CHAPTER 16

TECHNICAL SPECIFICATIONS

Chapter 16 is completely replaced in Amendment No. . Due to the effort required to revise the format from the original submittal, previous amendments were submitted as interim documents. Therefore, revision lines are not shown on the revised pages.

INDEX

DEFINITIONS

SECTION	PAGE
1.1 ACTION	1-1
1.2 AXIAL SHAPE INDEX	1-1
1.3 AZIMUTHAL POWER TILT	1-1
1.4 CHANNEL CALIBRATION	1-1
1.5 CHANNEL CHECK	1-1
1.6 CHANNEL FUNCTIONAL TEST	1-1
1.7 CONTAINMENT INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.10 DOSE EQUIVALENT I-131	1-2
1.11 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY	1-2
1.12 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-2
1.13 FREQUENCY NOTATION	1-2
1.14 IDENTIFIED LEAKAGE	1-2
1.15 OPERABLE - OPERABILITY	1-3
1.16 OPERATIONAL MODE - MODE	1-3
1.17 PHYSICS TESTS	1-3
1.18 PLANAR RADIAL PEAKING FACTOR - F_{xy}	1-3
1.19 PRESSURE BOUNDARY LEAKAGE	1-3
1.20 RATED THERMAL POWER	1-3
1.21 REACTOR TRIP SYSTEM RESPONSE TIME	1-3
1.22 REPORTABLE OCCURRENCE	1-4
1.23 SHIELD BUILDING INTEGRITY	1-4
1.24 SHUTDOWN MARGIN	1-4
1.25 SOFTWARE	1-4
1.26 STAGGERED TEST BASIS	1-4
1.27 THERMAL POWER	1-4
1.28 UNIDENTIFIED LEAKAGE	1-4

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION		PAGE
2.1 SAFETY LIMITS		
2.1.1	REACTOR CORE	2-1
2.1.1.1	DNBR	2-1
2.1.1.2	PEAK LINEAR HEAT RATE	2-1
2.1.2	REACTOR COOLANT SYSTEM PRESSURE	2-1
2.2 LIMITING SAFETY SYSTEM SETTINGS		
2.2.1	REACTOR TRIP SETPOINTS	2-2
2.2.2	CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS	2-2

BASES

SECTION		PAGE
2.1 SAFETY LIMITS		
2.1.1	REACTOR CORE	B 2-1
2.1.2	REACTOR COOLANT SYSTEM PRESSURE	B 2-2
2.2 LIMITING SAFETY SYSTEM SETTINGS		
2.2.1	REACTOR TRIP SETPOINTS	B 2-3
2.2.2	CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS	B 2-7

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.0 APPLICABILITY		3/4 0-1
3/4.1 REACTIVITY CONTROL SYSTEMS		
3/4.1.1	BORATION CONTROL	
	SHUTDOWN MARGIN - $T_{cold} > 210°F$	3/4 1-1
	SHUTDOWN MARGIN - $T_{cold} \leq 210°F$	3/4 1-3
	MODERATOR TEMPERATURE COEFFICIENT	3/4 1-4
	MINIMUM TEMPERATURE FOR CRITICALITY	3/4 1-6
3/4.1.2	BORATION SYSTEMS	
	FLOW PATHS - SHUTDOWN	3/4 1-7
	FLOW PATHS - OPERATING	3/4 1-8
	CHARGING PUMPS - SHUTDOWN	3/4 1-9
	CHARGING PUMPS - OPERATING	3/4 1-10
	BORATED WATER SOURCES - SHUTDOWN	3/4 1-11
	BORATED WATER SOURCES - OPERATING	3/4 1-13
	BORON DILUTION ALARMS	3/4 1-15
3/4.1.3	MOVABLE CONTROL ASSEMBLIES	
	CEA POSITION	3/4 1-22
	POSITION INDICATOR CHANNELS - OPERATING	3/4 1-25
	POSITION INDICATOR CHANNELS - SHUTDOWN	3/4 1-26
	CEA DROP TIME	3/4 1-27
	SHUTDOWN CEA INSERTION LIMIT	3/4 1-28
	REGULATING CEA INSERTION LIMITS	3/4 1-29

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.2 POWER DISTRIBUTION LIMITS		
3/4.2.1	LINEAR HEAT RATE	3/4 2-1
3/4.2.2	PLANAR RADIAL PEAKING FACTORS	3/4 2-2
3/4.2.3	AZIMUTHAL POWER TILT	3/4 2-3
3/4.2.4	DNBR MARGIN	3/4 2-5
3.4.2.5	RCS FLOW RATE	3/4 2-9
3/4.2.6	REACTOR COOLANT COLD LEG TEMPERATURE	3/4 2-10
3/4.2.7	AXIAL SHAPE INDEX	3/4 2-12
3/4.2.8	PRESSURIZER PRESSURE	3/4 2-13
3/4.3 INSTRUMENTATION		
3/4.3.1	REACTOR PROTECTIVE INSTRUMENTATION	3/4 3-1
3/4.3.2	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	3/4 3-16
3/4.3.3	MONITORING INSTRUMENTATION	
	RADIATION MONITORING INSTRUMENTATION	3/4 3-36
	INCORE DETECTORS	3/4 3-37
	SEISMIC INSTRUMENTATION	3/4 3-38
	METEOROLOGICAL INSTRUMENTATION	3/4 3-39
	REMOTE SHUTDOWN SYSTEM INSTRUMENTATION	3/4 3-40
	POST-ACCIDENT MONITORING INSTRUMENTATION	3/4 3-43
	CHLORINE DETECTION SYSTEMS	3/4 3-46
	FIRE DETECTION INSTRUMENTATION	3/4 3-47
	LOOSE-PART DETECTION INSTRUMENTATION	3/4 3-48
3/4.3.4	TURBINE OVERSPEED PROTECTION	3/4 3-49

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.4 REACTOR COOLANT SYSTEM		
3/4.4.1	REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	
	STARTUP AND POWER OPERATION	3/4 4-1
	HOT STANDBY	3/4 4-2
	HOT SHUTDOWN	3/4 4-3
	COLD SHUTDOWN - LOOPS FILLED	3/4 4-5
	COLD SHUTDOWN - LOOPS NOT FILLED	3/4 4-6
3/4.4.2	SAFETY VALVES	
	SHUTDOWN	3/4 4-7
	OPERATING	3/4 4-8
3/4.4.3	PRESSURIZER	3/4 4-9
3/4.4.4	STEAM GENERATORS	3/4 4-10
3/4.4.5	REACTOR COOLANT SYSTEM LEAKAGE	
	LEAKAGE DETECTION SYSTEMS	3/4 4-11
	OPERATIONAL LEAKAGE	3/4 4-12
3/4.4.6	CHEMISTRY	3/4 4-13
3/4.4.7	SPECIFIC ACTIVITY	3/4 4-16
3/4.4.8	PRESSURE/TEMPERATURE LIMITS	
	REACTOR COOLANT SYSTEM	3/4 4-20
	PRESSURIZER	3/4 4-23
	OVERPRESSURE PROTECTION SYSTEMS	3/4 4-24
3/4.4.9	STRUCTURAL INTEGRITY	3/4 4-25
3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)		
3/4.5.1	SAFETY INJECTION TANKS	3/4 5-1
3/4.5.2	ECCS SUBSYSTEMS - HOT STANDBY, STARTUP AND POWER OPERATION.	3/4 5-3
3/4.5.3	ECCS SUBSYSTEMS - HOT SHUTDOWN	3/4 5-6
3/4.5.4	REFUELING WATER TANK	3/4 5-7

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
	C-E ATMOSPHERIC TYPE CONTAINMENT	
3/4.6 CONTAINMENT SYSTEMS		
3/4.6.1	PRIMARY CONTAINMENT	
	CONTAINMENT INTEGRITY	3/4 6-1
	CONTAINMENT LEAKAGE	3/4 6-1
	CONTAINMENT AIR LOCKS	3/4 6-1
	CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEM	3/4 6-1
	INTERNAL PRESSURE	3/4 6-1
	AIR TEMPERATURE	3/4 6-1
	CONTAINMENT STRUCTURAL INTEGRITY	3/4 6-1
	CONTAINMENT VENTILATION SYSTEM	3/4 6-1
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	
	CONTAINMENT SPRAY SYSTEM	3/4 6-2
	IODINE REMOVAL SYSTEM	3/4 6-3
	CONTAINMENT COOLING SYSTEM	3/4 6-5
3/4.6.3	CONTAINMENT ISOLATION VALVES	3/4 6-6
3/4.6.4	COMBUSTIBLE GAS CONTROL	
	HYDROGEN ANALYZERS	3/4 6-10
	ELECTRIC HYDROGEN RECOMBINERS	3/4 6-10
	HYDROGEN MIXING SYSTEMS	3/4 6-10
	PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM	3/4 6-10
	VACUUM RELIEF VALVES	3/4 6-10
3/4.6.5	IODINE CLEANUP SYSTEM	3/4 6-11

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION | PAGE

3/4.7 PLANT SYSTEMS

SECTION		PAGE
3/4.7.1	TURBINE CYCLE	
	SAFETY VALVES	3/4 7-1
	EMERGENCY FEEDWATER SYSTEM	3/4 7-4
	CONDENSATE STORAGE TANK	3/4 7-5
	ACTIVITY	3/4 7-6
	MAIN STEAM LINE ISOLATION VALVES	3/4 7-8
3/4.7.2	STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION	3/4 7-9
3/4.7.3	COMPONENT COOLING WATER SYSTEM	3/4 7-10
3/4.7.4	SERVICE WATER SYSTEM	3/4 7-10
3/4.7.5	EMERGENCY COOLING POND	3/4 7-10
3/4.7.6	FLOOD PROTECTION	3/4 7-10
3/4.7.7	CONTROL ROOM EMERGENCY CLEANUP SYSTEM	3/4 7-10
3/4.7.8	ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM	3/4 7-10
3/4.7.9	SNUBBERS	3/4 7-11
3/4.7.10	SEALED SOURCE CONTAMINATION	3/4 7-14
3/4.7.11	FIRE SUPPRESSION SYSTEMS	
	FIRE SUPPRESSION WATER SYSTEM	3/4 7-15
	SPRAY AND/OR SPRINKLER SYSTEM	3/4 7-15
	CO_2 SYSTEMS	3/4 7-15
	HALON SYSTEM	3/4 7-15
	FIRE HOSE STATIONS	3/4 7-15
	YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES	3/4 7-15
3/4.7.12	FIRE BARRIER PENETRATIONS	3/4 7-16
3/4.7.13	AREA TEMPERATURE MONITORING	3/4 7-17
3/4.7.14	SHUTDOWN COOLING SYSTEM	3/4 7-18

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.8 ELECTRICAL PLANT SYSTEMS		
3/4.8.1	A.C. SOURCES	
	OPERATING	3/4 8-1
	SHUTDOWN	3/4 8-1
3/4.8.2	D.C. SOURCES	
	OPERATING	3/4 8-2
	SHUTDOWN	3/4 8-2
3/4.8.3	ONSITE POWER DISTRIBUTION SYSTEMS	
	OPERATING	3/4 8-3
	SHUTDOWN	3/4 8-3
3/4.8.4	ELECTRICAL EQUIPMENT PROTECTIVE DEVICES	
	CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES	3/4 8-4
	MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES	3/4 8-4
3/4.9 REFUELING OPERATIONS		
3/4.9.1	BORON CONCENTRATION	3/4 9-1
3/4.9.2	INSTRUMENTATION	3/4 9-2
3/4.9.3	DECAY TIME	3/4 9-3
3/4.9.4	CONTAINMENT BUILDING PENETRATIONS	3/4 9-4
3/4.9.5	COMMUNICATIONS	3/4 9-4
3/4.9.6	REFUELING MACHINE	3/4 9-4
3/4.9.7	CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING	3/4 9-4
3/4.9.8	SHUTDOWN COOLING AND COOLANT CIRCULATION	
	HIGH WATER LEVEL	3/4 9-5
	LOW WATER LEVEL	3/4 9-6
3/4.9.9	CONTAINMENT PURGE VALVE ISOLATION SYSTEM	3/4 9-7
3/4.9.10	WATER LEVEL - REACTOR VESSEL	3/4 9-8
3/4.9.11	WATER LEVEL - STORAGE POOL	3/4 9-9
3/4.9.12	STORAGE POOL AIR CLEANUP	3/4 9-10

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN 3/4 10-1

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS 3/4 10-2

3/4.10.3 REACTOR COOLANT LOOPS 3/4 10-3

3/4.10.4 CEA POSITION AND REGULATING CEA INSERTION LIMITS 3/4 10-4

3/4.10.5 SAFETY INJECTION TANKS 3/4 10-6

3/4.10.6 SAFETY INJECTION TANK PRESSURE 3/4 10-7

BASES

SECTION PAGE

3/4.0 APPLICABILITY . . . . B 3/4 0-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL . . . . B 3/4 1-1
3/4.1.2 BORATION SYSTEMS . . . . B 3/4 1-2
3/4.1.3 MOVABLE CONTROL ASSEMBLIES . . . . B 3/4 1-3

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE . . . . B 3/4 2-1
3/4.2.2 RADIAL PEAKING FACTORS . . . . B 3/4 2-2
3/4.2.3 AZIMUTHAL POWER TILT . . . . B 3/4 2-2
3/4.2.4 DNBR MARGIN . . . . B 3/4 2-3
3/4.2.5 RCS FLOW RATE . . . . B 3/4 2-3
3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE . . . . B 3/4 2-4
3/4.2.7 AXIAL SHAPE INDEX . . . . B 3/4 2-4
3/4.2.8 PRESSURIZER PRESSURE . . . . B 3/4 2-4

3/4.3 INSTRUMENTATION

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION . . . . B 3/4 3-1
3/4.3.3 MONITORING INSTRUMENTATION . . . . B 3/4 3-1
3/4.3.4 TURBINE OVERSPEED PROTECTION . . . . B 3/4 3-2

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION . . . . B 3/4 4-1
3/4.4.2 SAFETY VALVES . . . . B 3/4 4-2
3/4.4.3 PRESSURIZER . . . . B 3/4 4-2
3/4.4.4 STEAM GENERATORS . . . . B 3/4 4-2

BASES

SECTION		PAGE

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE . . . . B 3/4 4-2
3/4.4.6 CHEMISTRY . . . . B 3/4 4-3
3/4.4.7 SPECIFIC ACTIVITY . . . . B 3/4 4-3
3/4.4.8 PRESSURE/TEMPERATURE LIMITS . . . . B 3/4 4-4
3/4.4.9 STRUCTURAL INTEGRITY . . . . B 3/4 4-6

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS . . . . B 3/4 5-1
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS . . . . B 3/4 5-2
3/4.5.4 REFUELING WATER TANK . . . . B 3/4 5-3

C-E - ATMOSPHERIC TYPE CONTAINMENT

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT . . . . B 3/4 6-1
3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS . . . . B 3/4 6-1
3/4.6.3 CONTAINMENT ISOLATION VALVES . . . . B 3/4 6-1
3/4.6.4 COMBUSTIBLE GAS CONTROL . . . . B 3/4 6-1
3/4.6.5 IODINE CLEANUP SYSTEMS . . . . B 3/4 6-1

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE . . . . B 3/4 7-1
3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION . . . . . . B 3/4 7-2
3/4.7.3 COMPONENT COOLING WATER SYSTEM . . . . B 3/4 7-2
3/4.7.4 SERVICE WATER SYSTEM . . . . B 3/4 7-2
3/4.7.5 EMERGENCY COOLING POND . . . . B 3/4 7-2
3/4.7.6 FLOOD PROTECTION . . . . B 3/4 7-2
3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM . . . . B 3/4 7-3
3/4.7.8 ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM . . . . B 3/4 7-3
3/4.7.9 SNUBBERS . . . . B 3/4 7-3

SECTION PAGE

3/4.7.10 SEALED SOURCE CONTAMINATION . . . . B 3/4 7-3
3/4.7.11 FIRE SUPPRESSION SYSTEMS . . . . B 3/4 7-3
3/4.7.12 FIRE RATED ASSEMBLIES . . . . B 3/4 7-3
3/4.7.13 AREA TEMPERATURE MONITORING . . . . B 3/4 7-3
3/4.7.14 SHUTDOWN COOLING . . . . B 3/4 7-3

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1, 3/4.8.2. and 3/4.8.3 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS . . . . B 3/4 8-1
3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES . . . . B 3/4 8-1

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION . . . . B 3/4 9-1
3/4.9.2 INSTRUMENTATION . . . . B 3/4 9-1
3/4.9.3 DECAY TIME . . . . B 3/4 9-1
3/4.9.4 CONTAINMENT PENETRATIONS . . . . B 3/4 9-1
3/4.9.5 COMMUNICATIONS . . . . B 3/4 9-1
3/4.9.6 REFUELING MACHINE . . . . B 3/4 9-1
3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING . . . . B 3/4 9-1
3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION . . . . B 3/4 9-1
3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM . . . . B 3/4 9-2
3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL . . . . B 3/4 9-2
3/4.9.12 STORAGE POOL AIR CLEANUP SYSTEM . . . . B 3/4 9-2

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN . . . . B 3/4 10-1
3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS . . . . B 3/4 10-1
3/4.10.3 REACTOR COOLANT LOOPS . . . . B 3/4 10-1

INDEX

BASES

SECTION | PAGE

3/4.10.4 CEA POSITION AND REGULATING CEA INSERTION LIMITS . . . . . B 3/4 10-1

3/4.10.5 SAFETY INJECTION TANKS . . . . B 3/4 10-1

3/4.10.6 SAFETY INJECTION TANK PRESSURE . . . . B 3/4 10-2

DESIGN FEATURES

SECTION		PAGE
5.1 SITE		5-1
5.2 CONTAINMENT		5-1
5.3 REACTOR CORE		
5.3.1	FUEL ASSEMBLIES	5-1
5.3.2	CONTROL ELEMENT ASSEMBLIES	5-1
5.4 REACTOR COOLANT SYSTEM		
5.4.1	DESIGN PRESSURE AND TEMPERATURE	5-1
5.4.2	VOLUME	5-1
5.5 METEOROLOGICAL TOWER LOCATION		5-1
5.6 FUEL STORAGE		
5.6.1	CRITICALITY	5-1
5.6.3	DRAINAGE	5-1
5.6.4	CAPACITY	5-1
5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS		5-2

LIST OF FIGURES

		PAGE
3.1-1	ALLOWABLE MTC MODES 1 AND 2	3/4 1-5
3.1-2	MINIMUM BORATED WATER VOLUMES	3/4 1-12
3.1-3	CEA INSERTION LIMITS VS. THERMAL POWER	3/4 1-31
3.2-1	DNBR MARGIN OPERATING LIMIT BASED ON COLSS	3/4 2-7
3.2-2	DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (COLSS OUT OF SERVICE)	3/4 2-8
3.2-3	REACTOR COOLANT COLD LEG TEMPERATURE VS. CORE POWER LEVEL	3/4 2-11
3.4-1	DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131	3/4 4-19
3.4-2	REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITATIONS	3/4 4-21

LIST OF TABLES

		PAGE
1.1	FREQUENCY NOTATION	1-5
1.2	OPERATIONAL MODES	1-6
2.2-1	REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS . .	2-3
2.2-2	CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS	2-7
3.1-1	MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $K_{eff} > 0.98$ FOR MODES 3, 4, AND 5	3/4 1-17
3.1-2	MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.98 \geq K_{eff} > 0.97$ FOR MODES 3, 4, AND 5	3/4 1-18
3.1-3	MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.97 \geq K_{eff} > 0.96$ FOR MODES 3, 4, AND 5	3/4 1-19
3.1-4	MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.96 \geq K_{eff} > 0.95$ FOR MODES 3, 4, AND 5	3/4 1-20
3.1-5	MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.95 \geq K_{eff}$ FOR MODES 3, 4, AND 5	3/4 1-21
3.3-1	REACTOR PROTECTIVE INSTRUMENTATION	3/4 3-3
3.3-2	REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES	3/4 3-9
4.3-1	REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-12
3.3-3	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	3/4 3-17
3.3-4	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES	3/4 3-24
3.3-5	ENGINEERED SAFETY FEATURES RESPONSE TIMES	3/4 3-27
4.3-2	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-30
3.3-6	REMOTE SHUTDOWN SYSTEM INSTRUMENTATION	3/4 3-41
4.3-3	REMOTE SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-42
3.3-7	POST-ACCIDENT MONITORING INSTRUMENTATION	3/4 3-44

LIST OF TABLES

		PAGE
4.3-4	POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-45
3.4-1	REACTOR COOLANT SYSTEM CHEMISTRY	3/4 4-14
4.4-1	REACTOR COOLANT SYSTEM CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS	3/4 4-15
4.4-2	PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM	3/4 4-18
4.4-3	REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE	3/4 4-22
3.6-1	CONTAINMENT ISOLATION VALVE ACTUATION TIMES	3/4 6-7
3.7-1	STEAM LINE SAFETY VALVES PER LOOP	3/4 7-2
3.7-2	MAXIMUM ALLOWABLE VARIABLE OVERPOWER HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING TWO LOOP OPERATION WITH FOUR PUMPS OPERATING	3/4 7-3
4.7-1	SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM	3/4 7-7
3.7-3a	SAFETY-RELATED HYDRAULIC SNUBBERS	3/4 7-12
3.7-3b	SAFETY-RELATED MECHANICAL SNUBBERS	3/4 7-13
5.7-1	COMPONENT CYCLE OR TRANSIENT LIMITS	5-3
5.7-2	COMPONENT CYCLE OR TRANSIENT LIMITS	5-5

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

AZIMUTHAL POWER TILT - T_q

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 See Applicant's SAR.

CONTROLLED LEAKAGE

1.8 Not Applicable.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems other than reactor coolant pump controlled bleedoff flow, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

OPERABLE - OPERABILITY

1.15 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.16 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.17 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - Fxy

1.18 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.19 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

RATED THERMAL POWER

1.20 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.21 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE OCCURRENCE

1.22 See Applicant's SAR.

SHIELD BUILDING INTEGRITY

1.23 See Applicant's SAR.

SHUTDOWN MARGIN

1.24 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part length control element assembly position, and

b. All full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SOFTWARE

1.25 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation and procedures.

STAGGERED TEST BASIS

1.26 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.27 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.28 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleedoff flow.

TABLE 1.1

FREQUENCY NOTATION

Notation	Frequency
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
S.A	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER	COLD LEG TEMPERATURE, T_{cold}
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 500°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 500°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{cold} > 210°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 210°F
6. REFUELING***	$\leq$ 0.95	0	$\leq$ 135°F

* Excluding decay heat

** See further temperature restrictions in LCO #3.1.1.4.

*** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TECHNICAL SPECIFICATIONS

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

2.1.1.1 DNBR

The calculated DNBR of the reactor core shall be maintained $\geq$ 1.22.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor core has decreased to less than 1.22, be in HOT STANDBY within 1 hour and comply with the requirements of Specification 6.7.1.

2.1.1.2 PEAK LINEAR HEAT RATE

The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained $\leq$ 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour and comply with the requirements of Specification 6.7.1.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia reduce the Reactor Coolant System pressure to within its limit within 5 minutes and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant less conservative than the value shown in the Allowable Value column of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	See Applicant's SAR	See Applicant's SAR
2. Pressurizer Pressure - Low (2)	See Applicant's SAR	See Applicant's SAR
3. Steam Generator Level - Low (4)	See Applicant's SAR	See Applicant's SAR
4. Steam Generator Level - High (11)	See Applicant's SAR	See Applicant's SAR
5. Steam Generator Pressure - Low (3)	See Applicant's SAR	See Applicant's SAR
6. Containment Pressure - High	See Applicant's SAR	See Applicant's SAR
7. Reactor Coolant Flow - Low (8)	See Applicant's SAR	See Applicant's SAR
a) Floor (7)	See Applicant's SAR	See Applicant's SAR
b) Rate (7)	See Applicant's SAR	See Applicant's SAR
c) Band (7)	See Applicant's SAR	See Applicant's SAR
8. Local Power Density - High (5)	See Applicant's SAR	See Applicant's SAR
9. DNBR - Low (5)	See Applicant's SAR	See Applicant's SAR
B. Excore Neutron Flux		
1. Variable Overpower Trip (9)	See Applicant's SAR	See Applicant's SAR
a) Ceiling (10)	See Applicant's SAR	See Applicant's SAR
b) Rate (10)	See Applicant's SAR	See Applicant's SAR
c) Band (10)	See Applicant's SAR	See Applicant's SAR

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)	See Applicant's SAR	See Applicant's SAR
a) Startup and Operating	See Applicant's SAR	See Applicant's SAR
b) Shutdown	See Applicant's SAR	See Applicant's SAR
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
1. Pressurizer Pressure - High	See Applicant's SAR	See Applicant's SAR
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER: bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-6, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(7) FLOOR is the minimum value of the trip setpoint.
RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
BAND is the amount by which the trip setpoint is below the input signal unless limited by the rate or the floor.

(8) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(9) Percent of RATED THERMAL POWER. With one or more inoperable main steam line safety valves, refer to Specification 3.7.1.1 for the maximum variable overpower trip setpoint.

(10) CEILING is the maximum value of the trip setpoint.
RATE is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
BAND is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(11) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

Point ID Number	Program Label	Description	Allowable Value
60	FC1	Core coolant mass flow rate calibration constant	See Applicant's SAR
61	FC2	Core coolant mass flow rate calibration constant	See Applicant's SAR
62	CEANOP	CEAC/RSPT inoperable flag	See Applicant's SAR
63	TR	Azimuthal tilt allowance	See Applicant's SAR
64	TPC	Thermal power calibration constant	See Applicant's SAR
65	KCAL	Neutron flux power calibration constant	See Applicant's SAR
66	DNBRPT	DNBR pretrip setpoint	See Applicant's SAR
67	LPDPT	Local power density pretrip setpoint	See Applicant's SAR

TECHNICAL SPECIFICATION

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in the succeeding pages summarizes the reasons for the specifications of Section 2.0, but, in accordance with 10 CFR 50.36, are not a part of the Technical Specifications.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21.0 kw/ft to prevent fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrence, is limited to 1.22 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accomodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kw/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this safety limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, piping, and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of the design pressure. The Reactor Coolant System valves and fittings, are designed to either Section III of the ASME Code or ANSI B 31.7, Class I, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. See Applicant's FSAR for specific Code, Standard Editions, and Addenda. The safety limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences, infrequent incidents, and to assist the Engineered Safety Features Actuation System in mitigating the consequences of limiting faults. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analysis.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.22 and 21.0 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in System 80 applicable system descriptions and safety analyses.

Manual Reactor Trip

The manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat removal by the secondary system. During normal operation this trip's setpoint may be manually decreased to a minimum value of 100 psia as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at ≤400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The operator may manually bypass this trip when pressurizer pressure is below 400 psia; this bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at ≤200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin at least 10 minutes before emergency feedwater is required.

Local Power Density-High

The Local Power Density-High trip is provided to prevent the linear heat rate (kw/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design basis anticipated operational occurrence. The local power density is calculated in the Reactor Protective System utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. ΔT power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of a design bases anticipated operational occurrence. The DNBR - Low trip incorporates a low pressurizer pressure floor of (*) psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (ΔT) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

---

* See Applicant's SAR.

The DNBR, the trip variable, calculated by the CPC, incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.22 such that the decrease in calaculated core DNBR after the trip will not result in a violation of the DNBR safety limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

Parameter	Limiting Value
a. RCS Cold Leg Temperature-Low	
b. RCS Cold Leg Temperature-High	
c. Axial Shape Index-Positive	
d. Axial Shape Index-Negative	
e. Pressurizer Pressure-Low	
f. Pressurizer Pressure-High	See Applicant's SAR
g. Integrated Radial Peaking Factor-Low	
h. Integrated Radial Peaking Factor-High	
i. Quality Margin-Low	

Steam Generator Level-High

The Steam Generator Level-High trip provides protection in the event of excess feedwater flow. The setpoint for this trip is identical to the main steam isolation setpoint.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides protection against a reactor coolant pump sheared shaft event and a two-pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the said pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of linear heat rate or DNBR safety limits under the stated conditions.

Pressurizer Pressure-High (SPS)

The Supplementary Protection System (SPS) augments reactor protection against overpressurization by utilizing a separate and diverse trip logic from the Reactor Protection System for initiation of reactor trip. The SPS will initiate a reactor trip when pressurizer pressure exceeds a predetermined value.

2.2-2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

TECHNICAL SPECIFICATIONS

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/or associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation cannot be satisfied, except as provided in the associated ACTION requirements, within one hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements. Exceptions to these requirements are stated in the individual specifications.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{cold} GREATER THAN 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 6.0% delta k/k.

APPLICABILITY: MODES 1,2*, 3 and 4.

ACTION:

(a) With $K_{eff} \geq 1$
(i.e., Modes 1 and 2 critical)

Comply with Specification 3.1.3.6

(b) With the shutdown margin less than 6% delta k/k and $K_{eff} < 1$
(i.e., Modes 2, 3 and 4 subcritical)

Immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required shutdown margin is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 6.0% delta k/k:

a. Within one hour after detection of in inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% delta k/k at least once per 31 Effective Full Power Days. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - T_{cold} LESS THAN OR EQUAL TO 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

4.1.1.2.2 K_{eff} shall be determined to be equal to or less than 0.98 at least once per 24 hours, when the RCS water level is drained below the pressurizer low level instrument tap, by performing a reactivity balance considering the factors listed in 4.1.1.2.1b.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 Effective Full Power Days burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to 2/3 of the expected end-of-cycle core average burnup.

* With K_{eff} greater than or equal to 1.0.

See Special Exception 3.10.2.

SEE APPLICANT'S SAR

FIGURE 3.1-1

ALLOWABLE MTC MODES 1 AND 2

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature T_{cold} shall be greater than or equal to 552°F.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a Reactor Coolant System operating loop temperature T_{cold} less than 552°F, restore T_{cold} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{cold}) shall be determined to be greater than or equal to 552°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{cold} is less than 557°F.

* See Special Test Exception 3.10.5

With K_{eff} greater than or equal to 1.0.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. If only the spent fuel pool in Specification (3.1.2.5.a) is OPERABLE, or a flow path from the spent fuel pool via a gravity feed connection and a charging pump to the Reactor Coolant System.

b. If only the refueling water tank in Specification (3.1.2.5.b) is OPERABLE, a flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump or a low pressure safety injection pump to the Reactor Coolant System.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

At least once per 31 days be verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the Refueling Water Tank or the Spent Fuel Pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the Refueling Water Tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the Refueling Water Tank or the Spent Fuel Pool through CH-161 (Boric Acid Filter Isolation Valve) or CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% $\Delta k/k$ at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days be verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers a total of at least 40 gpm to the Reactor Coolant System.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump* or one high pressure safety injection pump or one low pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump or low pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* Whenever the reactor coolant level is below the bottom of the pressurizer in Mode 5, one and only one charging pump shall be OPERABLE by verifying at least once per 7 days that power is removed from the remaining charging pumps.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and

2. A boron concentration of between 4000 and 4400 ppm boron, and

3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2,

2. A boron concentration of between 4000 and 4400 ppm boron, and

3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and

2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

SEE APPLICANT'S SAR

FIGURE 3.1-2

MINIMUM BORATED WATER VOLUMES

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 6% $\Delta k/k$ at 210°F; restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least one per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume in the water source.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

BORON DILUTION ALARMS

LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE and set to alarm when the ratio of the measured flux to the flux at the specified K_{eff} is less than or equal to 4.7.

APPLICABILITY: MODES 3**, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm not OPERABLE:

   1. Determine the RCS boron concentration when entering MODE 3, 4, 5 or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Table 3.1-1, 3.1-2, 3.1-3, 3.1-4, or 3.1-5; depending upon the degree of subcriticality available by either boronometer or RCS sampling*.

b. With both startup channel high neutron flux alarms not OPERABLE:

   1. Determine the RCS boron concentration by BOTH boronometer and RCS sampling* when entering MODE 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Table 3.1-1, 3.1-2, 3.1-3, 3.1-4, or 3.1-5; depending upon the degree of subcriticality available by both boronometer and RCS sampling*. If one of the methods of determining the RCS boron concentration is not available, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one additional method for detecting a boron dilution is restored to OPERABLE status.

   2. When in MODE 5 with the RCS drained down to the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

---

* With one or more reactor coolant pumps (RCP) operating, the sample should be obtained from the hot leg. With no RCP operating, the sample should be obtained from the discharge line of the low pressure safety injection (LPSI) pump operating in the shutdown cooling mode.

** This specification is not applicable during the first 30 minutes after entering MODE 3 from MODE 2.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least:

   1. Once per 12 hours.

   2. Within 15 minutes after setting the alarm setpoints.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

TABLE 3.1-1: MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $K_{eff} > 0.98$ FOR MODES 3, 4, AND 5

OPERATIONAL MODE	TIME PERIOD IN HOURS AT WHICH MONITORING IS REQUIRED WITH THE FOLLOWING NUMBER OF CHARGING PUMPS OPERATING			
	0	1	2	3
3 (Hot Standby)	12.0	1.0	OPERATION NOT ALLOWED	
4 (Hot Shutdown)	12.0	1.0	OPERATION NOT ALLOWED	
5 (Cold Shutdown)	8.0	1.0	OPERATION NOT ALLOWED	
5 (RCS PARTIALLY DRAINED FOR SYSTEM REPAIRS)		OPERATION NOT ALLOWED		
6* (Refueling)	24.0	8.0	4.0	2.0

* A conservative value of an initial boron concentration is assumed which is bounded by the Technical Specification 3.9.1. Furthermore, during refueling the LPSI pumps should be used for any makeup operation. If it is necessary to use the charging pumps, the appropriate monitoring frequency above should be used.

TABLE 3.1-2: MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.98 \geq K_{eff} > 0.97$ FOR MODES 3, 4, AND 5

OPERATIONAL MODE	TIME PERIOD IN HOURS AT WHICH MONITORING IS REQUIRED WITH THE FOLLOWING NUMBER OF CHARGING PUMPS OPERATING			
	0	1	2	3
3 (Hot Standby)	12.0	2.5	1.0	0.5
4 (Hot Shutdown)	12.0	2.5	1.0	0.5
5 (Cold Shutdown)	8.0	2.5	1.0	0.5
5 (RCS PARTIALLY DRAINED FOR SYSTEM REPAIRS)†	8.0	0.5	OPERATION NOT ALLOWED	
6* (Refueling)	24.0	8.0	4.0	2.0

† The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

* A conservative value of an initial boron concentration is assumed which is bounded by the Technical Specification 3.9.1. Furthermore, during refueling the LPSI pumps should be used for any makeup operation. If it is necessary to use the charging pumps, the appropriate monitoring frequency above should be used.

TABLE 3.1-3: MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.97 \geq K_{eff} > 0.96$ FOR MODES 3, 4, AND 5

OPERATIONAL MODE	TIME PERIOD IN HOURS AT WHICH MONITORING IS REQUIRED WITH THE FOLLOWING NUMBER OF CHARGING PUMPS OPERATING			
	0	1	2	3
3 (Hot Standby)	12.0	3.5	1.5	1.0
4 (Hot Shutdown)	12.0	3.5	1.5	1.0
5 (Cold Shutdown)	8.0	3.5	1.5	1.0
5 (RCS PARTIALLY DRAINED FOR SYSTEM REPAIRS)†	8.0	1.0	OPERATION NOT ALLOWED	
6* (Refueling)	24.0	8.0	4.0	2.0

† The Technical Specifica.ion 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

* A conservative value of an initial boron concentration is assumed which is bounded by the Technical Specification 3.9.1. Furthermore, during refueling the LPSI pumps should be used for any makeup operation. If it is necessary to use the charging pumps, the appropriate monitoring frequency above should be used.

TABLE 3.1-4: MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.96 \geq K_{eff} > 0.95$ FOR MODES 3, 4, AND 5

OPERATIONAL MODE	TIME PERIOD IN HOURS AT WHICH MONITORING IS REQUIRED WITH THE FOLLOWING NUMBER OF CHARGING PUMPS OPERATING			
	0	1	2	3
3 (Hot Standby)	12.0	5.0	2.0	1.0
4 (Hot Shutdown)	12.0	5.0	2.0	1.0
5 (Cold Shutdown)	8.0	5.0	2.0	1.0
5 (RCS PARTIALLY DRAINED FOR SYSTEM REPAIRS)†	8.0	1.5	OPERATION NOT ALLOWED	
6* (Refueling)	24.0	8.0	4.0	2.0

† The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

* A conservative value of an initial boron concentration is assumed which is bounded by the Technical Specification 3.9.1. Furthermore, during refueling the LPSI pumps should be used for any makeup operation. If it is necessary to use the charging pumps, the appropriate monitoring frequency above should be used.

TABLE 3.1-5: MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $K_{eff} \leq 0.95$ FOR MODES 3, 4, AND 5

OPERATIONAL MODE	TIME PERIOD IN HOURS AT WHICH MONITORING IS REQUIRED WITH THE FOLLOWING NUMBER OF CHARGING PUMPS OPERATING			
	0	1	2	3
3 (Hot Standby)	12.0	6.0	3.0	1.5
4 (Hot Shutdown)	12.0	6.0	3.0	1.5
5 (Cold Shutdown)	8.0	6.0	3.0	1.5
5 (RCS PARTIALLY DRAINED FOR SYSTEM REPAIRS)†	8.0	2.0	OPERATION NOT ALLOWED	
6* (Refueling)	24.0	8.0	4.0	2.0

† The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

* A conservative value of an initial boron concentration is assumed which is bounded by the Technical Specification 3.9.1. Furthermore, during refueling the LPSI pumps should be used for any makeup operation. If it is necessary to use the charging pumps, the appropriate monitoring frequency above should be used.

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full length or part length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA is either:

    1. Restored to OPERABLE status within its above specified alignment requirements, or

    2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

        a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 6.6 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3, the THERMAL POWER Level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

        b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

    Otherwise, be in at least HOT STANDBY within 6 hours.

* See Special Test Exceptions 3.10.2 and 3.10.4.

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 6.6 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

    1. Restored to OPERABLE status within its above specified alignment requirements, or

    2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

        a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 6.6 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

        b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

    Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full length CEA inoperable due to causes other than addressed by ACTION a. above, and inserted beyond the Long Term Steady State Insertion Limits (Figure 3.1-3) but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

f. With one full length CEA inoperable due to causes other than addressed by ACTION a, above, ensure that the CEA is (1) within its above specified alignment requirements and (2) either fully withdrawn or, if in full length CEA group 5, within the Long Term Steady State Insertion Limits (Figure 3.1-3). Then operation in MODES 1 and 2 may continue.

g. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable PLCEA is maintained within 6.6 inches (indicated position) of all other PLCEAs in its group.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length and part length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours, except during time intervals when one CEAC is inoperable or when both CEACs are inoperable; then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted and each part length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

- a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,
- b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and
- c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

- a. Restore the inoperable position indicator channel to OPERABLE status, or
- b. Be in at least HOT STANDBY, or
- c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specification 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out"* limit.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

---

* CEA's are fully withdrawn ("Full Out") when withdrawn to at least 144.75 inches (193 steps).

POSITION INDICATOR CHANNEL - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4* and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

* With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 4.0 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. T_{cold} greater than or equal to 550°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 and 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 144.75 inches (193 steps).

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 144.75 inches (193 steps), except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to at least 144.75 inches (193 steps), or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 144.75 inches (193 steps).

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

* See Special Test Exception 3.10.2

With K_{eff} greater than or equal to 1.0.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence, specified overlap, and to the insertion limits## shown on Figure 3.1-3**, with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per 18 Effective Full Power Month.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-3 are not exceeded, or

---

* See Special Test Exceptions 3.10.2 and 3.10.4.

\# With K_{eff} greater than or equal to 1.0.

** CEA's are considered fully withdrawn in accordance with Figure 3.1-3 when withdrawn to at least 144.75 inches (193 steps).

\#\# Following a reactor power cutback in which (1) Regulating Group 5 is dropped or (2) Regulating Groups 4 and 5 are dropped and for cases (1) and (2) should the remaining Regulating Groups (Group 1, 2, 3, and 4) be sequentially inserted, the Transient Insertion Limit of Figure 3.1-3 can be exceeded for up to 2 hours. Also for cases (1) and (2), the specified overlap between Regulating Groups 3, 4 and 5 can be exceeded for up to 2 hours.

2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

c. With the regulating CEA group inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 18 Effective Full Power Months, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

SEE APPLICANT'S SAR

FIGURE 3.1-3

CEA INSERTION LIMITS VS. THERMAL POWER

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed 14.0 kw/ft.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kw/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is less than or equal to 14.0 kw/ft.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on 14.0 kw/ft.

3/4.2.2 RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With a $F^m xy$ exceeding a corresponding $F^c xy$, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

c. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 Effective Full Power Days.

* See Special Test Exception 3.10.2.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall not exceed the AZIMUTHAL TILT ALLOWANCE used in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the value used in the CPCs, but ≤ 0.10, either reduce the power tilt to within the value used in the CPC or adjust the AZIMUTHAL TILT ALLOWANCE used in the CPC to a value greater than or equal to the measured tilt.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10 then:

1. Due to misalignment of either a part length or full length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and verify that the variable overpower trip setpoint has been reduced as appropriate within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER. Subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within its limit above 20% of RATED THERMAL POWER as follows:

* See Special Test Exception 3.10.2.

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Aximuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 days Effective Full Power Days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR to within its limits within one hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) per Figure 3.2-1 or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated n all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calcualtions at least once per 31 days:

Burnup $(\frac{GWD}{MTU})$	DNBR Penalty (%)*
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

* The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

SEE APPLICANT'S SAR

FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS

SEE APPLICANT'S SAR

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE)

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 164.0×10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than its limit at least once per 12 hours.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg temperature shall be within the area of Acceptable Operation on Figure 3.2-3.

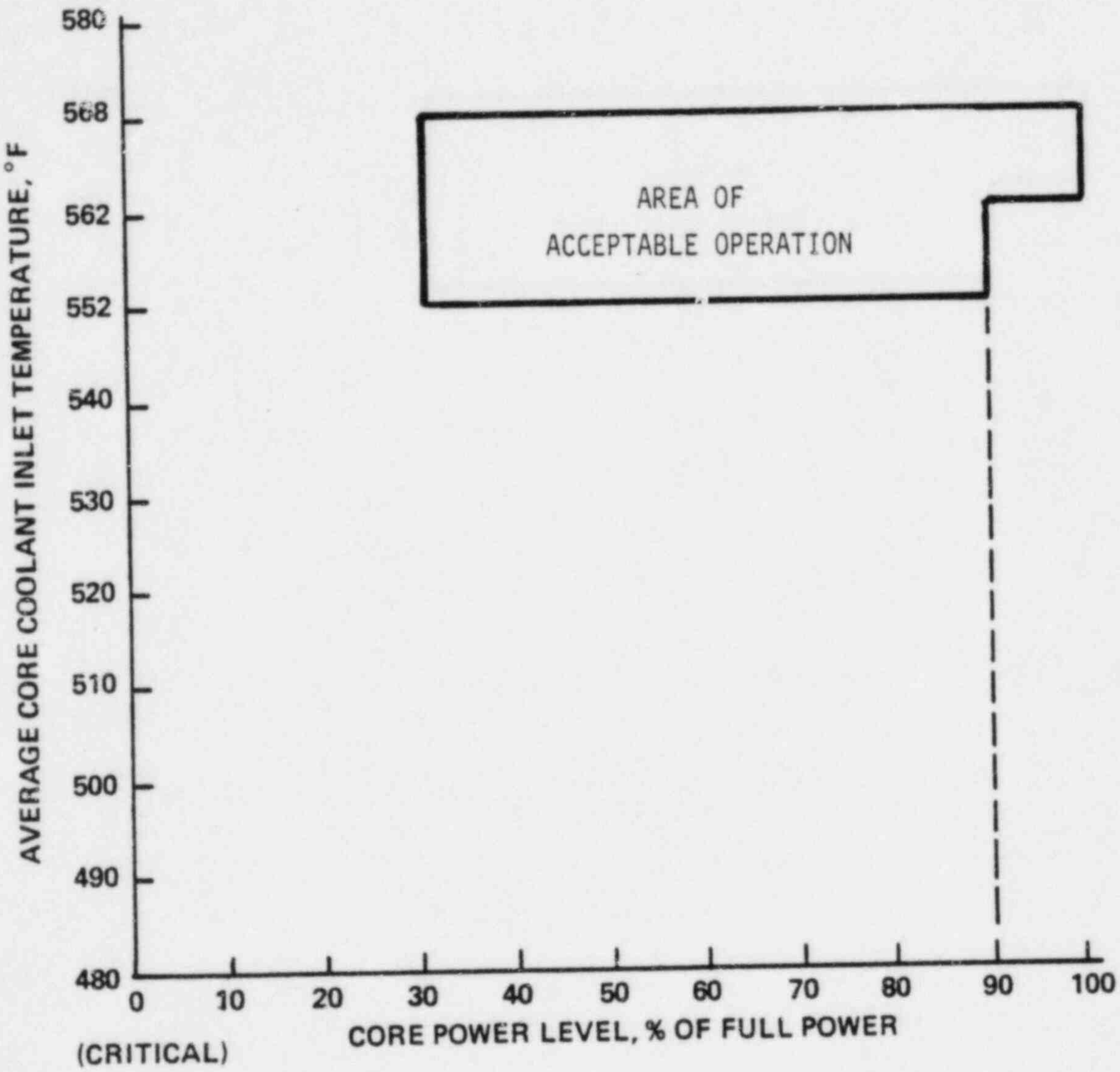
APPLICABILITY: MODE 1 above 30% of Rated Thermal Power.

ACTION:

With the Reactor Coolant Cold Leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg temperature shall be determined to be within its limit at least once per 12 hours.



C - E SYSTEM 80	REACTOR COOLANT COLD LEG TEMPERATURE vs CORE POWER LEVEL	Figure 3.2-3

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) outside its above limits, restore the core average ASI to within its limits within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

* See Special Test Exception 3.10.2

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 1815 psia and 2370 psia.

APPLICABILITY: MODE 1

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific reactor trip function as shown in the Total No. of Channels column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts AC (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volts DC with an applied input voltage of 5-10 volts DC.

b. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts DC.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restarts, Periodic Test Restart (Code 30) and Normal System Load (Code 33), shall not be included in this total.

4.3.1.6 The Core Protection Calculator System shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature Alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

RPS Functional Unit	Total No. Of Channels	Channels To Trip	Minimum Channels Operable	Applicable Modes	Action
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	2#, 3#
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	2#, 3#
9. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
B. Excore Neutron Flux					
1. Variable Overpower	4	2	3	1, 2	2#, 3#
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
b. Shutdown	4	2	3	3*, 4*, 5*	2#, 3#
	4	0	2	3, 4, 5	4

TABLE 3.3-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION

RPS Functional Unit	Total No. Of Channels	Channels To Trip	Minimum Channels Operable	Applicable Modes	Action
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calcualtors	4	2 (c)(d)	3	1, 2	2#, 3#, 7
D. Supplementary Protection System					
1. Pressurizer Pressure - High	4 (f)	2	4	1, 2	8
II. RPS LOGIC					
A. Matrix Logic	6	1	3	1, 2	1
	6	1	3	3*, 4*, 5*	8
B. Initiation Logic	4	2	4	1, 2,	5
	4	2	4	3*, 4*, 5*	8
III RPS ACTUATION DEVICES					
A. Reactor Trip Breaker	4 (f)	2	4	1, 2,	5
	4 (f)	2	4	3*, 4*, 5*	8
B. Manual Trip	4 (f)	2	4	1, 2	5
	4 (f)	2	4	3*, 4*, 5*	8

TABLE 3.3-1 (Cont'd)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\leq 10^{-4}$% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is $\geq$ 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER IS $\geq$ 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of, or associated with, one of the four Reactor Trip breakers, arranged in a selective 2 out of 4 configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.61. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Excore Nuclear Instrument - Linear Power (Subchannel or Linear)	Variable Overpower - Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator #1 Level - Low (ESF) Steam Generator #2 Level - Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator #1 Level - Low (ESF) Steam Generator #2 Level - Low (ESF)
5. Core Protection Calculator	Local Power Density - High DNBR - Low

ACTION 3 - With the number of channels OPERABLE One less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Excore Nuclear Instrument Linear Power (Subchannel or Linear)	Variable Overpower Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High (Narrow Range)	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator #1 Level - Low (ESF) Steam Generator #2 Level - Low (ESF)
4. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low Steam Generator #1 Level - Low (ESF) Steam Generator #2 Level - Low (ESF)
5. Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the reactor trip breaker of the inoperable channel may be closed for up to one hour in order to perform surveillance testing per Specification 4.3.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

   1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER.

   2. Within 4 hours:

      a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

      c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts (excluding auto restart codes 30 and 33) of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements restore the inoperable channel to OPERABLE status within 48 hours or open the affected reactor trip breaker(s) within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	$\leq$ ____ seconds
2. Pressurizer Pressure - Low	$\leq$ ____ seconds
3. Steam Generator Level - Low	$\leq$ ____ seconds
4. Steam Generator Level - High	$\leq$ ____ seconds
5. Steam Generator Pressure - Low	$\leq$ ____ seconds
6. Containment Pressure - High	$\leq$ ____ seconds
7. Reactor Coolant Flow - Low	$\leq$ ____ seconds
8. Local Power Density - High***	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ ____ seconds*
b. CEA Positions	$\leq$ ____ seconds**
c. CEA Positions: CEAC Penalty Factor	$\leq$ ____ seconds**
9. DNBR - Low***	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ ____ seconds*
b. CEA Positions	$\leq$ ____ seconds**
c. Cold Leg Temperature	$\leq$ ____ seconds##
d. Hot Leg Temperature	$\leq$ ____ seconds##
e. Primary Coolant Pump Shaft Speed	$\leq$ ____ seconds#
f. Reactor Coolant Pressure from Pressurizer	$\leq$ ____ seconds###
g. CEA Positions: CEAC Penalty Factor	$\leq$ ____ seconds**

TABLE 3.3-2 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

B. Excore Neutron Flux	
1. Variable Overpower	< ____ seconds*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ ____ seconds*
b. Shutdown	≤ ____ seconds*
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
1. Pressurizer Pressure - High	≤ ____ seconds
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

TABLE 3.3-2 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

* Neutron detectors are exempt from repsonse time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of the first electronic component in the channel.

** The CEA position transmitters are exempt from response time testing. The response time shall be measured from the input to the CPC, CEAC or signal isolator.

*** Response times are for hardware delays only.

The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input

Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time (P_T) of the slowest RTD shall be less than or equal to 6.0 seconds.

Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to (0.7) seconds.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
I. TRIP GENERATION				
A. Process				
1. Pressurizer Pressure - High	S	R	M	1, 2
2. Pressurizer Pressure - Low	S	R	M	1, 2
3. Steam Generator Level - Low	S	R	M	1, 2
4. Steam Generator Level - High	S	R	M	1, 2
5. Steam Generator Pressure - Low	S	R	M	1, 2, 3*, 4*
6. Containment Pressure - High	S	R	M	1, 2
7. Reactor Coolant Flow - Low	S	R	M	1, 2
8. Local Power Density - High	S	See Core Protection Calculation System		1, 2
9. DNBR - Low	S			1, 2
B. Excore Neutron Flux				
1. Variable Overpower	S	D (2, 4), M (3, 4) Q (4)	M	1, 2
2. Logarithmic Power Level - High	S	R (4)	M and S/U (1)	1, 2, 3, 4, 5, and *

TABLE 4.3-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
C. Core Protection Calculator System				
1. CEA Calculators	S	R	M, R (6)	1, 2
2. Core Protection Calculators	S	D (2, 4), R (4, 5) M (8), S (7)	M (9), R (6)	1, 2
D. Supplementary Protection System				
1. Pressurizer Pressure - High	S	R	M	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	M, S/U (1)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Cont'd)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) Each startup or when required with the reactor trip breakers closed and the CEA drive system capable or rod withdrawal, if not performed in the previous 7 days.

(2) Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level, CPC ΔT Power and CPC nuclear power signals to agree with calorimetric calculation if absolute difference is > 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practible to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total steady state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total steady state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

LIMITING CONDITION FOR OPERABILITY

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With the ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the operating and trip channel bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total operating bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. SAFETY INJECTION (SIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3, 4	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a), 4	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual SIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
II. CONTAINMENT ISOLATION (CIAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
2. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual CIAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
III. CONTAINMENT SPRAY (CSAS)					
A. Sensor/Trip Units					
1. Containment Pressure - High-High	4	2	3	1, 2, 3	13
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	18
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual CSAS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	17
IV. MAIN STEAM LINE ISOLATION (MSIS)					
A. Sensor/Trip Units					
1. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(b), 4	14*, 15*
2. Steam Generator Level - High	4/steam generator	2/steam generator	3/steam generator	1, 2, 3, 4	14*, 15*
3. Containment Pressure - High	4	2	3	1, 2, 3, 4	14*, 15*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	18
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual MSIS (Trip Buttons)	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	17

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
V. RECIRCULATION (RAS)					
A. Sensor/Trip Units					
1. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	14*, 15*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3, 4	18
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual RAS	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	17
VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1)					
A. Sensor/Trip Units					
1. Steam Generator #1 Level - Low	4	2	3	1, 2, 3	14*, 15*
2. Steam Generator #1 Pressure - Low	4	2	3	1, 2, 3(b)	14*, 15*
3. Steam Generator #2 Pressure - Low	4	2	3	1, 2, 3(b)	14*, 15*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	18
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual EFAS	4(c)	2(d)	4	1, 2, 3, 4	16
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	17

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
VII. EMERGENCY FEEDWATER (SG-2)(EFAS-2)					
A. Sensor/Trip Units					
1. Steam Generator #2 Level - Low	4	2	3	1, 2, 3	14*, 15*
2. Steam Generator #2 Pressure - Low	4	2	3	1, 2, 3(b)	14*, 15*
3. Steam Generator #1 Pressure - Low	4	2	3	1, 2, 3(b)	14*, 15*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	18
2. Initiation Logic	4[(c)]	2[(d)]	4	1, 2, 3, 4	12
3. Manual EFAS	4[(c)]	2[(d)]	4	1, 2, 3, 4	16
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	17

TABLE 3.3-3 (Cont'd)

TABLE NOTATION

(a) In MODES 3-6, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-6, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper 2 of 4 combination.

The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.61. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

TABLE 3.3-3 (Cont'd)

TABLE NOTATION

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level #1 - Low (ESF) Steam Generator Level #2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level #1 - Low (ESF) Steam Generator Level #2 - Low (ESF)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator Level #1 - Low (ESF) Steam Generator Level #2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level #1 - Low (ESF) Steam Generator Level #2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 14 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the minimum number of channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. SAFETY INJECTION (SIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	*	*
2. Pressurizer Pressure - Low	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
II. CONTAINMENT ISOLATION (CIAS)		
A. Sensor/Trip Units		
1. Containment Pressure - High	*	*
2. Pressurizer Pressure - Low	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
III. CONTAINMENT SPRAY (CSAS)		
A. Sensor/Trip Units		
1. Containment Pressure High - High	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

* See Applicant's SAR.

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
IV. MAIN STEAM LINE ISOLATION (MSIS)		
A. Sensor/Trip Units		
1. Steam Generator Pressure - Low	*	*
2. Steam Generator Level - High	*	*
3. Containment Pressure - High	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
1. Refueling Water Storage Tank - Low	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	*	*
2. Steam Generator ΔPressure (SG #2 > SG #1) - High	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
VII. EMERGENCY FEEDWATER (SG-2)(EFAS-2)		
A. Sensor/Trip Units		
1. Steam Generaotr #2 Level - Low	*	*
2. Steam Generator ΔPressure (SG #1 > SG #2) - High	*	*
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable
Containment Purge Valve Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable
f. CCAS	
Containment Cooling	Not Applicable
g. EFAS	
Emergency Feedwater Pumps	Not Applicable

TABLE 3.3-5 (Cont'd)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ ___*/___**
b. Containment Isolation	≤ ___*/___**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ ___*/___**
b. Containment Isolation	≤ ___*/___**
c. Main Steam Isolation	≤ ___*/___**
4. Containment Pressure--High-High	
a. Containment Spray	≤ ___*/___**
5. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ ___*/___**
6. Refueling Water Tank-Low	
a. Containment Sump Recirculation	≤ ___*/___**
7. Steam Generator Level-Low	
a. Emergency Feedwater	≤ ___*/___**
8. Steam Generator Water Level-High	
a. Main Steam Isolation	≤ ___*/___**
9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Emergency Feedwater Isolation from Ruptured Steam Generator	≤ ___*/___**
NOTE: Response time for Motor-Driven Auxiliary Feedwater Pumps on all S.I. signal starts	≤ (60.0)

TABLE 3.3-5 (Cont'd)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

** Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

*** See Applicant's SAR for Response Times.

TABLE 4.3-2

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
I. SAFETY INJECTION (SIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3, 4
2. Pressurizer Pressrue - Low	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual SIAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4
II. CONTAINMENT ISOLATION (CIAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High	S	R	M	1, 2, 3
2. Pressurizer Pressure - Low	S	R	M	1, 2, 3

TABLE 4.3-2 (Cont'd)

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
II. CONTAINMENT ISOLATION (CIAS) (Cont'd)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CIAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4
III. CONTAINMENT SPRAY (CSAS)				
A. Sensor/Trip Units				
1. Containment Pressure - High-High	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual CSAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4

TABLE 4.3-2 (Cont'd)

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
IV. MAIN STEAMLINE ISOLATION (MSIS)				
A. Sensor/Trip Units				
1. Steam Generator Pressure - Low	S	R	M	1, 2, 3, 4
2. Steam Generator Level - High	S	R	M	1, 2, 3, 4
3. Containment Pressure - High	S	R	M	1, 2, 3, 4
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual MSIS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4
V. RECIRCULATION (RAS)				
A. Sensor/Trip Units				
1. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3

TABLE 4.3-2 (Cont'd)

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
V. RECIRCULATION (RAS) (Cont'd)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual RAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4
VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1)				
A. Sensor/Trip Units				
1. Steam Generator #1 Level - Low	S	R	M	1, 2, 3
2. Steam Generator #1 Pressure - Low	S	R	M	1, 2, 3
3. Steam Generator #2 Pressure - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4

TABLE 4.3-2 (Cont'd)

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
VII. EMERGENCY FEEDWATER (SG-2)(EFAS-2)				
A. Sensor/Trip Units				
1. Steam Generator #2 Level - Low	S	R	M	1, 2, 3
2. Steam Generator #2 Pressure - Low	S	R	M	1, 2, 3
3. Steam Generator #1 Pressure - Low	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual EFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1), R(2)	1, 2, 3, 4

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(1) Testing of automatic actuation logic shall include energization/de-energization of each initiation relay and verification of proper operation of each initiation relay.

(2) Testing of the actuation relays shall include the energization/de-energization of each actuation relay and verification of proper operation of each actuation relay.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.1 Radiation Monitoring Instrumentation

See Applicant's SAR.

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.3 Seismic Instrumentation

See Applicant's SAR.

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.4 Meteorological Instrumentation

See Applicant's SAR.

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown controls and monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-6, either restore the inoperable channel to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

b. With one or more remote shutdown system instrumentation control circuits required by Table 3.3-6 inoperable, restore the inoperable circuit to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-3.

4.3.3.5.2 Each remote shutdown system instrumentation control circuit shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

TABLE 3.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

Instrument	Readout Location	Measurement Range	Minimum Channels OPERABLE
1. Logarithmic Neutron Channel	RSP*	2×10^{-8} - 200%	1
2. Reactor Coolant Hot Leg Temperature	RSP	50 - 750°F	1/loop
3. Pressurizer Pressure	RSP	15 - 3000 psia	1
4. Pressurizer Level	RSP	0 - 100%	1
5. Steam Generator Pressure	RSP	0 - 1385 psig	1/steam generator
6. Steam Generator Water Level	RSP	0 - 100%	1/steam generator
7. Refueling Water Tank Level	RSP	0 - 100%	1
8. Charging Flow	RSP	0 - 150 gpm	1
9. Charging Pressure	RSP	0 - 3000 psig	1

Control Circuits
Pressurizer Heater

* RSP - Remote Shutdown Panel

TABLE 4.3-3

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Logarithmic Neutron Channel	M	Q
2. Reactor Coolant Hot Leg Temperature	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Pressure	M	R
6. Steam Generator Water Level	M	R
7. Refueling Water Tank Level	M	R
8. Charging Flow	M	R
9. Charging Pressure	M	R

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-7, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-7; either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-4.

TABLE 3.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION

Instrument (Illustrational only)	Required Number of Channels	Minimum Channels OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1/loop
3. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1/loop
4. Pressurizer Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Generator Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator
8. Refueling Water Storage Tank Water Level	2	1
9. Auxiliary Feedwater Flow Rate	2	1
10. Reactor Cooling System Subcooling Margin Monitor	2	1
11. Safety Valve Position Indicator	2/valve	1/valve
12. Containment Water Level (Narrow Range)	2	1
13. Containment Water Level (Wide Range)	2	1

TABLE 4.3-4

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument (Illustrational Only)	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	M	R
4. Pressurizer Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Wide Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. Safety Valve Position Indicator	M	R
12. Containment Water Level (Narrow Range)	M	R
13. Containment Water Level (Wide Range)	M	R

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.7 Chlorine Detection Systems

See Applicant's SAR.

3/4.3.3.8 Fire Detection Instrumentation

See Applicant's SAR.

LOOSE - PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.8 Loose Part Detection Instrumentation

See Applicant's SAR.

TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.4 Turbine Overspeed Protection

See Applicant's SAR.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both Reactor Coolant loops and both Reactor Coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With less than the above required Reactor Coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required Reactor Coolant loops shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

* See Special Test Exception 3.10.3.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant Loops shall be in operation.

a. Reactor Coolant Loop (A) and its associated steam generator and at least one associated Reactor Coolant pump.

b. Reactor Coolant Loop (B) and its associated steam generator and at least one associated Reactor Coolant pump.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required Reactor Coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no Reactor Coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required Reactor Coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required Reactor Coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one Reactor Coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be $\geq$ 25% on the wide range level indicator at least once per 12 hours.

* All Reactor Coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loop shall be in operation.*

a. Reactor Coolant Loop #A and its associated steam generator and at least one associated Reactor Coolant pump,**

b. Reactor Coolant Loop #B and its associated steam generator and at least one associated Reactor Coolant pump,**

c. Shutdown Cooling Train #1,

d. Shutdown Cooling Train #2.

APPLICABILITY: MODE 4 ($T_{Cold} \leq 350°F$)

ACTION:

a. With less than the above required Reactor Coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no Reactor Coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

* All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to (a) (***)°F during cooldown or (b) (***)°F during heatup unless the secondary water temperature of each steam generator is less than 150°F above each of the Reactor Coolant System cold leg temperatures.

*** See Applicant's SAR.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 25% on the wide range level indicator at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant or shutdown cooling loop shall be verified to be in operation and circulating Reactor Coolant at a flowrate greater than or equal to 4000 gpm at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% on the wide range level indicator.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flowrate greater than or equal to 4000 gpm at least once per 12 hours.

One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

A reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to (**)°F during cooldown or (**)°F during heatup unless the secondary water temperature (saturation temperature corresponding to the steam generator pressure) of each steam generator is less than 150°F above each of the Reactor Coolant System cold leg temperatures.

* The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

** See Applicant's SAR.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flowrate greater than or equal to 4000 gpm at least once per 12 hours.

One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

* The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODE 4 (when the temperature of all of the RCS cold legs is greater than __**__ °F during cooldown or __**__ °F during heatup.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entry into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** See Applicant's SAR.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia $\pm$ 1%.*

APPLICABILITY: MODES 1, 2, 3

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with:

a. A steady state water volume less than or equal to 58% indicated level (1010 cu. ft) but greater than 26% indicated level (425 cu. ft.), and

b. At least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of the above required pressurizer heaters operable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kw at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:

a. the pressurizer heaters are automatically shed from the emergency power sources, and

b. the pressurizer heaters can be reconnected to their respective buses manually from the control room.

3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.4.4 Steam Generators

See Applicant's SAR.

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.4.5.1 Leakage Detection Systems

See Applicant's SAR

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through steam generators,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within its limit within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

3/4.4.6 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pres- sure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.6 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-1.

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY

Parameter	Steady State Limit	Transient Limit
Dissolved Oxygen*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
Fluoride	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm

* Limit not applicable with T_{cold} less than or equal to 250°F.

TABLE 4.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

Parameter	Sample and Analysis Frequency
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

* Not required with T_{cold} less than or equal to 250°F.

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provision of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT SHUTDOWN with T_{cold} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT SHUTDOWN with T_{cold} less than 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-2 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

   1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

---

* With T_{cold} greater than or equal to 500°F.

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-2.

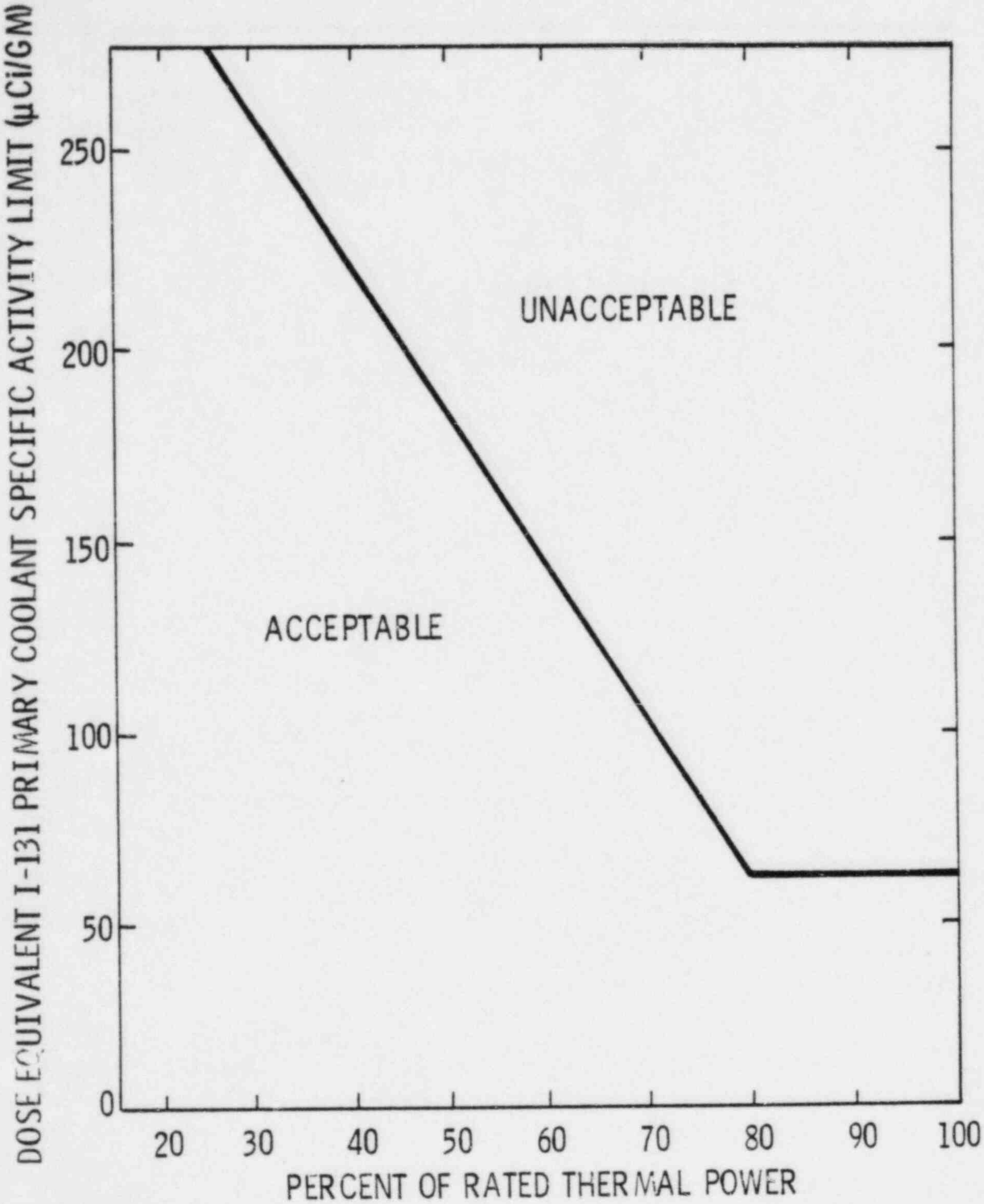
TABLE 4.4-2

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Type of Measurement And Analysis	Sample and Analysis Frequency	Modes in Which Sample And Analysis Required
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

\# Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.



C - E SYSTEM 80	DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY >1.0μCi/GRAM DOSE EQUIVALENT I-131	Figure 3.4-1

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 100°F/hr.

b. A maximum cooldown rate of 100°F/hr.

c. A maximum temperature change of 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-3. The results of these examinations shall be used to update Figure 3.4-2.

See Applicant's SAR

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITATIONS

TABLE 4.4-3

CAPSULE ASSEMBLY REMOVAL SCHEDULE
(Typical)

Capsule No.	Azimuthal Location (degree)	Lead Factor	Removal Time (EFPY)
1	38	1.5	Standby
2	43	1.5	Standby
3	137	1.5	4-5
4	142	1.5	Standby
5	230	1.5	12-15
6	310	1.5	18-24

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F/hr.,

b. A maximum cooldown rate of 200°F/hr.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than 4 reactor coolant pumps operating and for each cycle of auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both Shutdown Cooling System suction line relief valves with lift settings of less than or equal to ** psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: MODES 4, 5, and 6*. When the temperature of one or more of the RCS cold legs is less than or equal to:

a) (**) °F during cooldown
b) (**) °F during heatup

ACTION:

a) With one or more Shutdown Cooling System suction line relief valves inoperable restore the valves to OPERABLE status within 1 hour. Otherwise cooldown and depressurize. Do not start a Reactor Coolant Pump if the steam generator secondary water temperature (saturation temperature corresponding to the S/G pressure) is greater than the RCS cold leg temperature. Do not start any other pump which could overpressurize the RCS. A bubble shall be maintained in the pressurizer.

b) In the event the SCS suction line relief valves actuate to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves on the transient and any corrective action necessary to prevent recurrence.

c) The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during:

a) heatup with the RCS temperature less than or equal to (**) °F.

b) cooldown with the RCS temperature less than or equal to (**) °F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

* This Specification is not applicable with the Reactor Vessel head removed.

** See Applicant's SAR.

3/4.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a) With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b) With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210°F.

c) With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d) The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14 of October 27, 1971 (originally issued as Safety Guide 14 on October 27, 1971).

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Safety Injection System safety injection tank shall be OPERABLE with:

a. The isolation valve keylocked open and power removed.***

b. A contained borated water level of between 28% (1802 ft^3) and 72% 1914 ft^3) on narrow range indication.

c. A boron concentration of between 4000 and 4400 ppm of boron.

d. A nitrogen cover pressure of between 600 and 625 psig.

e. The nitrogen vent valves closed and power removed.**

f. The nitrogen vent valves capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3*+, and 4*+.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

\+ With pressurizer pressure $\geq$ 1750 psia. When pressurizer pressure is below 1750 psia, at least three safety injection tanks must be operable with a minimum pressure of 254 psig and a contained borated water volume of between 60% (1415 ft^3) level on wide range indication and 72% (1914 ft^3) level on narrow range indication. With pressurizer pressure below 1750 psia and four safety injection tanks operable, a minimum pressure of 254 psig and a contained borated water volume of between (962 ft^3) 39% level on wide range indication and 72% (1914 ft^3) level on narrow range indication is allowable.

* See Special Test Exception 3.10.6.

** N_2 vent valves may be cycled as necessary to maintain the required N_2 cover pressure per 3.5.1.d.

*** In MODE 4 operation for pressurizer pressure less than 430 psia the safety injection valves may be closed.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water level and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of $\geq$7% of tank narrow range level, by verifying the boron concentration of the safety injection tank solution is between 4000 and 4400 ppm.

c. At least once per 31 days when the RCS pressure is above 715 psia by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated the RCS pressure signals exceeds 515 psia, and

2. Upon receipt of a safety injection actuation test signal.

e. At least once per 18 months by verifying OPERABILITY of the RCS-SIT differential pressure alarm by simulating RCS pressure >715 psia with SIT pressure <600 psig.

f. At least once per 18 months, when the SIT's are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

3/4.5.2 ECCS SUBSYSTEMS

HOT STANDBY, STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
SI 604* SI 609*	High Pressure Hot Leg Injection Line Isolation	Closed

* Requirement applicable only if SI 604, SI 609, SI 321, SI 331 not each supplied by an independent and redundant emergency power source (four sources total).

b. At least once per 31 days by:

1. Verifying that each valve (manual, powered operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casing and accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash clothing, etc) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. The visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown; by:*

1. Verifying that each automatic valve in the flow path actuates to its correct position on a SIAS and RAS test signal, and

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open, the HPSI and LPSI pump minimum bypass recirculation flow line isolation valves close, and the LPSI pumps stop.

f. By verifying that each of the following pumps develop the indicated differential pressure at or greater than their respective minimum allowable bypass recirculation flow rates when tested pursuant to Specification 4.0.5:

1. High-Pressure Safety Injection pump $\geq$ 1800 psid.

2. Low Pressure Safety Injection pump $\geq$ 190 psid.

---

* The testing sequence shall not allow actual initiation of flow to the RCS.

g. By verifying that correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve's stroke testing or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

HPSI System Valve Number	LPSI System Valve Number	Hot Leg Injection Valve Number
a. SI-617, SI-616	a. SI-615, SI-306	a. SI-604
b. SI-627, SI-626	b. SI-625, SI-307	b. SI-609
c. SI-637, SI-636	c. SI-635	c. SI-321
d. SI-647, SI-646	d. SI-645	d. SI-331

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verify the following flow rates:

HPSI System - Single Pump	LPSI System - Single Pump
a. Injection Leg 1, 277 ± 5 gpm	a. Injection Leg 1, 2450 ± 50 gpm
b. Injection Leg 2, 277 ± 5 gpm	b. Injection Leg 2, 2450 ± 50 gpm
c. Injection Leg 3, 277 ± 5 gpm	c. Injection Leg 3, 2450 ± 50 gpm
d. Injection Leg 4, 277 ± 5 gpm	d. Injection Leg 4, 2450 ± 50 gpm

Simultaneous Hot Leg and Cold Injection - Single Pump

a. Hot Leg, 545 ± 20 gpm
b. Cold Legs Total 545 ± 20 gpm

ECCS SUBSYSTEMS

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem, comprised of the following shall be OPERABLE:

a. An OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODE 4

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements 4.5.2.

3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank (RWT) shall be OPERABLE with:

a. A minimum contained borated water volume as specified in Figure 3.1-2, of Specification 3.1.2.5,

b. A boron concentration between 4000 and 4400 ppm of boron, and

c. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the contained borated water volume in the tank, and

   2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the (outside) air temperature is outside the 60°F to 120°F range.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.6.1.1 CONTAINMENT INTEGRITY

See Applicant's SAR.

3/4.6.1.2 CONTAINMENT LEAKAGE

See Applicant's SAR.

3/4.6.1.3 CONTAINMENT AIR LOCKS

See Applicant's SAR.

3/4.6.1.4 CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZER SYSTEM

See Applicant's SAR.

3/4.6.1.5 INTERNAL PRESSURE

See Applicant's SAR.

3/4.6.1.6 AIR TEMPERATURE

See Applicant's SAR.

3/4.6.1.7 CONTAINMENT STRUCTURE INTEGRITY

See Applicant's SAR.

3/4.6.1.8 CONTAINMENT VENTILATION SYSTEM

See Applicant's SAR.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days be verifying that each valve (manual, power operated or automatic) in the flow path is positioned to take suction from the RWT on a Containment Spray Actuation test signal.

b. By verifying, that each pump develops the indicated differential pressure of $\geq$ 200 psid at or greater than the minimum allowable bypass recirculation flow rate when tested pursuant to Specification 4.0.5.

c. At least once per 31 days by verifying that the system piping is full of water from the RWT to the header isolation valves.

d. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a CSAS and RAS test signal.

   2. Verifying that upon a Recirculation Actuation Test Signal, the containment sump isolation valves open and that a recirculation mode flow path via on OPERABLE shutdown cooling heat exchanger is established.

   3. Verifying that each spray pump starts automatically on a CSAS test signal.

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Iodine Removal System shall be OPERABLE with:

a. A spray chemical addition tank containing a level of between 90% and 100% (816 to 896 gallons) of between 33% and 35% by weight N_2H_4 solution.

b. Two spray chemical addition pumps each capable of adding N_2H_4 solution from the spray chemical addition tank to a containment spray pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the Iodine Removal System inoperable restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Iodine Removal System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Iodine Removal System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

   1. Verifying the contained solution volume in the tank, and

   2. Verifying the concentration of the N_2H_4 solution by chemical analysis.

c. By verifying, that on recirculation flow, each Spray Chemical addition pump develops a discharge pressure of 100 psig when tested pursuant to Specification 4.0.5.

d. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a CSAS test signal.

   2. Verifying that each spray chemical addition pump starts automatically on a CSAS test signal.

e. At least once per 5 years by verifying each solution flow rate (to be determined during pre-operational tests) from the following drain connections in the spray additive system:

1. (Drain line location) 0.63 ± 0.02 gpm.
2. (Drain line location) 0.63 ± 0.02 gpm.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

See Applicant's SAR.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valve(s) specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1. Unless specifically permitted by the Applicant's SAR the containment purge system shall be closed when operating at RATED THERMAL POWER.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 and the containment purge valve specified in the Applicant's SAR inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by verifying that on its applicable actuation test signal, each isolation valve actuates to its required position. Valves which may be required to open or close following an accident will be actuated to demonstrate their capability to achieve both positions.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVE ACTUATION TIMES

Penetration Number	Valve Number	Description	Valve Location Relative to Containment	ESF Actuation Signal	Required Post-Accident Valve Position	Maximum Valve Actuation Time (sec)
A. Remotely Actuated Valves						
11	SI-331	Hot Leg Injection Valve	Outside	None	Open	10
12	SI-321	Hot Leg Injection Valve	Outside	None	Open	10
13 14 15 16	SI-616, 617 SI-626, 627 SI-636, 637 SI-646, 647	High Pressure Cold Leg Injection Valves	Outside	SIAS	Open	10
17 18 19 20	SI-615 SI-625 SI-635 SI-645	Low Pressure Cold Leg Injection Valves	Outside	SIAS	Open	10
23	SI-673	Containment sump isolation valve	Inside	RAS	Open	20
	SI-674	Containment sump isolation valve	Outside	RAS	Open	20
24	SI-675	Containment sump isolation valve	Inside	RAS	Open	20
	SI-676	Containment sump isolation valve	Outside	RAS	Open	20
27	SI-690	Shutdown Cooling Warmup bypass valve	Outside	None	Open or Closed	30
	SI-656	Shutdown Cooling isolation valve	Outside	None	Open or Closed	80
	SI-654	Shutdown Cooling isolation valve	Inside	None	Open or Closed	80

TABLE 3.6-1 (Cont'd)

CONTAINMENT ISOLATION VALVE ACTUATION TIMES

Penetration Number	Valve Number	Description	Valve Location Relative to Containment	ESF Actuation Signal	Required Post-Accident Valve Position	Maximum Valve Actuation Time (sec)
28	SI-691	Shutdown Cooling Warmup bypass valve	Outside	None	Open or Closed	30
	SI-655	Shutdown Cooling isolation valve	Outside	None	Open or Closed	80
	SI-653	Shutdown Cooling isolation valve	Inside	None	Open or Closed	80
29	SI-682	Safety Injection Tank fill and drain isolation valve	Inside	SIAS	Closed	5
40	CH-523 CH-516	CVCS Letdown Line Isolation Valves	Outside Inside	CIAS CIAS/SIAS	Closed Closed	5 5
41	CH-524	CVCS Charging Line Isolation Valves	Outside	None	Open or Closed	5
43	CH-505 CH-506	Reactor Coolant Pump Controlled Bleedoff Containment Isolation Valves	Outside Inside	CIAS CIAS	Closed Closed	5 5
44	CH-560 CH-561	Reactor Drain Tank Suction Isolation Valves	Inside Outside	CIAS CIAS	Closed Closed	5 5
45	CH-580	Reactor Makeup Water Supply Isolation Valve to the RDT	Outside	CIAS	Closed	5
57	CH-255	Seal Injection Containment Isolation Valve	Outside	None	Open or Closed	5

B. Manual Valves

29	SI-463	Safety Injection Tank Fill and Drain Isolation Valve	Outside	None	Closed	Not Applicable

TABLE 3.6-1 (Cont'd)

CONTAINMENT ISOLATION VALVE ACTUATION TIMES

Penetration Number	Valve Number	Description	Valve Location Relative to Containment	ESF Actuation Signal	Required Post-Accident Valve Position	Maximum Valve Actuation Time (sec)
41	CH-393 CH-854	CVCS Charging Line Isolation Valves	Inside Outside	None None	Closed Closed	Not Applicable Not Applicable
C. Check Valves						
11 12	SI-533 SI-523	Hot Leg Injection Line Isolation Valve	Inside	None	Open	Not Applicable
13 14 15 16	SI-113 SI-123 SI-133 SI-144	High Pressure Cold Leg Injection Line Isolation Valve	Inside	None	Open	Not Applicable
17 18 19 20	SI-114 SI-124 SI-134 SI-144	Low Pressure Injection Line Isolation Valves	Inside	None	Open	Not Applicable
41	CH-431 CH-433	CVCS Charging Line Isolation Valves	Inside	None	Open or Closed	Not Applicable
45	CH-494	Reactor Makeup Water Supply Isolation Valve to the RDT	Inside	None	Closed	Not Applicable
57	CH-835	Seal Injection Containment Isolation Valve	Inside	None	Open or Closed	Not Applicable

3/4.6.4 COMBUSTIBLE GAS CONTROL

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.6.4.1 HYDROGEN MONITORS

See Applicant's SAR.

3/4.6.4.2 ELECTRIC HYDROGEN RECOMBINERS

See Applicant's SAR.

3/4.6.4.3 HYDROGEN PURGE CLEANUP SYSTEM

See Applicant's SAR.

3/4.6.4.4 HYDROGEN MIXING SYSTEMS

See Applicant's SAR.

3/4.6.4.5 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM

See Applicant's SAR.

3/4.6.4.6 VACUUM RELIEF VALVES

See Applicant's SAR.

3/4.6.5 IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

See Applicant's SAR.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift setting as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3 and 4*.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam line code safety valves inoperable per steam generator, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam line code safety valves associated with the operating steam generator, otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provision of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

---

* Until the steam generators are no longer required for heat removal.

** The maximum number of inoperable safety valves on any operating steam generator is four (4).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING (+1%)*	MINIMUM RATED CAPACITY**
	Line No. 1	Line No. 2		
a.	See Applicants' SAR	See Applicants' SAR	1255 psig	904,000 lb/hr
b.	"	"	1255 psig	904,000 lb/hr
c.	"	"	1290 psig	931,000 lb/hr
d.	"	"	1290 psig	931,000 lb/hr
e.	"	"	1315 psig	950,000 lb/hr
f.	"	"	1315 psig	950,000 lb/hr
g.	"	"	1315 psig	950,000 lb/hr
h.	"	"	1315 psig	950,000 lb/hr
i.	"	"	1315 psig	950,000 lb/hr
j.	"	"	1315 psig	950,000 lb/hr

* The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

** Capacity is rated at lift setting +3% accumulation. These capacities are based on a minimum total capacity of 19,000,000 lb/hr at 1355 psig (1315 psig + 3% accumulation).

TABLE 3.7-2

MAXIMUM ALLOWABLE VARIABLE OVERPOWER HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING TWO LOOP OPERATION WITH FOUR PUMPS OPERATING

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Variable Overpower Level Trip Setpoint (Percent of RATED THERMAL POWER)
1	94.1
2	83.6
3	73.2
4	62.7

3/4.7.1.2 Emergency Feedwater System

See Applicant's SAR.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least __*__ feet (300,000 gallons).

APPLICABILITY: MODES 1, 2, 3

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the applicable alternative service water system with a water volume of at least 300,000 gallons as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with an OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The applicable alternative water source shall be demonstrated OPERABLE at least once per 12 hours by verifying:

a. That the alternate water source to Emergency Feedwater System isolation valves are either open or OPERABLE whenever the alternate water source is the supply source for the emergency feedwater purges, and

b. That the alternate water source contains a water level of at least __*__ feet (300,000 gallons) whenever the alternate water source is the supply source for the emergency feedwater pumps.

* See Applicant's SAR.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 micro- curies/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 2, 3, and 4 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2, 3, and 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than *°F when the pressure of the secondary coolant in the steam generator is greater than * psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to * psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above *°F.

SURVEILLANCE REQUIREMENTS

4.7.2 a. The pressure in the secondary side of the steam generators shall be determined to be less than * psig at least once per hour when the temperature of the secondary coolant is less than * °F.

---

* See Applicant's SAR.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

See Applicant's SAR.

3/4.7.4 SERVICE WATER SYSTEM

See Applicant's SAR.

3/4.7.5 EMERGENCY COOLING POND

See Applicant's SAR.

3/4.7.6 FLOOD PROTECTION

See Applicant's SAR.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

See Applicant's SAR.

3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM

See Applicant's SAR.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers listed in Tables 3.7-3a and 3.7-3b shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

See Applicant's SAR.

SURVEILLANCE REQUIREMENTS

See Applicant's SAR.

TABLE 3.7-3a

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
(1)	Steam Generator (36' above hot leg center line)	(1)	(1)	(1)
(1)	Reactor Coolant Pump (14' above hot leg center line)	(1)	(1)	(1)

See Applicant's SAR for other snubbers not in CESSAR scope.

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-3a provided that a revision to Table 3.7-3a is included with the next License Amendment request.

** Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-3A is included with the next License Amendment request.

(1) See Applicant's SAR.

TABLE 3.7-3b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

See Applicant's SAR - (None in CESSAR Scope)

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-3b provided that a revision to Table 3.7-3b is included with the next License Amendment request.

** Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-3b is included with the next License Amendment request.

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

See Applicant's SAR.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.7.11.1 FIRE SUPPRESSION WATER SYSTEM

See Applicant's SAR.

3/4.7.11.2 SPRAY AND/OR SPRINKLER SYSTEM

See Applicant's SAR.

3/4.7.11.3 CO_2 SYSTEMS

See Applicant's SAR.

3/4.7.11.4 HALON SYSTEM

See Applicant's SAR.

3/4.7.11.5 FIRE HOSE STATIONS

See Applicant's SAR.

3/4.7.11.6 YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

See Applicant's SAR.

3/4.7.12 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

See Applicant's SAR.

3/4.7.13 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

See Applicant's SAR.

3/4.7.14 SHUTDOWN COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low-pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within one hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within one hour or be in at least HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both shutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.14 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot legs, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is > (*) psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure > (*) psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure > 700 psia.

* See Applicant's SAR.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.8.1.1 OPERATING

See Applicant's SAR.

3/4.8.1.2 SHUTDOWN

See Applicant's SAR.

3/4.8.2 D.C. SOURCES

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.8.2.1 OPERATING

See Applicant's SAR.

3/4.8.2.2 SHUTDOWN

See Applicant's SAR.

3/4.8.3 ONSITE POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.8.3.1 OPERATING

See Applicant's SAR.

3/4.8.3.2 SHUTDOWN

See Applicant's SAR.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.8.4.1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

See Applicant's SAR.

3/4.8.4.2 MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION DEVICES

See Applicant's SAR.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling pool shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2150 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing $\geq$ 4000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2150 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling pool shall be determined by chemical analysis at least once per 72 hours.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two sources range neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

See Applicant's SAR.

3/4.9.5 COMMUNICATIONS

See Applicant's SAR.

3/4.9.6 REFUELING MACHINE OPERABILITY

See Applicant's SAR.

3/4.9.7 CRANE TRAVEL - SPENT FUEL POOL BUILDING

See Applicant's SAR.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

* The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation*.

APPLICABILITY: MODE 6* when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loop to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving an increase in reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

* The shutdown cooling loops may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

See Applicant' SAR.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet, 8 inches of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 STORAGE POOL AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

See Applicant's SAR.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s) or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODE 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEA'S at least once per 2 hours by consideration of at least the following factors:

---

* Operation in MODE 3 shall be limited to 6 consecutive hours.

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification (3.4.1.1) and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

3/4.10.4 CEA POSITION AND REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.10.5 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.10.5 The Safety Injection Tank isolation valve requirement of Specification 3.5.1a may be suspended during partial stroke testing of the low pressure safety injection check valves (SI-114, SI-124, SI-134, SI-144) provided:

a. That power to the isolation valve is restored and the SIAS signal is not overridden.

b. Only one isolation valve at a time is closed during the testing for no longer than one hour.

c. That the valve is opened with power removed before the next isolation valve is closed.

APPLICABILITY: While partial stroke testing of the low pressure injection check valves during normal plant operation.

ACTION: If requirement 3.5.1a was suspended to perform the Specification 3.10.5 partial stroke test and if any of the Specification 3.10.5 requirements are not met during the 3.10.5 partial stroke testing, the Limiting Condition for Operation shall revert to Specification 3.5.1 and the 3.5.1 ACTION shall be applicable.

SURVEILLANCE REQUIREMENTS

4.10.5.1 A valve alignment shall be performed within 4 hours following completion of testing to verify that all valves operated during this testing are restored to their normal positions and that power is removed to the SIT isolation valves.

3/4.10.6 SAFETY INJECTION TANK PRESSURE

LIMITING CONDITION FOR OPERATION

3.10.6 The safety injection tank (SIT) pressure of Specification 3.5.1d may be suspended for low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER;

b. The SITs have been filled per Specification 3.5.1b and pressurized to 175 to 225 psig below the RCS pressure;

c. All valves in the injection lines from the SITs to the RCS are open and the SITs are capable of injecting into the RCS if there is a decrease in RCS pressure.

APPLICABILITY: MODES 2 and 3

ACTION:

If all the SITs do not meet the level and pressure requirements of Specification 3.10.6, restore all the SITs to meet these requirements or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.6.1 The THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during low pressure PHYSICS TESTS.

4.10.6.2 Every 8 hours verify:

a. All the SITs levels meet the requirements of Specificaton 3.5.1b.

b. All the SITs pressures meet the requirements of Specificaton 3.10.8.

c. The valve alignment from the SITs to the RCS has not changed.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but, in accordance with 10CFR50.36, are not a part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required Containment Spray Systems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification have been provided for a limited number as specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test inter- vals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activites must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activites required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition taked precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, assuming the insertion of the regulating CEA's are within the limits set by 3.1.3.6, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS Tc. The most restrictive condition occurs at EOL, with Tc at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR safety analyses. With Tc less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and shutdown margin requirements are set to ensure that reactivity transients resulting from an inadvertant single CEA withdrawal event are minimal.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature and (5) to ensure consistency with the FSAR Safety Analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 6.0% Δk/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires (*) gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure considerations on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one OPERABLE charging pump, whenever the reactor coolant level is below the bottom of the pressurizer, is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% Δk/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F and 120°F. This condition requires (*) gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

The values of water volume, temperature and boron concentration in the refueling water tank are provided to ensure that the assumptions used in the initial conditions of the LOCA Safety Analyses remain valid.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

With the RCS temperature below 210°F while in Modes 5 and 6, a source of borated water is required to be available for reactivity control and makeup for losses due to contraction and evaporation. The requirement of 33,500 gallons of 4000 ppm borated water in either the refueling water tank or spent fuel pool ensures that this source is available.

BORON DILUTION ALARMS

The startup channel high neutron flux alarms alert the operator to an inadvertent boron dilution. Both channels must be operating to assure detection of a

* See Applicant's SAR.

boron dilution event by the high neutron flux alarms. If one or both of the alarms are inoperable at anytime, the bases for ACTION statements are as follows:

a. One startup channel high neutron flux alarm not operating:

   With only one startup channel high neutron flux alarm OPERABLE while in MODE 3, 4, 5, or 6, a single failure to the alarm could prevent detection of boron dilution. By periodic monitoring of the RCS boron concentration by either boronometer or RCS sampling, a decrease in the boron concentration during an inadvertent boron dilution event will be observed. This provides a diverse and redundant method of detection of boron dilution, with sufficient time for termination of the event before return to inadvertent criticality.

b. Both startup channel high neutron flux alarms not operating:

   When both startup channel high neutron flux alarms are inoperable, there is no means of alarming on high neutron flux when subcritical. Therefore, simultaneous use of boronometer and RCS sampling to monitor the RCS boron concentration provides diverse and redundant indications of an inadvertent boron dilution. This will allow detection with sufficient time for termination of boron dilution before return to inadvertent criticality.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This

distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted of fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{cold} greater than or equal to 550°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. Specifically, a programmed insertion schedule will be used to cycle the CEAs between the full out position ("FULL OUT" LIMIT) and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing any effects. To accommodate this programmed insertion schedule, the fully withdrawn position was redefined, in some cases, to be 144.75 inches (193 steps) or greater.

The establishment of LSSS and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short term behavior

relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors wihtin the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that 1) the minimum SHUTDOWN MARGIN is maintained, and 2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 14.0 kw/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPC's (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Specification 3.2.1 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty penalty factors plus those associated with start-up acceptance cr iteria are also included in the CPC's.

3/4.2.2 RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal or to greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured planar radial peaking factors after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT-T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is given by:

$$P_{tilt}/P_{untilt} = 1 + T_q$$

where AZIMUTHAL POWER TILT (T_q) is calculated from the following functional form: $T_q = T'_q \cdot g \cdot \cos(\theta - \theta_o)$

T'_q	is the tilt amplitude
g	is the normalized radial tilt function
θ	is the azimuthal angle
θ_o	is the azimuthal orientation of maximum tilt
P_{tilt}/P_{untilt}	is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPC's), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minimum DNBR Limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPC's. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

The DNBR penalty factors listed in Section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of core average AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and ESFAS instrumentation systems and by- passes ensure that 1) the associated ESFAS action and/or reactor trip will be ini- tiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems if required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of limiting fault and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every four hours. If the second CEAC fails, the CPC's in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If these margins are not maintained, a reactor trip will occur.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.2 IN-CORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10CFR50.

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

See Applicant's SAR.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

See Applicant's SAR.

3/4.3.4 TURBINE OVERSPEED PROTECTION

See Applicant's SAR.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.22 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations required that at least two loops (either shutdown cooling or RCS) be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two shutdown cooling loops be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. A flowrate of at least 4000 gpm will circulate one equivalent reactor coolant system volume of 12,097 cubic feet in approximately 23 minutes. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold leg less than or equal to * °F during cooldown or * °F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 150°F above each of the RCS cold leg temperatures.

* See Applicant's SAR.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE and during operations at low temperatures, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the reactor coolant system during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class IE heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhance the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4.4 STEAM GENERATORS

See Applicant's SAR.

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

See Applicant's SAR.

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value. A threshold value of 1 gpm is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The total steam generator tube leakage limit of 1 gpm, for both steam generators ensures that the dosage contribution from the tube leakage will be limited to less than Part 100 guidelines for infrequent and limiting fault events.

PRESSURE BOUNDARY LEAKAGE of any magnitude may be indicative of an impending failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensures that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentration to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Reducing Tc to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance require- ments provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. There thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the

inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate or interest must be analyzed on an individual basis.

The heatup and cooldown limit curves (Figure 3.4-2) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials for each Applicant are tested to determine their initial RT_{NDT}; the results of these tests will be shown in the Applicant's SAR. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence can be predicted using Chapter 5, Figure 5.3-6 and the recommendations of Regulatory Guide 1.99 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves shown on Figure 3.4-2 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, and Appendix H of 10CFR50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retraining materials, with the exception of the reactor pressure vessel, has been assumed to be 60°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limitations imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to * °F during cooldown or * °F during heatup. Either one of the two SCS suction line relief valves provide adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 150°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

3/4.4.9 STRUCTURAL INTEGRITY

The inspection programs for the safety-related ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

* See Applicant's SAR.

3/4.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the SIS safety injection tanks (SIT) ensures that a sufficient volume of borated water will be immediately forced into the reactor through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODES 1, 2, 3, or 4.

A minimum of 25% (1790 ft^3) and a maximum of 75% (1927 ft^3) on the SIT narrow range level instruments of borated water at normal operating conditions are used in the safety analysis as the volume in the SIT's. To allow for instrument accuracy, 28% (1802 ft^3) and 72% (1914 ft^3) on the narrow range instrument, are specified in the technical specification.

A minimum of 593 psig and a maximum pressure of 632 psig at normal operating conditions are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the technical specification.

For MODES 3 and 4 operation with pressurizer pressure less than 1750 psia a minimum of 57% (1361 ft^3) on the SIT wide range level instrument and a maximum of 75% (1927 ft^3) on the SIT narrow range level instrument of borated water per tank is required when 3 safety injection tanks are operable and a minimum of 36% (908 ft^3) on the SIT wide range level instruments and a maximum of 75% (1927 ft^3) on the SIT narrow range instruments is required when 4 safety injection tanks are operable at a minimum pressure of 235 psig. To allow for instrument inaccuracy, 60% (1415 ft^3) on the wide range instrument and 72% (1914 ft^3) on the narrow range instrument when 3 safety injection tanks are operable, and 39% (962 ft^3) on the wide range instrument and 72% (1914 ft^3) on the narrow range instrument when 4 SIT's are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

A boron concentration of 4000 ppm minimum and 4400 ppm maximum are used in the safety analysis.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SIT's by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SIT's.

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SIT's may still be vented during post LOCA long term cooling. Venting the SIT's provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCA's.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switches keylocked open. These precautions ensure that the SIT's are available during a Limiting Fault.

The limits of operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperature. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems with the RCS temperature equal to or above 350°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA, assuming the loss of one subsystem through any single failure consideration. Either subsystem in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the per cladding temperature within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides a long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis fo the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA*. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

*The following test conditions apply during flow balance tests ensure that the ECCS subsystems are adequately tested.

1) The pressurizer pressure is 15 psia.

2) The mini flow bypass recirculation lines are aligned for injection.

3) For the LPSI system, $\left\{\begin{matrix}\text{add}\\\text{subtract}\end{matrix}\right\}$ 3.2 gpm $\left\{\begin{matrix}\text{to}\\\text{from}\end{matrix}\right\}$ the 2450 gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level $\left\{\begin{matrix}\text{exceeds}\\\text{is less than}\end{matrix}\right\}$ the difference of RCS water level above the Cold Leg centerline.

The term "minimum bypass recirculation flow", as used in paragraphs 4.5.2.e.3 and 4.5.2.f, refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in paragraph 4.5.2.f verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the Refueling Water Tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10 percent margin is available to permit 20 minutes of Engineered Safeguard Features Pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control element assembly. The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analysis remain valid. These assumptions are consistent with the LOCA analysis.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

See Applicant's SAR.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the Iodine Removal system ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure adequate chemical available to remove iodine from the containment atmosphere following a LOCA.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

See Applicant's SAR.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC54 through GDC57 of Appendix A to 10CFR50. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Unless otherwise stated no containment purging should be done while operating at 100% of RATED THERMAL POWER. This is necessary as it is one of the assumptions used in the LOCA Safety analyses.

3/4.6.4 COMBUSTIBLE GAS CONTROL

See Applicant's SAR.

3/4.6.5 IODINE CLEANUP SYSTEMS

See Applicant's SAR.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1382 psig) of its design pressure of 1255 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 18,660,000 lbs/hr which is 104.6 percent of the total secondary steam flow of 17,830,000 lbs/hr at 100% RATED THERMAL POWER plus 2% uncertainity.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop, four pump operation

$$SP = \left(\frac{10-N}{10}\right) \times 104.6$$

where:

SP = reduced reactor trip setpoint in percent of Rated Thermal Power. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = total number of secondary safety valves for one steam generator.

N = the number of inoperable secondary safety valves on the steam generator with the greater number of inoperable valves.

104.6 = the ratio of the total relieving capacity of all twenty (20) secondary safety valves 18,660,000 lb/hr at 1397 psig - 50 psi pressure drop to the inlet of the safety valves) over the secondary steam flow at 100% Rated Thermal Load (17,830,000 lbs/hr) plus 2% uncertainity.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

See Applicant's SAR.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere with concurrent total loss of off-site power. The OPERABILITY of the condensate storage tank with the minimum water volume also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours followed by an orderly cooldown to the shutdown cooling entry temperature (350°F) with concurrent total loss of off-site power.

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumption used in the safety analyses.

3/4.7.1.5 MAIN STEAM ISOLATION VALVES

The OPERABILITY of the main steam isolation valves ensures that no more than one steam generator will blowdown in the event of a large steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitation to * °F and * psig are based on a steam generator RT_{NDT} of * °F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

See Applicant's SAR.

3/4.7.4 SERVICE WATER SYSTEM

See Applicant's SAR.

3/4.7.5 EMERGENCY COOLING POND

See Applicant's SAR.

3/4.7.6 FLOOD PROTECTION

See Applicant's SAR.

---

* See Applicant's SAR.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

See Applicant's SAR.

3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM

See Applicant's SAR.

3/4.7.9 SNUBBERS

The hydraulic snubbers included within the scope of CESSAR are identified on Table 3.7-3a of Section 3/4.7.9.

The hydraulic snubbers are required to be OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

The hydraulic snubbers included in the CESSAR scope are provided with seals fabricated from materials which have been demonstrated compatible with their operating environment. Inspection program intervals and acceptance criteria for the CESSAR scope snubbers will be included in the Applicant's SAR, consistent with the program set-up by the Applicant for other snubbers in the plant.

3/4.7.10 SEALED SOURCE CONTAMINATION

See Applicant's SAR.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

See Applicant's SAR.

3/4.7.12 FIRE BARRIER PENETRATIONS

See Applicant's SAR.

3/4.7.13 AREA TEMPERATURE MONITORING

See Applicant's SAR.

3/4.7.14 SHUTDOWN COOLING

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling subsystem inoperable for any reason minimize the time exposure of the plant to an accident event occuring concurrent with the failure of a component on the other shutdown cooling subsystem.

3/4.8.1 A.C. SOURCES

See Applicant's SAR.

3/4.8.2 D.C. SOURCES

See Applicant's SAR.

3/4.8.3 ONSITE POWER DISTRIBUTION

See Applicant's SAR.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

See Applicant's SAR.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform BORON CONCENTRATION is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2150 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

See Applicant's SAR.

3/4.9.5 COMMUNICATIONS

See Applicant's SAR.

3/4.9.6 REFUELING MACHINE OPERABILITY

See Applicant's SAR.

3/4.9.7 CRANE TRAVEL-SPENT FUEL POOL BUILDING

See Applicant's SAR.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation and circulating reactor coolant at a flowrate of $\geq$ 4000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the

reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) that the ΔT across the core will be maintained less than 75°F.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23' of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23' of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

See Applicant's SAR.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND WATER LEVEL - STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 STORAGE POOL AIR CLEANUP SYSTEM

See Applicant's SAR.

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. Although testing will be initiated from MODE 2, temporary entry into MODE 3 is necessary during some CEA worth measurements. A reasonable recovery time is available for return to MODE 2 in order to continue physics testing.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth, 2) determine the reactor stability index and damping factor under xenon oscillation conditions, 3) determine power distributions from non-normal CEA configurations, 4) measure rod shadowing factors, and 5) measure temperature and power coefficients.

This special test exception permits the MTC to exceed the limits of Specification 3.1.1.3 during the performance of physics tests.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality with less than four reactor coolant pumps in operation and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CEA POSITION AND REGULATING CEA INSERTION LIMITS

This special test exception permits the CEA's to be positioned beyond the insertion limits during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of a SIAS, therefore, system capability should not be affected.

3/4.10.6 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 300°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

TECHNICAL SPECIFICATIONS

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

See Applicant's SAR.

5.2 CONTAINMENT

See Applicant's SAR.

5.3 REACTOR CORE

5.3.1 FUEL ASSEMBLIES

The reactor core shall contain 241 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods or burnable poison rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of approximately 1900 grams uranium. Each burnable poison rod shall have a nominal active poison length of 136 inches. The initial core loading shall have a maximum enrichment of 3.30 weight percent U-235. Reloaa fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.0 weight percent U-235.

5.3.2 CONTROL ELEMENT ASSEMBLIES

The reactor core shall contain 76 full length and 13 part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

5.4.1 DESIGN PRESSURE AND TEMPERATURE

The reactor coolant system is designed and shall be maintained in accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements, and is also designed as follows:

a. For a pressure of 2500 psia, and

b. For a temperature of 650°F, except for the pressurizer which is 700°F.

5.4.2 VOLUME

The total water and steam volume of the reactor coolant system is 13,900 + 300/-0 cubic feet at a nominal T_{avg} of 593°F.

5.5 METEOROLOGICAL TOWER LOCATION

See Applicant's SAR.

5.6 FUEL STORAGE

See Applicant's SAR.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor Coolant System	500 system heatup and cooldown cycles at rates $\leq$ 100°F/hr.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 565°F; cooldown cycle - T_{avg} from $\geq$ 565°F to $\leq$ 200°F.
	500 pressurizer heatup and cooldown cycles at rates $\leq$ 200°F/hr.	Heatup cycle - Pressurizer temperature from $\leq$ 200°F to $\geq$ 653°F; cooldown cycle - Pressurizer temperature from $\geq$ 653°F to $\leq$ 200°F.
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psig with RCS temperature between 100°F and 400°F.
	480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow.	Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow.
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).
	1 complete loss of secondary pressure cycle.	Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle.
	15,000 power change cycles	Cycles from 15% to 100% full load, at a rate of 5% per minute, either increasing or decreasing. (30,000 cycles total)
	10^6 step changes of 100 psi and 10°F (20°F for surge line)	Pressure variations between the pressurizer pressure setpoint for backup heater actuation and spray valve opening. Temperature variations due to CEA controller; 2000 step change of 10% full power.

TABLE 5.7-1 (Cont'd)

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Pressurizer Spray Nozzle		Main Spray (4 pumps operating)
	Unlimited number of cycles	Main Spray (Less than 4 pumps operating) - with fluid $\Delta T_m \leq$ *°F.
		Auxiliary spray at various initial fluid temperatures with fluid $\Delta T_a \leq$ *°F.
	Calculate usage factor per Table 5.7-2	Main spray (less than 4 pumps operating) with fluid $\Delta T_m >$ *°F.
		Auxiliary spray with fluid $\Delta T_a >$ *°F.

ΔT_m = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.**

ΔT_a = The difference in temperature between the pressurizer and Auxiliary spray water as adjusted by the instrument correction factor.**

* See Applicant's SAR

** For instrument correction factors, see Applicant's SAR.

TABLE 5.7-2

COMPONENT CYCLIC OR TRANSIENT LIMITS

SEE APPLICANT'S SAR.