Table R Abnormal Rad Levels/Radiological Effluents

Category	General Emergency	Site Area Emergency	Alert	Notification of Unusual Event
Radiological Effluents	RG1 (Pg R-22) **IC Dose at the EMERGENCY PLAN BOUNDARY Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem Child Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.** Modes: ALL <u>EAL Threshold Value</u> (1 or 2 or 3 or 4 or 5) ***Note:*** *If dose assessment results are available at the time of declaration, the classification should be based on EAL 2 instead of EAL 1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.* 1. VALID Noble Gas vent stack monitor reading(s) that exceeds or is expected to exceed a site total release rate of 6.2E+9μCi/min for 15 minutes or longer and Dose Projections are not available. <u>OR</u> 2. VALID dose assessment using actual meteorology indicates projected doses greater than 1000 mrem TEDE or 5000 mrem child thyroid CDE at or beyond the EPB. <u>OR</u> 3. A VALID reading sustained for 15 minutes or longer on the RMS perimeter radiation monitoring system greater than 1000 mR/hr. (The RMS perimeter radiation monitoring system is only monitored when the TSC or EOF is activated) <u>OR</u> 4. Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour at or beyond the EPB. <u>OR</u> 5. Analyses of field survey samples indicate child thyroid dose commitment at or beyond the EPB of 5000 mrem assuming one hour of inhalation.	RS1 (Pg 26) **IC Dose at the EMERGENCY PLAN BOUNDARY Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mrem TEDE or 500 mrem Child Thyroid CDE for the Actual or Projected Duration of the Release.** Modes: ALL <u>EAL Threshold Value</u> (1 or 2 or 3 or 4 or 5) ***Note:*** *If dose assessment results are available at the time of declaration, the classification should be based on EAL 2 instead of EAL 1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.* 1. VALID Noble Gas vent stack monitor reading(s) that exceeds or is expected to exceed a site total release rate of 6.2E+8μCi/min for 15 minutes or longer and Dose Projections are not available. <u>OR</u> 2. VALID dose assessment using actual meteorology indicates projected doses greater than 100 mrem TEDE or 500 mrem child thyroid CDE at or beyond the EPB. <u>OR</u> 3. A VALID reading sustained for 15 minutes or longer on the RMS perimeter radiation monitoring system greater than 100mR/hr. (The RMS perimeter radiation monitoring system is only monitored when the TSC or EOF is activated) <u>OR</u> 4. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour at or beyond the EPB. <u>OR</u> 5. Analyses of field survey samples indicate child thyroid dose commitment at or beyond the EPB of 500 mrem assuming one hour of inhalation.	RA1 (Pg 30) **IC Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Technical Requirements Manual Limits for 15 Minutes or Longer.** Modes: ALL <u>EAL Threshold Value</u> (1 or 2 or 3 or 4) 1. VALID Noble Gas vent stack monitor reading(s) that exceeds a site total release rate of 2.0E+8 μCi/min and that is sustained for 15 minutes or longer. <u>OR</u> 2. Confirmed sample analyses for airborne releases indicate total site release rates for 15 minutes or longer resulting in dose rates at the SITE BOUNDARY of: A. Noble gases > 1.0E+5 mrem/year whole body, <u>OR</u> B. Noble gases > 6.0E+5 mrem/year skin, <u>OR</u> C. I-131, I-133, H-3, and particulates with half-lives greater than 8 days >3.0E+5 mrem/year to any organ (inhalation pathways only). <u>OR</u> 3. Confirmed sample analyses for liquid releases indicate concentrations in excess of 200 times the Technical Requirements Manual liquid effluent limits for 15 minutes or longer. <u>OR</u> 4. VALID reading on any liquid effluent monitor that exceeds two hundred times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer	RU1 (Pg 34) **IC Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Technical Requirements Manual Limits for 60 Minutes or Longer.** Modes: ALL <u>EAL Threshold Value</u> (1 or 2 or 3 or 4) 1. VALID Noble Gas vent stack monitor reading(s) that exceeds a total site release rate of 2.0E+6 μCi/min and that is sustained for 60 minutes or longer. <u>OR</u> 2. Confirmed sample analyses for airborne releases indicate total site release rates, with a release duration of 60 minutes or longer, resulting in dose rates at the SITE BOUNDARY of : A. Noble gases >1000 mrem/year whole body, <u>OR</u> B. Noble gases >6000 mrem/year skin, <u>OR</u> C. I-131, I-133, H-3 and particulates with half-lives greater than 8 days > 3000 mrem/year to any organ (inhalation pathway only). <u>OR</u> 3. Confirmed sample analyses for liquid releases indicate concentrations with a release duration of 60 minutes or longer in excess of two times the Technical Requirements Manual liquid effluent limits. <u>OR</u> 4. VALID reading on any liquid effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer

Table R Abnormal Rad Levels/Radiological Effluents

Abnormal Radiation Levels			RA2 (Pg 38) **IC Release of Radioactive Material or Increases in Radiation Levels within the Facility that Impedes Operation of Systems Required to Maintain Safe Operations or to Establish Or Maintain Cold Shutdown.** Modes: ALL <u>**EAL Threshold Value**</u> (1 or 2) 1. VALID radiation reading > 15 mR/hr in the Main Control Room, the Radwaste Control Room or both the Security Control Center (SCC) and Alternate Security Control Center (ASCC). <u>OR</u> 2. VALID radiation monitor readings > 10 R/hr in areas requiring infrequent access to maintain plant safety functions (Table R-1).	RU2 (Pg 40) **IC Unexpected Increase in Plant Radiation.** Modes: ALL <u>**EAL Threshold Value**</u> (1 or 2) 1. A. Uncontrolled water level decrease in the reactor refueling cavity, fuel transfer canal or spent fuel pool with all irradiated fuel assemblies remaining covered by water as indicated by any of the following on either unit: • Unexpected Fuel Pool Water Low Level alarm <u>OR</u> • Skimmer Surge Tank Low Level alarm <u>OR</u> • Visual observation of an uncontrolled water level drop below a fuel pool skimmer surge tank inlet, <u>OR</u> • Observation of water draining down the outside wall of Primary Containment. <u>AND</u> B. UNPLANNED VALID Refuel Floor Area Radiation Monitor (Table R-2) readings greater than 500 mR/hr. <u>OR</u> 2. UNPLANNED VALID Area Radiation Monitor readings increase by a factor of 1000 over normal* levels. *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.
Irradiated Fuel Accidents			RA3 (Pg 43) **IC Damage to Irradiated Fuel or Loss of Water Level that has or will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.** Modes: ALL <u>**EAL Threshold Value**</u> (1 or 2 or 3) 1. UNPLANNED VALID Refuel Floor Area Radiation Monitor (Table R-2) readings greater than 1 R/hr. <u>OR</u> 2. Water level < 22 feet above the RPV flange for the reactor refueling cavity that will result in irradiated fuel uncovering. <u>OR</u> 3. Water level < 22 feet above seated irradiated fuel for the spent fuel pool that will result in irradiated fuel uncovering.	

Table R-1
Reactor Building Radiation Monitors

	ARM Number	
RB Area Elevation (ft)	**High Range**	**ARM Channel Description**
749	52	RWCU Recirc Pp Access
	54	Fuel Pool Pump Area
719	50	CRD North
	51	CRD South
670	53	Remote Shutdown Room
645	48	HPCI PP Turbine Room
	57	RCIC PP Turbine Room
	55	RHR A C PP Room
	56	RHR B D PP Room

Table R-2
Refuel Floor Area Radiation Monitors

ARM Number	Description	Range (mR/hr)
14	Spent Fuel Pool Area	0.1 – 1000
47 (44 U-2)	Spent Fuel Pool Area	0.1 – 1000
49	Refuel Floor Area	10^2 - 10^6

ABLE F - FISSION PRODUCT BARRIER DEGRADATION

Barrier	1. Fuel Clad Barrier Pg 50 to 55		2. Reactor Coolant System Barrier Pg 56 to 64		3. Primary Containment Barrier Pg 65 to 74	
Parameter	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
a. Reactor Coolant Activity Level	Reactor Coolant activity > 300µCi/gm I-131 Dose Equivalent. **(Pg. 50)**	N/A	N/A	N/A	N/A	N/A
b. RPV Level	RPV Level < [redacted]". **(Pg. 51)**	RPV Level < [redacted]". **(Pg. 51)**	RPV Level < [redacted]". **(Pg. 56)**	N/A	N/A	Entry into EP-DS-002, "RPV and Primary Containment Flooding Procedure". **(Pg. 65)**
c. RCS Leak Rate or Containment Isolation Failure or Breach/Bypass	N/A	N/A	Unisolable Main Steamline break as indicated by the failure of both MSIVs in any one line to close. **(Pg. 57)** AND A. High MSL Flow AND High Steam Tunnel Temperature annunciators. OR B. Direct report of steam release.	1. Unisolable RCS Leakage > 50 gpm inside Primary Containment. **(Pg. 57)** OR 2. Unisolable primary system leakage outside Primary Containment as indicated by: A. Any Reactor Building area exceeds Max Normal Reactor Building Temperature Limit per Table F-1. OR B. Any Reactor Building area exceeds Max Normal Reactor Building Radiation Limit per Table F-2.	1. Failure of All automatic isolation valves in any one line penetrating Primary Containment to close AND a downstream pathway to the environment exists. **(Pg. 67)** OR 2. Intentional venting per EP-DS-004 is performed. OR 3. Unisolable primary system leakage outside Primary Containment as indicated by: A. Any Reactor Building areas exceed Max Safe Reactor Building Temperature Limit per Table F-3. OR B. Any Reactor Building areas exceed Max Safe Reactor Building Radiation Limits per Table F-4.	N/A
d. Drywell Pressure	N/A	N/A	Drywell Pressure ≥ [redacted] psig. AND Indication of a RCS leak inside drywell. **(Pg. 61)**	N/A	1. Rapid, unexplained decrease in Drywell Pressure following initial increase in pressure. **(Pg. 71)** OR 2. Drywell Pressure response not consistent with LOCA conditions indicating a containment breach.	1. Drywell Pressure > 53 psig and increasing. **(Pg. 71)** OR 2. Drywell Hydrogen or Suppression Chamber Hydrogen > 6% AND Drywell Oxygen or Suppression Chamber Oxygen > 5%.
e. Drywell Radiation	Containment High Range Rad Monitor reading > 3000 R/hr. **(Pg. 53)**	N/A	Containment High Range Rad Monitor reading > 7 R/hr. AND Indication of a RCS leak inside drywell **(Pg. 62)**	N/A	N/A	Containment High Range Rad Monitor reading > 40,000 R/hr. **(Pg. 73)**
f. Emergency Director/Recovery Manager Judgement	Any condition in the judgement of the Emergency Director or Recovery Manager that indicates Loss of the FUEL CLAD barrier. **(Pg. 54)**	Any condition in the judgement of the Emergency Director or Recovery Manager that indicates Potential Loss of the FUEL CLAD barrier. **(Pg. 54)**	Any condition in the judgement of the Emergency Director or Recovery Manager that indicates Loss of the RCS barrier. **(Pg. 64)**	Any condition in the judgement of the Emergency Director or Recovery Manager that indicates Potential Loss of the RCS barrier. **(Pg. 64)**	Any condition in the judgement of the Emergency Director or Recovery Manager that indicates Loss of the PRIMARY CONTAINMENT barrier. **(Pg. 74)**	Any condition in the judgement of the Emergency Director or Recovery Manager that indicates Potential Loss of the PRIMARY CONTAINMENT barrier. **(Pg. 74)**

Circle the X's in the table below for all applicable situations. Declare the EAL based upon all circled X's in any column.

Fission Product Barrier Status Table Modes: 1, 2, 3	FG1: General Emergency Loss Of ANY Two Barriers AND Loss OR Potential Loss Of Third Barrier.				FS1: Site Area Emergency Loss OR Potential Loss Of ANY Two Barriers.												FA1: Alert ANY Loss OR ANY Potential Loss Of EITHER Fuel Clad OR RCS.				FU1: Notification of Unusual Event ANY Loss OR ANY Potential Loss Of Primary Containment	
Fuel Clad – LOSS	X	X		X	X		X		X	X							X					
Fuel Clad – POTENTIAL LOSS			X			X		X			X	X						X				
RCS – LOSS	X	X	X		X			X					X	X					X			
RCS – POTENTIAL LOSS				X		X	X								X	X				X		
Primary Containment – LOSS	X		X	X					X		X		X		X						X	
Primary Containment – POTENTIAL LOSS		X								X		X		X		X						X

Table F-1

Max Normal Reactor Building Temperature Limit

RB Area Elevation (ft)	Area Temperature	Max Normal Temp (°F)
818	General Area	110
779	General Area	110
749	General Area RWCU-Pump Room RWCU-Heat Exch Room RWCU-Penetration Room	110 120 120 120
719	General Area Main Steam Line Tunnel	110 157
683	General Area HPCI Pipe Routing Area RCIC Pipe Routing Area	110 120 120
670	General Area	110
645	HPCI-Equip Area HPCI-Emerg Area Cooler	120 120
645	RCIC-Emerg Area Cooler RCIC-Equip Area	120 120
645	RHR Equip Area 1	110
645	RHR Equip Area 2	110
645	CS Pump Room A	110
645	CS Pump Room B	110
645	RB Sump Room	110

Table F-2

Max Normal Reactor Building Radiation Limit

RB Area Elevation (ft)	ARM Number	Arm Channel Description	Alarm Level
818	35 14 15 42 47 (44 U2)	Cask Stor Area Spent Fuel Crit Mon Refuel Floor North (South U2) Refuel Floor West Spent Fuel Crit Mon	Hi Alarm
749	8 10	RWCU Recirc PP Access Fuel Pool PP Area	Hi Alarm
719	5 6	CRD North CRD South	Hi Alarm
670	16	Remote Shutdown Room	Hi Alarm
645	3 2 25 1	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	Hi Alarm

Table F-3

Max Safe Reactor Building Temperature

RB Area Elevation (ft)	Area Temperature	**Max Safe Temp (°F)**
818	General Area	120
779	General Area	120
749	General Area RWCU-Pump Room RWCU-Heat Exch Room RWCU-Penetration Room	120 147 147 131
719	General Area Main Steam Line Tun	120 177
683	General Area HPCI Pipe Routing Area RCIC Pipe Routing Area	120 167 167
670	General Area	120
645	HPCI-Equip Area HPCI-Emerg Area Cooler	167 167
645	RCIC-Emerg Area Cooler RCIC-Equip Area	167 167
645	RHR Equip Area 1	142
645	RHR Equip Area 2	142
645	CS Pump Room A	142
645	CS Pump Room B	142
645	RB Sump Room	125

Table F-4

Max Safe Reactor Building Radiation Monitors

RB Area Elevation (ft)	**ARM Number**	**Arm Channel Description**	**Max Safe Radiation Levels Per EO-000-104 (R/HR)**
818	49	Refuel Floor Area	10
749	52 54	RWCU Recirc PP Access Fuel Pool Pump Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C Pump Room RHR B D Pump Room	10

BLE M - SYSTEMS MALFUNCTIONS

Category	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOTICE OF UNUSUAL EVENT
Loss of AC Power	**MG1** (Pg 79) IC **Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.** Modes: 1, 2, 3 <u>EAL Threshold Value</u> Loss of power from Startup Transformer 10 <u>AND</u> 20 to either unit. <u>AND</u> All 4.16 kV ESS Busses on either unit are de-energized. <u>AND</u> A. Restoration of at least two 4.16 kV ESS Busses on each unit within 4 hours is not likely. <u>OR</u> B. RPV Water Level <[redacted]".	**MS1** (Pg 81) IC **Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.** Modes: 1, 2, 3 <u>EAL Threshold Value</u> Loss of power from Startup Transformer 10 <u>AND</u> 20 to either unit. <u>AND</u> All 4.16 kV ESS Busses on either unit are de-energized. <u>AND</u> Failure to restore power to at least two 4.16KV ESS busses on each unit within fifteen minutes from the time of loss of both offsite and ON SITE AC power.	**MA1** (Pg 83) IC **AC Power Capability to Essential Busses Reduced to a Single Power Source for Greater than 15 Minutes such that Any Additional Single Failure Would Result in Station Blackout.** Modes: 1, 2, 3 <u>EAL Threshold Value</u> (1 or 2) 1. Loss of power from Startup Transformer 10 <u>AND</u> 20 to either unit for > 15 minutes. <u>AND</u> Onsite AC power is reduced to a single 4.16 kV ESS Bus on either unit. <u>OR</u> 2. Loss of power from Startup Transformer 10 <u>OR</u> 20 to either unit for > 15 minutes. <u>AND</u> Onsite AC power is not available.	**MU1** (Pg 85) IC **Loss of all Offsite Power to Essential Busses for Greater than 15 Minutes.** Modes: 1, 2, 3 <u>EAL Threshold Value</u> Loss of power from Startup Transformer 10 <u>AND</u> 20 to either unit for > 15 minutes.
Loss of DC Power		**MS2** (Pg 86) Loss of All Vital DC Power. Modes: 1, 2, 3 <u>EAL Threshold Value</u> Loss of all vital DC power to either unit based on less than 105 volts on the 125 VDC main distribution busses 1D612 (2D612), 1D622 (2D622), 1D632 (2D632), <u>AND</u> 1D642 (2D642) for > 15 minutes. NOTE: Busses do not trip on undervoltage condition.		
Failure of Reactor Protection System	**MG3** (Pg 87) IC **Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and there is Indication of an Extreme Challenge to the Ability to Cool the Core.** Modes: 1, 2 <u>EAL Threshold Value</u> Indication(s) exist that indicate that Reactor Protection System setpoint was exceeded. <u>AND</u> RPS, ARI, and Manual Scram/ARI fail to initiate and complete a scram that reduces reactor power to < [redacted]%. <u>AND</u> A. Reactor water level cannot be maintained > [redacted]". <u>OR</u> B. The combination of RPV Pressure and Suppression Pool Temperature cannot be maintained below the HCTL curve, Figure M-1. NOTE: Although the HCTL curve is not evaluated in EO-000-103 until the reactor is shutdown, the curve must be used to consider entry into this EAL.	**MS3** (Pg 91) IC **Failure of Reactor Protection System to Complete or Initiate an Automatic Reactor Scram once a Reactor Protection System Setpoint has been Exceeded and Manual Scram was NOT Successful.** Modes: 1, 2 <u>EAL Threshold Value</u> Indication(s) exist that indicate that Reactor Protection System setpoint was exceeded. <u>AND</u> RPS, ARI, and Manual Scram/ARI fail to initiate and complete a scram that reduces reactor power to < [redacted]%.	**MA3** (Pg 93) IC **Failure of Reactor Protection System to Complete or Initiate an Automatic Reactor Scram once a Reactor Protection System Setpoint has been Exceeded and Manual Scram was Successful.** Modes: 1, 2 <u>EAL Threshold Value</u> Indication(s) exist that indicate that Reactor Protection System setpoint was exceeded <u>AND</u> RPS automatic scram did not reduce reactor power to < [redacted]%. <u>AND</u> A Manual Scram or ARI initiates and reduces reactor power to < [redacted]%.	

TABLE M - SYSTEMS MALFUNCTIONS

Loss of Communications				**MU8** (Pg 106) **IC UNPLANNED Loss of all ON SITE or Offsite Communications Capabilities.** Modes: 1, 2, 3 **EAL Threshold Value** (1 or 2) 1. UNPLANNED loss of all ON SITE communications capability per Table M-1 affecting the ability to perform routine operations. OR 2. UNPLANNED loss of all offsite communications capability per Table M-1.
Technical Specifications				**MU9** (Pg 108) **IC Inability to Reach Required Shutdown within Technical Specification Limits.** Modes: 1, 2, 3 **EAL Threshold Value** Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.
Inadvertent Criticality				**MU10** (Pg 109) **IC Inadvertent Criticality.** Modes: 3 **EAL Threshold Value** An UNPLANNED extended positive period observed on nuclear instrumentation.

Figure M-1
HEAT CAPACITY TEMPERATURE LIMIT

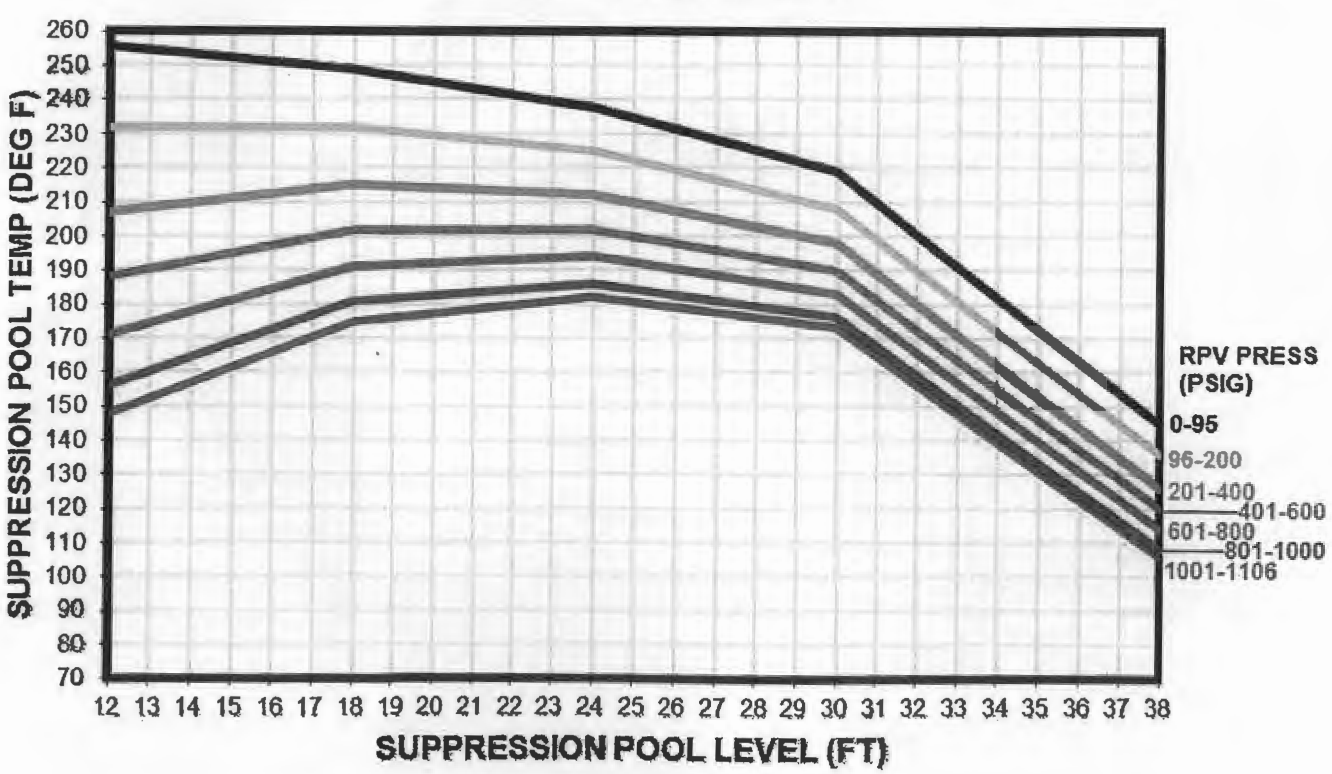


TABLE M-1 ON SITE/OFFSITE COMMUNICATIONS CAPABILITY		
SYSTEM	ON SITE	OFFSITE
UHF Radio	X	
Commercial Telephone Systems	X	X
Loss of dedicated conference lines to offsite agencies		X
FTS-2001 (ENS)		X
Plant PA System	X	
Plant Cellular Telephone	X	
Telecopy Transmittal		X
Sound Powered Phones	X	

Table M-2 Significant transients
SCRAM
Reactor Recirculation System Runback (>25% thermal power changes)
ECCS Initiations
UNPLANNED Thermal Power Changes > 10%
Load Reject > 25% electrical load

Table M-3 Safety System Annunciators/Indicators
ECCS
Containment Isolation
Reactor Trip
Process or Effluent Radiation Monitors
Electrical Distribution/Diesel Generators

Table M-4 Safety Function Indicators
Reactor Power-Nuclear Instrumentation Displays, Full Core Display, SIP panel displays
Decay Heat Removal-valve and pump indications for RHR, Core Spray, HPCI, RCIC, Suppression Pool level and Temperature
Containment Safety Functions-Pressure indication, Hydrogen/Oxygen concentrations, radiation levels,
Coolant System integrity-RPV level, RPV pressure, Containment pressure, Containment Radiation level

Table C Cold Shutdown/Refueling System Malfunctions

Category	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOTICE OF UNUSUAL EVENT
Loss of AC Power			**CA1** (Pg 162) IC **Loss of All Offsite Power and Loss of All ON SITE AC Power to Essential Busses.** Modes: 4, 5, D EAL Threshold Value Loss of power from Startup Transformer 10 AND 20 to either unit. AND All 4.16 kV ESS Busses on either unit are de-energized. AND Failure to restore power to at least two 4.16KV ESS busses on each unit within fifteen minutes from the time of loss of both offsite and ON SITE AC power.	**CU1** (Pg 163) IC **Loss of All Offsite Power to Essential Busses for Greater than 15 Minutes.** Modes: 4, 5 EAL Threshold Value Loss of power from Startup Transformer 10 AND 20 to either unit for > 15 minutes.
Loss of DC Power				**CU2** (Pg 164) IC **UNPLANNED Loss of Required DC Power for Greater than 15 Minutes.** Modes: 4, 5 EAL Threshold Value UNPLANNED loss of all vital DC power to either unit based on less than 105 volts on the 125 VDC main distribution busses 1D612 (2D612), 1D622 (2D622), 1D632 (2D632), AND 1D642 (2D642) AND Failure to restore power to one required DC bus within 15 minutes from the time of loss. Note: Busses do not trip on undervoltage condition.
Decay Heat Removal			**CA3** (Pg 165) IC **Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV.** Modes: 4, 5 EAL Threshold Value (1 or 2 or 3) 1. With Secondary Containment and RCS integrity[1] not established, an UNPLANNED event results in RCS temperature > [redacted]°F. OR 2. With Secondary Containment established and RCS integrity[1] not established, an UNPLANNED event results in RCS temperature > [redacted]°F for > 20 minutes[2]. OR 3. An UNPLANNED event results in RCS temperature > 200°F for > 60 minutes[2] or results in an RCS pressure increase of greater than 20 psig. [1]NOTE: By definition, in Mode 5 RCS integrity is not established. [2]NOTE: If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.	**CU3** (Pg 167) IC **UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV.** Modes: 4, 5 EAL Threshold Value (1 or 2) 1. An UNPLANNED event results in RCS temperature > 200°F, the Technical Specification cold shutdown temperature limit. OR 2. Loss of all RCS temperature and RPV level indication for > 15 minutes.

Table C Cold Shutdown/Refueling System Malfunctions

Category	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOTICE OF UNUSUAL EVENT
RCS Leakage/RCS Draindown	**CG4** (Pg 169) IC **Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV.** Modes: 4, 5 **EAL Threshold Value** (1 and 2 and 3) 1. Loss of RPV inventory as indicated by unexplained Drywell or Reactor Building Sump or Drywell Equipment Drain Tank level increase, or Suppression Pool level increase, or other indications of loss of RPV inventory, **AND** 2. RPV Level: A. < [redacted]" (TAF) for > 30 minutes **OR** B. RPV level cannot be monitored concurrent with indication of core uncovery for > 30 minutes as evidenced by one of the following: • Containment High Range Rad Monitor reading greater than or equal to 10R/hr, • Erratic Source Range Monitor Indication • Visual indication. **AND** 3. Indication of Containment challenged as indicated by one or more of the following: • Drywell Pressure > 53 psig and increasing OR • Drywell Hydrogen or Suppression Chamber Hydrogen > 6% AND Drywell Oxygen or Suppression Chamber Oxygen > 5% OR • Secondary Containment not established OR • Any Reactor Building area exceeds Max Safe Radiation Levels per Table C-1	**CS4** (Pg 173) IC **Loss of RPV Inventory Affecting Core Decay Heat Removal Capability.** Mode: 4 **EAL Threshold Value** (1 or 2) 1. With secondary Containment NOT established: (a or b) a. Loss of RPV inventory as indicated by RPV level < - 135" [redacted]. **OR** b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by: • Unexplained Drywell or Reactor Building Sump or Drywell Equipment Drain Tank level increase, or • Unexplained Suppression Pool level increase, or • Other unexplained indications of loss of RPV inventory **OR** 2. With secondary Containment established: (a or b) a . Loss of RPV inventory as indicated by RPV level [redacted]" (TAF) **OR** b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by either: • Unexplained Drywell or Reactor Building Sump or Drywell Equipment Drain Tank level increase, or • Unexplained Suppression Pool level increase, or • Other unexplained indications of loss of RPV inventory • Erratic Source Range Monitor Indication	**CA4** (Pg 175) IC **Loss of RCS Inventory.** Mode: 4 **EAL Threshold Value** (1 or 2) 1. Loss of RCS inventory as indicated by RPV level < -129" **OR** 2. Loss of RCS inventory as indicated by unexplained Drywell or Reactor Building Sump or Drywell Equipment Drain Tank level increase, or Suppression Pool level increase, or other indications of loss of RPV inventory, **AND** RCS level cannot be monitored for > 15 minutes.	**CU4** (Pg 177) IC **RCS Leakage.** Mode: 4 **EAL Threshold Value** (1 or 2) 1. Unidentified or pressure boundary leakage > 10 gpm. **OR** 2. Identified leakage > 25 gpm.

Table C Cold Shutdown/Refueling System Malfunctions

Category	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NOTICE OF UNUSUAL EVENT
Fuel Clad Degradation				**CU6** (Pg 185) **IC Fuel Clad Degradation.** Modes: 4, 5 EAL Threshold Value (1 or 2) 1. UNPLANNED increase in the following radiation monitor readings: • Refuel Floor radiation Monitor > 750 mr/hr • Refuel Platform ARM > 750 mr/hr • Refuel Floor Continuous Air Monitor > 300 DAC OR 2. Reactor coolant activity, determined by sample analysis ≥4 μCi/gm of I-131dose equivalent.
Loss of Communications				**CU7** (Pg 187) **IC UNPLANNED Loss of all ON SITE or Offsite Communications Capabilities.** Modes: 4, 5 EAL Threshold Value (1 or 2) 1. UNPLANNED loss of all ON SITE communications capability per Table C-2 affecting the ability to perform routine operations. OR 2. UNPLANNED loss of all offsite communications capability per Table C-2.
Inadvertent Criticality				**CU8** (Pg 189) **IC Inadvertent Criticality.** Modes: 4, 5 EAL Threshold Value An UNPLANNED extended positive period observed on nuclear instrumentation.

Table C-1

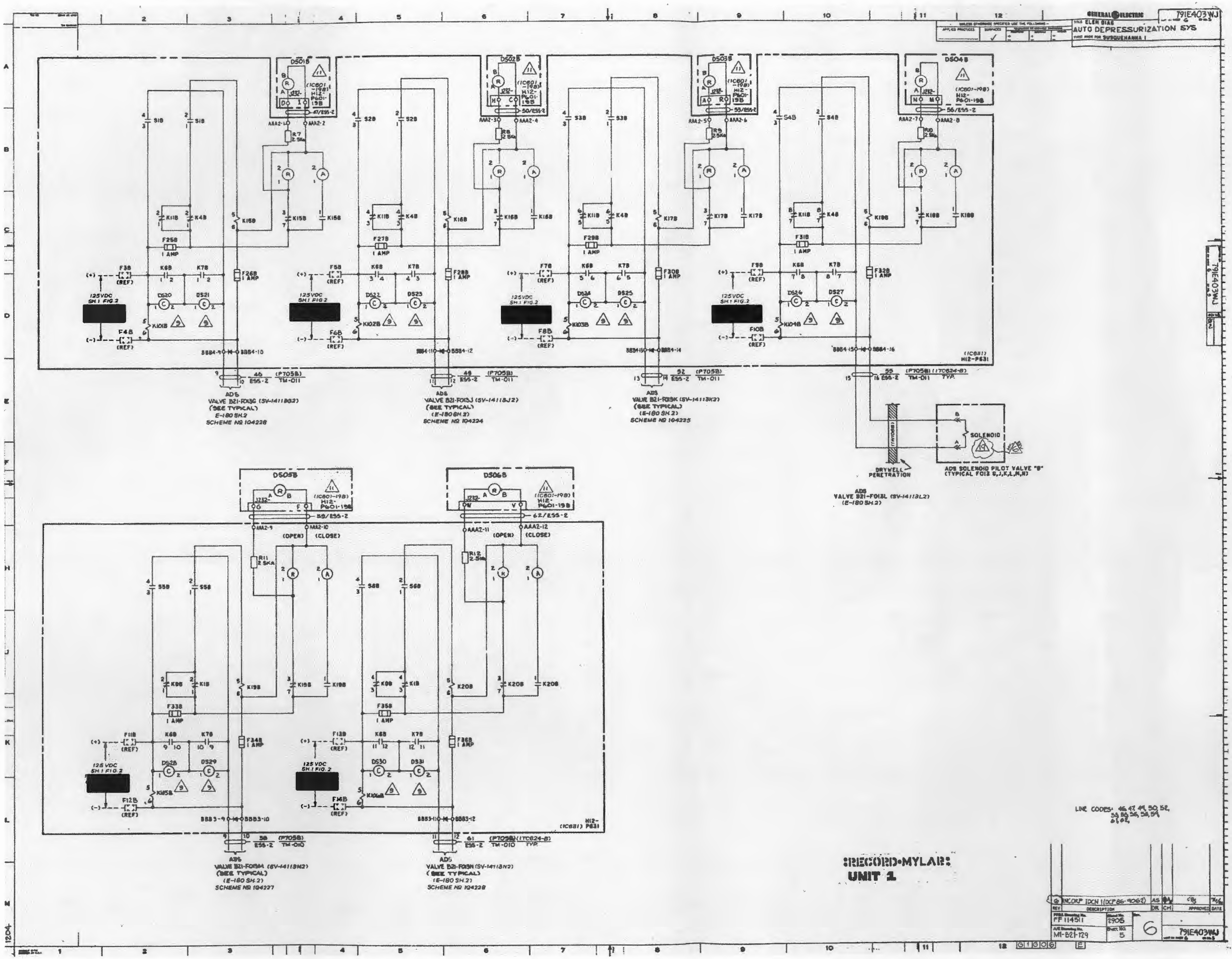
Reactor Building Radiation Monitors

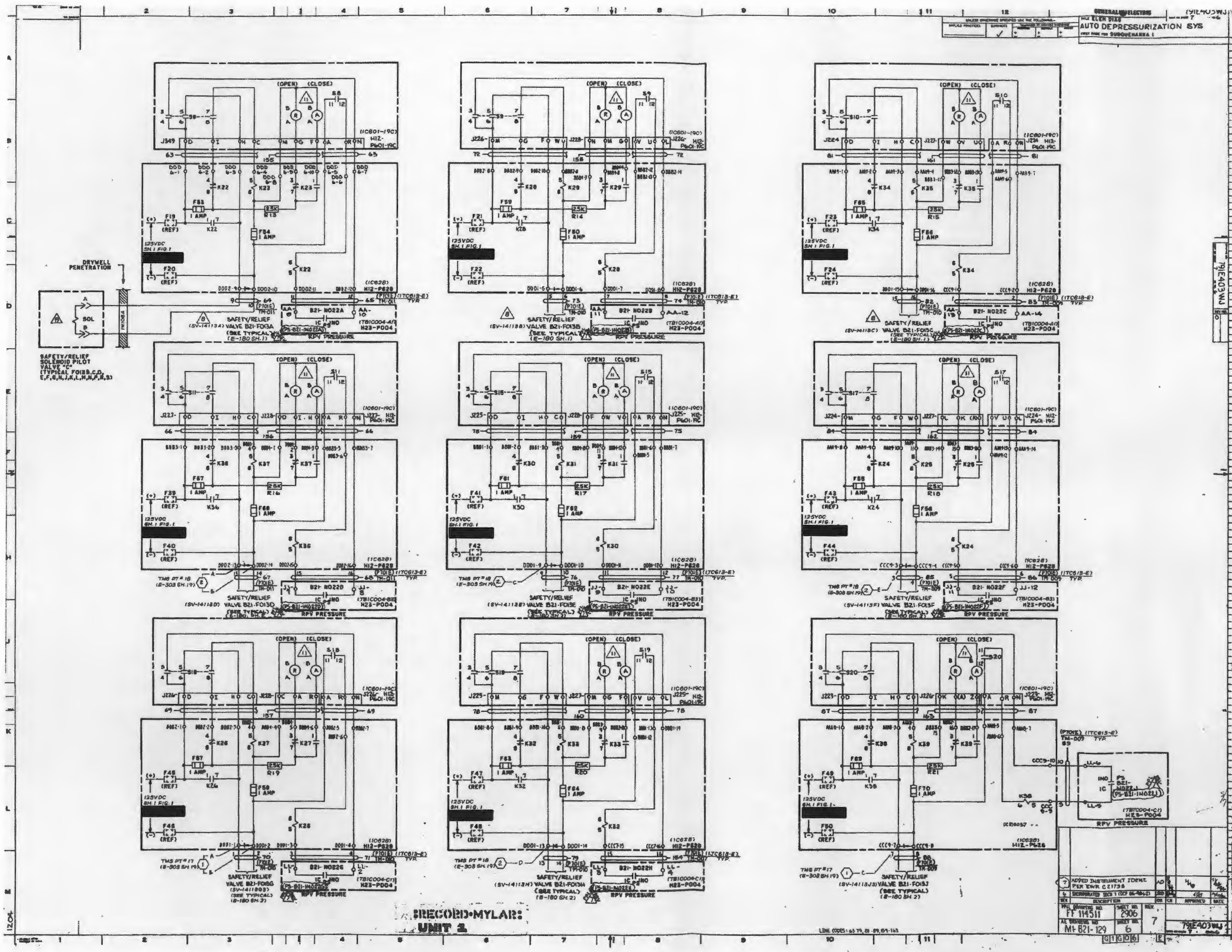
	ARM Number		Max Safe Radiation Levels Per EO-000-104 (R/HR)
RB Area Elevation (ft)	High Range	ARM Channel Description	
818	49	Refuel Floor Area	10
749	52 54	RWCU Recirc PP Fuel Pool PP Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	10

Table C-2

ON SITE/Offsite Communications Capability

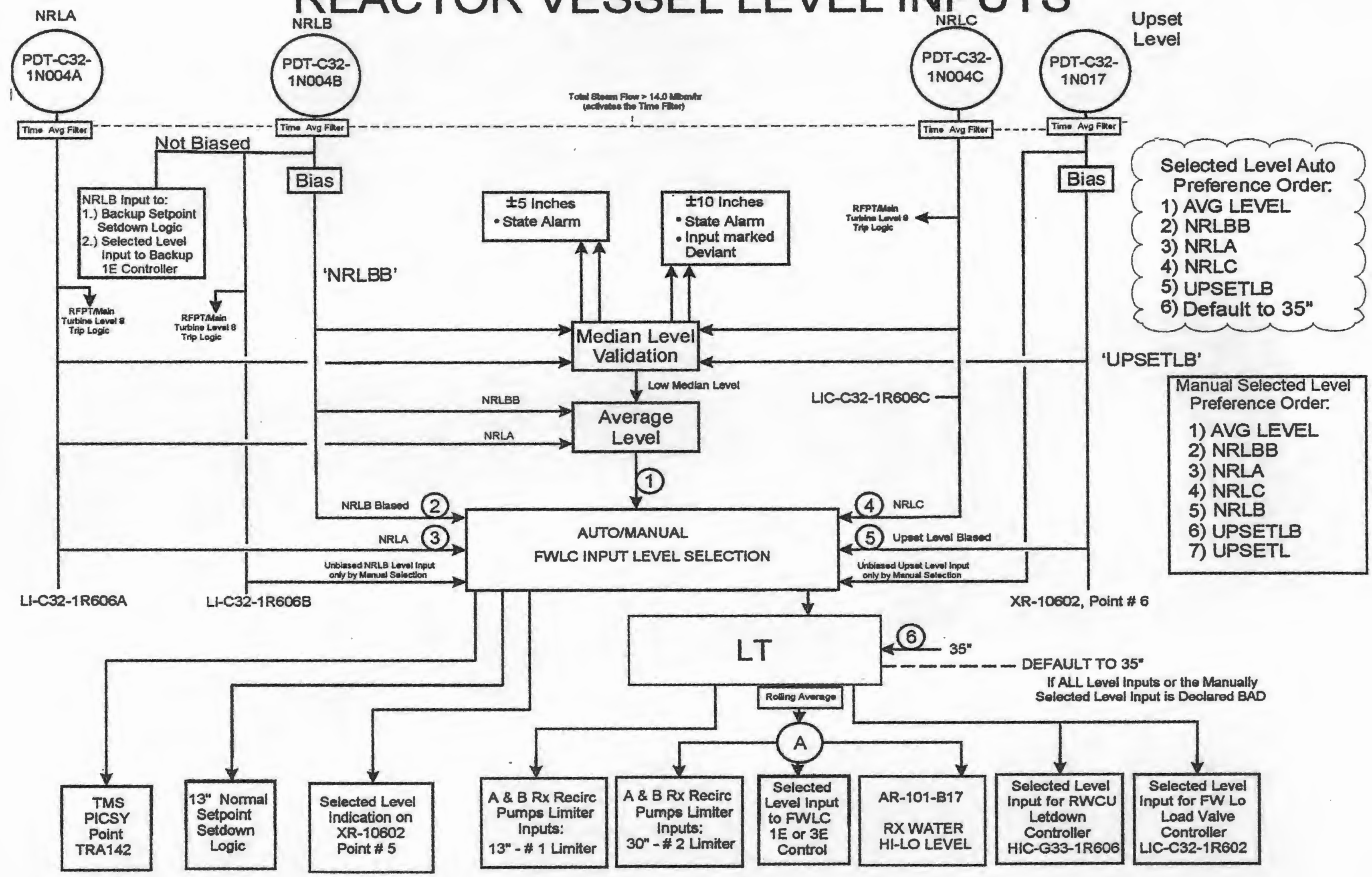
SYSTEM	ON SITE	OFF SITE
UHF Radio	X	
Commercial telephone systems	X	X
Loss of dedicated conference lines to the offsite agencies		X
FTS-2001 (ENS)		X
Plant PA System	X	
Plant cellular telephones	X	
Telecopy Transmittal		X
Sound powered phones	X	


GENERAL ELECTRIC
791E403WJ
ELEM DIAG
AUTO DEPRESSURIZATION SYS
FIRST MADE FOR SUSQUEHANNA I
DS01B
DS02B
DS03B
DS04B
DS05B
DS06B
125VDC SH.1 FIG.2
[redacted]
ADS VALVE B21-F013C (SV-14113G2) (SEE TYPICAL) (E-180 SH.2) SCHEME NO 104228
ADS VALVE B21-F013J (SV-14113J2) (SEE TYPICAL) (E-180 SH.2) SCHEME NO 104224
ADS VALVE B21-F013K (SV-14113K2) (SEE TYPICAL) (E-180 SH.2) SCHEME NO 104225
ADS SOLENOID PILOT VALVE "B" (TYPICAL F013 G,J,K,L,M,N)
DRYWELL PENETRATION
ADS VALVE B21-F013L (SV-14113L2) (E-180 SH.2)
ADS VALVE B21-F013M (SV-14113M2) (SEE TYPICAL) (E-180 SH.2) SCHEME NO 104227
ADS VALVE B21-F013N (SV-14113N2) (SEE TYPICAL) (E-180 SH.2) SCHEME NO 104228
LINE CODES: 46,47,49,50,52, 53,55,56,58,59, 61,62,
RECORD-MYLAR
UNIT 1
6 INCORP IDCN 1 (DCP 86-9062) AS
REV DESCRIPTION DR CH APPROVED DATE
FF 114511
2905
M1-B21-129
SHEET NO. 5
6
791E403WJ
G1G06
E
1204



GENERAL ELECTRIC
ELEM DIAG
AUTO DEPRESSURIZATION SYS
SUSQUEHANNA 1
791E403WJ
(OPEN) (CLOSE)
DRYWELL PENETRATION
SAFETY/RELIEF SOLENOID PILOT VALVE "C" (TYPICAL F013B,C,D, E,F,G,H,J,K,L,M,N,P,R,S)
125VDC SH.1 FIG.1
SAFETY/RELIEF VALVE B21-F013A
SAFETY/RELIEF VALVE B21-F013B
SAFETY/RELIEF VALVE B21-F013C
RPV PRESSURE
7 ADDED INSTRUMENT IDENT. PER EWR C21738
REV. DESCRIPTION APPROVED DATE
FF 114511
2906
7
MI-B21-129
6
G1G06
E
RECORD-MYLAR
UNIT 1
1204

REACTOR VESSEL LEVEL INPUTS



3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than [redacted] OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than [redacted] OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 AND 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	200 days cumulative operation in MODE 1
SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell <u>AND</u> Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

-----NOTES-----

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 05. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig
45	0.52
39	0.86
25	1.91
05	3.44

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when <800 psig are within established limits.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ [redacted] psig.	A.1 ---------------NOTE--------------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -------------------------------------	
	Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>	
	A.2 Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ [redacted] psig.	B.1 Restore charging water header pressure to ≥ [redacted] psig.	[redacted] discovery of Condition B concurrent with charging water header pressure < [redacted] psig
	<u>AND</u>	
	---------------NOTE--------------- B.2.1 Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ------------------------------------	
	Declare the associated control rod scram time "slow."	1 hour
	<u>OR</u>	
	B.2.2 Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. [redacted]	C.1 [redacted]	[redacted]
	AND	
	C.2 [redacted]	[redacted]
D. [redacted]	D.1 ---------------NOTE--------------- [redacted] --- [redacted]	[redacted]

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.d, 6.b, 7.a, and 7.b AND 24 hours for Functions other than Functions 2.a, 2.d, 6.b, 7.a and 7.b
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	Required Action and associated Completion Time of Condition A or B not met.	C.1	Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D.	As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1	Isolate associated main steam line (MSL).	12 hours
		<u>OR</u>		
		D.2.1	Be in MODE 3.	12 hours
		<u>AND</u>		
		D.2.2	Be in MODE 4	36 hours
E.	As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1	Be in MODE 2.	6 hours
F.	As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1	Isolate the affected penetration flow path(s).	1 hour
G.	As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	G.1	Isolate the affected penetration flow path(s).	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1	I.1 Declare associated standby liquid control subsystem (SLC) inoperable. OR I.2 Isolate the Reactor Water Cleanup System.	1 hour 1 hour
J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	J.1 Initiate action to restore channel to OPERABLE status. OR J.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2	----- 1. A test of all required contacts does not have to be performed 2. For Functions 2.e, 3.a, and 4.a, a test of all required relays does not have to be performed ----- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.6.1.6	-------------------------NOTE------------------------- 1. For Function 1.b. channel sensors are excluded. 2. Response time testing of isolating relays is not required for Function 5.a. -- Verify the ISOLATION SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -136 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 841 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 179 psid
d. Condenser Vacuum - Low	1 2[a], 3[a]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≥ 8.8 inches Hg vacuum
e. Reactor Building Main Steam Tunnel Temperature - High	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 184°F
f. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

(a) With any main turbine stop valve not closed.

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≥ 11.5 inches
b. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -45 inches
c. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥-136 inches
d. Drywell Pressure - High	1,2,3	2	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
e. SGTS Exhaust Radiation - High	1,2,3	1	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 31 mR/hr
f. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Δ Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 383 inches H_2O
b. HPCI Steam Supply Line Pressure - Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 20 psig
d. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤1.88 psig
e. HPCI Pipe Routing Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 174°F
f. HPCI Equipment Room Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 174°F
g. HPCI Emergency Area Cooler Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 174°F
h. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Δ Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤193 inches H_2O
b. RCIC Steam Supply Line Pressure - Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 20 psig
d. Drywell Pressure – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 1.88 psig
e. RCIC Pipe Routing Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 174°F
f. RCIC Equipment Room Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 174°F
g. RCIC Emergency Area Cooler Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 174°F
h. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5.	Reactor Water Cleanup (RWCU) System Isolation					
a.	RWCU Differential Δ Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 67 gpm
b.	RWCU Penetration Area Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 137°F
c.	RWCU Pump Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 154°F
d.	RWCU Heat Exchanger Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 154°F
e.	SLC System Initiation	1,2,3	2[(b)]	I	SR 3.3.6.1.5	NA
f.	Reactor Vessel Water Level -Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -45 inches
g.	RWCU Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 472 gpm
h.	Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(continued)

(b) SLC System Initiation only inputs into one of the two trip systems.

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≤ 108 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[(c)]	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≥ 11.5 inches
c. Manual Initiation	3,4,5	1[(c)]	G	SR 3.3.6.1.5	NA
7. Traversing Incore Probe Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5	≥ 11.5 inches
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig

(c) Only one trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.10 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE----------- Required Action A.2 shall be completed if this Condition is entered. ------------------------------ Requirements of the LCO not met in MODES 1, 2, and 3.	A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTE----------- Required Action C.2 shall be completed if this Condition is entered. ------------------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	------------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify: a. RCS pressure and RCS temperature are to the right of the most limiting curve specified in Figures 3.4.10-1 through 3.4.10-3; and b. --------------------NOTE-------------------- Only applicable when governed by Figure 3.4.10-2, Curve B, and Figure 3.4.10-3, Curve C. -- RCS heatup and cooldown rates are ≤ 100°F in any one hour period; and c. --------------------NOTE-------------------- Only applicable when governed by Figure 3.4.10-1, Curve A. -- RCS heatup and cooldown rates are ≤ 20°F in any one hour period.	30 minutes
SR 3.4.10.2	Verify RCS pressure and RCS temperature are to the right of the criticality limit (Curve C) specified in Figure 3.4.10-3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.10.3	-------------------------NOTE------------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.10.4	-------------------------NOTE------------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.10.5	-------------------------NOTE------------------------- Only required to be met in single loop operation when: a. THERMAL POWER ≤ 27% RTP; or b. The operating recirculation loop flow ≤ 21,320 gpm. -- Verify the difference between the bottom head coolant temperature and the RPV coolant temperature is ≤ 145°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.10.6	-------------------------NOTE------------------------- Only required to be met in single loop operation when the idle recirculation loop is not isolated from the RPV, and: a. THERMAL POWER ≤ 27% RTP; or b. The operating recirculation loop flow ≤ 21,320 gpm. -- Verify the difference between the reactor coolant temperature in the recirculation loop not in operation and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow.
SR 3.4.10.7	-------------------------NOTE------------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ≥ 70°F.	30 minutes
SR 3.4.10.8	-------------------------NOTE------------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are ≥ 70°F.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.10.9	---------------------------NOTE--------------------------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. ---	
	Verify reactor vessel flange and head flange temperatures are ≥ 70°F.	12 hours

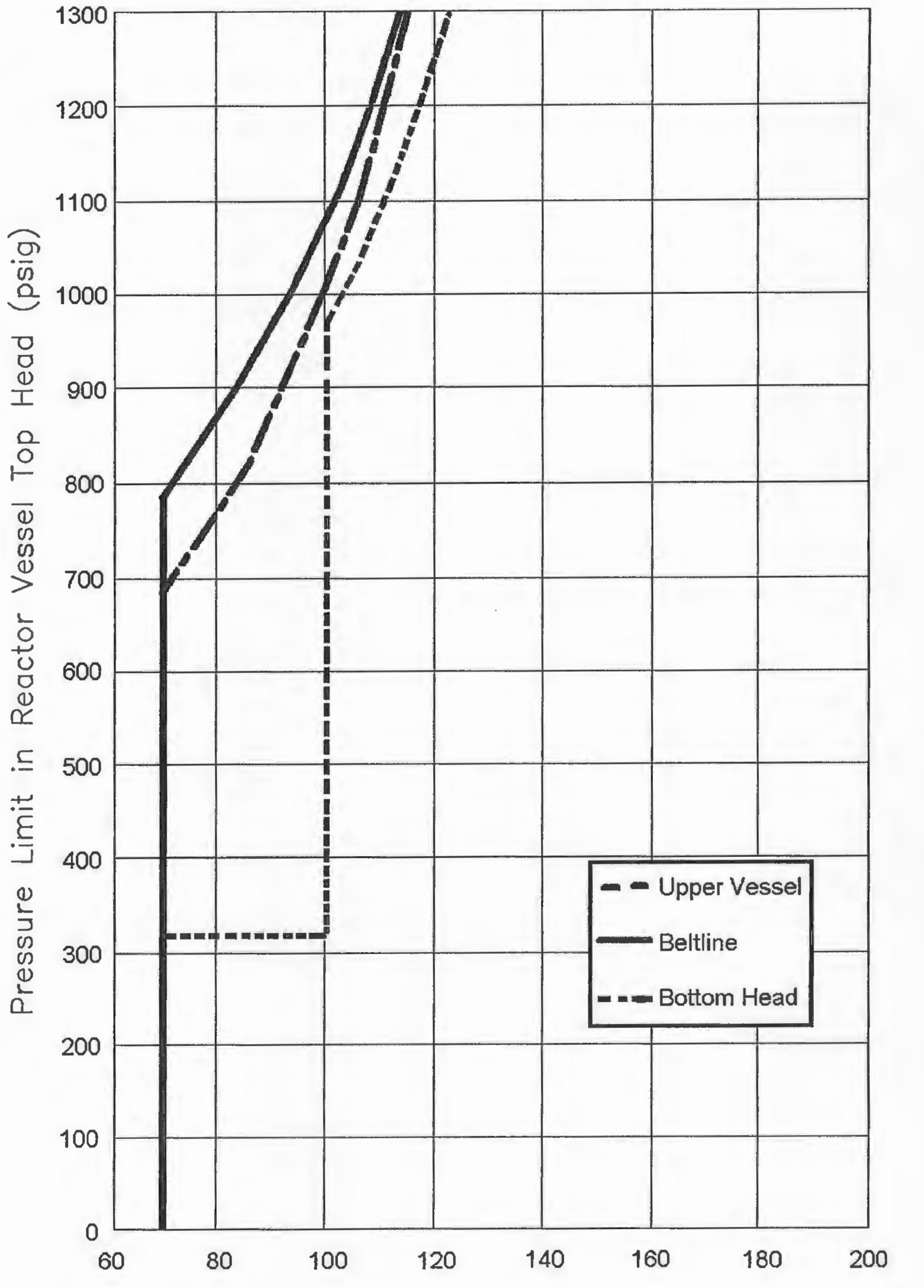


FIGURE 3.4.10-1
System Hydrotest Limit with Fuel in Vessel for 35.7 EFPY
(Curve A)

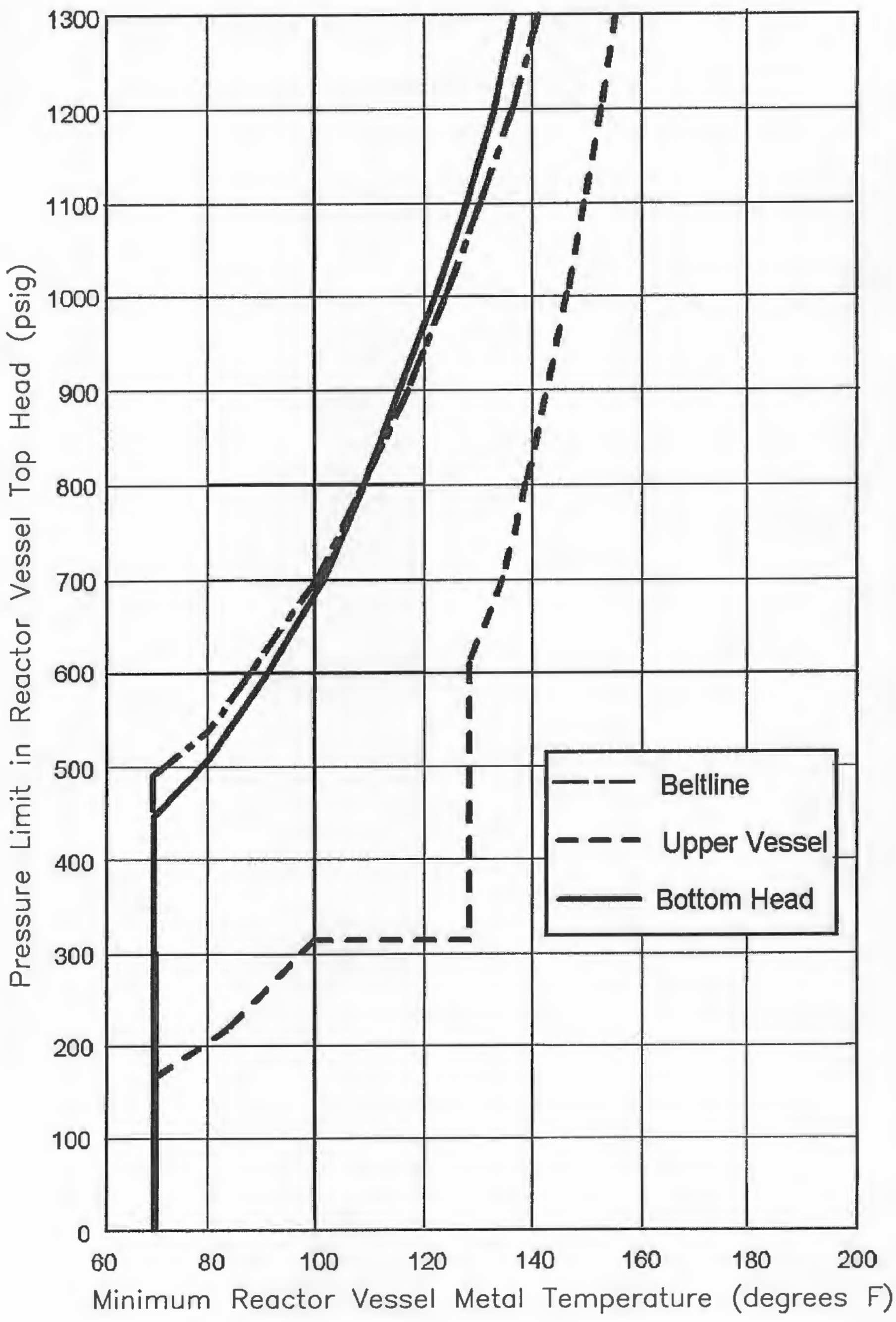


FIGURE 3.4.10-2
Non-Nuclear Heating Limit for 35.7 EFPY
(Curve B)

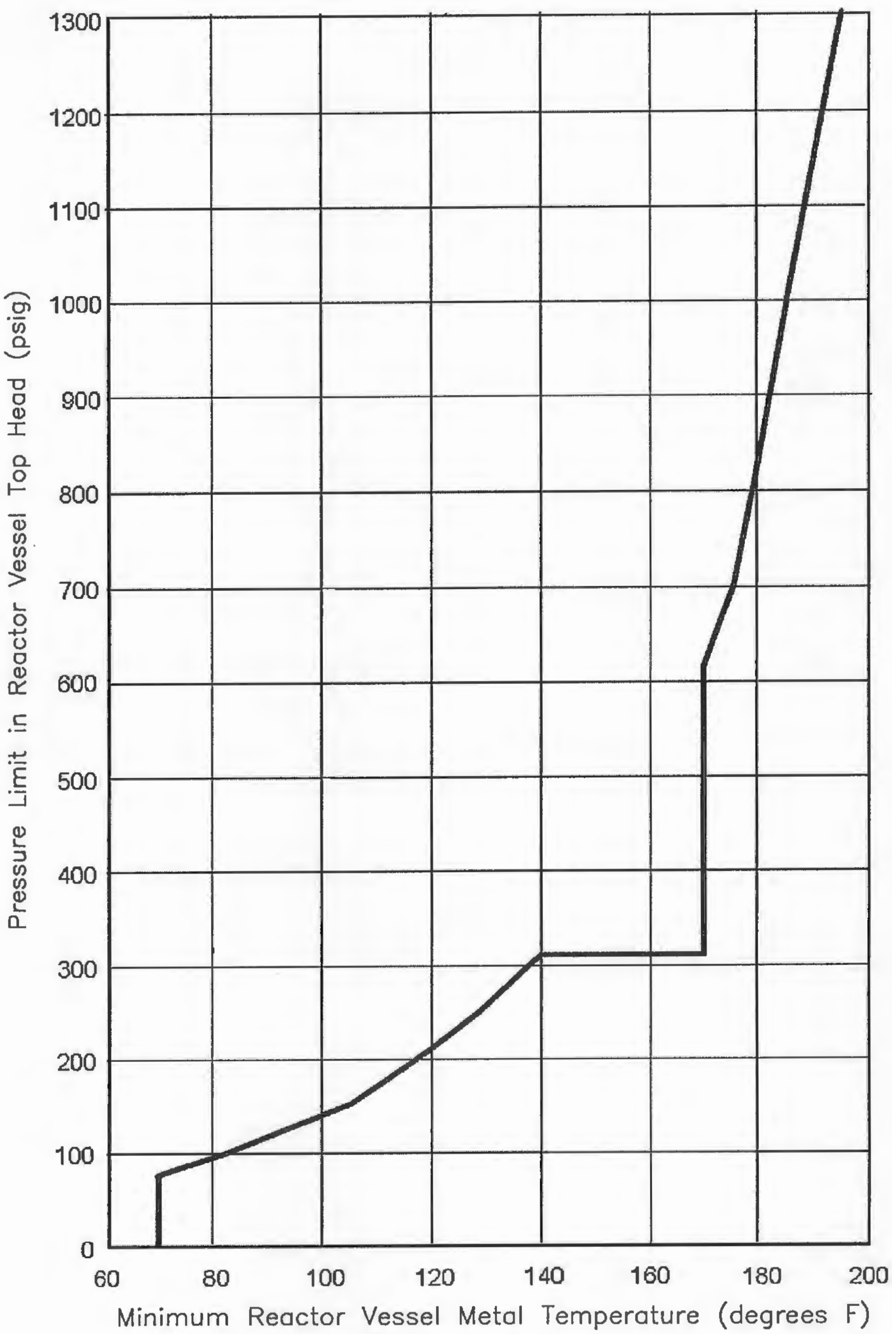


FIGURE 3.4.10–3
Nuclear (Core Critical) Limit for 35.7 EFPY
(Curve C)

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ------------NOTE------------ Only applicable to penetration flow paths with two PCIVs except for the H_2O_2 Analyzer penetrations. ------------------------------ One or more penetration flow paths with one PCIV inoperable except for purge valve leakage not within limit.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours except for main steam line AND 8 hours for main steam line
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. --------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment <u>AND</u> Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------NOTE------------- Only applicable to penetration flow paths with two PCIVs except for the H_2O_2 Analyzer penetrations. ----------------------------------- One or more penetration flow paths with two PCIVs inoperable except for purge valve leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. -------------NOTE------------- Only applicable to penetration flow paths with only one PCIV. ----------------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours except for excess flow check valves (EFCVs) <u>AND</u> 12 hours for EFCVs
	<u>AND</u>		
	C.2	-------------NOTE------------- Isolation devices in high radiation areas may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. -------------NOTE-------------- Only applicable to the H_2O_2 Analyzer penetrations. ------------------------------------ One or more H_2O_2 Analyzer penetrations with one or two PCIVs inoperable.	D.1	Isolate the affected penetration flow path by the use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	AND		
	D.2	Verify the affected penetration flow path is isolated.	Once per 31 days
E. Secondary containment bypass leakage rate not within limit.	E.1	Restore leakage rate to within limit.	4 hours
F. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limit.	F.1	Restore the valve leakage to within valve leakage limit.	24 hours
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met in MODE 1, 2, or 3.	G.1	Be in MODE 3.	12 hours
	AND		
	G.2	Be in MODE 4.	36 hours
H. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met for PCIV(s) required to be OPERABLE during MODE 4, 5 or Operations with the potential for draining the reactor vessel (OPDRVs).	H.1	Initiate action to suspend OPDRVs.	Immediately
	OR		
	H.2	Initiate action to restore valve(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.1	----------NOTES---------- 1. Only required to be met in MODES 1, 2, and3. 2. Not required to be met when the 18 and 24 inch primary containment purge valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. ------------------------------- Verify each 18 and 24 inch primary containment purge valve is closed.	31 days
SR 3.6.1.3.2	----------NOTES---------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. ------------------------------- Verify each primary containment isolation manual valve and blind flange that is located outside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.3	-----------------------------NOTES----------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity for each of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated and each automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	-------------------------NOTE------------------------- Only required to be met in MODES 1, 2 and 3. -- Perform leakage rate testing for each primary containment purge valve with resilient seals.	24 months
SR 3.6.1.3.7	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.9	Verify a representative sample of reactor instrumentation line EFCVs actuate to check flow on a simulated instrument line break.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.11	---------------------------NOTES--------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 15 scfh when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.3.12	---------------------------NOTES--------------------------- Only required to be met in MODES 1, 2, and 3. --- Verify leakage rate through each MSIV is ≤ 100 scfh and ≤300 scfh for the combined leakage including the leakage from the MS Line Drains, when the MSIVs are tested at ≥ 24.3 psig or P_a and the MS Line Drains are tested at P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.13	---------------------------NOTE--------------------------- Only required to be met in MODES 1, 2, and 3. -- Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program.

3.4 Reactor Coolant System

3.4.1 Reactor Coolant System Chemistry

TRO 3.4.1 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.1-1 and the conductivity recorder shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

--- NOTE ---

1. The provisions of TRO 3.0.4 are not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A Conductivity not within the limits specified in Table 3.4.1-1.	A.1 Perform TRS 3.4.1.1 and TRS 3.4.1.3. <u>AND</u> A.2 Perform TRS 3.4.1.4	Once per 8 hours Once per 24 hours
B. Conductivity or Chlorides not within the limits specified in Table 3.4.1-1 in MODE 1 but with conductivity less than 10 μmho/cm at 25°C and chloride concentration less than 0.5 ppm.	B.1 Restore parameter within limits	72 hours <u>AND</u> ≤ 336 hours/year cumulative time exceeding the limit
C. pH not within limits of Table 3.4.1-1 in MODE 1.	C.1 Restore pH within limits.	72 hours
D. Required Actions and associated Completion Times of Conditions B or C not met.	D.1 Be in MODE 2	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
------------------NOTE---------------- Required Actions shall be completed if this Condition is entered. -- E. Conductivity greater than or equal to 10 μmho/cm at 25°C or chloride concentration greater than or equal to than 0.5 ppm.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours ----------NOTE---------- Cooldown should be performed as rapidly as possible not to exceed the cooldown rate ---------------------------- 36 hours
F. Reactor Coolant Chemistry not within limits specified in Table 3.4.1-1 for MODE 2or 3	F.1 Restore parameter to within limits.	48 hours
G. Required Actions and Completion Time of Condition F not met.	G.1 Be in MODE 3 AND G.2 Be in MODE 4	12 hours 36 hours
H. -------------NOTE------------- Only applicable when in a MODE other than MODES 1, 2, or 3. ------------------------------------- Reactor Coolant Chemistry not within limits specified in Table 3.4.1-1.	H.1 Restore parameter to within limits	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------------- NOTE --------------- Required Actions shall be completed if this Condition is entered. -- I. Chloride concentration greater than 0.5 ppm for > 24 hours	I.1 Perform Engineering Evaluation of structural integrity	Prior to proceeding to MODE 3
J. Conductivity recorder inoperable.	J.1 Obtain in-line conductivity measurement or grab sample	----------NOTE--------- Applicable in MODES 1, 2, or 3. -------------------------- Once per 4 hours <u>AND</u> ----------NOTE---------- Applicable in other than MODES 1, 2, or 3. --------------------------- Once per 24 hours

TECHNICAL REQUIREMENT SURVEILLANCE

SURVEILLANCE		FREQUENCY
TRS 3.4.1.1	Analyze a sample of reactor coolant for chlorides	72 hours
TRS 3.4.1.2	Analyze a sample of reactor coolant for conductivity	72 hours
TRS 3.4.1.3	Analyze a sample of reactor coolant for pH	---------------NOTE--------------- Only required to be performed if reactor conductivity is greater than 1.0 μmho/cm at 25°C -- 72 hours
TRS 3.4.1.4	Perform CHANNEL CHECK of the continuous conductivity monitor	7 days

TABLE 3.4.1-1
REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

MODE	CONDUCTIVITY μmho/cm @ 25°C	CHLORIDE CONCENTRATION ppm	pH
1	≤ 1.0	≤ 0.2	$5.6 \leq pH \leq 8.6$
2 and 3	≤ 2.0	≤ 0.1	$5.6 \leq pH \leq 8.6$
At all times other than MODE 1, 2, or 3	≤ 10.0	≤ 0.5	$5.3 \leq pH \leq 8.6$