BEAVER VALLEY POWER STATION

UNIT NO. 2

TECHNICAL SPECIFICATIONS

BEAVER VALLEY

POWER STATION

UNIT 2

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

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UNIT NO. 2

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

1.1 DEFINED TERMS

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

1.2 THERMAL POWER:

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

1.3 RATED THERMAL POWER:

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2652 MWt.

1.4 OPERATIONAL MODE:

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

1.5 ACTION

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

1.6 OPERABLE - OPERABILITY:

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, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

1.7 REPORTABLE EVENT:

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

1.8 CONTAINMENT INTEGRITY:

CONTAINMENT INTEGRITY shall exist when:

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

   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 or Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

1.9 CHANNEL CALIBRATION

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 input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock 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.

1.10 CHANNEL CHECK:

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.

1.11 CHANNEL FUNCTIONAL TEST

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, interlocks, and/or trip functions.

1.12 CORE ALTERATION:

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.

1.13 SHUTDOWN MARGIN:

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

1.14 IDENTIFIED LEAKAGE:

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.

1.15 UNIDENTIFIED LEAKAGE:

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

1.16 PRESSURE BOUNDARY LEAKAGE:

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

1.17 CONTROLLED LEAKAGE:

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

1.18 QUADRANT POWER TILT RATION:

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.

1.19 DOSE EQUIVALENT I-131:

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/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 Regulatory Guide 1.109, Revision 1.

1.20 STAGGERED TEST BASIS:

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.

1.21 FREQUENCY NOTATION:

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

1.22 REACTOR TRIP SYSTEM RESPONSE TIME:

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.

1.23 ENGINEERED SAFETY FEATURE RESPONSE TIME:

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.

1.24 AXIAL FLUX DIFFERENCE:

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

1.25 PHYSICS TESTS:

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

1.26 $\overline{E}$ - AVERAGE DISINTEGRATION ENERGY:

$\overline{E}$ 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 noniodine activity in the coolant.

1.27 SOURCE CHECK:

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.28 PROCESS CONTROL PROGRAM:

A PROCESS CONTROL PROGRAM (PCP) shall be the manual or set of operating parameters detailing the program of sampling, analysis, and evaluation by which SOLIDIFICATION of radioactive wastes from liquid systems is assured. Requirements of the PCP are provided in Specification 6.14.

1.29 SOLIDIFICATION:

SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems into a form that meets shipping and burial ground requirements.

1.30 OFFSITE DOSE CALCULATION MANUAL (ODCM):

An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of off-site doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints. Requirements of the ODCM are provided in Specification 6.15.

1.31 GASEOUS RADWASTE TREATMENT SYSTEM:

A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.32 VENTILATION EXHAUST TREATMENT SYSTEM:

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

1.33 PURGE-PURGING:

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is required to purify the confinement.

1.34 VENTING:

VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

1.35 MAJOR CHANGES:

MAJOR CHANGES to radioactive waste systems, as addressed in Paragraph 6.16.2, (liquid, gaseous and solid) shall include the following:

1. Major changes in process equipment, components, structures and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2. Major changes in the design of radwaste treatment systems (liquid, gaseous and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3. Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4. Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

1.36 MEMBER(S) OF THE PUBLIC

MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries and persons who traverse portions of the site as the consequence of a public highway, railway, or waterway located within the confines of the site boundary. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

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 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
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

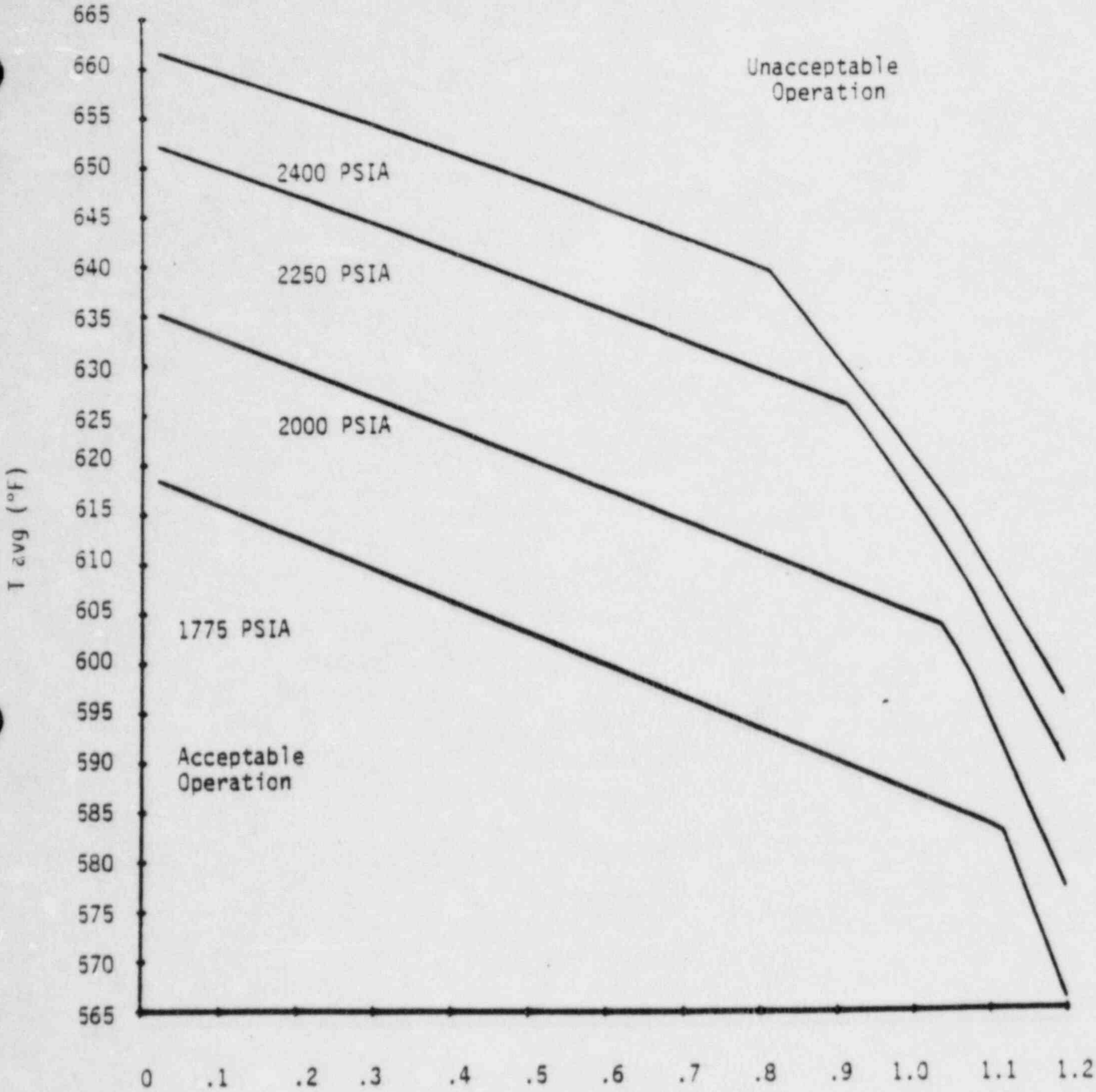
The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 3 loop operation and Figure 2.1-2 for 2 loop operation.

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.



Power (fraction of nominal)

Figure 2.1-1

Reactor Core Safety Limit - 3 Loops in Operation

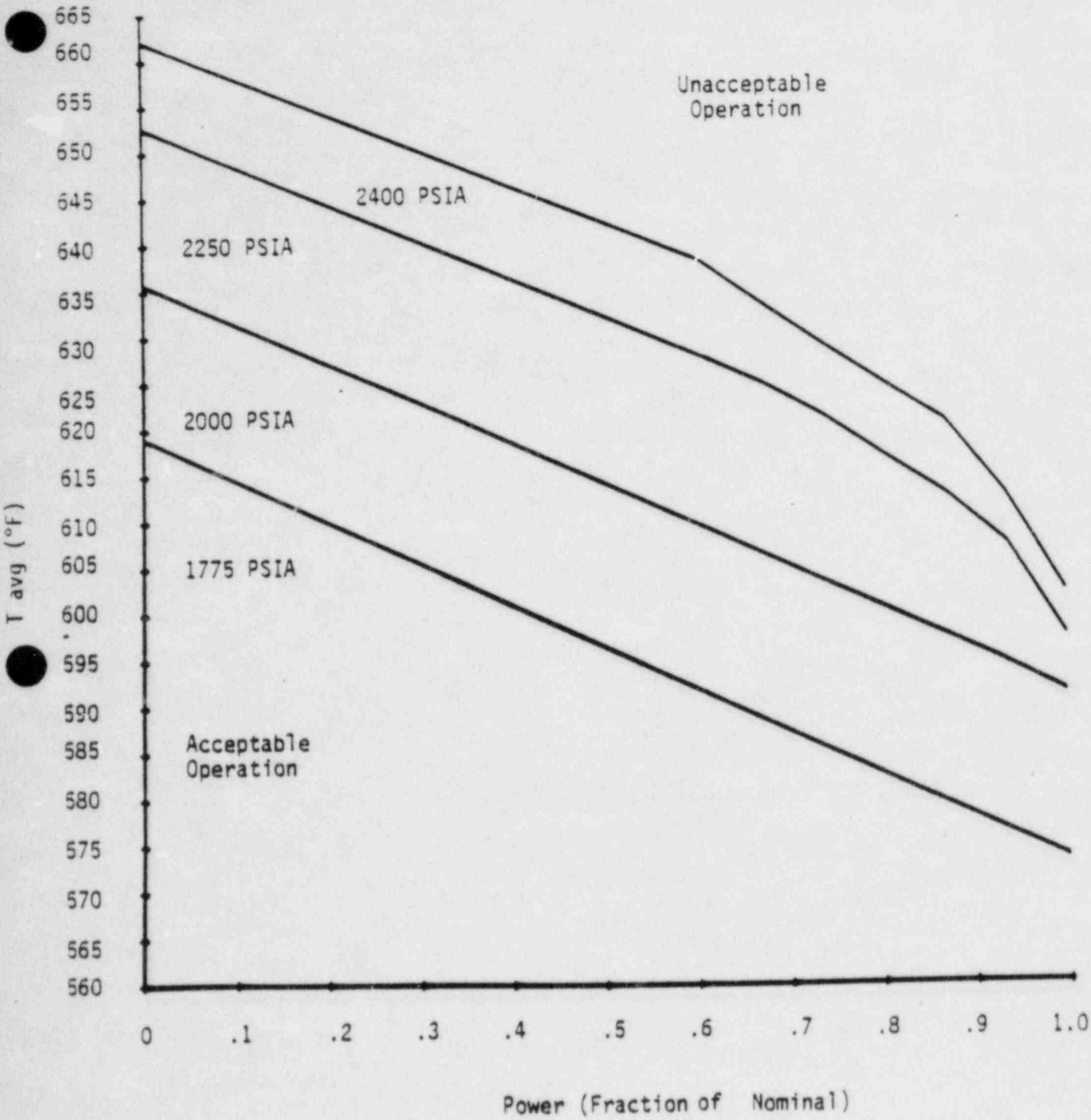


Figure 2.1-2

Reactor Core Safety - 2 Loops in Operation

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

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 five minutes.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

LIMITING CONDITION FOR OPERATION

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 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 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
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 Delta T	See Note 1	See Note 2
8. Overpower Delta T	See Note 3	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 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	≥ (later)% of narrow range instrument span-each steam generator	≥ (later)% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level
	≥ 25% of narrow range instrument span--each steam generator	≥ 24% of narrow range instrument span- each steam generator
15. Undervoltage-Reactor Coolant Pumps	≥ 75% $^{+1\%}_{-2\%}$ of nominal bus voltage-each bus	≥ 70% of nominal bus voltage-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 58.0 H_z ± 0.1 H_z -each bus	≥ 57.5 H_z -each bus
17. Turbine Trip		
a. Auto Stop Oil Pressure	45 psig	± 5 psig
b. Turbine Stop Valve	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	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

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
20. Reactor Trip System Interlocks (Based on Ascending Power)		
A. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ Amps	$\geq 6 \times 10^{-11}$ Amps
B. Power Range Neutron Flux, P-8	$\leq$ 30% RATED THERMAL POWER	$\leq$ 31% RATED THERMAL POWER
C. Power Range Neutron Flux, P-9	$\leq$ 70% RATED THERMAL POWER	$\leq$ 71% RATED THERMAL POWER
D. Power Range Neutron Flux, P-10 (Input to P-7)	> 9% RATED THERMAL POWER	< 12% RATED THERMAL POWER
E. Turbine Impulse Chamber Pressure, P-13 (Input to P-7)	$\leq$ 66 PSIG	$\leq$ 72 PSIG

TABLE 2.2-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE Delta T

$$\text{Delta T}\frac{(1 + t_1S)}{(1 + t_2S)}\left(\frac{1}{1 + t_3S}\right) \leq \text{Delta T}_0\ K_1 - K_2\frac{(1 + t_4S)}{(1 + t_5S)}\left[T\left(\frac{1}{1 + t_6S}\right) - T^1\right] + K_3(P - P^1) - f_1(\text{Delta I})$$

Where: Delta T = Measured Delta T by RTD Manifold Instrumentation;

$\frac{1 + t_1S}{1 + t_2S}$ = Lead-lag compensator on measured Delta T;

t_1, t_2 = Time constants utilized in lead-lag controller for Delta T, t_1 = 8 seconds, t_2 = 3 seconds;

$\frac{1}{1 + t_3S}$ = Lag compensator on measured Delta T;

t_3 = Time constants utilized in the lag compensator for Delta T, t_3 = 0 seconds;

Delta T_0 = Indicated Delta T at RATED THERMAL POWER;

K_1 = 1.2806 for 3 loop operation, 1.1657 for 2 loop operation;

K_2 = 0.01747;

TABLE 2.2-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION
(Continued)

NOTE 1: (Continued)

$\frac{1 + t_4 S}{1 + t_5 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

t_4, t_5 = Time constants utilized in the lead-lag controller for T_{avg}, t_4 = 30 seconds, t_5 = 4 seconds;

T = Average temperature, °F;

$\frac{1}{1 + t_6 S}$ = Lag compensator on measured T_{avg};

t_6 = Time constant utilized in the measured T_{avg} lag compensator, t_6 = 0 seconds;

T^1 $\leq$ 576.2°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.000823;

P = Pressurizer pressure, psig;

P^1 = 2235 psig (Nominal RCS operating pressure); and

TABLE 2.2-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION
(Continued)

NOTE 1: (Continued)

S = Laplace transform operator, $second^{-1}$;

and f_1 (Delta I) 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 -34% and + 10% f_1 (Delta I) = 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 -34% the Delta T Trip Setpoint shall be automatically reduced by 1.439% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +10% the Delta T Trip Setpoint shall be automatically reduced by 1.789% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.8%.

TABLE 2.2-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION
(Continued)

NOTE 3: OVERPOWER Delta T

$$\text{Delta T} \left(\frac{1 + {}_1S}{1 + {}_2S}\right) \left(\frac{1}{1 + {}_3S}\right) \leq \text{Delta } T_0 \quad K_4 - K_5 \left(\frac{{}_7S}{1 + {}_7S}\right) \left(\frac{1}{1 + {}_6S}\right) T - K_6 \left[T \left(\frac{1}{1 + {}_6S}\right) - T''\right] - f_2 \text{ (Delta T)}$$

Where: Delta T = As defined in Note 1;

$\frac{1 + t_1S}{1 + t_2S}$ = As defined in Note 1;

t_1, t_2 = As defined in Note 1;

$\frac{1}{1 + t_3S}$ = As defined in Note 1;

t_3 = As defined in Note 1;

Delta T_0 = As defined in Note 1;

K_4 = 1.0781;

TABLE 2.2-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION
(Continued)

NOTE 3: (Continued)

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

$\frac{t_7 S}{1 + t_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation;

t_7 = Time constants utilized in the rate-lag controller for T_{avg}, t_7 = 10 seconds;

$\frac{1}{1 + t_6 S}$ = As defined in Note 1;

t_6 = As defined in Note 1;

K_6 = 0.00115/°F for $T > T^{11}$ and $K_6 = 0$ for $T \leq T^{11}$;

T = As defined in Note 1;

T^{11} = As indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for Delta T instrumentation, $\leq$ 576.2°F);

TABLE 2.2-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION
(Continued)

NOTE 3: (Continued)

S = As defined in Note 1; and

f_2(Delta I) = 0 for all I

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0%.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

BASES

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 R-grid correlation. The R-grid 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 of DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

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

2.1 SAFETY LIMITS

REACTOR CORE

BASES

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 (Delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature Delta T trip will reduce the setpoint to provide protection consistent with core safety limits.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR COOLANT SYSTEM PRESSURE

BASES

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 percent (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B.31.1 which permit a maximum transient pressure of 120 percent (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 to demonstrate integrity prior to initial operation.

REACTOR TRIP SET POINTS:

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 values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its Setpoint Limit 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 to occur for each trip used in the accident 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 setpoint 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 control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

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 channels 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 DELTA T:

The Overtemperature Delta T 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 on Figures 2.1-1 and 2.1-2. 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.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

BASES

Operation with a reactor coolant loop out of service below the 3 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 2 loop operation exclusive of the Overtemperature Delta T setpoint. Two loop operation above the 3 loop P-8 setpoint is permissible after resetting the K1, K2, and K3 inputs to the Overtemperature Delta T channels and raising the P-8 setpoint to its 2 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 Delta T:

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T 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 percent of nominal full loop flow. Above 31 percent (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90 percent of nominal full loop flow. This latter trip will prevent the minimuim value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature Delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature T trip set point adjusted to the value specified for 2 loop operation, the P-8 trip at 70 percent RATED THERMAL POWER with a loop stop valve closed will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with 2 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 $\geq 1.55 \times 10^6$ pounds/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 setpoints assure a reactor trip signal is generated before the low flow trip setpoint 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-9. 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 FOR 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 two or more pump breakers above P-7. 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.

REACTOR TRIP SYSTEM INTERLOCKS

The Reactor Trip System Interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint Source Range level trips are automatically reactivated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip and the low setpoint Power Range reactor trips are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION

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

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

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

a. At least HOT STANDBY within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

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

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

3.0.5 When a system, subsystem, train, component or devide is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: 1) its corresponding normal or emergency power source is OPERABLE; and 2) all of its redundant system(s), subsystem(s), train(s), component(s), and device(s) are OPERABLE or likewise satisfy the requirements of this Specification. Unless both

LIMITING CONDITION FOR OPERATION

conditions 1) and 2) are satisfied within 2 hours, action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply, by placing it, as applicable, in:

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

This Specification is not applicable in MODES 5 or 6.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing 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).

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

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

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

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

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

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1.77% $\Delta k/k$.

APPLICABILITY:

MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN < 1.77% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 30 gpm of 7000 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 $\geq$ 1.77% $\Delta k/k$:

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 3.1.3.6.

c. When in MODE 2***, at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5 percent 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 3.1.3.6.

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* See Special Test Exception 3.10.1

** With $K_{eff} \geq 1.0$

*** With $K_{eff} < 1.0$

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

SURVEILLANCE REQUIREMENTS

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.

f. The Reactor Coolant System shall be borated to at least the cold shutdown boron concentration prior to manually blocking the Low Pressurizer Pressure Safety Injection Signal and shall remain at this boron concentration or greater at all times during which this signal is blocked.

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.

3/4.1.1 BORATION CONTROL

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 $\geq$ 30 gpm of 7000 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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the core shall be ≥ 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 core < 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 core shall be determined to be ≥ 3000 gpm 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 ≥ 3000 gpm through the core.

3/4.1.1 BORATION CONTROL

MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0×10^{-4} delta k/k/°F,

b. Less negative than -3.9×10^{-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 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 percent 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 3.10.4.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

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 when the reactor is critical.

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 > 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}) deviation alarm not reset.

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* See Special Test Exception 3.10.3.

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

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. The flow path from the boric acid storage system via a boric acid transfer pump to a charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7.a 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 3.1.2.7.b 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:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

2. Verifying that the temperature of the heat traced portion of the flow path is $\geq$ 65°F when a flow path from the boric acid tanks is used and the ambient air temperature of the Auxiliary Building is < 65°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.

3/4.1.2 BORATION SYSTEMS

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 one charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via one 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 percent $\Delta k/k$ 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:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - OPERATING

SURVEILLANCE REQUIREMENTS

2. Verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is $\geq$ 65°F when the ambient air temperature of the Auxiliary Building is < 65°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.

c. At least once per 18 months during shutdown by cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 As a minimum, one charging pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

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 The above required charging pump shall be demonstrated OPERABLE at least once per 31 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 2437* psig, and

c. Verifying pump operation for at least 15 minutes.

*To be verified during Pre-Operational Testing

3/4.1.2 BORATION SYSTEMS

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 within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 Two charging pumps shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of > 2437* psig, and

c. Verifying pump operation for at least 15 minutes.

*To be verified during Pre-Operational Testing

3/4.1.2 BORATION SYSTEMS

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 One boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1.a is OPERABLE.

APPLICABILITY:

MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1.a, 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 The above required boric acid transfer pump shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 107* psig, and

c. Verifying pump operation for at least 15 minutes.

*To be verified during Pre-Operational Testing

3/4.1.2 BORATION 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 3.1.2.2.a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2.a 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 percent $\Delta k/k$ 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 At least the above required boric acid pump shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 107* psig, and

c. Verifying pump operation for at least 15 minutes.

*To be verified during Pre-Operational Testing

3/4.1.2 BORATION SYSTEMS

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 with:

1. A minimum contained volume of 2315 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 217,000 gallons,

2. A minimum boron concentration of 2000 ppm, and

3. A minimum solution temperature of 45°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:

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.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is < 45°F.

3/4.1.2 BORATION SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system with:

1. A minimum contained volume of 13,390 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 859,248 gallons of water,

2. A boron concentration of between 2000 ppm and 2100 ppm, and

3. A minimum solution temperature of 45°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 and borated to a SHUTDOWN MARGIN equivalent to at least 1 percent k/k at 200°F within the next 6 hours; 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.

BORIC ACID TRANSFER PUMPS - OPERATING

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

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

3. Verifying the boric acid storage system solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the RWST ambient air temperature is < 45°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 shall be OPERABLE and positioned within ± 12 steps (indicated position, as determined in accordance with Specification 3.1.3.2) corresponding to their respective group demand counter 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 length rod inoperable or misaligned from the group demand counter position by more than ± 12 steps (indicated position determined in accordance with Specification 3.1.3.2), be in HOT STANDBY within 6 hours.

c. With one full length rod trippable but inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group demand counter position by more than ± 12 steps (indicated position determined in accordance with Specification 3.1.3.2), 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 remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod 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 (3.1.3.6) during subsequent operation, or

---

* See Special Test Exceptions 3.10.2 and 3.10.4

3/4.1.3 MOVEABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3. 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. The THERMAL POWER level is reduced to less than or equal to 75 percent of RATED THERMAL POWER within the hour and, within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85 percent of RATED THERMAL POWER.

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

   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.

   d. A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 Each shutdown and control rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

4.1.3.1.2 Each full length rod position shall be determined to be $\pm$ 12 steps of the associated group demand counter by verifying the individual rod position at least once per 12 hours except during intervals when the Rod Position Deviation monitor is inoperative, then verify the group position at least once per 4 hours.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes or from Cracked Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Systems Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY:

MODES 1 and 2*.

ACTION:

a. With a maximum of one rod position indicator per bank 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 less than 50 percent 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 less than 50 percent of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system 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 indication system at least once per 4 hours.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group demand position indicators shall be OPERABLE and capable of determining within ± 12 steps the demand position for each shutdown or control rod not fully inserted.

APPLICABILITY:

MODES 3*, 4*, and 5*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required group demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days when the reactor coolant system pressure is greater than 400 psig.

---

* With the reactor trip system breakers in the closed position.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be ≤ 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 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ≤ 65 percent of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 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.

c. At least once per 18 months.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 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 4.1.3.1.1, 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

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn by use of the group demand counters, and verified by the rod position indicators.

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 24 hours thereafter.

---

* See Special Test Exception 3.10.2 and 3.10.4

\# With $k_{eff} \geq 1.0$

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.1.3.6 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 4.1.3.1.1, either:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 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 at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 When the Rod Insertion Limit Monitor is OPERABLE, the deviation between the position indicated by the individual rod position instrument channel and the position indicated by the corresponding group demand indication shall be checked manually for each rod at least once per 24 hours. When the Rod Insertion Limit Monitor is inoperable, the deviation between indicated positions shall be checked** manually at least once per 4 hours.

---

* See Special Test Exception 3.10.2 and 3.10.4

\# With $k_{eff} \geq 1.0$

FIGURE 3.1-1

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER THREE LOOP OPERATION

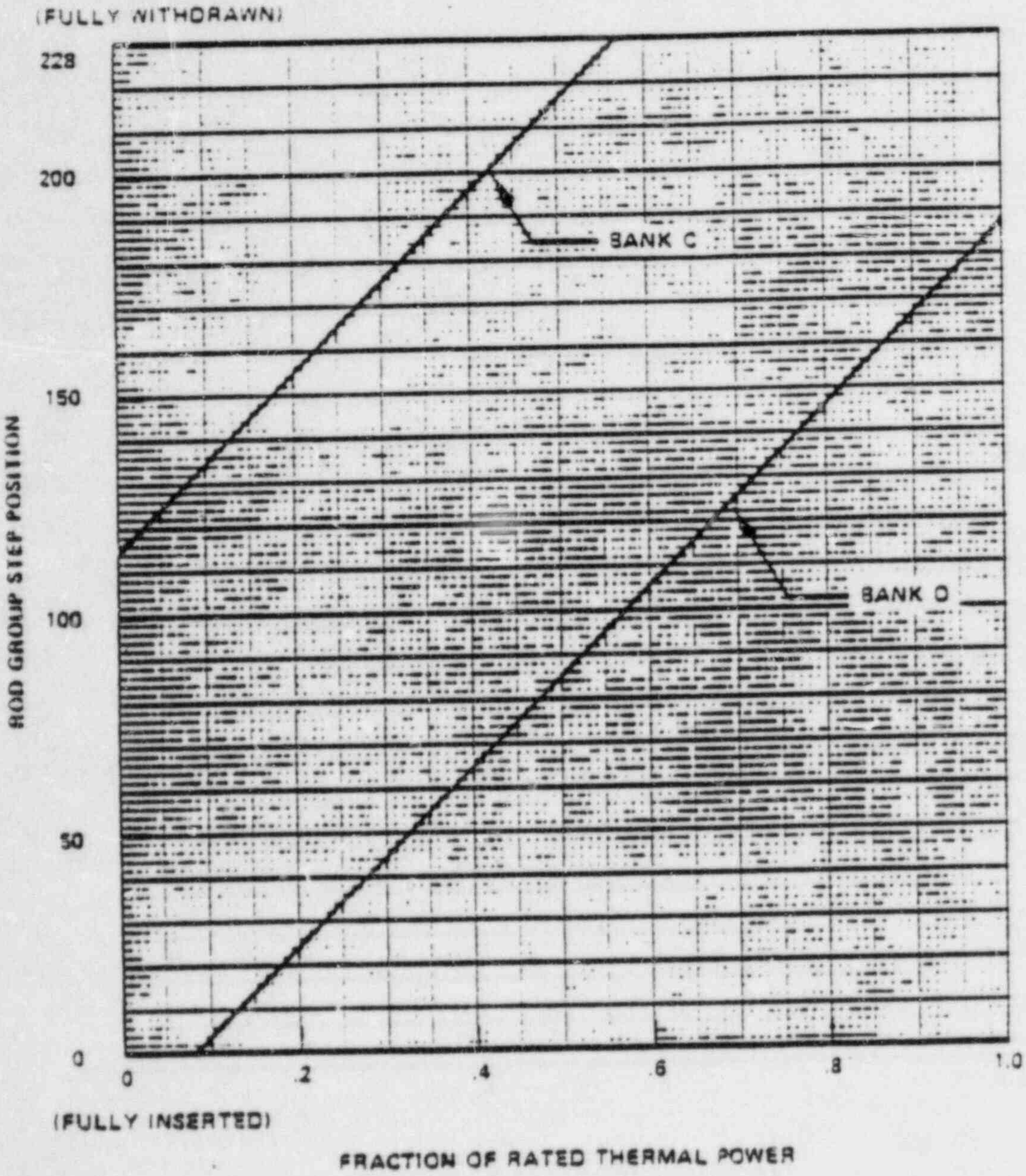
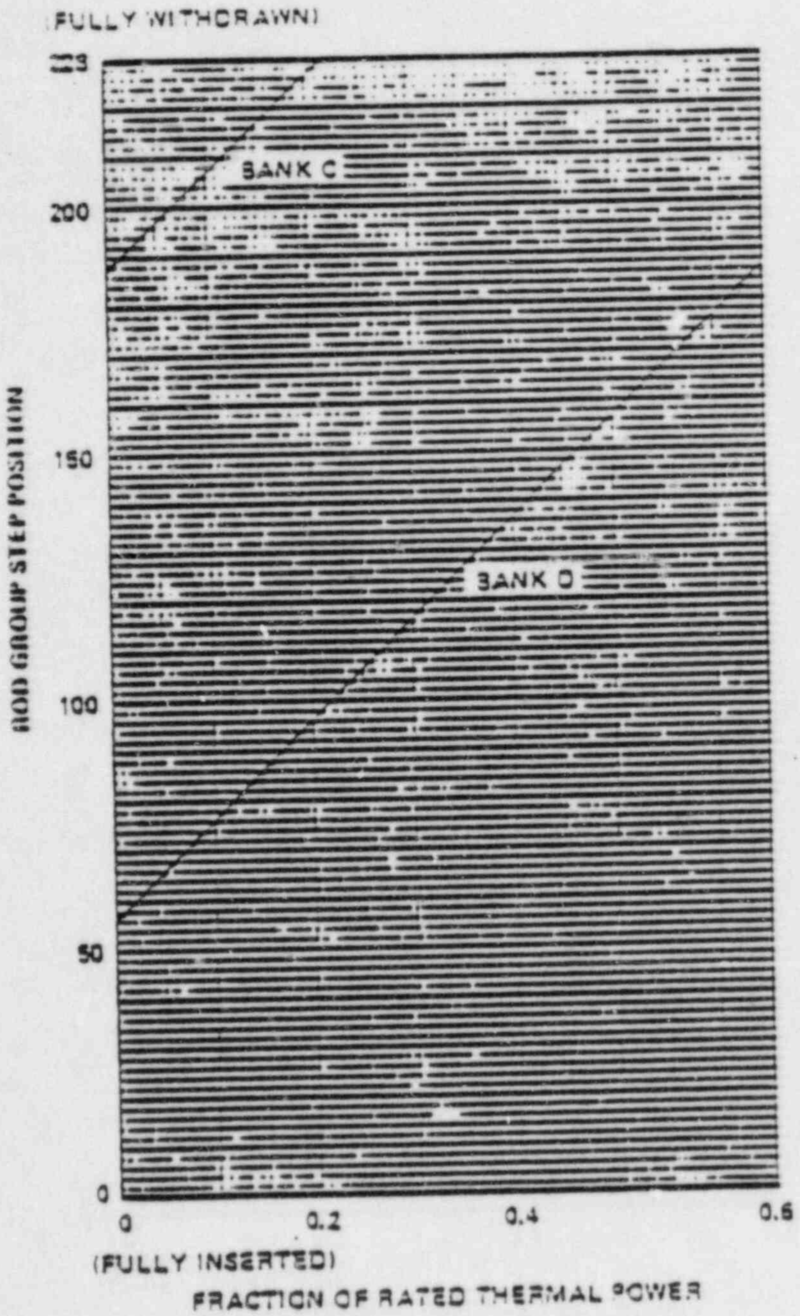


FIGURE 3.1-2

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER TWO LOOP OPERATION



3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1

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

APPLICABILITY:

MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER*.

ACTION:

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

1. Above 90 percent 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 percent of RATED THERMAL POWER.

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

   a. POWER OPERATION may continue provided:

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

* See Special Test Exception 3.10.2

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

2. The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55 percent 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.

b. THERMAL POWER shall not be increased above 90 percent of RATED THERMAL POWER unless the indicated AFD is within the $\pm$ 5 percent target band and ACTION a.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 $\pm$ 5 percent 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 percent 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.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

SURVEILLANCE REQUIREMENTS

4.2.1.2

The indicated AFD shall be considered outside of its ± 5 percent 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. POWER OPERATION outside of the ± 5 percent 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 percent 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 percent of RATED THERMAL POWER.

4.2.1.3

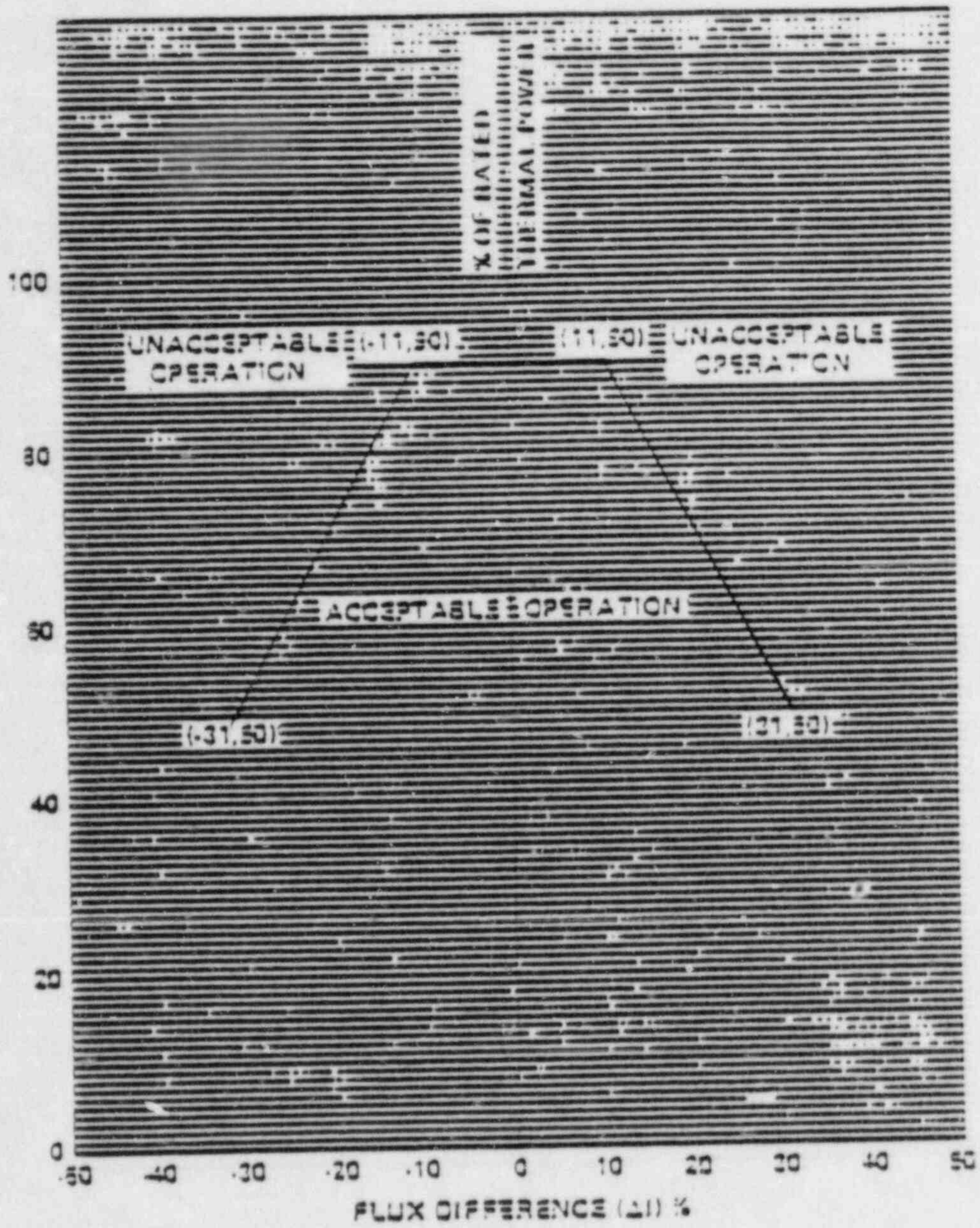
The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4

The target flux difference shall be updated at least once per 31 Effective Full Power Days be 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



3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.2 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:

$$F_Q(Z) \leq \frac{[2.18]}{P} [K(Z)] \text{ for } P>0.5$$

$$F_Q(Z) \leq [(4.36)] \quad [K(Z)] \text{ for } P\leq 0.5$$

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

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 percent for each 1 percent $F_Q(Z)$ exceeds the limit within 15 minutes and similiarly 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 T Trip Setpoints have been reduced at least 1 percent for each 1 percent $F_Q(Z)$ exceeds the limit. The Overpower T Trip Setpoint reduction shall be performed with the reactor subcritical.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; 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.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

SURVEILLANCE REQUIREMENTS

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 percent of RATED THERMAL POWER.

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

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:

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

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 :

SURVEILLANCE REQUIREMENTS

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} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.14.

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:

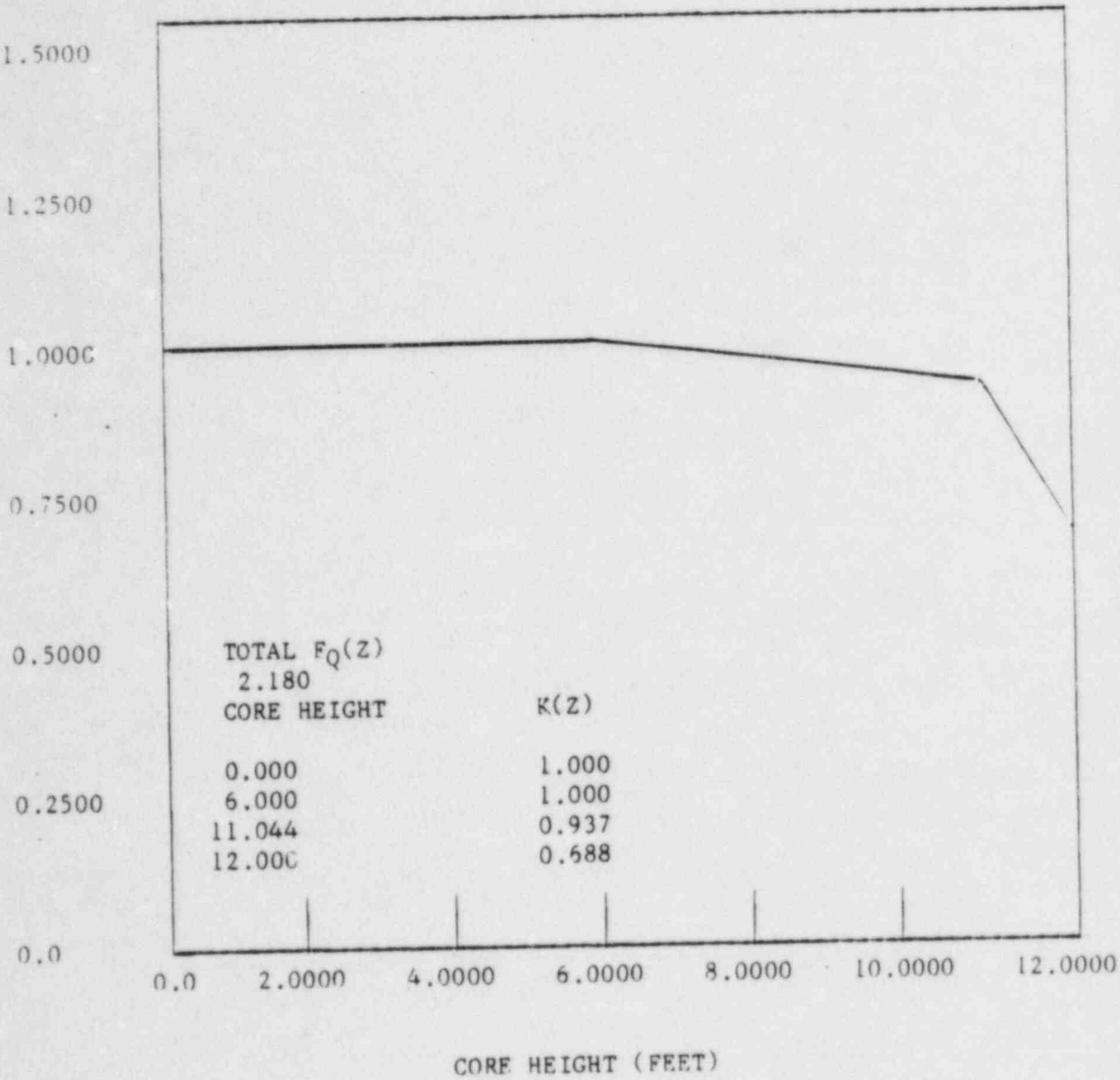
1. Lower core region from 0 to 15 percent, inclusive.
2. Upper core region from 85 to 100 percent, inclusive.
3. Grid plane regions at 17.8 $\pm$ 2 percent, 32.1 $\pm$ 2 percent, 46.4 $\pm$ 2%, 60.6 $\pm$ 2%, and 74.9 $\pm$ 2%, inclusive.
4. Core plane regions within $\pm$ 2 percent of core height ($\pm$ 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

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 percent to account for manufacturing tolerances and further increased by 5 percent to account for measurement uncertainty.

Figure 3.2-2

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



3/4.2.3 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:

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

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

APPLICABILITY:

MODE 1

ACTION:

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

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

b. Demonstrate through 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 percent 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; 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 percent of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75 percent of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95 percent or greater RATED THERMAL POWER.

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

SURVEILLANCE REQUIREMENTS

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

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

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

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

3/4.2.4 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 percent of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but $\leq$ 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 percent for each 1 percent 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 percent of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to $\leq$ 55 percent 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 percent of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95 percent or greater RATED THERMAL POWER.

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* See Special Test Exception 3.10.2.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

---

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

1. Reduce THERMAL POWER at least 3 percent for each 1 percent 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 reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55 percent 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 percent of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95 percent 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 or control rod:

1. Reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55 percent 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 percent of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified at 95 percent or greater RATED THERMAL POWER.

3/4.2.4 QUADRANT POWER TILT RATIO

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50 percent 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 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.

3/4.2.5 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 Pressurizer

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 percent 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 indicating 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: 3 Loops In Operation	LIMITS: 2 Loops In Operation and Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 581°F	$\leq$ 570°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq$ 265,500 gpm	$\geq$ 187,800 gpm

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

4.3.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. 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.

4.3.1.3 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 leat 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 Number 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, 3*, 4*, 5*	12
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1(1), 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(1), 2, 3*, 4*, 5*	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2(2), 3*, 4*, 5*	4
b. Shutdown	2	0	1	3, 4, and 5	5
7. Overtemperature Delta T					
a. Three Loop Operation	3	2	2	1, 2	2
b. Two Loop Operation	3	1**	2	1, 2	9
8. Overpower Delta T					
a. Three Loop Operation	3	2	2	1, 2	2
b. Two Loop Operation	3	1**	2	1, 2	9
9. Pressurizer Pressure-Low (Above P-7)	3	2	2	1, 2	7

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Pressurizer Pressure--High	3	2	2	1, 2	7
11. Pressurizer Water Level--High (Above P-7)	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 any 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 (Loop Stop Valves Open)	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 (Above P-7)	3-1/bus	2	2	1	7
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	3-1/bus	2	2	1	7

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Turbine Trip (Above P-9)					
a. Auto Stop Oil Pressure	3	2	2	1	7
b. Turbine Stop Valve Closure	4	4	4	1	8
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip (Above P-7)	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1, 2, 3*, 4*, 5*	1
22. Automatic Trip Logic	2	1	2	1, 2, 3*, 4*, 5*	1
23. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	1	2	3
b. Power Range Neutron Flux, P-8	4	2	3	1	12
c. Power Range Neutron Flux, P-9	4	2	3	1	12
d. Power Range Neutron Flux, P-10	4	2	3	1	12
e. Turbine Impulse Chamber Pressure, P-13	2	1	1	1	12

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.

(1) Trip function may be manually bypassed in this mode above P-10.

(2) Trip function may be manually buypassed in this mode 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.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5 percent of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing THERMAL POWER above 5 percent of RATED THERMAL POWER; otherwise reduce THERMAL POWER to less than 5 percent RATED THERMAL POWER within the following 6 hours.

b. Above 5 percent of RATED THERMAL POWER, operation may continue provided all of the following conditions are satisfied:

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

2. 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.

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

Table 3.3-1

TABLE NOTATION

ACTION 3 - 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 but below 5 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5 percent of RATED THERMAL POWER.

c. Above 5 percent 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 Specificlation 3.1.1.1 or 3.1.1.2, as applicable within 1 hour, and at least once per 12 hours thereafter.

ACTION 6 - Not Applicable.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. Less than or equal to 5 percent of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; restore the inoperable channel to operable status within 24 hours after increasing THERMAL POWER above 5 percent of RATED THERMAL POWER; otherwise reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the following 6 hours.

b. Above 5 percent of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

Table 3.3-1

TABLE NOTATION

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level above P-7, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 9 - With a channel associated with an operation 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 - Not applicable.

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.

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 (High and Low Setpoint)	$\leq$ 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 Delta T	$\leq$ 4.0 seconds*
8. Overpower Delta T	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 shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2

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	$\leq$ 1.5 seconds
17. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.9 seconds
18. Turbine Trip	
a. Auto Stop 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
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable
23. Reactor Trip System Interlocks	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				
a. High Setpoint	S	D(2), M(3) and Q(6)	M	1, 2
b. Low Setpoint	S	N/A	S/U(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	N/A	S/U(1), M(7)	1, 2, 3*, 4*, 5*
6. Source Range, Neutron Flux (Below P-10)	N/A	N/A	S/U(1), M(8)	2, 3*, 4*, 5*
7. Overtemperature Delta T	S	R	M	1, 2
8. Overpower Delta T	S	R	M	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R	M	1, 2
10. Pressurizer Pressure--High	S	R	M	1, 2
11. Pressurizer Water Level--High (Above P-7)	S	R	M	1, 2
12. Loss of Flow - Single Loop	S	R	M	1
13. Loss of Flow - Two Loops	S	R	N/A	1
14. Steam Generator Water Level--Low-Low	S	R	M	1, 2

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage - Reactor Coolant Pumps (Above P-7)	N/A	R	M	1
17. Underfrequency - Reactor Coolant Pumps (Above P-7)	N/A	R	M	1
18. Turbine Trip (Above P-9)				
a. Auto Stop 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 ESF	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, 5*
22. Automatic Trip Logic	N/A	N/A	M(5)	1, 2, 5*
23. Reactor Trip System Interlocks				
a. P-6	N/A	N/A	M(9)	1, 2
b. P-8	N/A	N/A	M(9)	1
c. P-9	N/A	N/A	M(9)	1
d. P-10	N/A	N/A	M(9)	1
e. P-13	N/A	N/A	M(9)	1

TABLE 4.3-1

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 imbalance above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

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

(5) - Each train tested every other month.

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

(7) - Below P-10.

(8) - Below P-6.

(9) - Required only when below Interlock Trip Setpoint.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The engineered safety feature actuation system 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 engineered safety feature actuation system 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 engineered safety feature actuation system instrumentation channel inoperable, take the action shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each engineered safety feature actuation 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 4.3-2.

4.3.2.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. 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.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF 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 ESF 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 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, 36
c. Containment Pressure - High	3	2	2	1, 2, 3	14
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	14
e. Low Steamline Pressure					
Three Loops Operating	3/loop	2/loop any loop	2/loop any loop	1, 2, 3#	14
Two Loops Operating	3/loop	2/loop any loop	2/loop any loop	1, 2, 3#	15

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
1.1 SAFETY INJECTION - TRANSFER FROM INJECTION TO THE RECIRCULATION MODE					
a. Automatic Actuation Logic Coincident with Safety Injection Signal	2	1	2	1, 2, 3	18
b. Refueling Water Storage Tank Level - Low	4	2	3	1, 2, 3	16
c. Refueling Water Storage Tank Level - Auto Quench Spray Flow Reduction	1	1	1	1, 2, 3	18

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
2. CONTAINMENT SPRAY					
a. Manual*	2 sets	1 set of 2 switches	2 sets	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
b. Phase "B" Isolation					
1) Manual*	2 sets (2 switches/sets)	1 set	2 sets	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

*Manual actuation of containment spray is accomplished by actuating either of two sets (two switches per set). Both switches in a set must be actuated to obtain a manually initiated containment depressurization signal per train.

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
4. STEAM LINE ISOLATION					
a. Manual	1/loop*	1/loop	1/loop*	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure - Intermediate High High	3	2	3	1, 2, 3	14
d. Low Steamline Pressure					
Three Loops Operating (Loop Stop Valves Open)	3/loop	2/loop any loop	2/loop any loop	1, 2, 3#	14
Two Loops Operating	3/loop	2/loop any operating loop	2/loop any operating loop	1, 2, 3#	15
e. High Steam Pressure Rate	3/loop	2/loop any loop	2/operating loop	3#, 4	37

*Additionally, there will be two sets of control switches (two momentary controls per set) on the main control board. Operating either set will actuate all three main steamline stop and bypass valves at the system level.

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
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level - High-High, P-14	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14
6. LOSS OF POWER					
a. 4.16kv Bus				1, 2, 3, 4	33
1) Loss of Voltage (trip feeder)	1/4.16kv Bus	1/4.16kv Bus	1/4kv Bus		
2) Loss of Voltage (start diesel)	1/4.16kv Bus	1/4.16kv Bus	1/4kv Bus		
b. Grid Degraded Voltage (4.16kv Bus)	2/4.16kv Bus	2/Bus	2/Bus	1, 2, 3, 4	34
c. Grid Degraded Voltage (480v Bus)	2/480v Bus	2/Bus	2/Bus	1, 2, 3, 4	34

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

Functional Unit	fotal No. of Channels	Channels to Trip	Minimum Channels Operable	Applicable Modes	Action
7. AUXILIARY FEEDWATER					
a. Steam Generator Water Level - Low-Low					
i. Start Turbine Driven Pump	3/steam generator	2/steam generator any steam generator	2/steam generator	1, 2, 3	14
ii. Start Motor Driven Pump	3/steam generator	2/steam generator any 2 steam generator	2/steam generator	1, 2, 3	14
b. Undervoltage-RCP Start Turbine Driven Pump	(3)-1/bus	2	2	1	14
c. S. I. Start Motor-Driven Pumps	See 1 above (all S.I. initiating functions and requirements)				
d. Turbine-Driven Pump Discharge Pressure Low Wit Steam Valves Open (Start Motor-Driven Pumps)	(2)-1/train	1	1	1, 2, 3	18
e. Trip of Main Feedwater Pumps Start Motor-Driven Pumps	1/pump	1	1	1, 2, 3	18
f. Emergency Bus Undervoltage Start Motor-Driven Pumps	1/bus	1	1	1, 2, 3	18

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
8. ESF INTERLOCKS					
a. Reactor Trip, P-4	2	1	2	1, 2, 3	38
b. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	38
c. Low-Low Tavg, P-12	3	2	2	1, 2, 3	38

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

Trip function automatically bypassed above P-11, and is bypassed below P-11 when Safety Injection on low steam pressure is not manually bypassed.

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 n COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing in accordance with Specification 4.3.2.1, provided the other channel is operable.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the tripped condition within 1 hour: restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT STANDBY within the following 6 hours.

b. Above P-11 and P-12, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

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 in accordance with Specification 4.3.2.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the bypass condition; restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT SHUTDOWN within the following 12 hours.

b. Above P-11 or P-12, demonstrate that the Minimum Channels OPERABLE requirement is met within 1 hour; operation may continue with the inoperable channel bypassed and one additional channel may bne bypassed for up to 2 hours for surveillance testing in accordance with Sepcification 4.3.2.1.

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 COLD SHUTDOWN within the following 30 hours.

ACTION 33 - With the number of OPERABLE Channels one less than the Total Number of Channels, the Emergency Diesel Generator associated with the 4kv Bus shall be declared inoperable and the ACTION Statements for Specifications 3.8.1.1 or 3.8.1.2, as appropriate, shall apply.

ACTION 34 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until the performance of the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 36 - The block of the automatic actuation logic introduced by a reset of safety injection shall be removed by resetting (closure) of the reactor trip breakers within one hour of an inadvertent initiation of safety injection providing that all trip input signals have reset due to stable plant conditions. Manual block permitted after Safety Injection System and P-4 reset. Otherwise, the requirements of action statement 13 shall have been met.

ACTION 37 - 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 a tripped condition within one hour.

b. The Minimum Channels OPERABLE requirements is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per specification 4.3.2.1.

ACTION 38 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) (bistable status lights or computer checks) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Trip Setpoint	Allowable Values
1. SAFETY INJECTION AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure - High	$\leq$ (later) psig	$\leq$ (later) psig
d. Pressurizer Pressure - Low	$\geq$ (later) psig	$\geq$ (later) psig
e. Steam Line Pressure - Low	$\geq$ (later) psig steam line pressure	$\geq$ (later) psig steam line pressure
1.1 SAFETY INJECTION - TRANSFER FROM INJECTION TO THE RECIRCULATION MODE		
a. Automatic Actuation Logic Coincident with Safety Injection Signal	Not Applicable	Not Applicable
b. Refueling Water Storage Tank Level - Low	19'2-1/2" $\pm$ 0'6"	19' -1/2" $\pm$ 1'0"
c. Refueling Water Storage Tank Level - Auto Quench Pump Trip	(later)	(later)

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	$\leq$ (later) psig	$\leq$ (later) 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 Actuation Logic	Not Applicable	Not Applicable
3) Containment Pressure - High High	$\leq$ (later) psig	$\leq$ (later) psig

E 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
TRIP SETPOINTS

Functional Unit	Trip Setpoint	Allowable Values
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure - Intermediate High High	$\leq$ (later) psig	$\leq$ (later) psig
d. Steam Line Pressure - Low	$\geq$ (later) psig steam line pressure	$\geq$ (later) psig steam line pressure
e. High Steam Pressure Rate	-100 psi	-110 psi
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level - High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Trip Setpoint	Allowable Values
6. LOSS OF POWER		
a. 1) 4.16kv Emergency Bus Undervoltage (Loss of Voltage) (Trip Feed)	75% $^{+1\%}_{-2\%}$ of nominal bus voltage with a 1 ± 0.1 second time delay	≥ 70% of nominal bus voltage with a 1 ± 0.1 second time delay
2) 4.16kv Emergency Bus (Start Diesel)	75% $^{+1\%}_{-2\%}$ of nominal bus voltage, 20 cycles ± 2 cycles	≥ 70% of nominal bus voltage, 20 cycles ± 2 cycles
b. 4.16kv Emergency Bus Undervoltage (Degraded Voltage)	90% $^{+3\%}_{-1\%}$ of nomianl bus voltage with a 90 ± 5 second time delay	≥ 89% of nominal bus voltage with a 90 ± 5 second time delay
c. 480v Emergency Bus Undervoltage (Degraded Voltage)	90% $^{+3\%}_{-1\%}$ of nominal bus voltage with a 90 ± 5 second time delay	≥ 89% of nominal bus voltage with a 90 ± 5 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
TRIP SETPOINTS

Functional Unit	Trip Setpoint	Allowable Values
7. AUXILIARY FEEDWATER		
a. Steam Generator Water Level - Low-Low	$\geq$ (later) of narrow range instrument span each steam generator	$\geq$ (later) of narrow range instrument span each steam generator
b. Undervoltage - RCP	$\geq$ 75% $^{+1\%}_{-2\%}$ of nominal bus voltage	$\geq$ 70% of nominal RCP bus voltage
c. S. I.	See 1 above (all S. I. Setpoints)	
d. Turbine-Driven Auxiliary Feed Pump Discharge Pressure Low with Steam Valves Open	Disch pressure > 468 psig with steam inlet valves open	Disch pressure > 452 psig with steam inlet valves open
e. Trip of Main Feedwater Pumps	Not Applicable	Not Applicable
f. Emergency Bus Undervoltage	$\leq$ (later) volts	$\leq$ (later) volts
8. ESF INTERLOCKS		
a. P-4	N/A	N/A
b. P-11	$\leq$ 2000 psig	$\leq$ 2010 psig
c. P-12	(later)	(later)

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 Vent and Purge Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
b. Containment Quench Spray Pumps	Not Applicable
Containment Quench Spray Valves	Not Applicable
Containment Isolation - Phase "B"	Not Applicable
c. Containment Isolation - Phase "A"	Not Applicable
d. Control Room Ventilation Isolation	Not Applicable
2. Containment Pressure - High	
a. Safety Injection (ECCS)	$\leq$ 27.0*
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0(1)
d. Containment Isolation - Phase "A"	$\leq$ (later)
e. Auxiliary Feedwater Pumps	Not Applicable
f. Service Water System	$\leq$ (later)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

Initiating Signal and Function	Response Time in Seconds
3. Pressurizer Pressure - Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/12.0#
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0(1)
d. Containment Isolation - Phase "A"	$\leq$ (later)
e. Auxiliary Feedwater Pumps	Not Applicable
f. Service River Water System	$\leq$ (later)
4. Steam Line Pressure - Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/12.0#
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0(1)
d. Containment Isolation - Phase "A"	$\leq$ (later)
e. Auxiliary Feedwater Pumps	Not Applicable
f. Service River Water System	$\leq$ (later)
g. Steam Line Isolation	$\leq$ 7.0
5. Containment Pressure - High-High	
a. Containment Quench Spray	$\leq$ (later)
b. Containment Isolation - Phase "B"	Not Applicable
c. Control Room Ventilation Isolation	$\leq$ (later)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

Initiating Signal and Function	Response Time in Seconds
6. Steam Generator Water Level - High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 7.0(1)
7. Containment Pressure - Intermediate High High	
a. Steam Line Isolation	$\leq$ 7.0
8. Steamline Pressure Rate - High-High	
a. Steam Line Isolation	$\leq$ 7.0
9. Loss of Power	
a. 4.16kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$ 10
b. 4.16kv and 480v Emergency Bus Undervoltage (Degraded Voltage	(later)
10. (Intentionally blank.)	
11. Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps**	$\leq$ 60.0
b. Turbine-driven Auxiliary Feedwater Pumps***	$\leq$ 60.0

NOTE: Response time for Motor-driven Auxiliary Feedwater Pumps on all S.I. signal starts.

** on 2/3 in 2/3 Steam Generators
*** on 2/3 any Steam Generators

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

Initiating Signal and Function	Response Time in Seconds
12. Undervoltage RCP	
a. Turbine-driven Auxiliary Feedwater Pumps	≤ 60.0
13. Emergency Bus Undervoltage	
a. Motor-driven Auxiliary Feedwater Pump	≤ 60.0
14. Trip of Main Feedwater Pumps	
a. Motor-driven Auxiliary Feedwater Pump	≤ 60.0
15. Turbine - Driven Auxiliary Feed Pump Discharge Pressure Low with Steam Valves Open	
a. Steam Driven Auxiliary Feed Pumps	≤ 60.0

NOTE: Response time for Motor-driven Auxiliary Feedwater Pumps on all S.I. signal starts.

** on 2/3 in 2/3 Steam Generators
*** on 2/3 any Steam Generators

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 and Low Head Safety Injection 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.

(1) Feedwater system overall response time shall include verification of each individual feedwater system valve shown in Table 3.6-1.

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 AND FEEDWATER ISOLATION				
a. Manual Initiation	N/A	N/A	M(1)	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	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Steam Line Pressure - Low	S	R	M	1, 2, 3
1. SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RE-RECIRCULATION MODE				
a. Automatic Actuation Logic Coincident with Safety Injection Signal	N/A	N/A	M(2)	1, 2, 3
b. Refueling Water Storage Tank Level - Low	S	R	M	1, 2, 3
c. Refueling Water Storage Tank Level - Auto Quench Pump Trip	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N/A	N/A	M(2)	1, 2, 3, 4
c. Contain Pressure - High High	S	R	M	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	M(1)	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	M(1)	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Logic	N/A	N/A	M(2)	1, 2, 3, 4
3) Contain Pressure - High-High	S	R	M	1, 2, 3
4. STEAM LINE ISOLATION				
a. Manual	N/A	N/A	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N/A	N/A	M(2)	1, 2, 3, 4
c. Containment Pressure - Intermediate High-High	S	R	M	1, 2, 3
d. Steam Line Pressure - Low	S	R	M	1, 2, 3
e. Steam Line Pressure Rate - High	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

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
6. LOSS OF POWER				
a. 4.16kv Emergency Bus Undervoltage (Loss of Voltage) Trip Feed and Start Diesel	N/A	R	M	1, 2, 3, 4
b. 4.16kv and 480v Emergency Bus Undervoltage (Degraded Voltage)	N/A	R	M	1, 2, 3, 4
7. AUXILIARY FEEDWATER				
a. Steam Generator Water Level - Low-Low	S	R	M	1, 2, 3
b. Undervoltage - RCP	S	R	M	1, 2
c. S. I.	See 1 above (all S. I. surveillance requirements)			
d. Turbine Driven Auxiliary Feed Pump Discharge Pressure Low With Steam Valves Open	N/A	R	R	1, 2, 3
e. Trip of Main Feedwater Pumps	N/A	N/A	R	1, 2, 3
f. Emergency Bus Undervoltage	N/A	R	R	1, 2, 3
8. ESF INTERLOCKS				
a. P-4	N/A	N/A	R	1, 2, 3
b. P-11	N/A	R	M	1, 2, 3
c. P-12	N/A	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Table Notation

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

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

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

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.

APPLICABLE:

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 SYSTEM

Instruments	Minimum Channels Operable	Applicable Modes	Alarm/Trip Setpoint	Measurement Range	Action
1. Area Monitors					
a. Fuel Pool Area 2RMF-RQI202	1	*	<15mR/hr	10^{-1} to 10^{4}mR/hr	19
b. Containment 2RMR-RQI201	1	1,2,3,4	(later)	1 to 10^{5}mR/hr	
2. Airborne Monitors					
a. Fuel Building Vent 2RMF-RQI301					
1) Particulate (I^{131})	1	**	($\leq$2xbackground)	10^{-10} to 10^{-5}uCi/cc	21
2) Gas (Xe^{133})	1	**	($\leq$2xbackground)	10^{-6} to 10^{-1}uCi/cc	21
b. Containment					
1) RCS Leakage 2RMR*RQI303					
a) Particulate (I^{131})	1	1,2,3,4	(later)	10^{-10} to 10^{-4}uCi/cc	20
b) Gas (Xe^{133})	1	1,2,3,4	(later)	10^{-6} to 10^{-1}uCi/cc	20

TABLE 3.3-6 (Continued)

RADIATION MONITORING SYSTEM

Instruments	Minimum Channels Operable	Applicable Modes	Alarm/Trip Setpoint	Measurement Range	Action
3. Process Monitors (Gaseous)					
a. Containment Purge Exhaust 2HVR*RQI104					
1) Gas (Xe^{133})	1	6	($\leq$2xbackground)	10^{-6} to 10^{-1}uCi/cc	22
b. Leak Collection Ventilation 2RMR-RQI301					
1) Particulate (I^{131})	1	1,2,3,4,5	(later)	10^{-10} to 10^{-6}uCi/cc	36
2) Gaseous (xe^{133})	1	1,2,3,4,5	(later)	10^{-6} to 10^{-1}uCi/cc	36

TABLE 3.3-6 (Continued)

RADIATION MONITORING SYSTEM

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 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specifications 3.9.12 and 3.9.13.

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.

Action 36 - With the Number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool.

TABLE 4.3-3

RADIATION MONITORING SYSTEM SURVEILLANCE

Instrument	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
1. Area Monitors				
a. Fuel Pool Area 2RMF-RQI202	S	R	M	*
b. Containment 2RMR-RQI201	S	R	M	1, 2, 3, 4
2. Airborne Monitors				
a. Fuel Building Vent 2RMF-RQI301	S	R	M	**
b. Containment 2RMR*RQI303	S	R	M	1, 2, 3
3. Process Monitors (Gaseous)				
a. Containment Purge Exhaust 2HVR*RQI104	S	R	M	6
b. Leak Collection Ventilation 2RMR-RQI301	S	R	M	1, 2, 3, 4, 5

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

3/4.3.3 MONITORING INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75 percent 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 axial flux offset detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of F^N_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 incore movable detection system shall be demonstrated OPERABLE by normalizing each detector output to be used within 24 hours prior to its use when required for:

a. Recalibration of the excore axial flux offset detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of F^N_H and $F_Q(Z)$.

3/4.3.3 MONITORING INSTRUMENTATION

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 the number of OPERABLE seismic monitoring instruments less than required by Table 3.3-7, restore the inoperable instrument(s) to OPERABLE status within 30 days.

b. 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.

c. 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 and a CHANNEL CALIBRATION performed within 24 hours 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

Instrument and Sensor Locations	Measurement Range	Minimum Instruments Operable
1. TRIAXIAL TIME-HISTORY ACCELOGRAPH		
a. Contain Mat el. 692'-11"	± 1g	1*
b. Contain Operating Floor el. 767'-10"	± 1g	1*
c. Switchyard	± 1g	1*
2. TRIAXIAL PEAK ACCELOGRAPH		
a. On top of RHS heat exchanger (2RSH-E21A)	± 1g	1
b. Six inch SI pipe (2SIS-006-269-1(A) el 741'-5"	± 1g	1
c. MCC*2-E03 el. 755'-6" (PAB)	± 1g	1
3. TRIAXIAL SEISMIC SWITCH		
a. Containment mat	N/A	1*
4. TRIAXIAL RESPONSE-SPECTRUM RECORDER		
a. Steam Generator Support cubicle No. 1 el. 718'-6"	0-1.2 g.	1
b. Center of PAB el. 718'-6"	0-1.2 g	1
c. MCC*2-E03 el. 755'-6"	0-1.2 g	1
d. Containment mat (accelerometer from the Time History Accelograph)	0-1.2 g	1

* With reactor control room indication.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument and Sensor Locations	Channel Check	Channel Calibration	Channel Functional Test
1. TRIAXIAL TIME-HISTORY ACCELOGRAPH			
a. Containment Mat. el. 692'-11"	M*	R	SA
b. Containment Operating floor el. 767'-10"	M*	R	SA
c. Switchyard	M*	R	SA
2. TRIAXIAL PEAK ACCELOGRAPHS			
a. On top RHS heat exchanger (2RHS-E21A)	N/A	R	N/A
b. Six inch SI pipe (2SIS-006-269-1(A)) el. 741'-5"	N/A	R	N/A
c. MCC*2E03 el 755-6"	N/A	R	N/A
3. TRIAXIAL SEISMIC SWITCH			
a. Containment mat	N/A	N/A	R

* Except seismic trigger.

TABLE 4.3-4 (Continued)
SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument and Sensor Locations	Channel Check	Channel Calibration	Channel Functional Test
4. TRIAXIAL RESPONSE SPECTRUM RECORDER			
a. Steam Generator Support Cubicle No. 1 el. 718'-6"	N/A	R	N/A
b. Center of PAB el. 718'-6"	N/A	R	N/A
c. MCC*2-E03 el. 755'-6"	N/A	R	N/A
d. Containment Mat (accelerometer from the Time-History Accelograph)	N/A	R	N/A

3/4.3.3 MONITORING 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 the number of OPERABLE meteorological monitoring channels less than required by Table 3.3-8, suspend all release of gaseous radioactive material from the radwaste gas decay tanks until the inoperable channel(s) is restored to OPERABLE status.

b. 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.

c. 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	Instrument Minimum Accuracy	Minimum Operable
1. WIND SPEED			
a. Nominal Elevation 500'		$\pm$0.5 mph*	Any
b. Nominal Elevation 150'		$\pm$0.5 mph*	2 of 3
c. Nominal Elevation 35'		$\pm$0.5 mph*	
2. WIND DIRECTION			
a. Nominal Elevation 500'		$\pm$5°	Any
b. Nominal Elevation 150'		$\pm$5°	2 of 3
c. Nominal Elevation 35'		$\pm$5°	
3. AIR TEMPERATURE Delta T			
a. T Elevation 500'-35'		$\pm$0.1°C	Any
b. T Elevation 150'-35'		$\pm$0.1°C	1 of 2

* Starting speed of anemometer shall be < 1 mph.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Channel Calibration
1. WIND SPEED		
a. Nominal Elevation 500'	D	SA
b. Nominal Elevation 150'	D	SA
c. Nominal Elevation 35'	D	SA
2. WIND DIRECTION		
a. Nominal Elevation 500'	D	SA
b. Nominal Elevation 150'	D	SA
c. Nominal Elevation 35'	D	SA
3. AIR TEMPERATURE Delta T		
a. Delta T Elevation 500'-35'	D	SA
b. Delta T Elevation 150'-35'	D	SA

3/4.3.3 MONITORING INSTRUMENTATION

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:

With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either:

a. Restore the inoperable channel to OPERABLE status within 30 days, or

b. Be in HOT SHUTDOWN within the next 12 hours.

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

INSTRUMENTS*	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Intermediate Range Nuclear Flux	10^{-11} to 10^{-3} Amps	1
2. Intermediate Range Startup Rate	-5 to +5 dpm	1
3. Source Range Nuclear Flux	10^{0} - 10^{6} cps	1
4. Source Range Startup Rate	-5 to +5 dpm	1
5. Reactor Coolant Temperature - Hot Leg	0-700°F	1
6. Reactor Coolant Temperature - Cold Leg	0-700°F	1
7. Pressurizer Pressure	1700-2500 psig	1
8. Pressurizer Level	0-100%	1
9. Steam Generator Pressure	0-1400 psig	1/Steam generator
10. Steam Generator Level	0-100%	1/Steam generator
11. RHR Return to Loop Temperature	50°-400°F	1
12. Auxiliary Feedwater Flow	0-400 gpm	1/Steam generator

* Emergency Shutdown Panel

TABLE 4.3-6

REMOTE SHUTDOWN MONITORINC INSTRUMENTATION SURVEILLANCE

INSTRUMENTS*	CHANNEL CHECK	CHANNEL CALIBRATION
1. Intermediate Range Nuclear Flux	M	N/A
2. Intermediate Range Startup Rate	M	N/A
3. Source Range Nuclear Flux	M	N/A
4. Source Range Startup Rate	M	N/A
5. Reactor Coolant Temperature - Hot Leg	M	R
6. Reactor Coolant Temperature - Cold Leg	M	R
7. Pressurizer Pressure	M	R
8. Pressurizer Level	M	R
9. Steam Generator Pressure	M	R
10. Steam Generator Level	M	R
11. RHR Return to Loop Temperature	M	R
12. Auxiliary Feedwater Flow	S/U$^{(1)}$	R

NOTE 1: Channel Check to be performed in conjunction with surveillance requirement 4.7.1.2.a.9 following an extended outage.

* Emegency Shutdown Panel

3/4.3.3 MONITORING INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY:

Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instruments less than required by Table 3.3-10:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.6.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per six months by performance of CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous six months.

4.3.3.6.2 The NFPA Code 72D Class A supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.6.3 The non-supervised circuits between the local panels in Specification 4.3.3.6.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

Table 3.3-10

FIRE DETECTION INSTRUMENTS

Location	Detectors: Smoke Total/Minimum Operable	Detectors: Heat Total/Minimum Operable
1. Reactor Containment		
a. Smoke		
Zone 64 (cable penetration area)	19/10	
Zone 65 (2RHS*P21A,B)	2/1	
b. Sprinkler/deluge		
2RHS*P21A		3/2
2RHS*P21B		3/2
cable penetration area		6/3
2. Control Building		
a. Smoke		
Zone 11 (comm., instrument, and rely below floor)	16/8	
Zone 10 (instrument and relay room ceiling mount)	14/7	
Zone 23 (cable spreading room)	23/12	
Zone 17 (control room)	12/6	
Zone 18 (computer room)	2/1	
Zone 19 (equipment room)	2/1	
Zone 20 (equipment room duct mount)	2/1	
Zone 14 (west comm. room)	2/1	
Zone 12 (MCC room)	2/1	
b. Carbon Dioxide		
Zone 1 (instrument and relay room)		14/7
Zone 1 (cable spreading area)		10/5

Table 3.3-10 (Continued)

FIRE DETECTION INSTRUMENTS

Location	Detectors: Smoke Total/Minimum Operable	Detectors: Heat Total/Minimum Operable
3. Service Building		
a. Smoke		
Zone 01 (emergency swgr wert)	24/12	
Zone 02 (emergency swgr east)	22/11	
Zone 03 (battery rooms 2-1 to 2-4)	8/1 per room	
Zone 4A (service building cable trays north)	24/12	
Zone 4B (service building cable trays south)	23/12	
b. Carbon Dioxide		
Zone 4 (cable spreading area)		26/13
4. Diesel Generator Building (EG2-2)		
a. Smoke		
Zone 62 (die. generator north)	4/2	
b. Carbon Dioxide		
Zone 6 (die. generator room)		7/4
5. Diesel Generator Building (EG2-1)		
a. Smoke		
Zone 62 (die. generator south)	4/2	
b. Carbon Dioxide		
Zone 5 (die. generator room)		7/4
6. Primary Intake Structure		
Protection provided by the BVPS-1 control room detection and alarms.		

Table 3.3-10 (Continued)

FIRE DETECTION INSTRUMENTS

Location	Detectors: Smoke Total/Minimum Operable	Detectors: Heat Total/Minimum Operable
7. Safeguards Building		
a. Smoke		
Zone 26 (pump area north)	6/3	
Zone 27 (pump area south)	6/3	
Zone 28 (swgr north)	2/1	
Zone 28 (swgr north, duct mount)	1/1	
Zone 29 (swgr south)	2/1	
Zone 29 (swgr south, duct mount)	1/1	
b. Sprinkler/deluge		
2FWE*P22		2/1
2FWE*P23A		2/1
2FWE*P23B		2/1
8. Fuel and Decontamination Building		
a. Smoke		
Zone 66 (2FNC*P21A,B)	2/1	
9. Primary Auxiliary Building		
a. Smoke		
Zone 51B (charging pumps)	13/7	
Zone 51A (2CCP*P21A,B,C and storage area)	19/10	
Zone 52B (supp. leak collection filters)	19/10	
b. Sprinkler/deluge		
2CCP*P21A,B,C		8/4

Table 3.3-10 (Continued)

FIRE DETECTION INSTRUMENTS

Location	Detectors: Smoke Total/Minimum Operable	Detectors: Heat Total/Minimum Operable
10. Cable Vault and Rod Control Area		
a. Smoke		
Zone 30 (MCC*2-E3, MCC*2-E05)	13/7	
Zone 31 (MCC*2-E14, MCC*2-E06)	9/5	
Zone 32 (2RCP*H2A,B,D,E)	21/11	
Zone 53 (Auxiliary Building relay room	3/2	
Zone 52 (Auxiliary Building cable tunnel)	6/3	
Zone 50 (Auxiliary Building cable area)	5/3	
b. Carbon Dioxide		
Zone 3 (Auxiliary Building cable tunnel)		13/7
Zone 3 (cable vault and rod control)		22/11
Zone 2A (cable vault and rod control east)		7/4
Zone 2 (cable vault and rod control west)		13/7
Zone 2 (cable tunnel)		8/4
11. Cable Tunnel		
a. Smoke		
Zone 16 (cable tunnel north)	7/4	
Zone 36 (cable tunnel north)	2/1	
Zone 22 (cable tunnel south)	7/4	
Zone 15 (cable tunnel south)	3/2	

Table 3.3-10 (Continued)

FIRE DETECTION INSTRUMENTS

Location	Detectors: Smoke Total/Minimum Operable	Detectors: Heat Total/Minimum Operator
b. Carbon Dioxide		
Zone 1 (cable tunnel-control to PAB)		14/7
Zone 1 (cable tunnel-PAB)		13/7

3/4.3.3 MONITORING INSTURMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY:

Modes 1, 2, 3, 4.

ACTION:

a. With one chlorine detection systems inoperable, restore one of the inoperable detection systems to OPERABLE status within 7 days, or within the next 6 hours, initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operations.

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTION TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

3/4.3.3 MONITORING INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, and 3

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in the channel designation column in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours except for the PORV(s) which may be isolated in accordance with Specification 3.4.11.a.

b. With the number of OPERABLE accident monitoring instrumentation channels less the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

Instrument	Required Number of Channels Operable	Minimum Channels Operable
1. Pressurizer Water Level	3	2
2. Auxiliary Feedwater Flow Rate	1 per steam generator	1 per steam generator
3. Reactor Coolant System Subcooling Margin Monitor	1	0
4. PORV Position Indicator	2/valve	1/valve
5. PORV Block Valve Position Indicator	1/valve	0/valve
6. Safety Valve Position Indicator	2/valve	1/valve
7. Safety Valve Temperature Detector	1/valve	0/valve

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Channel Calibration
1. Pressurizer Water Level	M	R
2. Auxiliary Feedwater Flow Rate	SU(1)	R
3. Reactor Coolant System Subcooling Margin Monitor	M	R
4. PORV Position Indicator	M	R
5. PORV Block Valve Position Indicator	M	R
6. Safety Valve Position Indicator	M	R
7. Safety Valve Temperature Detector	M	R

(1) Channel Check to be performed in conjunction with Surveillance Requirement 4.7.1.2.a.9 following an extended plant outage.

3/4.3.3 MONITORING INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip septoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY:

During releases through the flow paths

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification immediately suspend the release of radioactive liquid effluents monitored by the affected channel or correct the alarm/trip setpoint.

b. With one or more radioactive liquid effluent monitoring instrumentation channels INOPERABLE, take the action shown in Table 3.3-12 or conservatively reduce the alarm setpoint. Exert best effort to return the instruments to OPERABLE status within 30 days. If unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the INOPERABILITY was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Action
1. Gross radioactivity monitor providing alarm and automatic termination of release		
a) Liquid waste process effluent monitor 2SGC-RQI100	1	23
2. Gross BETA or GAMMA radioactivity monitors providing alarm but not providing automatic termination of release		
None		
3. Flow rate measurement devices		
a) Liquid radwaste effluent 2SGC-FT100 and FI100	1	25
4. Tank level indicating devices		
a) RWST (2QSS*LT100A,B)	1	26

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATION

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may be resumed provided that prior to initiating a release:

a) At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1 and;

b) At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least one per 8 hours grab samples are analyzed for gross radioactivity (BETA or GAMMA) at a Lower Limit of Detection (LLD) of at least 10^{-7} uCi/ml.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least one per 4 hours during actual releases. Pump curves may be used to estimate flow.

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional
1. Gross radioactivity monitor providing alarm and automatic termination of release				
a) Liquid waste process effluent 2SGC-RQI100	D	P(5)	R(3)	Q(1)
2. Gross BETA or GAMA radioactivity monitors providing alarm but not providing automatic termination of release				
None				
3. Flow rate measurement devices				
a) Liquid radwaste effluent 2SGC-FT100 and FI100	D(4)	N/A	R	Q
4. Tank level indicating devices				
a) RWST (2QSS*LT100A,B)	D*	N/A	R	Q

* During liquid additions to the tank

TABLE 4.3-12 (Continued)

TABLE NOTATION

1. The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and Control Room Alarm Annunciation occurs if any of the following conditions exist:

    1. Instrument indicates measured levels above the alarm/trip setpoint.

    2. Downscale failure.

    3. Instrument controls not set in operate mode.

2. The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

    1. Instrument indicates measured levels above the alarm/trip setpoint.

    2. Downscale failure.

    3. Instrument controls are not set in operate mode.

3. The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per 18 months. This can normally be accomplished during refueling outages. (Existing plants may substitute previously established calibration procedures for this requirement.)

4. CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

5. A source check may be performed utilizing the installed means or flashing the detector with a portable source to obtain an upscale increase in the existing count rate to verify channel response.

3/4.3.3 MONITORING INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY:

During releases through the monitored flow paths.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.11.2.1 are met, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or correct the alarm/trip set point.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels INOPERATIVE, take the ACTION shown in Table 3.3-13 or conservatively reduce the alarm set point. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the INOPERABILITY was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the requirements shown in Table 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Parameter	Action
1. Elevated Release 2HVS*RQI109			
a. Noble gas activity monitor	1	Radioactivity Rate Measurement	29, 30*
b. Iodine Sampler	1		32
c. Particulate Sampler	1		32
d. Process flow rate monitor	1	Process Flow Rate Measurement	28
e. Sampler flow rate monitor	1	Sampler Flow Rate Measurement	28
2. Decontamination Building Vent 2RMQ-RQI301			
a. Noble gas activity monitor	1	Radioactivity Rate Measurement	29
b. Iodine sampler	1		32
c. Particulate sampler	1		32
d. Process flow rate monitor	1	Process Flow Rate Measurement	28
e. Sampler flow rate monitor	1	Sampler Flow Rate Measurement	28
3. Ventilation Vent 2HVS-RQI101			
a. Noble gas activity monitor	1	Radioactivity Rate Measurement	29, 30
b. Iodine sampler	1		32
c. Particulate sampler	1		32
d. Process flow monitor	1	Process Flow Rate Measurement	28
e. Sampler flow monitor	1	Sampler Flow Rate Measurement	28
4. Waste Gas Storage Vault 2RMQ-RQI303			
a. Noble gas activity monitor	1	Radioactivity Rate Measurement	29
b. Iodine sampler	1		32
c. Particulate sampler	1		32
d. Process flow rate monitor	1	Process Flow Rate Measurement	28
e. Sampler flow rate monitor	1	Sampler Flow Rate Measurement	28

TABLE [illegible]3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Parameter	Action
5. Condensate Polishing Building Vent 2HVLK-RQI112			
a. Noble gas activity monitor	1	Radioactivity Rate Measurement	29
b. Iodine Sampler	1		32
c. Particulate Sampler	1		32
d. Process flow rate monitor	1	Process Flow Rate Measurement	28
e. Sampler flow rate monitor	1	Sampler Flow Rate Measurement	28
6. Gaseous Waste/Process Vent System Rm-GW-108 A & B (BVPS Unit 1)			
a. Noble gas activity monitor	1	Radioactivity Rate Measurement	27,30
b. Iodine Sampler Cartridge	1		32
c. Particulate Activity Monitor	1		32
d. System Effluent Flow Rate Measuring Device (FR-GW-108)	1	System Flow Rate Measurement	28
e. Sampler Flow Rate Measuring Device	1	Sampler Flow Rate Measurement	28
7. Waste Gas Holdup System Explosive Gas Monitoring			
a. Oxygen Monitor	2	31	

* During purging of reactor containment via this pathway

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATION

ACTION 27 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank may be released to the environment provided that prior to initiating the release:

1. At least two independent samples of the tank's content are analyzed, and

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 30 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, immediately suspend PURGING of Reactor Containment via this pathway.

ACTION 31 - With the number of channels OPERABLE one less than required by the MINIMUM Channels OPERABLE requirement, operation of this system may continue provided grab samples are obtained every 4 hours and analyzed within the following 4 hours during additions to a tank.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 or sampled and analyzed once every 8 hours.

ACTION 35 - See Surveillance 4.11.2.5.1.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
1. Elevated Release 2HVS-RQI109				
a. Noble gas activity monitor	D	M(5) P(5)***	R(3)	Q(2)
b. Iodine Sampler	W	N/A	N/A	N/A
c. Portible sampler	W	N/A	N/A	N/A
d. Process flow rate monitor	D	N/A	R	Q
e. Sampler flow rate monitor	D	N/A	R	Q
2. Decontamination Building Vent 2RMQ-RQI301				
a. Noble gas activity monitor	D	M(5)	R(3)	Q(2)
b. Iodine sampler	W	N/A	N/A	N/A
c. Particulate sampler	W	N/A	N/A	N/A
d. Process flow rate	D	N/A	R	Q
e. Sample flow rate	D	N/A	R	Q
3. Ventilation Vent 2HVS-RQI101				
a. Noble gas activity monitor	D	M(5) P(5)***	R(3)	Q(2)
b. Iodine sampler	W	N/A	N/A	N/A
c. Portible sample	W	N/A	N/A	N/A
d. Process flow rate monitor	D	N/A	R	Q
e. Sample flow rate monitor	D	N/A	R	Q
4. Waste Gas Storage Vault 2RMQ-RQI303				
a. Noble gas activity monitor	D	M(5)	R(3)	Q(2)
b. Iodine sample	W	N/A	N/A	N/A
c. Particulate sample	W	N/A	N/A	N/A
d. Process flow rate monitor	D	N/A	R	Q
e. Sample flow rate monitor	D	N/A	R	Q

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
5. Condensate Polishing Building Vent 2HVR-RQI112				
a. Noble gas activity monitor	D	M(5)	R(3)	Q(2)
b. Iodine Sampler	W	N/A	N/A	N/A
c. Particulate sampler	W	N/A	N/A	N/A
d. Process flow rate monitor	D	N/A	R	P
e. Sample flow rate monitor	D	N/A	R	Q
6. Gaseous Waste/Process Vent System RM-GW-108 A & B (BVPS Unit 1)				
a. Noble gas activity monitor	P	P(5)	R(3)	Q(1)
b. Iodine sampler cartridge	W(6)	N/A	N/A	N/A
c. Particulate activity monitor	W	N/A	N/A	N/A
d. System effluent flow rate measuring device (FR-GW-108)	P	N/A	R	Q
e. Sampler flow rate measuring device	D*	N/A	R	Q
7. Waste Gas Holdup System Explosive Gas Monitoring				
a. Oxygen Monitor	D	N/A	Q(4)	M

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm/trip setpoint.
b. Downscale failure.
c. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm/trip setpoint.
b. Downscale failure.
c. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. This can normally be accomplished during refueling outages.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and
b. Four volume percent oxygen, balance nitrogen.

(5) A source check may be performed utilizing the installed means or flashing the detector with a portable source to obtain an upscale increase in the existing count rate to verify channel response.

(6) Compliance is demonstrated in Table 4.11-2.

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:

MODES 1 and 2*.

ACTION:

Above P-7, comply with the following ACTIONS:

a. With one reactor coolant loop and associated pump not in operation, subsequent STARTUP and POWER OPERATION above 25 percent of RATED THERMAL POWER may proceed provided:

1. The following actions have been completed with the reactor subcritical:

a) Reduce the overtemperature ΔT trip setpoint to the value specified in Specification 2.2.1 for 2 loop operation.

* See Special Test Exception 3.10.5.

LIMITING CONDITION FOR OPERATION

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

1. Overpower Delta T channel.

2. Overtemperature Delta T channel.

c) Change the P-8 interlock setpoint from the value specified in Table 3.3-1 to ≤ 70 percent of RATED THERMAL POWER when the reactor coolant stop valves in the non-operations loop are closed.

2. Thermal Power is restricted to ≤ 65 percent of RATED THERMAL POWER when the reactor coolant stop valves in the non-operating loop are closed.

3. With one reactor coolant loop and associated pump not in operation, subsequent STARTUP and PROPER OPERATION at or below 31% of RATED THERMAL POWER may proceed provided the following reactor trip system channels, associated with the loop not in operation, have been placed in their tripped condition with the reactor subcritical:#

1. Overpower Delta T channel

2. Overtemperature Delta T channel

These channels may be placed in the bypass condition for up to 8 hours during surveillance testing of the overpower and overtemperature Delta T channels of the active loops.

3/4.4.1 NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

Below P-7:

a. With $K_{eff} \geq 1.0$, operation below P-7 may proceed provided at least two reactor coolant loops and associated pumps are in operation.

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

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant loop and associated pump not in operation, at least once per 7 days determine that:

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

b. If P-8 interlock setpoint was reset for 2 loop operation, the P-8 interlock setpoint is $\leq$ 70 percent of RATED THERMAL POWER when the reactor coolant stop valves in the non-operating loop are closed.

4.4.1.1.2 The power to each of the reactor coolant system loop stop valves shall be verified to be removed at least once per 31 days during operation in Modes 1 or 2.

3/4.4.1 REACTOR COOLANT LOOPS

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2

At least two of the reactor coolant loops listed below shall be OPERABLE and in operation*:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump.

APPLICABILITY:

MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.1.2

At least two cooling loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3

a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump*,
2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump*,
3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump*.
4. Residual Heat Removal Pump (A) and a heat exchanger**,
5. Residual Heat Removal Pump (B) and a second heat exchanger**.

b. At least one of the above coolant loops shall be in operation***.

APPLICABILITY:

MODES 4 and 5.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to (later)°F unless the secondary water temperature of each steam generator is less than (later)°F above each of the RCS cold leg temperatures.

** The normal or emergency power source may be inoperable in MODE 5.

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

3/4.4.1 REACTOR COOLANT LOOPS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5, and by verifying that each residual heat removal pump develops a differential pressure of $\geq$ (later) psi on recirculation flow.

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

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level equivalent to 12 percent narrow range at least once per 12 hours.

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

3/4.4.1 REACTOR COOLANT LOOPS

ISOLATED LOOP

LIMITING CONDITION FOR OPERATION

3.4.1.4 The boron concentration of an isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops, except when the loop is drained for maintenance.

APPLICABILITY:

MODES 1, 2, 3, 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loops stop valves; either increase the boron concentration of the isolated loop to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours with the unisolated portion of the RCS borated to a SHUTDOWN MARGIN equivalent to at least 1 percent delta k/k at 200°F.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The boron concentration of an isolated loop shall be determined to be greater than or equal to the boron concentration of the operating loops within 30 minutes prior to opening either the hot leg or cold leg stop valves of an isolated loop.

3/4.4.1 REACTOR COOLANT LOOPS

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.5 A reactor coolant loop shall remain isolated until:

a. The isolated loop has been operating on a recirculation flow of $\geq$ 125 gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops.

b. The reactor is subcritical by at least 1 percent delta k/k.

APPLICABILITY:

All Modes*

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

4.4.1.5.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.5.2 The reactor shall be determined to be subcritical by at least 1 percent delta k/k within 30 minutes prior to opening the cold leg stop valve.

* With fuel in the reactor vessel

3/4.4.2 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 ± 1 percent.

APPLICABILITY:

MODES 4 and 5.

ACTION:

a. 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.

b. Failure of a pressurizer code safety valve to operate when required shall be reported to the Commission within 30 days pursuant to Specification 6.9.2.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

3/4.4.3 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 ± 1 percent.

APPLICABILITY:

MODES 1, 2, and 3.

ACTION:

a. 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.

b. Failure of a pressurizer code safety valve to operate when required shall be reported to the Commission within 30 days pursuant to Specification 6.9.2

SURVEILLANCE REQUIREMENTS

4.4.3 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2485 psig ± 1 percent, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1980 Edition.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 150 kw of pressurizer heaters and with a steam bubble.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the pressurizer inoperable due to less than 150 kw of heaters supplied by an emergency bus, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The power supply for these pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters and measuring circuit current to verify capacity.

3/4.4.5 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

4.4.5.1 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. Steam generator tubes shall be examined in accordance with the method prescribed in Article 8 - "Eddy Current Examination of Tubular Products," as contained in ASME Boiler and Pressure Vessel Code, Section V - "Non-destructive Examination," and referenced in ASME Boiler and Pressure Vessel Code - Appendix IV of the 1980 Edition through Winter 1980 Addenda of Section XI - "Inservice Inspection of Nuclear Power Plant Components." The tubes selected for each inservice inspection shall include at least 3 percent 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 percent 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.

SURVEILLANCE REQUIREMENTS

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50 percent 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 percent 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 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes.
C-3	More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent 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 All Volatile Treatment (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.

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS

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 or repaired because it may become unserviceable prior to the next inspection and is equal to 40 percent 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 (repair 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

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported pursuant to Specification 6.6 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		
Number 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	Repair 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., repair 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 add'l S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Add'l 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

TABLE 4.4-2 (Continued)

STEAM GENERATOR TUBE INSPECTION

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

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 discharge flow measurement system or narrow range level instrument, and

c. Containment atmosphere gaseous activity monitoring system.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:

1) The other two above required leakage detection systems are OPERABLE, and

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours:

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

b. W.ch the containment sump discharge flow measurement system and narrow range level instrument inoperable, restore at least one inoperable system 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.

c. The provisions of Specification 3.0.4 are not applicable in Modes 1, 2 and 3.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous monitoring system by performing a CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

b. Containment sump discharge flow measurement system by performing a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Logging the narrow range level indication every 12 hours.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

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 not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System.

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

e. 28 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakage shall be demonstrated to be within each of the above limits by:

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

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

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 $\pm$ 20 psig at least once per 31 days with the modulating valve fully open.

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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.3 This Technical Specification intentionally blank.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

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 analysis 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 operations 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 3.4-2

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

CONTAINMENT	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
Dissolved Oxygen	3 times per 7 days*	72 hours
Chloride	3 times per 7 days	72 hours
Fluoride	3 times per 7 days	72 hours

---

* Not required with $T_{avg} \leq 250°F$.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

b. $\leq$ 100/E uCi/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 uCi/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 uCi/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 HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > 100/E uCi/gram, be in 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 uCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ uCi/gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. Immediately notify the Commission pursuant to 10CFR50.72 (declaration of any of the Emergency Classes specified in the Emergency Preparedness Plan). Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2 containing the results of the specific activity analyses together with the following information:

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* With $T_{avg} \geq$ 500°F.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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.

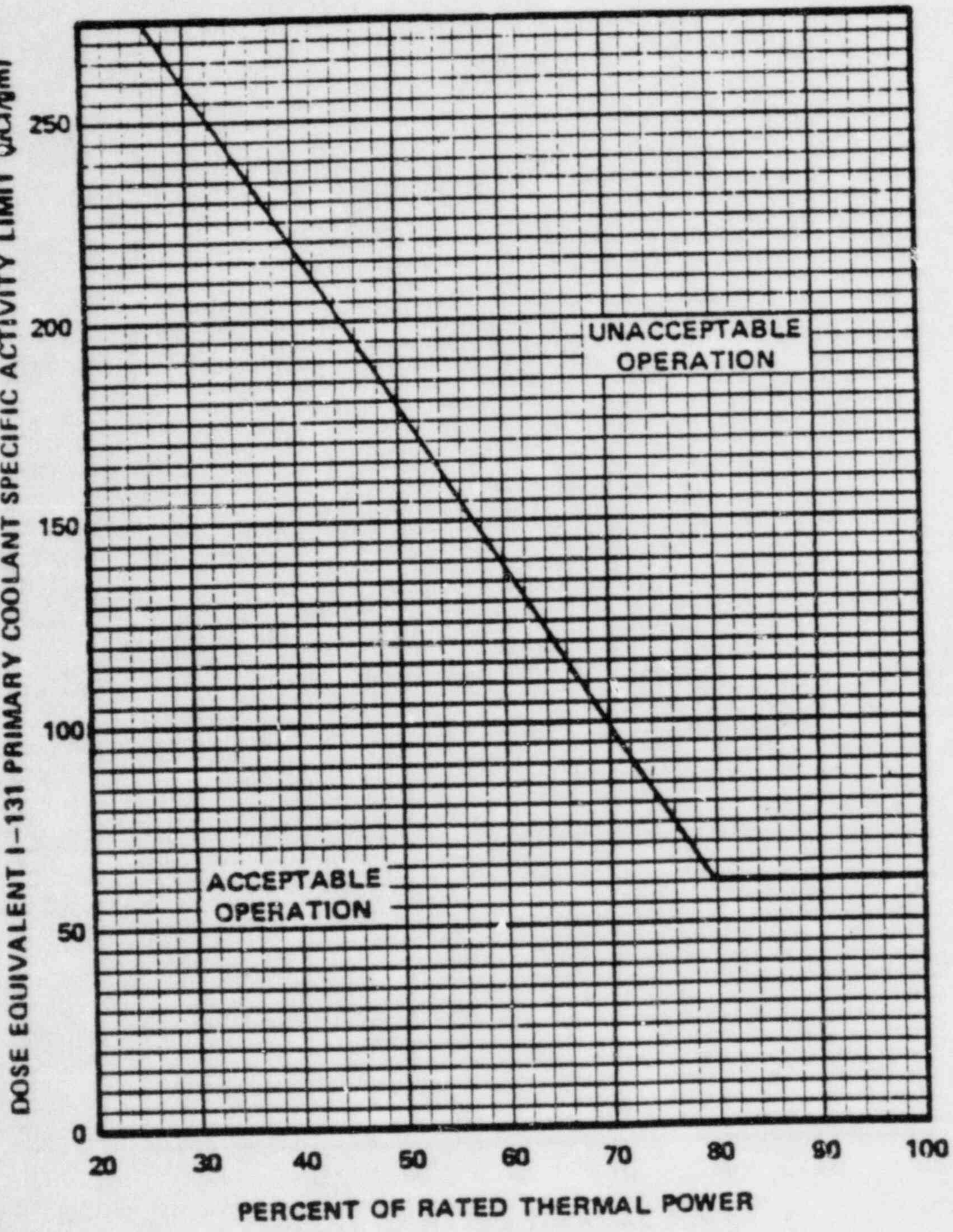
TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SURVEILLANCE REQUIRED
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples.	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 uCi/gram DOSE EQUIVALENT I-131 or $100/\overline{E}$ uCi/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.

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 uCi/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:

MODES 1, 2*, 3, 4, and 5.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. 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.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

* See Special Test Exception 3.10.3.

3/4.4.9 PRESSURE TEMPERABURT LIMITS

RFACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

c. 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 examinctions shall be used to update Figures 3.4-2 and 3.4-3.

FIGURE 3.4-2

BEAVER VALLEY UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 10 EFPY

MATERIAL PROPERTY BASIS

Controlling Material	:	Plate Metal
Copper Content	:	Conservatively Assumed to be 0.10 wt%
Phosphorus Content	:	0.010 wt%
RT_{NDT} Initial	:	60°F.
RT_{NDT} After 10 EFPY	:	1/4T, 139°F. 3/4T, 114°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

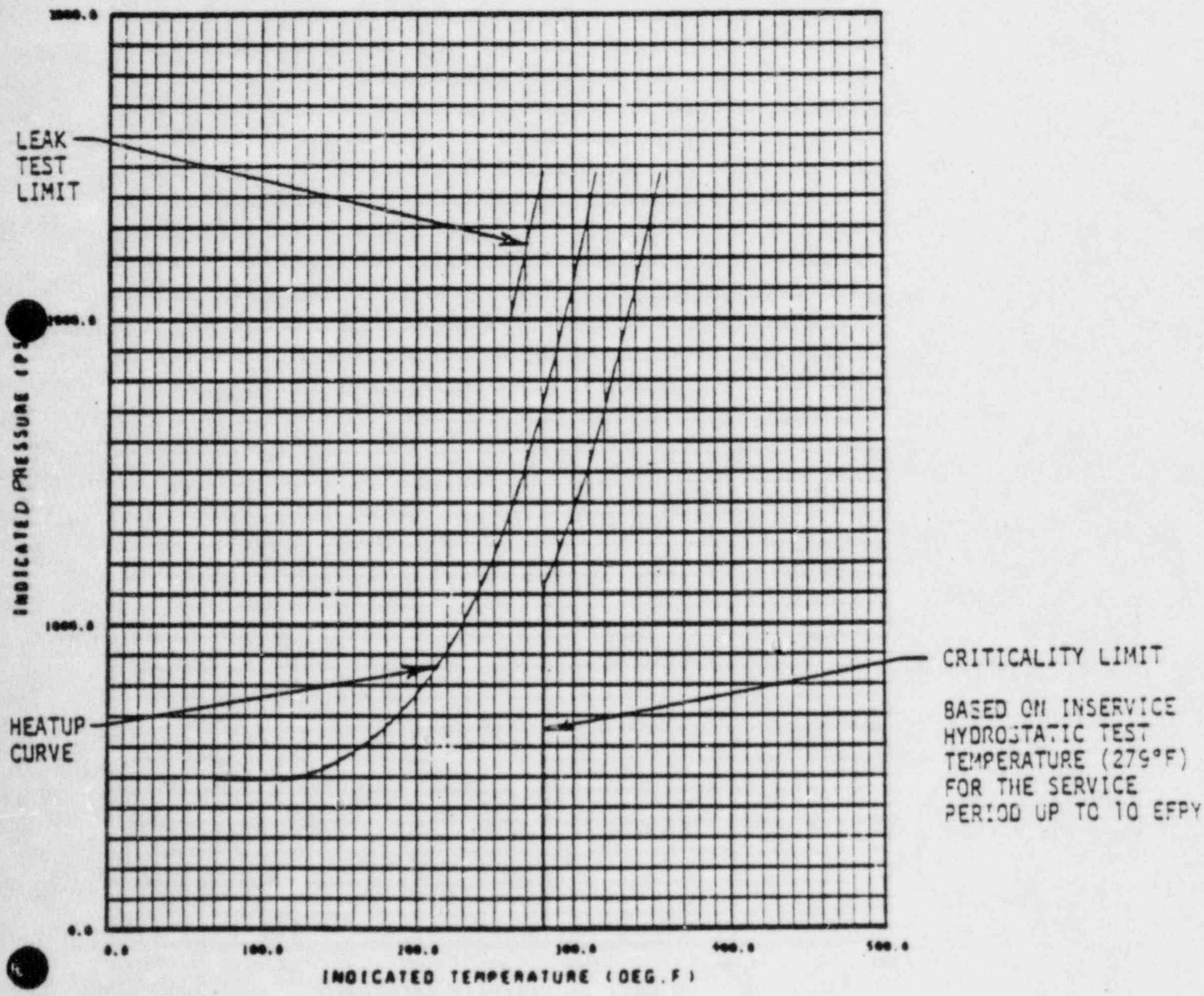


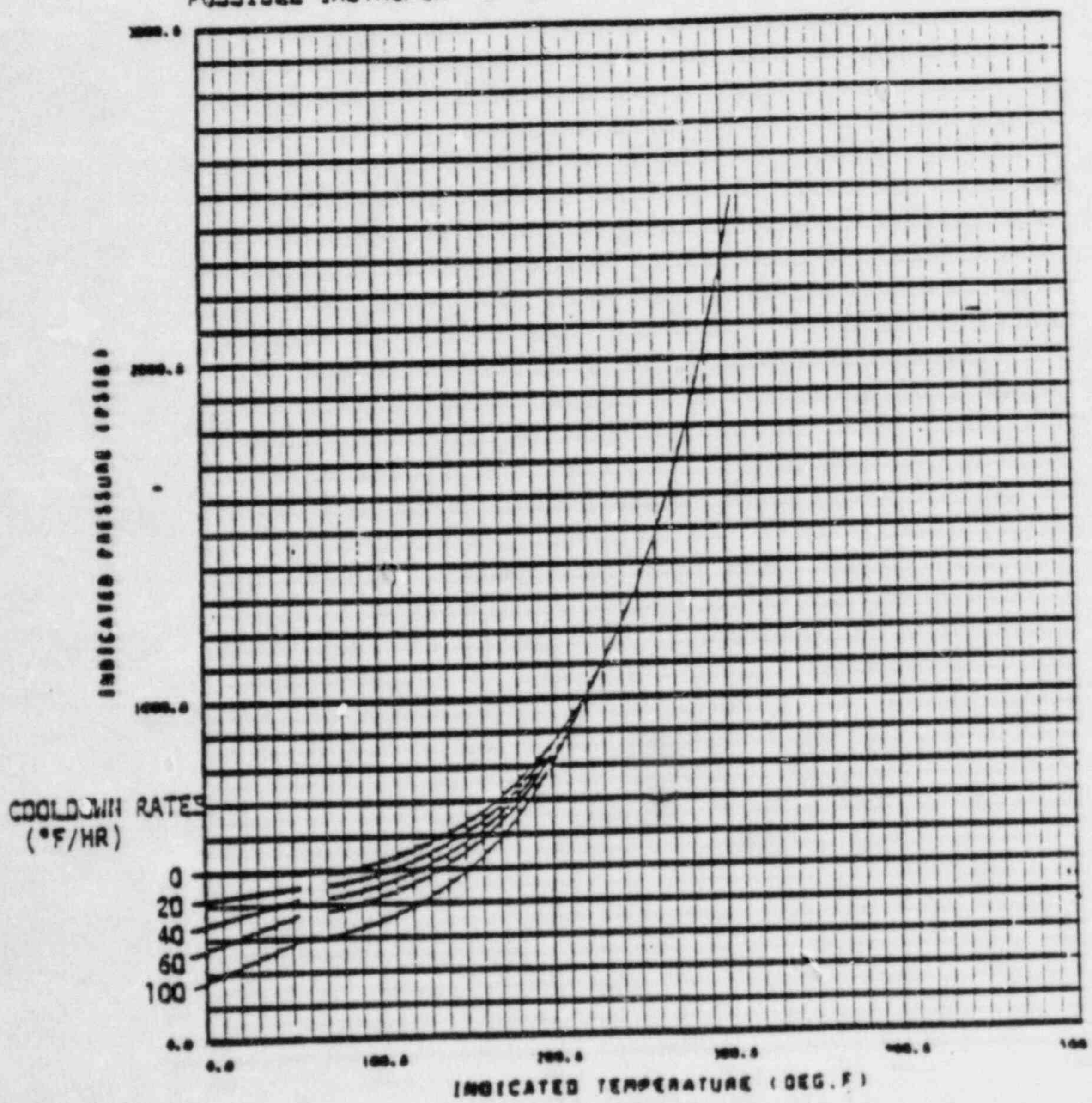
FIGURE 3.4-3

BEAVFR VALLEY UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 10 EFPY

MATERIAL PROPERTY BASIS

Controlling Material	:	Plate Metal
Copper Content	:	Conservatively Assumed to be 0.10 wt%
Phosphorus Content	:	0.010 wt%
RT_{NDT} Initial	:	60°F.
RT_{NDT} After 10 EFPY	:	1/4T, 139°F.
		3/4T, 114°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS



3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

b. A maximum cooldown of 200°F in any one hour period, and

c. 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 analysis 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 steady state operations.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

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 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5.

3/4.4.11 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.11 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, and 3.

ACTION:

a. With less than 2 PORV(s) operable, within 1 hour either restore two PORV(s) to OPERABLE status or close the associated block valves(s) and remove power from the block valves(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valves(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. Failure of a power operated relief valve to operate when required shall be reported to the Commission within 30 days pursuant to Specification 6.9.2

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL CHECK of the position indication, excluding valve operation, and

b. By performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

4.4.11.3 The power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between (later) and (later) gallons of borated water,

c. Between 1900 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between (later) and (later) 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 at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 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

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

---

* Pressurizer Pressure above 1000 psig.

3/4.5.1 ACCUMULATORS

SURVEILLANCE REQUIREMENTS

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1 percent 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 control circuit is disconnected by removal of the plug in the lock out jack from the circuit.

d. Verifying at least once per 18 months that each accumulator isolation valve opens automatically under each of the following conditions:

1. When the RCS pressure exceeds 2000 psig.

2. Upon receipt of a Safety Injection test signal.

4.5.1.2 Each accumulator water level and pressure alarm channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of a CHANNEL FUNCTIONAL TEST.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Low head safety injection pump, and

c. 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 subsystems 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 operator control circuits disconnected by removal of the plug in the lock out circuit from each circuit.

Valve Number	Valve Function	Valve Position
a. 2SIS*MOV8889	LHSI to hot legs	Closed
b. 2SIS*MOV869A	HHSI to hot leg	Closed
c. 2SIS*MOV869B	HHSI to hot leg	Closed
d. 2SIS*MOV836	HHSI to cold leg	Closed
e. 2SIS*MOV841	HHSI to cold leg	Open
f. 2CHS*MOV8132A	HHSI pump disch X-conn	Open
g. 2CHS*MOV8132B	HHSI pump disch X-conn	Open
h. 2CHS*MOV8133A	HHSI pump disch X-conn	Open
i. 2CHS*MOV8133B	HHSI pump disch X-conn	Open

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

1. Verifying that each centrifugal charging pump:

   a. Starts (unless already operating) from the control room,

   b. Develops a discharge pressure of $\geq$ 2437* psig on recirculation flow,

   c. Operates for at least 15 minutes.

2. Verifying that each low head safety injection pump:

   a. Starts (unless already operating) from the control room,

   b. Develops a discharge pressure $\geq$ 103* psig on recirculation flow,

   c. Operates tor at least 15 minutes.

3. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

*To be verified during Pre-Operational Testing

SURVEILLANCE REQUIREMENTS

4. 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.

5. Verifying that each ECCS subsystem is aligned to receive electrical power from separate OPERABLE emergency buses.

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. 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. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection signal.

3. Verifying that the centrifugal charging pump and low head safety injection pumps start automatically upon receipt of a safety injection signal.

3/4.5.3 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 low head safety injection pump, and

c. 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. 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.

LIMITING CONDITION FOR OPERATION

3.5.4 This Technical Specification intentionally blank.

LIMITING CONDITION FOR OPERATION

3.5.5 This Technical Specification intentionally blank

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 36 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.

2. All equipment hatches are closed and sealed.

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

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to La, (0.10) percent by weight of the containment air per 24 hours at Pa (44.7 psig), or

b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C tests, as identified in Table 3.6-1, when pressurized to Pa (44.7 psig).

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 La, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 La, 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. A Type A test (Overall integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at Pa (44.7 psig).

b. If any periodic Type A test fails to meet .75 La, 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 La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 La, at which time the above test schedule may be resumed.

* Exception to 10CFR50 Appendix J.III.D.1(a)

CONTAINMENT LEAKAGE

SURVEILLANCE REQUIREMENTS

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 La.

   2. Has a duration sufficient to establish accurately the change in leakage rate 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 at Pa (44.7 psig).

d. Type B and C tests shall be conducted with gas at Pa (44.7 psig) at intervals no greater than 24 months except for tests involving.

   1. Air locks,

   2. Penetrations using continuous leakage monitoring systems, and

   3. Valves pressurized with fluid from a seal system.

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

f. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J. Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 Pa (49.2 psig) and the seal system capacity is adequate to maintain system pressure for at least 30 days.

g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analysis shall be performed to determine the inaccuracy of the measured leakage rates due to maximum measurement accuracy and instrument repeatability; the measured leakage rates shall be adjusted to include the measurement error.

3/4.6.1 PRIMARY CONTAINMENT

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 less than or equal to 0.05 La at Pa, (44.7 psig).

APPLICABILITY:

Modes 1, 2, 3, and 4.

ACTION:

With a containment air lock inoperable, restore the inoperable 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.

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT AIR LOCKS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each containment entry, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage when the gap between the door seals is pressurized for at least 2 minutes to:

1. Personnel airlock $\geq$ 44.7 psig

2. Emergency air lock $\geq$ 10.0 psig

or, by quantifying the total air lock leakage to insure the requirements of 3.6.1.3.b are met.

b. By conducting overall air lock leakage tests, at not less than P_a (44.7 psig), and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months, # and

2. Upon completion of maintenance which has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 18 months during shutdown by verifying:

1. Only one door in each air lock can be opened at a time, and

2. No detectable seal leakage when the volume between the emergency air lock shaft seals is pressurized to greater than or equal to 44.7 psig for at least 2 minutes.

The provisions of Specification 4.0.2 are not applicable.

* Exemption to Appendix J of 10 CFR 50.

3/4.6.1 PRIMARY CONTAINMENT

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained greater than or equal to 9.0 psia and within the acceptable operation range (below and to the left of the RWST water temperature limit lines) shown on Figure 3.6-1 as a function of RWST water temperature and service water temperature.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure less than 9.0 psia or above the applicable RWST water temperature limit line shown on Figure 3.6-1, 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.

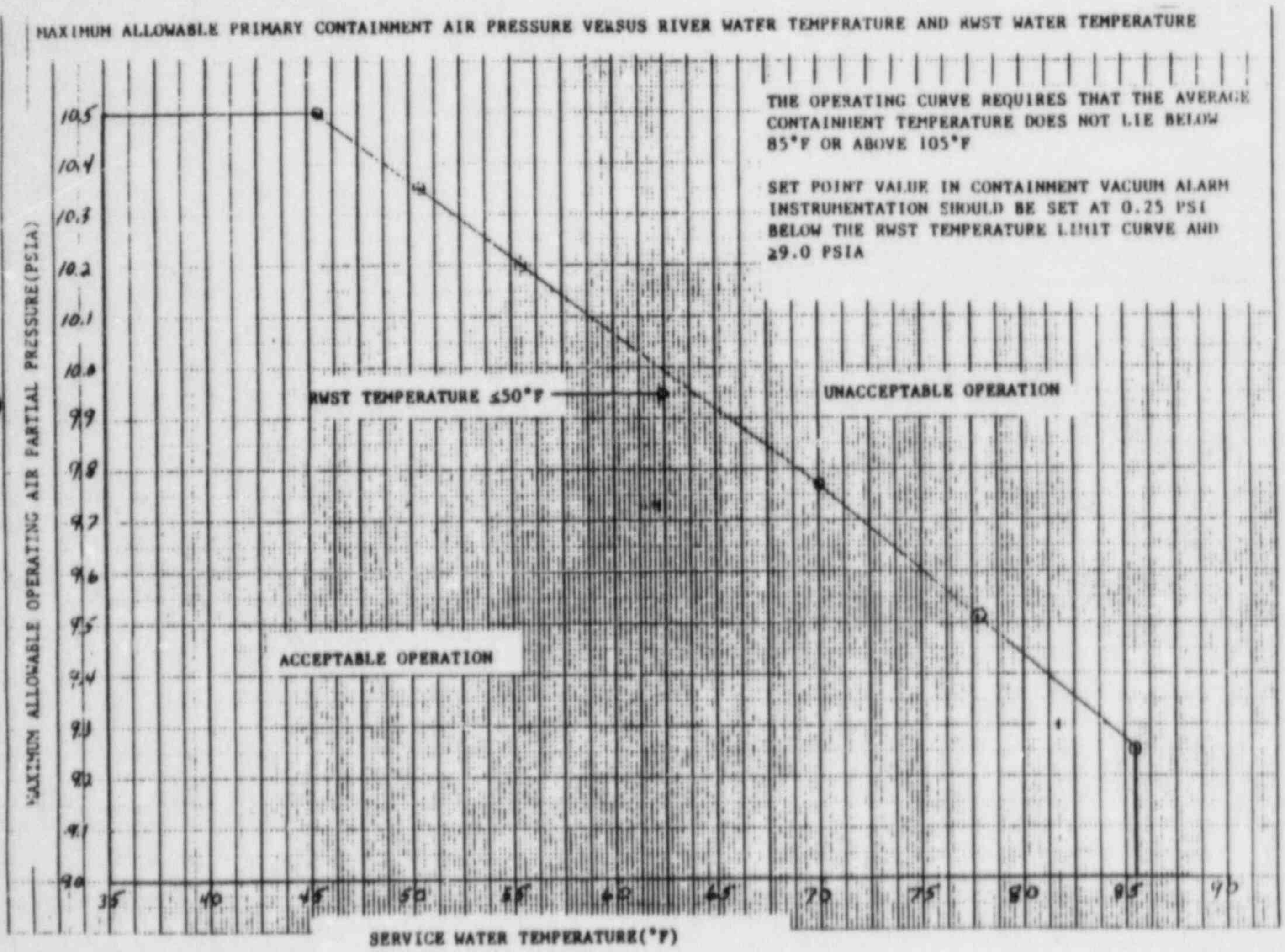
SURVEILLANCE REQUIREMENTS

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

FIGURE 3.6-1

MAXIMUM ALLOWABLE PRIMARY CONTAINMENT AIR PRESSURE

VERSUS SERVICE WATER TEMPERATURE AND RWST WATER TEMPERATURE



3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained less than or equal to 105°F and greater than or equal to 85°F.

APPLICABILITY:

MODES 1, 2, 3, and 4

ACTION:

With the containment average air temperature greater than 105°F or below 85°, restore the average air temperature to within the limit shown 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 any five temperatures from the following locations and shall be determined at least once per 24 hours:

Location	Elevation
RC Annulus,	730'-7"
RHR Cub	801'-6"
SG 21C Cub	701'-6"
SG 21C Cub	745'-6"
SG 21C Cub	865'-0"
Pzr Cub	802'-0"
RC Annulus	740'-7"
Pzr Cub Stairway	746'-0"
SG 21B Cub	701'-6"
SG 21B Cub	865'-0"
SG 21B Cub	730'-0"
RC Annulus	736'-11"
Reactor Head Storage Area	802'-0"
RC Annulus	701'-6"
RC Annulus	777'-4"
SG 21A Cub	701'-6"
SG 21A Cub	865'-0"
SG 21A Cub	726'-6"
RC Annulus	740'-10"
SG 21C Cub	727'-0"

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION 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 4.6.1.6.1.

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 Liner Plate and Concrete - The structural integrity of the containment liner plate and concrete shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by:

a. A visual inspection of the accessible surfaces and verifying no apparent changes in appearance or other abnormal degradation.

b. A visual inspection of accessible containment liner test channels prior to each Type A containment leakage rate test.

c. A visual inspection of the dome area prior to each Type A containment leakage rate test to insure the integrity of the protective coating.

4.6.1.6.2 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 reports shall be submitted pursuant to Specification 6.9.2 within 90 days after completion. This report shall include a description of the condition of the liner plate and concrete, the inspection procedure, the tolerances on cracking and the corrective actions taken.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two separate and independent containment quench spray subsystems shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, 3, and 4

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

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

1. Starting each spray pump.
2. Verifying, that on recirculation flow, when tested in accordance with the requirements of Section 4.0.5, each quench spray pump develops a discharge pressure of $\geq$ 172* psig at a flow of $\geq$ 3000* gpm.
3. Verifying that each spray pump operates for at least 15 minutes.
4. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.
5. 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.
6. Verifying the temperature of the borated water in the refueling water storage tank is within the limits shown on Figure 3.6-1.

*Will be verified during Pre-Operational Testing.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

SURVEILLANCE REQUIREMENTS

b. At least once per 18 months during shutdown:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

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

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

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

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent containment recirculation spray subsystems shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, 3, and 4

ACTION:

With one containment recirculation spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days 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

4.6.2.2 Each containment recirculation spray subsystem shall be demonstrated OPERABLE.

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

1. Manually starting each spray pump and verifying the pump shaft rotates.

2. Verifying correct position of all accessible manual valves not locked, sealed or otherwise secured in position, and all remote or automatically operated valves in each recirculation spray subsystem flow path.

3. Cycling each testable power-operated or automatic valve in the flow path through at least one complete cycle of full travel.

4. 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 verifying that on a Containment Pressure HI-3 (High/High-High) test signal, each recirculation spray pump starts automatically after a 628 ± 3 second delay.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT RECIRCULATION SPRAY SYSTEM

SURVEILLANCE REQUIREMENTS

c. At least once per 18 months, during shutdown, by verifying that on recirculation flow, each recirculation spray pump develops a discharge pressure of $\geq$ 96* psig at a flow of $\geq$ 3500* gpm.

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

1. Cycling each power-operated (excluding automatic) valve in the flow path not testable during plant operation, through at least one complete cycle of full travel.

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

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

* To be verified during pre-operational testing.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CHEMICAL ADDITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The chemical addition system shall be OPERABLE with:

a. A chemical addition tank containing at least 8,500 gallons of between 23 and 25 percent by weight NaOH solution, and

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

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With the chemical addition system inoperable, restore the system to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore the chemical addition system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The chemical addition 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 31 days on a STAGGERED TEST BASIS by:

1. Starting each injection pump.

2. Verifying that each injection pump operates for at least 15 minutes.

3. Cycling each testable power-operated or automatic valve in the flow path through at least one complete cycle of full travel. w

4. Verifying that on recirculation, each injection pump develops a flo between 55 and 60 gpm.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CHEMICAL ADDITION SYSTEM

SURVEILLANCE REQUIREMENTS

c. At least once per 6 months by:

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

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

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

1. Cycling each valve in the chemical addition system flow path that is not testable during plant operation, through at least one complete cycle of full travel.

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

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

c. Isolate the affected penetration within 6 hours by use of at least one closed manual valve or blind flange, or

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Cycling each OPERABLE power-operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

2. Cycling each weight or spring loaded check valve testable during plant operation through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is $<$ 1.2 psid and opens when the differential pressure in the direction of flow is $\geq$ 1.2 psid but less than 6.0 psid.

SURVEILLANCE REQUIREMENTS

b. Immediately 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 the applicable cycling test, above, and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during 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 Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.

d. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring the isolation time.

e. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is $\leq$ 1.2 psid and opens when the differential pressure in the direction of flow is $\geq$ 1.2 psid but less than 6.0 psid.

f. Cycling each manual valve not locked, sealed, or otherwise secured in the closed position through at least one complete cycle of full travel.

TABLE 3.6-1

CONTAINMENT PENETRATIONS

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
1	Comp Cool from Res Heat Exch	(1)(B) 2CCP*MOV157-2	50	(1)(B) 2CCP*MOV157-1 2CCP*RV105	28 N/A
2	Comp Cool to Res Heat Exch	(1)(B) 2CCP*MOV150-2	50	(1)(B) 2CCP*MOV150-1 2CCP*RV102	28 N/A
4	Comp Cool to Res Heat Exch	(1)(B) 2CCP*MOV151-2	50	(1)(B) 2CCP*MOV151-1 2CCP*RV103	28 N/A
5	Comp Cool from Res Heat Exch	(1)(B) 2CCP*MOV156-2	50	(1)(B) 2CCP*MOV156-1 2CCP*RV104	28 N/A
6	SPARE				
7	High Head Safety Injection	(2) 2SIS*83	N/A	(2) 2SIS*MOV869A	10 (4)
9	SPARE				
11	Instrument Air	(A) 2IAC*MOV133	30	(A) 2IAC*MOV134	30
13	SPARE				
14	Chill & Service Wtr to Cont. Air Recirc Cooling Coils	(B) 2SWS*MOV153-2	22	(B) 2SWS*MOV153-1 2SWS*RV153	22 N/A
15	CHARGING	2CHS*31 2CHS*RV8144	N/A	2CHS*MOV289	10
16	SPARE				

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
17	High Head Safety Injection	(2) 2SIS*84	N/A	(2) 2SIS*MOV869B	10 (4)
19	Seal Water from Reactor Coolant Pump	2CHS*MOV378 2CHS*[illegible]	10 N/A	(A) 2CHS*MOV381	10
20	Safety Injection Accumulator Makeup	2SIS*42	N/A	2SIS*41	N/A
21	Chill & Service Wtr from Cont. Air Recirc Cooling Coils	(B) 2SWS*MOV155-2	22	(B) 2SWS*MOV155-1 2SWS*RV155	22 N/A
22	SPARE				
23	SPARE				
24	Residual Heat Removal to Refueling Water Tank	2RHS*107	N/A	2RHS*15 2RHS*RV100	N/A N/A
25	Chill & Service Wtr from Cont. Air Recirc	(B) 2SWS*MOV154-2	22	(B) 2SWS*MOV154-1 2SWS*RV154	22 N/A
27	Chill & Service Wtr to Cont. Air Recirc	(B) 2SWS*MOV152-2	22	(B) 2SWS*MOV152-1 2SWS*RV152	22 N/A
28	Reactor Coolant Letdown	(A) 2CHS*AOV200A (A) 2CHS*AOV200B (A) 2CHS*AOV200C 2CHS*HCV142 2CHS*RV103	10 10 10 40 N/A	(A) 2CHS*AOV204	10

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
29	Pri Dr Trans Pump Disch	(A) 2DGS*AOV108A	1.5	(A) 2DGS*AOV108B 2DGS*RV115	1.5
30	SPARE				
31	SPARE				
32	SPARE				
33	SPARE				
34	High Head Injection Line	(2) 2SIS*94	N/A	(2) SIS*MOV836 (2) 2SIS*MOV840	10 (4) 8
35	Inj Seal Wtr to Reactor Coolant Pump	2CHS*474 2CHS*RV260A	N/A N/A	2CHS*MOV308A	10 (4)
36	Inj Seal Wtr to Reactor Coolant Pump	2CHS*476 2CHS*260B	N/A N/A	2CHS*MOV308B	10 (4)
37	Inj Seal Wtr to Reactor Coolant Pump	2CHS*475 2CHS*RV260C	N/A N/A	2CHS*MOV308C	10 (4)
38	Sump Pump Discharge	(A) 2DAS*AOV100A	1.5	(A) 2DAS*AOV100B 2DAS*RV110	1.5
39	St Gen Blowdown			(2)(A) 2BDG*AOV100A-1	10
40	St Gen Blowdown			(2)(A) 2BDG*AOV100B-1	10
41	St Gen Blowdown			(2)(A) 2BDG*AOV100C-1	10

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
42	Service Air	2SAS*15	N/A	2SAS*14	N/A
43	Air Monitor Sample	2CVS*93	N/A	(A) 2CVS*SOV102	1
44	Air Monitor Sample	(1)(A) 2CVS*SOV153B	8	(1)(A) 2CVS*SOV153A	8
45	Primary Grade Water	2RCS*72	N/A	(A) 2RCS*AOV519 2RCS*RV100	10 N/A
46	Loop Fill	2CHS*72 2CHS*RV160	N/A N/A	2CHS*FCV160	10
47	SPARE				
48	Primary Vent Header	(A) 2VRS*AOV109A-2	1.5	(A) 2VRS*AOV109A-1	1.5
49	Nitrogen Supply Manifold	2RCS*68	N/A	2RCS*AOV101	10
50	SPARE				
51	SPARE				
52	SPARE				
53	Nitrogen Manifold	(A) 2GNS*AOV101-2	10	(A) 2GNS*AOV101-1	10

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
55	Leakage Detection			(2) 2LMS*SOV953	8
	Press Relief Tank	(1)(A) 2SSR*SOV130A-1	0.3	(1)(A) 2SSR*SOV130A-2	0.3
	Accumulator Water Sample	(A) 2SSR*AOV109A-1	0.3	(A) 2SSR*AOV109A-2 2SSR*RV117	0.3 N/A
	Hydrogen Analyzer	(1) 2HCS*SOV136A	0.3	(1) 2HCS*SOV136B	0.3
56	Cold Leg Sample	(A) 2SSR*AOV120A-1	0.3	(A) 2SSR*AOV102A-2 2SSR*RV118	0.3 N/A
	Hot Leg Sample	(1)(A) 2SSR*SOV128A-1	0.3	(1)(A) 2SSR*SOV128A-2 2SSR*RV120	0.3 N/A
	Pressurizer Liquid Space Sample	(a) 2SSR*AOV100A-1	0.3	(A) 2SSR*AOV100A-2 2SSR*RV119	0.3 N/A
	Blowdown Sample			(2)(A) 2SSR*AOV117A	2
57	Leak Detection			(2) 2LMS*SOV950	8
	Blowdown Sample			(2)(A) 2SSR*AOV117B	2
	Pressurizer Vapor Space Sample	(A) 2SSR*AOV112A-1	0.3	(A) 2SSR*AOV112A-2 2SSR*RV121	0.3 N/A
	Hydrogen Analyzer	(1) 2HCS*SOV135A	0.3	(1) 2HCS*SOV135B	0.3
59	Instrument Air Containment	2IAC*22	N/A	(A) 2IAC*MOV130	30

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
60	High Head Safety Injection Discharge	(2) 2SIS*132	N/A	(2) 2SIS*MOV8888B	15
61	Low Head Safety Injection Discharge	(2) 2SIS*130	N/A	(2) 2SISMOV*8889	15
62	Low Head Safety Injection Discharge	(2) 2SIS*133	N/A	(2) 2SIS*MOV8888A	15
63	Quench Pump Discharge	2QSS*4	N/A	(1) 2QSS*MOV101A 2QSS*RV101A	60 (4) N/A
64	Quench Pump Discharge	2QSS*3	N/A	(1) 2QSS*MOV101B 2QSS*RV101B	60 (4) N/A
65	Fuel Transfer Tube			2ISC*102	N/A
66	Recirc Spray Pump Suction			(2) 2RSS*MOV155A	22 (4)
67	Recirc Spray Pump Suction			(2) 2RSS*MOV155C	22 (4)
68	Recirc Spray Pump Suction			(2) 2RSS*MOV155D	22 (4)
69	Recirc Spray Pump Suction			(2) 2RSS*MOV155B	22 (4)
70	Recirculation Pump Discharge	2RSS*29	N/A	(2) 2RSS*MOV156A 2RSS*RV156A	60 (4) N/A
71	Recirculation Pump Discharge	2RSS*31	N/A	(2) 2RSS*MOV156C 2RSS*RV156C	60 (4) N/A

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
73	Main Steam System "A"	Closed System	N/A	(2) 2MSS*HYV101A	5
		Closed System	N/A	(2) 2MSS*AOV102A	-
		Closed System	N/A	(2) 2MSS*SOV105A	8
		Closed System	N/A	(2) 2MSS*SV101A	N/A
		Closed System	N/A	(2) 2MSS*SV102A	N/A
		Closed System	N/A	(2) 2MSS*SV103A	N/A
		Closed System	N/A	(2) 2MSS*SV104A	N/A
		Closed System	N/A	(2) 2MSS*SV105A	N/A
	Steam Drains System	Closed System	N/A	(2) 2SDS*AOV111A-1	2.6
		Closed System	N/A	(2) 2SDS*SOV129B	1
	Steam Vent System	Closed System	N/A	(2) 2SVS*PCV101A	N/A
		Closed System	N/A	(2) 2SVS*HCV104	N/A

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
74	Main Steam System "B"	Closed System	N/A	(2) 2MSS*HYV101B	5
		Closed System	N/A	(2) 2MSS*AOV102B	-
		Closed System	N/A	(2) 2MSS*SOV105B	8
		Closed System	N/A	(2) 2MSS*SV101B	N/A
		Closed System	N/A	(2) 2MSS*SV102B	N/A
		Closed System	N/A	(2) 2MSS*SV103B	N/A
		Closed System	N/A	(2) 2MSS*SV104B	N/A
		Closed System	N/A	(2) 2MSS*SV105B	N/A
	Steam Drains System	Closed System	N/A	(2) 2SDS*AOV111B-1	2.6
		Closed System	N/A	(2) 2SDS*SOV129B	1
	Steam Vent System	Closed System	N/A	(2) 2SVS*PCV101B	N/A
		Closed System	N/A	(2) 2SVS*HCV104	N/A

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
75	Main Steam System "C"	Closed System	N/A	(2) 2MSS*HYV101C	5
		Closed System	N/A	(2) 2MSS*AOV102C	-
		Closed System	N/A	(2) 2MSS*SOV105C	8
		Closed System	N/A	(2) 2MSS*SV101C	N/A
		Closed System	N/A	(2) 2MSS*SV102C	N/A
		Closed System	N/A	(2) 2MSS*SV103C	N/A
		Closed System	N/A	(2) 2MSS*SV104C	N/A
		Closed System	N/A	(2) 2MSS*SV105C	N/A
	Steam Drains System	Closed System	N/A	(2) 2SDS*AOV111C-1	2.6
		Closed System	N/A	(2) 2SDS*SOV129B	1
	Steam Vent System	Closed System	N/A	(2) 2SVS*PCV101C	N/A
		Closed System	N/A	(2) 2SVS*HCV104	N/A
76	Feed Water "A"	Closed System	N/A	(2) 2FWS*HYV157A (2) 2FWS*28	5 N/A
77	Feed Water "B"	Closed System	N/A	(2) 2FWS*HYV157B (2) 2FWS*29	5 N/A
78	Feed Water "C"	Closed System	N/A	(2) 2FWS*HYV157C (2) 2FWS*30	5 N/A

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
79	Aux Feed "A"	(2) 2FWE*99	N/A	(2) 2FWE*HCV100E	40
				(2) 2FWE*HCV100F	40
				(2) 2FWE*42A	N/A
				(2) 2FWE*42B	N/A
80	Aux Feed "B"	(2) 2FWE*100	N/A	(2) 2FWE*HCV100C	40
				(2) 2FWE*HCV100D	40
				(2) 2FWE*43A	N/A
				(2) 2FWE*43B	N/A
83	Aux Feed "C"	(2) 2FWE*101	N/A	(2) 2FWE*HCV100A	40
				(2) 2FWE*HCV100B	40
				(2) 2FWE*44A	N/A
				(2) 2FWE*44B	N/A
87	Hydrogen Recombiner Discharge	2HCS*120	N/A	(1) 2HCS*MOV117	30
88	Hydrogen Recombiner Discharge	2HCS*119	N/A	(1) 2HCS*MOV116	30
89	SPARE				
90	Purge Duct Exhaust	(5) 2HVR*MOD23B	N/A	(5) 2HVR*MOD23A	N/A
91	Purge Duct Supply	(5) 2HVR*MOD25B	N/A	(5) 2HVR*MOD25A	N/A
				(5) 2HVR*DMP206	N/A
92	Hydrogen Recombiner Isolation		N/A	(1) 2HCS*SOV114B	8
				(1) 2HCS*SOV115B	8
	Reactor Cont. Vacuum Pump Suction			(A) 2CVS*SOV151B	8
				(A) 2CVS*SOV152B	8

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
93	Hydrogen Recombiner Isolation		N/A	(1) 2HCS*SOV114A (1) 2HCS*SOV115A	8 8
	Reactor Cont. Vacuum Isolation			(A) 2CVS*SOV151A (A) 2CVS*SOV152A	8 8
94	Ejector Suction	2CVS*151	N/A	2CVS*151-1	N/A
96	SPARE				
97	Leakage Detection			(2) 2LMS*SOV952	8
	Blowdown Sample			(2)(A) 2SSR*AOV117C	2
	Liquid Sample	(1)(A) 2SSR*SOV129A-1	0.3	(1)(A) 2SSR*SOV129A-2 2SSR*RV122	0.3 N/A
	Hydrogen Analyzer	(1) 2HCS*SOV133B	0.3	(1) 2HCS*SOV134B	0.3
98	SPARE				
99	Hose Rack Supply	2FPW*761	N/A	(A) 2FPW*AOV206	12
100	SPARE				
101	RC Pump Deluge	2FPW*753	N/A	(A) 2FPW*AOV205	12
103	Reactor Cavity Purif Inlet	2FNC*121	N/A	2FNC*38	N/A
104	Reactor Cavity Purif Outlet	2FNC*122	N/A	2FNC*9	N/A

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
105	Leak Detection			(2) 2LMS*SOV951	8
	Leak Detection			2LMS*51	N/A
	Hydrogen Analyzer	(1) 2HCS*SOV133A	0.3	(1) 2HCS*SOV134A	0.3
106	Safety Inj. Test Line	(A) 2SIS*MOV842	10	(A) 2SIS*AOV889 2SIS*RV175	10 N/A
108	SPARE				
110	SPARE				
113	Boron Injection	(2) 2SIS*95	N/A	(2) 2SIS*MOV867C (2) 2SIS*MOV867D	10 (4) 10 (4)
114	Recirculation Pump Discharge	(2) 2RSS*32	N/A	(2) 2RSS*MOV156D (2) 2RSS*RV156D	60 (4) N/A
115	Recirculation Pump Discharge	(2) 2RSS*30	N/A	(2) 2RSS*MOV156B (2) 2RSS*RV156B	60 (4) N/A
116	Fire Protection Filter B	2FPW*388	N/A	(A) 2FPW*AOV221	1.5
117	Fire Protection Filter A	2FPW*382	N/A	(A) 2FPW*AOV204	1.5
118	Quench Spray System	2QSS*267	N/A	(1) 2QSS*SOV100A (1) 2QSS*SOV100B	7 (4) 7 (4)

TABLE 3.6-1 (Cont)

Penet. No. Area	Identification/Description	Inside Valve	Maximum Stroke Time (Sec)	Outside Valve	Maximum Stroke Time (Sec)
Primary Containment Personnel Air Lock 2PHS-PAL 1					
	Equalizing Valve	(7) 2PHS*112	N/A		
	Equalizing Valve	(7) 2PHS*113	N/A		
	Equalizing Valve	(7) 2PHS*101	N/A		
	Equalizing Valve			(7) 2PHS*110	N/A
	Equalizing Valve			(7) 2PHS*111	N/A
	Equalizing Valve			(7) 2PHS*100	N/A
Emergency Containment Air Lock 2PHS*EAL1					
	Equalizing Valve	(7) 2PHS*202	N/A		
	Equalizing Valve			(7) 2PHS*201	N/A

NOTES:

(A) Containment Isolation Phase A.
(B) Containment Isolation Phase B.

(1) May be opened on an intermittent basis under administrative control.

(2) Not subject to Type C leakage tests.

(3) May be leakage tested with water as the test fluid.

(4) Maximum opening time.

(5) Applicability: During CORE ALTERATIONS or movement of irradiated fuel within containment. The provisions of Specification 3.0.4 are not applicable. The containment Purge Exhaust and Supply valves will be locked shut during operation in modes 1, 2, 3, and 4.

(*6) Not subject to the requirments of Specification 3/4.6.3. Listed in TABLE 3.6-1 for information only.

(7) Tested under Type "B" testing.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two separate and independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY:

MODES 1 and 2.

ACTION:

With one hydrogen analyzer inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 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:

a. Performing a CHANNEL CALIBRATION using sample gas containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

3/4.6.4 COMBUSTIBLE GAS CONTROL

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two separate and 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 HOT STANDBY within the next 12 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 at a flow rate of $\geq$ 50 scfm that the heater outlet 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 using containment atmospheric air at a pressure of $\leq$ 13 psia and a flow rate of $\geq$ 50 scfm, that the heater temperature increases to $\geq$ 1,100°F within 5 hours and is maintained for at least 4 hours.

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

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge system shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY:

MODES 1 and 2.

ACTION:

With the containment hydrogen purge system inoperable, restore the hydrogen purge system to OPERABLE status within 30 days or be in HOT STANDBY within 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the purge fan operates for at least 15 minutes.

b. At least once per 18 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove $\geq$ 99 percent of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the purge system at a flow rate of 50 cfm $\pm$ 10 percent.

2. Verifying that the HEPA filter banks remove $\geq$ 99 percent of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the purge system at a flow rate of 50 cfm $\pm$ 10 percent.

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of $\geq$ 90 percent for radioactive methyl iodine at an air flow velocity of 0.11 ft/sec $\pm$ 20 percent with an inlet methyl iodide concentration of 0.15 to 0.5 mg/m^3, $\geq$ 95 percent relative

humidity, and $\geq$ 190°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. 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 50 cfm $\pm$ 10 percent during system operation.

c. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the purge system at a flow rate of 50 cfm $\pm$ 10 percent.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

3.6.5.1 The inside and outside manual isolation valves in the steam jet air ejector suction line shall be closed.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With the inside or outside manual isolation valve in the steam jet air ejector suction line not closed, restore the valve to the closed position within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 The steam jet air ejector suction line outside manual isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 350°F and at least once per 31 days thereafter.

4.6.5.1.2 The steam jet air ejector suction line inside manual isolation valve shall be determined to be sealed or locked in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 350°F.

LIMITING CONDITION FOR OPERATION

3.6.5.2 This Technical Specification intentionally blank.

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 3 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 Setpoint trip 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 2 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 Setpoint trip 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 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 4.7-1, in accordance with Section XI of the ASME boiler and Pressure Vessel Code, 1980 Edition.

TABLE 3.7-1

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	87
2	65
3	43

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 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	56
2	42
3	28

* 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 Diameter
a. SV-MS101A, B, & C	1075 psig	4.515 inches
b. SV-MS102A, B, & C	1085 psig	4.515 inches
c. SV-MS103A, B, & C	1095 psig	4.515 inches
d. SV-MS104A, B, & C	1110 psig	4.515 inches
e. SV-MS105A, B, & C	1125 psig	4.515 inches

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY:

MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the three auxiliary feedwater pumps (two capable of being powered from separate emergency 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.

b. With the motor driven auxiliary feedwater pump supplying the redundant header inoperable, realign the two remaining auxiliary feedwater pumps to the separate headers within 2 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each pump from the control room.

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS

2. Verifying that:

   a. Each motor driven pump develops a discharge pressure of $\geq$ 1335* psig on recirculation flow, and

   b. The steam turbine driven pump develops a discharge pressure of $\geq$ 1335* psig on recirculation flow when the secondary steam pressure is greater than 600 psig.

3. Verifying that each pump operates for at least 15 minutes.

4. Cycling each testable power operated valve in the flow path through at least one complete cycle of full travel.

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

6. Reverifying the requirements of Technical Specification surveillance 4.7.1.2.a.5 by a second and independent operator.

7. Establish and maintain constant communications between the control room and the auxiliary feed pump room while any normal discharge valve is closed during surveillance testing.

8. Verifying operability of each Service Water System auxiliary supply valve by cycling each manual Service Water System to Auxiliary Feedwater System valve through one complete cycle.

9. Following an extended plant outage verify Auxiliary Feedwater Flow from TK-210 to the Steam Generators with the Auxiliary Feedwater Valves in their normal alignment.

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

   1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

   2. Verifying that each automatic valve in the flow path actuates to it correct position on a test signal.

*To be verified during Pre-Operational Testing

3/4.7.1 TURBINE CYCLE

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS

3. Verifying that each pump starts automatically upon receipt of a test signal.

3/4.7.1 TURBINE CYCLE

PRIMARY PLANT DEMINERALIZED WATER (PPDW)

LIMITING CONDITION FOR OPERATION

3.7.1.3 The primary plant demineralized water storage tank shall be OPERABLE with a minimum contained volume of 140,000 gallons.

APPLICABILITY:

MODES 1, 2, and 3.

ACTION:

With less than 140,000 gallons of water in the PPDW storage tank, within 4 hours either:

a. Restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the reactor plant service water system as a backup supply to the auxiliary feedwater pumps and restore the PPDW storage tank water volume to within its limit within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The PPDW storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

ACTIVITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system >0.10uCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 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	Minimum Frequency
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples.
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.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

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 after:

a) The inoperable isolation valve is restored to OPERABLE status, or

b) The isolation valve is maintained closed;

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

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full-stroke closure within 5 seconds on any closure actuation signal, while in HOT STANDBY with $T_{avg} \geq 515F$ during each reactor shutdown except that verification of full closure within 5 seconds need not be determined more often than once per 92 days.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 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 ≤ 200 psig within 30 minutes, and

b. Perform an analysis 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 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 in the steam generator is < 70°F.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two primary component cooling water loops shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, 3 and 4.

ACTION:

With less than two primary component cooling water loops 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 At least two primary component cooling water loops shall be demonstrated OPERABLE:

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

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

2. Verifying that each pump develops at least 93 percent of the discharge pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.

3. Verifying that each pump operates for at least 15 minutes.

4. Cycling each testable power operated or automatic valve servicing safety related equipment through at least one complete cycle of full travel.

5. Verifying that each valve (manual, power operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, in its correct position.

b. At least once per 18 months during shutdown, by cycling each power operated valve servicing safety related equipment that is not testable during plant operation, through at least one complete cycle of full travel.

3/4.7.4 SERVICE WATER SYSTEM (SWS)

LIMITING CONDITION FOR OPERATION

3.7.4 At least two service water loops supplying safety related equipment shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With less than two SWS loops 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 At least two SWS loops shall be demonstrated OPERABLE:

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

1. Starting (unless already operating) each pump from the control room.
2. Verifying that each pump develops at least (later) psig discharge pressure with the discharge valve open.
3. Verifying that each pump operates for at least 15 minutes.
4. Cycling each testable power operated or automatic valve servicing safety related equipment through at least one complete cycle of full travel.
5. 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 cycling each power operated valve servicing safety related equipment that is not testable during plant operation, through at least one complete cycle of full travel.

3/4.7.5 ULTIMATE HEAT SINK - OHIO RIVER

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 654 Mean Sea Level, at the intake structure, and

b. An average water temperature of $\leq$86°F.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Ohio River exceeds 695 feet Mean Sea Level at the intake structure.

APPLICABILITY:

At all times.

ACTION:

With the water level at the intake structure above elevation 695 feet Mean Sea Level:

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

b. Initiate and complete within 8 hours, the following flood protection measures:

1. Install and seal the flood doors in the intake structure.

SURVEILLANCE REQUIREMENTS

4.7.6 The water level at the 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 690 feet Mean Sea Level, and

b. Measurement at least once per 2 hours when the water level is equal to to or above elevation 690 feet Mean Sea Level.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7 The following control room emergency habitability systems shall be OPERABLE:

a. The emergency ventilation system, and

b. The bottled air pressurization system.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With one control room emergency habitability system inoperable, restore the system 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 The emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 104°F.

b. At least once per 31 days by initiating flow through the HEPA filter and charcoal absorber train and verifying that the train operates for 15 minutes.

c. At least once per 12 months or after 720 hours of system operation and 1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or 2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 3) 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 1000 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 1000 CFM $\pm$ 10%.

SURVEILLANCE REQUIREMENTS

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of $\geq$ 90% for radioactive methyl iodide at an air flow velocity of 0.45 ft/sec $\pm$ 20% with an inlet methyl iodide concentration of 0.05 to 0.15 mg/m^3, $\geq$ 95% relative humidity, and $\geq$125°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. 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 1000 cfm $\pm$ 10 percent during system operation.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8.3 inches Water Gauge while operating the ventilation system at a flow rate of 1000 cfm $\pm$ 10 percent.

2. Verifying that on a containment isolation signal, the system automatically starts in 60 minutes and diverts its inlet flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 10 bottles of air, each pressurized to at least 1825 psig.

b. At least once per 18 months by verifying that the system will supply at least 800 cfm of air to maintain the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

LIMITING CONDITION FOR OPERATION

3.7.8 Two SLCRS exhaust air filter trains shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With one SLCRS exhaust air filter train 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.

SURVEILLANCE REQUIREMENTS

4.7.8 Each SLCRS exhaust air filter train shall be demonstrated OPERABLE:

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

1. 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 12 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove $\geq$99 percent 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 59,000 cfm $\pm$ 10 percent.

2. Verifying that the HEPA filter banks remove 99 percent 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 59,000 cfm $\pm$ 10 percent.

SURVEILLANCE REQUIREMENTS

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of $\geq$90 percent for radioactive methyl iodide at an air flow velocity of 0.9 ft/sec $\pm$ 20 percent with an inlet methyl iodide concentration of 0.05 to 0.15 mg/m^3, $\geq$95 percent relative humidity, and $\geq$125°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. 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 59,000 cfm $\pm$ 10 percent during system operation.

c. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <7.83 inches Water Gauge while operating the ventilation system at a flow rate of 59,000 cfm $\pm$ 10 percent.

   2. Verifying that the air flow distribution to each HEPA filter and charcoal adsorber is within $\pm$ 20 percent of the averaged flow per unit.

   3. Verifying that the SLCRS flow is diverted through the filter train on a Containment Isolation - Phase "A" signal.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of those quantities of by-product material listed in 10 CFR 30.71 or $\geq$ 0.1 microcuries of any other material, including alpha emitters, 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 limit 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 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.

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

a. Sources in use (excluding startup sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials.

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

3/4.7.9 SEALED SOURCE CONTAMINATION

SURVEILLANCE REQUIREMENTS

b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources - Each sealed startup source shall be tested prior to being subjected to core flux and following repair or maintenance to the source.

4.7.9.3 Reports - A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 5 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removsable contamination.

LIMITING CONDITION FOR OPERATION

3.7.10 This Technical Specification intentionally blank.

LIMITING CONDITION FOR OPERATION

3.7.11 This Technical Specification intentionally blank.

3/4.7.12 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.12 All snubbers shall be OPERABLE.

APPLICABILITY:

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

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.12.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.12 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after four months, but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7-4a and 3.7-4b. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25 percent from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

* These systems are defined as those portions or subsystems required to prevent releases in excess of 10CFR100 limits.

SURVEILLANCE REQUIREMENTS

No. Inoperable Snubbers Per Inspection Period	Subsequent Visual Inspection Period**#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups; those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify:

1. that there are no visible indications of damage or impaired OPERABILITY,

2. attachments to the foundation or supporting structure are secure, and

3. in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that:

1. the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and

2. the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.12.d or 4.7.12.e, as applicable.

** The inspection interval shall not be lengthened more than one step at a time.

The provisions of Specification 4.0.2 are not applicable.

However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (of at least 10 snubbers or at least 10 percent, whichever is less) of the total of each type of snubber in use in the plant, shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance critera of Specification 4.7.12.d or 4.7.12.e, an additional 10 snubbers or at least 10 percent, whichever is less of that type of snubber shall be functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25 percent of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle;
2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)
3. Snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers as in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

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* Permanent or other exemptions from functional testing for individual snubbers inthese categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion or their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS

If a spare snubber has been installed in place of a failed snubber, the spare snubber shall be retested. Test results of this snubber may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under unidirectional dynamic load, the ability of the snubber to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under unidirectional dynamic load, the ability of the snubber to withstand load without displacement shall be verified.

SURVEILLANCE REQUIREMENTS

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by specification 6.10.2.m.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SSWS)

LIMITING CONDITION FOR OPERATION

3.7.13 At least one Standby Service Water System shall be OPERABLE.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With less than one SSWS system OPERABLE, restore at least one system 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

4.7.13 At least one SSWS system shall be demonstrated OPERABLE.

a. At least once per 31 days by:

1. Starting each pump from its control station.

2. Verifying that each pump develops at least (later) psig discharge pressure while pumping through its test flow line.

3. Verifying that each pump operates for at least 15 minutes.

4. Cycling its power operated discharge valve through at least one complete cycle of full travel.

b. At least once per 18 months during shutdown by starting a Standby Service Water System pump, shutting down one Service Water System, pump, and verifying that the Standby Service Water Subsystem provides at least 8,000 gpm cooling water to that portion of the Primary Service Water System under test for at least 2 hours.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14.1 The fire suppression water system shall be OPERABLE with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header.

b. An OPERABLE flow path capable of taking suction from the Ohio River and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser required to be OPERABLE per Specifications 3.7.14.2 and 3.7.14.4.

c. The booster fire pump, with a capacity of 625 gpm with its discharge aligned to Unit 2 hose rack stations for the safety-related equipment areas.

d. An OPERABLE flow path capable of taking suction from the Service Water System and transferring the water through distribution piping to the first valve ahead of the water flow alarm device on each hose standpipe required to be OPERABLE per Specification 3.7.14.4.

APPLICABILITY:

At all times.

ACTION:

a. With one 2500 gpm pump inoperable, verify the operability of the booster fire pump, and restore the inoperable equipment to OPERABLE status within 7 days or, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the booster fire pump inoperable verify the operability of two 2500 gpm pumps, and restore the inoperable equipment to OPERABLE status within 7 days or, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

c. With the fire suppression water system otherwise inoperable:

1. Establish a backup fire suppression water system within 24 hours, and,

2. Submit a Special Report in accordance with Specification 6.9.2:

   a) By telephone with 24 hours,

   b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

   c) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.14.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. By verifying that the Surveillance Requirements per BVPS 1 Technical Specification 4.7.14.1.1 are met for the two fire pumps (FP-P-1 and FP-P-2) and any other applicable portions of the Unit 1 fire loop which supply the Unit 2 fire loop.

b. At least once per 31 days by starting the booster fire pump and operating it for at least 15 minutes.

c. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

d. At least once per 12 months by performance of a system flush to be performed in conjunction with the Unit 1 system flush, (Unit 1 Technical Specification Surveillance Requirement 4.7.14.1.1.c).

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by:

   1. Verifying that the booster fire pump develops at least 625 gpm at a system head of 250 feet,

FIRE SUPPRESSION WATER SYSTEM

SURVEILLANCE REQUIREMENTS

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that the booster fire pump maintains the fire suppression water system pressure $\geq$ 90 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Section 11, Chapter 5 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.14.1.2 The fire pump diesel engine shall be demonstrated OPERABLE by verifying that the Surveillance Requirements per BVPS 1 Technical Specification 4.7.14.1.2 are met.

4.7.14.1.3 The fire pump diesel starting 24 volt battery bank and charger shall be demonstrated OPERABLE by verifying that the Surveillance Requirements per BVPS 1 Technical Specification 4.7.14.1.3 are met.

3/4.7 PLANT SYSTEMS

3/4.7.14 FIRE SUPPRESSION SYSTEMS

SPRAY SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.2 The following spray systems shall be OPERABLE:

a. Auxiliary Feed Pumps; 2FWS*P22, P23A&B, Safeguards Building.

b. Component Cooling Pumps; 2CCP*P21A,B, and C, Auxiliary Building.

c. Residual Heat Removal Pumps; 2RHS*P21A and B, Reactor Containment.*

d. Charcoal Filters; 2HVS*FLTA 205A and B, 208A and B, Auxiliary Building.

APPLICABILITY:

Whenever equipment in the spray protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.14.2 Each of the above required spray systems 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 accessible during plant operation is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

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*With a containment area spray system inoperable, check this area during scheduled containment entries in Modes 1-4 and once per shift in Modes 5 and 6.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

SPRAY SYSTEMS

SURVEILLANCE REQUIREMENTS

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a manual test signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

c. At least once per 3 years by performing an airflow test through each open spray header and verifying each open spray nozzle is unobstructed.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

LOW PRESSURE CO_2 SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14.3 The 10 ton low pressure CO_2 system (2-FPD System No. 2) serving the following areas shall be OPERABLE with a minimum level of 91 percent and a minimum pressure of 295 psig in one of the two 10 ton storage tanks.

a. Cable Spreading Area Control Building

b. Instrumentation Room Control Building

c. Communication Room Control Building

d. Cable Tunnel Between Control and Auxiliary Buildings

e. Cable Tunnel Auxiliary Building

f. Cable Vault and Rod Control Areas

g. Cable Spreading Area Service Building

h. Emergency Diesel Generator Building Room

APPLICABILITY:

Whenever equipment in the low pressure CO_2 protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required low pressure CO_2 systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

3/4.7.14 FIRE SUPPRESSION SYSTEMS

LOW PRESSURE CO_2 SYSTEM

SURVEILLANCE REQUIREMENTS

4.7.14.3 The above required low pressure CO_2 system shall be demonstrated OPERABLE:

a. At least once per 7 days be verifying the two CO_2 storage tank levels and pressures, and

b. At least once per 18 months by verifying:

   1. The system valves and associated ventilation dampers actuate manually and automatically, upon receipt of a simulated actuation signal, and

   2. Flow from each nozzle during a "Puff Test."

3/4.7.14 FIRE SUPPRESSION SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.14.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY:

Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within one hour (4 hours for containment hose stations) if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.14.4 Each of the above required fire hose stations shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the fire hose stations accessible during plant operation to assure all required equipment is at the station.

b. At least one per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Replacement of all degraded gaskets in couplings.

c. At least one per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig above maximum fire main operating pressure.

TABLE 3.7-5

FIRE HOSE STATIONS

Location	Isolating Valve No.	Elevation	Hose Rack Number
Decon Bldg	182	735' 6"	201
Fuel Bldg	181	752' 5"	202
Fuel Bldg	180	752' 5"	203
Fuel Bldg	179	735' 6"	204
Diesel 2-2 Bldg	523	732' 6"	205
Diesel 2-1 Bldg	526	732' 6"	206
Cable Vault/Contig Area	566	718' 6"	207
Cable Vault/Contig Area	568	718' 6"	208
Cable Vault/Contig Area	558	735' 6"	209
Cable Vault/Contig Area	564	735' 6"	210
Cable Vault/Contig Area	560	755' 6"	211
Cable Vault/Contig Area	562	755' 6"	212
Cable Vault/Contig Area	570	773' 6"	213
Service Bldg	138	780' 6"	214
Service Bldg	137	760' 6"	215
Service Bldg	135	760' 6"	216
Service Bldg	139	745' 6"	217
Service Bldg	142	745' 6"	218
Service Bldg	144	730' 6"	219
Service Bldg	146	730' 6"	220
Auxiliary Bldg	195	710' 6"	241
Auxiliary Bldg	199	773' 6"	242
Auxiliary Bldg	316	755' 6"	243

TABLE 3.7-5

FIRE HOSE STATIONS (CONT'D.)

Location	Isolation Valve No.	Elevation	Hose Rack Number
Auxiliary Bldg	317	735' 6"	244
Auxiliary Bldg	320	718' 6"	245
Auxiliary Bldg	324	773' 6"	246
Auxiliary Bldg	325	755' 6"	247
Auxiliary Bldg	326	735' 6"	248
Auxiliary Bldg	327	718' 6"	249
Auxiliary Bldg	331	773' 6"	250
Auxiliary Bldg	332	755' 6"	251
Auxiliary Bldg	333	735' 6"	252
Auxiliary Bldg	340	718' 6"	253
Containment	772	735' 6"	258
Containment	774	735' 6"	259
Containment	770	735' 6"	260
Containment	777	717' 0"	261
Containment	779	717' 0"	262
Containment	776	717' 0"	263
Containment	783	692' 11"	264
Containment	785	692' 11"	265
Containment	781	692' 11"	266
Containment	764	767' 10"	267
Containment	766	767' 10"	268
Containment	768	767' 10"	269
Auxiliary Bldg	577	712' 6"	270

TABLE 3.7-5

FIRE HOSE STATIONS (CONT'D.)

Location	Isolation Valve No.	Elevation	Hose Rack Number
Auxiliary Bldg	579	712' 6"	271
Cable Tunnel	575	735' 6"	272
Relay Room	573	735' 6"	273
Control Bldg	549	725' 6"	274
Control Bldg	551	725' 6"	275
Control Bldg	553	712' 10"	276
Control Bldg	555	707' 6"	277
Control Bldg	545	712' 6"	278
Safeguards Area	516	737' 6"	287
Safeguards Area	514	737' 6"	288
Safeguards Area	518	718' 6"	289
Safeguards Area	512	718' 6"	290

3/4.7.14 FIRE SUPPRESSION SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.5 The following Halon systems shall be OPERABLE:

a. Control Building Computer Room

b. Control Building West Communications Room

APPLICABILITY:

Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE STATUS within 14 days or, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2, within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system ot OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.14.5 Each of the above required Halon systems 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 is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight and pressure to be at least 90% of full charge pressure.

c. At least once per 18 months by:

1. Verifying the system actuates manually and automatically, upon receipt of a simulated actuation signal, and

2. Visually inspect each header and nozzle to verify their integrity.

3/4.7.15 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.15 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices, in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable and piping penetration seals and ventilation seals) shall be OPERABLE.

APPLICABILITY:

At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol until the functional capability of the barrier is restored.

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

SURVEILLANCE REQUIREMENTS

4.7.15 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assemblies.

b. Performing a visual inspection of each fire window/fire damper and associated hardware.

c. Performing a visual inspection of at least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample with no apparent changes in appearance or abnormal degradation is found.

3/4.7.15 FIRE RATED ASSEMBLIES

SURVEILLANCE REQUIREMENTS

4.7.15.2 Each of the above required fire doors * shall be verified OPERABLE by inspecting the automatic holdopen, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The position of each closed Fire door at least once per 24 hours.

b. That doors with automatic holdopen and release mechanisms are free of obstructions at least once per 24 hours.

*Security alarm fire doors are not included in the above surveillance requirements, since they are checked per security requirements.

3/4.7.16 TERRESTRIAL ECOLOGICAL SURVEY

LIMITING CONDITION FOR OPERATION

3.7.16 The effects of cooling tower drift on terrestrial biota shall be determined.

APPLICABILITY:

At all times.

ACTION:

An assessment of the data, including copies of the latest photographs, gathered in this program element shall be submitted at the end of each alternate year in accordance with Section 5.6.1.

SURVEILLANCE REQUIREMENTS

4.7.16

a. The Terrestrial Ecological Survey program element shall assess the potential impact of the cooling tower drift on the terrestrial vegetation of the site and its vicinity by comparison and evaluation of infrared aerial photographs taken once every two years. Location, extent, and severity of any stressed areas shall be documented and related to the meteorological data. Incipient impacts of cooling tower drift on the terrestrial vegetation of the site and its vicinity shall be detected with color infrared aerial photography. Interpretation of the infrared photographs shall include ground reconnaissance of selected areas.

   Aerial photos shall be taken on color infrared film at a scale of 1 inch = 400 feet. Photos shall be taken between 11 A.M. and 2 P.M. EDT during the middle of the growing season and as close as possible to the same date during alternate years. The flight direction shall remain the same as preoperational flights. Photographs shall be free of cloud shadows. The film processor will insure that processing methods and conditions shall remain the same throughout the study. A flight log shall be compiled when the photographs are taken and processed.

b. Areas with the greatest and least potential for being affected by cooling tower drift shall be selected and compared. The location, extent, and severity of any stressed area shall be documented and related to meteorological data. The possible role of cooling tower drift in the development of stressed areas shall be assessed.

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, and

b. Two separate and independent diesel generators each with:

1. Separate day and engine-mounted fuel tanks containing a minimum of 900 gallons of fuel,

2. A separate fuel storage system containing a minimum of 53,225 gallons of fuel, and

3. A separate fuel transfer pump.

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.5 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 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.5 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 COLD SHUTDOWN within the next 36 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

c. With two of the above required A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.5 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

LIMITING CONDITION FOR OPERATION

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4 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 COLD SHUTDOWN within the next 36 hours.

d. With two 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 one of the inoperable diesel generators to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1 Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the unit circuit to the system circuit.

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 the day tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment,

4. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day and engine-mounted tank,

SURVEILLANCE REQUIREMENTS

5. Verifying the diesel starts from ambient condition,

6. Verifying the generator is synchronized, loaded to greater than or equal to 4,238 kW (continuous rating) and operates for at least 60 minutes, and

7. Verifying the diesel generator is aligned to provide standby power to the associated emergency 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 it's manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a load of greater than or equal to 825 kW (largest single emergency load, a standby service water pump) without tripping,

3. Simulating a loss of offsite power in conjunction with a safety injection signal, and:

   a. Verifying de-energization of the emergency busses and load shedding from the emergency busses,

   b. Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads.

4. Verifying that on a loss of power to the emergency busses, all diesel generator trips, except engine overspeed, generator differential and overcurrent, are automatically disabled,

5. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 4,238 kW,

6. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 4,535kW,

7. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within $\pm$ 10 percent of its required value.

3/4.8.1 A.C. SOURCES

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. One diesel generator with:

1. Day and engine-mounted fuel tanks containing a minimum of 900 gallons of fuel.

2. A fuel storage system containing a minimum of 52,325 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 4.8.1.1.2.a.6.

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 with tie breakers open between redundant busses:

4160 volt Emergency Bus #2AE and 480 volt Emergency Bus #2N

4160 volt Emergency Bus #2DF and 480 volt Emergency Bus #2P

120 volt A.C. Vital Bus #I

120 volt A.C. Vital Bus #II

120 volt A.C. Vital Bus #III

120 volt A.C. Vital Bus #IV

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.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator.

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Bus

2 - 120 volt A.C. Vital Busses

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.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE.

TRAIN "A" (orange) consisting of 125-volt D.C. busses No. 2-1 & 2-3, 125-volt D.C. battery banks 2-1 & 2-3 & charger 2-1 & inverter 2-3.

TRAIN "B" (purple) consisting of 125-volt D.C. busses No. 2-2 & 2-4, 125-volt D.C. battery banks 2-2 & 2-4 & charger 2-2 & inverter 2-4.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank 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 of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its asociated battery bank by performing Surveillance Requirement 4.8.2.3.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 3.8-1 is not met, declare the battery inoperable.

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 indicated power availability.

4.8.2.3.2 Each 125-volt battery bank, charger, and inverter shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 3.8-1 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 127.8 - volts on float charge.

SURVEILLANCE REQUIREMENTS

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 - volts, or battery overcharge with battery terminal voltage above 150 - volts, by verifying that:

1. The parameters in Table 3.8-1 meet the Category B limits.

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of every tenth cell of connected cells is above 60°F.

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 resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms; and

4. The battery charger will supply at least (100) amperes at 140 -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 or simulated emergency loads for the 2-hour design duty cycle when the battery is subjected to a battery service test.[1]

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80 percent of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months, during shutdown, performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85 percent of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10 percent of rated capacity from its average on previous performance tests, or is below 90 percent of the manufacturer's rating.

[1] Load testing conducted pursuant to IEEE 450-1980.

TABLE 3.8-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥1.200(b)	≥ 1.195 Average of all connected cells > 1.205	Not more than .020 below the average of all connected cells Average of all connected cells ≥ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than (2) amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the batery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

Numbers in parentheses assume a manufacturer's recommended full charge specific gravity of 1.215.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

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. bus systems, and

2 - 125-volt battery bank and chargers/inverters associated with the above D.C. bus systems.

APPLICABILITY:

MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus system OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus system 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 battery bank and chargers/inverters shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.

3/4.9 REFUELING OPERATIONS

3/4.9.1 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 percent k/k conservative allowance for uncertainties, or

b. A boron concentration of 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 $\geq$ 30 gpm of 7000 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.

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* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

3/4.9.2 INSTRUMENTATION

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 lest once per 12 hours during CORE ALTERATIONS.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 150 hours.

APPLICABILITY:

During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 150 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 150 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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. Exhausting at less than or equal to 7500 cfm through OPERABLE Containment Purge and Exhaust Isolation Valves with isolation times as specified in Table 3.6-1 to OPERABLE HEPA filters and charcoal adsorbers of the Supplemental Leak Collection and Release System (SLCRS).

APPLICABILITY:

During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirement of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition within 150 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

SURVEILLANCE REQUIREMENTS

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE by:

a. Verifying the flow rate through the SLCRS at least once per 24 hours when the system is in operation,

b. Testing the Containment Purge and Exhaust Isolation Valves per the applicable portions of Specification 4.6.3.2, and

c. Testing the SLCRS per Specification 4.7.8.

3/4.9.5 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.

3/4.9 REFUELING OPERATIONS

3/4.9.6 MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive 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 cutoff limit less than or equal to 2850 pounds.

b. The Auxiliary hoist used for latching and unlatching drive 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 drive 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 drive 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 150 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 700 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 3000 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 Crane interlocks and physical stops which prevent crane travel with loads in excess of 3000 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.

3/4.9.8 COOLANT CIRCULATION

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) 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 residual heat removal loop may be removed from operation for up to 4 hours per 8 hour period during the performance of Ultrasonic In-service Inspection inside the reactor vessel nozzles provided there is at least 23 feet of water above the top of the reactor vessel flange.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm at least once per 4 hours when making boron dilution changes and $\geq$ 1000 gpm for decay heat removal when the Reactor Coolant System is in the drained down condition within the loops.

3/4.9.8 COOLANT CIRCULATION

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY:

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

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

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* The normal or emergency power source may be inoperable for each RHR loop.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY:

During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Containment Purge and Exhaust Isolation System inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust Isolation System shall be demonstrated OPERABLE within 150 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high-high radiation test signal from each of the containment radiation monitoring instrumentation channels.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY:

During movement of fuel assemblies or 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 movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies seated in the storage racks.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours. 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 area.

3/4.9.12 FUEL BUILDING VENTILATION SYSTEM - FUEL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.9.12 The fuel building ventilation system shall be operating and discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers during either:

a. Fuel movement within the spent fuel storage pool, or

b. Crane operation with loads over the spent fuel storage pool.

APPLICABILITY:

When irradiated fuel which was decayed less than 60 days is in the fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The fuel building ventilation system shall be verified to be operating with all building doors closed within 2 hours prior to the initiation of and at least once per 12 hours during either fuel movement within the fuel storage pool or crane operation with loads over the fuel storage pool.

3/4.9.13 FUEL BUILDING VENTILATION SYSTEM - FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 The Supplemental Leak Collection and Release System (SLCRS) portion of the fuel building ventilation system shall be OPERABLE.

APPLICABILITY:

Whenever irradiated fuel is in the storage pool.

ACTION:

Without the SLCRS portion of the fuel building ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel building ventilation system is restored to OPERABLE status. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The SLCRS portion of the fuel building ventilation system shall be demonstrated OPERABLE by:

1. Verifying that on a high-high radiation signal, the system automatically directs its exhaust flow through the HEPA filters and charcoal adsorber banks of the SLCRS at least once per 18 months.

2. Testing the SLCRS per Specification 4.7.8.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN margin requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY:

MODE 2.

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 30 gpm of 7000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 30 gpm of 7000 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 rod either partially or fully withdrawn shall be determined at least once per 2 hours.

2. Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50 percent withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.2 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, 3.1.3.5, 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 $\leq$ 85 percent 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, 3.1.3.5, 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 $\leq$ 85 percent of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

2. The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

1. Specification 4.2.2, at least once per 12 hours.

2. Specification 4.2.3, at least once per 12 hours.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.3 PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$ 25 percent of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figures 3.4-2 and 3.4-3.

APPLICABILITY:

MODE 2.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the unacceptable region of operation on Figures 3.4-2 and 3.4-3, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3

1. The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figures 3.4-2 and 3.4-3 at least once per hour.

2. The THERMAL POWER shall be determined to be $\leq$ 5 percent of RATED THERMAL POWER at least once per hour.

3. Each Intermediate and Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

3/4.10.4 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 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 percent of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$ 25 percent of RATED THERMAL POWER, and

APPLICABILITY:

MODE 2.

ACTION:

With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4

1. The THERMAL POWER shall be determined to be $\leq$ 5 percent of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

2. Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

3/4.10.5 NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.5 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 at $\leq$ 25 percent 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

4.10.5

1. The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

2. Each Intermediate and 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.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.6 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.6 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided:

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY:

MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the position indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.6 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree within 12 steps when the rods are stationary.

* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (See Figure 5.1-2) shall be limited to the concentration specified in 10 CFR 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} uCi/ml total activity.

APPLICABILITY:

At all times.

ACTION:

a. With the concentration of radioactive material released from the site to unrestricted areas exceeding the above limits, immediately restore concentration within the above limits.

b. Immediately notify the Commission pursuant to 10 CFR 50.72 (declaration of any of the Emergency Classes specified in the EPP), and

c. Submit a Special Report to the Commission within 30 days in accordance with Specification 6.9.2

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

SURVEILLANCE REQUIREMENTS

4.11.1.1

1. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1*.

2. The results of radioactive analysis shall be used in accordance with the methods of the ODCM to assure that the concentration at the point of release are maintained within the limits of Specification 3.11.1.1.

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* Radioactive liquid discharges are normally via batch modes. Turbine Building Drains shall be monitored as specified in paragraph 4.11.1.1.3. Recirculation Drain Pump discharge shall be monitored as specified in paragraph 4.11.1.1.4.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

SURVEILLANCE REQUIREMENTS

4.11.1.1.3

When the gross activity of the secondary coolant is greater than 10^{-5} uCi/ml and the Turbine Building Transfer Pumps (2DGS-P42, 2DBS-P43, 2DBS-P44) are not pumping their sumps to the steam generator blowdown tank (2SGC-TK21B) grab samples shall be taken for each sump discharge from the turbine building. The sample shall be analyzed for gross activity at a sensitivity of at least 10^{-7} uCi/ml and recorded in the plant records. Water volume discharged shall be estimated from the number of pump operations unless alternate flow or volume instrumentation is provided.**

4.11.1.1.4

Prior to the Recirculation Drain Pump(s), 2DAS-P215A/B discharging to catch basin number 16 a grab sample will be taken. The samples will be analyzed for gross activity at a sensitivity of at least 10^{-7} uCi/ml and recorded in the plant records. Water volume discharged shall be estimated from the number of pump operations unless alternate flow or volume instrumentation is provided.***

** Refer to Figure 5.1-3 for the location and drain paths to the river for the normal drains from the Turbine Building into manhole 4, catch basin-6 and catch basin-11.

*** Refer to Figure 5.1-3 for the location and drain paths to the river for the discharge of the recirculation Drain Pump(s) into catch basin-16.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Mimimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml) (a)
A. Batch Waste Release Tanks (d)	P Each Batch (h)	P Each Batch (h)	Principal Gamma Emitters (f)	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M (h)	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch (h)	M Composite (b)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch (h)	Q Composite (b)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Release (e) (g)	Grab Sample (g)	W Composite (c)	Principal Gamma Emitters (f)	5×10^{-7}
			I-131	1×10^{-6}
	Grab Sample (g)	M	Dissolved and Entrained Gases (Gamma Gases)	1×10^{-5}
	Grab Sample (g)	M Composite (c)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Grab Sample (g)	Q Composite (c)	Sr-89 Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1

NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95 percent probability with 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation)

$$LLD = \frac{4.66\ S_b}{(E)\ (V)\ (2.22)\ (Y)\ \exp\ (-\lambda\ \text{Delta T})}$$

where

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume);

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per disintegration);

V is the sample size (in units of mass or volume)

2.22 is the number of disintegration per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide;

Delta T is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on a unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g. potassium in milk samples). Typical values of E, V, Y, and T should be used in the calculations.

The LLD is defined as a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

TABLE 4.11-1

NOTATION

(continued)

b. A composite sample is one in which the quantity of liquid sampled as proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentration of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release exists when the discharge of liquid wastes is from a discrete volume. Prior to sampling for analysis, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

e. A continuous release exists when the discharge of liquid wastes is from a nondiscrete volume; e.g. from a volume of a system having an input flow during the continuous release. This is applicable to the Turbine Building drains when the secondary coolant gross radioactivity (beta and gamma) is greater than 10^{-5} uCi/ml.

f. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should be reported as "less than" the nuclide's LLD, and should not be reported as being present at the LLD level for that nuclide. The "less than" values should not be used in the required dose calculations. When unusual circumstances result in LLD's higher than required the reasons shall be documented in the semi-annual Radioactive Effluent Release Report.

g. Whenever there is primary to secondary leakage occuring, sampling is done for turbine building drain effluents by means of grab samples taken every hour during the period of discharge and analyzed for gross radioactivity (beta and gamma) at a sensitivity of at least 10^{-7} uCi/ml and recorded in the plant records, along with the flow rate. Primary to secondary leakage is considered to be occurring whenever measurements indicate that secondary coolant gross radioactivity (beta and gamma) is greater than 10^{-5} uCi/ml. In addition, two plant personnel shall check release calculations to verify that the limits of 3.11.1.1 and 3.11.1.2 are not exceeded.

TABLE 4.11

NOTATION

(continued)

h. Whenever the Recirculation Drain Pump(s) are discharging to catch basin 16 sampling will be performed by means of a grab sample taken every hour during pump operation.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose committments to MEMBER(S) OF THE PUBLIC from radioactive materials in liquid effluents released from the site (see Figure 5.1-2) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases, and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits. (This Special Report shall also include

1. the results of radiological analyses of the drinking water source and,

2. the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act).*

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

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* Applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. (3 miles down stream only).

3/4.11.1 LIQUID EFFLUENTS

DOSE

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations - Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be used to reduce the radioactive materials in each liquid waste batch prior to its discharge when the projected doses due to liquid effluent releases from the site (See Figure 5.1-2) when averaged over 31 days would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

a. With liquid waste being discharged without treatment and exceeding the limits specified, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to operational status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUELNTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to ≤ 8.5 curies, excluding tritium and dissolved or entrained noble gases.

a. Refueling Water Storage Tank

b. Miscellaneous temporary outside radioactive liquid storage tanks.

APPLICABILITY:

At all times.

ACTION:

a. With the quantity of radioactive material in the above tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2 and include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in the above tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in the unrestricted areas (see Figure 5.1-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be ≤ 500 mrem/year to the total body and ≤ 3,000 mrem/year to the skin, and

b. The dose rate limit, inhalation pathway only, for I-131, tritium and all radionuclides in particulate form (excluding C-14) with half-lives greater than 8 days shall be ≤ 1,500 mrem/year to any organ.

APPLICABILITY:

At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately decrease the release rate to comply with the above limit(s).

b. Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.1

1. The dose rate due to noble gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

2. The dose rate, inhalation pathway only, for I-131, tritium and all radionuclides in particulate form (excluding C-14) with half-lives greater than 8 days in gaseous effluents, shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
B. Containment Exhaust	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. Ventilation Systems[h] 1. Process Vent 2. Elevated Release Point 3. Ventilation Vent 4. Aux. Bldg. Exh. Hood (Emergency Use Only) 5. Cond. Polish. Bldg. Vent. Exh. 6. Decon. Bldg. Vent. Exh. 7. Waste Gas Storage Vault Vent. Exh. 8. Turbine Building Exhaust Release from Radioiodine and Particulates (Airborne) may be limited to the Inhalation Pathway only.	M[b,c,e] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}

TABLE 4.11-2 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)
D. All systems listed above which produce continuous release.*	Continuous[f]	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1×10^{-6}

* Release from radioiodine and particulates (Airborne) may be limited to the inhalation pathway only.

TABLE 4.11-2
(Continued)

TABLE NOTATION

a. The Lower Limit of Detection (LLD) is defined in Table Notation (a) of Table 4.11-1 of Specification 4.11.1.1.

b. When reactor coolant system activity exceeds the limits stated in Specification 3.4.8, analyses shall be performed once every 24 hours during startup, shutdown and 25 percent load changes and 72 hours after achieving the maximum steady state power operation unless continuous monitoring is provided.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling and analyses shall also be performed at least once per 24 hours, during startup, shutdown and 25 percent load changes and 72 hours after achieving the maximum steady state power operation when RCS activity exceeds the limits in Specification 3.4.8 unless continuous monitoring is provided. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The average ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specification 3.11.2.1, 3.11.2.2, and 3.11.2.3.

g. The principal gamma emitters for which the LLD Specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLD's high than required, the reasons shall be documented in the semi-annual effluent report.

h. Only when this release path is in use.

3/4.11.2 GASEOUS EFFLUENTS

DOSE, NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose in unrestricted areas (See Figure 5.1-1) due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to $\leq$ 5 mrad for gamma radiation and $\leq$ 10 mrad for beta radiation.

b. During any calendar year, to $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation.

APPLICABILITY:

At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the causes(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be within the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions shall be determined in accordance with the ODCM at least once every 31 days.

3/4.11.2 GASEOUS EFFLUENTS

DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to MEMBER(S) OF THE PUBLIC from radioiodines and radioactive materials in particulate form (excluding C-14), and radionuclides (other than nobles gases) with half-lives greater than 8 days in gaseous effluents released from the site (See Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter to $\leq$ 7.5 mrem to any organ, and

b. During any calendar year to $\leq$ 15 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

a. With the calculated dose from the release of radioiodines, radioactive materials in particulate form (excluding C-14), and radionuclides (other than noble gases) with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission with 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the causes(s) for exceeding the limit and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations. Cumulative dose contributions shall be determined in accordance with the ODCM at least once every 31 days.

3/4.11.2 GASEOUS EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-1), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-1) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY:

At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

 1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

 2. Action(s) taken to restore the inoperable equipment to operational status, and

 3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

GASEOUS WASTE STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.5 The quantity of radioactivity contained in each gaseous waste storage tank shall be limited to 20,000 curies noble gases (considered as Xe-133).

APPLICABILITY:

At all times.

ACTION:

a. With the quantity of radioactive material in any gaseous waste storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. Submit a Special Report to the Commission within 30 days pursuant to Specification 6.9.2 and include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5

1. The quantity of radioactive material contained in each gaseous waste storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank when the Gaseous Waste Storage Tank Radiation Monitor (2GWS-RQI104) is not operable.

2. The Gaseous Waste Storage Tank Radiation Monitor (2GWS-RQI104) operability shall be determined in accordance with Table 4.3-13 unless sampling pursuant to Specification 4.11.2.5.1 is being conducted.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of oxygen in the waste gas holdup system shall be limited to $\leq$ 2 percent by volume whenever the hydrogen concentration exceeds 4 percent by volume.

APPLICABILITY:

At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system > 2 percent by volume, immediately suspend all additions of waste gases to the gaseous waste decay tank and reduce the concentration of oxygen to $\leq$ 4 percent within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4 percent by volume and the hydrogen concentration greater than 2 percent by volume, immediately suspend all additions of waste gases to the affected tank and reduce the concent ation of oxygen to less than or equal to 2 percent by volume wit in 12 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are no: applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentrations of oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10 or monitoring in conjunction with its associated action statement.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used, as applicable, to solidify and package radioactive wastes, and to ensure meeting the requirements of 10 CFR 20 and of 10 CFR 71. Methods utilized to meet these requirements shall be described in facility procedures and in the Process Control Program (PCP).

APPLICABILITY:

At all times.

ACTION:

a. With the applicability requirements of 10 CFR 20 and 10 CFR 71 not satisfied, suspend affected shipments of solid radioactive wastes from the site.

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

SURVEILLANCE REQUIREMENTS

4.11.3

1. Prior to shipment, solidification shall be verified in accordance with Station Operating Procedures.

2. The semi-annual Radioactive Effluent Release Report in Specification 6.9.1.12 shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume;

b. Total curie quantity (determined by measurement or estimate);

c. Principal radionuclides (determined by measurement or estimate);

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms);

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity); and

f. Solidification agent (e.g., cement, urea formaldeyde).

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitment to MEMBER(S) OF THE PUBLIC from all facility releases is limited to $\leq$ 25 mrem to the total body or any organ (except the thyroid, which is limited to $\leq$ 75 mrem) for a calendar year.

APPLICABILITY:

At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, prepare and submit a special report to the commission pursuant to Specification 6.9.2 defining the corrective action and limit the subsequent releases such that the dose or dose commitment to MEMBER(S) OF THE PUBLIC is limited to $\leq$ 25 mrem to the total body or any organ (except thyroid, which is limited to $\leq$ 75 mrem for a calendar year. This special report shall describe the steps to be taken or modifications necessary to prevent a recurrence. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR 190 Standard.

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

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations. Cummulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, and 3.11.2.3.b, and in accordance with the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY:

At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.)

b. With the level of radioactivity in an environmental sampling medium at one or more of the locations specified in Table 3.12-1 exceeding the limits of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of affected calendar quarter a Report pursuant to Specification 6.9.1.9, which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits of Table 3.12-2 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Report.

When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Limit Level (1)}} + \frac{\text{Concentration (2)}}{\text{Limit Level (2)}} + \ldots \geq 1.0$$

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

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c. With milk or fresh leafy vegetable samples unavailable from the required number of locations selected in accordance with Specification 3.12.2 and listed in the ODCM, obtain replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1 and the ODCM provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations, if available.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MOITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type of Frequency (a) of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	5 Locations 1. One sample form a control location 10-20 miles distant and in the least prevalent wind direction.	Continuous operation of sampler with sample collection at least weekly.	Each radioiodine canister. Analyze for I-131; Particulate sampler. Analyze for gross beta weekly(b); Perform gamma isotopic analysis composite (by location).
	2. One sample from vicinity of community having the highest calculated annual average ground level D/Q.		Sample at least quarterly.
2. DIRECT RADIATION	40 Locations. $\geq$ 2 TLD or a pressurized ion chamber at each location.	Continuous measurement with collection at least quarterly.	Gamma dose, quarterly.

(a) Analysis frequency same as sampling frequency unless otherwise specified.

(b) Particulate samples are not counted for $\geq$ 24 hours after filter change. Perform gamma isotopic analysis on each sample when gross beta is > 10 times yearly mean of control samples.

** Sample locations are given on figures and table in Offsite Dose Calculation Manual (ODCM).

TABLE 3.12-1
(Continued)

RADIOLOGICAL ENVIRONMENTAL MOITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type of Frequency (a) of Analysis
3. WATERBORNE			
a. Surface	2 Locations 1. One sample upstream. 2. One sample downstream.	Composite* sample collected over a period not to exceed one month.	Gamma isotopic analysis of each composite sample; Tritium analysis of composite sample at least quarterly.
b. Drinking	2 Locations	Composite* sample collected over a period not to exceed 2 weeks	I-131 analysis of each composite sample; Gamma isotopic analysis of composite sample (by location) monthly; Tritium analysis of composite sample quarterly.
c. Groundwater	N/A - No wells in lower elevations between plant and river.		
d. Sediment from Shoreline	1 Location	Semi-Annually	Gamma Isotopic analysis semi-annually.

* Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.

** Sample locations are shown on figures and tables in the Offsite Dose Calculation manual (ODCM).

(a) Analysis frequency same as sampling frequency unless otherwise specified.

TABLE 3.12-1
(Continued)

RADIOLOGICAL ENVIRONMENTAL MOITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type of Frequency (a) of Analysis
4. INGESTION			
a. Milk	4 Locations (c) 1. Three samples selected on basis of highest potential thyroid dose using milch census data. 2. One local large dairy.	At least bi-weekly when animals are on pasture; at least monthly at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fishing	2 Locations	Semi-annual. One sample of available species.	Gamma isotopic analysis on edible portions.
c. Food Products (Leafy Vegetables)	4 Locations 1. Three Locations within 5 miles. 2. One control location.	Annually at time of harvest.	Gamma isotopic analysis and I-131 analysis on edible portion.

** Sample locations are shown on figures and tables in the Offsite Dose Calculation manual (ODCM).

(a) Analysis frequency same as sampling frequency unless otherwise specified.

(c) Other dairies may be included as control station or for historical continuity. These would not be modified on basis of milch animal census.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water pCi/l	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg. wet)	Milk (pCi/l)	Broad Leaf Vegetables (pCi/Kg. wet)
H-3	2×10^4 (a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

(a) For drinking water samples. This is a 40 CFR 141 value.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a]

Analysis	Water pCi/l	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/Kg. wet)	Milk (pCi/l)	Food Products (pCi/Kg. wet)	Sediment (pCi/Kg. dry)
Gross Beta	4	1×10^{-2}				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-95	30[c]					
NB-95	15[c]					
I-131	1[b]	7×10^{-2}		1	60	
Cs-134	15	5×10^{-2}	130	15	60	150
Cs-137	18	6×10^{2}	150	18	80	180
Ba-140	60[c]			60		
La-140	15[c]			15		

NOTE: This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall be identified and reported.

TABLE 4.12-1
(Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95 percent probability with 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 s_b}{(E)\ (V)\ (2.22)\ (Y)\ \exp(-\lambda\ \text{Delta T})}$$

where:

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume);

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute);

E is the counting efficiency (as counts per disintegration);

V is the sample size (in units of mass or volume);

2.22 is the number of disintegrations per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide;

Delta T is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of S_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., pottassium 40 in milk samples). Typical values of E, V, Y and delta T should be used in the calculations.

The LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as *a posteriori* (after the fact) limit for a particular measurement.

b. LLD for drinking water.

c. If parent and daughter are totaled, most restrictive LLD should be applied.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. (For elevated releases defined in Regulatory Guide 1.111, (Revision 1) July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.)

APPLICABILITY:

At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the new location(s).

b. With a land use census identifying a milch animal location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1 prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the new location. This report will also include copies of the revised figures and tables to be placed in the ODCM. the new location shall be added to the radiological environmental monitoring program within 30 days, if possible. The milk sampling program shall include samples from the three active milch animal locations, having the highest calculated dose or dose commitment. Any replaced location may be deleted from this monitoring program after (October 31) of the year in which the land use census was conducted.

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

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* Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

3/4.12.2 LAND USE CENSUS

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey,* aerial survey, or by consulting local agriculture authorities.

* Confirmation by telephone is equivalent to door-to-door.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program.

APPLICABILITY:

At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The results of analyses performed as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Report.

BASES

FOR

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Condition 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 if one accumulator is inoperable. Under the terms of Specification 3.0.3, if more than one accumulator is inoperable, the unit is required to be in at least HOT STANDBY within 1 hour and in an least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems, to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable: Under the terms of Specification 3.0.3, if both of the required Containment Spray Systems are inoperable, the unit is required to be in at least HOT STANDBY within 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in at least COLD SHUTDOWN in the next 24 hours. It is assumed that the unit is brought to the required MODE within the required times by promptly initiating and carrying out the appropriate ACTION statement.

3.0.4 This specification provides that entry into an OPERABLE 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.

3.0.5 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of the specification permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for a 72 hour out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components, and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, action is required in accordance with this specification.

As a further example, Specification 3.8.1.1 requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. The ACTION statement provides a 24 hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits, would also be inoperable. This would dictate invoking the applicable ACTION statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE,, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, action is required in accordance with this specification.

In MODES 5 or 6 Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations.

The 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 sat forth the criteria for determinaton 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.

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.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedance over the ASME Boiler and Pressure Vessal Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal opration. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedance over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

BASES

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 1.77 percent 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 accident analysis assumptions. With T_{avg} < 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1 percent $\Delta k/k$ shutdown margin provides adequate protection.

The purpose of borating to the cold shutdown boron concentration prior to blocking safety injection is to preclude a return to criticality should a steam line break occur during plant heatup or cooldown.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

BORON DILUTION

BASES

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 9370 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therfore be within the capability for operator recognition and control.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

MODERATOR TEMPERATURE COEFFICIENT (MTC)

BASES

The limitations on Moderator Temperature Coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The Surveillance Requirement for measurement of the MTC at the beginning and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1 BORATION CONTROL

MINIMUM TEMPERATURE FOR CRITICALITY

BASES

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 pressurizer is capable of being in an OPERABLE status with a steam bubble,
3. The reactor pressure vessel is above its minimum RT_{NDT} temperature, and
4. The protective instrumentation is within its normal operating range.

3/4.1.2 BORATION SYSTEMS

BASES

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 1.0 percent delta k/k after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at EOL from full power equilibrium xenon conditions and requires 13,390 gallons of 7000 ppm borated water from the boric acid storage tanks or 58,965 gallons of 2000 ppm borated water from the refueling water storage tank. The associated technical specification limit on the refueling water storage tank volume of water while operating has been established at 859,248 gallons to account for reactivity considerations, the NPSH requirements of the ECCS system, and the water required for containment spray operation.

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 boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1 percent delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2315 gallons of 7000 ppm borated water from the boric acid storage tanks or 10,196 gallons of 2000 ppm borated water from the refueling water storage tank. The associated technical specification limit on refueling water storage tank volume of water while below 200°F has been established at 217,000 gallons to account for reactivity considerations and the NPSH requirements of the ECCS system.

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.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

BASES

The specifications of this section ensure that:

1. acceptable power distribution limits are maintained,

2. the minimum SHUTDOWN MARGIN is maintained, and

3. the potential effects of rod misalignment on associated accident analyses are limited.

OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within $\pm$ 12 steps (indicated position), of the respective group demand counter position. OPERABILITY of the control rod position indicators is required to determine control rod position 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 design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are re-evaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 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.

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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

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 Conditions I events provides assurance that the initial conditions asumed 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.

The limits on AXIAL FLUX DIFFERENCE assure that the F_Q (Z) upper bound envelope of 2.18 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. 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 oth - THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWF alue bvy the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

BASES

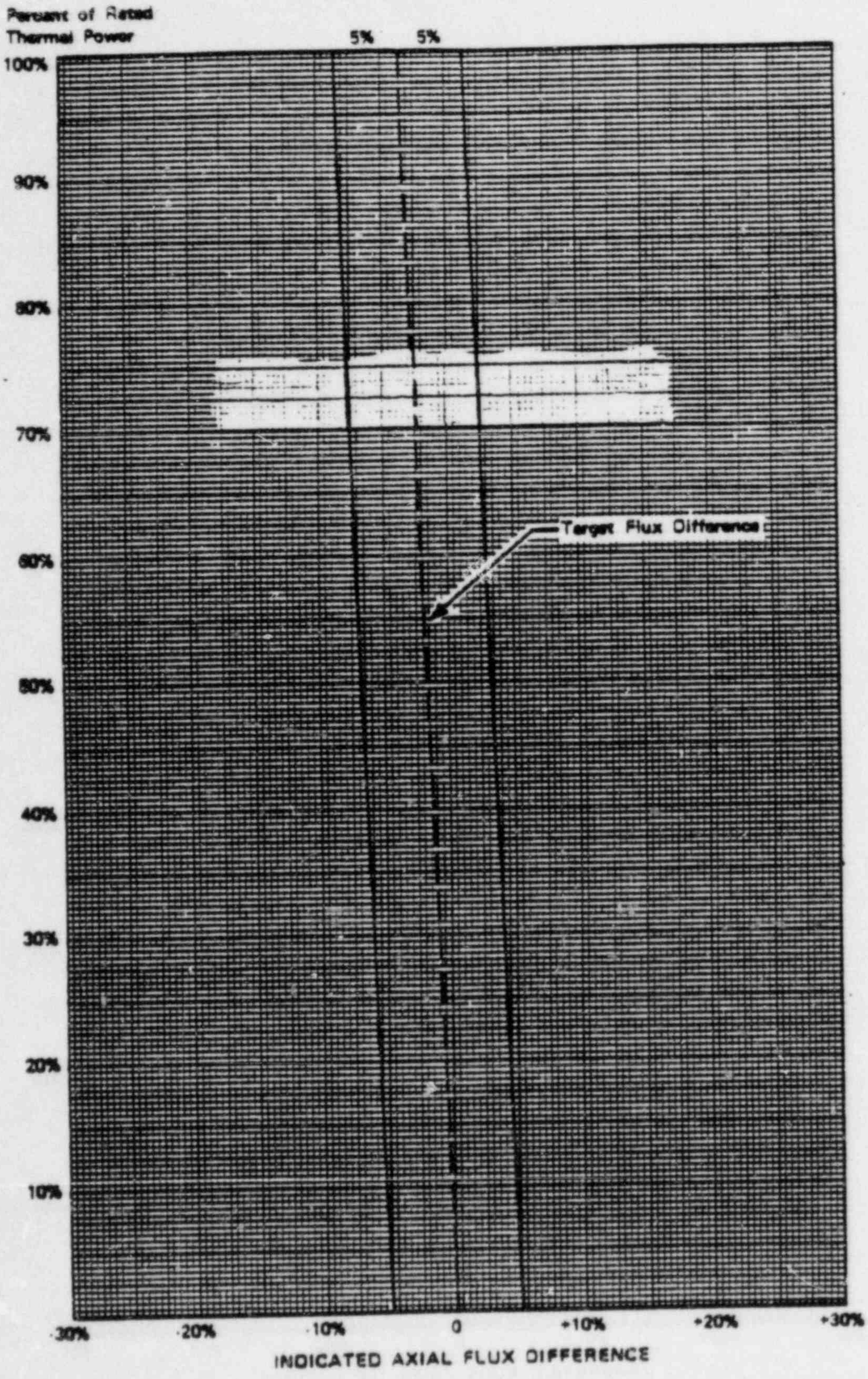
Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5 percent target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER 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 percent and 90 percent of RATED THERMAL POWER. For THERMAL POWER levels between 15 percent and 50 percent 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 on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90 percent of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50 percent and 90 percent and 15 percent and 50 percent RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL



3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

3/4.2.3. NUCLEAR ENTHALPY FACTOR - $F^N_{\Delta H}$

BASES

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 ECCS acceptance criteria limit of 2200°F.

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 ± 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 3.1.3.4 and 3.1.3.5 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 through d above, are maintained.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

3/4.2.3 NUCLEAR ENTHALPY FACTOR - $F^N_{\Delta H}$

BASES

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate experimental error allowance for a full core map taken with the incore detector flux mapping system and 3 percent is the appropriate allowance for manufacturing tolerance.

The specified limit of $F^N_{\Delta H}$ contains an 8 percent allowance uncertainties which means that normal, full power, three loop operation will result in $F^N_{\Delta H}$ is less than or equal to 1.55/1.08.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties (less than 3% for the worst case which occurs at a burnup of 33,000 MWD/MTU). This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28

2) Grid Spacing (Ks) of 0.046 vs. 0.059

3) Thermal Diffusion Coefficienc of 0.038 vs. 0.059

4) DNBR Multiplier of 0.865 vs. 0.88

5) Pitch reduction

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limit.

The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

BASES

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.

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 maximum allowed, power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

BASES

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 through 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 INSTRUMENTATION

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

BASES

The OPERABILITY of the Engineered Safety Feature Actuation System instrumentation and interlocks ensure that:

1) The associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint,

2) The specified coincidence logic is maintained,

3) Sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and

4) Sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, 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 actuation 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.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

BASES

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - actuates turbine trip, closes main feedwater valves on T avg below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 Above the setpoint P-11 automatically reinstates safety injection actuation on Low pressurizer pressure, automatically blocks steamline isolation on high steam pressure rate, enables safety injection and steamline isolation on (Loop Stop Valve Open) with low steamline pressure, and enables auto actuation of the pressurizer PORVs.

Below the setpoint P-11 allows the manual block of safety injection actuation on low pressurizer pressure, allows manual block of safety injection and steamline isolation on (Loop Stop Valve Open) with low steamline pressure and enabling steamline isolation on high steam pressure rate, automatically disables auto actuation of the pressurizer PORVs unless the Reactor Vessel Over Pressure Protection System is in service.

P-12 Above the setpoint P-12 automatically reinstates an arming signal to the steam dump system. Below the setpoint P-12 blocks steam dump and allows manual bypass of the steam dump block to cooldown with condenser dump valves.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

BASES

The OPERABILITY of the radiation monitoring channels ensures that:

1) The radiation levels are continually measured in the areas served by the individuals channels;

2) The alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and

3) Sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of the TMI Action Plan Requirements," October, 1980.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

MOVABLE INCORE DETECTORS

BASES

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 determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or F^N_{H}, a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the Quadrant Power Tilt Ratio when one Power Range Channel is inoperable.

3/4.3.3 MONITORING INSTRUMENTATION

MOVABLE INCORE DETECTORS

BASES

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 determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$, a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the Quadrant Power Tilt Ratio when one Power Range Channel is inoperable.

3/4.3.3 MONITORING INSTRUMENTATION

SEISMIC INSTRUMENTATION

BASES

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 a measured response to that used in the design basis for the facility and is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes."

3/4.3.3 MONITORING INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

BASES

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 and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs."

3/4.3.3 MONITORING INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

BASES

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside the control room. This capability is required in the event control room habitability is lost and this capability is consistent with GENERAL DESIGN CRITERIA 19 of 10 CFR 50.

3/4.3.3 MONITORING INSTRUMENTATION

FIRE DETECTION

BASES

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3 MONITORING INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

BASES

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January, 1977.

3/4.3.3 MONITORING INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

BASES

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on seleted plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to assess Plant Conditions During and Following an Accident", December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3 MONITORING INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

BASES

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive materials in liquid effluent during actual or potential liquid releases. The alarm/trip setpoint for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of GENERAL DESIGN CRITERIA 60, 63 and 64 of Appendix A to 10 CFR 50.

3/4.3.3 MONITORING INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

BASES

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated. in accordance with the procedure in the ODCM to ensure the alarm/trip will occur prior to exceeding the limits of 10 CFR 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of GENERAL DESIGN CRITERIA 60, 63 and 64 of Appendix A to 10CFR 50.

3/4.4.1 REACTOR COOLANT LOOPS

BASES

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. In MODES 1 and 2, with one reactor coolant loop not in operation, THERMAL POWER is restricted to $\leq$ 30 percent of RATED THERMAL POWER until the Overtemperature T 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 (10 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (30 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, the FSAR safety analyses assumed two reactor coolant pumps operating.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this Specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCP's to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.1 REACTOR COOLANT LOOPS

BASES

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop immediately prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delays isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

BASES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is deisgned to relieve 345,000 pounds per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressure.

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 setpoint 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 Subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Code, dated July 1974.

3/4.4.4 PRESSURIZER

BASES

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

BASES

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

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 parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter 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 plant 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.

Wastage-type defects are unlikely with the All Volatile Treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 40 percent 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 percent 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 6.6 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

LEAKAGE DETECTION SYSTEMS

BASES

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."

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

BASES

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 akage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 28 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2,235 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 not isolated from the RCS 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. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.7 CHEMISTRY

BASES

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.

3/4.4.8 SPECIFIC ACTIVITY

BASES

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 ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 uCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 uCi/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

BASES

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 3.9 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. Subsequently, 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 temperture at the end of 10 EFPY.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

BASES

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, copper content and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" or the Westinghouse copper trend curves shown by Figure 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 10 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature). RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, RT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper) present in reactor vessel steels. The Westinghouse trend curves which show the effect of fluence and copper and phosphorus contents on delta RT_{NDT} for reactor vessel steels are shown in Figure B 3/4.4-2.

Given the copper content of the most limiting material, the radiation-induced delta RT_{NDT} can be estimated from Figure B 3/4.4-2. Fast-neutron fluence (E > 1 Mev) at the 1/4 T (wall thickness) and 3/4 T (wall thickness) vessel locations are given as a function of full-power service life in Figure B 3/4.4-1. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to RT_{NDT}.

The preirradiation fracture-toughness properties of the Beaver Valley Unit 2 reactor vessel materials are presented in Table B 3/4.4-1. The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup and cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code.[2] The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T-RT_{NDT} + 160)] \quad (4\text{-}1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G to the ASME Code[2] as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \quad (4\text{-}2)$$

where

K_{IM} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is a function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical.

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2 ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendix, "Rules for Construction of Nuclear Vessels," Appendix G. "Protection Against Non-ductile Failure," pp. 559-569, 1980 Edition, American Society of Mechanical Engineers, New York, 1983.

BASES

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From equation (4-2), the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady- e stat operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and insures conservative operation of the system for the entire cooldown period.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

BASES

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature.

During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

BASES

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H, 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 RT_{NDT} determined from the surveillance capsule is higher than the calculated RT_{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 for reactor criticalilty and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	CODE NO.	MATERIAL SPEC. NO.	C_4 %	P %	T_{NDT} %	50 FT/LB 35 MIL TEMP %	RT_{NDT} %	USE FT/LBS.
Closure Head Dome	B9008-1	A533B, CL.1	.13	.013	-20	50	-10	137
Closure Head Flange	B9002-1	A508 CL.7	---	.012	-10	<40	-10	136
Vessel Flange	B9001-1	A508 CL.7	---	.010	0	<10	0	132.5
Inlet Nozzle	B9011-1	A508 CL.7	---	.006	0	<10	0	104
Inlet Nozzle	B9011-2	A508 CL.7	---	.010	10	<10	10	115
Inlet Nozzle	B9011-3	A508 CL.7	---	.009	20	<40	20	122
Outlet Nozzle	B9012-1	A508 CL.7	---	.007	-10	0	-10	137
Outlet Nozzle	B9012-2	A508 CL.7	---	.006	-10	0	-10	121
Outlet Nozzle	B9012-3	A508 CL.7	---	.008	-10	0	-10	112
Nozzle Shell	B9003-1	A533B, CL.1	.13	.008	-10	110	50	91
Nozzle Shell	B9003-2	A533B, CL.1	.12	.009	0	120	60	79.5
Nozzle Shell	B9003-3	A533B, CL.1	.13	.008	-10	110	50	97.5
Inter. Shell	B9004-1	A533B, CL.1	.07	.010	0	120	60	83
Inter. Shell	B9004-2	A533B, CL.1	.07	.007	-10	100	40	75.5
Lower Shell	B9005-1	A533B, CL.1	.08	.009	-50	88	28	82
Lower Shell	B9005-2	A533B, CL.1	.07	.009	-40	93	33	77.5
Bottom Head Torus	B9010-1	A533B, CL.1	.15	.007	-30	56	-4	97
Bottom Head Dome	B9009-1	A533B, CL.1	.14	.007	-30	35	-25	116
Weld (Inter. & Lower Shell Long. Seams & Girth Seam)*			.08	.008	-30	<30	-30	144.5
HAZ (Plate B9004-2)			---	----	-80	40	-20	76

* Same heat of wire and lot of flux used in all seams including surveillance weldment.

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE
(E > 1 Mev AS A FUNCTION OF FULL POWER SERVICE LIFE)

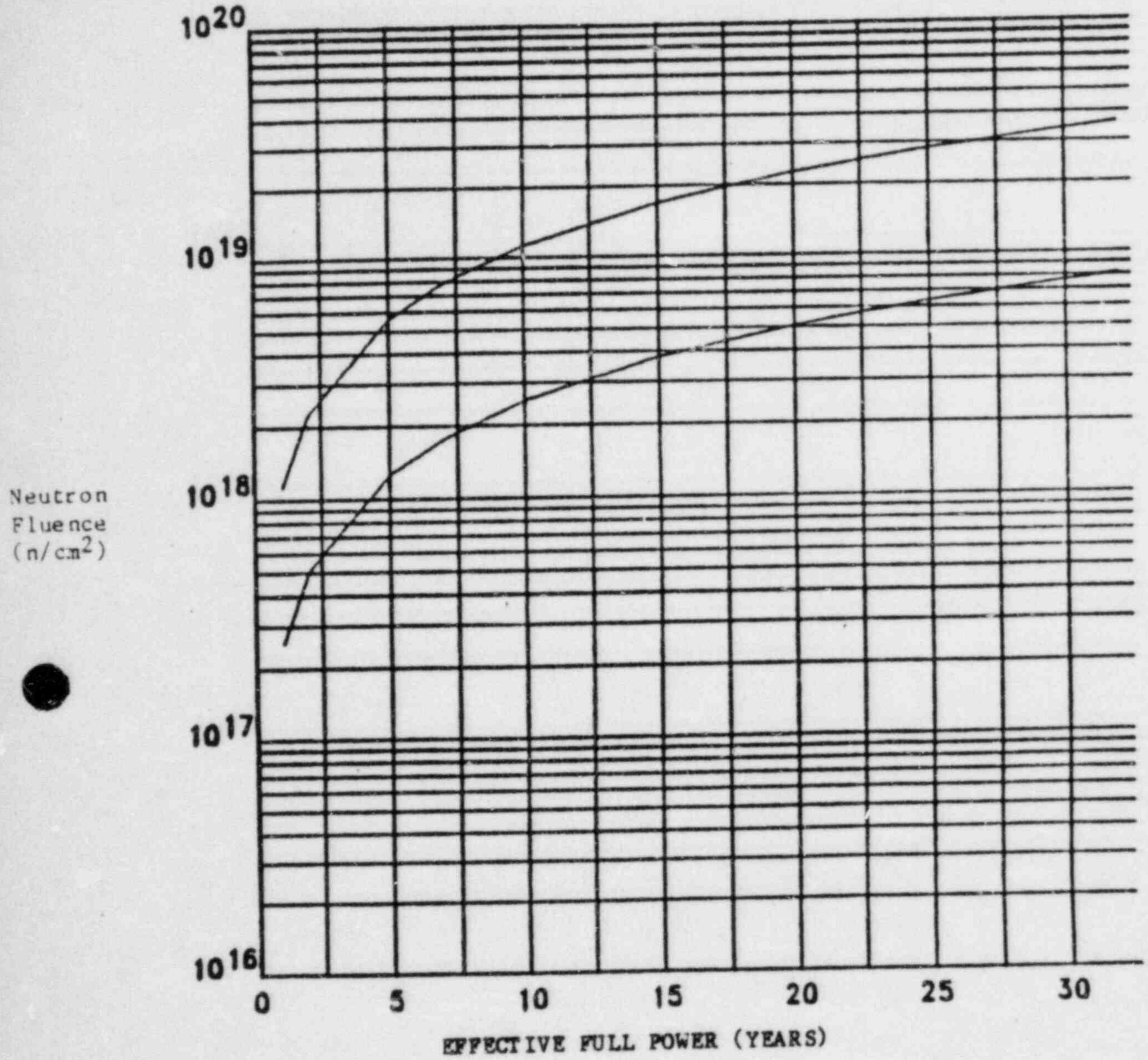
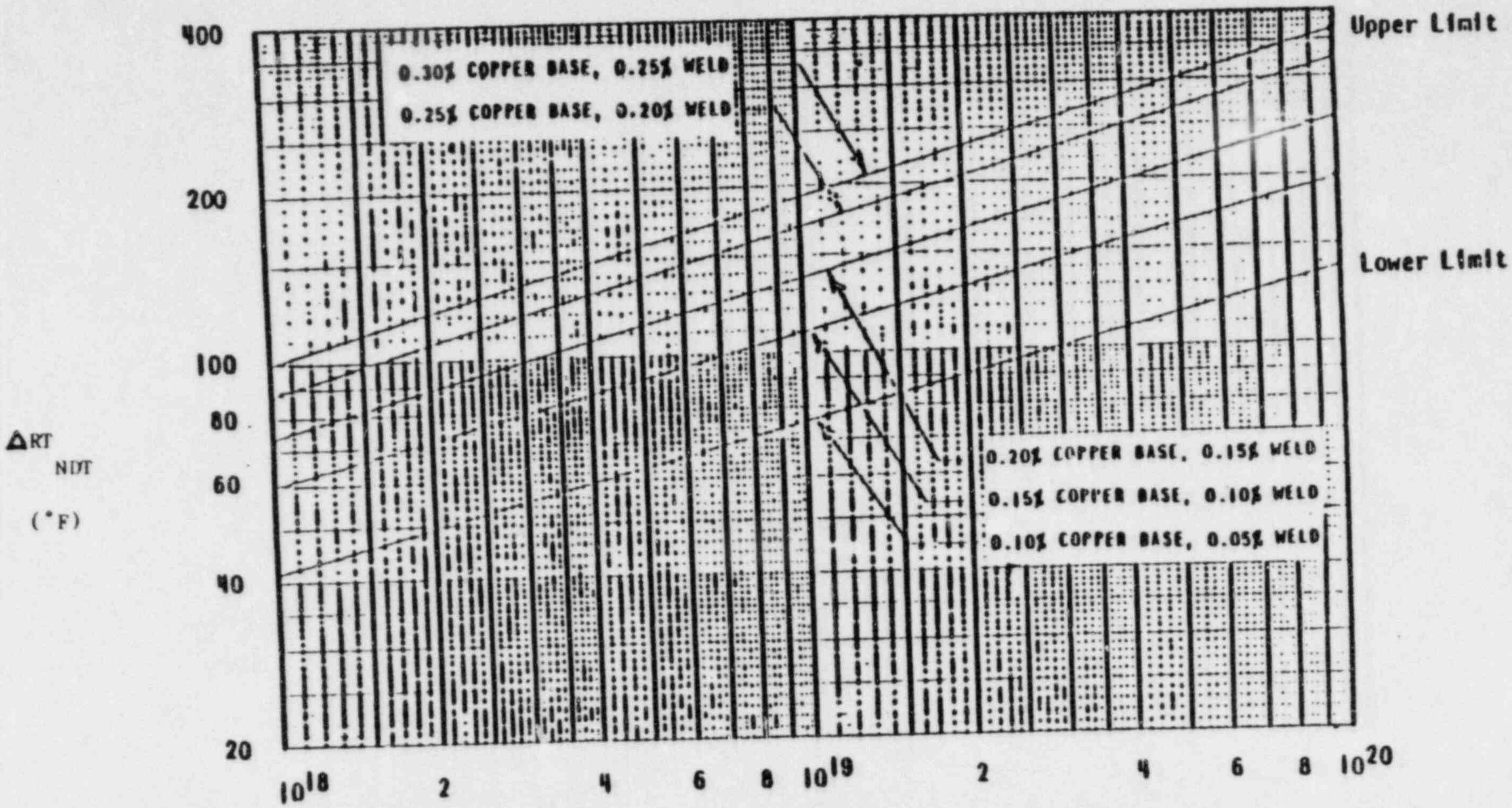


FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER ON SHIFT OF ΔRT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F



3/4.4.10 STRUCTURAL INTEGRITY

BASES

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

3/4.4.11 RELIEF VALVES

BASES

The relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

BASES

The OPERABILITY of each of the RCS accumulators 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 accident analysis are met.

The limit of one hour for operation with an inoperable accumulator 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

BASES

The OPERABILITY of two separate and 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 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.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

BASES

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 PRIMARY CONTAINMENT

CONTAINMENT LEAKAGE

BASES

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analysis at the peak accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 La 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 PRIMARY CONTAINMENT

CONTAINMENT AIR LOCKS

BASES

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment lea rate. Surveillance testing of the air lock seals provides 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 PRIMARY CONTAINMENT

INTERNAL PRESSURE/AIR TEMPERATURE

BASES

The limitations on containment internal pressure and average air temperature as a function of RWST and service water temperature ensure that:

1) The containment structure is prevented from exceeding its design negative pressure of 8 psia,

2) The containment peak pressure does not exceed the design pressure of 45 psig during LOCA conditions, and

3) The containment pressure is returned to subatmospheric conditions following a LOCA.

The containment internal pressure limits shown as a function of RWST and service water temperature describe the operational envelope that will:

1) Limit the containment peak pressure to less than its design value of 45 psig, and

2) Ensure the containment internal pressure returns subatmospheric within 60 minutes following a LOCA.

The limits on the parameters of Figure 3.6-1 are consistent with the assumptions of the accident analyses.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT STRUCTURAL INTEGRITY

BASES

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 44.6 psig in the event of a LOCA. The visual and Type A leakage tests are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEM

CONTAINMENT QUENCH SPRAY SYSTEM

CONTAINMENT RECIRCULATION SYSTEM

BASES

The OPERABILITY of the containment spray systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analysis.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CHEMICAL ADDITION SYSTEM

BASES

The OPERABILITY of the chemical addition 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.

These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

BASES

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 of a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

BASES

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 or the purge system is capable of controlling the expected hydrogen generation associated with:

1. Zirconium-water reactions,

2. Radiolytic decomposition of water,

3. Corrosion of metals within containment.

These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

BASES

The closure of the manual isolation valves in the suction of the steam jet air ejector ensures that:

1. the containment internal pressure may be maintained within its operation limits by the mechanical vacuum pumps, and

2. the containment atmosphere is isolated from the outside environment in the event of a LOCA.

These valves are required to be closed for containment isolation.

3/4.7 TURBINE CYCLE

3/4.7.1 TURBINE CYCLE SAFETY VALVES

BASES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 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 percent 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 and Winter 1972 Addenda. The total relieving capacity for all valves on all of the steam lines is 110 percent of the total secondary steam flow at 100 percent 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 3 loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 2 loop operation:

$$SP = \frac{(X) - (Y)(U)}{X} \times (70)$$

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

3/4.7.1 TURBINE CYCLE SAFETY VALVES

BASES

(109) = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation

(70) = maximum percent of RATED THERMAL POWER permissible by P-8 setpoint for 2 loop operation with stop valves closed

X = Total relieving capacity of all safety valves per steam line in pounds/hour

Y = Maximum relieving capacity of one safety valve in pounds/hour

3/4.7.1 TURBINE CYCLE

AUXILIARY FEEDWATER SYSTEM

BASES

The OPERABILITY of the auxiliary feedwater pumps 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 350 gpm at a pressure of 1133 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1133 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 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

PRIMARY PLANT DEMINERALIZED WATER (PPDW)

BASES

The OPERABILITY of the PPDW storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 350°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to atmosphere.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

ACTIVITY

BASES

The limitations on the secondary coolant system specific activity ensure that the resultant offsite 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 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

MAIN STEAM LINE ISOLATION VALVES

BASES

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 the containment in the event the steam line rupture occurs within the 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.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

BASES

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 steam generator average impact values taken at 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

BASES

The OPERABILITY of the primary 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 analysis.

3/4.7.4 SERVICE WATER SYSTEMS

BASES

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.

3/4.7.5 ULTIMATE HEAT SINK - OHIO RIVER

BASES

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either:

1) Provide normal cooldown of the facility, or

2) To mitigate the effects or accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

3/4.7.6 FLOOD PROTECTION

BASES

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 feet Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

BASES

The OPERABILITY of the control room ventilation 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 operations.

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 Criteria 19 of Appendix "A", 10 CFR 50.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

BASES

The OPERABILITY of the SLCRS ensures that potential radioactive materials leaking from equipment are processed and filtered before release to the atmosphere at an elevated point. The operation of this system and the resultant effect on offsite dosage calculations was not assumed in the accident analyses.

3/4.7.9 SEALED SOURCE CONTAMINATION

BASES

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3/4.7.12 SNUBBERS

BASES

All 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 similar event initiating dynamic loads. 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 visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed, may be used as a new reference point to determine the next inspection.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling or 18 month intervals not to exceed two years. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

3/4.7.12 SNUBBERS

BASES

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SSWS)

BASES

The OPERABILITY of the SSWS ensures that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that a barge explosion at the station's intake structure or any other extremely remote event would render all of the normal Service Water System supply pumps inoperable.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

BASES

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression systems consist of the water system, sprinklers, CO_2 Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire supports.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight of the tanks. The Halon systems are indoor systems not susceptible to outdoor weather conditions. The systems are dry pipe (rust is not expected) gas suppression systems.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24 hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.15 FIRE RATED ASSEMBLIES

BASES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. The design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays, and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their operability.

3/4.7.16 TERRESTRIAL ECOLOGICAL SURVEY

BASES

The objective of the Terrestrial Ecological Survey is to determine the effects of cooling tower drift on terrestrial biota. The Terrestrial Ecological Survey program element involves long term monitoring.

Cooling tower drift could lead to ecological effects that would appear as vegetation stresses on infrared aerial photographs.

3/4.8.1, 3/4.8.2 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

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 Criterion 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 safety analyses and are based upon maintaining at least one redundant set of 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 unit status.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Standard 450-1980, "IEEE Recommended Practice for maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 3.8-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 3.8-1 is permitted for up to 7 days. During this 7 day period:

1. The allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability;

2. The allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing;

3. The allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and

4. The allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

BASES

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 uncertanties, is sufficient to prevent reactor criticality during refueling operations.

3/4.9.2 INSTRUMENTATION

BASES

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

BASES

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor 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

BASES

The requirements on containment penetration closure and OPERABILITY of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere within 10CFR100 limits. 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

BASES

The requirements 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

BASES

The OPERABILITY requirements for the manipulator crane and auxiliary hoist ensure that:

1. Manipulator crane and auxiliary hoist will be used for the movement of fuel assemblies and drive rod assemblies,

2. The crane has sufficient load capacity to lift a fuel assembly and the hoist has sufficient capacity to lift a drive rod assembly,

3. The core internals and pressure vessel are protected from excessive lifting force in the event they are engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

BASES

The restriction on movement of loads in excess of the normal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analysis.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

BASES

The requirement that at least one residual heat removal (RHR) 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 throughout the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two (RHR) loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

BASES

The OPERABILITY of this system insures that the containment vent and purge penetrations will be automatically isolated upon detection of high-high radiation levels within the containment. The integrity of the containment penetrations of this system is required to restrict the release of radioactive material from the containment atmosphere to accessible levels which are less than those listed in 10 CFR 100. Applicability in MODE 5, although not an NRC requirement, will provide additional protection against small releases of radioactive material from the containment during maintenance activities.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

BASES

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99 percent of the assumed 10 percent 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 BUILDING VENTILATION SYSTEM - FUEL MOVEMENT

BASES

The limitations on the storage pool ventilation system ensures 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 analysis.

3/4.9.13 FUEL BUILDING VENTILATION SYSTEM - FUEL STORAGE

BASES

The spent fuel pool area ventilation system is non-safety related and only recirculates air throughout the fuel building. The SLCRS portion of the ventilation filters the fuel building air upon receipt of a high-high radiation signal.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGINS

BASES

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 occuring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

BASES

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 PRESSURE/TEMPERATURE LIMITATIONS - REACTOR CRITICALITY

BASES

This special test exception permits the reactor to be critical at less than or equal to 5 percent of RATED THERMAL POWER during low temperature PHYSICS TESTS required to measure such parameters as control rod worth and SHUTDOWN MARGIN.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.4 PHYSICS TESTS

BASES

This special test exception permits the physics tests to be performed at less than or equal to 5 percent of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation.

3/4.10.5 NO FLOW TESTS

BASES

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.

3/4.10.6 POSITION INDICATION SYSTEM - SHUTDOWN

BASES

This special test exception permits the position indication systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time is derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the position indication systems remain OPERABLE.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

BASES

This Specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluent to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR 20 Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within:

1. The Section II A design objectives of Appendix I, 10 CFR 50 to a individual, and

2. The limits of 10 CFR 20.106e to the population.

The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe^{135} is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

DOSE

BASES

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I to 10 CFR 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The Action statements provides the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in liquid effluents will be kept "as low as reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operation, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the dose due to the actual release rates of radioactive materials in liquid effluent are consistent with the methodolog provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of g, Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I, "Revision 1, October, 1977, and Regulatory Guide 1.113 "Estimatin Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", Revision 1, April, 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

This Specification applies to the release of liquid effluents from Beaver Valley Power Station, Unit No. 2. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1 LIQUID EFFLUENTS

LIQUID WASTE TREATMENT

BASES

The requirements that are the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This Specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and design objective given in Section II.D of Appendix I to 10 CFR 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendiox I, 10 CFR 50 , for liquid effluents. This Specification applies to Beaver Valley Power Station - Unit 2.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

BASES

Restricting the quantity of radioactive material contained in the specified tank provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

DOSE RATE

BASES

This specification is provided to ensure that the dose at anytime at the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II, of 10 CFR 20 (10 CFR 20.106(b). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to $\leq$ 500 mrem/year to the total body or to $\leq$ 3,000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to $\leq$ 1,500 mrem/year for the nearest cow to the plant.

This specification applies to the release of gaseous effluents from Beaver Valley Power Station, Unit No. 2. For units with shared radwaste treatment system, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2 GASEOUS EFFLUENTS

DOSE, NOBLE GASES

BASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially under-estimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided 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 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled reactors," Revision 1, July, 1977. The ODCM equations provided for determining the air doses at the exclusion area boundary are based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111. This specification applies to the release of gaseous effluents from Beaver Valley Power Station, Unit 2.

3/4.11.2 GASEOUS EFFLUENTS

DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

BASES

This specification is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I to 10 CFR 50. The Limiting Condition for Operation are the guides set forth in Section II.C of Appendix I.

The Action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Rector Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form, and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which are examined in the development of these calculations are:

1. Individual inhalation of airborne radionuclides,
2. Deposition of radionuclides onto green leafy vegetation with subsequent consumption by man,
3. Deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and
4. Deposition on the ground with subsequent exposure of man. This specification applies to radioactive material in particulate form and radionuclides other than noble gases released from Beaver Valley Power Station, Unit 2.

3/4.11.2 GASEOUS EFFLUENTS

GASEOUS RADWASTE TREATMENT

BASES

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonable achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and design objective Section II.D of Appendix I to 10 CFR 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR 50, for gaseous effluents. This specification applies to gaseous radwaste from Beaver Valley Power Station, Unit 2.

3/4.11.2 GASEOUS EFFLUENTS

GASEOUS WASTE STORAGE TANKS

BASES

Restricting the quantity of radioactivity contained in each gaseous waste storage tank provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting total body exposure to an individual located at the nearest exclusion area boundary for two hours immediately following the onset of the release will not exceed 0.5 rem. The specified limit restricting the quantity of radioactivity contained in each gaseous waste storage tank was specified to ensure that the total body exposure resulting from the postulated release remained a suitable fraction of the reference value set forth in 10 CFR 100.11(a)(1).

3/4.11.2 GASEOUS EFFLUENTS

BASES

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Isolation of the affected tank for purposes of purging and/or discharge permits the flammable gas concentrations of the tank to be reduced below the lower explosive limit in a hydrogen rich system. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR 50.

3/4.11.3 SOLID RADIOACTIVE WASTE

BASES

This specification implements the requirements of 10 CFR 50.36a and General Design Criteria 60 of Appendix A of 10 CFR 50 and requires the system be used whenever solid radwastes require processing and packaging prior to being shipped offsite. The process parameters used in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

BASES

This Specification is provided to meet the dose limitations of 40 CFR 190. The Specification requires the preparation and submittal of a Special Report, in lieu of any other report, whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 nuclear reactors, it is highly unlikely that the resultant dose to MEMBER(S) OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to MEMBER(S) OF THE PUBLIC for the calendar year to within the 40 CFR 190 limits. For the purpose of the Special Report, it may be assumed that the dose commitment to MEMBER(S) OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

3/4.12.1 MONITORING PROGRAM

BASES

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways for those radionuclides which lead to the highest potential radiation exposures of MEMBER(S) OF THE PUBLIC resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operation experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

3/4.12.2 LAND USE CENSUS

BASES

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring programs are made if required by the results of this census. The best survey information from the door-to-door survey,* aerial survey, or by consultig with local agriculture authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used:

a. That 20 percent of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and

b. A vegetation yield of 2 kg/square meter.

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* Confirmation by telephone is equivalent to door-to-door.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

BASES

The requirement for participation in an Interlaboratory Comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed a part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE:

SITE BOUNDARY FOR GASEOUS EFFLUENTS:

5.1.1

The site boundary for gaseous effluents shall be as shown in Figure 5.1-1. Release points are shown on Figure 5.1-2.

SITE BOUNDARY FOR LIQUID EFFLUENTS:

5.1.2

The site boundary for liquid effluents shall be as shown in Figure 5.1-1. Release points are shown on Figure 5.1-2.

EXCLUSION AREA:

5.1.3

The exclusion area shall be as shown in Figure 5.1-3.

LOW POPULATION ZONE:

5.1.4

The low population zone shall be as shown in Figure 5.1-4.

FLOOD CONTROL:

5.1.5

The flood control provisions (dikes, levees, etc.) shall be designed and maintained in accordance with the original design provisions contained in Section 3.4.1 of the FSAR.

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 = 126 feet.

b. Nominal inside height = 185 feet.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of foundation mat = 10 feet.

f. Nominal thickness of vertical portion of steel liner = 3/8 inch.

g. Nominal thickness of steel liner dome portion = 1/2 inch.

h. Net free volume = 1.73×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE:

5.2.2

The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 369°F.

PENETRATIONS:

5.2.3

Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE:

FUEL ASSEMBLIES:

5.3.1

The reactor core shall contain 157 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 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.3 weight percent U-235.

CONTROL ROD ASSEMBLIES:

5.3.2

The reactor core shall contain 48 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be hafnium. All control rods shall be clad with stainless steel tubing.

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 5.2 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 9370 ± 100 cubic feet at a nominal T_{avg} of 576°F.

5.5 EMERGENCY CORE COOLING SYSTEMS:

The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY:

5.6.1

The spent fuel storage racks are designed and shall be maintained with a minimum of 10.4375 inch center-to-center distance between fuel assemblies placed in the storage racks. With spent fuel of a maximum enrichment of 3.6% by weight UO_2, the fuel pool filled with pure water at 32°F, the fuel stored in the worst feasible geometric configuration, and with the worst case seismic deflection, K_{eff} will be less than .95.

DRAINAGE:

5.6.2

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 742'-1".

CAPACITY:

5.6.3

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1,088 fuel elements.

5.7 SEISMIC CLASSIFICATION

5.7.1

Those structures, systems and components identified as Category I items in Section 3.7 of the FSAR shall be designed and maintained to the original design provisions with allowance for normal degradation pursuant to the applicable surveillance requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1

The meteorological tower shall be located as shown on Figure 5.1-1.

FIGURE 5.1-1

SITE BOUNDARY FOR GASEOUS EFFLUENTS AND LIQUID FOR THE BEAVER VALLEY POWER STATION

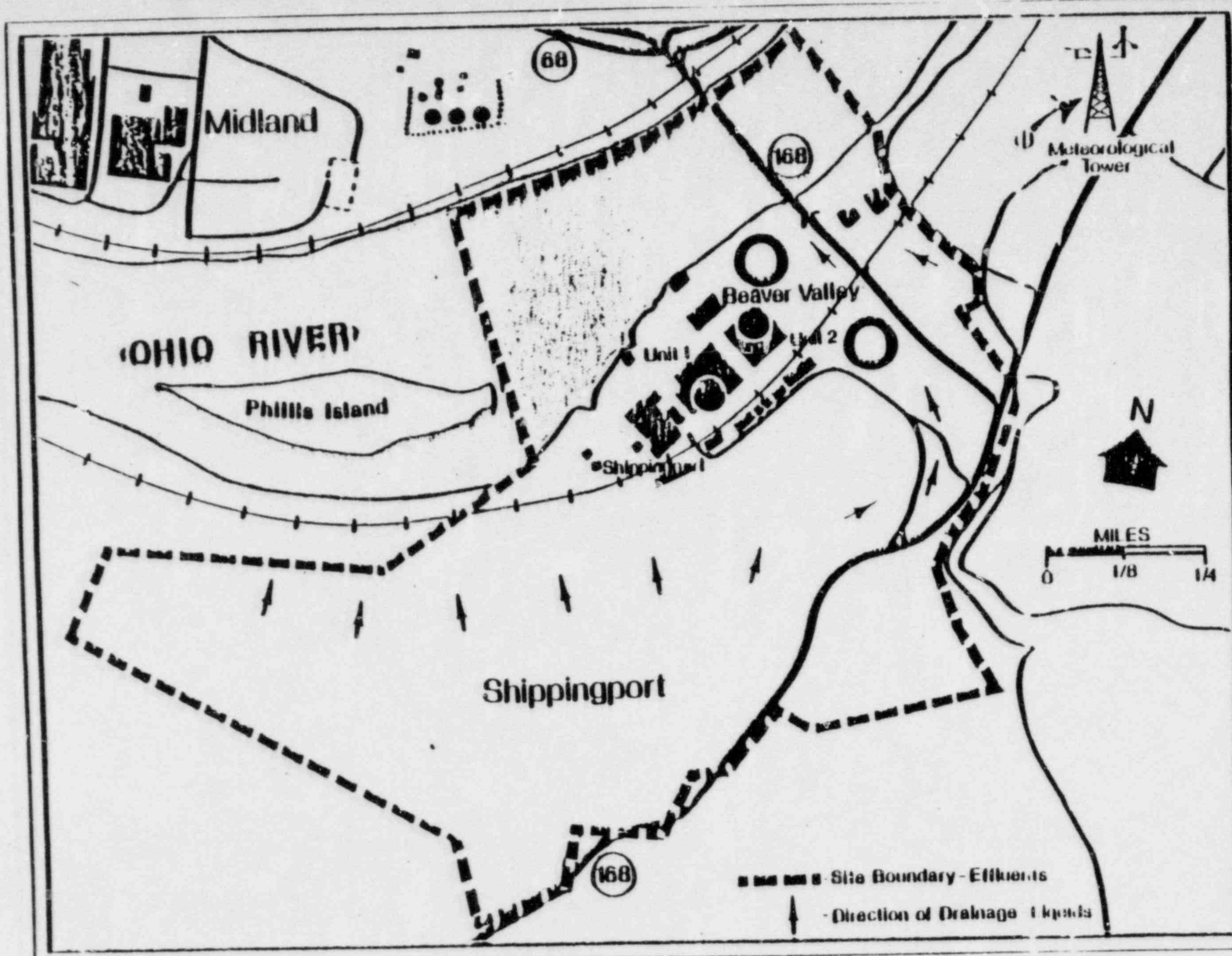


FIGURE 5.1-2

RADIOACTIVE LIQUID AND GASEOUS RELEASE POINTS

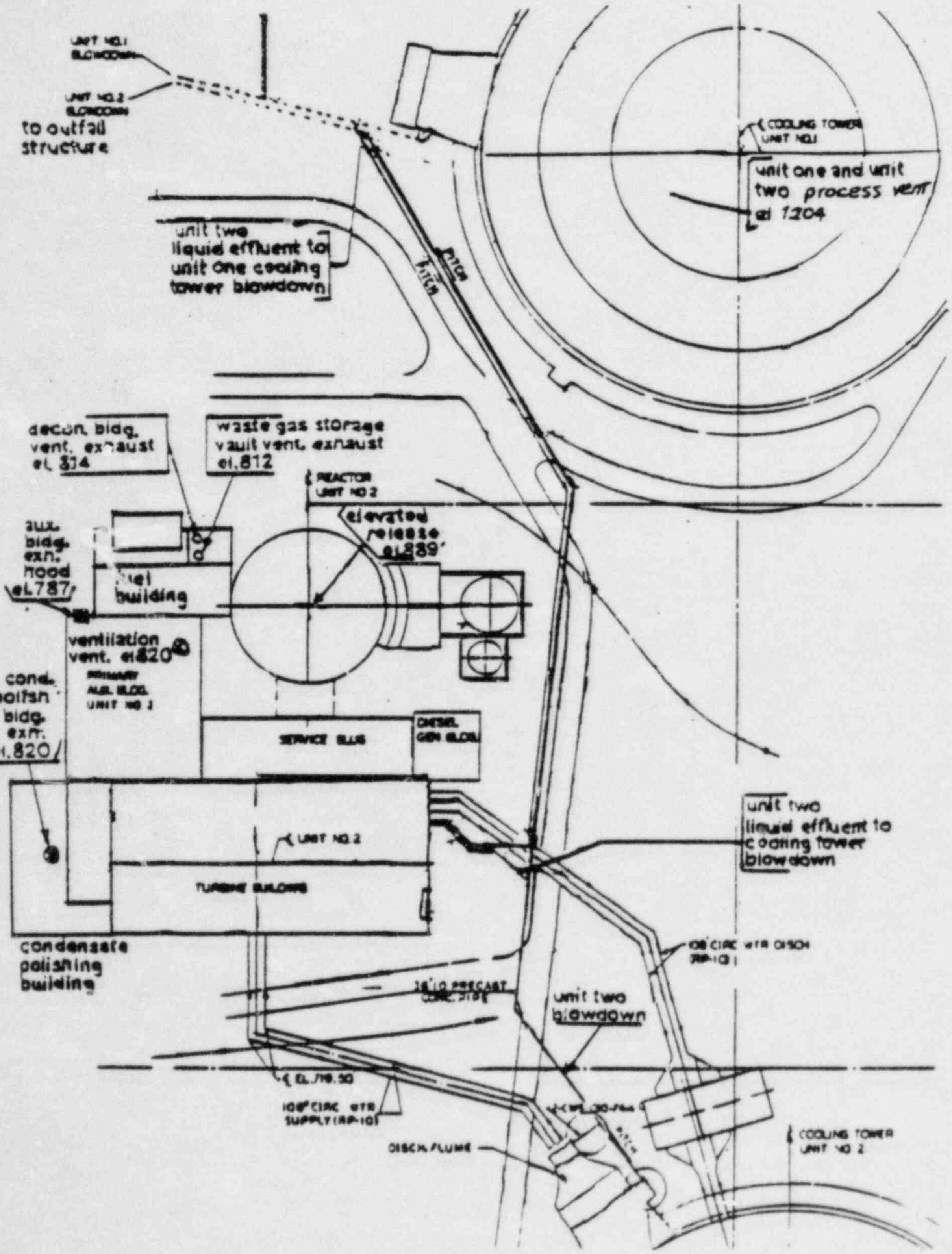


FIGURE 5.1-3

EXCLUSION AREA BOUNDARY - BEAVER VALLEY POWER STATION UNIT NO. 2

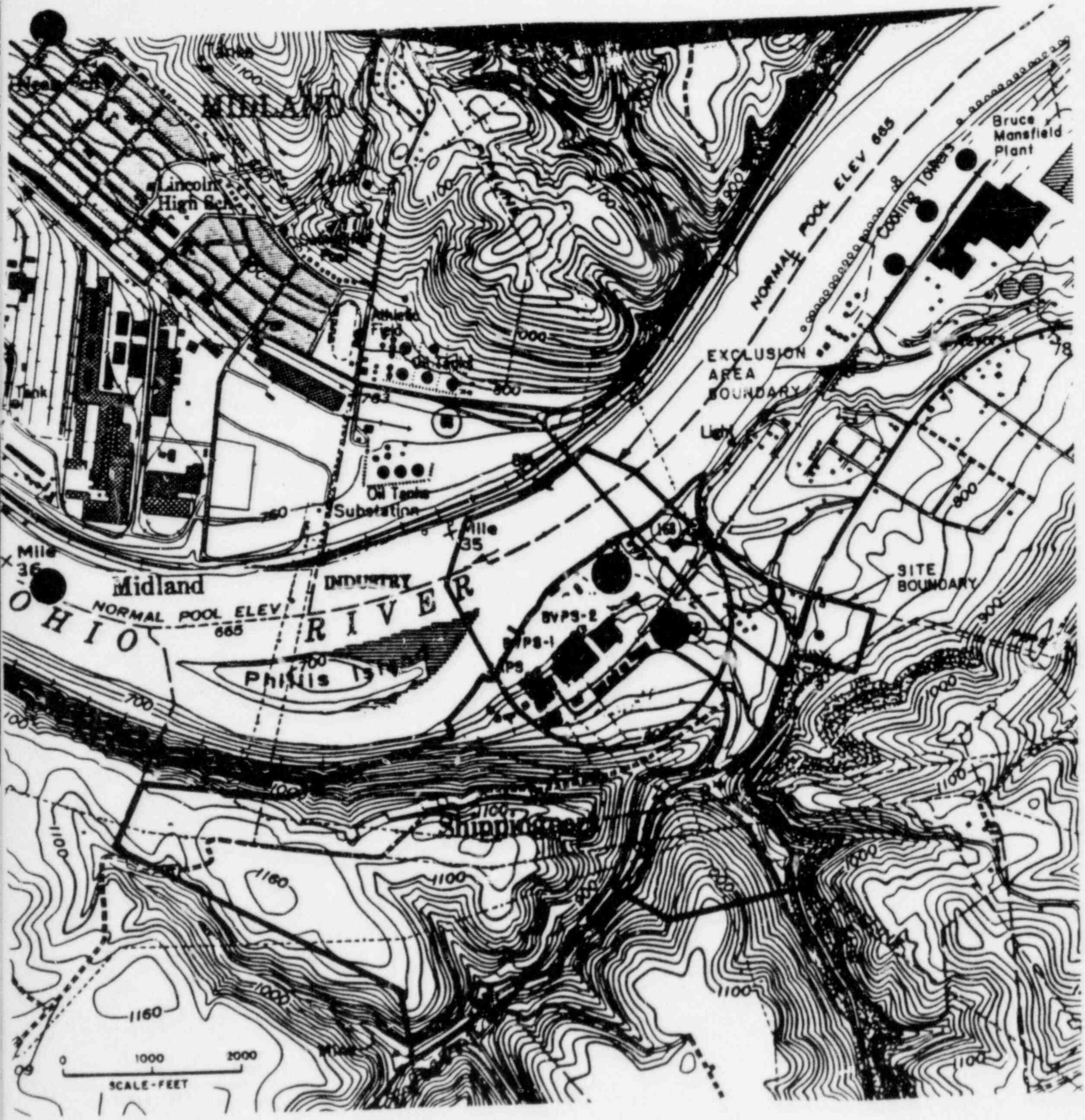
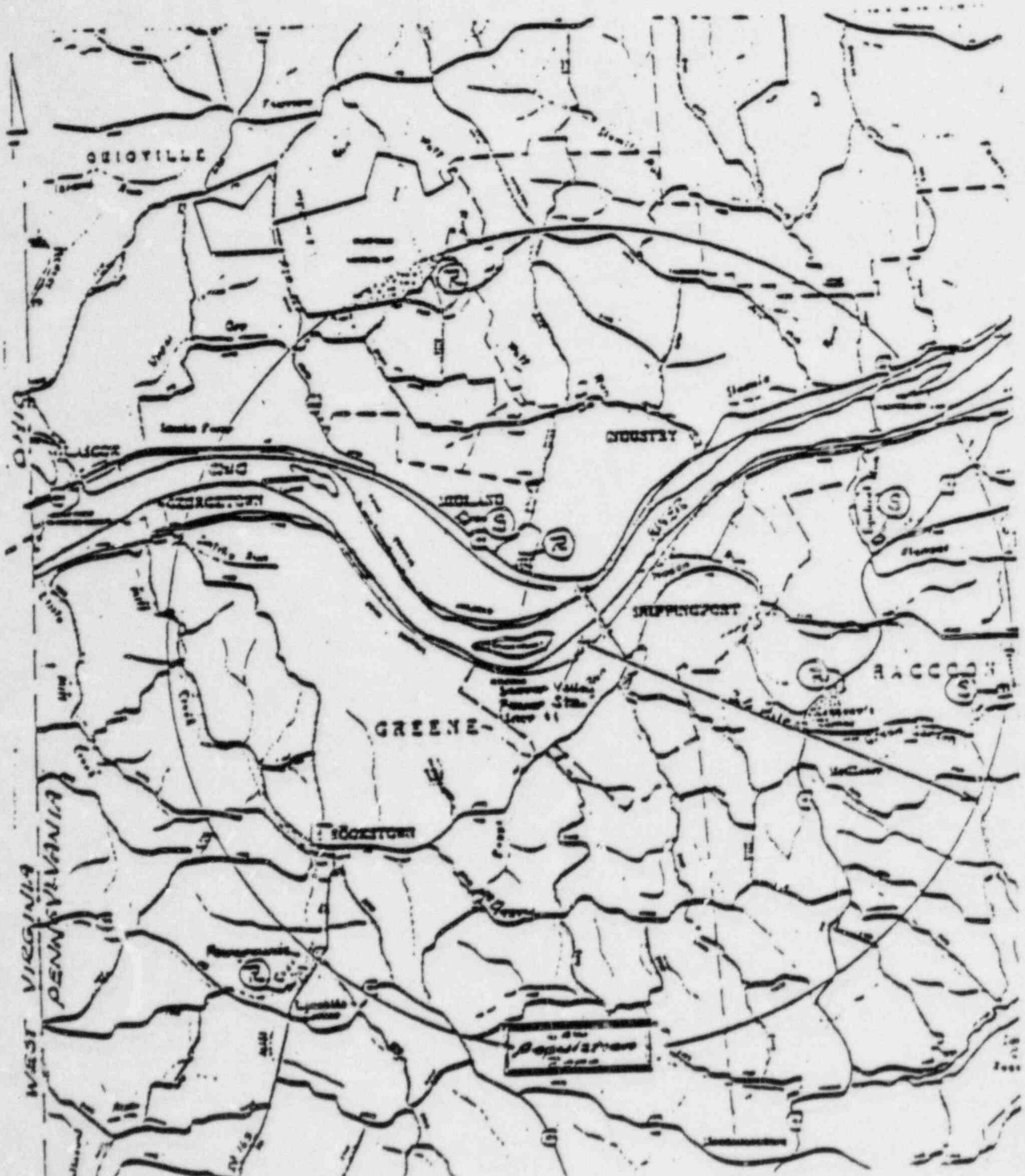


FIGURE 5.1-4

LOW POPULATION ZONE - BEAVER VALLEY POWER STATION UNIT NO. 2



SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Superintendent of Operations shall be responsible for overall unit 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 unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF:

6.2.2 The Unit(s) organization shall be as shown in 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 for each unit 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 startup, 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.

f. A Fire Brigade of at least 5 members shall be maintained on site at all times. The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.

g. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; senior reactor operators, reactor operators, radiation control technicians, auxiliary operators, meter and control repairmen, and all personnel actually performing work on safety related equipment.

The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Superintendent or predesignated alternate, or higher levels of management. Authorized deviations to the working hour guidelines shall be documented and available for NRC review.

FIGURE 6.2-1

(later)

FIGURE 6.2-2

(later)

TABLE 6.2-1

MINIMUM BVPS UNIT 1 and UNIT 2 SHIFT CREW COMPOSITION[a]

License Category Qualification	Applicable Modes[c] Both Units 1, 2, 3, 4	One Unit 1, 2, 3, 4 / One Unit 5, 6	Both Units 5, 6
Senior Reactor Operator (SRO)[b][e]	2[f]	2[d][f]	1[d][h]
Reactor Operator (RO)	3[g]	3[g]	2
Nonlicensed Auxiliary Operator	3[g]	3[g]	3[g]
Shift Technical Advisor (STA)	1[h][i]	1[h][i]	0
Individual Qualified in Radiation Protection Procedures	1[h]	1[h]	1[h]
Rad/Chem Technician	1[h]	1[h]	1[h]

NOTES:

[a] Except for the Shift Supervisor, the shift crew composition may be one 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. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

[b] Includes the licensed SRO serving as the Shift Supervisor.

[c] Operational Mode Definitions:

MODE 1 - Power Operation
MODE 2 - Start-up
MODE 3 - Hot Standby
MODE 4 - Hot Shutdown
MODE 5 - Cold Shutdown
MODE 6 - Refueling

[d] Does not include the SRO assigned during MODE 6 to directly supervise core operations.

[e] During any absence of the Shift Supervisor from the Control Room while the unit is in Mode 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control

Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in Mode 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

(f) Minimum of 2 individuals for each unit; each individual may fill the same position on both units if qualified on both units.

(g) Minimum of 2 individuals for each unit; one of two individuals may fill the same position on both units if qualified on both units.

(h) Minimum of 1 individual for each unit; one individual may fill the same position on both units if qualified on both units.

(i) One of two required individuals filling the SRO positions may also fill the STA position, if qualified.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility and Radiation Protection staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiological Operations Coordinator who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Director Nuclear Division Training 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.

6.4.2 A Training Program for the Emergency Squad shall be maintained under the direction of the Director Nuclear Division Training and shall meet or exceed the requirements of Section 27 of the NEPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 Onsite Safety Committee (OSC)

FUNCTION:

6.5.1.1 The OSC shall function to advise the Station Superintendent on all matters related to nuclear safety.

COMPOSITION:

6.5.1.2 The OSC shall be composed of the:

Chairman:	Chief Engineer
Member:	Senior Licensed Operator
Member:	Radiation Control Foreman
Member:	Maintenance Engineer
Member:	Project Engineer - Nuclear Engineering Department
Member:	Senior Testing or Study Projects Coordinator
Member:	Shift Technical Advisor
Member:	Chemist
Member:	Quality Control Engineer
Member:	I&C Supervisor

NOTE:

The chairman of the OSC shall appoint an individual from each of the above listed job categories to serve as a member of the OSC for a period of at least 6 months.

NOTE:

OSC members shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The SRO shall meet the qualifications of Section 4.2.2 and the Maintenance Engineer will meet the qualifications of Section 4.2.3.

ALTERNATES:

6.5.1.3 All alternate members shall be appointed in writing by the OSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSC activities at any one time.

MEETING FREQUENCY:

6.5.1.4 The OSC shall meet at least once per calendar month and as convened by the OSC Chairman or his designated alternate.

QUORUM:

6.5.1.5 A quorum of the OSC shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES:

6.5.1.6 The OSC 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 Plant Superintendent 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" Tecnnical 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 Manager of Nuclear Operations and to the Chairman of the Offsite Review Committee.

f. Review all REPORTABLE EVENTS.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Offsite Review Committee.

AUTHORITY:

6.5.1.7 The OSC shall:

a. Recommend to the Plant Superintendent 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 Manager of Nuclear Operations and the Offsite Review Committee of disagreement between the OSC and the Plant Superintendent; however, the Plant Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS:

6.5.1.8 The OSC shall maintain written minutes of each meeting and copies shall be provided to the Manager of Nuclear Operations and Chairman of the Offsite Review Committee.

6.5.2 Offsite Review Committee (ORC)

FUNCTION:

6.5.2.1 The ORC 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 ORC shall be composed of the:

Chairman:	Vice President, Nuclear Division
Vice Chairman:	Manager, Nuclear Safety and Licensing
Member:	Manager, Nuclear Engineering
Member:	Manager, Nuclear Operations
Member:	Manager, Regulatory Affairs, Beaver Valley Power Station Unit No. 2
Member:	Senior Project Engineer, Nuclear Engineering Department
Member:	Manager, Nuclear Support Services
Member:	Site Service Manager, Westinghouse Electric Corporation
Member:	Manager, Quality Assurance
Member:	Director, Environmental and Radiological Safety Programs
Member:	Outside Consultant, Chemistry and Radiochemistry

ALTERNATES:

6.5.2.3 All alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ORC activities at any one time.

CONSULTANTS:

6.5.2.4 Consultants shall be utilized as determined by the ORC Chairman to provide expert advise to the ORC.

MEETING FREQUENCY:

6.5.2.5 The ORC 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 ORC shall consist of the Chairman or his designated alternate and at least 4 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 ORC shall review:

a. The safety evaluation 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 in Technical Specifications or licenses.

e. Violations of applicable statutes, 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. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structure systems, or components.

i. Reports and meetings minutes of the OSC.

j. The results of the environmental monitoring program prior to submittal of the Annual Environmental Operating Report.

AUDITS:

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the ORC. 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 methods of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the 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 12 months.

f. The Facility Security Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the ORC or the Vice President, Nuclear.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUTHORITY:

6.5.2.9 The ORC shall report to and advise the Vice President, Nuclear Division on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

RECORDS:

6.5.2.10 Records of ORC activities shall be prepared, approved and distributed as indicated by the following:

a. Minutes of each ORC meeting shall be prepared for and approved by the ORC Chairman 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 ORC Chairman 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, Nuclear Division and to the management positions responsible for the areas audited within 30 days after completion of the audit.

d. The Vice President, Nuclear Division shall review all recommendations of the ORC.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified in accordance with 10CFR50.72 and/or a report submitted pursuant to the requirements of Section 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the OSC, and the results of this review shall be submitted to the ORC and the Vice President, Nuclear Group.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Superintendent of Operations and the the ORC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the On-Site Safety Committee (OSC). 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 ORC and the Manager of Nuclear Operations 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.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the OSC and approved by the Plant Superintendent, predesignated alternate or a predesignated Department Manager to whom the Plant Superinten .nt has assigned in writing the responsibility for review and approval of specific subjects considered by the committee, as applicable.

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 (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the OSC and approved by the Plant Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES:

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORTS:

6.9.1.1 A summary report of plant startup and power escalation testing will be submitted following 1) receipt of an operating license, 2) amendment to the license involving a planned increase in power level, 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and 4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within 1) 90 days following completion of the startup test program, 2) 90 days following resumption or commencement of commercial power operation, or 3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

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1 A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions[2] (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. Documentation of all challenges to the pressurizer power operated relief valves (PORV's) or pressurizer safety valves.

c. A report on the non-radiological environmental surveillance programs (Alternating years only). 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 shall be provided. 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.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, submitted no later than the 15th of each month following the calendar month covered by the report.

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2 This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

6.9.1.7 This item intentionally blank.

6.9.1.8 This item intentionally blank.

6.9.1.9 This item intentionally blank.

ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT[3]:

6.9.1.10 Routire radiological environmental operating reports covering the operation of the unit during the previous calendai year shall be submitted prior to May 1 of each year and will include reporting any deviations not reported under 6.9.2 with respect to the Radiological Effluent Technical Specifications.

3 A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to both units.

6.9.1.11 The annual radiological environmental 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. The reports shall also include the results of the land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, 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.

TABLE 6.9-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility ______________________ Docket No. ______________

Location of Facility ______________________ Reporting Period ______________
(County, State)

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limits of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range[b]	Locations with Highest Annual Mean: Name Distance and Direction	Locations with Highest Annual Mean: Mean (f)[b] Range[b]	Control Locations Mean (f)[b] Range[b]	Number of REPORTABLE OCCURRENCES

[a] Nominal Lower Limit of Detection (LLD) as defined in table notation c. of Table 4.12-1 of Specification 4.12.1.1.

[b] Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses (f).

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program required by Specification 3.12.3.

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[4]:

6.9.1.12 Routine radioactive effluent release reports covering the operating of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.1.13 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, Revision 1, June, 1974, "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," with date summarized on a quarterly basis following the format of Appendix B thereof.

In addition, the radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed; wind direction, atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This report shall also include an assessment of the radiation doses from radioactive effluents to MEMBER(S) OF THE PUBLIC due to their activities inside the site boundary (Figure 5.1-1 and 5.1-2) during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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4 A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real individual from reactor releases for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1. The SKYSHINE Code (available from Radiation Shielding Information Center, ORNL) is acceptable for calculating the dose contribution from direct radiation due to N-16.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with the ODCM.

The radioactive effluent release reports shall also include any licensee initiated changes to the ODCM made during the 6 month period.

RADIAL PEAKING FACTOR LIMIT REPORT:

6.9.1.14 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Director of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention Chief of the Core Performance Branch, United States Nuclear Regulatory Commission, Washington, DC 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

BI-ANNUAL ENVIRONMENTAL OPERATING REPORT:

6.9.1.15 A report on the terrestrial environmental surveillance program for the previous 24 months of operation shall be submitted to the Regional Director of the Office of Inspection and Enforcement (with copy to Director of Nuclear Reactor Regulation) 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 nonradiological and environmental surveillance activities (Section 3.7.16) for the report period.

A comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the [illegible]erved impacts of the plant operation on the environment shall be provided. 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.

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 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.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Sealed source leakage in excess of limits, Specification 4.7.9.1.3.

f. Fire Detection Instrumentation, Specification 3.3.3.6.

g. Fire Suppression Systems, Specifications 3.7.14.1, 3.7.14.2, 3.7.14.3 and 3.7.14.5.

h. Miscellaneous reporting requirements specified in the Action Statements for Radiological Effluent Technical Specifications.

i. Failure of Pressurizer PORVs, Specification 3.4.11.

j. Failure of Pressurizer Safety Valves, Specifications 3.4.2 and 3.4.3.

k. RCS Specific Activity, Specification 3.4.8.

l. Containment Inspection Report, Specification 4.6.1.6.2.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five (5) 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 EVENTS.

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 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 designed for a limited number of transients or cycles.

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 OSC and the ORC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lines of all hydraulic and mechanical snubbers listed on Table 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

n. Records of analyses required by the Radiological Environmental Monitoring Program.

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 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hour, but less than 1000 mrem/hour, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiological Work Permit* or Radiological Access Control Permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by a facility health physics supervisor in the Radiological Work Permit or Radiological Access Control Permit.

6.12.2 The requirements of 6.12.1, above, also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hour. 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 Shift Supervisor on duty and/or a facility health physics supervisor.

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* Health physics personnel, or personnel escorted by health physics personnel in accordance with approved emergency procedures, shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.13 PROCESS CONTROL PROGRAM (PCP)

FUNCTION:

6.13.1 The PCP shall be a manual containing the processing steps and a set of established process parameters detailing the program of sampling, analysis, and evaluation within which solidification of radioactive wastes is assured, consistent with Specification 3.11.3.1 and the surveillance requirements of these Technical Specifications.

LICENSEE INITIATED CHANGES:

6.13.2 Shall become effective upon review and acceptance by the OSC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

FUNCTION:

6.14.1 The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

LICENSEE INITIATED CHANGES:

6.14.2 Shall become effective upon review and acceptance by the OSC.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (LIQUID, GASEOUS, AND SOLID)

FUNCTION:

6.15.1 The radioactive waste treatment systems (liquid, gaseous and solid) are those systems described in the facility Final Safety Analysis Report or Hazards Summary Report, and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the LCO's set forth in Specifications 3.11.1.1, 3.11.1.2, 3.11.1.3, 3.11.1.4, 3.11.2.1, 3.11.2.2, 3.11.2.3, 3.11.2.4., 3.11.2.5, 3.11.2.6, 3.11.3.1, and 3.11.4.1.

6.15.2 Major changes as defined in Section 1 to the radioactive waste systems (liquid, gaseous and solid) shall be made by the following method:

a. Licensee initiated changes:

1. If a permanent facility change is made to a radioactive treatment system that could result in an increase in the volume or activity discharged, the Commission shall be informed by the inclusion of a suitable discussion of each change in the Annual 10 CFR 50.59 Report for the period in which the changes were made. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made (in accordance with 10 CFR 50.59);

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change will be submitted which shows the predicted increase of releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected increase in the maximum exposures to an individual in the unrestricted area from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted increase of releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period the changes were made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the OSC.

2. The change shall become effective upon review and acceptance by the OSC.

6.15.3 Background of what constitutes "major changes" to radioactive waste systems (liquid, gaseous, and solid).

a. Background

1. 10 CFR 50, Section 50.34a(a) requires that each application to construct a nuclear power reactor provide a description of the equipment installed to maintain control over radioactive material in gaseous and liquid effluents produced during normal reactor operations including operational occurrences.

2. 10 CFR 50, Section 50.34a(b)(2) requires that each application to construct a nuclear power reactor provide an estimate of the quantity of radionuclides expected to be released annually to unrestricted areas in liquid and gaseous effluents produced during normal reactor operation.

3. 10 CFR 50, Section 50.34a(3) requires that each application to construct a nuclear power reactor provide a description of the provisions for packaging, storage and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

4. 10 CFR 50, Section 50.34A(3)(c) requires that each application to operate a nuclear power reactor shall include 1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems and 2) a revised estimate of the information required in (b)(2) if the expected releases and exposures differ significantly from the estimate submitted in the application for a construction permit.

5. The Regulatory staff's Safety Evaluation Report and amendments thereto issued prior to the issuance of an operating license contains a description of the radioactive waste systems installed in the nuclear power reactor and a detailed evaluation (including estimated releases of radioactive materials in liquid and gaseous waste and quantities of solid waste produced from normal operation, estimated annual maximum exposures to an individual in the unrestricted area and estimated exposures to the general population) which shows the capability of these systems to meet the appropriate regulations.

6.16 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Manager of Nuclear Safety and Licensing delegates the responsibility for the Radiological Environmental Monitoring Program to the Director, Environmental and Radiological Safety Programs of his designated alternate.

The Director, Environmental and Radiological Safety Programs is responsible for administering the offsite Radiological Environmental Monitoring Program. He shall determine that the sampling program is being implemented as described to verify that the environment is adequately protected under existing procedures. He shall also have the responsibility for establishing, implementing, maintaining and approving offsite environmental program sampling, analysis and calibration procedures.