system consists of two redundant Category I hydrogen recombiners backed up by the nonsafety-related primary containment purge system.

System Operation

Schematic arrangement of the system mechanical equipment and instrumentation and a description of the system design and operation are provided in Section 6.2.5. Instrumentation specifications are listed in Table 7.3-9. The control logic is shown on Figure 6.2-72K. Instrument locations are identified on Figures 6.2-72a and 6.2-72b.

Detailed instrumentation requirements for the system are described in Section 6.2.5.5.

7.3.1.1.7 Reactor Building Heating, Ventilating, and Air Conditioning System

System Function

The reactor building HVAC system includes the drywell cooling subsystem, the primary containment purge subsystem, and the HVAC system for all other areas of the reactor building (Section 9.4.2).

The drywell cooling subsystem is not an ESF system. The primary containment purge subsystem supplies filtered air to the drywell and suppression chamber for purge of radioactive gases and particulates and for ventilation during reactor shutdown and refueling periods. The HVAC system for all other areas of the reactor building provides conditioned air to the various areas of the reactor building, limits the release of radioactivity in conjunction with the SGTS, and maintains a reactor building pressure of 0.25 in W.G. negative with respect to the outside atmosphere.

<u>System Operation</u>

Schematic arrangements of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 9.4.2. The control logic is shown on Figure 9.4-9. Instrument specifications are listed in Table 7.3-10, and the appropriate allowable values are listed in Technical Specifications. Instrument locations are identified on Figure 9.4-8. The detailed instrumentation requirements for the reactor building HVAC system are described in Section 9.4.2.5.

7.3.1.1.8 Service Water System

<u>System Function</u>

The purpose of the SWP system is to provide a reliable source of cooling water for plant auxiliaries that are essential to safe reactor shutdown during and following a design basis accident (LOCA).

<u>System Operation</u>

Schematic arrangement of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 9.2.1. The instrumentation specifications are listed in Table 7.3-11, and the appropriate allowable values are listed in the TRM. The SWP system control logic is shown on Figure 9.2-2.

The detailed instrumentation requirements for the system are described in Sections 9.2.1.5 and 9.2.5.5.

7.3.1.1.9 Service Water Pump Bays Ventilation System

System Function

The purpose of the service water pump bays ventilation system is to provide cooling and ventilation of the service water pumps and motors located in the pump bays.

System Operation

Schematic arrangements of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 9.4.7. The instrumentation specifications are listed in Table 7.3-12. The service water pump bays ventilation system control logic is shown on Figure 9.4-17.

The detailed instrumentation requirements of the service water pump bays ventilation system are described in Section 9.4.7.5.

7.3.1.1.10 Control Building Heating, Ventilating, and Air Conditioning System

System Function

The purpose of the control building HVAC system is to provide an environment suitable for habitation in the main control room and associated areas. A positive pressure is maintained in the main control room and relay room to prevent in-leakage of outside air and air from other areas. The system also has the capability to reduce radioactivity in the outside air intake and to remove smoke from the main control room and other major areas of the control building. It also provides negative pressure in the battery rooms to prevent potential hydrogen buildup.

System Operation

The main control room ventilation system intake air is provided with two radiation monitoring channels per trip system. Control room emergency filtration is initiated when both channels of one trip system are in a trip condition. A detectable inoperable condition or downscale channel produces a high radiation signal (trip condition). A manual trip condition is produced for manually-induced inoperability.

Schematic arrangements of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 9.4. The control logic is shown on Figure 9.4-4. Instrument specifications are listed in Table 7.3-13, and the appropriate allowable values are listed in Technical Specifications. Instrument locations are identified on Figure 9.4-1.

The detailed instrumentation requirements for the control building HVAC system are described in Section 9.4.1.5.

7.3.1.1.11 Control Building Chilled Water System

System Function

The control building chilled water system supplies chilled water to the control building air conditioning units servicing the main control room, the relay room, the remote shutdown room, and the computer room.

System Operation

Schematic arrangement of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 9.4.10. Instrumentation specifications are listed in Table 7.3-14. The control logic is shown on Figure 9.4-20. Instrumentation locations are identified on Figure 9.4-1a.

The detailed instrumentation requirements for the system are described in Section 9.4.10.1.5.

7.3.1.1.12 Standby Power System

System Function

The standby power system provides a self-contained, independent source of ac electrical power that is capable of supplying sufficient power for those electrical loads required for safe shutdown. The standby power system consists of three diesel generators and associated auxiliary systems as described in Section 8.3 and Sections 9.5.4 through 9.5.8.

System Operation

Schematic arrangements of system mechanical equipment and instrumentation and a description of system design and operation are provided in Sections 9.5.4 through 9.5.8. The instrumentation specifications for the fuel oil storage and transfer system are listed in Table 7.3-15. The emergency diesel generator protection and control logic is shown on Figure 9.5-41. The standby station service control logic is shown on Figure 8.3-5. Instrument locations are identified on Figure 9.5-40.

7.3.1.1.13 Diesel Generator Building Heating, Ventilating and Air Conditioning System

System Function

The purpose of the diesel generator building HVAC system is to provide cooling and ventilation of the building general areas and the diesel generator control rooms; limit the maximum ambient temperature when the diesel generators are running; and provide normal ventilation for removal of combustible fumes.

System Operation

Schematic arrangements of system mechanical equipment and instrumentation and a description of system design and operation are provided in Section 9.4.6. The instrumentation specifications are listed in Table 7.3-16. The control logic is shown on Figure 9.4-16. Instrument locations are identified on Figure 9.4-15a.

The detailed instrumentation requirements for the system are described in Section 9.4.6.5.

7.3.1.2 Design Basis

The ESF systems are designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Chapter 15 and Appendix A identify and evaluate events that jeopardize the fuel barrier and RCPB. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are also presented in Chapter 15. See Section 7.2.1.4.7 for minimum performance requirements for RPS instrumentation and controls.

7.3.1.2.1 Variables Monitored to Provide Protective Action

The following variables are monitored to initiate protective actions by the ESF and supporting systems:

1. HPCS:

    a. Reactor vessel low water level (trip level 2).

    b. High drywell pressure.

2. ADS:

    a. Reactor vessel low water level (trip level 3).

    b. Reactor vessel low water level (trip level 1).

3. LPCS and LPCI:

    a. Reactor vessel low water level (trip level 1).

    b. High drywell pressure.

4. PCRVICS:

    a. Reactor vessel low water level (trip level 3).

    b. Reactor vessel low water level (trip level 2).

    c. Reactor vessel low water level (trip level 1).

    d. Deleted.

    e. Main steam line tunnel high ambient and differential temperature.

    f. Main steam line high flow.

    g. Turbine inlet low steam pressure.

    h. RWCU system high differential flow.

    i. RWCU system equipment area high ambient temperature.

    j. RHR area high ambient temperature.

    k. Main condenser low vacuum trip.

l. High drywell pressure.

m. Reactor vessel high pressure.

n. Main steam line (turbine building area) high ambient temperature.

o. SLCS actuated.

p. RWCU system nonregenerative heat exchanger high outlet temperature.

q. Reactor building pipe chase ambient temperature.

r. Reactor building ambient temperature.

s. Containment purge system exhaust (via SGTS) high radiation.

5. RHR containment spray cooling mode: High drywell pressure.

6. RHR suppression pool cooling mode:

   a. Suppression pool temperature.

   b. High drywell pressure.

   c. Reactor vessel low water level (trip level 1).

7. ESF support systems: The variables monitored for ESF support systems are discussed in sections referenced from 7.3.1.1.5 through 7.3.1.1.13.

The plant conditions that require protective action involving the ESF systems are described in Chapter 15 and Appendix 15A.

7.3.1.2.2 Location and Number of Sensors

Table 7.3-17 describes the total and minimum number of sensors required to monitor safety-related variables. There are no sensors in the ESF systems that have a spatial dependence.

The following definitions were used in generating the data contained in Table 7.3-17:

Channel A group of devices sufficient to produce only one trip signal for only one parameter. A channel consists of all components from sensor through trip unit (or relay, when provided). A channel loses its identity when combined with one or more additional channels.

Trip-System A system consists of one or more trip logic arrangements, each in turn consisting of one or more channels.

Trip Function This is defined as a parameter which is used to control the action of a particular actuated device (e.g., pumps and valves). Each trip function is achieved by one or more trip systems. A trip function may be accomplished by the action of a single channel or the simultaneous action of more than one channel, depending upon the system design.

Minimum Operable Channels The number of channels required by plant Technical Specifications to be operable before the Operator needs to take actions as specified by the Technical Specifications. The minimum number of channels does not stand alone, but must be considered along with all applicable action statements and clarifying notes contained in the Technical Specifications.

Normal Channels The number of channels provided for in the design for each trip system function. The number of normal channels provided must at least be equal to the minimum number of channels required by the plant Technical Specifications. The number of normal channels can exceed the minimum number to provide for additional operational flexibility, e.g., test and maintenance.

7.3.1.2.3 Prudent Operational Limits

Operational limits for each safety-related variable trip setting are selected with sufficient margin that a spurious ESF system initiation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or the nuclear system process barrier, is kept within acceptable bounds.

7.3.1.2.4 Margin

Adequate margin between safety limits and instrument setpoints is provided to allow for instrument error. The appropriate allowable values are listed in Technical Specifications. The bases are discussed in the Technical Specifications Bases.

7.3.1.2.5 Levels

Levels requiring protective action are established in Technical Specifications.

7.3.1.2.6 Range of Transient, Steady-State, and Environmental Conditions

Refer to Section 3.11 and the Environmental Qualification Program documents for environmental conditions. Refer to Sections 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to ESF instrumentation and controls. All ESF instrumentation and controls are specified and purchased to withstand the effects of energy supply extremes.

7.3.1.2.7 Malfunctions, Accidents, and Other Unusual Events That Could Cause Damage to Safety Systems

Chapter 3 describes the following accidents and events: floods, storms, tornadoes, earthquakes, fires, and pipe breaks outside containment. LOCA events are discussed in Chapters 6 and 15. Each of these events is discussed below for the ESF systems.

Floods

The buildings containing ESF system components are protected against floods as described in Section 3.4.

Storms and Tornadoes

All buildings, except the turbine building containing ESF system components, are protected against storms and tornadoes as described in Section 3.3.

Earthquakes

All structures, except the turbine building, containing ESF system components have been seismically qualified as described in Sections 3.7 and 3.8. Seismic qualification of instrumentation and electrical equipment is discussed in Section 3.10.

Fires

The functions of the ESF systems are protected from the effects of fire as described in Section 9.5.1.

LOCA

The ESF system components located inside the drywell and functionally required during and/or following a LOCA have been environmentally qualified to remain functional as discussed in Section 3.11.

Missiles

Protection for safety-related components is described in Section 3.5.

7.3.1.2.8 Minimum Performance Requirements

Minimum performance requirements for ESF instrumentation and controls are provided in Technical Specifications.

7.3.1.3 Final System Drawings

The final system drawings, including piping and instrumentation diagrams (P&ID) and functional control diagrams (FCD)/control logic diagrams have been provided for the ESF systems in the FSAR. Functional and architectural design differences between the PSAR and FSAR are listed in Tables 1.3-8 and 1.3-9.

7.3.2 Analysis

7.3.2.1 ESF Systems - Instrumentation and Controls

Chapters 6 and 15 evaluate the individual and combined capabilities of the ESF systems. The ESF systems are designed in such a way that a loss of instrument air, a plant load rejection, or a turbine trip will not prevent the completion of the safety function.

Originally, the FMEA of: 1) the BOP instrumentation and control components of the ECCS (HPCS, LPCS, and LPCI), and 2) the SGTS, CGCS, reactor building HVAC, service water, service water pump bays ventilation, control building HVAC, control building chilled water, standby power, and diesel generator building HVAC systems were contained in the Unit 2 FMEA document, which is historical.

FMEAs for plant systems are now performed and controlled by the design process.

7.3.2.1.1 Conformance to 10CFR50 Appendix A

The general design criteria conformance discussions provided in Section 3.1 apply to the ESF systems as specified in Table 7.1-3.

7.3.2.1.2 Conformance to IEEE Standards

The IEEE Standards that apply to the ESF systems are specified in Table 7.1-3. The following conformance discussions apply specifically to ESF systems. Refer to Section 7.1.2.2 for conformance discussions applying generically to all safety-related systems.

7.3.2.1.2.1 Conformance to IEEE-279-1971

Paragraph 4.1 The ESF systems automatically initiate the appropriate protective actions, whenever the parameters described in Section 7.3.1.2 (Variables Monitored to Provide Protective Action) reach predetermined limits, with precision and reliability assuming the full range of conditions and performance discussed in Section 7.3.1.2.

Paragraph 4.2 ESF systems are not required to meet the single-failure criterion on an individual system (division) basis. However, on a network basis, the single-failure criterion does apply to assure the completion of a protective function. Redundant sensors, wiring, logic, and actuated devices are physically and electrically separated in such a way that a single failure will not prevent the protective function. Refer to Section 8.3.1.4 for a complete description of the Unit 2 separation criteria.

Paragraph 4.3 Components used in the ESF systems have been carefully selected on the basis of suitability for the specific application. Ratings have been selected with sufficient conservatism to ensure against significant deterioration during anticipated duty over the lifetime of the plant. Furthermore, a quality control and assurance program has been implemented and documented by equipment vendors to comply with the requirements set forth in 10CFR50 Appendix B. For a further discussion of the quality of ESF system components and modules, refer to Sections 3.2 and 3.11.

Paragraph 4.4 Vendor certification confirms that the sensors associated with each of the ESF trip variables, manual switches, and trip logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.

Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In situ operational testing of these sensors, channels, and the entire protection system will be performed during the preoperational test phase.

For a complete discussion of ESF equipment qualification, refer to Sections 3.5, 3.6, 3.10, and 3.11.

Paragraph 4.5 For a discussion of ESF system channel integrity under all extremes of conditions described in Section 7.3.1.2, refer to Sections 3.10, 3.11, 8.2.1, and 8.3.1.

Paragraph 4.6 ESF system channel independence is maintained through the application of the Unit 2 separation criteria (Section 8.3.1.4).

Paragraph 4.7 There are no ESF system and control system interactions.

Paragraph 4.8 The ESF variables are direct measures of the desired variables requiring protective actions. Refer to Sections 7.3.1.1.1 through 7.3.1.1.11.

Paragraph 4.9 Refer to Section 7.3.2.1.3, Regulatory Guide 1.22.

Paragraph 4.10 Refer to Section 7.3.2.1.3, Regulatory Guide 1.22.

Paragraph 4.11 During periodic testing of any one ESF system channel, a sensor may be valved out of service and returned to service under the administrative control procedures. Since only one sensor is valved out of service at any given time during the test interval, protective action capability for ESF system automatic initiation is maintained through the remaining redundant instrument channels.

Paragraph 4.12 The ESF system contains the following operating bypasses. The PCRVICS has two bypasses:

1. Main steam line low pressure operating bypass is imposed by means of the reactor mode switch in all modes except RUN. The mode switch cannot be left in these positions above 10 percent of rated power without initiating a scram. Therefore, the bypass is removed by the normal reactor operating sequence.

2. The low condenser vacuum bypass is imposed by means of a manual bypass switch. Bypass removal is accomplished manually by placing the manual switch in the NORMAL position or by placing the mode switch in the RUN position. The bypass is removed automatically when the turbine stop valve is more than 90 percent open.

<u>Paragraph 4.13</u> For a discussion of bypass and inoperability indication, refer to Section 7.1.2.3, Regulatory Guide 1.47.

<u>Paragraph 4.14</u> Access to means of bypassing any safety action or function for the ESF systems is under the administrative control of the Control Room Operator. The Operator is alerted to bypasses as described in Section 7.1.2.3, Regulatory Guide 1.47.

Control switches that allow safety system bypasses are keylocked. All keylock switches in the control room are designed so that their keys can only be removed when the switches are in the ACCIDENT or SAFE position. All keys will normally be removed from their respective switches during operation and maintained under strict administrative control.

<u>Paragraph 4.15</u> There are no multiple setpoints within the ESF systems.

<u>Paragraph 4.16</u> Each of the automatically initiated ESF system control logics seal in electrically and remain energized after initial conditions return to normal. Deliberate Operator action is required to return (reset) an ESF system logic to normal.

For Unit 2, upon actuation of an ESF signal, all components proceed to their safety position. To reset a component, two distinct Operator actions are necessary: one to reset the actuation signal, and one to reset each component. See Section 1.10, Item II.E.4.2, Position 4, and Attachment 1.10-1.

Paragraph 4.17 Refer to the discussion of RG 1.62 in Section 7.3.2.1.3.

Paragraph 4.18 All accesses to ESF system setpoint adjustments, calibration controls, and test points are maintained under administrative control.

Paragraph 4.19 ESF protective actions are directly indicated and identified by annunciators located in the main control room and a typed record is available from the process computer.

Paragraph 4.20 The ESF systems are designed to provide the Operator with accurate and timely information pertinent to their status. They do not introduce signals that could cause anomalous indications confusing to the Operator.

Paragraph 4.21 The ESF systems are designed to permit repair or replacement of components. Recognition and location of a failed component will be accomplished during periodic testing or by annunciation in the main control room.

Paragraph 4.22 The ESF control panels are identified by nameplates. The nameplate shows the division to which each panel or rack is assigned, and identifies system(s) in the control panel. The system to which each relay belongs is identified on the relay panels. All wiring and cabling outside of panels are labeled to indicate divisional assignment as well as system assignment.

7.3.2.1.2.2 Conformance to IEEE-338-1971

The ESF systems are fully testable during normal operation in conformance with IEEE-338. For further discussion of how the system designs conform, refer to Section 7.3.2.1.3, Regulatory Guide 1.22.

Operation of each instrument channel is testable from the sensor to final logic relay. The pressure sensor may be valved out of service and test pressures applied to the sensor to check operation of the complete instrument channel from sensor to trip unit. The channels monitoring the same variable may be cross compared. A calibration module may be used to test the trip unit of each channel. These tests will not interfere with automatic operation of the system if required by an initiation signal.

Periodic testing is performed in accordance with plant surveillance procedures. These procedures establish the administrative control for removing only one instrument channel at a time from service. Plant surveillance procedures establish frequency schedule and documentation required for the testing. Testing is performed at intervals so that credible failure may be detected and repaired before system reliability is reduced.

7.3.2.1.2.3 Conformance to IEEE-384-1974

The criteria for independence of IEEE-279-1971 Paragraph 4.6, as further defined in IEEE-384-1974, are met as described in Section 7.3.2.1.2.1.

7.3.2.1.3 Conformance to Regulatory Guides

Regulatory guides that apply to the ESF systems are specified in Table 7.1-3. The following conformance discussions apply specifically to the ESF systems. Refer to Section 7.1.2.3 for conformance discussions applying generically to all safety-related systems. The extent of compliance with all applicable regulatory guides is provided in Section 1.8.

<u>Regulatory Guide 1.22</u> While not a design basis, the extent of compliance to verify the operability of each system component is as follows: The ESF systems instrumentation and controls are capable of being tested during normal plant operation, unless that testing is detrimental to plant availability. Testing of safety-related pressure sensors is accomplished by valving out each sensor one at a time and applying a simulated input. This verifies the operability of the sensor contacts, the sensor setpoint, and the associated logic components in the main control room. Functional operability of temperature sensors may be verified by readout comparisons, applying a heat source to the locally mounted temperature sensing elements, or by continuity testing.

For the HPCS, LPCS, and LPCI, testing for functional operability of the control logic relays can be accomplished by use of plug-in test jacks and switches in conjunction with single sensor tests.

Four test jacks are provided to allow ADS logic testing, one for each logic channel. During testing only one logic should be actuated at a time. However, when the test plug is plugged into one channel, the complementary channel of that trip system is automatically rendered inoperative. Therefore, inadvertent ADS actuation cannot occur even if both channels are improperly placed in the test mode simultaneously. An alarm is provided if a test plug is inserted in both channels in a division at the same time. Operation of the test plug switch and the permissive contacts will close one of the two series relay contacts in the valve solenoid circuit. This will cause a panel light to indicate proper channel operation.

Annunciation is provided in the main control room whenever a test plug is inserted into a jack to indicate to the Operator that an ECCS is in a test status. Operability of air-operated valves (AOVs), solenoid-operated valves (SOVs), and MOVs is verified by actuating the valve control switches and monitoring the position change by position indicating lights at the control switch. When the ADS SRVs are tested at pressure, valve opening and closing is confirmed by observing reactor pressure, the temperature of each safety relief valve discharge line (SRVDL), and SRV position indicator lights that are fed by acoustic monitors mounted on the SRVDLs. It is also permissible to test the ADS SRVs by decoupling the actuator from the valve stem and visually verifying actuator movement by energizing the individual control solenoids with the reactor at low pressure, essentially atmospheric during a shutdown. There would be no change in reactor pressure or increase in tailpipe temperature.

ESF systems have indications, status displays, annunciation, and computer printouts that aid the Control Room Operator during periodic system tests to verify component operability.

Regulatory Guide 1.53 Refer to IEEE-279, Paragraph 4.2, Section 7.3.2.1.2.1. In addition, conformance of the ESF systems to this regulatory guide is described as follows.

A. ECCS

Position C.1 The ECCS consists of four systems: HPCS, ADS, LPCS, and LPCI (three loops). No single component, power supply bus, or circuit failure in any one system or loop will prevent the ECCS from functioning. Thus, on a system level basis, sufficient systems or loops are available to maintain the ECCS protective function. With the exception of the ADS, no system or loop (LPCI) is designed on an individual basis to be single-failure-proof. The ECCS systems are described in Sections 6.3.1.1, 6.3.2, and 6.3.3.

Position C.2 The ECCS systems are all fully testable.

Position C.3 A single push-button switch will initiate both LPCS and LPCI loop A (Division I equipment). A second push-button switch initiates LPCI loops B and C (Division II equipment). With these exceptions, redundant system or loops control equipment is not shared and is separate from other system components.

Position C.4 Except for LPCS and LPCI initiation logic, failure of a single system logic or actuator will not prevent other systems or loops from performing their protection function. Redundancy in Division I and Division II provides single-failure protection for LPCS and LPCI.

B. PCRVICS

Position C.1 Redundant sensors are used for each logic channel input circuit. Combination of relay logic is achieved through proper isolation devices maintaining independence of each logic channel. Section 7.3.1.1.2 provides a detailed discussion of the relay logic and isolation functions.

Position C.2 There are no known undetectable failures in the design of this system.

Position C.3 The reactor mode switch which supplies signals to four redundant channels of the PCRVICS conforms to the separation criteria. This switch consists of four separate cam-driven units, each one serving only one channel or reactor control. Each individual unit is enclosed in a metallic enclosure. The contacts of each switch unit are used by its dedicated channel only. Circuits leaving each unit are run in conduits separate from other circuits.

Position C.4 A single component failure or a logic failure will not disable a redundant sensor and its associated logic channel. Thus, the PCRVICS protection function is maintained through redundancy.

C. RHR System - Non-LPCI Modes

Position C.1 The RHR system consists of three loops (A, B, and C), each operating on a separate pump. RHR loop A is on a separate divisional power from RHR loops B and C, providing the required redundancy for the various ESF modes of operation. These ESF modes of operation are described in the following sections: Section 7.3.1.1.1.4 (LPCI), Section 7.3.1.1.3 (containment spray), and Section 7.3.1.1.4 (suppression pool cooling). The normal shutdown cooling mode, described in Section 7.4.1.3, is a safe shutdown system, not an ESF system. No single failure either in system design or power source will result in the loss of the shutdown cooling mode because Unit 2 is capable of using normal shutdown cooling through the recirculation and RHR loops or one of the two alternate shutdown cooling modes. The three non-LPCI RHR modes use loops A and B.

Position C.2 Each RHR loop is fully testable.

Position C.3 The redundant RHR loops use separate control switches which are not shared.

Position C.4 Failure of a single loop logic will not disable the system function, since a redundant loop is available on a separate divisional power source.

Regulatory Guide 1.62 The HPCS, LPCS, and the Division II LPCI systems can be manually initiated at the system level from the main control room by actuation of an armed push button for each system. The LPCS push button also initiates the Division I LPCI system. The ADS and PCRVICS are manually initiated at the system (division) level by actuation of two armed push buttons (one for each logic channel).

The RCSCM is manually initiated at the system (division) level by actuation of the RHR pump start control switch and by opening the containment spray and/or suppression chamber spray valves.

The RCSCM is manually initiated from the main control room by initiation of system pump and valve controls.

The SGTS, CGCS, reactor building HVAC system, SWP system, service water pump bays ventilation system, control building HVAC system, control building chilled water system, standby power system, and diesel generator building HVAC system equipment are manually initiated at the system (division) level by actuation of individual control switches in the main control room.

Actuation of the system level manual initiation switches simulates actions of automatic or manual (individual equipment initiation) system actuation.

Review of NSSS ESF systems and Category I BOP systems revealed that the logic for manual initiation in several instances is interlocked with permissive logic from various sensors. This permissive logic is dependent upon the same sensors as those used for automatic initiation of the system. Each of these systems, however, has redundancy and/or diversity in its design such that a single-failure of a system will not prevent the safety function (protective action) from being initiated and carried to completion by the manual or automatic initiation of the redundant or diverse portion of the system.

In some cases, Category I BOP systems use interlocks to provide protection for safety-related equipment, i.e., limit switches of dampers interlocked with starting of fans. In each case, redundant equipment is available so that no single failure in the manual, automatic, or common portion of the protection system will prevent initiation by manual or automatic means of the redundant portion of the system. Additionally, an objective of the surveillance program is to address high-pressure/low-pressure interlocks in the surveillance testing.

The system design meets the single-failure criterion by providing redundancy so that a single-failure will not prevent initiation of the protective action. FMEAs have been performed to verify the above for all BOP Category I systems. Originally, the FMEA was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

ESF and RCIC Reset Controls

The ADS reset controls return each ADS SRV to its closed position. This deliberate Operator action closing the ADS SRVs B22-F013 C, H, K, M, N, R, and U to prevent or limit inadvertent reactor depressurization is considered an allowed exception to IE 80-06 compliance. Besides this exception, there is no deviation from the guidance indicated in IE Bulletin 80-06.

In addition, RCIC (not considered an ESF) was reviewed and found to be in conformance with the guidance of IE Bulletin 80-06.

TABLE 7.3-1

HIGH-PRESSURE CORE SPRAY SYSTEM
INSTRUMENTATION SPECIFICATIONS

HPCS Function	Instrument	Instrument Range*
Reactor vessel low water level (level 2)(1)	Level transmitter	0 - 750" H_2O
High drywell pressure	Pressure transmitter	0 - 150" H_2O
Reactor vessel high water level (level 8)(1)	Level transmitter	0 - 750" H_2O
Pump discharge pressure	Pressure transmitter	0 - 3,000 psig
HPCS flow	Flow transmitter	0 - 25" H_2O
Suppression pool high water level	Level transmitter	0 - 100" H_2O
Condensate storage tank level (located in condensate storage tank suction line)	Level transmitter	0 - 150" H_2O

* See Technical Specifications for Allowable Values.
(1) Instrument zero equal to 380.7 in above vessel zero.

TABLE 7.3-2

AUTOMATIC DEPRESSURIZATION SYSTEM
INSTRUMENTATION SPECIFICATIONS

ADS Functions	Instrument	Instrument Range*
Reactor vessel low water level (level 1)[(1)]	Level transmitter	70 - 220.2" H_2O
Reactor vessel low water level (level 3)[(1)]	Level transmitter	0 - 150" H_2O
LPCS permissive	Pressure transmitter	0 - 300 psig
RHR permissive	Pressure transmitter	0 - 300 psig
ADS time delay	Timer	4 - 120 sec

* See Technical Specifications for Allowable Values.
(1) Instrument zero equal to 380.7 in above vessel zero.

TABLE 7.3-3

LOW-PRESSURE CORE SPRAY
INSTRUMENT SPECIFICATIONS

Function	Instrument	Instrument Range[1]
Reactor vessel low water level (level 1)[2]	Level transmitter	0 - 750" H_2O
High drywell pressure	Pressure transmitter	0 - 150" H_2O
Injection valve differential pressure	Differential pressure transmitter	0 - 1,000 psid
Pump minimum flow bypass	Flow transmitter	0 - 30" H_2O

(1) See Technical Specifications for Allowable Values.
(2) Instrument zero equal to 380.7 in above vessel zero.

TABLE 7.3-4

LOW-PRESSURE COOLANT INJECTION
INSTRUMENTATION SPECIFICATIONS

LPCI Function	Instrument	Instrument Range*
Reactor vessel low water level (level 1)[(1)]	Level transmitter	70 - 220.2" H_2O
High drywell pressure	Pressure transmitter	0 - 138.6" H_2O
Injection valve differential pressure	Differential pressure transmitter	0 - 700 psid
Pump minimum flow bypass	Flow transmitter	0 - 30" H_2O

* See Technical Specifications for Allowable Values.
(1) Instrument zero equal to 380.7 in above vessel zero.

TABLE 7.3-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM INSTRUMENT SPECIFICATIONS

PCRVICS Function	Instrument	Instrument Range[1]
Reactor vessel low water level 2[2]	Level transmitter	0 - 750" H_2O
High main steam line tunnel ambient temperature	Temperature switch	50 - 350°F
High main steam line flow	Differential pressure transmitter	0 - 300 psi
Main steam line low pressure (reactor mode switch in RUN)	Pressure transmitter	0 - 3,000 psig
Reactor vessel low water level 3[2]	Level transmitter	0 - 150" H_2O
High drywell pressure	Pressure transmitter	0 - 750" H_2O
Vessel high pressure	Pressure transmitter	0 - 3,000 psig
RWCU system high differential flow	Flow transmitter	0 - 750" H_2O 0 - 150" H_2O
RWCU system equipment area high ambient temperatures	Temperature switch	50 - 350°F
Reactor vessel low water level 1[2]	Level transmitter	0 - 750" H_2O
RWCU system nonregenerative heat exchanger outlet high temperature	Temperature switch	-50 - 150°F

TABLE 7.3-5 (Cont'd.)

PCRVICS Function	Instrument	Instrument Range[1]
Standby liquid control system actuated	Manual switch	N/A
Main steam line tunnel high differential temperature	Temperature switch	0 - 150°F
Main steam line area high temperature (turbine building)	Temperature switch	50 - 350°F
Main condenser low vacuum trip	Pressure transmitter	0 - 55" Hg abs
RHR equipment area high ambient temperature	Temperature switch	50 - 350°F
Containment purge exhaust (via SGTS) high radiation[3]	Radiation monitor	10^{-7} - 10^{-1} system uCi/cc

(1) See Technical Specifications for Allowable Values.
(2) Instrument zero equal to 380.7 in above vessel zero.
(3) Low end of range is the design value. Actual range observed is subject to background readings.

TABLE 7.3-6

RHR CONTAINMENT SPRAY COOLING MODE
INSTRUMENTATION SPECIFICATIONS

Function	Instrument	Instrument Range
High drywell pressure	Pressure transmitter	0 - 138.6" H_2O
Reactor vessel low water level (level 1)[1]	Level transmitter	70 - 220.2" H_2O

[1] Instrument zero equal to 380.7 in above vessel zero.

TABLE 7.3-7

RHR SUPPRESSION POOL COOLING MODE
INSTRUMENTATION SPECIFICATIONS

RSPCM Function	Instrument	Instrument Range
Reactor vessel low water level (level 1)[1]	Level transmitter	70 - 220.2" H_2O
High drywell pressure	Pressure transmitter	0 - 138.6" H_2O
Suppression pool temperature high	Temperature recorder	50 - 250°F

[1] Instrument zero equal to 380.7 in above vessel zero.

TABLE 7.3-8

STANDBY GAS TREATMENT SYSTEM
INSTRUMENTATION SPECIFICATIONS

SGTS Function	Instrument	Instrument Range[1]
Reactor vessel water level (level 2)	Level transmitter	-5 to 205"
Drywell pressure	Pressure transmitter	-2.5 - 2.5 psig
Reactor building above and below the refueling floor exhaust radiation[2]	Radiation monitor	10^{-7} - 10^{-1} uCi/cc

(1) See Technical Specifications for Allowable Values.
(2) Low end of the range is the design value. Actual value observed is subject to background readings (gas channel only).

TABLE 7.3-9

COMBUSTIBLE GAS CONTROL SYSTEM
INSTRUMENTATION SPECIFICATIONS

CGCS Function	Instrument	Instrument Range
Primary containment hydrogen concentration high activity	Containment monitoring system hydrogen analyzer	0 - 30%

TABLE 7.3-10

REACTOR BUILDING HVAC SYSTEM
INSTRUMENTATION SPECIFICATIONS

Reactor Building HVAC Function	Instrument	Instrument Range[1]
Reactor building above and below refueling floor exhaust radiation[2]	Radiation monitor	10^{-7} - 10^{-1} uCi/cc

[1] See Technical Specifications for Allowable Values.
[2] Low end of range is the design value. Actual range observed is subject to area background (gas channel only).

TABLE 7.3-11

SERVICE WATER SYSTEM
INSTRUMENTATION SPECIFICATIONS

Service Water Function	Instrument	Instrument Range
Service water to TBCLCW system pressure	Pressure transmitter	0 - 100 psig
	Pressure switch	0 - 100 psig
RBCLCW-RCS pumps pressure	Pressure transmitter	0 - 200 psig
Service water pumps discharge flow	Flow transmitter	0 - 12,000 gpm
HPCS emergency diesel generator water header pressure	Pressure transmitter	0 - 100 psig[1]
Emergency diesel generator water header pressure	Pressure transmitter	0 - 100 psig
Control building air conditioning chiller SW outlet temperature	Temperature transmitter	35 - 130°F
Control building chiller inlet water temperature low	Temperature switch	32 - 100°F
SW header flow to lake	Flow transmitter	0 - 35,000 gpm
SW header discharge pressure	Pressure transmitter	0 - 100 psig
Reactor building emergency ventilation recirculation air temperature high	Temperature indicating switch	-40 - 180°F

TABLE 7.3-11 (Cont'd.)

Service Water Function	Instrument	Instrument Range
Electrical tunnels north unit cooler suction temperature high/low	Temperature switch high Temperature switch low	45 - 120°F 45 - 120°F
Discharge bay water level high/high	Level switch	N/A[1]
SW pump suction bay level low	Level switch	N/A[1]
Intake tunnel 2 water temperature low	Temperature switch	25 - 80°F[1]
RHR heat exchanger SW discharge radiation[2]	Radiation monitor	10^{-7} - 10^{-1} uCi/ml
SW effluent loop A&B radiation[2]	Radiation monitor	10^{-7} - 10^{-1} uCi/ml[3]

[1] See TRM for Allowable Values.
[2] Low end of range is the design value. Actual range observed is subject to area background.
[3] See Offsite Dose Calculation Manual for Allowable Values.

TABLE 7.3-12

SERVICE WATER PUMP BAYS VENTILATION
SYSTEM INSTRUMENTATION SPECIFICATIONS

Service Water Pump Bays Ventilation Functions	Instrument	Instrument Range
Space temperature high	Temperature indicating switch	-40 - 180°F

TABLE 7.3-13

CONTROL BUILDING HVAC SYSTEM
INSTRUMENTATION SPECIFICATIONS

Control Building HVAC Function	Instrument	Instrument Range
Control building air conditioning supply air radiation(2)	Radiation monitor	10^{-7} - 10^{-1} uCi/cc(1)
Control room air conditioning return air temperature	Temperature switch	50 - 105°F
Control room and office supply air temperature	Temperature controller	-40 - 120°F
Control room and office supply air flow	Flow switch	0 - 16,745 cfm
Filter heater inlet air temperature	Temperature transmitter	0 - 100°F
Filter heater outlet air temperature	Temperature transmitter	5 - 105°F
Filter train outlet air temperature	Temperature transmitter	15 - 115°F

(1) See Technical Specifications for Allowable Values.
(2) Low end of range is the design value. Actual range observed is subject to area background.

TABLE 7.3-14

CONTROL BUILDING CHILLED WATER SYSTEM
INSTRUMENTATION SPECIFICATION

Control Building Chilled Water Function	Instrument	Instrument Range
Control building chilled water flow	Flow transmitter	0 - 350 gpm
Control building chilled water expansion tank water level	Level switch (float tandem)	N/A
Control building main control room humidity	Moisture transmitter	0 - 100% R.H.
Control building main control room temperature	Resistance temperature detector	65° - 104°F
Control building main control room air conditioning return air temperature	Resistance temperature detector	50° - 105°F
Control building chilled water tank water level	Level switch (float)	N/A

TABLE 7.3-15

STANDBY POWER SYSTEM
INSTRUMENTATION SPECIFICATIONS

Standby Power System Function	Instrument	Instrument Range
Emergency diesel generator fuel oil day tank level (high-high, low-low)	Level switch (tandem displacer)	N/A
Emergency diesel generator fuel oil day tank level (high, low)	Level switch (displacer)	N/A
Emergency diesel generator fuel oil day tank level	Level indicating transmitter	0 - 40" W.C.
Emergency diesel generator fuel oil storage tank level (high, low)	Level switch (tandem displacer)	N/A
Emergency diesel generator fuel oil storage tank level	Level indicating transmitter	0 - 120" W.C.
Fuel oil transfer pump strainer differential pressure (high)	Differential pressure indicating switch	0 - 30" W.C.
Fuel oil transfer pumps flow (low)	Flow indicating switch	0 - 20 gpm

TABLE 7.3-16

DIESEL GENERATOR BUILDING HVAC
INSTRUMENTATION SPECIFICATIONS

Diesel Generator Building HVAC Function	Instrument	Instrument Range
Emergency diesel generator room ventilation exhaust fan intake airflow	Flow switch	0 - 45,500 cfm
Emergency diesel generator room temperature (for air dampers control)	Resistance temperature detector	50° - 150°F
Emergency diesel generator room temperature (heater control)	Temperature indicating switch	-40° - 120°F
Emergency diesel generator room temperature	Resistance temperature detector	50° - 150°F
Emergency diesel generator normal exhaust fan airflow	Flow switch	0 - 1,540 cfm
Diesel generator makeup air fan airflow	Flow switch	0 - 4,620 cfm
Diesel generator building corridor room temperature	Temperature indicating controller	-40° - 120°F
Diesel generator building makeup air heater airflow	Flow switch	0 - 4,620 cfm
Diesel generator building makeup air filter differential pressure	Differential indicating switch	0 - 2" W.C.
Diesel generator building control room temperature (for diesel generator building control room unit cooler)	Temperature indicating switch	-40° - 180°F