50-272

APPENDIX A/TECHNICAL SPECIFICATIONS TO
LICENSE NO. DPR-70
APPENDIX B/ENVIRONMENTAL TECHNICAL SPECIFICATIONS
to LICENSE NO. DPR-70
SALEM UNIT #1 dtd 8-13-76

— NOTICE —

THE ATTACHED FILES ARE OFFICIAL RECORDS OF THE DIVISION OF DOCUMENT CONTROL. THEY HAVE BEEN CHARGED TO YOU FOR A LIMITED TIME PERIOD AND MUST BE RETURNED TO THE RECORDS FACILITY BRANCH 016. PLEASE DO NOT SEND DOCUMENTS CHARGED OUT THROUGH THE MAIL. REMOVAL OF ANY PAGE(S) FROM DOCUMENT FOR REPRODUCTION MUST BE REFERRED TO FILE PERSONNEL.

DEADLINE RETURN DATE ______________

______________ ______________

______________ ______________

______________ ______________

RECORDS FACILITY BRANCH

SALEM NUCLEAR GENERATING STATION
UNIT 1
TECHNICAL SPECIFICATIONS

APPENDIX "A"
TO
LICENSE NO. DPR - 70

AUGUST 13, 1976

ISSUED BY THE UNITED STATES NUCLEAR REGULATORY COMMISSION

SALEM NUCLEAR GENERATING STATION

UNIT 1

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. DPR-70

INDEX

DEFINITIONS

SECTION | PAGE

1.0 DEFINITIONS

Defined Terms.. 1-1
Thermal Power.. 1-1
Rated Thermal Power.................................. 1-1
Operational Mode..................................... 1-1
Action... 1-1
Operable - Operability............................... 1-1
Reportable Occurrence................................ 1-1
Containment Integrity................................ 1-2
Channel Calibration.................................. 1-2
Channel Check.. 1-2
Channel Functional Test.............................. 1-3
Core Alteration...................................... 1-3
Shutdown Margin...................................... 1-3
Identified Leakage................................... 1-3
Unidentified Leakage................................. 1-3
Pressure Boundary Leakage............................ 1-4
Controlled Leakage................................... 1-4
Quadrant Power Tilt Ratio............................ 1-4
Dose Equivalent I-131................................ 1-4
Staggered Test Basis................................. 1-4
Frequency Notation................................... 1-5
Reactor Trip System Response Time.................... 1-5
Engineered Safety Feature Response Time.............. 1-5
Axial Flux Difference................................ 1-5
Physics Tests.. 1-5
$\bar{E}$-Average Disintegration Energy.............. 1-5

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION | Page

2.1 SAFETY LIMITS

Reactor Core.. 2-1

Reactor Coolant System Pressure............................. 2-1

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Trip System Instrumentation Setpoints................. 2-4

BASES

SECTION | Page

2.1 SAFETY LIMITS

Reactor Core.. B 2-1

Reactor Coolant System Pressure............................. B 2-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Trip System Instrumentation Setpoints................. B 2-3

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 - $T_{avg} > 200°F$	3/4 1-1
	Shutdown Margin - $T_{avg} \leq 200°F$	3/4 1-3
	Boron Dilution	3/4 1-4
	Moderator Temperature Coefficient	3/4 1-5
	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 Pump - Shutdown	3/4 1-10
	Charging Pumps - Operating	3/4 1-11
	Boric Acid Transfer Pumps - Shutdown	3/4 1-12
	Boric Acid Transfer Pumps - Operating	3/4 1-13
	Borated Water Sources - Shutdown	3/4 1-14
	Borated Water Sources - Operating	3/4 1-16
3/4.1.3	MOVABLE CONTROL ASSEMBLIES	
	Group Height	3/4 1-18
	Position Indicator Channels	3/4 1-20
	Rod Drop Time	3/4 1-21
	Shutdown Rod Insertion Limit	3/4 1-22
	Control Rod Insertion Limits	3/4 1-23
	Part Length Rod Insertion Limits	3/4 1-26

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		Page
3/4.2 POWER DISTRIBUTION LIMITS		
3/4.2.1	AXIAL FLUX DIFFERENCE	3/4 2-1
3/4.2.2	HEAT FLUX HOT CHANNEL FACTOR	3/4 2-5
3/4.2.3	NUCLEAR ENTHALPY HOT CHANNEL FACTOR	3/4 2-9
3/4.2.4	QUADRANT POWER TILT RATIO	3/4 2-11
3/4.2.5	DNB PARAMETERS	3/4 2-13
3/4.3 INSTRUMENTATION		
3/4.3.1	REACTOR TRIP SYSTEM INSTRUMENTATION	3/4 3-1
3/4.3.2	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION	3/4 3-14
3/4.3.3	MONITORING INSTRUMENTATION	
	Radiation Monitoring Instrumentation	3/4 3-35
	Movable Incore Detectors	3/4 3-39
	Seismic Instrumentation	3/4 3-40
	Meteorological Instrumentation	3/4 3-43
	Remote Shutdown Instrumentation	3/4 3-46
3/4.4 REACTOR COOLANT SYSTEM		
3/4.4.1	REACTOR COOLANT LOOPS	
	Normal Operation	3/4 4-1
3/4.4.2	SAFETY VALVES - SHUTDOWN	3/4 4-4
3/4.4.3	SAFETY VALVES - OPERATING	3/4 4-5
3/4.4.4	PRESSURIZER	3/4 4-6
3/4.4.5	STEAM GENERATORS	3/4 4-7
3/4.4.6	REACTOR COOLANT SYSTEM LEAKAGE	
	Leakage Detection Systems	3/4 4-14
	Operational Leakage	3/4 4-15

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		Page
3/4.4.7	CHEMISTRY	3/4 4-17
3/4.4.8	SPECIFIC ACTIVITY	3/4 4-20
3/4.4.9	PRESSURE/TEMPERATURE LIMITS	
	Reactor Coolant System	3/4 4-24
	Pressurizer	3/4 4-29
3/4.4.10	STRUCTURAL INTEGRITY	
	ASME Code Class 1, 2 and 3 Components	3/4 4-30
3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)		
3/4.5.1	ACCUMULATORS	3/4 5-1
3/4.5.2	ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$	3/4 5-3
3/4.5.3	ECCS SUBSYSTEMS - $T_{avg} < 350°F$	3/4 5-6
3/4.5.4	BORON INJECTION SYSTEM	
	Boron Injection Tank	3/4 5-7
	Heat Tracing	3/4 5-8
3/4.5.5	REFUELING WATER STORAGE TANK	3/4 5-9

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.6 CONTAINMENT SYSTEMS		
3/4.6.1	PRIMARY CONTAINMENT	
	Containment Integrity	3/4 6-1
	Containment Leakage	3/4 6-2
	Containment Air Locks	3/4 6-5
	Internal Pressure	3/4 6-6
	Air Temperature	3/4 6-7
	Containment Structural Integrity	3/4 6-8
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	
	Containment Spray System	3/4 6-9
	Spray Additive System	3/4 6-10
	Containment Cooling System	3/4 6-11
3/4.6.3	CONTAINMENT ISOLATION VALVES	3/4 6-12
3/4.6.4	COMBUSTIBLE GAS CONTROL	
	Hydrogen Analyzers	3/4 6-18
	Electric Hydrogen Recombiners	3/4 6-19
3/4.7	**PLANT SYSTEMS**	
3/4.7.1	TURBINE CYCLE	
	Safety Valves	3/4 7-1
	Auxiliary Feedwater System	3/4 7-5
	Auxiliary Feed Storage Tank	3/4 7-7
	Activity	3/4 7-8
	Main Steam Line Isolation Valves	3/4 7-10
	Secondary Water Chemistry	3/4 7-11

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.7.2	STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION........	3/4 7-14
3/4.7.3	COMPONENT COOLING WATER SYSTEM..........................	3/4 7-15
3/4.7.4	SERVICE WATER SYSTEM....................................	3/4 7-16
3/4.7.5	FLOOD PROTECTION..	3/4 7-17
3/4.7.6	CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM.........	3/4 7-18
3/4.7.7	AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM.......	3/4 7-22
3/4.7.8	SEALED SOURCE CONTAMINATION.............................	3/4 7-26
3/4.7.9	HYDRAULIC SNUBBERS......................................	3/4 7-28
3/4.8 ELECTRICAL POWER SYSTEMS		
3/4.8.1	A.C. SOURCES	
	Operating...	3/4 8-1
	Shutdown..	3/4 8-5
3/4.8.2	ONSITE POWER DISTRIBUTION SYSTEMS	
	A.C. Distribution - Operating...........................	3/4 8-6
	A.C. Distribution - Shutdown............................	3/4 8-7
	125-Volt D.C. Distribution - Operating..................	3/4 8-8
	125-Volt D.C. Distribution - Shutdown...................	3/4 8-10
	28-Volt D.C. Distribution - Operating...................	3/4 8-11
	28-Volt D.C. Distribution - Shutdown....................	3/4 8-13

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
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 9-5
3/4.9.6	MANIPULATOR CRANE OPERABILITY	3/4 9-6
3/4.9.7	CRANE TRAVEL - FUEL HANDLING AREA	3/4 9-7
3/4.9.8	COOLANT CIRCULATION	3/4 9-8
3/4.9.9	CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM	3/4 9-9
3/4.9.10	WATER LEVEL-REACTOR VESSEL	3/4 9-10
3/4.9.11	STORAGE POOL WATER LEVEL	3/4 9-11
3/4 9.12	FUEL HANDLING AREA VENTILATION SYSTEM	3/4 9-12
3/4.10 SPECIAL TEST EXCEPTIONS		
3/4.10.1	SHUTDOWN MARGIN	3/4 10-1
3/4.10.2	GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS	3/4 10-2
3/4.10.3	PHYSICS TESTS	3/4 10-3
3/4.10.4	NO FLOW TESTS	3/4 10-4

INDEX

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 AXIAL FLUX DIFFERENCE................................. B 3/4 2-1

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT
CHANNEL FACTORS....................................... B 3/4 2-4

3/4.2.4 QUADRANT POWER TILT RATIO............................. B 3/4 2-5

3/4.2.5 DNB PARAMETERS.. B 3/4 2-6

BASES

SECTION PAGE

3/4.3 INSTRUMENTATION

3/4.3.1 PROTECTIVE INSTRUMENTATION.............................. B 3/4 3-1

3/4.3.2 ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION........ B 3/4 3-1

3/4.3.3 MONITORING INSTRUMENTATION.............................. B 3/4 3-1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS.................................. B 3/4 4-1

3/4.4.2 and 3/4.4.3 SAFETY VALVES.............................. B 3/4 4-1

3/4.4.4 PRESSURIZER.. B 3/4 4-2

3/4.4.5 STEAM GENERATORS....................................... B 3/4 4-2

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE......................... B 3/4 4-3

3/4.4.7 CHEMISTRY.. B 3/4 4-4

3/4.4.8 SPECIFIC ACTIVITY...................................... B 3/4 4-5

3/4.4.9 PRESSURE/TEMPERATURE LIMITS............................ B 3/4 4-6

3/4.4.10 STRUCTURAL INTEGRITY.................................. B 3/4 4-12

BASES

SECTION | PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS.. B 3/4 5-1

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS............................ B 3/4 5-1

3/4.5.4 BORON INJECTION SYSTEM................................. B 3/4 5-2

3/4.5.5 REFUELING WATER STORAGE TANK (RWST)..................... B 3/4 5-2

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 6-3

3/4.6.3 CONTAINMENT ISOLATION VALVES........................... B 3/4 6-3

3/4.6.4 COMBUSTIBLE GAS CONTROL................................ B 3/4 6-4

BASES

SECTION PAGE

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE.. B 3/4 7-1

3/4.7.2 STEAM GENERATOR PRESSURE/TEMERATURE LIMITATION.......... B 3/4 7-4

3/4.7.3 COMPONENT COOLING WATER SYSTEM.......................... B 3/4 7-4

3/4.7.4 SERVICE WATER SYSTEM.................................... B 3/4 7-4

3/4.7.5 FLOOD PROTECTION.. B 3/4 7-5

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM.......... B 3/4 7-5

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM........ B 3/4 7-5

3/4 7.8 SEALED SOURCE CONTAMINATION............................. B 3/4 7-5

3/4.7.9 HYDRAULIC SNUBBERS...................................... B 3/4 7-6

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES.. B 3/4 8-1

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS....................... 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 BUILDING PENETRATIONS....................... B 3/4 9-1

3/4.9.5 COMMUNICATIONS.. B 3/4 9-1

BASES

SECTION		PAGE
3/4.9.6	MANIPULATOR CRANE OPERABILITY	B 3/4 9-2
3/4.9.7	CRANE TRAVEL - SPENT FUEL STORAGE BUILDING	B 3/4 9-2
3/4.9.8	COOLANT CIRCULATION	B 3/4 9-2
3/4.9.9	CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF 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 9-3
3/4.9.12	FUEL HANDLING AREA VENTILATION SYSTEM	B 3/4 9-3
3/4.10	SPECIAL TEST EXCEPTIONS	
3/4.10.1	SHUTDOWN MARGIN	B 3/4 10-1
3/4.10.2	GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS	B 3/4 10-1
3/4.10.3	PHYSICS TESTS	B 3/4 10-1
3/4.10.4	NO FLOW TESTS	B 3/4 10-1

INDEX

DESIGN FEATURES

SECTION | PAGE

5.1 SITE

Exclusion Area... 5-1

Low Population Zone...................................... 5-1

5.2 CONTAINMENT

Configuration.. 5-1

Design Pressure and Temperature.......................... 5-4

5.3 REACTOR CORE

Fuel Assemblies.. 5-4

Control Rod Assemblies................................... 5-4

5.4 REACTOR COOLANT SYSTEM

Design Pressure and Temperature.......................... 5-4

Volume... 5-5

5.5 METEOROLOGICAL TOWER LOCATION........................... 5-5

5.6 FUEL STORAGE

Criticality.. 5-5

Drainage... 5-5

Capacity... 5-5

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT...................... 5-6

INDEX

ADMINISTRATIVE CONTROLS

SECTION	PAGE
6.1 RESPONSIBILITY	6-1
6.2 ORGANIZATION	
Offsite	6-1
Facility Staff	6-1
6.3 FACILITY STAFF QUALIFICATIONS	6-5
6.4 TRAINING	6-5
6.5 REVIEW AND AUDIT	
6.5.1 STATION OPERATIONS REVIEW COMMITTEE	
Function	6-5
Composition	6-5
Alternates	6-5
Meeting Frequency	6-6
Quorum	6-6
Responsibilities	6-6
Authority	6-7
Records	6-7
6.5.2 NUCLEAR REVIEW BOARD	
Function	6-7
Composition	6-8
Alternates	6-8

SECTION PAGE

Consultants.. 6-8
Meeting Frequency.. 6-8
Quorum.. 6-9
Review.. 6-9
Audits.. 6-10
Authority.. 6-10
Records.. 6-11

6.6 REPORTABLE OCCURRENCE ACTION.. 6-11

6.7 SAFETY LIMIT VIOLATION.. 6-12

6.8 PROCEDURES.. 6-12

6.9 REPORTING REQUIREMENTS

6.9.1 ROUTINE REPORTS AND REPORTABLE OCCURRENCES 6-13
6.9.2 SPECIAL REPORTS.. 6-13

6.10 RECORD RETENTION.. 6-14

6.11 RADIATION PROTECTION PROGRAM.. 6-15

6.12 RESPIRATORY PROTECTION PROGRAM
6.12.1 ALLOWANCE.. 6-15
6.12.2 PROTECTION PROGRAM.. 6-16
6.12.3 REVOCATION.. 6-18

6.13 HIGH RADIATION AREA.. 6-18

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3338 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, 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).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified as a reportable occurrence in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 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

1.10 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

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 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.

SHUTDOWN MARGIN

1.13 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 rod position, and

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

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, 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.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 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.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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."

STAGGERED TEST BASIS

1.20 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,

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

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 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 loss of stationary gripper coil voltage.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.23 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.

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

PHYSICS TESTS

1.25 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 13 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\overline{E}$ 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.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

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.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for 4 and 3 loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

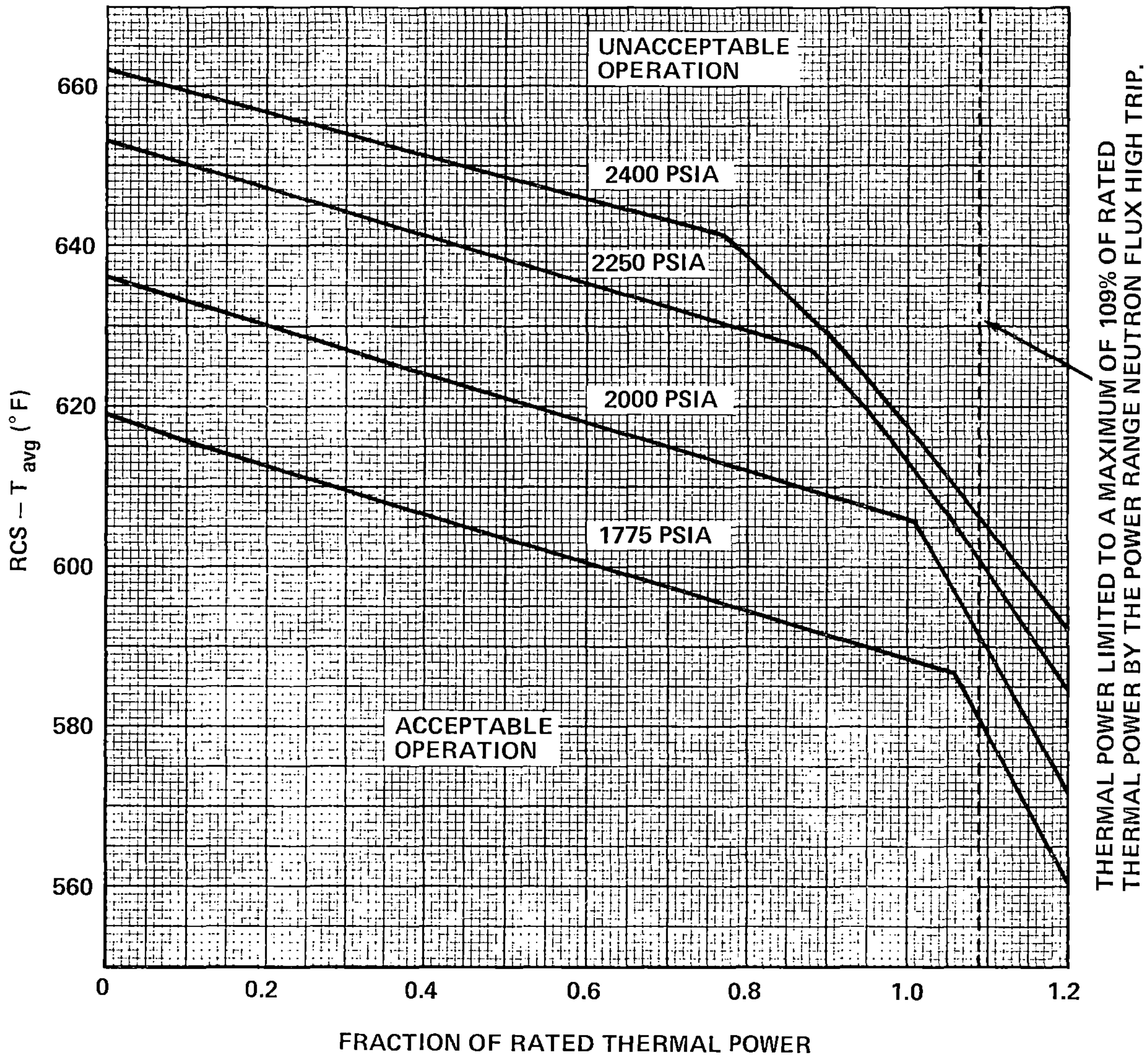


FIGURE 2.1-1. REACTOR CORE SAFETY LIMIT – FOUR LOOPS IN OPERATION

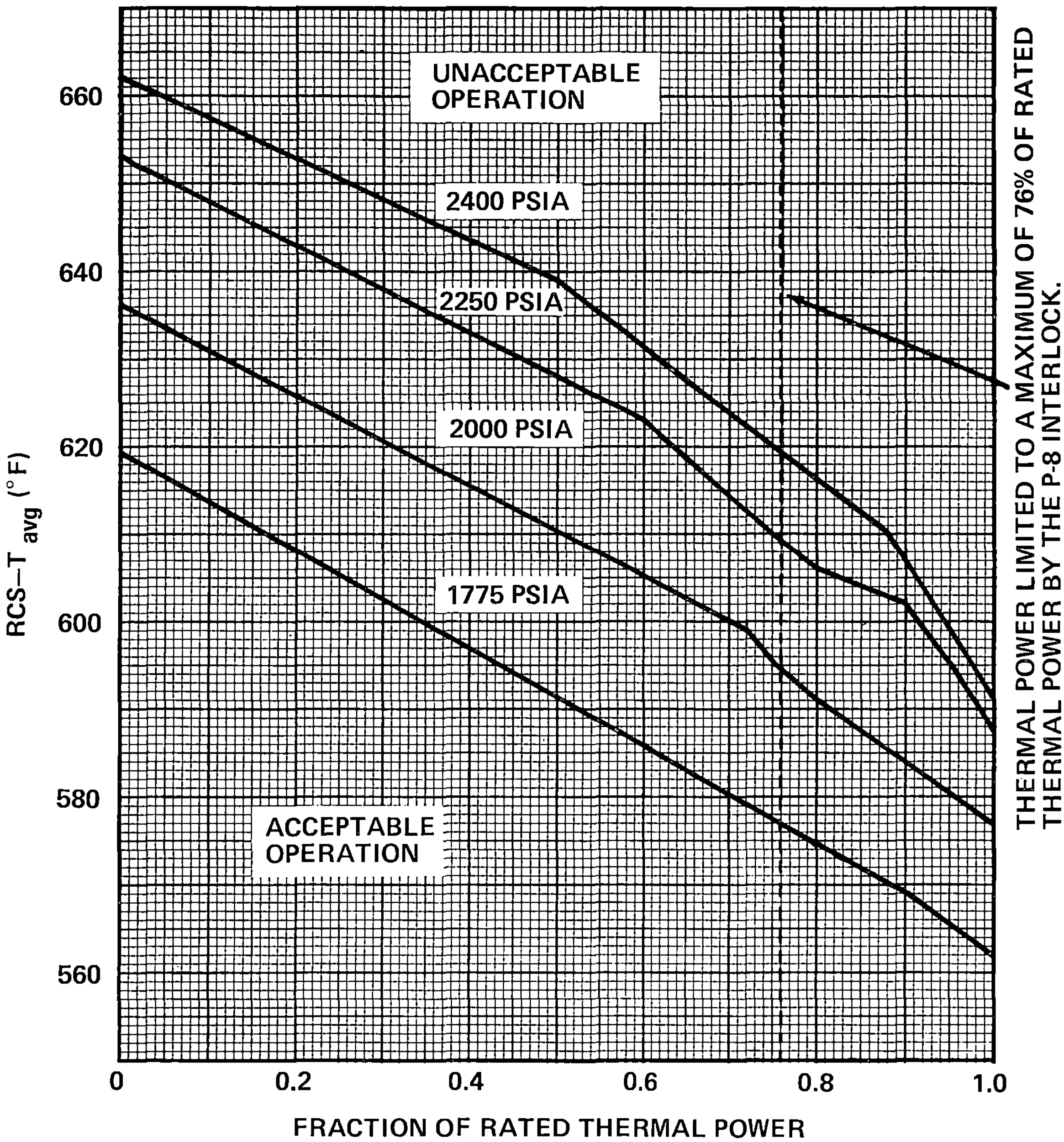


FIGURE 2.1-2. REACTOR CORE SAFETY LIMIT – THREE LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system 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 trip system 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 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 88,500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	$\geq$ 5% of narrow range instrument span-each steam generator	$\geq$ 4% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq$ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span--each steam generator	$\leq$ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pumps	$\geq$ 2900 volts-each bus	$\geq$ 2850 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	$\geq$ 56.5 Hz - each bus	$\geq$ 56.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 45 psig
B. Turbine Stop Valve Closure	$\leq$ 15% off full open	$\leq$ 15% off full open
18. Safety Injection Input from SSPS	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) (T - T') + K_3 (P - P') - f_1(\Delta I)\right]$

where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 30 secs, τ_2 = 4 secs.

S = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops	Operation with 3 Loops
$K_1 = 1.164$	$K_1 = 1.05$
$K_2 = 0.01434$	$K_2 = 0.01434$
$K_3 = 0.00073$	$K_3 = 0.00073$

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -23 percent and +10 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 1.26 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +10 percent, the ΔT trip setpoint shall be automatically reduced by 1.34 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T") - f_2(\Delta I)]$

where:

ΔT_o = Indicated ΔT at rated power

T = Average temperature, °F

$T"$ = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

K_4 = 1.080

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00119 for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 4 percent.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel 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 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.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

SAFETY LIMITS

BASES

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.2\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

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 and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 1955 Edition while the valves are designed to ANSI B 16.5, MSS-SP-66-1964, or ASME Section III-1968, which permit maximum transient pressures of up to 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

NOTE

The summary statements contained in this section provided the bases for the specifications of Section 2.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the 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.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 9 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for multiple control rod drop accidents. The analysis of a single control rod drop accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback. This transient will not result in a DNBR of less than 1.30, therefore single rod drop protection is not required.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 4 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature ΔT set point. Three loop operation above the 4 loop P-8 set point is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature ΔT channels and raising the P-8 set point to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature ΔT trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with 3 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by $\geq 1.42 \times 10^6$ lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of any one pump breaker above P-8 or the opening of two or more pump breakers below P-8. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

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 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the facility shall be placed in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery. Exceptions to these requirements shall be stated in the individual specifications.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability 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. For the time period from issuance of the Facility Operating License to the start of facility commercial operation, inservice testing of 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 1974 Edition, and Addenda through Winter 1975, except where specific written relief has been granted by the Commission.

b. For the time period following start of facility commercial operation, 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 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).

Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements. The provisions of Specification 4.0.2 are not applicable to surveillance intervals associated with inservice inspection and testing activities required by Section XI of the above ASME Boiler and Pressure Vessel Code and applicable Addenda.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - Tavg > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1.6% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 1.6% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 20,100 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1.6% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod 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 control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of specification 3.1.3.5.

---

*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

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

1. Reactor coolant system boron concentration,

2. Control rod 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% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). 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 - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 1.0% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 1.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 20,100 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1.0% $\Delta k/k$:

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

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

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

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: All MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system $<$ 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be $\geq$ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying $\geq$ 3000 gpm through the reactor coolant system.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. $\leq 0 \times 10^{-4}$ $\Delta k/k/°F$, and

b. Less negative than -5.0×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 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.

4.1.1.4.2 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 RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

*With $K_{eff} \geq 1.0$
#See Special Test Exception 3.10.3

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 541°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 541°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_{avg} is less than 551°F with the T_{avg} - T_{ref} Deviation Alarm not reset.

#With $K_{eff} \geq 1.0$.

REACTIVITY CONTROL SYSTEMS

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. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is $\geq$ 145°F when a flow path from the boric acid tanks is used.

b. 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path 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

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is $\geq$ 145°F.

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

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 1% $\Delta k/k$ at 200°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.

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid transfer pump shall be OPERABLE if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid transfer pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump 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 1% $\Delta k/k$ at 200°F; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained volume of 835 gallons,
2. Between 20,100 and 21,800 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 9690 gallons,
2. A minimum boron concentration of 2000 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source 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

4.1.2.7 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,
2. Verifying the water level of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is < 35°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained volume of 5106 gallons,
2. Between 20,100 and 21,800 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 350,000 gallons of water,
2. A minimum boron concentration of 2000 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system 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 1% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage 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

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the water level of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is < 35°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, and all part length rods which are inserted in the core, shall be OPERABLE and positioned within ± 12 steps (indicated position) of their bank demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods 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 HOT STANDBY within 6 hours.

b. With more than one full or part length rod inoperable or misaligned from the bank demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full or part length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) An analysis of the potential ejected rod worth is performed within 72 hours and the rod worth is determined to be $\leq$ 0.95% Δk at zero power and $\leq$ 0.21% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

* See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

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

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) The THERMAL POWER level is reduced to $\leq$ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown, control and part length control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within $\pm$ 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 541°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

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

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to $\leq$ 71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With $K_{eff} \geq 1.0$.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figures, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3

#With $K_{eff} \geq 1.0$.

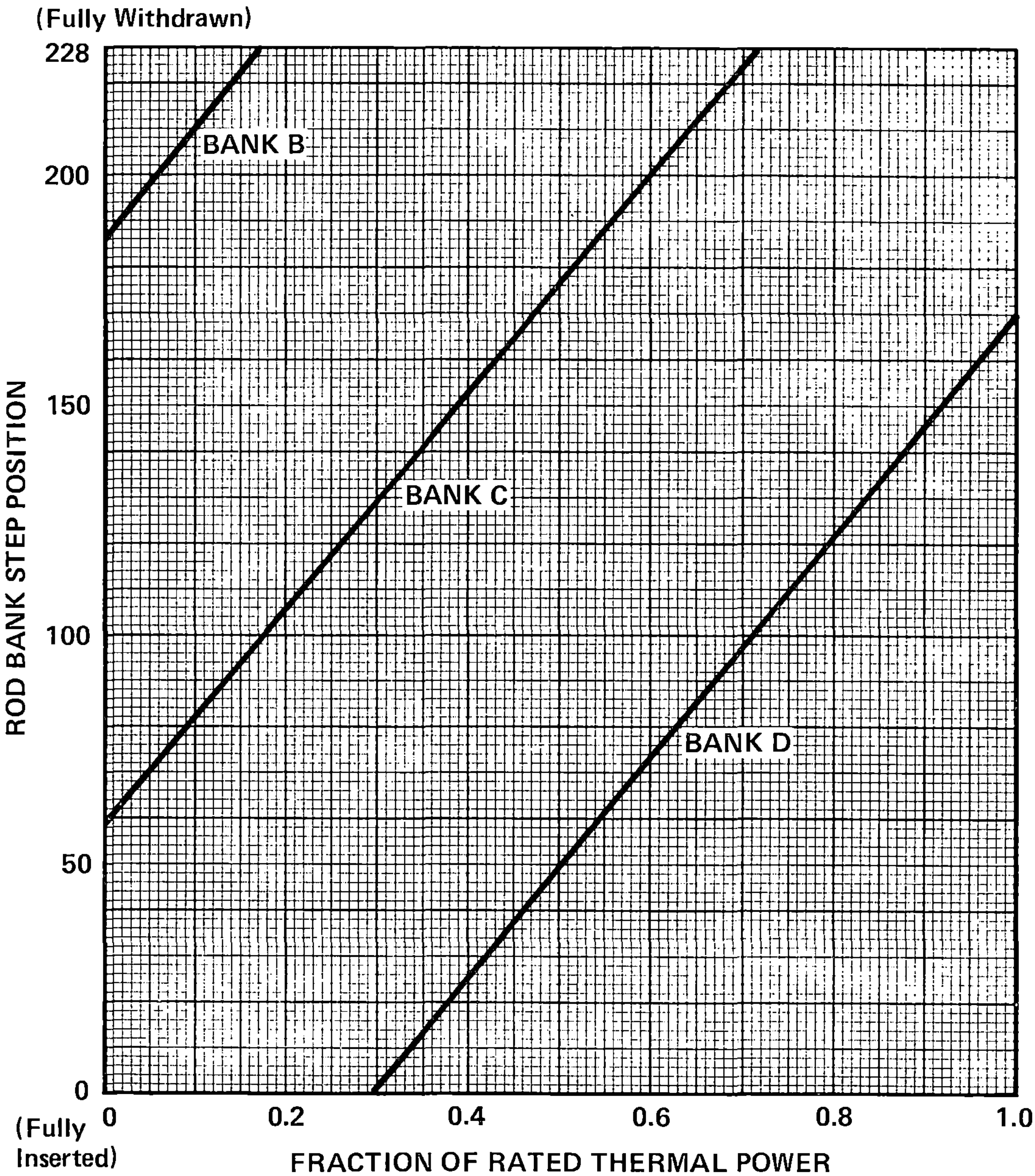


Figure 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

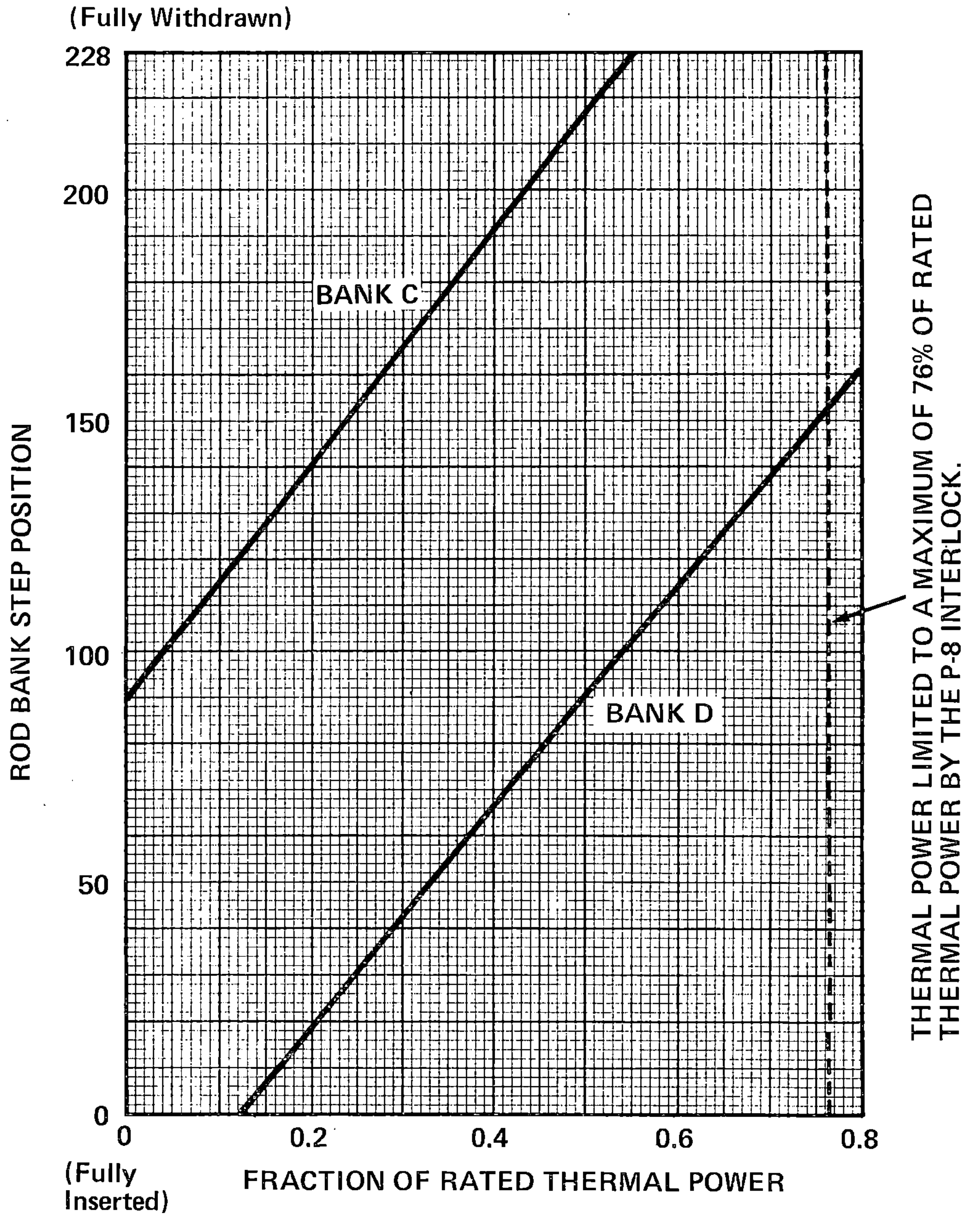


FIGURE 3.1-2 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER THREE LOOP OPERATION

PART LENGTH ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 All part length rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*

ACTION:

With a maximum of one part length rod not fully withdrawn, within one hour either:

a. Fully withdraw the rod, or

b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each part length rod shall be determined to be fully withdrawn by:

a. Verifying the position of the part length rod prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER, and

b. Verifying, at least once per 31 days, that electric power has been disconnected from its drive mechanism by physical removal of a breaker from the circuit.

* See Special Test Exceptions 3.10.2 and 3.10.3

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE shall be maintained within a ±5% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ±5% target band about the target flux difference and with THERMAL POWER:

   1. Above 90% of RATED THERMAL POWER, within 15 minutes:

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

   2. Between 50% and 90% of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the ±5% target band for more than 1# hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2
#A 2-hour penalty deviation is permissible during tests performed as part of the Augmented Startup Test Program.

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 5% target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ± 5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days with all part length control rods fully withdrawn. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

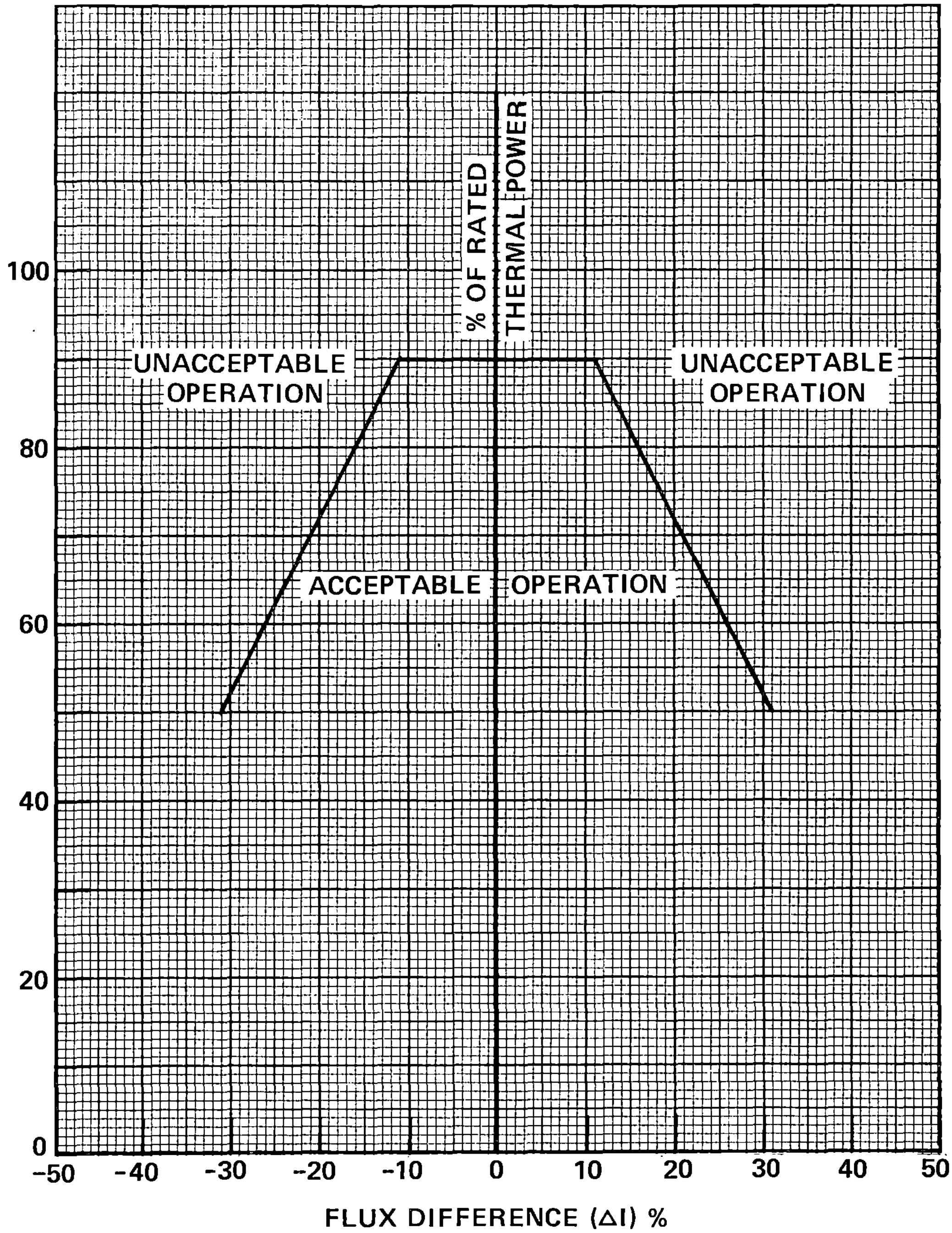


FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [(4.64)] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} \; [1+0.2(1-P)]$$

where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} :

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing either bank "D" control rods or any part length rods, and

2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 47.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

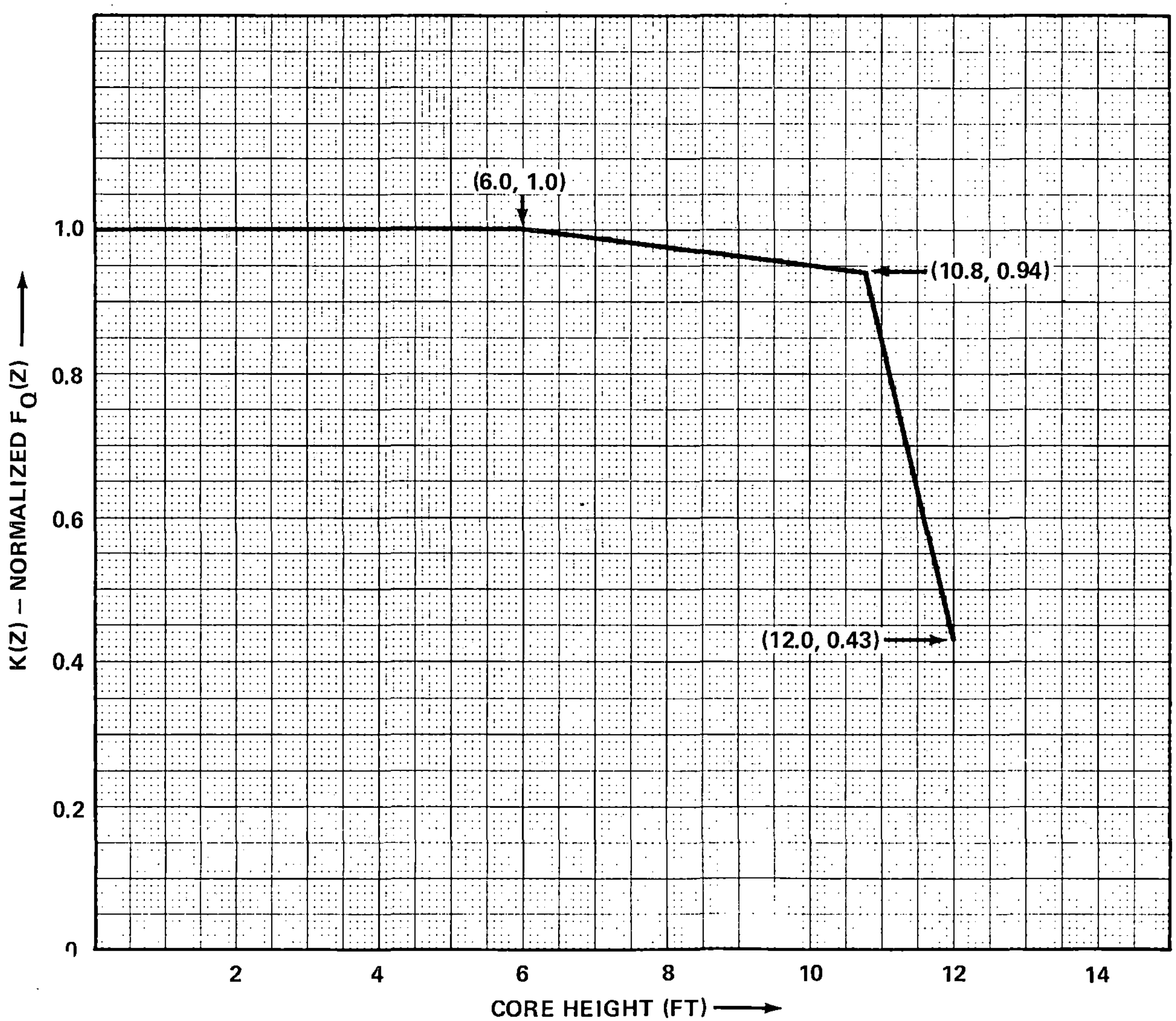


FIGURE 3.2-2

K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\ [1.0 + 0.2]\ (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

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

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER 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 QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown, control or part length rod:

   1. Reduce THERMAL POWER at least 3% for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER 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 QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown, control or part length rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. 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 QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter 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

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation	LIMITS 3 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 581°F	$\leq$ 572°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System	$\geq$ 349,200 gpm	$\geq$ 278,100 gpm

*Limit not applicable during either a THERMAL POWER ramp incease in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks 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

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

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

4.3.1.1.3 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 logic train such that both logic trains are tested at least once per 36 months and 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.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	2#
Three Loop Operation	4	1**	3	1, 2	9
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	2#
Three Loop Operation	4	1**	3	1, 2	9
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	1/2 twice	4	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	1/2 twice	4	1	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Turbine Trip					
A. Low Autostop Oil Pressure	3	2	2	1	7#
B Turbine Stop Valve Closure	4	4	3	1	6#
19. Safety Injection Input from SSPS	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10
B. Above P-7	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1, 2*	1
22. Automatic Trip Logic	2	1	2	1, 2*	1

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

c. Either, THERMAL POWER is restricted to $\leq$ 75% of RATED THERMAL and the Power Range, Neutron Flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

TABLE 3.3-1 (Continued)

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above P-6 but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above P-6, operation may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-1 (Continued)

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - 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 HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels < 6×10^{-11} amps.	P-6 prevents or defeats the manual block of source range reactor trip.
P-7	With 2 of 4 Power Range Neutron Flux Channels $\geq$ 11% of RATED THERMAL POWER or 1 of 2 Turbine impulse chamber pressure channels $\geq$ a pressure equivalent to 11% of RATED THERMAL POWER.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level.

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-8	With 2 of 4 Power Range Neutron Flux channels $\geq$ 36% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip on low coolant flow in a single loop.
P-10	With 3 of 4 Power range neutron flux channels < 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 6.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

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

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14. Steam Generator Water Level--Low-Low	$\leq$ 2.0 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M	1, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	N.A.	R(6)	S/U(1)	2, 3, 4 and 5
7. Overtemperature ΔT	S	R	M	1, 2
8. Overpower ΔT	S	R	M	1, 2
9. Pressurizer Pressure--Low	S	R	M	1, 2
10. Pressurizer Pressure--High	S	R	M	1, 2
11. Pressurizer Water Level--High	S	R	M	1, 2
12. Loss of Flow - Single Loop	S	R	M	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow - Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from SSPS	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	M(5) and S/U(1)	1, 2*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2*

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual SSPS functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks 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 an 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

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

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 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 logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per 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 FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14*
d. Pressurizer Pressure - Low with Pressurizer Level-Low	3 pressure and 3 level	1 pressure coincident with 1 level	2 pressure and 2 level	1, 2, 3#	14*
e. Differential Pressure Between Steam Lines - High				1, 2, 3##	
Four Loops Operating	3/steam line	2/steam line any steam line	2/steam line		14*
Three Loops Operating	3/operating steam line	1###/steam line, any operating steam line	2/operating steam line		15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Flow in Two Steam Lines-High				1, 2, 3##	
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line		15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low				1, 2, 3##	
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops		14*
Three Loops Operating	1 T_{avg}/ operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops		14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		15
2. CONTAINMENT SPRAY					
a. Manual	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	18
2) From Safety Injection Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Phase "B" Isolation					
1) Manual	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	18
2) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
3) Containment Pressure--High-High	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Manual	2	1	2	1, 2, 3, 4	17
2) Containment Atmosphere Radioactivity-High	4	1	2	1, 2, 3, 4	17
4. STEAM LINE ISOLATION					
a. Manual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
d. Steam Flow in Two Steam Lines--High					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line		15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low				1, 2, 3##	
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops		14*
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		15
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. TO CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops		14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water level-- High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14*
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC)	3	2	3	1, 2, 3, 4	13
7. UNDERVOLTAGE, VITAL BUS	3	2	3	1, 2, 3	14*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST, provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-3 (Continued)

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - 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.

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels $\geq$ 1925 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure coincident with low pressurizer water level.
P-12	With 3 of 4 T_{avg} channels $\geq$ 545°F.	P-12 prevents or defeats manual block of safety injection actuation high steam line flow and low steam line pressure.
	With 2 of 4 T_{avg} channels < 541°F.	Allows manual block of safety injection actuation on high steam line flow and low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	≤ 4.7 psig	≤ 5.2 psig
d. Pressurizer Pressure--Low Coincident with Pressurizer Water Level--Low	≥ 1765 psig ≥ 5%	≥ 1755 psig ≥ 4.0%
e. Differential Pressure Between Steam Lines--High	≤ 100 psi	≤ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low or Steam Line Pressure--Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load $T_{avg} \geq 543°F$ ≥ 500 psig steam line pressure	≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load $T_{avg} \geq 541°F$ ≥ 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 23.5 psig	$\leq$ 24 psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation Logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High	$\leq$ 23.5 psig	$\leq$ 24 psig
c. Containment Ventilation Isolation		
1. Manual	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Atmosphere Radioactivity		
a) Particulate	≤ 2 x background	≤ 2 x background
b) Gaseous	≤ 2 x background	≤ 2 x background
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	≤ 23.5 psig	≤ 24 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low or Steam Line Pressure--Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load	≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load
	T_{avg} ≥ 543°F	T_{avg} ≥ 541°F
	≥ 500 psig steam line pressure	≥ 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	$\leq$ 67% of narrow range instrument span each steam generator	$\leq$ 68% of narrow range instrument span each steam generator
6. UNDERVOLTAGE, VITAL BUS	$\geq$ 70% of bus voltage	$\geq$ 65% of bus voltage

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Fan Cooler	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Ventilation Isolation	Not applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0*
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 13.0#/48.0##

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low with Pressurizer Level-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/13.0#
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 48.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq$ 15.0#/25.0##
b. Reactor Trip (from SI)	$\leq$ 5.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq$ 20.0#/30.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 15.0#/50.0##
h. Steam Line Isolation	$\leq$ 10.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 14.0#/48.0##
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Fan Cooler	$\leq$ 40.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\# Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	R	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	1, 2. 3
d. Pressurizer Pressure--Low Coincident with Pressurizer Water Level--Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	R	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	R	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	R	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M(3)	1, 2, 3
c. Containment Ventilation Isolation				
1) Manual	N.A.	N.A.	R	1, 2, 3, 4
2) Containment Radio-activity-High	S	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}-- Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) LOGIC				
a. Inputs	N.A.	N.A.	M	1, 2, 3, 4
b. Logic, Timing and Outputs	N.A.	N.A.	M(1)	1, 2, 3, 4
7. UNDERVOLTAGE, VITAL BUS	S	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each logic channel shall be tested at least once per 62 days on a STAGGERED TEST BASIS. The CHANNEL FUNCTION TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within $\pm$ 10% of its design interval.

(2) Each train or logic channel shall be tested at least every other 31 days.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area	1	*	$\leq$ 15 mR/hr	10^{-1} - 10^{4} mR/hr	19
2. PROCESS MONITORS					
a. Containment					
1) Gaseous Activity	1#				
a) Purge & Pressure-Vacuum Relief Isolation		1, 2, 3, 4 & 6	$\leq$ 2 x background	10^{1} - 10^{6} cpm	22
b) RCS Leakage Detection		1, 2, 3 & 4	N/A	10^{1} - 10^{6} cpm	20
2) Air Particulate Activity	1#				
a) Purge & Pressure-Vacuum Relief Isolation		1, 2, 3, 4 & 6	$\leq$ 2 x background	10^{1} - 10^{6} cpm	22
b) RCS Leakage Detection		1, 2, 3 & 4	N/A	10^{1} - 10^{6} cpm	20
3) Fixed Filter Iodine-Purge & Pressure - Vacuum Relief Isolation	1	1, 2, 3, 4 & 6	$\leq$ 2 x background	10^{1} - 10^{6} cpm	22

* With fuel in the storage pool or building.
# Channel may be removed from service and used for monitoring plant stack effluent rather than for monitoring containment atmosphere for up to 8 hours per 24 hour interval while either purging the containment atmosphere or venting a gas decay tank.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area	S	R	M	*
2. PROCESS MONITORS				
a. Containment				
1) Gaseous Activity				
a) Purge & Pressure - Vacuum Relief Isolation	S	R	M	1, 2, 3, 4 & 6
b) RCS Leakage Detection	S	R	M	1, 2, 3 & 4
2) Air Particulate Activity				
a) Purge & Pressure - Vacuum Relief Isolation	S	R	M	1, 2, 3, 4 & 6
b) RCS Leakage Detection	S	R	M	1, 2, 3 & 4
3) Fixed Filter Iodine - Purge & Pressure - Vacuum Relief Isolation	S	R	M	1, 2, 3, 4 & 6

*With fuel in the storage pool or building.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

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

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$

ACTION:

With the movable 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 movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output to be used during its use when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelographs		
a. Reactor Containment, Elev. 81'	0 - 1g	1
b. Reactor Containment, Elev. 130'	0 - 1g	1
c. Auxiliary Building, Elev. 122'	0 - 1g	1
2. Triaxial Peak Accelographs		
a. Reactor Containment, Elev. 86'5"	0 - 1g	1
b. Reactor Containment, Elev. 136'3"	0 - 1g	1
c. Auxiliary Building, Elev. 122'5"	0 - 1g	1
d. Fuel Handling Building, Elev. 130'	0 - 1g	1

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. Reactor Containment, Elev. 81'	M*	R	SA
b. Reactor Containment, Elev. 130'	M*	R	SA
c. Auxiliary Building, Elev. 122'	M*	R	SA
2. Triaxial Peak Accelographs			
a. Reactor Containment, Elev. 86'5"	NA	R	NA
b. Reactor Containment, Elev. 136" 3"	NA	R	NA
c. Auxiliary Building, Elev. 122'5"	NA	R	NA
d. Fuel Handling Building, Elev. 130'	NA	R	NA

*Except seismic trigger

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a. Meteorological Tower	Nominal Elev. 33'	
b. Meteorological Tower	Nominal Elev. 150'	any 2 of 3
c. Meteorological Tower	Nominal Elev. 300'	
2. WIND DIRECTION		
a. Meteorological Tower	Nominal Elev. 33'	
b. Meteorological Tower	Nominal Elev. 150'	any 2 of 3
c. Meteorological Tower	Nominal Elev. 300'	
3. AIR TEMPERATURE - DELTA T		
a. Meteorological Tower	Nominal Elev. 150' - 33'	1 of 2
b. Meteorological Tower	Nominal Elev. 300' - 33'	

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 33'	D	SA
b. Nominal Elev. 150'	D	SA
c. Nominal Elev. 300'	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 33'	D	SA
b. Nominal Elev. 150'	D	SA
c. Nominal Elev. 300'	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 150' - 33'	D	SA
b. Nominal Elev. 300' - 33'	D	SA

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 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 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 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 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Pressurizer Pressure	Hot Shutdown Panel 213	1700-2500 psig	1
2. Pressurizer Level	Hot Shutdown Panel 213	0 - 100%	1
3. Steam Generator Pressure	Hot Shutdown Panel 213	0 - 2500 psig	1/steam generator
4. Steam Generator Level	Hot Shutdown Panel 213	0 - 100%	1/steam generator

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Pressurizer Pressure	M	R
2. Pressurizer Level	M	R
3. Steam Generator Pressure	M	R
4. Steam Generator Level	M	R

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS

NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

Above P-7, comply with either of the following ACTIONS:

a. With one reactor coolant loop and associated pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to less than 36% of RATED THERMAL POWER and the following ESF instrumentation channels associated with the loop not in operation, are placed in their tripped condition within 1 hour:

   1. T_{avg} -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

   2. Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

   3. Steam Flow-High Channel used for Safety Injection.

   4. Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

b. With one reactor coolant loop and associated pump not in operation, subsequent STARTUP and POWER OPERATION above 36% of RATED THERMAL POWER may proceed provided:

   1. The following actions have been completed with the reactor in at least HOT STANDBY:

      a) Reduce the overtemperature ΔT trip setpoint to the value specified in Specification 2.2.1 for 3 loop operation.

*See Special Test Exception 3.10.4.

ACTION (Continued)

b) Place the following reactor trip system and ESF instrumentation channels, associated with the loop not in operation, in their tripped conditions:

1) Overpower ΔT channel.

2) Overtemperature ΔT channel.

3) T_{avg} -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

4) Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

5) Steam Flow-High channel used for Safety Injection.

6) Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

c) Change the P-8 interlock setpoint from the value specified in Table 3.3-1 to $\leq$ 76% of RATED THERMAL POWER.

2. THERMAL POWER is restricted to $\leq$ 71% of RATED THERMAL POWER.

ACTION (Continued)

Below P-7:

a. With $K_{eff} \geq 1.0$, operation may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. With $K_{eff} < 1.0$, operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant or residual heat removal pump.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.1 With one reactor coolant loop and associated pump not in operation, at least once per 31 days determine that:

a. The applicable reactor trip system and/or ESF actuation system instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions, and

b. If the P-8 interlock setpoint has been reset for 3 loop operation, its setpoint is $\leq$ 76% of RATED THERMAL POWER.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG $\pm$ 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG $\pm$ 1%.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

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

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble.

APPLICABILITY: MODES 1 and 2

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

    1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

2. Tubes in those areas where experience has indicated potential problems.

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to the inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions.

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

SURVEILLANCE REQUIREMENTS (Continued)

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4–2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C–1	None	N/A	N/A	N/A	N/A
	C–2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C–1	None	N/A	N/A
			C–2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug defective tubes
					C–3	Perform action for C–3 result of first sample
			C–3	Perform action for C–3 result of first sample	N/A	N/A
	C–3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC pursuant to specification 6.9.1	All other S. G.s are C–1	None	N/A	N/A
			Some S. G.s C–2 but no additional S. G. are C–3	Perform action for C–2 result of second sample	N/A	N/A
			Additional S. G. is C–3	Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. Either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

b. Containment sump level and containment fan cooler condensate flow rate (if being used) monitoring systems-performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 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 all steam generators and 500 gallons per day through any one steam generator,

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

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 $\pm$ 20 psig.

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 limits 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

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

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

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

SURVEILLANCE REQUIREMENTS (Continued)

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2230 $\pm$ 20 psig and valve 1CV71 is fully closed,

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 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 $\leq$ 500 psig, 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 pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

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	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-3

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 $T_{avg} \leq 250°F$

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram.

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

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 μCi/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 operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/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 STANDBY with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4a 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:

*With $T_{avg} \geq$ 500°F.

ACTION: (Continued)

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

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 de-gassing operations, 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 μCi/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 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 4.4-4.

TABLE 4.4-4

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 $\overline{E}$ 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 $100/\overline{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 & 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.

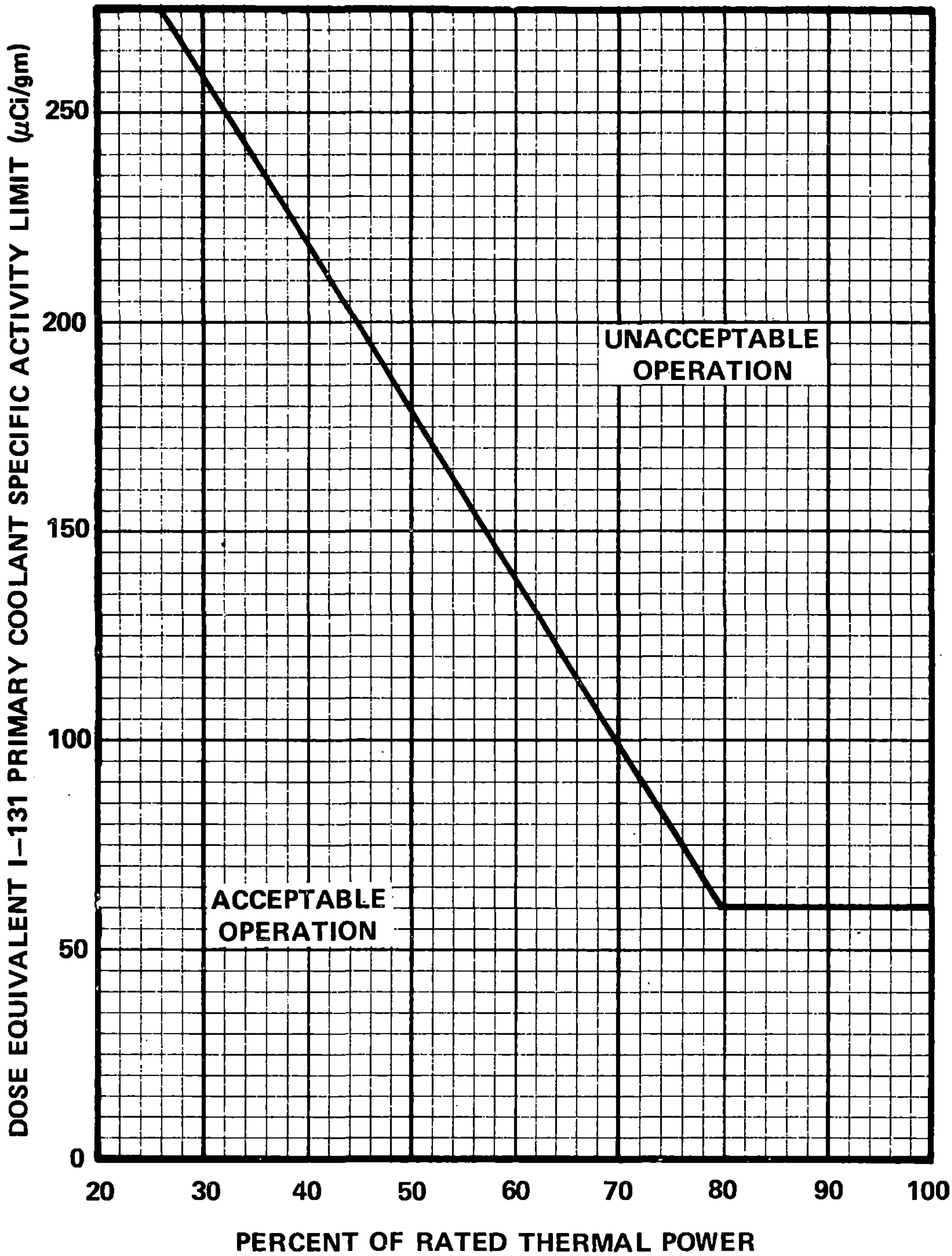


FIGURE 3.4-1

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period, and

c. A maximum temperature change of $\leq$ 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

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 operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

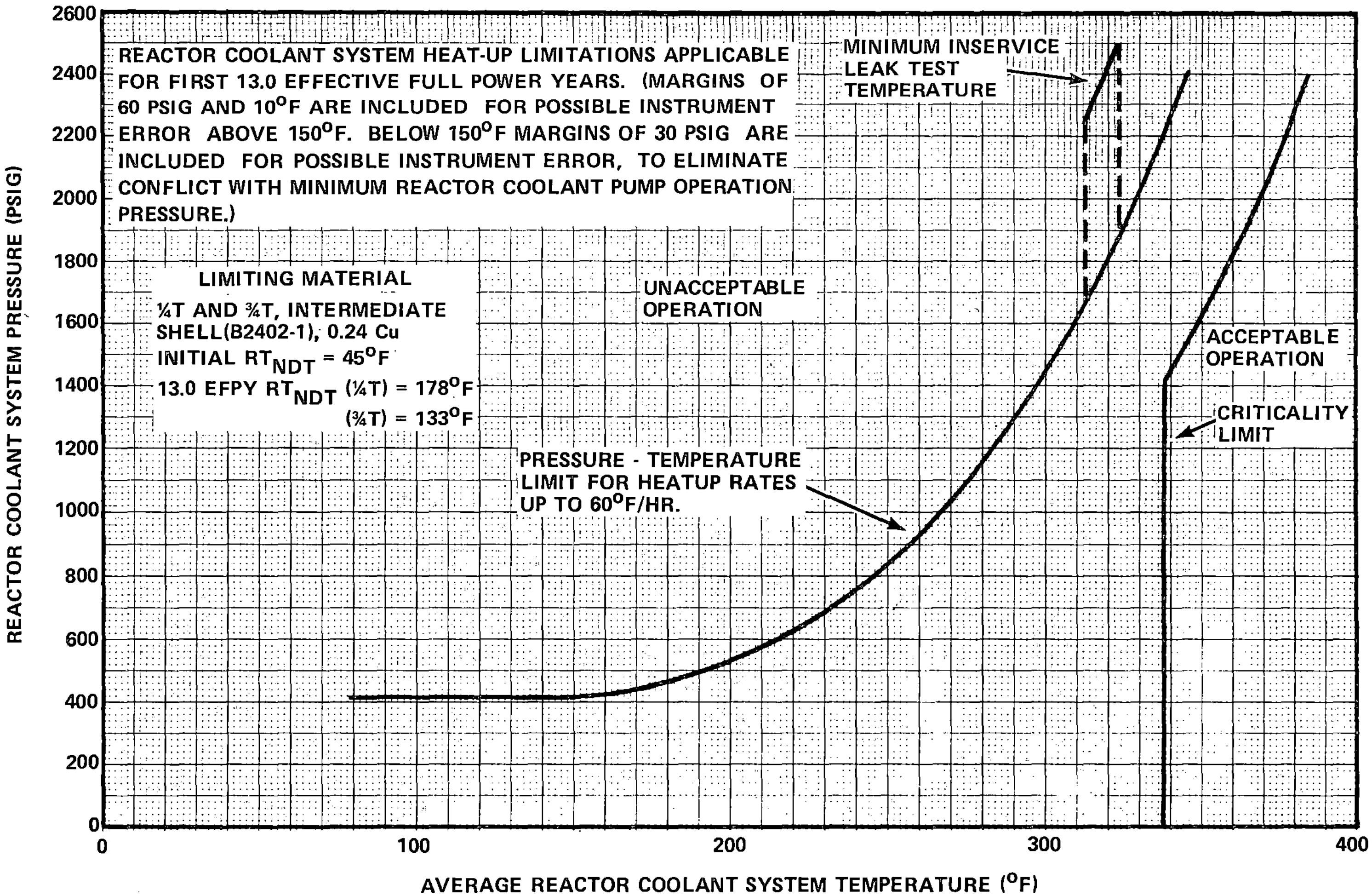


FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE – TEMPERATURE LIMITS VERSUS 60°F/HOUR RATE – CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

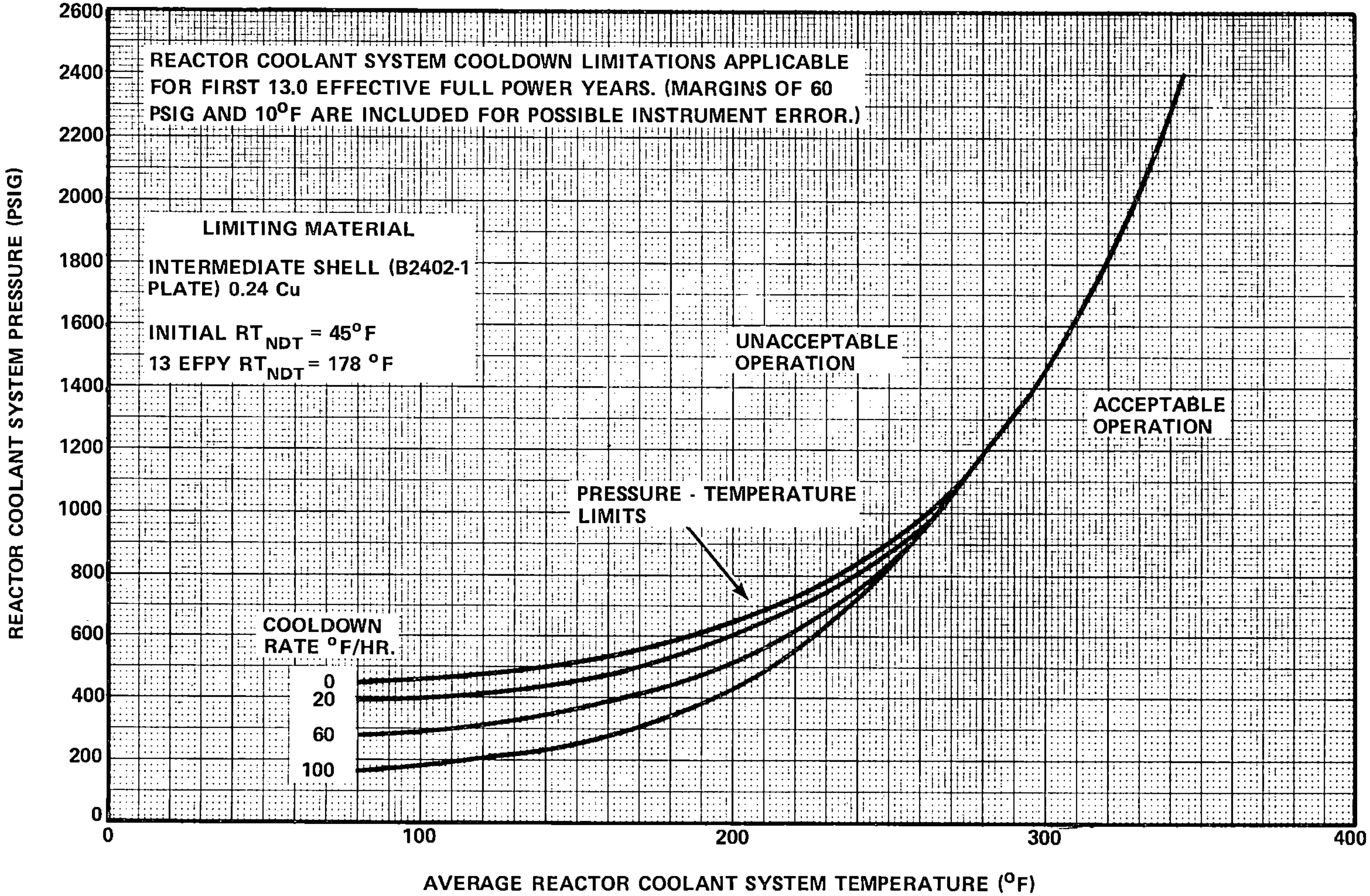


FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE — TEMPERATURE LIMITS VERSUS COOLDOWN RATES

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1. Capsule T	Replacement of 1st Region (Postirradiation test)
2. Capsule Y	5 years (Postirradiation test)
3. Capsule Z	10 years (Postirradiation test)
4. Capsule V	10 years (Reinsert in Capsule T location)
5. Capsule X	10 years (Reinsert in Capsule Y location)
6. Capsule U	10 years (Reinsert in Capsule Z location)
7. Capsule S	15 years (Postirradiation test)
8. Capsule W	20 years (Reinsert in Capsule S location)

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup or cooldown of 200°F in any one hour period, and

b. A maximum spray water temperature differential of 320°F.

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

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

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 200°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(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

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, Revision 1, August 1975.

4.4.10.1.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The steam generator channel heads shall be ultrasonic inspected during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material. The stainless steel clad surfaces of the steam generator channel heads shall also be 100% visually inspected during the above outages and a television camera shall be used to make a videotape recording of the condition of this cladding. Each videotape shall be compared with those obtained during the previous outages to determine that the cladding does not show any abnormal degradation.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained volume of between 6380 and 6657 gallons of borated water,

c. A minimum boron concentration of 1900 PPM, and

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 12 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of $\geq$ 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically upon receipt of a safety injection test signal.

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

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 HOT SHUTDOWN within the next 12 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.

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
a. 1 SJ 69	a. RHR pump suction	a. open
b. 1 SJ 30	b. SI pump suction	b. open
c. 11 SJ 40	c. SI discharge to hot legs	c. closed
d. 12 SJ 40	d. SI discharge to hot legs	d. closed
e. 1 RH 26	e. RHR discharge to hot legs	e. closed
f. 11 SJ 49	f. RHR discharge to cold legs	f. open
g. 12 SJ 49	g. RHR discharge to cold legs	g. open
h. 1 CS 14	h. Spray additive tank discharge	h. open
i. 1 SJ 135	i. SI discharge to cold legs	i. open
j. 1 SJ 67	j. SI recirc. line isolation	j. open
k. 1 SJ 68	k. SI recirc. line isolation	k. open
l. 11 SJ 44	l. Containment sump isolation valve	l. closed
m. 12 SJ 44	m. Containment sump isolation valve	m. closed

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

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. This visual inspection shall be performed:

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

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

d. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 580 psig.

   2. 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 safety injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. 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.

SURVEILLANCE REQUIREMENTS

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

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum contained volume of 900 gallons of borated water,

b. Between 20,100 and 21,800 ppm of boron, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the water level through a recirculation flow test at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be $\geq$ 145°F at least once per 8 hours; otherwise, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be $\geq$ 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water.

b. A minimum boron concentration of 2000 ppm, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage 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

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is < 35°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY 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

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

   1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1., and

   2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

*Except vents, drains, test connections, etc. which are (1) one inch nominal pipe diameter or less, (2) located inside the containmnet, and (3) locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed at least once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.10 percent by weight of the containment air per 24 hours at design pressure, (47.0 psig).

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests as identified in Table 3.6-1, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at design pressure (47.0 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet .75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet .75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a.

2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at design pressure (47.0 psig).

d. Type B and C tests shall be conducted with gas at design pressure (47.0 psig) at intervals no greater than 24 months except for tests involving air locks.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

SURVEILLANCE REQUIREMENTS (Continued)

f. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at design pressure (47.0 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. *After each opening, except when the airlock is being used for multiple entries, then at least once per 72 hours by verifying the door seal leakage rate is $\leq 0.01\ L_a$ when the volume between the door seals is pressurized to and maintained at greater than or equal to design pressure (47.0 psig),

b. At least once per 6 months by conducting an overall air lock leakage test at design pressure (47.0 psig) and by verifying that the overall air lock leakage rate is within its limit, and

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*Exemption to Appendix "J" of 10 CFR 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.5 and +0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any 5 of the 10 following locations and shall be determined at least once per 24 hours:

Location (Containment Perimeter)

a. Elev. 84' - North
b. Elev. 106' - North
c. Elev. 78' - Northeast
d. Elev. 84' - East
e. Elev. 106' - East
f. Elev. 136' - South
g. Elev. 84' - South
h. Elev. 106' - Southwest
i. Elev. 121' - West
j. Elev. 136' - West

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Concrete Surfaces The structural integrity of the concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the concrete exterior surfaces or the concrete crack patterns. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.6.2 Liner Plate The structural integrity of the containment liner plate shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the plate and verifying no apparent changes in appearance or other abnormal degradation.

4.6.1.6.3 Reports An initial report of any abnormal degradation of the containment structure detected during the above required tests and inspections shall be made within 10 days after completion of the surveillance requirements of this specification and the detailed report shall be submitted pursuant to Specification 6.9.1 within 90 days after completion. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

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 RWST and transferring suction to the RHR pump discharge.

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 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 18 months during shutdown, by:

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

   2. Verifying that each spray pump starts automatically on a Containment High-High pressure test signal.

c. At least once per 5 years by:

   1. Performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

   2. Verifying a spray additive tank eductor flow rate of 35 $\pm$ 3.5 gpm to each containment spray system with the spray pump operating in the recirculation mode.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of at least 2000 gallons of not less than 30 percent by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the spray additive 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 spray additive 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 spray additive system shall be demonstrated OPERABLE:

a. At least once per 31 days also 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 solution level in the tank, and

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

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment High-High pressure test signal.

d. At least once per 5 years by verifying a NaOH solution flow rate of 7.3 ± 0.7 gpm from the spray additive tank through sample valve 1CS61 with the spray additive tank at 2.5 ± 0.5 psig.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Three independent groups of containment cooling fans shall be OPERABLE with two fan systems to each of two groups and one fan system to the third group.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one group of containment cooling fans inoperable, restore the inoperable fan group to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Starting (unless already operating) each fan from the control room.

   2. Verifying that each fan operates for at least 15 minutes.

   3. Verifying a cooling water flow rate of $\geq$ 700 gpm to each cooler.

b. At least once per 18 months by verifying that on a safety injection test signal:

   1. Each fan starts automatically on low speed.

   2. The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is $\geq$ 2500 gpm.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable 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 each 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

4.6.3.1.1 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 by performance of a cycling test and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 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:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a feedwater isolation test signal, each feedwater isolation valve isolates to its isolation position.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each Purge and Pressure-Vacuum Relief valve actuates to its isolation position.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
A. PHASE "A" ISOLATION		
1. 1 PR 17*	Pressurizer Relief Tk.-Gas Analyzer Conn.	$\leq$ 10 Sec.
2. 1 PR 18*	Pressurizer Relief Tk.-Gas Analyzer Conn.	$\leq$ 10 Sec.
3. 1 NT 25*	Pressurizer Relief Tk.-N_2 Connection	$\leq$ 10 Sec.
4. 1 WR 80*	Pressurizer Relief Tk.-Primary Water Conn.	$\leq$ 10 Sec.
5. 1 CV 3	CVCS - Letdown Line	$\leq$ 10 Sec.
6. 1 CV 4	CVCS - Letdown Line	$\leq$ 10 Sec.
7. 1 CV 5	CVCS - Letdown Line	$\leq$ 10 Sec.
8. 1 CV 7	CVCS - Letdown Line	$\leq$ 10 Sec.
9. 1 CV 68	CVCS - Charging Line	$\leq$ 10 Sec.
10. 1 CV 69	CVCS - Charging Line	$\leq$ 10 Sec.
11. 1 CV 284	CVCS - RCP Seals	$\leq$ 10 Sec.
12. 1 CV 116	CVCS - RCP Seals	$\leq$ 10 Sec.
13. 1 CC 215	Comp. Cooling to Excess Letdown Hx	$\leq$ 10 Sec.
14. 1 CC 113	Comp. Cooling to Excess Letdown Hx	$\leq$ 10 Sec.
15. 1 WL 96*	RC Drain Tk - Gas Analyzer Conn.	$\leq$ 10 Sec.
16. 1 WL 97*	RC Drain Tk - Gas Analyzer Conn.	$\leq$ 10 Sec.
17. 1 WL 98*	RC Drain Tk - Vent Header Conn.	$\leq$ 10 Sec.
18. 1 WL 99*	RC Drain Tk - Vent Header Conn.	$\leq$ 10 Sec.
19. 1 WL 108*	RC Drain Tk - N_2 Connection	$\leq$ 10 Sec.
20. 1 WL 12*	RC Drain Tank Pumps	$\leq$ 10 Sec.
21. 1 WL 13*	RC Drain Tank Pumps	$\leq$ 10 Sec.
22. 1 NT 32*	Accumulator N_2 Supply	$\leq$ 10 Sec.
23. 1 SJ 123*	SI Test Line	$\leq$ 10 Sec.
24. 1 SJ 60*	SI Test Line	$\leq$ 10 Sec.
25. 1 SJ 53*	SI Test Line	$\leq$ 10 Sec.
26. 1 SS 103*	Accumulator Sampling	$\leq$ 10 Sec.
27. 1 SS 27*	Accumulator Sampling	$\leq$ 10 Sec.
28. 1 SS 104*	RC Sampling	$\leq$ 10 Sec.
29. 1 SS 33*	RC Sampling	$\leq$ 10 Sec.
30. 11 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

A. PHASE "A" ISOLATION (Continued)

VALVE NUMBER	FUNCTION	ISOLATION TIME
31. 12 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.
32. 13 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.
33. 14 SS 94*	SG Blowdown Sampling	$\leq$ 10 Sec.
34. 1 SS 107*	Pressurizer Liquid Sampling	$\leq$ 10 Sec.
35. 1 SS 49*	Pressurizer Liquid Sampling	$\leq$ 10 Sec.
36. 1 SS 110*	Pressurizer Steam Sampling	$\leq$ 10 Sec.
37. 1 SS 64*	Pressurizer Steam Sampling	$\leq$ 10 Sec.
38. 1 VC 7	Containment Radiation Sampling	$\leq$ 10 Sec.
39. 1 VC 8	Containment Radiation Sampling	$\leq$ 10 Sec.
40. 1 VC 11	Containment Radiation Sampling	$\leq$ 10 Sec.
41. 1 VC 12	Containment Radiation Sampling	$\leq$ 10 Sec.
42. 11 CA 330	Instrument Air Supply	$\leq$ 10 Sec.
43. 12 CA 330	Instrument Air Supply	$\leq$ 10 Sec.
44. 1 DR 29	Demineralized Water Supply	$\leq$ 10 Sec.
45. 11 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
46. 12 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
47. 13 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
48. 14 GB 4	Steam Generator Blowdown	$\leq$ 10 sec.
49. 1 WL 16	Containment Sump Discharge	$\leq$ 10 sec.
50. 1 WL 17	Containment Sump Discharge	$\leq$ 10 sec.
51. 1 FP 147*	Fire Protection System	$\leq$ 10 sec.

B. PHASE "B" ISOLATION

VALVE NUMBER	FUNCTION	ISOLATION TIME
1. 1 CC 118	Component Cooling to RCP	$\leq$ 10 sec.
2. 1 CC 117	Component Cooling to RCP	$\leq$ 10 sec.
3. 1 CC 187	Component Cooling to RCP	$\leq$ 10 sec.
4. 1 CC 136	Component Cooling to RCP	$\leq$ 10 sec.
5. 1 CC 190	Component Cooling to RCP	$\leq$ 10 sec.
6. 1 CC 131	Component Cooling to RCP	$\leq$ 10 sec.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
B. PHASE "B" ISOLATION (Continued)		
7. 11 MS 7	Main Steam Drain	≤ 10 Sec.
8. 12 MS 7	Main Steam Drain	≤ 10 Sec.
9. 13 MS 7	Main Steam Drain	≤ 10 Sec.
10. 14 MS 7	Main Steam Drain	≤ 10 Sec.
11. 11 MS 18	Main Steam Bypass	≤ 10 Sec.
12. 12 MS 18	Main Steam Bypass	≤ 10 Sec.
13. 13 MS 18	Main Steam Bypass	≤ 10 Sec.
14. 14 MS 18	Main Steam Bypass	≤ 10 Sec.
C. FEEDWATER ISOLATION		
1. 11 BF 19	Main Feedwater Isolation	≤ 8 Sec.
2. 12 BF 19	Main Feedwater Isolation	≤ 8 Sec.
3. 13 BF 19	Main Feedwater Isolation	≤ 8 Sec.
4. 14 BF 19	Main Feedwater Isolation	≤ 8 Sec.
5. 11 BF 40	Main Feedwater Isolation	≤ 8 sec.
6. 12 BF 40	Main Feedwater Isolation	≤ 8 Sec.
7. 13 BF 40	Main Feedwater Isolation	≤ 8 Sec.
8. 14 BF 40	Main Feedwater Isolation	≤ 8 Sec.
D. CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF		
1. 1 VC 1*	Purge Supply	≤ 2 Sec.
2. 1 VC 2*#	Purge Supply	≤ 2 Sec.
3. 1 VC 3*#	Purge Exhaust	≤ 2 Sec.
4. 1 VC 4*	Purge Exhaust	≤ 2 Sec.
5. 1 VC 5*	Pressure-Vacuum Relief	≤ 2 Sec.
6. 1 VC 6*#	Pressure-Vacuum Relief	≤ 2 Sec.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

E. MANUAL

	VALVE NUMBER	FUNCTION	ISOLATION TIME
1.	(2 valves)#	Pressurizer Dead-Weight Calibrator	Not Applicable
2.	11 CV 98#	CVCS - RCP Seals	Not Applicable
3.	12 CV 98#	CVCS - RCP Seals	Not Applicable
4.	13 CV 98#	CVCS - RCP Seals	Not Applicable
5.	14 CV 98#	CVCS - RCP Seals	Not Applicable
6.	1 SJ 71#	CVCS Flushing Connection	Not Applicable
7.	11 SS 93*#	Steam Generator Sampling	Not Applicable
8.	12 SS 93*#	Steam Generator Sampling	Not Applicable
9.	13 SS 93*#	Steam Generator Sampling	Not Applicable
10.	14 SS 93*#	Steam Generator Sampling	Not Applicable
11.	1 SA 118#	Compressed Air Supply	Not Applicable
12.	1 WL 190#	Refueling Canal Supply	Not Applicable
13.	1 SF 36#	Refueling Canal Supply	Not Applicable
14.	1 WL 191#	Refueling Canal Discharge	Not Applicable
15.	1 SF 22#	Refueling Canal Discharge	Not Applicable
16.	1 VC 9*#	Containment Radiation Sampling	Not Applicable
17.	1 VC 10*#	Containment Radiation Sampling	Not Applicable
18.	1 VC 13*#	Containment Radiation Sampling	Not Applicable
19.	1 VC 14*#	Containment Radiation Sampling	Not Applicable
20.	- #	Fuel Transfer Tube	Not Applicable

*May be opened on an intermittent basis under administrative control.

#Not subject to Type C leakage tests.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. Zero volume percent hydrogen, balance purging air without any free hydrogen, and

b. Two volume percent hydrogen, balance air.

ELECTRIC HYDROGEN RECOMBINERS - W

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to $\geq$ 700°F within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.)

   3. Verifying during a recombiner system functional test that the heater sheath temperature increases to $\geq$ 1200°F within 5 hours and is maintained for at least 4 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

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 associated with each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux 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.

c. The provisions 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.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	64
3	42

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	60
2	44
3	29

* At least two safety valves shall be OPERABLE on the non-operating steam generator.

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER				LIFT SETTING (± 1%)	ORIFICE SIZE (sq. inches)
	Loop A	Loop B	Loop C	Loop D		
a.	11MS11	12MS11	13MS11	14MS11	1125 psig	16.0
b.	11MS12	12MS12	13MS12	14MS12	1120 psig	16.0
c.	11MS13	12MS13	13MS13	14MS13	1110 psig	16.0
d.	11MS14	12MS14	13MS14	14MS14	1100 psig	16.0
e.	11MS15	12MS15	13MS15	14MS15	1070 psig	16.0

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate vital busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate vital busses and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the steam turbine driven pump develops a discharge pressure of $\geq$ 1500 psig on recirculation flow when the secondary steam supply pressure is greater than 750 psig.

2. 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.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the motor driven pump flow path actuates to its correct position on a pump discharge pressure test signal.

2. Verifying that each motor driven pump starts automatically upon receipt of each of the following test signals:

   a) Loss of main feedwater pumps.

   b) Safeguards sequence signal.

   c) Steam Generator Water Level -- Low-Low from one steam generator.

3. Verifying that the steam turbine driven pump starts automatically upon receipt of each of the following test signals:

   a) Loss of offsite power.

   b) Steam Generator Water Level -- Low-Low from two steam generators.

AUXILIARY FEED STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feed storage tank (AFST) shall be OPERABLE with a minimum contained volume of 200,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the auxiliary feed storage tank inoperable, within 4 hours either:

a. Restore the AFST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of a demineralized water or a fire protection/domestic water storage tank as a backup supply to the auxiliary feedwater pumps and restore the auxiliary feed storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

4.7.1.3.2 A demineralized water storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the tank contains $\geq$ 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

4.7.1.3.3 A fire protection/domestic water storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the tank contains $\geq$ 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$0.10 μCi/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/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 4.7-2.

TABLE 4.7-2

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:

MODES 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided;

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

SECONDARY WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The secondary water chemistry shall be maintained within the limits of Table 3.7-3.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(To be determined in a manner set forth in the bases in approximately six months and to be imposed by a change to this Specification.)

SURVEILLANCE REQUIREMENTS

4.7.1.6 The secondary water chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.7-3.

TABLE 3.7-3

SECONDARY WATER CHEMISTRY CONTROL PARAMETERS

Water Sample Location	Total Cation Conductivity μmhos/cm^2 @ 25°C Limits	pH @ 25°C Limits	Free Hydroxide ppm $CaCO_3$ Limits	Total Suspended Solids Limits
Condenser Condensate	*	N.A.	N.A.	N.A.
Steam Generator Blowdown	*	*	*	*

* Limits to be established in approximately 6 months based upon test program described in bases.

TABLE 4.7-3

SECONDARY WATER CHEMISTRY SURVEILLANCE REQUIREMENTS

Water Sample Location	Total Cation Conductivity $\mu mhos/cm^2$ @ 25°C Frequency	pH @ 25°C Frequency	Free Hydroxide ppm $CaCO_3$ Frequency	Total Suspended Solids Frequency
Condenser Condensate	*	N.A.	N.A.	N.A.
Steam Generator Blowdown	*	*	*	*

* Surveillance frequencies to be established in approximately 6 months based upon test program described in bases.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be $>$ 70°F when the pressure of either coolant in the steam generator is $>$ 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to $\leq$ 200 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 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be $<$ 200 psig at least once per hour when the temperature of either the primary or secondary coolant is $<$ 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two service water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on Safeguards Initiation signal.

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Delaware River exceeds 10.5' Mean Sea Level USGS datum, at the service water intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level at the service water intake structure above elevation 10.5' Mean Sea Level USGS datum, close all watertight doors within 2 hours.

b. With the water level at the service water intake structure above elevation 11.5' Mean Sea Level USGS datum, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The water level at the service water intake structure shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 10.5' Mean Sea Level USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 10.5' Mean Sea Level USGS datum.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The control room emergency air conditioning system shall be OPERABLE with:

a. Two fans,

b. One cooling coil,

c. One charcoal adsorber and HEPA filter train, and

d. At least two isolation dampers in each outside air intake duct.

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

ACTION:

a. With one fan inoperable, restore the inoperable fan to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the cooling coil inoperable, restore the cooling coil to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the filter train inoperable, restore the filter train to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With only one isolation damper in an outside air intake duct OPERABLE, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The control room emergency air conditioning system shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least one hour and maintains the control room air temperature $\leq$ 120°F with each fan operating for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 7410 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 7410 cfm $\pm$ 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 7410 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510 - 1975 (130°C, 95% R.H.); or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510 - 1975 (130°C, 95% R.H.) and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

   a) Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510 - 1975 while operating the ventilation system at a flow rate of 7410 cfm $\pm$ 10%, and

   b) Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510 - 1975 while operating the ventilation system at a flow rate of 7410 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber bank is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 7410 cfm $\pm$ 10%.

2. Verifying that on a safety injection test signal or control room area high radiation test signal, the system automatically actuates in the recirculation mode by closing off the outside air supply and diverting air flow through the HEPA filter and charcoal adsorber bank.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the filter system at a flow rate of 7410 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filter system at a flow rate of 7410 cfm $\pm$ 10%.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 Two Auxiliary Building exhaust air HEPA filter trains, associated with the one charcoal adsorber bank, and at least two exhaust fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the above required HEPA filter trains inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the charcoal adsorber bank inoperable, restore the charcoal adsorber bank to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With only one exhaust fan OPERABLE, restore at least two exhaust fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each Auxiliary Building exhaust air HEPA filter train and above required exhaust fan shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 21,400 cfm ± 10 % and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the ventilation system to the facility vent, including leakage through the ventilation system diverting valves, is ≤ 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.

3. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

5. Verifying a system flow rate of 21,400 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by either:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510 - 1975 (130°C, 95% R.H.); or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510 - 1975 (130°C, 95% R.H.) and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

   a) Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510 - 1975 while operating the ventilation system at a flow rate of 21,400 cfm $\pm$ 10%, and

   b) Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510 - 1975 while operating the ventilation system at a flow rate of 21,400 cfm $\pm$ 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 21,400 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 21,400 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 21,400 cfm $\pm$ 10%.

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.8.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources and fission detectors previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials.

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source or detector.

4.7.8.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4.7.9 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9.1 All hydraulic snubbers listed in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Hydraulic snubbers shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7-4 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7-4 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

b. Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the

SURVEILLANCE REQUIREMENTS (Continued)

snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

c. At least once per 18 months, during shutdown, a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-4, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lb. capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1	No. 11 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
2	No. 11 Steam Generator. Reactor Containment, Elev. 128' 6"	I	Yes	Yes
3	No. 11 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
4	No. 11 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
5	No. 12 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
6	No. 12 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
7	No. 12 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
8	No. 12 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes

TABLE 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
9	No. 13 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
10	No. 13 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
11	No. 13 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
12	No. 13 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
13	No. 14 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
14	No. 14 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
15	No. 14 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes
16	No. 14 Steam Generator, Reactor Containment, Elev. 128' 6"	I	Yes	Yes

TABLE 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1.	No. 11 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
2.	No. 11 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
3.	No. 12 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
4.	No. 12 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
5.	No. 13 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
6.	No. 13 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
7.	No. 14 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No
8.	No. 14 Main Steam Isolation Valve Penetration Area, Elev. 115' 5"	A	No	No

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4 provided that a revision to Table 3.7-4 is included with the next License Amendment request.

** Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months $\pm$ 25%
1	12 months $\pm$ 25%
2	6 months $\pm$ 25%
3 or 4	124 days $\pm$ 25%
5, 6, or 7	62 days $\pm$ 25%
≥ 8	31 days $\pm$ 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system (vital bus system), and

b. Three separate and independent diesel generators with:

   1. Separate day tanks containing a minimum volume of 130 gallons of fuel, and

   2. A common fuel storage system consisting of two storage tanks, each containing a minimum volume of 20,000 gallons of fuel, and two fuel transfer pumps.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours

* One inoperable fuel transfer pump is equivalent to one inoperable diesel generator.

ACTION (Continued)

and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two or more of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least two of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore three diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system (vital bus system) shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) vital bus supply from one 13/4 kv transformer to the other 13/4 kv transformer.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in its day tank.

   2. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in $\leq$ 10 seconds.

   3. Verifying the generator is synchronized, loaded to $\geq$ 1400 kw, and operates for $\geq$ 60 minutes.

   4. Verifying the diesel generator is aligned to provide standby power to the associated vital busses.

b. At least once per 18 months during shutdown by:

   1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

   2. Verifying the generator capability to reject a load of 785 $\pm$ 40 kw without tripping,

   3. Simulating a loss of offsite power in conjunction with a safety injection test signal, and:

      a) Verifying de-energization of the vital busses and load shedding from the vital busses.

      b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the vital busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

SURVEILLANCE REQUIREMENTS (Continued)

c) Verifying that all diesel generator trips, except engine overspeed, lube oil pressure low, generator breaker failure protection, and generator differential, are automatically bypassed upon loss of voltage on the emergency bus and/or safety injection actuation signal.

4. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 2665 kw.

5. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2750 kw.

4.8.1.1.3 The diesel fuel oil storage and transfer system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying the level in each of the above required 20,000 gallon fuel storage tanks.

2. Verifying that both fuel transfer pumps can be started and transfer fuel from the 20,000 gallon storage tanks to the day tanks.

b. At least once per 92 days by verifying that a sample of diesel fuel from each of the above required 20,000 gallon fuel storage tanks is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators with:

1. Separate day tanks containing a minimum volume of 130 gallons of fuel,

2. A common fuel storage system containing a minimum volume of 20,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.3.

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A. C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators:

4 kvolt Vital Bus # 1A

4 kvolt Vital Bus # 1B

4 kvolt Vital Bus # 1C

460 volt Vital Bus # 1A and associated control centers

460 volt Vital Bus # 1B and associated control centers

460 volt Vital Bus # 1C and associated control centers

230 volt Vital Bus # 1A and associated control centers

230 volt Vital Bus # 1B and associated control centers

230 volt Vital Bus # 1C and associated control centers

115 volt Vital Instrument Bus # 1A and Inverter

115 volt Vital Instrument Bus # 1B and Inverter

115 volt Vital Instrument Bus # 1C and Inverter

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two A.C. electrical bus trains shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator with each train consisting of:

1 - 4 kvolt Vital Bus

1 - 460 volt Vital Bus and associated control centers

1 - 230 volt Vital Bus and associated control centers

1 - 115 volt Instrument Bus

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated power availability.

125-VOLT D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE:

TRAIN 1A consisting of 125-volt D.C. bus No. 1A, 125-volt D.C. battery No. 1A and at least one full capacity charger.

TRAIN 1B consisting of 125-volt D.C. bus No. 1B, 125-volt D.C. battery No. 1B and at least one full capacity charger.

TRAIN 1C consisting of 125-volt D.C. bus No. 1C, 125-volt D.C. battery No. 1C and at least one full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 125-volt D.C. bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery and/or its charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

4.8.2.3.2 Each 125-volt battery and above required charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of the pilot cell is between the minimum and maximum level indication marks.

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is $\geq$ 1.200.

3. The pilot cell voltage is $\geq$ 2.08 volts

4. The overall battery voltage is $\geq$ 125 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is $\geq$ 2.17 volts under float charge and has not decreased more than 0.23 volts from the value observed during the original acceptance test.

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is $\geq$ 1.200 and has not decreased more than 0.02 from the value observed during the previous test.

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material.

3. The battery charger will supply at least 200 amperes at 125 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 8 hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

125-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D. C. electrical equipment and bus shall be energized and OPERABLE:

2 - 125-volt D.C. busses, and

2 - 125-volt batteries, each with at least one full capacity charger, associated with each of the above D.C. busses.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

28-VOLT D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The following D.C. bus trains shall be energized and OPERABLE:

TRAIN 1A consisting of 28-volt D.C. bus No. 1A, 28-volt D.C. battery No. 1A and at least one full capacity charger.

TRAIN 1B consisting of 28-volt D.C. bus No. 1B, 28-volt D.C. battery No. 1B, and at least one full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 28-volt D.C. bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 28-volt D.C. battery and/or its charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

4.8.2.5.2 Each 28-volt battery and above required charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of the pilot cell is between the minimum and maximum level indication marks.

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F, and full electrolyte level, is $\geq$ 1.200.

3. The pilot cell voltage is $\geq$ 2.08 volts.

4. The overall battery voltage is $\geq$ 27 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is $\geq$ 2.17 volts under float charge and has not decreased more than 0.23 volts from the value observed during the original acceptance test.

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is $\geq$ 1.200 and has not decreased more than 0.02 from the value observed during the previous test.

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The battery charger will supply at least 150 amperes at 28 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 8 hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

ELECTRICAL POWER SYSTEMS

28-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.6 As a minimum, the following D. C. electrical equipment and bus shall be energized and OPERABLE:

1 - 28-volt D.C. bus, and

1 - 28-volt battery and at least one full capacity charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.6.1 The above required 28-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

4.8.2.6.2 The above required 28-volt batteries and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.5.2.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of $\geq$ 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

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 $\geq$ 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 2000 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

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 control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

---

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be 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:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

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

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

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

ACTION:

With the reactor subscritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment isolation valves per the applicable portions of Specification 4.6.3.1.2.

REFEELING OPERATIONS

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

REFUELING OPERATIONS

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and

2. An overload cut off limit $\leq$ 2850 pounds.

b. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.

CRANE TRAVEL - FUEL HANDLING AREA

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2500 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The overload cutoff which prevents crane travel with loads in excess of 2500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal 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.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm at least once per 24 hours.

CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Pressure-Vacuum Relief isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the Containment Purge and Pressure-Vacuum Relief isolation system inoperable, close each of the Purge and Pressure-Vacuum Relief penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provision of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Pressure-Vacuum Relief isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Pressure-Vacuum Relief isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

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 movements of fuel assemblies or control rods.

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet 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 requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

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.

FUEL HANDLING AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Handling Area ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no Fuel Handling Area ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until the Fuel Handling Area ventilation system is restored to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

   1. Verifying that with the ventilation system operating at a flow rate of 19,490 cfm $\pm$ 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the ventilation system to the facility vent, including leakage through the ventilation system diverting valves, is $\leq$ 1% when the ventilation system is tested by admitting cold DOP at the storage pool ventilation system intake.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 19,490 cfm $\pm$ 10%.

3. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 19,490 cfm $\pm$ 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R. H.). The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to thickness of the bed.

5. Verifying a system flow rate of 19,490 cfm, $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510 - 1975 (130°C, 95% R.H.); or

2. Verifying within 31 days after removal that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510 - 1975 (130°C, 95% R.H.) and the samples are prepared by either:

SURVEILLANCE REQUIREMENTS (Continued)

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510 - 1975 while operating the ventilation system at a flow rate of 19,490 cfm $\pm$ 10%, and

b) Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510 - 1975 while operating the ventilation system at a flow rate of 19,490 cfm $\pm$ 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 4 inches Water Gauge while operating the ventilation system at a flow rate of 19,490 cfm $\pm$ 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that on a high radiation test signal, the system automatically directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

4. Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of $\geq$ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in place in accordance with ANSI N510-1975 while operating the filter train at a flow rate of 19,490 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filter train at a flow rate of 19,490 cfm $\pm$ 10%.

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 control rod worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s), and

b. All part length rods are withdrawn to at least the 180 step position and OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion or the part length rods not within their withdrawal limits, immediately initiate and continue boration at $\geq$ 10 gpm of 20,100 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor sub-critical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 10 gpm of 20,100 ppm boric acid solution 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 rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod 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 The part length rods shall be demonstrated OPERABLE by moving each part length rod $\geq$ 10 steps within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85%# of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

#A THERMAL POWER limit of 90% of RATED THERMAL POWER is permissible during tests performed as part of the Augmented Startup Test Program.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set $\leq$ 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The summary statements contained in this section provide the bases for the specifications of Sections 3.0 and 4.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36.

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 ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.1 calls for each Reactor Coolant System accumulator to be OPERABLE and provides explicit ACTION requirements when one accumulator is inoperable. Under the terms of Specification 3.0.3, if more than one accumulator is inoperable, the facility is required to be in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 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 provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

APPLICABILITY

BASES

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.

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 tolerance values, taken either individually or consecutively over 3 test intervals, 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 activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

APPLICABILITY

BASES

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 a part of these technical specifications.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

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, 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 T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} 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 1.6% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} < 200°F$, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses 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 thoughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°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 P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

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, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°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 all operating conditions of 1.6% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 5106 gallons of 20,100 ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000 ppm borated water from the refueling water storage tank. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

BASES

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,100 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 541$°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.

BASES

Control rod positions and OPERABILITY of the rod 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 restriction prohibiting part length rod insertion ensures that adverse power shapes and rapid local power changes which may effect DNB considerations do not occur as a result of part length rod insertion during operation.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ±5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD are derived from the plant nuclear instrumentation system through the AFD Monitor Alarm. A control room recorder continuously displays the auctioneered high flux difference and the target band limits as a function of power level. A first alarm is received any time the auctioneered high flux difference exceeds the target band limits. A second alarm is received if the AFD exceeds its allowable limits for a cumulative time of one hour during any 24 hour time period starting with the occurrence of the first alarm. Time outside the target band is graphically presented on the strip chart.

Figure B 3/4 2-1 shows a typical monthly target band.

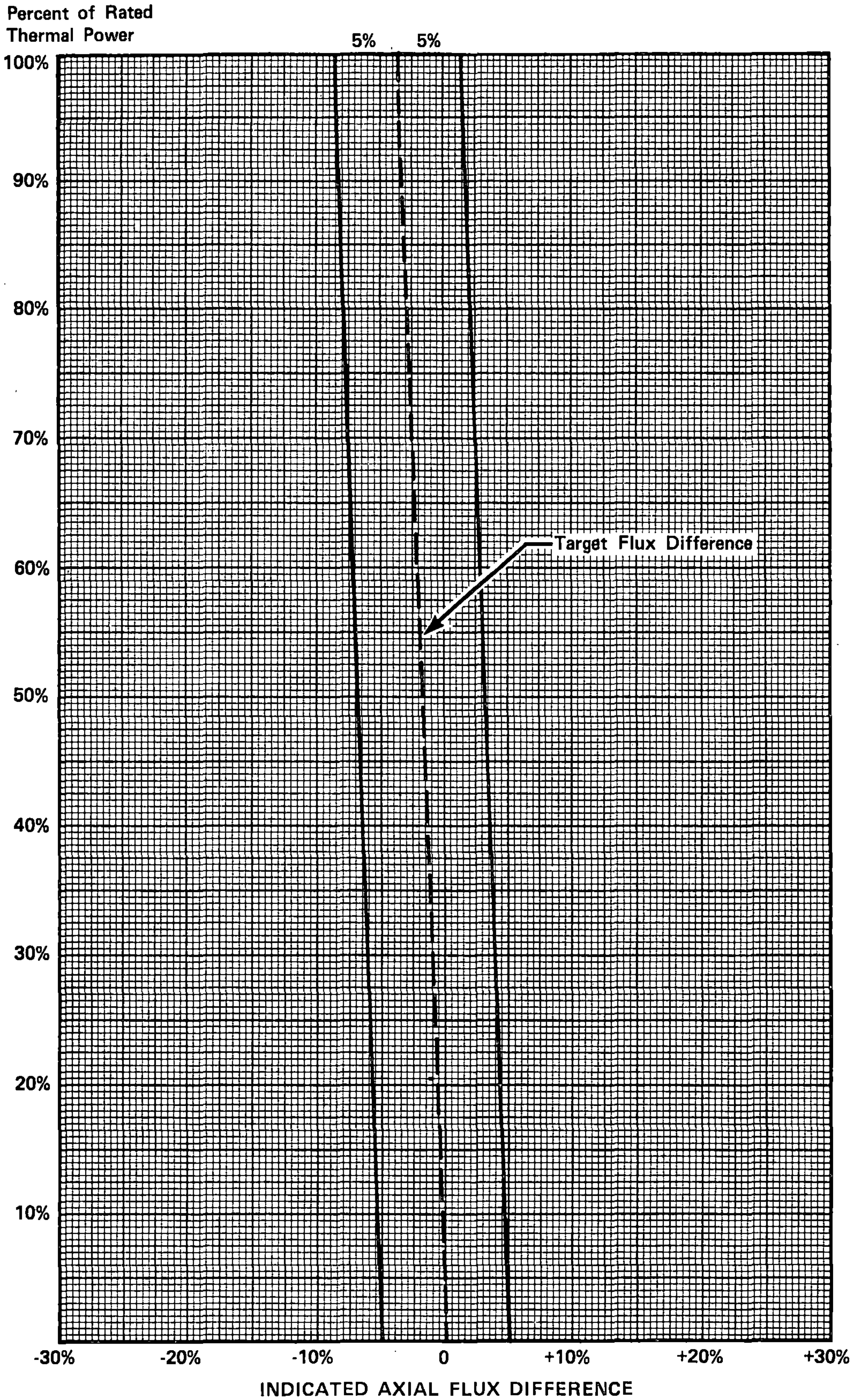


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds 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 for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

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 accident 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

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

RADIATION MONITORING INSTRUMENTATION (Continued)

by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable 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. The OPERABILITY of this system is demonstrated by irradiating each detector used and normalizing its respective output.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

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 of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to <36 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.30. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (36 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. 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, an operating RHR 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 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or 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 Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

3/4.4.5 STEAM GENERATORS (Continued)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 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 of less than 1 GPM. This threshold value 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The CONTROLLED LEAKAGE limitation restricts operation when the total flow from the reactor coolant pump seals exceeds 40 GPM with valve 1CV71 fully closed at a nominal RCS pressure of 2230 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 GPM for all steam generators (but not more than 500 gpd for any steam generator) ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces 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 concentrations 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.

BASES

3/4.4.8 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. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Salem site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-31, 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 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 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.

Reducing T_{avg} to <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 requirements 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.

3/4.4.9 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 Section 4.1.5 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. These 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 of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves, Figure 3.4-3, are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 13 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. 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 and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of 13 EFPY, 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 E185-70, 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 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 pressurizer heatup and cooldown 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.

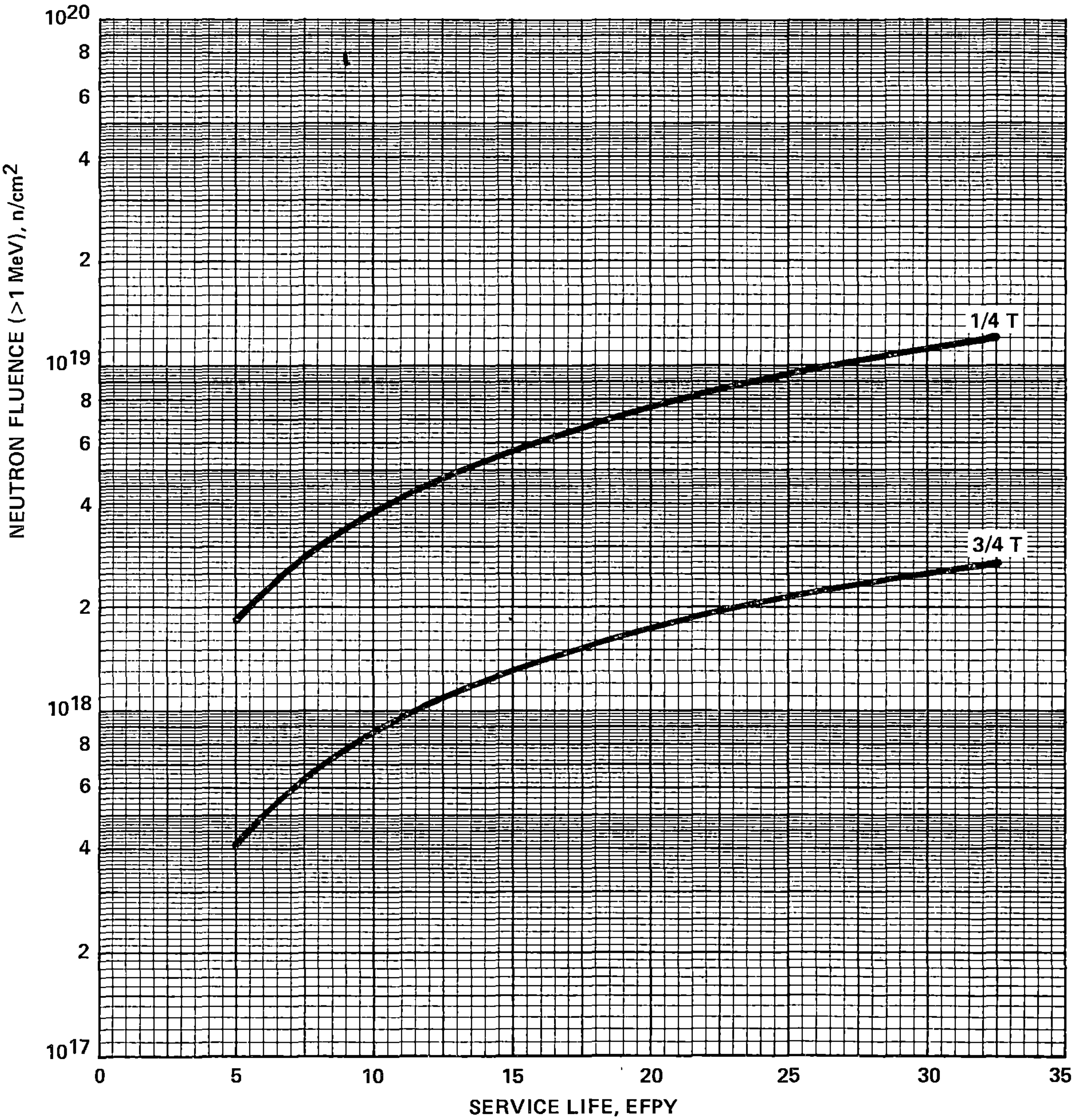


FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E >1 MeV) AS A FUNCTION OF
FULL POWER SERVICE LIFE

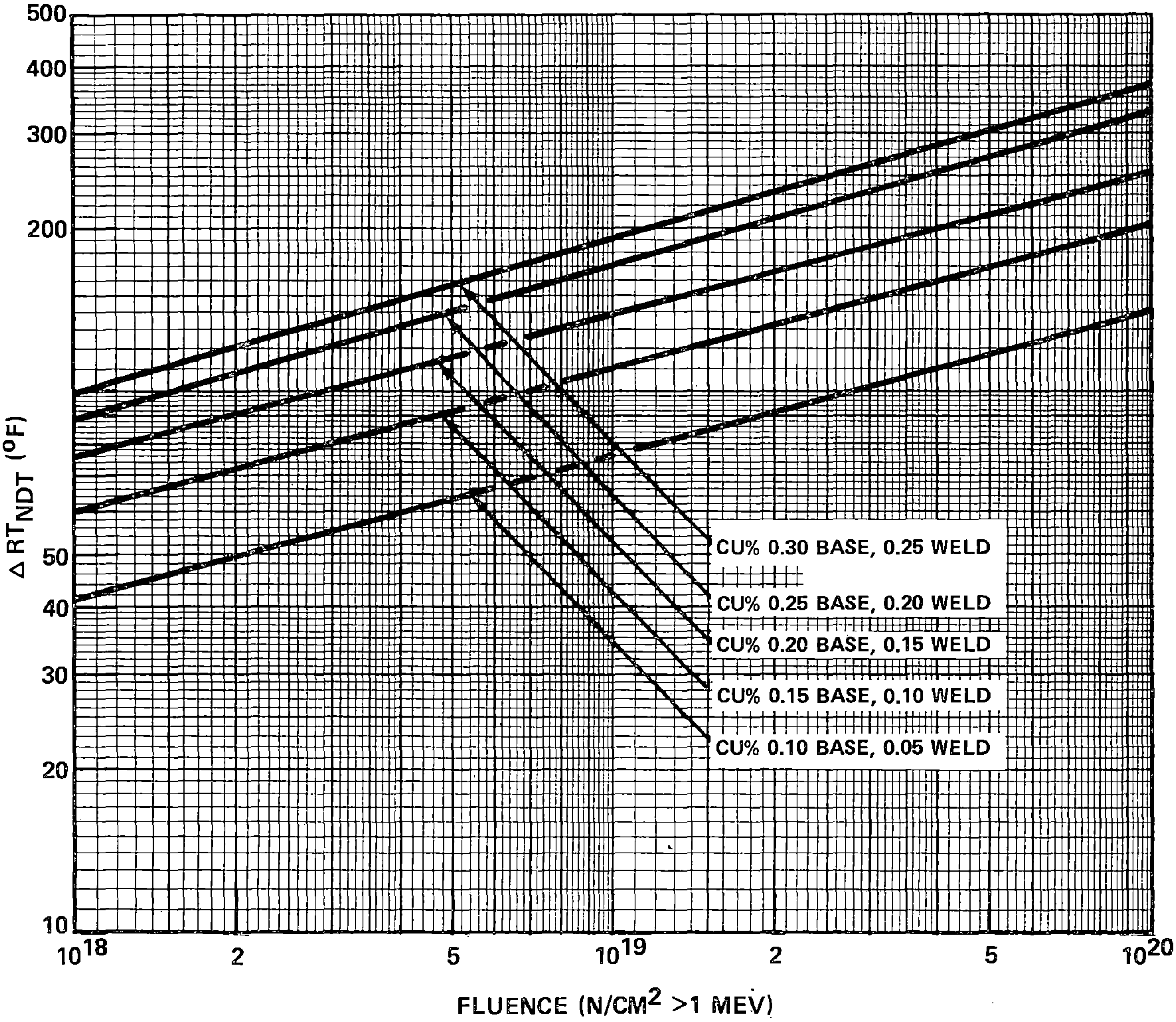


Figure B 3/4.4-2

Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessels Exposed to 550° F Temperature

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	RTNDT F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS	
CL.HD.DOME	01DB	A533B1	20	011	-30	79	117*	57	105	68**	
CL.HD.SEG.	02DC	A533B1	13	018	-20	69	96*	36	120	78**	
CL.HD.SEG.	02DD	A533B1	16	012	-30	65	96*	36	115	75**	
CL.HD.SEG.	02DE	A533B1	16	009	-50	46	75*	15	130	85**	
CL.HD.FLG.	03DF	A508,2	NA	010	28$	2	18*	28	177	115**	
VES.SH.FLG	04DG	A508,2	NA	009	50$	-20	20*	50	138	90**	
INLET NOZ.	06DH	A508,2	NA	010	55$	23	41*	55	143	93**	
INLET NOZ.	06DI	A508,2	NA	011	46$	6	30*	46	153	99**	
INLET NOZ.	06DJ	A508,2	NA	010	47$	17	35*	47	157	102**	
INLET NOZ.	06DK	A508,2	NA	010	9$	-3	-8*	9	160	104**	
OUTLET NOZ	07DL	A508,2	NA	010	60$	75	158*	98	75	48**	
OUTLET NOZ	07DM	A508,2	NA	011	60$	75	184*	124	75	48**	
OUTLET NOZ	07DN	A508,2	NA	013	60$	-10	34*	60	108	70**	
OUTLET NOZ	07DO	A508,2	NA	012	60$	-7	32*	60	124	80**	
UPPER SHL.	08DP	A533B1	NA	012	-30	67	99*	39	109	71**	
UPPER SHL.	08DR	A533B1	NA	011	0	60	164*	44	120	78**	
UPPER SHL.	08DS	A533B1	NA	011	-10	94	138*	78	90	58**	
INTER.SHL.	09DT	A533B1	24	010	-30	53	105	45	99	80	SURV
INTER.SHL.	09DU	A533B1	24	010	-30	55	55	-5	109	90	
INTER.SHL.	09DV	A533B1	22	011	-40	-20	57	-3	125	104	

$ ESTIMATED (60 F OR 100FT-LB TEMP, WHICHEVER IS LESS)
/$ ESTIMATED (0 F OR 30FT-LB TEMP, WHICHEVER IS HIGHER)
* ESTIMATED (77FT-LB/54 MIL TEMP FOR LONGITUDINAL DATA)
** ESTIMATED (65 PER CENT OF LONGITUDINAL SHELF)
SURV MATERIAL SELECTED FOR SURVEILLANCE PROGRAM (PROBABLE ACCORDING TO E185)

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	RTNDT F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS
LOWER SHL.	10DW	A533B1	19	011	-40	45	77*	17	138	90**
LOWER SHL.	10DX	A533B1	19	012	-70	58	89*	29	124	81**
LOWER SHL.	10DY	A533B1	19	010	-40	46	93*	33	124	81**
BOT.HD.SEG	12DZ	A533B1	10	009	10	28	71*	11	117	76**
BOT.HD.SEG	12EA	A533B1	11	010	-50	40	76*	16	131	85**
BOT.HD.SEG	12EB	A533B1	12	008	10	27	64*	10	118	76**
BOT.HD.DOM	13EC	A533B1	15	010	-20	37	84*	24	104	68**
WELD	14ED	WELD	16	019	0/$	NA	-38	0	NA	97
HAZ CORE	15ED	HAZ	NA	NA	NA	-28	NA	NA	107	NA

/$ ESTIMATED (0 F OR 30FT-LB TEMP, WHICHEVER IS LESS)
* ESTIMATED (77FT-LB/54 MIL TEMP FOR LONGITUDINAL DATA)
** ESTIMATED (65 PER CENT OF LONGITUDINAL SHELF)

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for 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.

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator 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 independent ECCS subsystems 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 operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures 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 long term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21000 ppm boron.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST 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 RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the containment peak pressure does not exceed the design pressure of 47 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 44.8 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 45.1 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 44.4 psig in the event of a LOCA. The visual inspections of the concrete and liner and the Type A leakage test are sufficient to demonstrate this capability.

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 accident analyses.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

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. 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.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment.

3/4.7 PLANT SYSTEMS

BASES

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% of its design pressure of 1085 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 Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 16,655,268 lbs/hr which is 115 percent of the total secondary steam flow of 14,459,360 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

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 Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

76 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 3 loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 440 gpm at a pressure of 1150 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 880 gpm at a pressure of 1150 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 AUXILIARY FEED STORAGE TANK

The OPERABILITY of the auxiliary feed 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 the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

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. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a 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 accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

A test program will be conducted during approximately the first 6 months of operation after initial criticality to establish the appropriate limits on the secondary water chemistry parameters and to determine the appropriate frequencies for monitoring these parameters. The results of this test program will be submitted to the Commission for review. The Commission will then issue a revision to this specification specifying the limits on the chemistry parameters and the frequencies for monitoring these parameters.

SECONDARY WATER CHEMISTRY (Continued)

The test program will include an analysis of the chemical constitutent of the river water at the point of intake for the Salem Nuclear Generating Station. The analysis shall identify the various traces of ions which upon concentration may have the potential for inducement for stress corrosion in the steam generator tubing. The test program shall also evaluate the efficiency of the water treatment systems in the Salem facility for removal of such ions and the potential for addition of other ions resulting from the treatment method. The test program shall analyze concentration phenomena and the concentration rates in the steam generator and the secondary water system and shall consider concentration in the recirculating cooling water system.

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 limitations of 70°F and 200 psig are based on average steam generator impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. The limit of elevation 10.5' Mean Sea Level is based on the elevation above which facility flood control measures are required to provide protection to safety related equipment.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

The OPERABILITY of the control room emergency air conditioning system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 10 of Appendix "A", 10 CFR 50.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM

The OPERABILITY of the auxiliary building exhaust air filtration system ensures that radioactive materials leaking from the ECCS equipment following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.9 HYDRAULIC SNUBBERS

The hydraulic snubbers are required 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 only snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment (only ethylene propylene compounds to date) is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during an inspection of these snubbers determines the time interval for the next required inspection of these snubbers. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in high radiation zones or in especially difficult to remove locations (as identified in Table 3.7-4) may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

BASES

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) 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. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

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 accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 COOLANT CIRCULATION

The requirements that at least one residual heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

3/4.9.9 CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND 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 FUEL HANDLING AREA VENTILATION SYSTEM

The limitations on the fuel handling area ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod 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.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods 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 control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation.

3/4.10.4 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 210 feet.

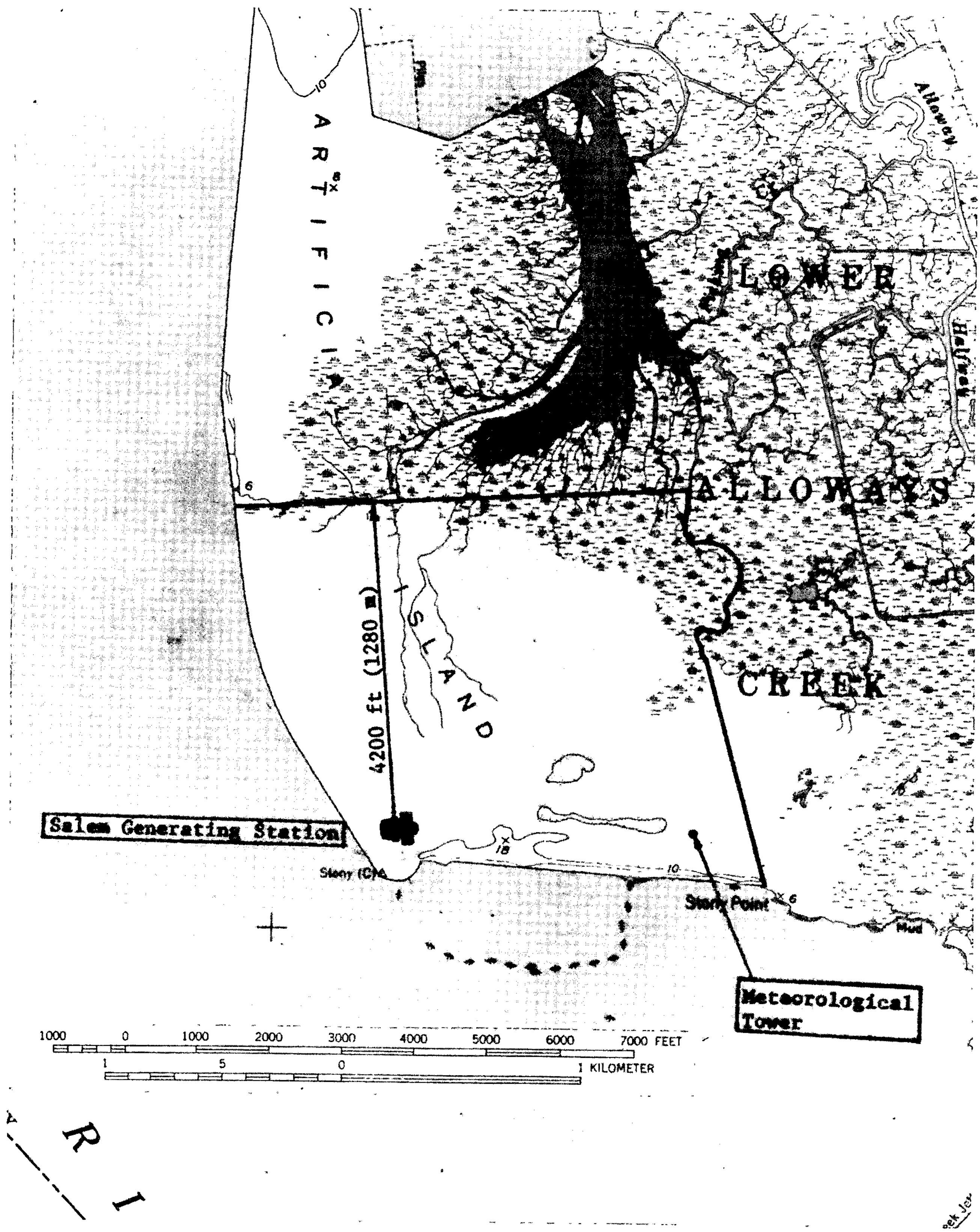
c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 3.5 feet.

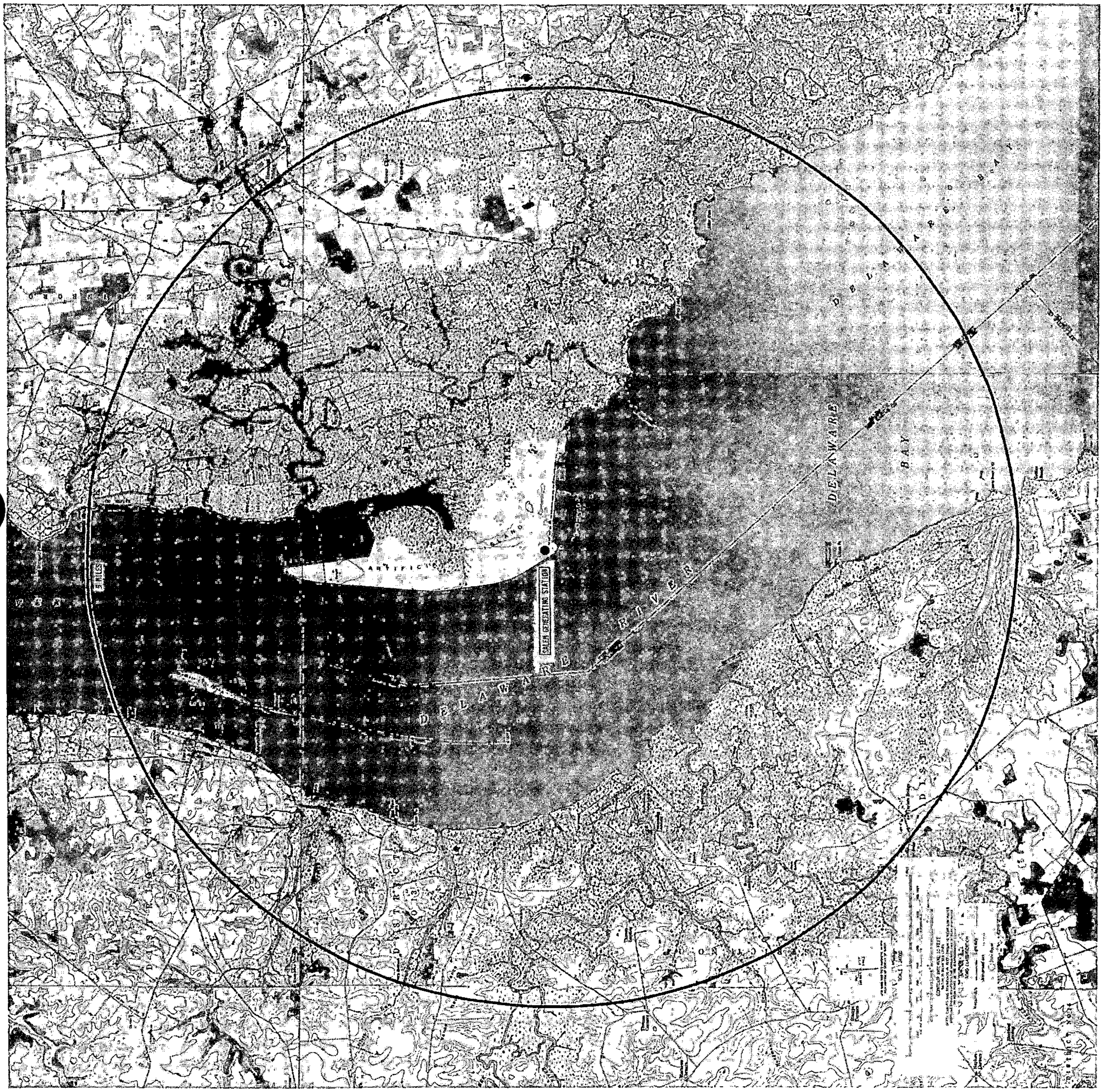
e. Minimum thickness of concrete floor mat = 16 feet.

f. Nominal thickness of steel liner = 1/4 to 1/2 inch.

g. Net free volume = 2.62×10^6 cubic feet.



EXCLUSION AREA
FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 47 psig and an air temperature of 271°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 143.7 inches and contain a nominal total weight of 1743 grams uranium. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and 8 part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The part length control rod assemblies shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The balance of the void length in the part length rods shall contain aluminum oxide.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,811 $\pm$ 100 cubic feet at a nominal T_{avg} of 576.7°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The k_{eff} of $\leq$ 0.95 includes a conservative allowance of 3.3% $\Delta k/k$ for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 124'8".

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 264 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 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	200 heatup cycles at $\leq$ 100°F/hr and 200 cooldown cycles at $\leq$ 100°F/hr (pressurizer cooldown at $\leq$ 200°F/hr).	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 542°F. Cooldown cycle - T_{avg} from $\geq$ 542°F to $\leq$ 200°F.
	80 loss of load cycles.	Without immediate turbine or reactor trip.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	200 large step decreases in load.	50% of RATED THERMAL POWER step load decrease with steam dump.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe > 13.5 inches equivalent diameter.
	Operating Basis Earthquake Design Basis Earthquake	50 cycles 10 cycles; 0.20g horizontal, 0.136g vertical.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests	Pressurized to ≥ 3107 psig.
Secondary System	1 steam line break	Break in a steam line > 6 inches equivalent diameter.
	5 hydrostatic pressure tests	Pressurized to ≥ 1356 psig.
	10 turbine roll tests	Turbine roll on pump heat resulting in plant cooldown > 100°F/hr.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

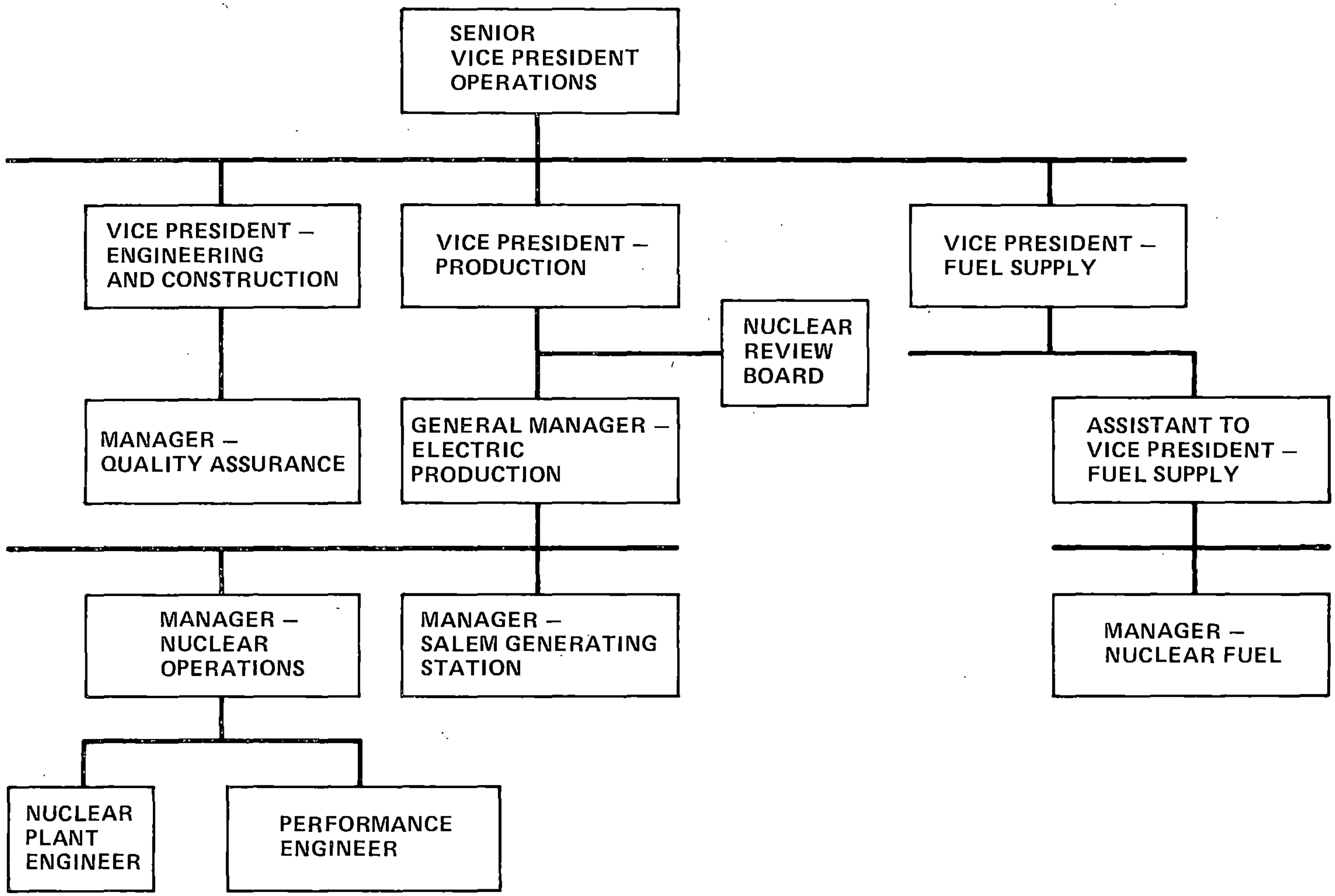


FIGURE 6.2-1. OFFSITE ORGANIZATION FOR FACILITY MANAGEMENT AND TECHNICAL SUPPORT

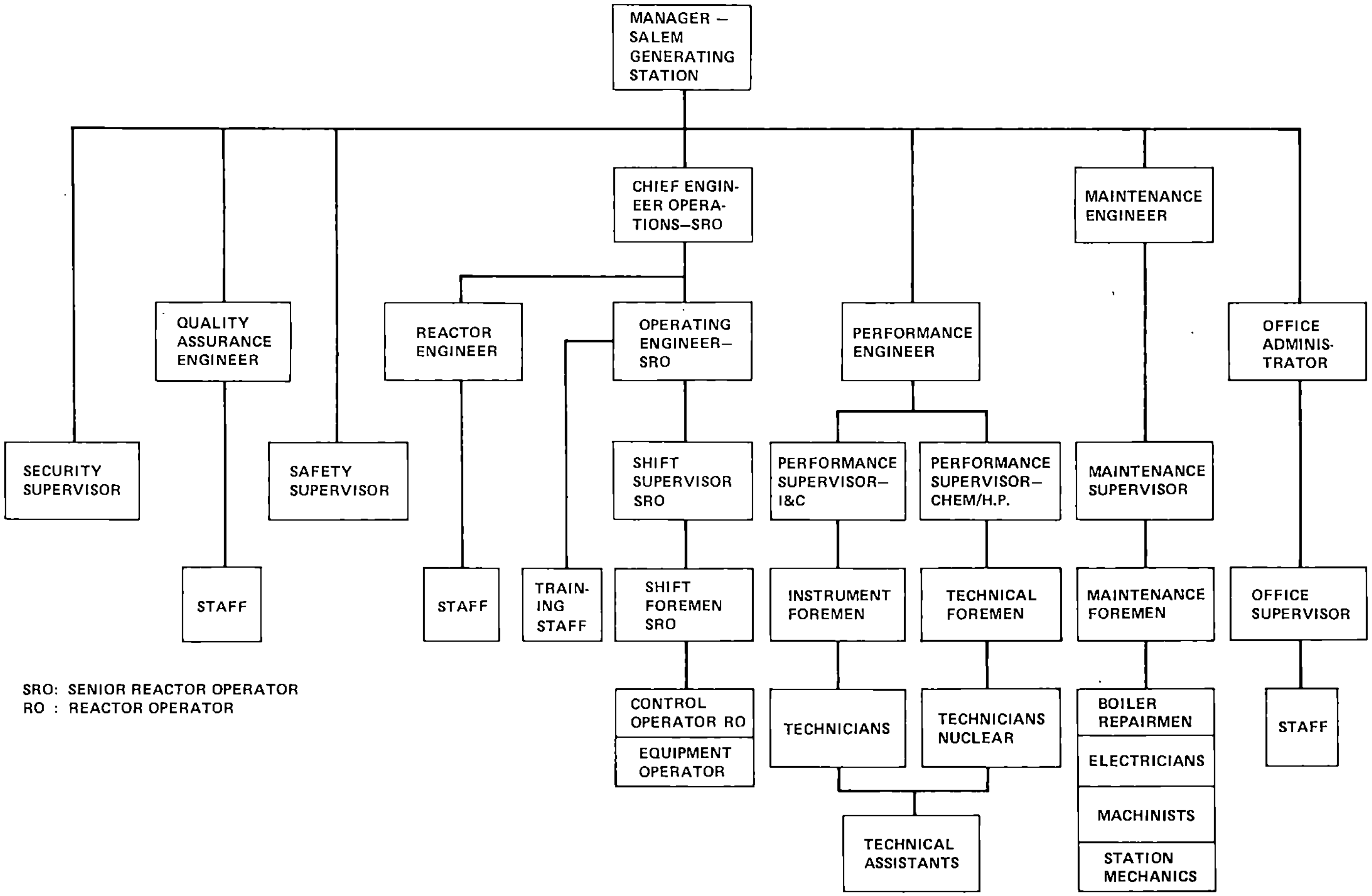


FIGURE 6.2-2. FACILITY ORGANIZATION – SALEM NUCLEAR GENERATING STATION – UNIT 1

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Performance Supervisor - Chemistry/HP who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Chief Engineer and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 STATION OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.1.1 The Station Operations Review Committee shall function to advise the Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Station Operations Review Committee shall be composed of the:

Chairman:	Chief Engineer
Vice Chairman:	Maintenance Engineer
Member:	Operating Engineer
Member:	Performance Engineer
Member:	Reactor Engineer
Member:	Shift Supervisor
Member:	Performance Supervisor - I&C
Member:	Performance Supervisor - Chemistry/HP
Member:	Maintenance Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SORC shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Station Operations Review Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Station Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the General Manager - Electric Production and to the Chairman of the Nuclear Review Board.

f. Review of events requiring 24 hour written notification to the Commission.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Nuclear Review Board.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear Review Board.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear Review Board.

AUTHORITY

6.5.1.7 The Station Operations Review Committee shall:

a. Recommend to the Station Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the General Manager-Electric Production and the Nuclear Review Board of disagreement between the SORC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Station Operations Review Committee shall maintain written minutes of each meeting and copies shall be provided to the General Manager-Electric Production and Chairman of the Nuclear Review Board.

6.5.2 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.2.1 The Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The NRB shall be composed of the:

Chairman:	General Manager-Electric Production
Vice Chairman:	General Manager-Construction
Member:	General Manager-Projects
Member:	Manager-Nuclear Operations
Member:	Assistant to General Manager-Fuel Supply
Member:	Manager-Quality Assurance
Member:	Project Manager-Hope Creek
Member:	Manager-Salem Generating Station

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.2.5 The NRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of NRB shall consist of the Chairman or his designated alternate and at least 4 NRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The NRB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this operating license.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Station Operations Review Committee.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NRB. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. Any other area of facility operation considered appropriate by the NRB or the Vice President-Production.

AUTHORITY

6.5.2.9 The NRB shall report to and advise the Vice President-Production on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NRB meeting shall be prepared, approved and forwarded to the Vice President-Production within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President-Production within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice President-Production and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the SORC and submitted to the NRB and the General Manager-Electric Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the General Manager-Electric Production and to the NRB within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NRB and the General Manager-Electric Production within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the SORC and approved by the Station Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SORC and approved by the Station Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

6.9 REPORTING REQUIREMENTS (Continued)

d. Seismic event analysis, Specification 4.3.3.3.2.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the NRB.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 RESPIRATORY PROTECTION PROGRAM

ALLOWANCE

6.12.1 Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column I, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in §20.101. These materials shall be subject to applicable process and other engineering controls.

PROTECTION PROGRAM

6.12.2 In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in 6.12.1 above, are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for OPERABILITY immediately prior to use.

4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U.S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U.S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

REVOCATION

6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1.a above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Watch Foreman on duty.

TABLE 6.12-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION(7)	MODES(1)	PROTECTION FACTORS(2) PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE(3)	GUIDES TO SELECTION OF EQUIPMENT* BUREAU OF MINES NATIONAL INSTITUTE FOR OCCUPATIONAL SAFETY AND HEALTH APPROVALS (or schedule superseding for equipment type listed)
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask(4)	NP	5	30 CFR Part 11 Subpart K
Facepiece, full	NP	100	30 CFR Part 11 Subpart K
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	30 CFR Part 11 Subpart J
Facepiece, full	CF	1,000	30 CFR Part 11 Subpart J
Facepiece, full	D	100	30 CFR Part 11 Subpart J
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart J
Hood	CF	(5)	30 CFR Part 11 Subpart J
Suit	CF	(5)	(6)
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full	D	100	30 CFR Part 11 Subpart H
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart H
Facepiece, full	R	100	30 CFR Part 11 Subpart H
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	30 CFR Part 11 § 11.63(b)

TABLE 6.12-1 (Continued)

TABLE NOTATION

[1] See the following symbols:

- CF: continuous flow
- D: demand
- NP: negative pressure (i.e., negative phase during inhalation)
- PD: pressure demand (i.e., always positive pressure)
- R: recirculating (closed circuit)

[2](a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency [above 99.9% removal efficiency by U.S. Bureau of Mines type dioctyl phthalate (DOP) test] particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

[3] Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote [5], below, concerning supplied-air suits and hoods.

TABLE 6.12-1 (Continued)

TABLE NOTATION

[4] Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

[5] Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

[6] No approval schedules current available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

[7] Only for shaven faces and where nothing interferes with the seal of tight fitting facepieces against the skin. (Hoods and suits are excepted.)

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines or the National Institute for Occupational Safety and Health according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Bureau of Mines or the National Institute for Occupational Safety and Health in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

APPENDIX B

TO

OPERATING LICENSE DPR-70

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

FOR

SALEM

NUCLEAR GENERATING STATION

UNITS 1 AND 2

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DOCKET NOS. 50-272, 50-311

AUG 13 1976

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

Section	Title	Page
1.0	DEFINITIONS, ABBREVIATIONS AND NOTES	1.1-1
1.1	DEFINITIONS	1.1-1
1.2	ABBREVIATIONS	1.2-1
1.3	NOTES	1.3-1
2.0	LIMITING CONDITIONS FOR OPERATION	2.1-1
2.1	THERMAL	2.1-1
2.1.1a	MAXIMUM ΔT ACROSS CONDENSER	2.1-1
2.1.1b	MAXIMUM ΔT ACROSS CONDENSER DURING PUMP OUTAGE	2.1-2
2.1.2	MAXIMUM DISCHARGE TEMPERATURE	2.1-4
2.1.3	RATE OF CHANGE OF DISCHARGE TEMPERATURE	2.1-7
2.2	CHEMICAL	2.2-1
2.2.1	BIOCIDES	2.2-1
2.2.2	SUSPENDED SOLIDS	2.2-4
2.2.3	pH	2.2-5
2.3	RADIOACTIVE EFFLUENTS	2.3-1
2.3.1	SPECIFICATIONS FOR LIQUID WASTE EFFLUENTS	2.3-2
2.3.2	SPECIFICATIONS FOR LIQUID WASTE SAMPLING AND MONITORING	2.3-4
2.3.3	SPECIFICATIONS FOR GASEOUS WASTE EFFLUENTS	2.3-8
2.3.4	SPECIFICATIONS FOR GASEOUS WASTE SAMPLING AND MONITORING	2.3-13
2.3.5	SPECIFICATIONS FOR SOLID WASTE HANDLING AND DISPOSAL	2.3-19
3.0	ENVIRONMENTAL SURVEILLANCE	3.0-1
3.1	NON-RADIOLOGICAL SURVEILLANCE	3.1-1
3.1.1	ABIOTIC	3.1-1
3.1.1.1	Chlorine	3.1-1
3.1.1.2	Dissolved Gases	3.1-2
3.1.1.3	Suspended Solids	3.1-3
3.1.1.4	Other Chemicals	3.1-4
3.1.1.5	Chemical Releases	3.1-6
3.1.1.6	Meteorological Monitoring	3.1-7

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.1.2	BIOTIC	3.1-9
3.1.2.1	General Ecological Survey	3.1-9
3.1.2.2	Impingement of Organisms	3.1-15
3.1.2.3	Entrainment of Plankton Organisms	3.1-17
3.1.2.4	References	3.1-21
3.2	RADIOLOGICAL SURVEILLANCE	3.2-1
4.0	SPECIAL SURVEILLANCE AND STUDY ACTIVITIES	4.1-1
4.1	EXPERIMENTAL ENTRAINMENT STUDIES	4.1-1
4.2	THERMAL AND CHEMICAL RESPONSES OF ESTUARINE ORGANISMS	4.2-1
4.2.1	GENERAL	4.2-1
4.2.2	TEMPERATURE PREFERENCE STUDIES	4.2-2
4.2.3	TEMPERATURE AND CHEMICAL AVOIDANCE STUDIES	4.2-3
4.2.4	COLD SHOCK STUDIES	4.2-4
4.3	THERMAL PLUME MAPPING	4.3-1
4.4	INTAKE VELOCITY STUDY	4.4-1
5.0	ADMINISTRATIVE CONTROLS	5.1-1
5.1	RESPONSIBILITY	5.1-1
5.2	ORGANIZATION	5.2-1
5.3	REVIEW AND AUDIT	5.3-1
5.3.1	NUCLEAR REVIEW BOARD (NRB)	5.3-1
5.3.2	STATION OPERATIONS REVIEW COMMITTEE (SORC)	5.3-1
5.4	ACTION TO BE TAKEN IF A LIMITING CONDITION FOR OPERATION IS EXCEEDED	5.4-1
5.5	PROCEDURES	5.5-1
5.6	PLANT REPORTING REQUIREMENTS	5.6-1
5.6.1	ROUTINE REPORTS	5.6-1
5.6.1.1	Annual Environmental Operating Report	5.6-1
5.6.1.2	Radioactive Effluents Release Report	5.6-3

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
5.6.2	NONROUTINE REPORTS.	5.6-3
5.6.2.1	Nonroutine Environmental Operating Reports	5.6-3
5.6.2.2	Nonroutine Radiological Environmental Operating Report	5.6-4
5.6.2.3	Nonroutine Radioactive Effluent Reports.	5.6-7
5.6.3	CHANGES IN ENVIRONMENTAL TECHNICAL SPECIFICATIONS.	5.6-7
5.7	RECORDS RETENTION.	5.7-1

LIST OF TABLES

Table No.	Title	Page
2.3-1	Radioactive Liquid Sampling and Analysis	2.3-20
2.3-2	Radioactive Gaseous Waste Sampling and Analysis	2.3-21
2.3-3	Liquid Waste System Specifications	2.3-22
2.3-4	Gaseous Waste System Specifications	2.3-23
2.3-5	Gamma and Beta Dose Factors	2.3-24
3.1-1	Water Quality Analysis Parameters	3.1-22
3.1-2	Summary of Aquatic, Terrestrial and Aerial Sampling Programs	3.1-23
3.1-3	Anticipated Chemical Waste Discharge	3.1-24
3.2-1	Operational Environmental Radiological Monitoring Program	3.2-6
3.2-2	Sensitivity Levels for Environmental Sample Analyses	3.2-9
5.6-1	Environmental Radiological Monitoring Program Summary	5.6-9

LIST OF FIGURES

Figure No.	Title	Page
3.1-1	Biological Sampling Stations in the Vicinity of Artificial Island	3.1-25
5.2-1	Organization Chart Showing Corporate Interrelationships	5.2-2
5.2-2	Mechanical Division Organization Chart	5.2-3
5.2-3	Station Organization Chart	5.2-4

1.0 DEFINITIONS, ABBREVIATIONS AND NOTES

1.1 DEFINITIONS

1. AMBIENT TEMPERATURE

   Temperature of the river unaffected by localized waste heat discharge; temperature of the river outside the designated mixing zone.

2. AMPEROMETRIC TITRATION

   Specific adaptation of polarographic principles which are used to measure the total residual chlorine or to differentiate between the free and combined available chlorine.

3. CALIBRATION

   Use of a known quantity of a measured parameter to determine the accuracy of the measuring instrument.

4. CHLORINE DEMAND

   The amount of chlorine required to oxidize substances in the water which reduce free chlorine.

5. COMBINED AVAILABLE CHLORINE RESIDUAL

Residual consisting of mono-, di-, and trichloromines.

6. CONDENSER

Shall include the three condenser shells utilized in the Circulating Water System for each unit.

7. CONDENSER OUTLET TEMPERATURE

The average condenser outlet circulating water temperature of those condenser sections in service measured as per DISCHARGE TEMPERATURE.

8. CONDENSER SHELL

A single heat exchanger in the Circulating Water System which includes two inlet and outlet water boxes and two tube bundles.

9. CONTROL STATION

Sample location that is far enough away from the station that it will not be affected by radiological emissions or other station releases.

10. DISCHARGE TEMPERATURE

The average temperature of the six 84-inch ID discharge lines for each unit. The temperatures are measured at a point approximately 20 feet downstream from the condenser outlet water boxes.

11. DISCHARGE VELOCITY

The average of the velocities from the three 120-inch ID circulating water discharge pipes from each unit. This number is calculated, based on discharge cross-sectional area and pump flow.

12. EMERGENCY NEED FOR POWER

An emergency need for power shall be considered to exist if the system is unlikely to meet the demand after the licensee has attempted to satisfy its requirements by operating all other available base load units.

13. ENVIRONMENTAL SAMPLES

Samples of soil, air, water, biota, or biological material collected for the purposes of analysis.

14. ENVIRONMENTAL SIGNIFICANCE

Exceeding a report level, or when, in the opinion of the Station Superintendent, an event which causes an adverse impact on the environment has occurred.

15. FREE AVAILABLE CHLORINE RESIDUAL

Residual consisting of hypochlorite ions (OCl), hypochlorous acid (HOCl), or molecular chlorine (Cl_2).

16. FUNCTIONAL TEST

Use of a simulated signal or check source to determine instrument operability.

17. GAMMA SCAN (GAMMA SPECTROSCOPY)

Identification of gamma emitting isotopes, using a multi-channel analyzer.

18. INDICATOR STATION

Sample location where any adverse environmental effects resulting from station operation could be perceived.

19. INSTRUMENT CHECK

Visual inspection of a monitor readout.

20. INTAKE TEMPERATURE

The temperature of the circulating water as measured at the inlet to the condenser shells; the temperature in the service water headers to the nuclear area and turbine area.

21. NORMAL OPERATION

Steady state operation at any power level; includes operation with up to 10% of condenser tubes blocked.

22. REPORT LEVEL

The numerical level of an environmental parameter below which the environmental impact is considered reasonable based on available information.

23. SPECIAL STUDY PROGRAMS

Environmental study programs designed to evaluate the impact of station operation on an environmental parameter.

24. TOTAL AVAILABLE CHLORINE RESIDUAL

Sum of free and combined available chlorine residuals.

1.2 ABBREVIATIONS

1.	EPA	United States Environmental Protection Agency
2.	JTU	Jackson Turbidity Units
3.	MDA	Minimum Detectable Activity
4.	MHT	Mean High Tide
5.	MLT	Mean Low Tide
6.	MT	Mean Tide
7.	NPDES	National Pollutant Discharge Elimination System
8.	NRC	United States Nuclear Regulatory Commission

1.3 <u>NOTES</u>

1. The Enviornmental Technical Specifications are limitations, conditions and requirements which are considered necessary for the protection of the environment. Safety Technical Specifications are imposed upon plant operation in the interest of the health and safety of the public.

   <u>Compliance with the Safety Technical Specifications shall have precedence over the Environmental Technical Specifications at all times.</u>

2. Each Monitoring Requirement shall be performed within the specified time interval, unless otherwise noted, with a maximum allowable extension not to exceed 25% of the monitoring interval.

   This extension also applies to all sampling, instrument check, calibration and functional test frequencies.

   These provisions provide allowable tolerances for performing monitoring activities beyond those specified in the nominal monitoring interval. These tolerances are necessary to provide operational flexibility because of scheduling, performance considerations and environmental influences.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 THERMAL

2.1.1a MAXIMUM ΔT ACROSS CONDENSER DURING NORMAL OPERATION

Objective

To limit thermal stress to the aquatic ecosystem by limiting the maximum ΔT across the condenser during normal operation.

Specification

The maximum ΔT across the condenser shall not exceed 16.5°F during normal operation with all six circulating water pumps operating. In the event that the specification is exceeded corrective action shall be taken to reduce the ΔT to within specification. Such corrective action could include cleaning condenser water boxes or reduction of unit power level, unless an emergency need for power exists.

Monitoring Requirement

The temperature differential across the condenser shall be monitored every hour utilizing the computer printout of the intake and discharge temperature measurements. The intake temperature is measured at each of the two inlets to each condenser shell. The discharge temperature is measured at a point downstream of the condenser in each of the two 84-inch ID discharge lines from each condenser shell. The range of this instrumentation is 0 - 150°F and the system accuracy is ± 0.5°F.

If the plant computer is out of service, the intake and discharge temperatures shall be monitored every two hours utilizing local reading instrumentation until the plant computer is returned to service.

Bases

The condenser cooling water system was designed to minimize thermal stress to organisms which may be entrained in the cooling water discharge. The overall impact on all species is not expected to be significant for the following reasons:

1. Comparatively small amounts of water are utilized for cooling purposes compared to tidal flow (on the order of 1%).

2. No thermal blockages for the Delaware estuary are predicted by the thermal plume model.

3. Studies show almost total survival among most potentially entrainable important species found in the vicinity of the plant exposed for 10 minutes to a ΔT of 16.5°F. During normal plant operation the period of entrainment will be less than 4 minutes.

2.1.1b MAXIMUM ΔT ACROSS CONDENSER DURING PUMP OUTAGE

Objective

To limit thermal stress to the aquatic ecosystem by limiting the maximum ΔT across the condenser during pump outage.

Specification

1. The maximum ΔT across the condenser shall not exceed 16.5°F for more than 24 consecutive hours because of scheduled maintenance and inspection.

2. The maximum ΔT across the condenser shall not exceed 16.5°F for more than 72 consecutive hours for reasons of pump failure.

3. At no time will the ΔT across the condenser exceed 27.5°F. In the event that either specification is exceeded, corrective action shall be taken to reduce the ΔT to within specification. Such corrective action could include cleaning condenser water boxes or reduction of limit power level, unless an emergency need for power exists.

Monitoring Requirement

The temperature differential across the condenser shall be monitored every hour utilizing the computer printout of the intake and discharge temperature measurements. The intake temperature is measured at each of the two inlets to each condenser shell. The discharge temperature is measured at a point downstream of the condenser in each of the two 84-inch ID discharge lines from each condenser shell. The range of this instrumentation is 0 - 150°F and the system accuracy is $\pm$ 0.5°F.

If the plant computer is out of service, the intake and discharge temperatures shall be monitored every two hours utilizing local reading instrumentation until the plant computer is returned to service.

Bases

The condenser cooling water system was designed to operate with a ΔT that would minimize thermal stress to organisms. The U.S. Environmental Protection Agency has set a limit of 27.5°F as a maximum ΔT permitted under the NPDES permit number NJ 0005622. The overall impact of occasional operating delta temperatures in excess of 16.5°F is not expected to be significant for the following reasons:

1. Less cooling water would be required during operation at the higher ΔT than during normal operation.

2. No thermal blockages for the Delaware estuary are predicted by the thermal plume model.

3. Studies show a relatively high survival among potentially entrainable important species found in the vicinity of the plant exposed for 10 minutes to a ΔT of 27.5°F. During a pump outage the period of entrainment will be less than 8 minutes.

2.1.2 MAXIMUM DISCHARGE TEMPERATURE

Objective

To limit thermal stress to the aquatic ecosystem by limiting the plant discharge water temperature.

Specification

1. The maximum condenser discharge water temperature shall not exceed 104°F for more than two consecutive hours within any 24 hour period with all six circulating water pumps in operation.

2. In the event that fewer than six circulating water pumps are in operation, the maximum condenser discharge water temperature shall not exceed 115°F for more than eight consecutive hours within any 24 hour period.

3. In the event specifications 2.1.2.1 or 2.1.2.2 are exceeded corrective action shall be taken to reduce the condenser discharge water temperature to within specification. Such corrective action could include cleaning condenser water boxes or reduction of unit power level, unless an emergency need for power exists.

Monitoring Requirement

Discharge temperature shall be monitored every hour utilizing the average of the computer printout of the discharge temperature measurements. The discharge temperature is measured at a point downstream of the condenser in each of the two 84-inch ID discharge lines from each condenser shell. The range of this instrumentation is 0-150°F and accuracy is $\pm$ 0.5°F.

If the plant computer is out of service, the discharge temperature shall be monitored every two hours utilizing local reading instrumentation until the plant computer is returned to service.

Bases

Ichthyological Associates (IA) studies performed from June 1968 through December 1973 show 25 records of river temperatures $\geq$ 84°F. Twenty-one of these records were at river surface and only one was at night. From 1970 through 1973 the U.S. Geological Survey temperature sensor at Reedy Island recorded temperatures $\geq$ 84°F on only four dates. The earliest calendar date of record by IA was June 26 (in 1969); the latest was September 7 (in 1973).

During this period for potentially high temperatures, phytoplankton, zooplankton, and ichthyoplankton have been annually collected. These non-motile organisms encountering the plume can experience mortality only if lethal time-temperature histories are experienced. At the predicted velocity range, effects on drifting organisms and passive life stages of motile spieces are minimized by short exposure time. Effects on sessile benthos in the near field (outside the area of maximum temperature and velocity and scouring) will be negligible since the plume will be primarily a surface phenomenon.

Motile organisms encountering a thermal plume will either (1) pass through it, (2) prefer it over ambient conditions, or (3) avoid it. Avoidance can be considered detrimental in that its result is similar to a loss of potential habitat. However, only a minor portion of the total available habitat will be made unavailable by the temperature-velocity pattern. Delaware River organisms will not be isolated from environmental conditions presently available to them. The discharge velocity will exclude most motile organisms from the maximum temperatures.

2.1.3 RATE OF CHANGE OF DISCHARGE TEMPERATURE

Objective

To minimize thermal stress to the aquatic environment due to sudden changes in water temperature.

Specification

1. The rate of change of discharge temperature shall not exceed 8°F per hour during normal plant shutdowns.

   If this specification is exceeded, the rate of reduction of plant power level shall be reduced such that the rate of change of discharge temperature is within specification.

   This limitation may be exceeded for brief periods as necessary to protect plant equipment and for certain safeguard operations which cannot be limited or negated by plant operation. These safeguard operations include automatic plant trips and compliance with the Safety Technical Specifications.

2. Both units shall not be intentionally shut down concurrently during the period of November through April. This specification is not applicable if shutdown is required to protect the health and safety of the public or for compliance with the Safety Technical Specifications.

Monitoring Requirement

Same as Specification 2.1.2, except that the discharge temperature shall be monitored every 15 minutes during power reductions of greater than 25% of full power.

Bases

All organisms have lower lethal temperatures. In temperate latitudes, such lethal temperatures are generally reached only when the ambient water temperature approaches freezing. The phenomenon of "cold shock" has been found to be most severe during the period of low ambient water temperatures ($\leq$ 40°F). The likelihood of reaching lower lethal temperatures can be minimized by maintaining a heated discharge during the period when ambient temperatures are < 40°F. The potential for cold shock and its effects will be minimized since the thermal effluent from one unit will compensate for possible shutdown of the other unit.

2.2 CHEMICAL

2.2.1 BIOCIDES

Objective

To insure that the chlorine residual released from the Circulating Water and Service Water Systems is controlled and will not have an adverse effect on the natural aquatic environment of the receiving waters.

Specification

1. The concentration of free chlorine in the Circulating Water System and Service Water System shall not be greater than 1.0 mg/liter at the outlet of the final heat exchanger.

   If this specification is exceeded, the chlorine addition rate shall be reduced as necessary to operate within the specification.

2. Circulating Water and Service Water pump intakes shall not be chlorinated more than 3 times per day. Chlorination periods shall not exceed 30 minutes. Chlorination of more than 3 Circulating Water pump intakes at one time shall not be permitted.

Monitoring Requirement

The outlet water boxes of the condenser shells that are being chlorinated shall be continuously monitored for free chlorine residual during treatment. The Service Water System shall be monitored at the 30-inch supply header to the turbine generator area during treatment.

The continuous monitoring (during treatment) shall be performed using a Wallace & Tiernan Series 50-236 free chlorine residual analyzer equipped with a strip chart recorder. The Circulating Water System and the Service Water System each have a separate free chlorine residual analyzer.

The chlorine monitors shall be calibrated once per month with an amperometric titrator and using ASTM Methods D-1235 and D-142, to 0.01 ppm accuracy.

If the chlorine monitors are inoperable, free chlorine residual shall be determined by manual analysis of a grab sample taken during the chlorination cycle.

Bases

The Water Quality Certificate issued by the Delaware River Basin Commission for Salem Nuclear Generating Station limits the free chlorine residual in circulating water discharged from the plant to maximum of 0.1 mg/liter. This also conforms to EPA-NPDES requirements of 0.2 to 0.5 mg/liter.

Intermittent treatment of cooling circuits in fresh and brackish water environments with a biocide (chlorine, sodium hypochlorite) is a reliable method for maintaining these circuits free from fouling.

It has been determined from past experience that treatment with chlorine at a concentration of 0.5 mg/liter free chlorine residual at the heat exchanger outlet (e.g., condenser) for 30 minutes three times a day is usually sufficient for maintaining system cleanliness although higher concentrations in the heat exchangers may be needed periodically. The discharge will be diluted sufficiently, however, to maintain the free chlorine residual discharged to the river at 0.1 mg/liter or less.

The circulating water will be chlorinated by controlled injection of sodium hypochlorite into the intake water to the condensers. Three of the twelve intakes are chlorinated at a time as a group. The period of chlorination will be no greater than 30 minutes and will be done 3 times per day. The rate of sodium hypochlorite addition is controlled to maintain a 1.0 mg/liter free chlorine residual or less at the condenser outlet. The discharge is diluted with unchlorinated water and the free chlorine residual of the discharge to the river will therefore be less than 0.1 mg/liter.

The service water system will be chlorinated at a frequency not to exceed 3 times a day for periods of not greater than 30 minutes, and not at the same time as the Circulating Water System. The concentration of free chlorine residual at the outlet of the final heat exchanger will be determined and shall be maintained at 1.0 mg/liter or less. Consequently, the concentration at the discharge to the river will be less than 0.1 mg/liter.

2.2.2 SUSPENDED SOLIDS

Objective

To insure that suspended solids released from Non-Radioactive Chemical Waste Disposal System are controlled and will not have an adverse effect on the natural aquatic environment of the receiving waters.

Specification

The average suspended solids concentration in the effluent from the Non-Radioactive Chemical Liquid Waste Disposal System shall not exceed 25 mg/liter on an annual basis.

Monitoring Requirement

A grab sample shall be taken once per day from the collecting basin discharge pipe and analyzed for suspended solids using a method which is acceptable to EPA. The sample shall be taken at the in-line pH monitoring probe in the discharge pipe. Samples shall be taken during periods of actual discharge and only on days when the collecting basin is discharged.

Bases

The non-radioactive chemical liquid waste basin is licensed to operate under permits issued for Industrial Waste Treatment Plants by the State of New Jersey. The suspended solids limitation is that which is required by the permits.

The filtration/gravimetric method is the present means recognized by EPA for the measurement of suspended solids. However, Proposed Water Quality Information, Vol. II, October 1973, by EPA, page 88, states that "accuracy data on actual samples cannot be obtained at this time."

2.2.3 pH

Objective

To insure that the pH of the effluent released from the Non-Radioactive Chemical Waste Disposal System is controlled and will not have an adverse effect on the natural aquatic environment of the receiving waters.

Specification

The pH of the Non-Radioactive Chemical Liquid Waste Disposal System shall be within the range of 6.5 to 8.5 pH units. If this specification is exceeded, the discharge shall be terminated until the pH is corrected to within Specification.

Monitoring Requirement

The pH of effluents released frm the collecting basin shall be monitored continuously at the pump discharge using an in-line pH probe with an accuracy of ± 2%.

Bases

The 6.5 to 8.5 pH limit on the Non-Radioactive Chemical Liquid Waste Disposal System has been set by the New Jersey Department of Environmental Protection and the Delaware River Basin Commission.

The collecting basin discharge valve is controlled by the continuously operating pH probe. The valve will not open unless collecting basin pH is within the range of 6.5 to 8.5 pH units.

No significant change in the background pH of the river water is expected due to the operation of the Salem Station.

2.3 LIMITING CONDITIONS FOR OPERATION

Radioactive Effluents

Objective

To define the limits and conditions for the controlled release of radioactive materials in liquid and gaseous effluents to the environs to ensure that these releases are as low as practicable. These releases should not result in radiation exposures in unrestricted areas greater than a few percent of natural background exposures. The concentrations of radioactive materials in effluents shall be within the limits specified in 10 CFR Part 20.

To ensure that the releases of radioactive material above background to unrestricted areas be as low as practicable, the following design objectives apply:

For liquid wastes:

a. The annual dose above background to the total body or an organ of an individual from all reactors at a site should not exceed 5 mrem in an unrestricted area.

b. The annual total quantity of radioactive materials in liquid waste, excluding tritium and dissolved gases, discharged from each reactor should not exceed 5 Ci.

For gaseous wastes:

c. The annual total quantity of noble gases above background discharged from the site should result in an air dose due to gamma radiation of less than

10 mrad, and an air dose due to beta radiation of less than 20 mrad, at any location near ground level which could be occupied by individuals at or beyond the boundary of the site.

d. The annual total quantity of all radioiodines and radioactive material in particulate forms with half-lives greater than eight days, above background, from all reactors at a site should not result in an annual dose to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem.

e. The annual total quantity of iodine-131 discharged from each reactor at a site should not exceed 1 Ci.

2.3.1 <u>SPECIFICATIONS FOR LIQUID WASTE EFFLUENTS</u>

a. The concentration of radioactive materials released in liquid waste effluents from all reactors at the site shall not exceed the values specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas.

b. The cumulative release of radioactive materials in liquid waste effluents, excluding tritium and dissolved gases, shall not exceed 10 Ci/reactor/calendar quarter.

c. The cumulative release of radioactive materials in liquid waste effluents, excluding tritium and dissolved gases, shall not exceed 20 Ci/reactor in any 12 consecutive months.

d. During release of radioactive wastes, the effluent control monitor shall be set to alarm and to initiate the automatic closure of each waste isolation valve prior to exceeding the limits specified in 2.3.1.a above.

e. The operability of each automatic isolation valve in the liquid radwaste discharge lines shall be demonstrated quarterly.

f. The equipment installed in the liquid radioactive waste system shall be maintained and shall be operated to process radioactive liquid wastes prior to their discharge when the projected cumulative release could exceed 1.25 Ci/reactor/calendar quarter, excluding tritium and dissolved gases.

g. The maximum radioactivity to be contained in any liquid radwaste tank that can be discharged directly to the environs shall not exceed 10 Ci, excluding tritium and dissolved gases.

h. If the cumulative release of radioactive materials in liquid effluents, excluding tritium and dissolved gases, exceeds 2.5 Ci/reactor/calendar quarter, the licensee shall make an investigation to identify the causes for such releases, define and initiate a program of action to reduce such releases to the design objective levels listed in Section 2.3, and report these actions to the NRC in accordance with Specification 5.6.2.3.1.

i. An unplanned or uncontrolled offsite release of radioactive materials in liquid effluents in excess of 0.5 curies requires notification. This notification shall be in accordance with Specification 5.6.2.3.3.

2.3.2 SPECIFICATIONS FOR LIQUID WASTE SAMPLING AND MONITORING

a. Plant records shall be maintained of the radioactive concentration and volume before dilution of liquid waste intended for discharge and the average dilution flow and length of time over which each discharge occurred. Sample analysis results and other reports shall be submitted by Section 5.6.1 of these Specifications. Estimates of the sampling and analytical errors associated with each reported value shall be included.

b. Prior to release of each batch of liquid waste, a sample shall be taken from that batch and analyzed for the concentration of each principle gamma emitter in accordance with Table 2.3-1 to demonstrate compliance with Specification 2.3.1 using the flow rate into which the waste is discharged during the period of discharge.

c. Sampling and analysis of liquid radioactive waste shall be performed in accordance with Table 2.3-1. Prior to taking samples from a monitoring tank, at least two tank volumes shall be recirculated.

d. The radioactivity in liquid wastes shall be continuously monitored and recorded during releases. Whenever these monitors are inoperable for a period not to exceed 72 hours, two independent samples of each tank to be discharged shall be analyzed and two plant personnel shall independently check valving prior to the discharge. If these monitors are inoperable for a period exceeding 72 hours, no release from a liquid waste tank shall be made and any release in progress shall be terminated.

e. The flow rate of liquid radioactive waste shall be continuously measured and recorded during release.

f. All liquid effluent radiation monitors shall be calibrated at least quarterly by means of a radioactive source which has been calibrated to a National Bureau of Standards source. Each monitor shall also have a functional test monthly and an instrument check prior to making a release.

g. The radioactivity in steam generator blowdown shall be continuously monitored and recorded. Whenever these monitors are inoperable, the blowdown flow shall be diverted to the waste management system and the direct release to the environment terminated.

Bases

The release of radioactive materials in liquid waste effluents to unrestricted areas shall not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as practicable in accordance with the requirements of 10 CFR Part 50.36a. These specifications provide reasonable assurance that the resulting annual dose to the total body or any organ of an individual in an unresricted area will not exceed 5 mrem. At the same time, these specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that by using this operational flexibility under unusual operating conditions, and exerting every effort to keep levels of radioactive material in liquid wastes as low as practicable, the annual releases will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience taking into account a combination of variables including defective fuel, primary system leakage, primary to secondary system leakage, steam generator blowdown and the performance of the various waste treatment systems, and are consistent with 10 CFR Part 50.36a.

Specification 2.3.1.a requires the licensee to limit the concentration of radioactive materials in liquid waste effluents released from the site to levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas. This specification provides assurance that no member of the general public will be exposed to liquid containing radioactive materials in excess of limits considered permissible under the Commission's Regulations.

Specifications 2.3.1.b and 2.3.1.c establish the upper limits for the release of radioactive materials in liquid effluents. The intent of these Specifications is to permit the licensee the flexibility of operation to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the levels normally achievable when the plant and the liquid waste treatment systems are functioning as designed. Releases of up to these levels will result in concentrations of radioactive material in liquid waste effluents at small percentages of the limits specified in 10 CFR Part 20.

Consistent with the requirements of 10 CFR Part 50, Appendix A, Design Criterion 64, Specifications 2.3.1.d and 2.3.1.e require operation of suitable equipment to control and monitor the releases of radioactive materials in liquid wastes during any period that these releases are taking place.

Specification 2.3.1.f requires that the licensee maintain and operate the equipment installed in the liquid waste systems to reduce the release of radioactive materials in liquid effluents to as low as practicable consistent with the requirements of 10 CFR Part 50.36a. Normal use and maintenance of installed equipment in the liquid waste system provides reasonable assurance that the quantity released will not exceed the design objective. In order to keep releases of radioactive materials as low as practicable, the specification requires operation of equipment whenever it appears that the projected cumulative discharge rate will exceed one-fourth of this design objective annual quantity during any calendar quarter.

Specification 2.3.1.g restricts the amount of radioactive material that could be inadvertently released to the environment to an amount that will not exceed the Technical Specification limit.

In addition to limiting conditions for operation listed under Specifications 2.3.1.b and 2.3.1.c, the reporting requirements of Specification 2.3.1.h delineate that the licensee shall identify the cause whenever the cumulative release of radioactive materials in liquid waste effluents exceeds one-half the design objective annual quantity during any calendar quarter and describe the proposed program of action to reduce such releases to design objective levels on a timely basis. This report must be filed within 30 days following the calendar quarter in which the release occurred as required by Specification 5.6.2 of these Technical Specifications.

Specification 2.3.1.i provides for reporting spillage or release events which, while below the limits of 10 CFR Part 20, could result in releases higher than the design objectives.

The sampling and monitoring requirements given under Specification 2.3.2 provide assurance that radioactive materials in liquid wastes are properly controlled and monitored in conformance with the requirements of Design Criteria 60 and 64. These requirements provide the data for the licensee and the Commission to evaluate the plant's performance relative to radioactive liquid wastes released to the environment. Reports on the quantities of radioactive materials released in liquid waste effluents are furnished to the Commission according to Section 5.6.1 of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The points of release to the environment to be monitored in Section 2.3.2 include all the monitored release points as provided for in Table 2.3-3.

2.3.3 SPECIFICATIONS FOR GASEOUS WASTE EFFLUENTS

The terms used in these Specifications are as follows:

subscripts v, refers to vent releases

i, refers to individual noble gas nuclide

(Refer to Table 2.4-5 for the noble gas nuclides considered)

Q_T = the total noble gas release rate (Ci/sec)

= $\sum_i Q_i$ sum of the individual noble gas radionuclides determined to be present by isotopic analysis

$\overline{K}$ = the average total body dose factor due to gamma emission (rem/yr per Ci/sec)

$\overline{L}$ = the average skin dose factor due to beta emissions (rad/yr per Ci/sec)

$\overline{M}$ = the average air dose factor due to beta emissions (rad/yr per Ci/sec)

$\overline{N}$ = the average air dose factor due to gamma emissions (rad/yr per Ci/sec)

The values of $\overline{K}$, $\overline{L}$, $\overline{M}$ and $\overline{N}$ are to be determined each time isotopic analysis is required as delineated in Specification 2.3.4. Determine the following using the results of the noble gas radionuclide analysis:

$\overline{K} = (1/Q_T) \sum_i Q_i K_i$

$\overline{L} = (1/Q_T) \sum_i Q_i L_i$

$\overline{M} = (1/Q_T) \sum_i Q_i M_i$

$\overline{N} = (1/Q_T) \sum_i Q_i N_i$

Where the values of K_i, L_i, M_i and N_i are provided in Table 2.3-5, and are site dependent gamma and beta dose factors:

Q = the measured release rate of the radioiodines and radioactive materials in particulate forms with half-lives greater than eight days.

a. (1) The release rate limit of noble gases from the site shall be such that

$$2.0 \left[Q_{Tv} \bar{K}_v \right] \leq 1$$

and

$$0.33 \left[Q_{Tv} \left(\bar{L}_v + 1.1 \bar{N}_v \right) \right] \leq 1$$

(2) The release rate limit of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days, released to the environs as part of the gaseous wastes from the site shall be such that:

$$1.5 \times 10^5 \, Q_v \leq 1$$

b. (1) The average release rate of noble gases from the site during any calendar quarter shall be such that:

$$13 \left[Q_{Tv} \bar{N}_v \right] \leq 1$$

and

$$6.3 \left[Q_{Tv} \bar{M}_v \right] \leq 1$$

(2) The average release rate of noble gases from the site during any 12 consecutive months shall be:

$$25 \left[Q_{Tv} \bar{N}_v \right] \leq 1$$

and

$$13 \left[Q_{Tv} \bar{M}_v \right] \leq 1$$

(3) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter shall be such that

$$13 \left[1.5 \times 10^5 Q_v \right] \leq 1$$

(4) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any period of 12 consecutive months shall be such that:

$$25 \left[1.5 \times 10^5 Q_v \right] \leq 1$$

(5) The amount of iodine-131 released during any calendar quarter shall not exceed 2 Ci/reactor.

(6) The amount of iodine-131 released during any period of 12 consecutive months shall not exceed 4 Ci/reactor.

c. Should any of the conditions of 2.3.3.c(1), (2) or (3) listed below exist, the licensee shall make an investigation to identify the causes of the release rates, define and initiate a program of action to reduce the release rates to design objective levels listed in Section 2.3 and report these actions to the NRC within 30 days from the end of the quarter during which the releases occurred.

(1) If the average release rate of noble gases from the site during any calendar quarter is such that:

$$50 \left[Q_{Tv}\bar{N}_v \right] \quad > 1$$

or

$$25 \left[Q_{Tv}\bar{M}_v \right] \quad > 1$$

(2) If the average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter is such that:

$$50 \left[1.5 \times 10^5 Q_v \right] \quad > 1$$

(3) If the amount of iodine-131 released during any calendar quarter is greater than 0.5 Ci/reactor.

d. During the release of gaseous wastes from the primary system waste gas holdup system the effluent monitors listed in Table 2.3-4 shall be operating and set to alarm and to initiate the automatic closure of the waste gas discharge valve prior to exceeding the limits specified in 2.3.3.a above. The operability of each automatic isolation valve shall be demonstrated quarterly.

e. The maximum activity to be contained in one waste gas storage tank shall not exceed 41,000 curies (considered as Xe-133).

f. An unplanned or uncontrolled offsite release of radioactive materials in gaseous effluents in excess of 5 curies of noble gas or 0.02 curie of radioiodine in gaseous form requires notification. This notification shall be in accordance with Section 5.6.2.3.3.

2.3.4 SPECIFICATIONS FOR GASEOUS WASTE SAMPLING AND MONITORING

a. Plant records shall be maintained and reports of the sampling and analyses results shall be submitted in accordance with Section 5.6 of these Specifications. Estimates of the sampling and analytical error associated with each reported value should be included.

b. Gaseous releases to the environment (Table 2.3-4), except from the turbine building ventilation exhaust and as noted in Specification 2.3.4.c, shall be continuously monitored for gross radioactivity and the flow continuously measured and recorded. Whenever these monitors are inoperable, grab samples shall be taken and analyzed daily for gross radioactivity. If these monitors are inoperable for more than seven days, these releases shall be terminated.

c. During the release of gaseous wastes from the primary system waste gas holdup system, the gross activity monitor, the iodine collection device, and the particulate collection device shall be operating.

d. All waste gas effluent monitors shall be calibrated at least quarterly by means of a known radioactive source which has been calibrated to a National Bureau of Standards source. Each monitor shall have a functional test at least monthly and instrument check at least daily.

e. Sampling and analysis of radioactive material in gaseous waste, including particulate forms and radioiodines shall be performed in accordance with Table 2.3-2.

Bases

The release of radioactive materials in gaseous waste effluents to unrestricted areas shall not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as practicable in accordance with the requirements of 10 CFR Part 50.36a. These specifications provide reasonable assurance that the resulting annual air dose from the site due to gamma radiation will not exceed 10 mrad, and an annual air dose from the site due to beta radiation will not exceed 20 mrad from noble gases, that no individual in an unrestricted area will receive an annual dose greater than 15 mrem from fission product noble gases, and that the annual dose to any organ of an indivdual from radio-iodines and radioactive material in particulate form with half-lives greater than eight days will not exceed 15 mrem per site.

At the same time these specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the pubic is provided with a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. Even with this operational flexibility under unusual operating conditions, if the licensee exerts every effort to keep levels of radioactive material in gaseous waste effluents as low as practicable, the annual releases will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience taking into account a combination of system variables including defective fuel, primary system leakage, primary to secondary system leakage, steam generator blowdown and the performance of the various waste treatment systems.

Specification 2.3.3.a(1) limits the release rate of noble gases from the site so that the corresponding annual gamma and beta dose rate above background to an individual in an unrestricted area will not exceed 500 mrem to the total body or 3000 mrem to the skin in compliance with the limits of 10 CFR Part 20.

For Specification 2.3.3.a(1), gamma and beta dose factors for the individual noble gas radionuclides have been calculated for the plant gaseous release points and are provided in Table 2.3-5. The expressions used to calculate these dose factors are based on dose models derived in Section 7 of <u>Meteorology and Atomic Energy</u>-1968 and model techniques provided in Draft Regulatory Guide 1.AA.

Dose calculations have been made to determine the site boundary location with the highest anticipated dose rate from noble gases using onsite meteorological data and the dose expressions provided in Draft Regulatory Guide 1.AA. The dose expression considers the release point location, building wake effects, and the physical characteristics of the radionuclides.

The offsite location with the highest anticipated annual dose from released noble gases is 1270 meters in the North direction.

The release rate Specifications for a radioiodine and radioactive material in particulate form with half-lives greater than eight days are dependent on existing radionuclide pathways to man. The pathways which were examined for these Specifications are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, and 3) deposition onto grassy areas where milch animals graze with consumption of the milk by man. Methods for estimating doses to the thyroid via these pathways are described in Draft Regulatory Guide 1.AA. The offsite location with the highest anticipated thyroid dose rate from radioiodines and radioactive material in particulate form with half-lives greater than eight days was determined using onsite meteorological data and the expressions described in Draft Regulatory Guide 1.AA.

Specification 2.3.3.a(2) limits the release rate of radioiodines and radioactive material in particulate form with half-lives greater than eight days so that the corresponding annual thyroid dose via the most restrictive pathway is less than 1500 mrem.

For radioiodines and radioactive material in particulate form with half-lives greater than eight days, the most restrictive location is a dariy farm located 6600 meters in the NW direction (vent X/Q = 1.1×10^{-7} sec/m^3).

Specification 2.3.3.b establishes upper offsite levels for the releases of noble gases and radioiodines and radioactive material in particulate form with half-lives greater than eight days at twice the design objective annual quantity during any calendar quarter, or four times the design objective annual quantity during any period of 12 consecutive months. In addition to the limiting conditions for operation of Specifications 2.3.3.a and 2.3.3.b, the reporting requirements of 2.3.3.c provide that the cause shall be identified whenever the release of gaseous effluents exceeds one-half the design objective annual quantity during any calendar quarter and that the proposed program of action to reduce such release rates to the design objectives shall be described.

Specification 2.3.3.d requires that suitable equipment to monitor and control the radioactive gaseous releases is operating during any period these releases are taking place.

Specification 2.3.3.e limits the maximum quantity of radioactive gas that can be contained in a waste gas storage tank. The calculation of this quantity should assume instantaneous ground release, a X/Q based 5 percent meteorology, the average gross energy is 0.19 Mev per disintegration (considering Xe-133 to be the principal emitter) and exposure occurring at the minimum site boundary radius using a semi-infinite cloud model. The calculated quantity will limit the offsite dose above background to 0.5 rem or less, consistent with Commission guidelines.

Specification 2.3.3.f provides for reporting release events which, while below the limits of 10 CFR Part 20, could result in releases higher than the design objectives.

The sampling and monitoring requirements given under Specification 2.3.4 provide assurance that radioactive materials released in gaseous waste effluents are properly controlled and monitored in conformance with the requirements of Design Criteria 60 and 64. These requirements provide the data for the licensee and the Commission to evaluate the plant's performance relative to radioactive waste effluents released to the environment. Reports on the quantities of radioactive materials released in gaseous effluents are furnished to the Commission on the basis of Section 5.6.1 of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The points of release to the environment to be monitored in Section 2.3.4 include all the monitored release points as provided for in Table 2.3-4.

Specification 2.3.4.b excludes monitoring the turbine building ventilation exhaust since this release is expected to be a negligible release point. Many PWR reactors do not have turbine building enclosures. To be consistent in this requirement for all PWR reactors, the monitoring of gaseous releases from turbine buildings is not required.

2.3.5 SPECIFICATIONS FOR SOLID WASTE HANDLING AND DISPOSAL

a. Measurements shall be made to determine or estimate the total curie quantity and principal radionuclide composition of all radioactive solid waste shipped offsite.

b. Reports of the radioactive solid waste shipments, volumes, principal radionuclides, and total curie quantity, shall be submitted in accordance with Section 5.6.1.

Bases

The requirements for solid radioactive waste handling and disposal given under Specification 2.3.5 provide assurance that solid radioactive materials stored at the plant and shipped offsite are packaged in conformance with 10 CFR Part 20, 10 CFR Part 71, and 49 CFR Parts 170-178.

TABLE 2.3-1

RADIOACTIVE LIQUID SAMPLING AND ANALYSIS

Liquid Source	Sampling Frequency and Analysis	Type of Activity Analysis	Detectable Concentrations (μCi/ml)[a]
A. Monitor Tank Releases	Each Batch	Principal Gamma Emitters	5×10^{-7} [b]
	One Batch/Month	Dissolved Gases [f]	10^{-5}
	Weekly Composite[c]	Ba-La-140, I-131	10^{-6}
	Monthly Composite[c]		
		H-3	10^{-5}
		Gross α	10^{-7}
	Quarterly Composite[c]	Sr-89, Sr-90	5×10^{-8}
B. Primary Coolant	Weekly[d]	I-131, I-133	10^{-6}
C. Steam Generator Blowdown	Weekly[e]	Principal Gamma Emitters	5×10^{-7} [b]
		Ba-La-140, I-131	10^{-6}
	One Sample/Month	Dissolved Gases [f]	10^{-5}
	Monthly Composite[e]		
		H-3	10^{-5}
		Gross α	10^{-7}
	Quarterly Composite[e]	Sr-89, Sr-90	5×10^{-8}

[a] The detectability limits for activity analysis are based on the technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable, and when nuclides are measured below the stated limits, they should also be reported.

[b] For certain mixtures of gamma emitters, it may not be possible to measure radionuclides in concentrations near their sensitivity limits when other nuclides are present in the sample in much greater concentrations. Under these circumstances, it will be more appropriate to calculate the concentrations of such radionuclides using measured ratios with those radionuclides which are routinely identified and measured.

[c] A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged.

[d] The power level and cleanup or purification flow rate at the sample time shall also be reported.

[e] To be representative of the average quantities and concentrations of radioactive materials in liquid effluents, samples should be collected in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite should be thoroughly mixed in order for the composite sample to be representative of the average effluent release.

[f] For dissolved noble gases in water, assume a MPC of 4×10^{-5} μCi/ml of water.

TABLE 2.3-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS

Gaseous Source	Sampling Frequency and Analysis	Type of Activity Analysis	Detectable Concentrations ($\mu Ci/ml$)[a]
A. Waste Gas Decay Tank Releases	Each Tank to be Released	Principal Gamma Emitters	10^{-4} [b]
		H-3	10^{-6}
B. Containment Purge Releases	Each Purge	Principal Gamma Emitters	10^{-4} [c]
		H-3	10^{-6}
C. Condenser Air Ejector	Monthly	Principal Gamma Emitters	10^{-4} [b, c]
		H-3	10^{-6}
D. Environmental Release Points	Monthly (Gas Samples)	Principal Gamma Emitters	10^{-4} [b, c]
		H-3	10^{-6}
	Weekly (Charcoal Sample)	I-131	10^{-12}
	Monthly (Charcoal Sample)	I-133, I-135	10^{-10}
	Weekly (Particulates)[d]	Principal Gamma Emitters (Ba-La-140, I-131 and others)	10^{-11}
	Monthly Composite[d] (Particulates)		
		Gross α	10^{-11}
	Quarterly Composite[d] (Particulates)	Sr-89, Sr-90	10^{-11}

[a] The above detectability limits for activity analysis are based on technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable, and when nuclides are measured below the stated limits, they should also be reported.

[b] For certain mixtures of gamma emitters, it may not be possible to measure radionuclides at levels near their sensitivity limits when other nuclides are present in the sample at much higher levels. Under these circumstances, it will be more appropriate to calculate the levels of such radionuclides using observed ratios with those radionuclides which are measurable.

[c] Analyses shall also be performed following each refueling, startup, or similar operational occurrence which could alter the mixture of radionuclides.

[d] To be representative of the average quantities and concentrations of radioactive materials in particulate form released in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent stream.

TABLE 2.3-3

SALEM STATION LIQUID WASTE SYSTEM
LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLERS REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Radiation Alarm	Auto Control to Isolation Valve	Gross Activity Continuous Monitor	Grab Sample Station	Measurement						High Liquid Level Alarm
					Gross Activity	I	Dissolved Gases	Alpha	H-3	Isotopic Analysis	
Miscellaneous Monitor Tanks[b]				X		X	X	X	X	X	X
Chemical Drain Tank				X		X	X	X	X	X	X
Laundry and Hot Shower Tanks[a]				X		X	X	X	X	X	X
Primary Coolant System				X		X					
Liquid Radwaste Discharge Pipe	X	X	X		X						
Steam Generator Blowdown System	X		X	X	X	X	X	X	X	X	
Outdoor Storage Tanks (potentially radioactive) Primary Water Storage Tank Refueling Water Storage Tank				X[c]	X					X	X
Component Cooling Systems	X		X		X						
Turbine Building Sumps (Floor Drains)				X[d]	X					X	X

[a] The contents of the Laundry and Hot Shower Tanks are sampled, analyzed, and then filtered prior to release through the liquid radwaste discharge pipe.

[b] Includes Waste Monitor Tanks, Waste Monitor Holdup Tank, CVCS Monitor Tanks.

[c] Grab sample to be taken and analyzed each 8 hours whenever tank leakage exists.

[d] Grab sample to be taken and analyzed each 8 hours whenever the gross activity in the secondary coolant system exceeds 10^{-5} uCi/ml (Except H-3).

TABLE 2.3-4

SALEM STATION GASEOUS WASTE SYSTEM

LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLERS REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Radiation Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Measurement: Noble Gas	I	Particulate	H-3	Alpha
Waste Gas Decay Tanks				X	X	X	X	X	X
Condenser Air Removal System	X		X	X	X	X	X	X	X
Plant Vent	X		X[b]	X	X	X	X	X	X
Building Ventilation Systems									
Reactor Containment Building (whenever there is flow)	X	X	X[d]	X	X	X	X	X	X
Auxiliary Building and Radwaste Area[a]				X[c]	X	X	X	X	X
Fuel Handling & Storage Building[a]				X[c]	X	X	X	X	X
Turbine Gland Seal Condenser[a]				X[c]	X	X	X	X	X
Waste Gas Discharge Line	X	X	X						

a Since these process streams or building ventilation systems are routed to the plant vent, the need for a continuous monitor at the individual discharge point to the main exhaust duct is eliminated. One continuous monitor at the final release point is sufficient.

b Continuously monitored. Also includes continuous iodine, noble gas and particulate monitors which are in service during waste gas decay tank releases and containment purging operations.

c Grab sample stations from which monthly gas samples (Table 2.3-2) are to be taken. Also, grab samples should be taken and measured to determine the process stream or building ventilation system source whenever an unexplained increase is indicated by the plant vent sampler-monitors.

d Includes continuous noble gas monitor which monitors this location at all times other than waste decay tank releases and containment purging operations.

TABLE 2.3-5

GAMMA AND BETA DOSE FACTORS FOR

SALEM, UNITS 1 AND 2

Noble Gas Radionuclide	K_{iv} Total Body $\frac{(rem/yr)}{(Ci/sec)}$	L_{iv} Skin $\frac{(rem/yr)}{(Ci/sec)}$	M_{iv} Beta Air $\frac{(rad/yr)}{(Ci/sec)}$	N_{iv} Gamma Air $\frac{(rad/yr)}{(Ci/sec)}$
	$X/Q = 1.2 \times 10^{-6}$ sec/m^3 @ 1270 Meters, North — Dose Factors for Vent			
Kr-83m	8.6×10^{-5}	0	0.043	0.35
Kr-85m	0.97	1.8	1.0	2.4
Kr-85	0.012	1.6	0.012	2.3
Kr-87	3.0	12	3.1	12
Kr-88	7.4	2.8	7.8	3.5
Kr-89	1.3	12	1.4	13
Xe-131m	0.34	0.57	0.43	1.3
Xe-133m	0.26	1.2	0.36	1.8
Xe-133	0.31	0.37	0.38	1.3
Xe-135m	1.2	0.85	1.3	0.89
Xe-135	1.4	2.2	1.5	2.9
Xe-137	0.18	15	0.19	15
Xe-138	2.9	5.0	3.0	5.7

3.0 ENVIRONMENTAL SURVEILLANCE

The objective of the Environmental Surveillance Program is to determine the effects of plant operation on the ecosystem. The program is designed to accomplish this objective through periodic sampling and analyses of key parameters in the vicinity of the Salem Nuclear Generating Station. The key parameters selected for monitoring are those that could reasonably be expected to be affected by plant operation.

Comparison of surveillance data with preoperational "base-line" levels will reveal changes and trends that could be attributed to plant operation. As operating experience and surveillance data are obtained, the Environmental Surveillance Program will be modified to reduce or eliminate surveillance of those parameters that have not been significantly affected by plant operation.

3.1 NON RADIOLOGICAL SURVEILLANCE

3.1.1 ABIOTIC

3.1.1.1 Chlorine

Objective

To determine the concentration of free and total residual chlorine in the station effluent water in an effort to maintain an optimum chlorination program for prevention of heat exchanger fouling while minimizing the environmental impact on the receiving waters.

Specification

Grab samples shall be taken weekly (weather permitting) during a chlorination cycle and analyzed for free and total residual chlorine. The samples shall be taken in the vicinity of the circulating water discharge, from the station intake water and from a point that is outside and downstream of the discharge water mixing zone.

Reporting Requirement

In the event the analysis of the sample taken from the point outside and downstream of the discharge water mixing zone indicates that the total residual chlorine at the point exceeds the ambient total residual chlorine level in the river by 0.1 mg/liter, a report shall be made in accordance with Specification 5.6.2.

Bases

This monitoring program will provide data on the chlorine demand of the receiving water as well as the concentration of fouling organisms present. These parameters are subject to seasonal variation and will aid in maintaining an optimum chlorination program for prevention of heat exchanger fouling.

3.1.1.2 Dissolved Gases

Objective

To ascertain that the dissolved oxygen level is not depressed to the extent that it may be harmful to the indigenous population of the receiving waters as a result of station operation.

Specification

The dissolved oxygen levels shall be monitored once per month (weather permitting) utilizing the Winkler titration method in accordance with APHA specification 218B 13th edition. Grab samples shall be taken at the intake structure, the outfall of the discharge and a point outside and downstream of the mixing zone. The samples shall be taken at depths of 10 feet, 8 feet, and 18 feet, respectively. The standard deviation is expected to be within ± 0.1 mg/l.

Reporting Requirement

If dissolved oxygen level is found to be less than 6 mg/l at the discharge, a comparison study of the intake, discharge and downstream dissolved oxygen levels shall be conducted to determine if the oxygen depression has been caused by station operation. If it is so determined, a report shall be made in accordance with Specification 5.6.2.

Bases

Monthly analyses of dissolved oxygen will aid in differentiating between normal seasonal fluctuations and changes due to station operation.

The 6 mg/liter limitation is required by the Water Quality Certificate issued by the Delaware River Basin Commission.

3.1.1.3 Suspended Solids

Objective

To determine the effect of plant operation on suspended solids in the receiving waters.

Specification

Suspended solids shall be monitored once per month (weather permitting). Grab samples shall be taken at the intake structure, the outfall of the discharge and at a point outside and downstream of the mixing zone. The samples shall be taken at depths of 10 feet, 8 feet, and 18 feet, respectively. These samples shall be analyzed for suspended solids by means of a method acceptable to EPA. Dissolved solids shall not be monitored.

Reporting Requirement

Reporting levels shall be developed after the initial phases of plant operation. Post-operational data will be related to preoperational data to yield norms from which report levels will be established. An initial evaluation of the reporting levels will be provided in the first annual report.

Bases

Monthly analyses of suspended solids will aid in differentiating between normal seasonal fluctuations, changes due to tide, wind and current, and those due to station operation.

The filtration/gravimetric method is the present means recognized by EPA for the measurement of suspended solids. However, Proposed Water Quality Information, Vol. II, October 1973, EPA, page 88 states that "accuracy data on actual samples cannot be obtained at this time."

Dissolved solids will not be measured since none of the applicable Regulatory agencies have issued guidelines for this parameter for the Delaware River zone in the vicinity of the Station site.

3.1.1.4 Other Chemicals

Objective

To determine the effects of plant operation on the quality of the receiving waters.

Specification

Grab samples shall be taken once per month (weather permitting) and analyzed for the parameters listed in Table 3.1-1. The samples shall be taken at the intake structure, the outfall of the discharge and at a point outside and downstream of the mixing zone. The samples shall be taken at depths of 10 feet, 8 feet, and 18 feet, respectively. These samples shall be analyzed for the parameters listed in Table 3.1-1 by a method acceptable to EPA.

Reporting Requirement

Reporting levels will be developed after the initial phases of plant operation. Post-operational data will be related to preoperational data to yield norms from which report levels will be established.

Bases

This monitoring program will serve to determine the effect of station operation on the quality of the receiving water. An evaluation of the program, after six months of full power operation, will be performed and those parameters which can be shown to be not significantly affected by station operation will be eliminated from the monitoring program subsequent to NRC staff review and approval. This program is in conformance with NPDES requirements.

The utilization of tests prescribed by EPA will insure the employment of current, state-of-the-art methods and accuracies.

3.1.1.5 Chemical Releases

Objective

To insure that chemical releases from the plant are identified by compound and quantity.

Specification

A physical inventory of identifiable chemicals, excluding spent laboratory reagents and condenser tube corrosion products, discharged directly to the river shall be maintained and submitted as part of the annual report.

Monitoring Specifications

The physical inventory of identifiable chemicals, excluding spent laboratory reagents, discharged directly to the river can be in the form of an estimation of discharge quantity by purchase order and inventory differential.

Bases

Documenting the discharge of chemicals shall allow comparison with estimated releases given in Table 3.1-3. If chemical usage differs significantly (a factor of 3) from that indicated in the table (3.1-3) or if the need for other chemicals is determined, then a brief assessment of environmental impact shall be made.

Spent chemical reagents from the chemistry laboratories are not to be included in the reporting requirement because of their small quantities and insignificant concentrations in liquids released. Condenser tube corrosion products are not a part of this chemical inventory.

3.1.1.6 Meteorological Monitoring

Objective

The objective of meteorological monitoring is to adequately measure and document meteorological conditions at the site.

Specification

The meteorological monitoring system shall conform to the recommendations in Regulatory Guide 1.23, Onsite Meteorological Programs, dated February 17, 1972, and consist of instruments to measure wind speed and direction, air temperature and vertical air temperature differences at heights above ground that are representative of atmospheric conditions that exist at all gaseous effluent release points, as described in Section 3.3.3.4 of the Safety Technical Specifications.

Reporting Requirements

Meteorological data shall be summarized and reported in a format consistent with the recommendations of Regulatory Guide 1.21, Measuring, Evaluating, and Reporting

Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, Revision 1, dated June, 1974 and Regulatory Guide 1.23, Onsite Meteorological Programs, dated February 17, 1972, and observations in a form consistent with National Weather Service procedures. Summaries of data and observations shall be available to the U. S. Nuclear Regulatory Commission upon request. If the outage time of any of the required meteorological instruments exceeds seven consecutive days, the total outage time and dates of outage, the cause of the outage, and the instrument(s) involved shall be reported within 30 days of the initial time of the outage to the Director, Office of Nuclear Reactor Regulation. Any modifications to the meteorological monitoring program as described above shall have the written approval of the Director, Office of Nuclear Reactor Regulation, prior to initiation of the modification.

Bases

The collection of meteorological data at the plant site will provide information which may be used to develop atmospheric diffusion parameters to estimate potential radiation doses to the public resulting from actual routine or accidental releases of radioactive materials to the atmosphere. A meteorological data collection program as described above is necessary to meet the requirements of subparagraph 50.36a (a) (2) of 10 CFR Part 50, Appendix E to 10 CFR Part 50, and 10 CFR Part 51.

3.1.2 BIOTIC

3.1.2.1 General Ecological Survey

The primary objective of this survey is to determine the effect of plant operation on the ecology and environment of the Delaware River Estuary and environs. The preoperational biological monitoring was initiated in 1968 and monitoring will be continued for 5 years after Unit No. 2 becomes operational. The program shall be discontinued only after approval by NRC staff. These studies will serve as a basis for assessment of the effects of plant operation on the ecology.

Study Plan

The study area includes the Delaware River Estuary and some tributaries within an approximate 10-mile radius of the station. The biological parameters monitored are listed in Table 3.1-2 and the general sampling locations are shown in Figure 3.1-1.

Physiochemical parameters will be monitored in the various sampling programs and will typically include dissolved oxygen, temperature, salinity, pH, and water transparency.

Specification

1. Aquatic Studies

   a. Phytoplankton

      Replicate phytoplankton samples shall be taken with a Van Dorn sampler. Surface and near bottom samples shall be collected biweekly (weather permitting) within the study area illustrated in Figure 3.1-1.

These samples shall be examined quantitatively for chlorophyll a using the spectrophotometric method of Lorenzen.[1] The standard error of cholorophyll a analysis at the 5 ug level is ± 0.18 ug for the mean of two determinations.[2,3]

Generic identifications and enumerations shall be conducted. Distributions exhibited by the dominant taxa shall be emphasized.

A productivity study shall be performed bimonthly (weather permitting) at appropriate control and thermally affected stations along with chlorophyll a analysis. Dissolved oxygen concentrations shall be determined. The azide modification of the Winkler method[4] shall be used to determine dissolved oxygen levels in both light and dark bottles as well as in control samples. The precision of oxygen determination based on replicate samples is ± 0.01 mg oxygen/liter.

b. Ichthyoplankton

Ichthyoplankton samples shall be collected monthly (weather permitting) within the study area illustrated in Figure 3.1-1. Surface and near-bottom samples shall be collected at all stations. In addition, mid-water samples shall be collected at offshore stations where depth exceeds 20 ft. Replicate samples shall be taken at selected stations.

All samples shall be collected with metered 1/2 m plankton nets (0.5-mm mesh) towed at a constant speed sufficient to keep the upper net at the surface. Bottom nets will be equipped with depressors to facilitate sampling near bottom.

Ichthyoplankton shall be identified and enumerated. Identification will be to species or to the lowest taxonomic level which specimen condition permits. Results shall be expressed as number of organisms per cubic meter.

c. Zooplankton - Microplankton

Microplankton samples shall be collected 2 ft below the surface and just above the bottom with a plankton pump filter system using No. 20 nets (0.08 mm mesh) within the study area illustrated in Figure 3.1-1. Additionally, integrated (surface to bottom) samples will be taken near the intake and at two stations on a transect extending 1.5 miles offshore. The sampler design is modified from one described by Icanberry and Richardson.[5] It consists of a reinforced suction hose coupled to an air-tight Plexiglas cyliner. Water is pulled through the hose and cylinder by a portable gasoline centrifugal pump. The volume of water filtered shall be measured inboard of the cylinder by a flowmeter.

Samples shall be collected monthly (weather permitting) at all stations. Additional samples shall be collected at 4-hour intervals over a 12-hour period once per month (weather permitting) on the three-station transect extending west from the plant site.

Replicate subsamples shall be counted in a counting cell with a compound microscope. Results will be presented as numbers of organisms per cubic meter.

Most zooplankters shall be identified to species. Poorly preserved or immature specimens shall be identified to the lowest taxonomic level which their condition permits.

Although all zooplankton organisms collected shall be identified, the emphasis in reports shall be on dominant species.

d. <u>Zooplankton - Macroplankton</u>

Macroplankton analyses shall be performed on samples taken under Specification 3.1.2.1.1.b (Ichthyoplankton), and in accordance with Specification 3.1.2.1.1.c (Zooplankton - Microplankton).

e. <u>Benthos</u>

Benthos collections shall be made monthly to bimonthly (weather permitting) within the study area illustrated in Figure 3.1-1. Based on variability observed in preoperational samples, three replicates shall be taken at each station. Samples shall be collected with a

Ponar grab sampler which samples an area 0.05 m^2 to a depth of approximately 15 cm. Most benthic organisms shall be identified to species. For specimens damaged in sampling, identification shall be to the lowest possible taxonomic level. The organisms shall be counted, dried, and weighed.

f. Blue Crab

Commercial crabbers shall be censused throughout the crabbing season (usually May through November) by means of daily questionnaires which ask data on the number of pots checked, number of bushels of hard crab, and number of individual moulting crab taken. The numbers of soft crab, mating crab, and egg-bearing female crab observed in pot catches are also noted. Additionally data on blue crab shall be collected monthly by interviewing and accompanying selected crabbers during their operations. Crab are also collected in the course of the fisheries sampling programs.

g. Fish

Fishes will be sampled by seine, trawl, and gill net within the area illustrated in Figure 3.1-1. Sites shall be sampled on a biweekly to quarterly schedule (weather permitting) throughout the year. Appropriate stations and zones shall be sampled during daylight and at night. Trawl hauls in the river zones shall be of 10-minute duration with a 16-ft semi-balloon otter trawl and in the creeks they shall be of 5-minute duration with a 9-ft semi-ballon otter trawl. Trawl hauls shall be made at a uniform speed, traveling with the tide. Seine collections shall be made parallel to the shore line. Seines shall be used in combination and may include a 1/4-inch mesh, 25-ft bag seine; a 1/4-inch mesh, 10-ft flat seine; a 1/8-inch mesh, 10-ft flat seine; and a 1/2-inch mesh, 225-ft seine. Fishes shall be identified and enumerated by species, and representative subsamples shall be measured for length.

Gill nets shall be fished in the spring to sample populations of anadromous fishes. Gill net gangs of stretched mesh sizes 5-1/2 inches and 3-1/8 inches shall be drifted after being set perpendicular to the current. Specimens shall be identified to species, sexed, measured, and weighed.

2. Terrestrial Studies

Studies of the Terrestrial Environment shall include:

1. Monitoring of nesting by the diamondback terrapin on Sunken Ship Cove Beach and in regions outside the thermal plume.
2. A monthly (weather permitting) bird survey in the area of Artificial Island.
3. Monitoring of occurrence and nesting of the osprey and southern bald eagle within a general 5-mile radius of the station.

Reporting Requirement

Reporting levels shall be developed after one year of full power operation of Unit 2. Post-operational data will be related to preoperational norms from which report levels will be established.

Bases

All biological parameters sampled will provide background data for determining the environmental effects of station operation. Results of the operational studies will be compared with preoperational studies by statistical methods. The various sampling locations were selected on the basis of their representative distribution throughout the region. As the data from these sites are analyzed, it will be determined whether additional sites are needed or old sites can be eliminated. The frequency of sampling has been established in much the same manner.

3.1.2.2 Impingement of Organisms

Objective

The principal objectives of the impingement study are to: (1) determine the species composition, and (2) quantify the numbers of fishes and other organisms which become impinged on the circulating and service water intake screens.

Specification

Within 10 days after achieving commercial operation at 100% electrical output of 1090 MWe the impingement monitoring described in this specification shall be initiated.

Impinged organisms shall be sampled for three 24-hour periods per week. The total weight of the 24-hour sample shall be determined. All fishes and other organisms shall be identified by species. If the sample is greater than 100 lbs., two random subsamples of 50 lbs. each shall be taken from each sampling period. All specimens in the two subsamples shall be identified by species and each series weighed and used to compute total weight per species for each sample. All fish of a given species shall be used to compute modal length, maximum length, and modal weight. If greater than 100 fish are collected during a sample, a random subsample of at least 100 fish shall be used for the above listed measurements. Estimates of the total number of each species impinged per 24-hour sampling period shall be computed. Number of circulation pumps in operation during sampling shall be recorded.

After one year of full power operation of Unit 1, the impingement data shall be analyzed. Suggested changes in sampling frequency shall be submitted for review and approval by the staff prior to implementation.

Reporting Requirement

Monthly results from this study shall be submitted to the NRC within 20 days after the end of the month. Report of each sample period shall contain the following information: the data of sample, the species collected, the number impinged in 24 hours for each species, the modal length for each species, the maximum length for each species, the modal weight for each species, and the number of pumps operating during sample collection. A summary of the impingement study shall be included in the annual environmental report.

Bases

This survey and subsequent data analyses will aid in verification of the effectiveness of the intake design in minimizing impingement of organisms.

3.1.2.3 Entrainment of Plankton Organisms

Objective

The objective of the entrainment study is to determine the effects of operation of the Circulating Water System on planktonic organisms.

Specifications

In order to obtain estimates of numbers of organisms passing through the circulating water system, a sampling program shall be implemented. The first phase of the program consists of sampling the plankton during the initial start-up of the Circulating Water System. Ambient temperature water shall be circulated through the various components of the system.

Since there will be no heat added to the water at this time, mortalities due solely to mechanical damage and pressure may be assessed. During the second phase, sampling shall be continued with varying heat load rejection to the Circulating Water System under different station operating conditions. This will provide an estimate of organism mortality related to the various operating modes of the Circulating Water System.

Intake samples shall be taken immediately before the intake structure; discharge samples shall be taken through sampling ports. During the testing and start-up phases of plant operation, sampling shall be regulated by the station operating schedule. Within 10 days after achieving commercial operation at 100% electrical output of 1090 MWe the sampling described below shall be done monthly.

Samples from the same water mass shall be obtained from the intake and discharge by coordinating their collection with circulating water passage time. Whenever possible entrainment sampling will be coordinated with the collection of river samples on the transect extending westward from the site.

Physiochemical parameters to be monitored at sampling will, whenever possible, include water temperature, dissolved oxygen, salinity, and pH.

Phytoplankton Studies

Replicate samples shall be taken with a Van Dorn sampler. Surface, mid-depth, and bottom samples shall be collected monthly during a 24-hour period at 4-hour intervals from in front of the intake trash racks. Discharge samples shall be collected through sampling ports.

Phytoplankton samples shall be analyzed quantitatively for chlorophyll a using the techniques presented in Specification 3.1.2.1.1.a. Additionally systematic studies shall be done to determine dominant taxa.

Microzooplankton Studies

Surface, near-bottom and integrated (near-bottom to surface) samples shall be collected monthly during a 24-hour period at 4-hour intervals. These samples shall be taken at the intake with the filter-pump system described in Specification 3.1.2.1.1.c and at the discharge through sampling ports. Samples will be immediately stained with neutral red to enable separation of live and dead zooplankton. A dead organism count and specimen identification to the lowest possible taxon shall be done in the laboratory. An additional discharge sample shall be collected, stained with neutral red, and maintained in a water bath at ambient river temperature. This sample shall be periodically monitored over a 12-hour period to determine latent mortality.

Ichthyoplankton and Macrozooplankton Studies

Ichthyoplankton shall be sampled at the intake and discharge biweekly during the appropriate season as determined by preoperational studies. Samples shall be taken over 24-hour periods at 4-hour intervals.

Replicate samples shall be collected at the intake with a high capacity pump sampler if present experimental efforts prove its suitability for sampling. If this pump sampler does not prove acceptable, metered plankton nets will be fished at surface, mid-depth, and near bottom in front of the intake structure. Samples of discharge water shall be taken through sampling ports.

Specimens collected shall be identified to the lowest possible taxonomic level, and densitites shall be calculated. Immediate mortality shall be determined for intake and discharge samples based on the following criteria:

Live: Swimming vigorously, no apparent orientation problems, behavior normal.

Stunned: Swimming erratically, struggling and swimming on side, some twitching but motile.

Dead: No vital life signs, body or opercular movement, no response to gentle probing.

Specimens determined to be alive or stunned shall be held separately for 12- to 24-hour periods at ambient river temperature to determine latent mortality.

After one year of full power operation of Unit No. 1, the entrainment data should be analyzed. Suggested changes in sampling frequency shall be submitted to NRC for review and approval by the staff prior to implementation.

Reporting Requirement

Report levels shall be developed from the data collected during the first year of operation.

Bases

This study and subsequent data analyses will aid in determining whether passage through the Circulating Water System will have a deleterious effect on planktonic organisms.

3.1.2.4 References

1. Lorenzen, C. J. 1967. Determination of chlorophyll and phaeo-pigments spectrophotometric equations. Limnol. and Oceanogr. 12:343-346.

2. Strickland, J. D. H., and T. R. Parsons. 1968. A manual of seawater analysis. Bull. 125, Fish. Res. Bd. Canada. 203 pp.

3. Strickland, J. D. H., and T. R. Parsons. 1968. A practical handbook of seawater analysis. Bull. 167, Fish. Res. Bd. Canada. 311 p.

4. American Public Health Association, et al. 1974. Standard Methods for the Examination of Water and Waste Water. American Public Health Association. 874 pp.

5. Icanberry, J. W., and R. W. Richardson. 1973. Quantitative sampling of live zooplankton with a filter-pump system. Limnol. and Oceanogr. 18(2): 333-335.

TABLE 3.1-1

WATER QUALITY ANALYSIS PARAMETERS

Parameter	PPM, as	Parameter	PPM, as
Solids, Non-Filterable (Diss.)	-	Nitrate (NO_3)	NO_3
Solids, Filterable (Susp.)	-	Conductivity (µmhos)	-
Solids, Total Volatile	-	Turbidity (JTU)	-
Calcium (Ca)	$CaCo_3$	Reducing Substances	H_2S
Magnesium (Mg)	$CaCO_3$	Chemical Oxygen Demand	COD
Sodium (Na)	$CaCO_3$	Total Organic Carbon	C
Potassium (K)	$CaCO_3$	Chlorine Demand, 30 Sec.	Cl
Iron, Total (Fe)	Fe	Chlorine Residual, Free	Cl
Copper, Total (Cu)	Cu	Chlorine Residual, Combined	Cl
Manganese (Mn)	Mn	Chlorine Demand, 3 Min.	Cl
Zinc (Zn)	Zn	Biochemical Oxygen Demand	BOD
Chromium (Cr)	Cr	Phenol	-
Ammonia (NH_3)	NH_3	Carbon Dioxide, Free	CO_2
Kjeldahl Nitrogen	N	Sulfides	S
Chloride (Cl)	$CaCO_3$	Dissolved Oxygen	O_2
Chloride (Cl)	NaCl	Phenophathalein Alkalinity	$CaCO_3$
Sulfate (SO_4)	$CaCO_3$	Methyl Orange Alkalinity	$CaCO_3$
Sulfate ($SO_4^=$)	SO_4	pH	-
Silica (SiO_2)	SiO_2	Phosphate (PO_4)	PO_4

TABLE 3.1-2

SUMMARY OF AQUATIC, TERRESTRIAL AND AERIAL SAMPLING PROGRAM

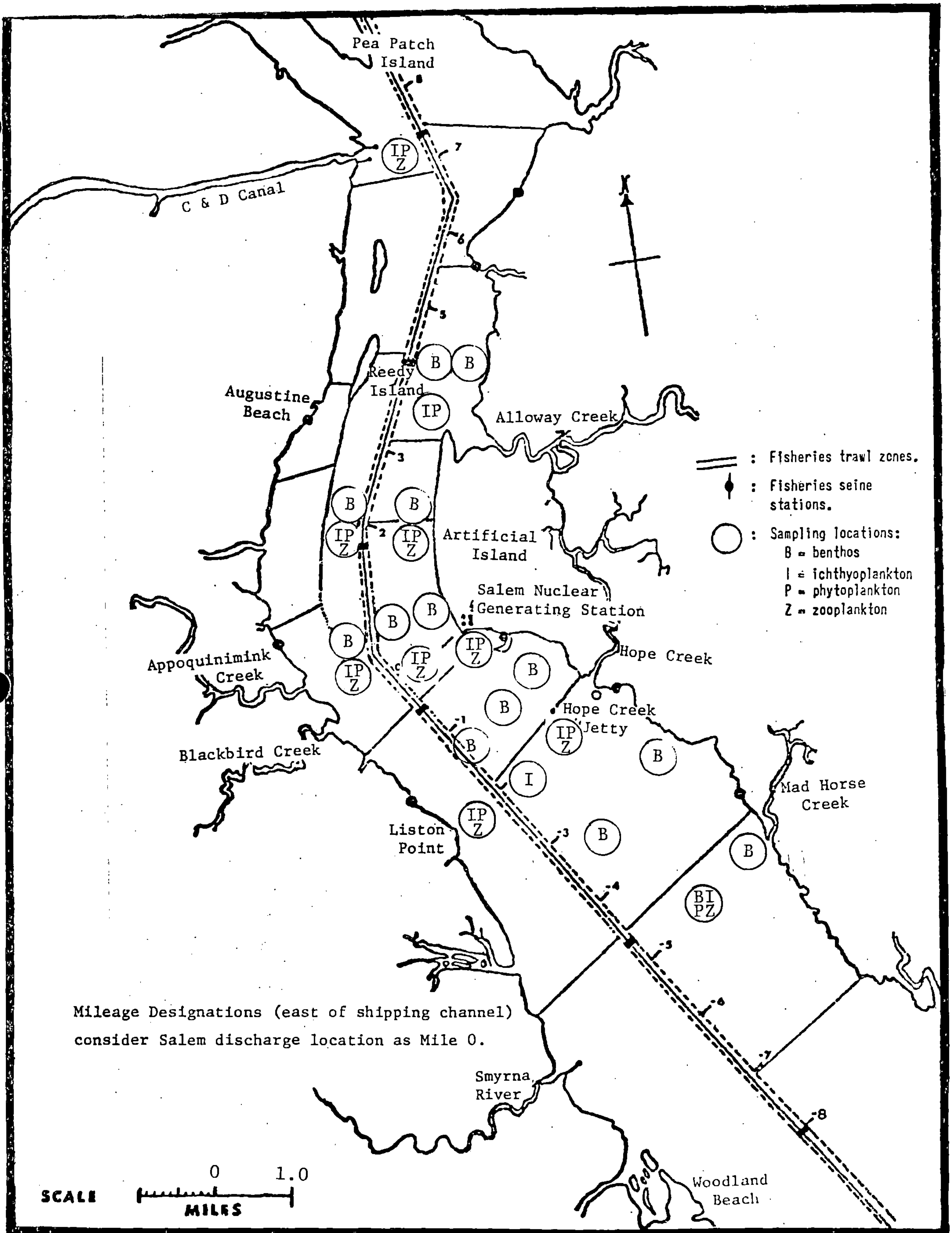
Sample	Method	Sampling Frequency*	Area Sampled Relative to Station (Mile 0) North	South
Aquatic				
Phytoplankton	Water bottles and other gear as appropriate	Biweekly to quarterly	to 7.5 miles;	to 5.0 miles
Zooplankton	Metered filter pump system fitted to plankton net	Monthly	7.5	5.0
Benthos	Ponar grab	Monthly to bimonthly	4.5	5.0
Blue Crab	Trawl haul, commercial crabbers	Biweekly to quarterly	8.5	9.0
Fisheries	Seines (Estuary)	Biweekly to monthly	6.5	4.0
	Trawls (Estuary)	Biweekly to monthly	8.5	9.0
	Trawls (Creek)	Biweekly to monthly	5.0	2.0
	Gill nets (Estuary)	Biweekly to quarterly	8.5	9.0
	Seines (Creek)	Biweekly to quarterly	5.0	0.0
Ichthyoplankton	Metered plankton net	Biweekly to monthly	7.5	5.0
Terrestrial and Aerial				
Birds	Visual observations	Biweekly to quarterly	Within 3-5 mile radius of site	
Mammals	Visual observations	Biweekly to quarterly	Within 3-5 mile radius of site	

*In the appropriate season.

TABLE 3.1-3

ANTICIPATED CHEMICAL WASTE DISCHARGE (SALEM STATION)

CHEMICAL CONSTITUENT	AVE.NATURAL CONC.IN WATER (mg/l)	AVE.NET AMOUNT DISCHARGED (lbs/day)	AVE.NET INCREASE (mg/l)	AVERAGE INCREASE (%)	MAX.NET AMOUNT DISCHARGED (lbs/day)	MAX.NET INCREASE (mg/l)	MAXIMUM INCREASE (%)
Cl_2,residual,free	0	870	<0.1	--	870	<0.1	--
Calcium as Ca	100	135	5.1×10^{-3}	5.1×10^{-3}	374	1.4×10^{-2}	1.4×10^{-2}
Magnesium as Mg	240	56	2.1×10^{-3}	8.8×10^{-4}	134	5.0×10^{-3}	2.1×10^{-3}
Sodium as Na	2000	600	2.2×10^{-2}	1.1×10^{-3}	1338	5.0×10^{-2}	2.5×10^{-3}
Potassium as K	70	55	2.0×10^{-3}	2.9×10^{-3}	142	5.3×10^{-3}	7.6×10^{-3}
Copper as Cu	0.082	--	--	--	110	4×10^{-3}	4.9
Sulfate as SO_4	570	1590	5.8×10^{-2}	0.01	3218	1.2×10^{-1}	2.1×10^{-2}
Chloride as Cl	3700	138	5.1×10^{-3}	1.4×10^{-4}	350	1.3×10^{-2}	3.5×10^{-4}
Nitrate as NO_3	5.6	2.4	9.0×10^{-5}	2×10^{-3}	25.6	9.6×10^{-4}	1.7×10^{-2}
Silica as SiO_2	--	46	1.7×10^{-3}	--	92	3.5×10^{-3}	--
Phosphate as PO_4	0.66	11	4.1×10^{-4}	0.06	40	1.5×10^{-3}	0.23
Volatile - Amines	--	4.2	1.5×10^{-3}	--	9.3	3.4×10^{-4}	--
Hydrazine	0	0.04	1.5×10^{-6}	--	0.05	1.9×10^{-6}	--
Suspended Solids	170	<1000	<0.04	--	--	--	--



PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Biological Sampling Stations in the Vicinity of Artificial Island FIG. 3.1-1

3.2 RADIOLOGICAL SURVEILLANCE

Objective

An environmental radiological monitoring program shall be conducted to assist in verifying that radioactive effluent releases are within allowable limits and that plant operations have no detrimental effects on the environment.

Specification

1. Environmental samples shall be collected and analyzed in accordance with Table 3.2-1. The sample locations are described in Table 3.2-1.

2. The analytical techniques used shall be such that the detection capabilities in Table 3.2-2 are achieved.

3. Reports shall be submitted in accordance with the requirements of Specification 5.6.

4. During the seasons that animals producing milk for human consumption are on pasture, samples of fresh milk shall be obtained from these animals at the locations and frequencies shown in Table 3.2-1, and analyzed for their radioiodine content (calculated as I-131). Analyses shall be performed within eight days of sampling. Suitable analytical procedures shall be used to determine the radioiodine concentration to a sensitivity of 0.5

picocuries per liter of milk at the time of sampling. For activity levels at or above 0.5 picocuries per liter, the determinate error of the analysis shall be within ± 25%. Results shall be reported, with associated calculated error, as picocuries of I-131 per liter of milk at the time of sampling.

5. A census of milk animals within a 1-mile radius from the plant site or within the 15 mrem/yr isodose line, whichever is larger, shall be conducted at the beginning and at the middle of each grazing season by using a door to door or equivalent counting technique to determine their location and number with respect to the site. A census shall be conducted within a 5-mile radius for cows and goats, with enumeration by using referenced information from county agricultural agents or other reliable sources.

If it is learned from this census that milk animals are present at a location which yields a calculated infant thyroid dose greater than any other sampled locations, the new location shall be added to the surveillance program. The infant thyroid dose shall be calculated using the equations and assumptions presented in Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I dated March 1976. Following the addition of any new location, a sampling location yielding a lower calculated dose may then be dropped from the surveillance program at the end of the grazing season during which the census was conducted.

Any locations from which milk can no longer be obtained may be dropped from the surveillance program. The Commission shall be notified in writing that milk animals are not longer present at the location or sufficient quantity of milk cannot be obtained or there exists a lack of cooperation in obtaining samples. A new sampling location shall be specified to replace the dropped location. The new location shall be one which represents the next greatest calculated infant thyroid dose.

If the calculated dose to a child's thyroid at any location where there is an animal producing milk for human consumption exceeds 15 mrem/yr, milk sampling shall be done weekly with I-131 analysis being performed on each sample.

6. A census of gardens producing fresh leafy vegetables for human consumption (e.g., lettuce, spinach, etc.) shall be conducted near the end of the growing season to determine their location with respect to the site. This census is limited to gardens having an area of 500 square feet or more and shall be conducted under the following conditions:

   1. Within a 1-mile radius of the plant site, enumeration by a door-to-door or equivalent counting technique.

2. If no milk-producing animals are located in the vicinity of the site, as determined by Specification 4 of Section 3.2, the census described in item 1. above shall be extended to a distance of 5 miles from the site.

If this census indicates the existence of a garden at a location yielding a calculated thyroid dose greater than that from the previously sampled garden, the new location shall replace the garden previously having the maximum iodine concentration. Also, any location from which fresh leafy vegetables can no longer be obtained may be dropped from the surveillance program. The NRC shall be notified in writing that such vegetables are no longer grown at that location.

7. Deviations shall be permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or malfunctions of automatic sampling equipment. In the case of the latter, corrective action shall be completed prior to the end of the next sampling period, if possible. Any location from which environmental monitoring program samples can no longer be reasonably obtained may be dropped from the surveillance program. The NRC shall be notified in writing of the reasons for this action. Any location which is dropped shall be replaced by a suitable alternate location.

Reporting Requirement

An annual report shall be submitted in accordance with the requirements of Specification 5.6.1.

Non-routine reports shall be submitted, as required, in accordance with Specification 5.6.2.

Bases

The magnitude and fluctuation of radioactivity levels in the environment surrounding the plant have been determined during implementation of the pre-operational environmental radiation monitoring program. This information serves as a solid baseline for evaluating any changes in environmental radioactivity levels during plant operation. The operational environmental radiation monitoring program was derived using the preoperational environmental radiation monitoring program as a basis.

The monitoring program utilizes a series of sampling locations which were determined by consideration of the spatial distribution of station effluents, including areas where concentrations of effluents in the environment are expected to be greatest, site meteorology, population distribution and ease of access to the sampling stations. The selection of sampling media was based on an evaluation of potential critical pathways of radiation exposure to man.

Concurrent sampling at control and indicator stations permits plant-produced radionuclides to be distinguished from other sources of radionuclides.

TABLE 3.2-1

OPERATIONAL ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

EXPOSURE PATHWAY	STATION CODE	LOCATION	COLLECTION METHOD & FREQUENCY	TYPE & FREQUENCY OF ANALYSIS
I. AIRBORNE				
(a) PARTICULATES	10D1	3.9 mi SSW of vent	Continuous low volume air sampler. Sample collected every week along with filter change	composite Sr 89 composite Sr 90 performed quarterly **Gross beta analysis on each weekly sample done weekly**** Composition Gamma scan quarterly
	2S1	on site		
	1F1	5.8 mi N of vent		
	2F2	8.7 mi NNE of vent		
	→ 3H3	110 mi NE of station		
	16E1	4.1 mi NNW of vent		
(b) IODINE	10D1	3.9 mi SE of vent	A TEDA impregnated charcoal flow-through cartridge is connected to air particulate air sampler and is collected weekly.	Iodine 131 analyses are performed weekly
	16E1	4.1 mi NNW of vent		
	2F2	NJ; 8.7 mi NNE of vent		
	2S1	on site		
	→ 3H3	110 mi NE of vent		
II. SOIL	10D1	3.9 mi SSW of vent	10 soil plugs to a depth of 6" over an area of 25 ft^2 are composited and sealed in a plastic bag at each location* A sample will be collected from each location once every 3 years	Gamma spectrometry performed on each sample on collection Sr-90 analyses on one sample from each location on collection
	16E1	4.1 mi NNW of vent		
	→ 3G1	16.6 miles NE of vent		
	→ 3H3	110 mi NE of station		
	1F1	5.8 mi N of vent		
	2F2	NJ; 8.7 mi NNE of vent		
	5D1	3.5 mi E of vent		
	2F1	5 mi NNE of vent		
	2E1	4.4 mi NNE of vent		
	2S1**	on site		

* Soil samples are taken in accordance with procedures outlined in HASL-300 (Rev. 5/73). If a suitable sample cannot be obtained at a location, a sample shall be obtained from a new location. The NRC shall be notified in writing of the new sample location.

→ Control Station

** Gamma spectrometry shall be performed if gross beta exceeds four times the control station value.

TABLE 3.2-1 (Cont.)

OPERATIONAL ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

EXPOSURE PATHWAY	STATION CODE	LOCATION	COLLECTION METHOD & FREQUENCY	TYPE & FREQUENCY OF ANALYSIS
III. DIRECT				
	10D1	3.9 mi SSW of vent		
	16E1	4.1 mi NNW of vent		
	2F2	8.7 mi NNE of vent		
	13F1	Middletown, Del; 9.8 miles W of vent		
	1F1	5.8 mi N of vent		
			2 dosimeters will be collected from each location quarterly	Gamma dose-quarterly
	→ 3G1	16.6 miles NE of vent		
	→ 3H1	32 mi NE of vent		
	→ 2H1	38.5 mi NNE of vent		
	6S1	.2 miles ESE of vent		
	7S1	Station personnel gate		
	14D1	3.9 mi WNW of vent		
	10S1	Cooling water inlet; 150 ft SSW of vent	At least one dosimeter collected from this location quarterly	
IV. WATER				
	11A1	Approximately 650 ft SW of vent		
(a) SURFACE	→ 12C1	2-1/2 mi WSW of vent	Two gallon sample to be collected monthly providing winter icing conditions allow sample collection	Gamma scan monthly H-3, Sr-89 and Sr-90 analyses of quarterly composites
	7E1	1 mi W of Mad Horse Creek; 4.5 mi SE of vent		
(b) GROUND	4S1	on site		
	→ 3E1	4.5 mi NE of vent	Two gallon grab sample is collected monthly	Gamma scan - QC H-3 analyses are done monthly
(c) DRINKING	2F3 (raw)	Salem Water Co.; 8 mi NNE of vent	50 ml aliquot is taken daily and composited to a monthly sample of two gallons	Gross beta monthly Gamma scan - QC H-3 monthly Sr 89 and Sr 90 analyses on quarterly composites

→ Control Station

TABLE 3.2-1 (Cont.)

OPERATIONAL ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

EXPOSURE PATHWAY	STATION CODE	LOCATION	COLLECTION METHOD & FREQUENCY	TYPE & FREQUENCY OF ANALYSES
V.. AQUATIC				
BENTHOS	7E1	1 mi W of Mad Horse Creek; 4.5 mi SE of Vent	A benthos sample consisting of benthic organisms and associated sediment is taken semiannually.	Gamma spectrometry of each sample semi-annually; Sr-89 and Sr-90 semi-annually on organisms, Sr-90 semi-annually on sediment
	→ 12C1	2-1/2 mi WSW of vent		
	11A1	Outfall area; 650' SW of vent		
VI. INGESTION				
(a) MILK	15F1	5.2 mi NW of vent	**Four gallon grab sample of fresh milk is collected from each farm semi-monthly. Collected weekly if calculated dose exceeds 15 mrem to child's thyroid.**	Gamma scan monthly; Sr-89 and Sr-90 monthly; I-131 monthly I-131 weekly if calculated dose exceeds 15 mrem to child's thyroid
	2F1	5 mi NNE of vent		
	5F1	6.5 mi E of vent		
	14F1	5.5 mi WNW of vent		
	→ 3G1	16.6 miles NE of vent		
(b) FISH	11A1	Outfall area; 650' SW of vent	Two key samples of fish are sealed in plastic bag or jar and frozen semi-annually or when in season	Gamma scan of edible portion on collection
	→ 12C1	2-1/2 mi WSW of vent		
(c) CRAB	11A1	Outfall area; 650' SW of vent	Two key samples of crab are sealed in a plastic bag or jar and frozen semi-annually or when in season	Gamma scan of edible portion on collection
	→ 12C1	West bank opposite Artificial Island, 2-1/2 mi WSW of vent		
(d) FRUITS or VEGETATION	→ 1G1	10.2 miles N of vent	Samples are collected during the normal harvest season, sealed in plastic, and frozen if perishable. Sufficient sample is collected to yield 500 grams of dry weight and done annually	Radioiodine determination of green leafy vegetables on collection Gamma scan on collection
	2E1	4,45 mi NNE of vent		
	2F1	3 mi NNE of vent Other locations may be substituted if a farm discontinues to grow the samples of concern.		
(e) GAME	XXX	Station vicinity east side of estuary	Muskrats are skinned and frozen semi-annually	Gamma scan on edible portion only on collection
	→ XXX	West side of estuary, 3-5 mi from vent		
	XXX	Within 10 mi of Station	**Beef portion of cow is*** sampled and frozen semi-annually**	

QC = Quarterly composite

XXX = location given at time of collection

*** = This sample is subject to availability of slaughtered cow

→ Control Station

TABLE 3.2-2

SENSITIVITY LEVELS FOR ENVIRONMENTAL SAMPLE ANALYSES

SAMPLE TYPE	TYPE OF ANALYSIS	SENSITIVITY*
Air Particulates	Gross beta	4.4×10^{-14} µCi/ml
	Gamma scan	1×10^{-14} " "
	Sr 89	5×10^{-15} " "
	Sr 90	1×10^{-15} " "
Air Iodine	I 131	4×10^{-14} " "
Soil	Gamma scan	1×10^{-7} µCi/g-dry
	Sr 90	5×10^{-8} " "
Thermoluminescent Dosimeters	Gamma	approx. 5mrem/yr
Surface water	Gamma scan	1×10^{-9} µCi/ml
	Tritium	2×10^{-7} " "
	Sr 89	5×10^{-9} " "
	Sr 90	1×10^{-9} " "
Ground water	Gamma scan	1×10^{-9} " "
	Tritium	2×10^{-7} " "
Drinking water	Gross beta	1×10^{-9} " "
	Gamma scan	1×10^{-9} " "
	Tritium	2×10^{-7} " "
	Sr 89	5×10^{-9} " "
	Sr 90	1×10^{-9} " "
Benthos	Gamma scan	1×10^{-7} µCi/g-dry
	Sr 89	5×10^{-7} " "
	Sr 90	1×10^{-7} " "
Fish	Gamma scan	8×10^{-8} µCi/g-wet

TABLE 3.2-2 (Cont'd)

SENSITIVITY LEVELS FOR ENVIRONMENTAL SAMPLE ANALYSES

SAMPLE TYPE	TYPE OF ANALYSIS	SENSITIVITY*
Milk	Gamma scan	1×10^{-8} μCi/ml
	Sr 89	5×10^{-9} " "
	Sr 90	1×10^{-9} " "
	I 131	5×10^{-10} " "
Fruits and Vegetables	Gamma scan	5×10^{-8} μCi/g-wet
	I 131	5×10^{-8} " "
Meat	Gamma scan	8×10^{-8} " "
Game	Gamma scan	8×10^{-8} " "

*The sensitivity of the gamma scan analysis are for Cs 134 and Cs 137 and are at a 95% confidence level.

4.0 SPECIAL SURVEILLANCE AND STUDY ACTIVITIES

4.1 EXPERIMENTAL ENTRAINMENT STUDIES

Objective

To estimate the effect of rapid temperature and pressure changes in the Circulating Water System on ichthyoplankton and zooplankton.

Specification

Responses to short duration increases in temperature and pressure which closely approximate those in the Circulating Water Systems will be determined for the more common entrainable organisms in the vicinity of Artificial Island.

The following species of fishes and crustaceans will be tested (contingent upon their availability): white perch, Morone americana; striped bass, Morone saxatilis; alewife, Alosa pseudoharengus; blueback herring, Alosa aestivalis; scud, Gammarus sp.; opossum shrimp, Neomysis americana; sand shrimp, Crangon septemspinosa; and grass shrimp, Palaemonetes pugio. Other species will be tested as they are available and as scheduling permits. Size range of test organisms will be between 2 and 50 mm (0.07 and 1.96 inches) total length.

Organisms will be considered acclimated to prevailing conditions at their point of capture and will be held under similar conditions for periods up to about one week prior to testing. Larval fishes will be hatched in the laboratory under conditions similar to those in the spawning areas.

Tests will be conducted in a rigid transparent PVC apparatus in which the effects of temperature and pressure can be evaluated independently and concurrently.

Temperature will be measured with a standardized mercury thermometer (precision, 0.5 C) within the test chamber and pressure with a Robertshaw test gauge (precision, 1/2 mm Hg) connected to the atmosphere in the test chamber.

Other variables monitored will include salinity, pH, and oxygen content. Salinity of the water will be measured with a salinometer (precision, 0.1 ppt) prior to testing. Determination of pH will be made with a pH meter (precision, 0.1 pH unit) before testing. Oxygen content of the water will be verified with an oxygen meter (precision, 0.1 mg O_2/l) prior to testing.

Water used in testing will be taken from Appoquinimink Creek at high tide. This approximates the water quality of the Delaware River in the vicinity of Artificial Island at high tide.

Test organisms will be exposed to various combinations of test conditions. Acclimation temperature will vary seasonally within the range 5° to 30°C (41-86°F). Test salinities will be appropriate levels within the range 0 to 12 ppt, pH between 7.0 and 8.0, and oxygen content near air saturation. Temperature increases for test organisms will be from 7.5° to 15°C (13.5-27°F) above ambient (acclimation). Test organisms will be exposed to pressures from 69 to 180 mm Hg absolute (-1.4-20.1 psig) in a sequence simulating passage through the condenser cooling system.

A general control group will be placed in a holding container. A handling control will receive standard handling in the apparatus, but will not be exposed to changes in either temperature or pressure. Test organisms will be exposed to one of three experimental conditions: pressure changes only, temperature changes only, or both temperature and pressure changes concurrently. Observations will be made on the test organisms during and immediately after testing and at appropriate intervals through a 96-hour period to determine immediate and long-term effects.

These studies will be conducted for a period of 1 year after Unit 1 becomes operational.

Reporting Requirements

Results of these studies shall be reported in accordance with Specification 5.6.1.

Bases

These studies and subsequent data analyses will aid in determining whether the temperature and pressure conditions in the Circulating Water System will have a deleterious effect on entrainable organisms.

4.2 THERMAL AND CHEMICAL RESPONSES OF ESTUARINE ORGANISMS

Objective

The principal objectives of these studies are to determine: (1) temperature preference, (2) temperature and chemical avoidance, and (3) cold shock temperatures for the various common fishes and macroinvertebrates in this region of the Delaware River estuary.

Specification

4.2.1 GENERAL

All studies will be conducted throughout the year under near air-saturated levels of dissolved oxygen at appropriate salinities within a range of 0.0 to 12.0 ppt and acclimation temperatures within a range of 5° to 30°C (41-86°F). Levels of temperature, salinity, dissolved oxygen, light, and pH, as well as size of test specimens will be measured and recorded for all tests in all studies. Water for all studies will be taken from Appoquinimink Creek, a tributary of the Delaware River, at high tide. This source approximates the water quality of the Delaware River in the vicinity of Artificial Island at high tide.

Species to be tested in all studies will include the bay anchovy, *Anchoa mitchilli*; the Atlantic silverside, *Menidia menidia*; the white perch, *Morone americana*; and the grass shrimp, *Palaemonetes pugio*. Other fishes and macroinvertebrates will also be tested depending upon availability. Specimens to be tested will be acclimated to the approximate temperatures and salinity at which they were captured for a period of 24 to 96 hours prior to testing.

These studies will be conducted for a period of one year after Unit 1 becomes operational.

4.2.2 TEMPERATURE PREFERENCE STUDIES

Temperature preferences will be determined in a horizontal temperature gradient in a trough approximately 13 ft long, 6 inches wide, and 1 ft deep, having a stainless steel bottom in the central 12 ft. The trough will be partially enclosed with polyethylene sheeting for light control. Tests will be generally conducted at a lighting of either 4 or 40 foot-candles at the surface of the water. Light level will be measured with a foot-candle meter whose precision is 2 foot-candles.

Initially the trough will be filled with water of the acclimation salinity and temperature of the test specimens to a depth which will permit horizontal but restrict vertical movement. Specimens will then be randomly placed in the trough and allowed to remain for several hours prior to the establishment of a thermal gradient. The horizontal thermal gradient may extend to 10°C above and below the acclimation temperature. Observations will be made at 5- or 10-minute intervals, depending on specimen activity. The temperature at the position of each specimen will be recorded using thermistors (placed at 6-inch intervals along the trough) which are connected to a temperature readout. The precision of the readout is 0.5°F. The test will be concluded when the specimens select the same temperature for four successive observation times.

4.2.3 TEMPERATURE AND CHEMICAL AVOIDANCE STUDIES

The avoidance design to be employed in these studies is a modification of the design employed first by Shelford and Allee.[1] In this modified design a control and a replicate are determined simultaneously. The apparatus is constructed such that, in thermal tests, water of differing temperatures flows into the opposing ends of a divided trough and then drains at the center. In chemical tests, various concentrations of the compound are substituted for the temperature increase. Due to the sharp gradient at the center drain the apparatus is effectively divided into quadrants. The water temperature (or chemical composition) is the same in diagonally opposed quadrants, but different in those directly opposed. One set of diagonally opposed quadrants is designated as experimental, the other set as control. Temperatures in the directly opposed quadrants are increased in step gradients, with the experimental quadrants being 3° to 5°F higher than the control quadrants. In the chemical tests, only the chemical concentration in the experimental quadrant is increased. Equal numbers of specimens are placed in each quadrant. The length of time spent by each specimen in each respective quadrant is continuously measured over a 5- to 15-minute test period. This results in a frequency distribution which is then analyzed statistically to determine the significance of the response to the chemical concentration or temperature increase. Tests begin at ambient temperature and continue through the step gradient until a significant avoidance response is given in both subtroughs. Responses to chlorine (both free and combined states) will be determined. Responses to other chemical compounds will be determined as needed or recommended.

Oxygen and pH will be monitored throughout all tests. The precision of the oxygen measurements is 0.1 mg/l; that of pH is 0.1 pH unit. Free and combined chlorine residuals will be determined by amperometric titration or an equivalent method. The precision of these measurements is 0.01 ppm and the limit of detectability is 0.02 ppm chlorine. The thermal conditions in temperature tests will be monitored by a multichannel temperature recorder connected to thermocouples at 6-inch intervals. The precision of the temperature measurements is 0.5°F. If more suitable instrumentation becomes available, it may be employed. The trough is enclosed for light level regulation as well as to permit movement around the trough area. Due to the increase in specimen activity which accompanies temperature increases, observations for temperature tests will be made via closed-circuit television.

4.2.4 COLD SHOCK STUDIES

Responses to sudden decreases in temperature will be determined over test periods of 96 hours. Test organisms will be exposed to instantaneous change from ambient temperatures to levels which permit estimation of the median lethal temperature decrease. Mortality will be defined as a total cessation of gill movement and when prodding with a glass rod elicits no response.

Reporting Requirements

Results of these experimental programs will be reported in accordance with Specification 5.6.1.

Bases

The results of these studies will aid in estimating the effects of the thermal plume on the motile organisms in the receiving waters.

References

1. Shelford, V. E., and W. C. Allee. 1913. The Reactions of Fishes to Gradients of Dissolved Atmospheric Gases. Jour. Exptl. Zool. 14: 207-266.

4.3 THERMAL PLUME MAPPING

Objective

To insure a minimal environmental impact to aquatic life due to the thermal discharge resulting from station operation.

Specification

A field survey will be performed throughout a complete tidal cycle to determine the characteristics of the thermal plume. This survey will be performed within the first year following the simultaneous full power operation of Units 1 and 2.

Reporting Requirement

The thermal plume survey results will be reported in accordance with Specification 5.6.1

Bases

Extensive model work has been performed by Pritchard-Carpenter, Consultants (see Dispersion and Cooling of Waste Heat Released into the Delaware River Estuary, a study performed for Salem Nuclear Generating Station utilizing the U.S. Army Corps of Engineers' model at Vicksburg, Mississippi, Salem FSAR, Appendix A.4), to determine the best discharge location and configuration for minimum thermal impact and recirculation. Attempts have been made by PSE&G to continually monitor river temperature. The current in the river, river traffic and theft of buoys has necessitated a review of the need for continual thermal monitoring. It is the opinion of Pritchard-Carpenter, Consultants, that continuous thermal monitoring is of no value. Pilfered and damaged equipment, together with system malfunctions, make continuous monitoring impractical. It is believed that a single set of accurate data is ample for model verification.

4.4 INTAKE VELOCITY STUDY

Objective

To measure the intake velocity in the vicinity of the traveling screens for both the cooling and service water intakes.

Specification

A one time study shall be performed to measure the intake velocity in the vicinity of the traveling screens for both the cooling and service water intakes. Measurements shall be taken at 10 points in front of each screen during both high and low tidal conditions. All pumps shall be operating normally during the period of measurement.

This study shall be performed within the first year following the simultaneous full power operation of Units 1 and 2.

Reporting Requirement

The intake velocity study results shall be reported in accordance with Specification 5.6.1.

Bases

The results of this study will aid in estimating the effects of intake velocity on entrainment and impingement levels reported.

5.0 ADMINISTRATIVE CONTROLS

5.1 RESPONSIBILITY

5.1.1 The implementation of the surveillance programs, including sampling, sample analysis, evaluation of results and the preparation of required reports is the responsibility of the Nuclear Licensing and Environmental Studies Group in the Mechanical Division of the Engineering Department. This group is responsible for the assignment of personnel to the above functions, for assurance that appropriate written procedures, as described in Section 5.5.1, are utilized in the surveillance program activities and for assuring the quality of surveillance program results, as described in Section 5.5.3.

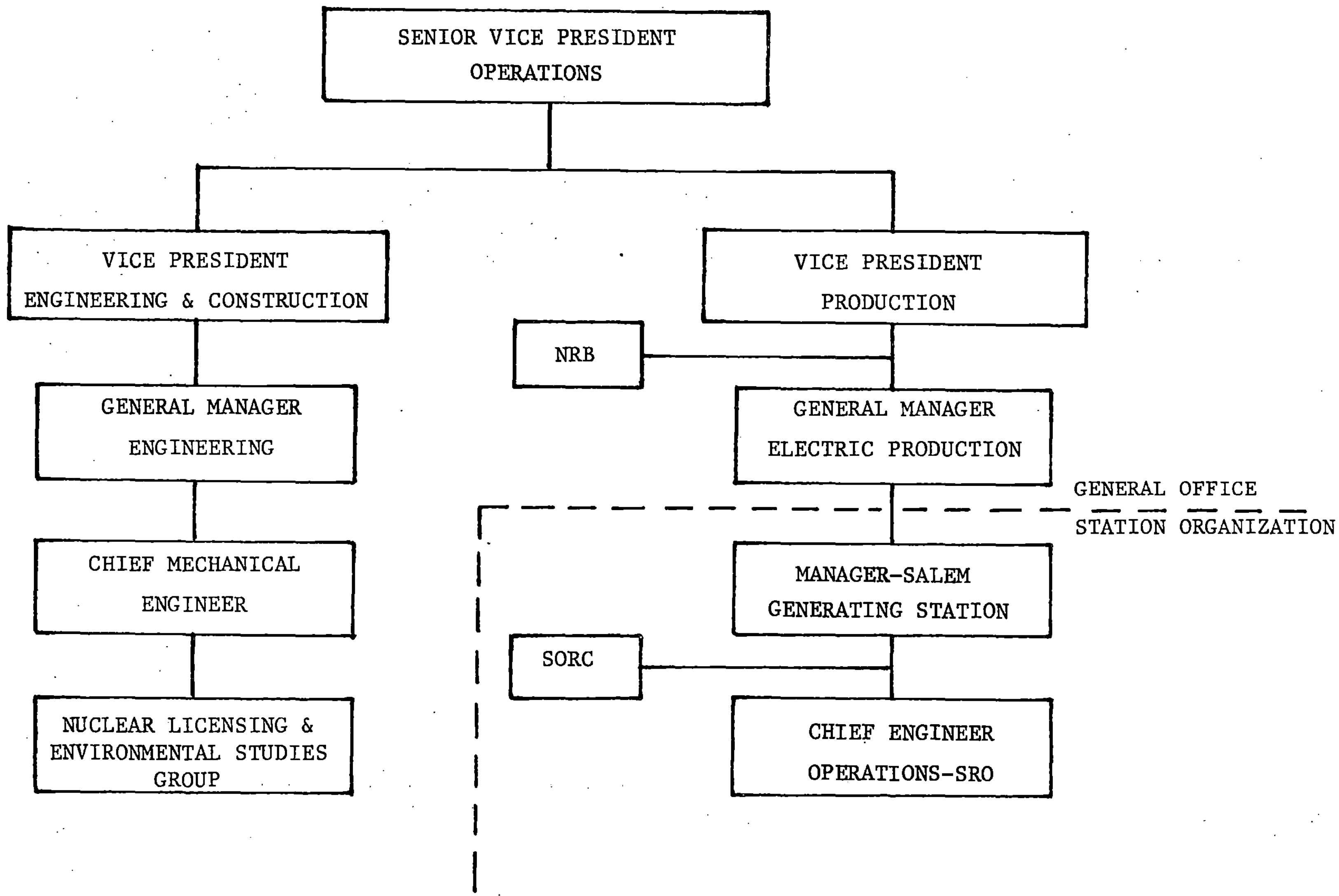
5.1.2 The Station Manager or his delegated alternate is responsible for operating the plant in compliance with the limiting conditions for operation as specified in the Environmental Technical Specifications. His responsibility includes assurance that plant activities are conducted in such a manner as to provide continuing protection to the environment and that personnel performing such activities use appropriate written procedures as described in Section 5.5.

5.2 ORGANIZATION

5.2.1 Figure 5.2-1 identifies the corporate relationship between the Nuclear Licensing and Environmental Studies Group and the Station Manager. Figure 5.2-2 identifies the organization of the Mechanical Division of the Engineering Department and Figure 5.2-3 identifies the Production Department Station Organization.

5.2.2 The Nuclear Review Board (NRB) and Station Operations Review Committee (SORC) are shown in Figure 5.2-1. They are advisory groups to the Vice President - Production and Station Manager, respectively. These groups perform the independent review and audit activities for these Environmental Technical Specifications. The SORC is composed of Station supervisory personnel and the NRB is composed of management and technical support personnel from the general office staff. The Station Manager is a member of the NRB. There is no direct relationship between the two groups except that the NRB is charged with the responsibility to review SORC activities.

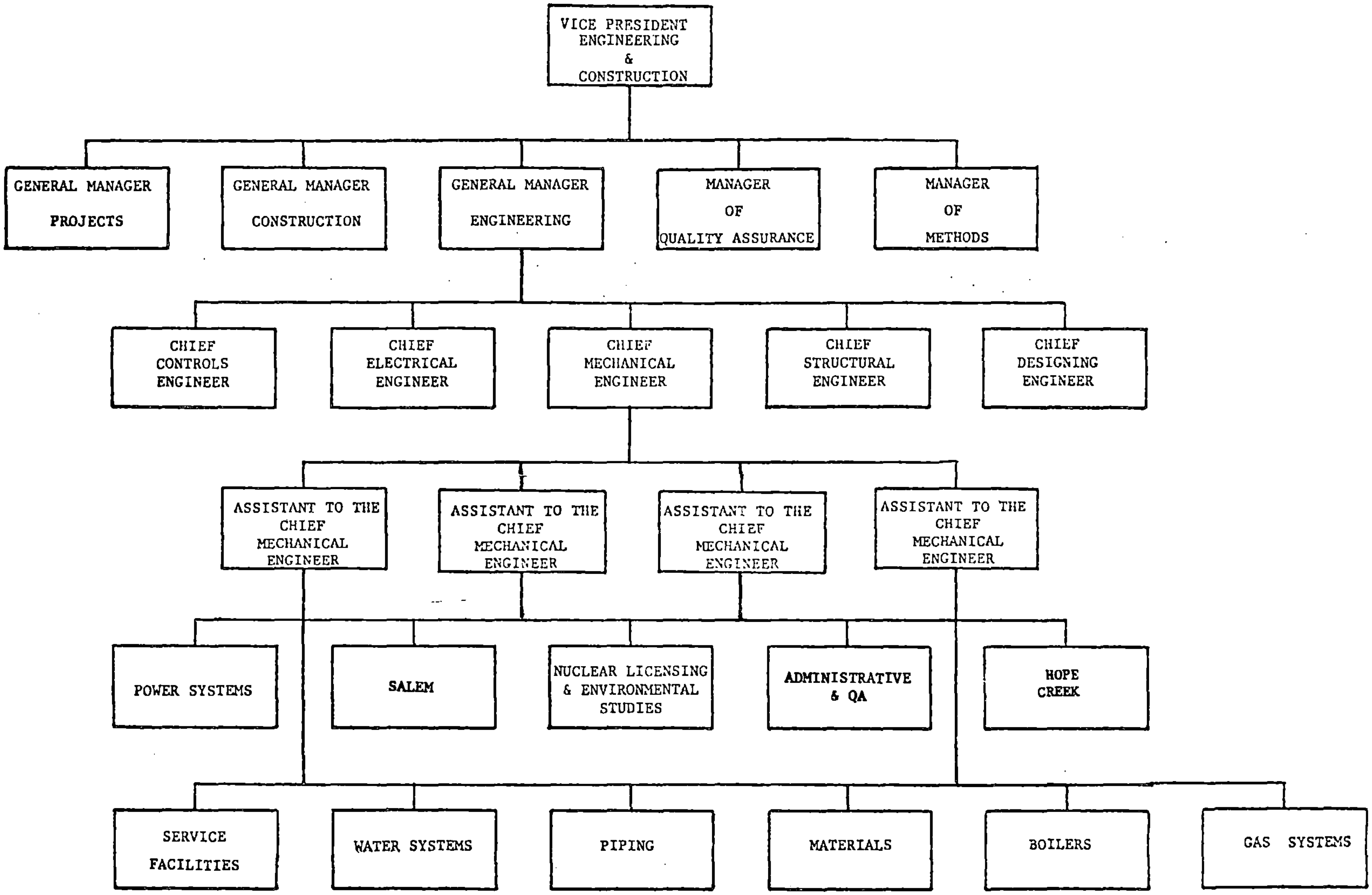
Detailed descriptions of these organizations are presented in the Salem FSAR, Chapter 12, "Conduct of Operations."



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Organization Chart Showing Corporate Interrelationships

FIG. 5.2-1



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

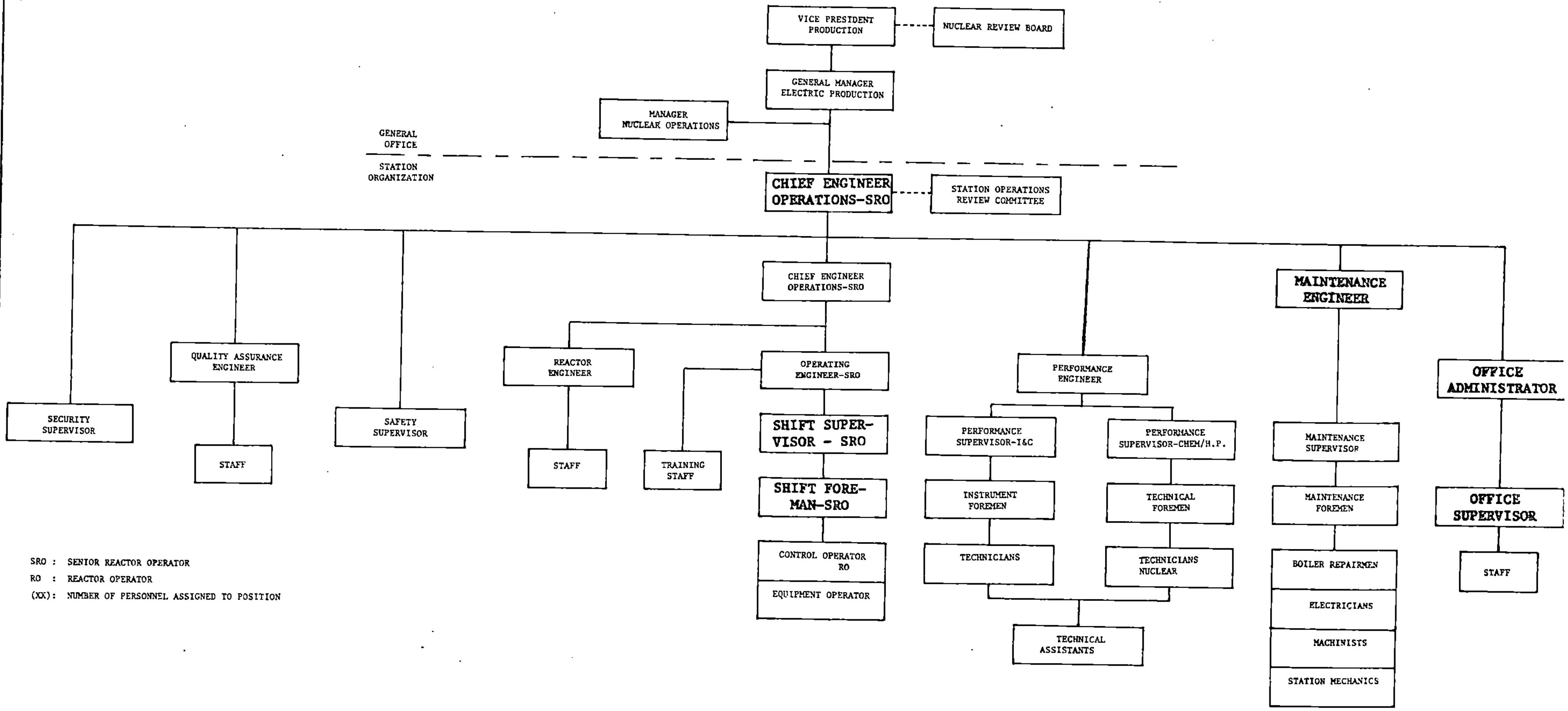
Mechanical Division Organization Chart

FIG. 5.2-2

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

STATION ORGANIZATION CHART

FIG. 5.2-3



SRO : SENIOR REACTOR OPERATOR
RO : REACTOR OPERATOR
(XX): NUMBER OF PERSONNEL ASSIGNED TO POSITION

5.3 REVIEW AND AUDIT

5.3.1 NUCLEAR REVIEW BOARD (NRB)

The NRB shall have the following responsibilities concerning the environmental impact of the plant:

1. NRB shall review:
    a. Proposed Environmental Technical Specification changes or license amendments.
    b. Violations of Environmental Technical Specifications.
    c. Environmental Monitoring Program and Evaluations.
    d. Routine and non-routine reports required by the Environmental Technical Specifications.

2. The NRB shall, at least once each year, conduct (or cause to have conducted) and evaluate audits of:
    a. Plant operation to assure Environmental Technical Specification compliance.
    b. Monitoring program sampling practices to assure they adhere to program schedule and appropriate procedures.

5.3.2 STATION OPERATIONS REVIEW COMMITTEE (SORC)

The SORC shall have the following responsibilities concerning the environmental impact of the plant:

1. Review plant procedures which have a potential impact on the environment.

2. Review proposed changes to the Environmental Technical Specifications.

3. Review environmental monitoring program results and evaluations.

4. Review routine and non-routine reports required by Section 5.6 prior to their submittal to the Commission.

5. Investigate all violations of Environmental Technical Specifications and recommend corrective action to prevent recurrence.

5.4 ACTION TO BE TAKEN IF A LIMITING CONDITION FOR OPERATION IS EXCEEDED

5.4.1 Remedial action as permitted by the Environmental Technical Specification shall be taken until the condition can be met.

5.4.2 Exceeding a limiting condition for operation shall be investigated by the Station Operation Review Committee.

5.4.3 A report for each occurrence shall be prepared and submitted as specified in Section 5.6.2.

5.5 PROCEDURES

5.5.1 Detailed written procedures, including applicable check lists and instructions, shall be prepared and followed for all activities involved in carrying out the Environmental Technical Specifications. Procedures for the environmental surveillance and special study programs described in Sections 3 and 4 shall be prepared by personnel responsible for the particular monitoring program. Procedures shall include sampling, data recording and storage, instrument calibration, measurements and analyses, and actions to be taken when limits are approached or exceeded. Testing frequency of any alarms shall be included. These frequencies shall be determined from experience with similar instruments in similar environments and from manufacturers' technical manuals.

5.5.2 In addition to the procedures specified in Section 5.5.1, the plant operating procedures shall include provisions to ensure the plant and all its systems and components are operated in compliance with the limiting conditions for operation established as part of the Environmental Technical Specifications.

5.5.3 Procedures will be established to assure quality results. Procedures will include:

a. Audits to assure organizations performing program activities are following policy directives and are using the appropriate written instructions.

b. A corrective action plan that identifies, controls and corrects deficiencies.

c. A plan to investigate anomalous or suspect results.

5.6 PLANT REPORTING REQUIREMENTS

5.6.1 ROUTINE REPORTS

5.6.1.1 Annual Environmental Operating Report

1.a. Nonradiological Report

A report on the environmental surveillance programs for the previous 12 months of operation shall be submitted to the Director of the Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation) as a separate document within 90 days after January 1 of each year. The period of the first report shall begin with the date of initial criticality. The report shall include summaries, interpretations, and statistical evaluation of the results of the non-radiological environmental surveillance activities (Section 3.0) and the environmental monitoring programs required by limiting conditions for operation (Section 2.0) for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

b. Reports to Other Agencies

Copies of routine reports required by Federal, State, local, and regional authorities for the protection of the environment shall be submitted to the Director, Office of Nuclear Reactor Regulation, USNRC, for information.

2. Radiological Report

a. A report on the radiological environmental surveillance programs for the previous 12 months of operation shall be submitted to the Director of the Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation) as a separate document within 90 days after January 1 of each year. The period of the first report shall begin with the date of initial criticality. The reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

b. Results of all radiological environmental samples taken shall be summarized on an annual basis following the format of Table 5.6-1. In the event that some results are not available within the 90 day period, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

5.6.1.2 Radioactive Effluents Release Report

1. A report on the radioactive discharges released from the site during the previous 6 months of operation shall be submitted to the Director of the Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation) within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in Reference 1, with data summarized on a quarterly basis following the format of Appendix B thereof.

2. The report shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter as outlined in Reference 1, with data summarized on a quarterly basis following the format of Appendix B thereof. Calculated offsite dose to humans resulting from the release of effluents and their subsequent dispersion in the atmosphere shall be reported as recommended in Reference 1.

5.6.2 NONROUTINE REPORTS

5.6.2.1 Nonroutine Environmental Operating Reports

A report shall be submitted in the event that (a) a limiting condition for operation is exceeded (as specified in Section 2.0, "Limiting Conditions for Operation"), (b) a report level is reached as specified in Section 3.0, "Environmental Surveillance"), or (c) an unusual or important event occurs that causes a significant environmental impact, that affects potential environmental impact from plant operation, or that has high public or potential public interest concerning environmental impact from plant operation.

Reports shall be submitted under one of the report schedules described below.

1. Prompt Report.* Those events requiring prompt reports within 24 hours by telephone, telegraph, or facsimile transmission to the Director of the Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation).

2. 30-Day Report.* Those events not requiring a prompt report shall be reported within 30 days by a written report to the Director of the Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation).

Copies of non-routine reports required by Federal, State, local, and regional authorities for the protection of the environment shall be reported to the Director, Office of Nuclear Reactor Regulation, for information.

5.6.2.2 Nonroutine Radiological Environmental Operating Report

1. Anomalous Measurement Report

If a confirmed measured level of radioactivity in any environmental medium exceeds 10 times the control station value, a written report shall be

*Written 10-day and 30-day reports and, to the extent possible, the preliminary telephone, telegraph, or facsimile reports shall (a) describe, analyze, and evaluate the occurrence, including extent and magnitude of the impact, (b) describe the cause of the occurrence, and (c) indicate the corrective action (including any significant changes made in procedures) taken to preclude repetition of the occurrence and to prevent similar occurrences involving similar components or systems.

Note: The significance of an unusual or apparently important event with regard to environmental impact may not be obvious or fully appreciated at the time of occurrence. In such cases, the NRC shall be informed promptly of changes in the assessment of the significance of the event and a corrected report shall be submitted as expeditiously as possible.

submitted to the Director of the NRC Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation) within 10 days after confirmation. This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

2. Milk Pathway Measurements*

   a. If milk samples collected over a calendar quarter show average I-131 concentrations of 7.0 picocuries per liter or greater, a written report and a plan shall be submitted within 30 days advising the Commission of the proposed action to ensure the plant-related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any member of the general public.

3. Fresh Leafy Vegetable and Inhalation Pathway Measurements*

   a. If green leafy vegetable samples collected over a calendar quarter show average I-131 concentrations of 220 picocuries per kilogram or greater, a written report and a plan shall be submitted within 30 days advising the Commission of the proposed action to ensure the plant-related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any member of the general public.

   b. If air samples collected over a calendar quarter show average concentrations of I-131 of 2 picocuries per cubic meter or greater, a written

*Milk pathway measurements apply when this pathway is controlling with respect to atmospheric radioiodine releases. If the milk pathway is not controlling then the reporting requirements for the fresh leafy vegetable and inhalation pathway are in effect.

report and a plan shall be submitted within 30 days advising the Commission of the proposed action to ensure the plant related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any member of the general public.

c. If statistically significant variations of offsite environmental radionuclide concentrations with time are observed, a comparison of these results with effluent releases shall be provided in the Annual Operating Report.

5.6.2.3 Nonroutine Radioactive Effluent Reports

1. Liquid Radioactive Wastes Report. If the cumulative releases of radioactive materials in liquid effluents, excluding tritium and dissolved gases, should exceed one-half the design objective annual quantity during any calendar quarter, the licensee shall make an investigation to identify the causes of such releases and define and initiate a program of action to reduce such releases to the design objective levels. A written report of these actions shall be submitted to the NRC within 30 days from the end of the quarter during which the release occurred.

2. Gaseous Radioactive Wastes Report. Should the conditions (a), (b), and (c) listed below exist, the licensee shall make an investigation to identify the causes of the release rates and define and initiate a program of action to reduce the release rates to design objective levels. A written report of these actions shall be submitted to the NRC within 30 days from the end of the quarter during which the releases occurred.

a. If the average release rate of noble gases for the site during any calendar quarter exceeds one-half the design objective annual quantity.

b. If the average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter exceeds one-half the design objective annual quantity.

c. If the amount of iodine-131 released during any calendar quarter is greater than 0.5 Ci/reactor.

3. Unplanned or Uncontrolled Release Report. Any unplanned or uncontrolled offsite release of radioactive materials in excess of 0.5 curie in liquid or in excess of 5 curies of noble gases or 0.02 curie of radioiodines in gaseous form requires notification. This notification must be made by a written report within 30 days to the NRC. The report shall describe the event, identify the causes of the unplanned or uncontrolled release and report actions taken to prevent recurrence.

5.6.3 CHANGES IN ENVIRONMENTAL TECHNICAL SPECIFICATIONS

5.6.3.1 A report shall be made to the Commission prior to implementation of a change in plant design, in plant operation, or in procedures described in Section 5.5 if the change would have a significant effect on the environment or involves an environmental matter or question not previously reviewed and evaluated by the Commission. The report shall include a description and evaluation of the changes and a supporting benefit-cost analysis.

5.6.3 Request for changes in environmental technical specifications shall be submitted to the Director, Office of Nuclear Reactor Regulation, for review and authorization. The request shall include an evaluation of the environmental impact of the proposed change and a supporting benefit-cost analysis.

5.6.3.3 Changes or additions to permits and certificates required by Federal, State, local, and regional authorities for the protection of the environment will be reported. When the required changes are submitted to the concerned agency for approval, they will also be submitted to the Director, Office of Nuclear Reactor Regulation, for information. The submittal will include an evaluation of the environmental impact of the change.

References

1. Regulatory Guide 1.21, Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, Revision 1, June 1974.

TABLE 5.6-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility ______________________ Docket No. ______________

Location of Facility ______________________ Reporting Period ______________
(County, State)

Medium or Pathway Sampled (Unit of Measurement)	Analysis and Total Number of Analyses Performed	Lower Limit of Detection a/ (LLD)	All Indicator Locations Mean b/ Range b/	Location with Highest Annual Mean Name Distance and Direction	Location with Highest Annual Mean Mean b/ Range b/	Control Locations Mean b/ Range b/	Number of Nonroutine Reported Measurements c/
Air Particulates (pCi/m^3)	β 416	0.003	0.08 (200/312) (0.05-2.0)	Middletown 5 miles 340°	0.10 (5/52) (0.08 - 2.0)	0.08 (8/104) (0.05-1.40)	1
	γ 32						
	^{137}Cs	0.003	0.05 (4/24) (0.03-0.13)	Smithville 2.5 miles 160°	0.08 (2/4) (0.03 - 0.13)	MDL	4
	^{140}Ba	0.003	0.03 (2/24) (0.01-0.08)	Podunk 4.0 miles 270°	0.05 (2/4) (0.01 - 0.08)	0.02 (1/8)	1
	^{89}Sr 40	0.002	MDL	-	-	MDL	0
	^{90}Sr 40	0.0003	MDL	-	-	MDL	0
Fish pCi/kg (dry weight)	γ 8						
	^{137}Cs	80	MDL	-	MDL	90 (1/4)	0
	^{134}Cs	80	MDL	-	MDL	MDL	0
	^{60}Co	80	120 (3/4) (90-200)	River Mile 35 Podunk River	See Column 4	MDL	0

a/ Minimum Detectable Level (MDL) is defined as $3\sigma_b$ where 6σ is defined as the standard deviation of the background count.

b/ Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses

c/ Nonroutine reported measurements are defined in Section 5.6.2b.

d/ Note: The example data are provided for illustrative purposes only.

5.7 RECORDS RETENTION

5.7.1 Records and logs relative to the following areas shall be made and retained for the life of the plant:

a. Records and drawings detailing plant design changes and modifications made to system and equipment as described in Section 5.6.3.

b. Records of all data from environmental monitoring, surveillance, and special surveillance and study activities required by these environmental technical specifications.

5.7.2 All other records and logs relating to the environmental technical specifications shall be retained for five years following logging or recording.