SECTION 1.0

DEFINITIONS

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1.0 DEFINITIONS

DEFINED TERMS

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

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 600 MWt.

OPERATIONAL MODE

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

ACTION

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

OPERABLE - OPERABILITY

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

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified as a reportable occurrence in Specification 6.9.4.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.2.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 The air lock is OPERABLE pursuant to Specification 3.6.1.3, and

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

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

CORE ALTERATION

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

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. 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 not to interfere with the operation of leakage detection systems and not to be PRESSURE BOUNDARY LEAKAGE, or

c. Main Coolant System leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

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

PRESSURE BOUNDARY LEAKAGE

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

DOSE EQUIVALENT I-131

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

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

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

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

FREQUENCY NOTATION

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

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and authorized under the provisions of 10 CFR 50.59, or otherwise approved by the Commission.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.26 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95% of the total non-iodine activity in the coolant.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 2%	$\geq$ 330°F
2. STARTUP	$\geq$ 0.99	$\leq$ 2%	$\geq$ 330°F
3. HOT STANDBY	< 0.99	0	$\geq$ 330°F
4. HOT SHUTDOWN	< 0.99	0	330°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

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

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, Main Coolant System pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for 4 and 3 loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop cold leg temperature and THERMAL POWER has exceeded (is above and to the right of) the appropriate Main Coolant System pressure line, be in HOT STANDBY within 1 hour.

MAIN COOLANT SYSTEM PRESSURE

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

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

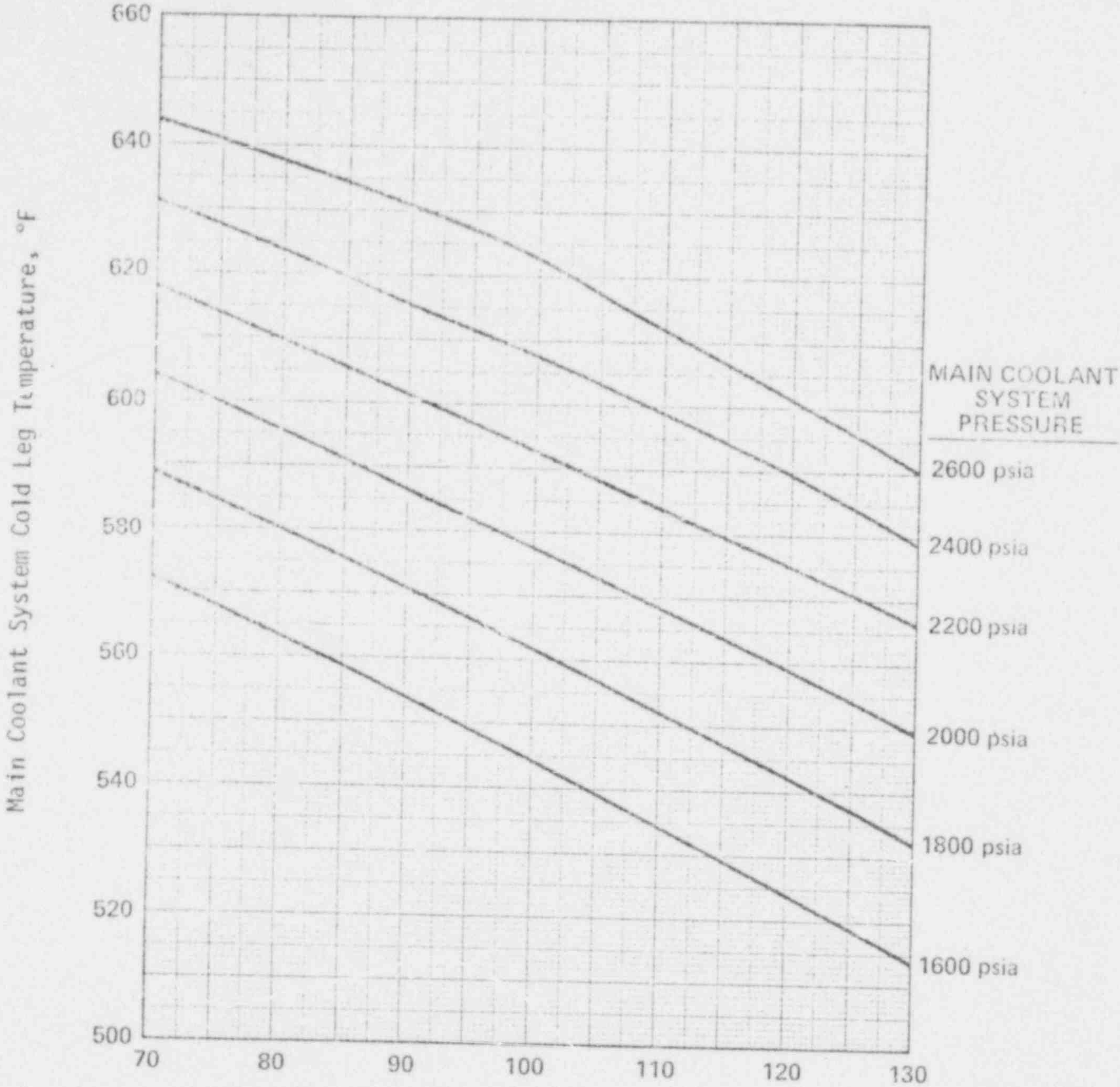
ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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



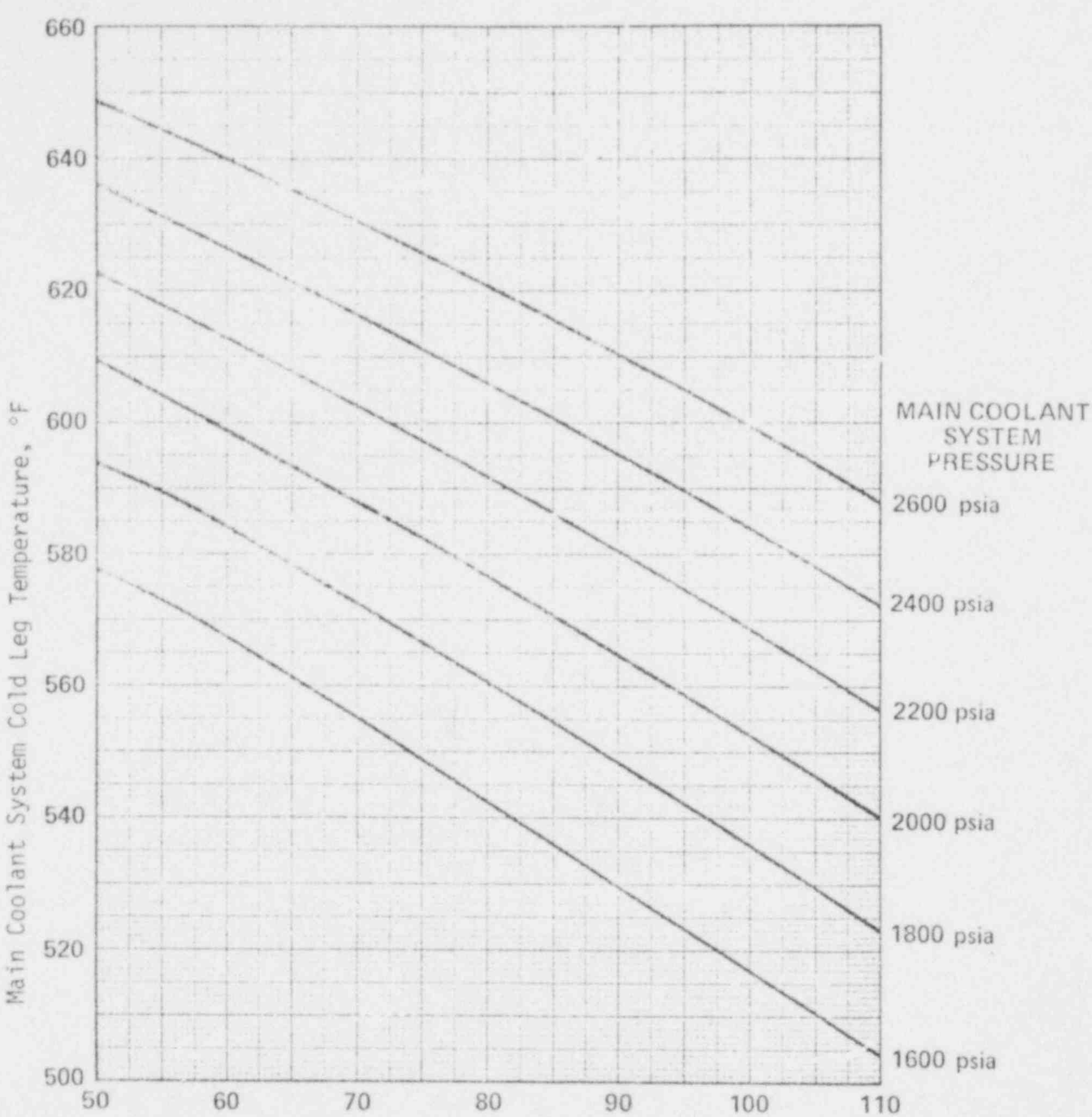
Indicated Reactor Power, Percent

REACTOR CORE SAFETY LIMIT - ALL LOOPS IN OPERATION

CORE XII

FIGURE 2.1-1

POOR ORIGINAL



Indicated Reactor Power, Percent

REACTOR CORE SAFETY LIMIT - 3 LOOPS IN OPERATION

CORE XII

FIGURE 2.1-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protective 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 protective system instrumentation trip setpoint less conservative than the value shown in the Trip Setpoint 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 PROTECTIVE SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. Manual Reactor Trip	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 35% of RATED THERMAL POWER
	High Setpoint - $\leq$ 108% of RATED THERMAL POWER with 4 main coolant pumps operating
	High Setpoint - $\leq$ 81% of RATED THERMAL POWER with 3 main coolant pumps operating
3. Intermediate Power Range, Neutron Flux	High Setpoint - $\leq$ 108% of RATED THERMAL POWER with 4 main coolant pumps operating
	High Setpoint - $\leq$ 81% of RATED THERMAL POWER with 3 main coolant pumps operating
4. Intermediate Range, High Startup Rate	$\leq$ 5.2 decades/minute
5. Source Range, Neutron Flux	Not Applicable
6. Low Main Coolant Flow (steam generator ΔP)	$\geq$ 80% of Design Flow
7. Low Main Coolant Flow (main coolant pump current)	$\geq$ 240 Amperes, $\leq$ 960 Amperes

TABLE 2.2-1 (continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
8. Low Pressurizer Pressure	$\geq$ 1800 psig
9. Low Main Coolant System Pressure	$\geq$ 1800 psig
10. High Pressurizer Water Level	$\leq$ 200 inches
11. Low Steam Generator Water	$\geq$ - 13*
12. Turbine Trip	Not Applicable
13. Generator Trip	Not Applicable

*Where 0 inches corresponds to 10" above the feed ring centerline.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the main 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 main coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

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

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Main Coolant System pressure and cold leg temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. Because of flow instability, DNB may occur prematurely should the core exit quality become too great. The limiting core exit quality for preventing flow instability is taken conservatively as 0.08.

The limiting hot channel factors used in determining the thermal limit curves are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion.

The curves are based on the following nuclear hot channel factors at or above 3 or 4 loop THERMAL POWER: F^N_q of 2.71 for EXXON Nuclear fuel and 2.70 for Gulf United fuel; $F^N_{\Delta H}$ of 1.88 for Gulf United fuel and 1.75 for EXXON Nuclear fuel; and a reference cosine with a peak of 1.43 for axial power shape.

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.

2.1.2 MAIN COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Main Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the main coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer and pumps are designed to Section VIII of the ASME Boiler and Pressure Vessel Code for Nuclear Power Plant, including all addenda through 1956, which permits a maximum transient pressure of 110%, 2735 psig, of design pressure. Pressure relief devices must be provided that will prevent pressure from exceeding 110 percent of the design pressure. The Main Coolant System piping and valves are designed to ANSI (formerly ASA) Standards, Power Piping Code, Section B31.1, 1955 Edition, and B16.5, 1957 Edition, respectively, which allows the design to be based on normal operating pressure and temperature and also allows exceeding the design conditions for periods of time. The stress level can be increased 15 percent above the Code allowable design value for not more than 10 percent of the design life and up to 20 percent above the allowable for up to 1 percent of the design life. Since normal plant operating pressure is 2000 psig, there is no conflict with either design condition. The setting of the Main Coolant System safety valves could allow pressure to increase to 2560 psig during a transient. The amount of time this condition is expected to exist is well within the allowances of B31.1. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Main Coolant System is hydrotested at 3435 psig, 138% of design pressure, to demonstrate integrity prior to initial operation.

BASES

2.2.1 REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint limits specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and Main Coolant System are prevented from exceeding their safety limits.

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 and Intermediate Power Range, Neutron Flux

The Power Range and Intermediate Power Range Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by pressurizer water level protective circuitry. The Power Range low set point provides additional protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed above 15 MWe and is manually reinstated at a power level below 15 MWe. The low setpoint trip is not assumed in the accident analysis.

The prescribed setpoint, with allowances for errors, is consistent with the trip point used in the accident analysis. The lower setting for three loop operation provides the protection at the reduced power level equivalent to that provided by the setting for four loop operation at full power.

Intermediate Range, Neutron Flux, High Startup Rate

The Intermediate Range High Startup Rate trip provides protection to limit the rate of power increase during low power conditions in the event of an uncontrolled rod withdrawal.

Low Main Coolant Flow (Steam Generator ΔP)

The Low Main Coolant Flow trips provide core protection in the event of a loss of one or more main coolant pumps.

Above a power of 15 MWE, with main coolant pumps operating, an automatic reactor trip will occur if the flow in any two loops drops below 80% of nominal full loop flow and, with 3 main coolant pumps operating, automatic reactor trip will occur if the flow in any single operating loop drops below 80% of nominal full loop flow. The setpoints specified are consistent with the value assumed in the accident analysis.

Low Main Coolant Flow (Main Coolant Pump Current)

The Low Main Coolant Flow trips provide core protection in the event of a loss of one or more main coolant pumps.

Above a power of 15 MWE, with 4 main coolant pumps operating, an automatic trip will occur if the main coolant pump motor current is outside the limits on any two pumps, and with 3 main coolant pumps operating, automatic trip will occur if the main coolant pump motor current is outside the limits on any operating pump. The setpoints specified are consistent with the value assumed in the accident analysis.

Main Coolant System and Pressurizer Pressure

The Main Coolant System and Pressurizer Low Pressure trips are provided to prevent operation in the pressure range in which DNBR is less than 1.30 ensuring that the thermal and hydraulic limits assumed in the accident analysis are not exceeded. These Low Pressure trips provide protection by tripping the reactor in the event of a loss of main coolant pressure.

Pressurizer High Water Level

The Pressurizer High Water Level trip ensures protection against Main Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble, prevents water relief through the pressurizer safety valves, and provides core protection for an uncontrolled rod withdrawal incident or loss of load accident.

Steam Generator Water Level

The Low Steam Generator Water Level trip provides core protection by preventing operation with the steam generator water level below the minimum volum 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 provide more than the 6 minutes assumed in the accident analysis for starting delays of the emergency boiler feedwater system in the 0.1 square foot LOCA analysis.

Turbine and Generator Trip

A Turbine or Generator Trip causes a direct reactor trip when operating above 15 MWE. 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 is required to enhance the overall reliability of the Reactor Protection System.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification.

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

3/4. REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 4.72% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN < 4.72% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 26 gpm of 2200 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 $\geq$ 4.72% $\Delta k/k$:

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(s) 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 4 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

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

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

*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Main Coolant System boron concentration,

2. Control rod position,

3. Main Coolant System average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 0.8% $\Delta k/k$ 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.

4.1.1.1.3 Whenever the reactor is shut down, before any operation which might result in a change of reactivity, a control rod group shall be withdrawn to a height sufficient to provide a reactivity worth of 1% for emergency shutdown capability. If for any reason this is not practical, the Main Coolant System shall be borated to provide 5% $\Delta k/k$ SHUTDOWN MARGIN with all control rods inserted.

4.1.1.1.4 During a reactor startup in which core reactivity or control rod positions for criticality are not established, a plot of inverse multiplication rate (or count rate) versus rod position shall be made.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN (with all control rods inserted) shall be $\geq$ 5.0% $\Delta k/k$.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the SHUTDOWN MARGIN (with all control rods inserted) < 5.0% $\Delta k/k$, immediately continue boration at $\geq$ 26 gpm of > 2200 ppm boric acid solution or equivalent and establish and maintain CONTAINMENT INTEGRITY until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN (with all control rods inserted) shall be determined to be $\geq$ 5.0% $\Delta k/k$:

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. Main Coolant System boron concentration,
2. Control rod position,
3. Main Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

SHUTDOWN MARGIN

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.2.2 Whenever the reactor is shut down, before any operation which might result in a change of reactivity, a control rod group shall be withdrawn to a height sufficient to provide a reactivity worth of 1% for emergency shutdown capability. If for any reason this is not practical, the main coolant system shall be borated to provide 5% $\Delta k/k$ SHUTDOWN MARGIN with all control rods inserted.

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 Main Coolant System boron concentration shall not be reduced unless:

a. The flow rate of main coolant to the reactor pressure vessel is $\geq$ 950 gpm;

b. The maximum reactivity insertion rate due to boron concentration reduction is $\leq 1.5 \times 10^{-4}$ $\Delta k/k$ per second; and

c. Main coolant temperature is $\geq$ 250°F.

APPLICABILITY All MODES.

ACTION:

a. With the flow rate of main coolant to the reactor pressure vessel < 950 gpm, immediately suspend all operations involving a reduction in boron concentration of the Main Coolant System.

b. With the maximum reactivity insertion rate due to main coolant boron concentration reduction in excess of the limit, immediately suspend boron concentration reduction, and verify the required SHUTDOWN MARGIN within one hour.

c. With the main coolant temperature < 250°F, immediately suspend boron concentration reduction and verify the required SHUTDOWN MARGIN within one hour.

SURVEILLANCE REQUIREMENTS

4.1.1.3

a. The flow rate of main coolant to the reactor pressure vessel shall be determined to be $\geq$ 950 gpm within one hour prior to the start of and at least once per hour during a reduction in the Main Coolant System boron concentration by either:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying at least one main coolant pump is in operation or

2. Verifying that the shutdown cooling system is in operation and supplying $\geq$ 950 gpm to the reactor pressure vessel.

b. Isolation valves of ion exchangers capable of reducing main coolant boron concentration shall be verified to be locked closed at least once per 31 days except when the ion exchanger is in use for boron removal.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Negative at hot zero power;

b. More negative than -0.5×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER; and

c. Less negative than -3.82×10^{-4} $\Delta k/k/°F$.

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% of RATED THERMAL POWER, after each fuel loading.

b. When restarting from the first shutdown longer than 72 hours after >60% of core life.

*With $K_{eff} \geq 1.0$

#See Special Test Exception 3.10.4

FLOW PATHS - REFUELING

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, the flow path from the boric acid mix tank via a gravity feed connection and at least two charging pumps to the Main Coolant System shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the above required flow path inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the above required flow path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 The above required flow path 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 $\geq$ 150°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.

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A flow path from the boric acid mix tank via the gravity feed connection and a charging pump to the Main Coolant System, if only the boric acid mix tank in Specification 3.1.2.10a is OPERABLE, or

b. The flow path from the safety injection tank via a charging pump to the Main Coolant System, if only the safety injection tank in Specification 3.1.2.10b is OPERABLE.

APPLICABILITY: MODE 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least one of the above required flow paths shall de 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 once complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the temperature of the heat traced portion of the flow path is $\geq$ 150°F when a flow path from the boric acid mix tank is OPERABLE.

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The flow path from the boric acid mix tank via the gravity feed connection and a charging pump to the Main Coolant System, and

b. The flow path from the safety injection tank via a charging pump to the Main Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the flow path from either the boric acid mix tank or the safety injection tank inoperable, provided the other flow path is OPERABLE, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN (all control rods inserted) equivalent to at least 5% $\Delta k/k$ at 200°F within the next 6 hours; restore the inoperable flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.3 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 once complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the temperature of the heat traced portion of the flow path is > 150°F when a flow path from the boric acid mix tanks is OPERABLE.

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.

CHARGING PUMPS - REFUELING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With less than two charging pumps OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least two charging pumps are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least the above required charging pumps shall be demonstrated OPERABLE at least once per 31 days by:

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

b. Verifying that each pump develops a discharge flow $\geq$ 26 gpm, and

c. Verifying pump operation for at least 15 minutes.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one charging pump in the boron injection flow path required by Specification 3.1.2.2 shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.5 At least 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 the pump develops a discharge pressure $>$ 30 psig in excess of the Main Coolant System pressure at a flow $\geq$ 26 gpm, and

c. Verifying pump operation for at least 15 minutes.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 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 HOT STANDBY and borated to a SHUTDOWN MARGIN (all control rods inserted) equivalent to at least 5% $\Delta k/k$ at 200°F 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

4.1.2.6 At least two charging pumps shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

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

b. Verifying that each pump develops a discharge pressure > 30 psig in excess of Main Coolant System pressure at a flow $\geq$ 26 gpm,

c. Verifying that each pump operates for at least 15 minutes, and

d. Verifying that the pump suction and discharge valves are open.

BORIC ACID MIX TANK GRAVITY FEED CONNECTION - SHUTDOWN AND REFUELING

LIMITING CONDITION FOR OPERATION

3.1.2.7 The boric acid mix tank gravity feed connection shall be OPERABLE if the flow path from the boric acid mix tank of Specification 3.1.2.1 or 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no mix tank gravity feed connection OPERABLE as required to complete the flow path of Specification 3.1.2.1 or 3.1.2.2a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the boric acid mix tank gravity feed connection is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required boric acid mix tank gravity feed connection shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

   2. 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.

b. At least once per 18 months, during shutdown, by demonstrating the gravity feed connection flow to be $\geq$ 26 gpm.

BORIC ACID MIX TANK GRAVITY FEED CONNECTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 The boric acid mix tank gravity feed connection in the boron injection flow path required by Specification 3.1.2.3 shall be OPERABLE if the flow path from the boric acid mix tank in Specification 3.1.2.3 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the boric acid mix tank gravity feed connection inoperable, restore the boric acid gravity feed connection to OPERABLE STATUS within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN (all control rods inserted) equivalent to 5% $\Delta k/k$ at 200°F; restore the boric acid gravity feed connection to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 The above required boric acid mix tank gravity feed connection shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

    2. 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.

b. At least once per 18 months, during shutdown, by demonstrating the gravity feed connection flow to be $\geq$ 26 gpm.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - REFUELING

LIMITING CONDITION FOR OPERATION

3.1.2.9 As a minimum, the boric acid mix tank and associated heat tracing shall be OPERABLE with:

a. A minimum contained volume of 1500 gallons, equivalent to a tank level $\geq$ 3.6 feet,

b. 12 to 12.5% by weight boric acid solution, and

c. A minimum solution temperature of 150°F.

APPLICABILITY: MODE 6.

ACTION:

With the boric acid mix tank inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the boric acid mix tank is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.9 The boric acid mix tank shall be demonstrated OPERABLE at least once per 7 days by:

a. Verifying the boron concentration of the water,

b. Verifying the water level of the tank, and

c. Verifying the boric acid mix tank solution temperature.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. The boric acid mix tank and associated heat tracing with:

1. A minimum contained volume of 1500 gallons, equivalent to a tank level $\geq$ 3.6 feet,

2. 12 to 12.5% by weight boric acid solution, and

3. A minimum solution temperature of 150°F.

b. The safety injection tank (SIT) with:

1. A minimum contained volume of 117,000 gallons, equivalent to a tank level of $\geq$ 25.5 feet,

2. A minimum boron concentration of 2200 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODE 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the water level of the tank, and

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the boric acid mix tank solution temperature when it is the source of borated water.

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The boric acid mix tank and associated heat tracing with:

1. A minimum contained volume of 1500 gallons, equivalent to a tank level of $\geq$ 3.6 feet,

2. 12 to 12.5% by weight boric acid solution,

3. A minimum solution temperature of 150°F.

b. The safety injection tank (SIT) with:

1. A minimum contained volume of 117,000 gallons of water, equivalent to a tank level of $\geq$ 25.5 feet,

2. A minimum boron concentration of 2200 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either the boric acid mix tank or the safety injection tank inoperable, provided the other required source is OPERABLE, restore the inoperable tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN (all control rods inserted) equivalent to at least 5% $\Delta k/k$ at 200°F; restore the inoperable tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.11 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

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

   3. Verifying the boric acid mix tank solution temperature.

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

3/4.1.3 MOVABLE CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods which are inserted in the core shall be OPERABLE and positioned within $\pm$ 8 inches (indicated position) of every other rod in their group.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more control 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 control rod inoperable or misaligned from any other rod in its group by more than $\pm$ 8 inches (indicated position), be in HOT STANDBY within 6 hours.

c. With one control rod inoperable or misaligned from any other rod in its group by more than $\pm$ 8 inches (indicated position), POWER OPERATION may continue provided that within one hour either:

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

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

      a) An analysis of the potential ejected rod worth is performed within 3 days and the rod worth is determined to be $\leq$ 1.0% Δk at zero power and $\leq$ 0.5% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

*See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

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

c) The THERMAL POWER level is reduced to $\leq$ 75% of THERMAL POWER allowable for the Main Coolant pump combination within one hour and within the next 4 hours the Power Range and Intermediate Power Range Neutron Flux high trip setpoint is reduced to $\leq$ 108% of the 75% of allowable THERMAL POWER, or

d) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 8 inches of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each control rod shall be determined to be within the limit by verifying the individual rod positions at least once per 4 hours.

4.1.3.1.2 Each control rod not fully inserted shall be determined to be OPERABLE by movement of at least 4 inches in any one direction at least once per 31 days.

4.1.3.1.3 The maximum reactivity insertion rate due to withdrawal of the highest worth control rod group shall be determined not to exceed 1.5×10^{-4} $\Delta k/k$ per second at least once per 18 months.

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All control rod primary and secondary position indicator channels shall be OPERABLE and capable of determining the control rod positions within ± 3 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

   2. Reduce THERMAL POWER to < 50% of THERMAL POWER allowable for the main coolant pump combination within 8 hours.

b. With a maximum of one secondary position indicator per group inoperable either:

   1. Verify that all primary rod position indicators for the affected group are OPERABLE, or

   2. Reduce THERMAL POWER to < 50% of THERMAL POWER allowable for the main coolant pump combination within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the primary position indication system and the secondary position indicator channels agree within 3 inches at least once per 4 hours.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual control rod drop time from the fully withdrawn position shall be $\leq$ 2.5 seconds from loss of stationary gripper coil voltage to 6-inch coil entry with:

a. $T_{avg} \geq 511°F$, and

b. All main coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

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

b. With the rod drop times within limits but determined with 3 main coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to $\leq$ 75% of RATED THERMAL POWER. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

1.3.3 The rod drop time of control 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 would affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be withdrawn to at least 87 inches.

APPLICABILITY: MODES 1* and 2*#

ACTION:

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

a. Withdraw the rod to within the limit, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be withdrawn to within the limit:

a. Within 15 minutes prior to withdrawal of any rods in regulating groups A & B during an approach to reactor criticality,

b. At least once per 4 hours thereafter.

---

*See Special Test Exceptions 3.10.2 and 3.10.4.
#With $K_{eff} \geq 1.0$.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control groups shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

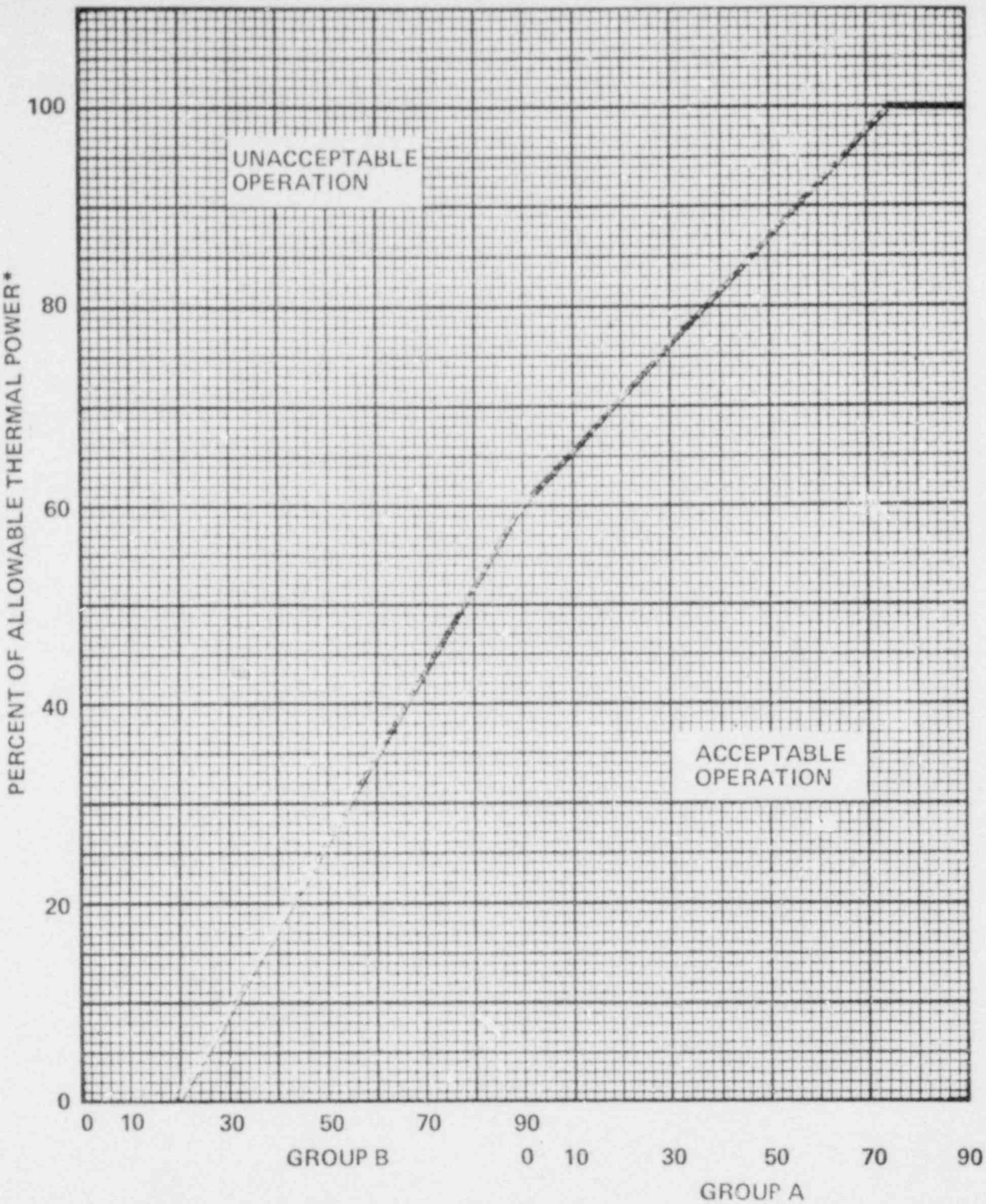
b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control group shall be determined to be within the insertion limits at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4
#With $K_{eff} \geq 1.0$.



Control Rod Position, Inches Withdrawn

*Allowable THERMAL Power based on the main coolant pump combination in operation.

FIGURE 3.1-1

PEAK LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The peak linear heat generation rate (LHGR) shall not exceed the limits of Figure 3.2-1 during steady state operation.

APPLICABILITY: MODE 1

ACTION:

With the peak LHGR exceeding the limits of Figure 3.2-1;

a. Within 15 minutes reduce THERMAL POWER to not more than that fraction of the THERMAL POWER allowable, for each fuel type and for the main coolant pump combination in operation, as expressed below:

$$\text{Fraction of THERMAL POWER} = \frac{\text{Limiting LHGR}}{\text{Peak Full Power LHGR}}$$

b. Within 4 hours reduce the Power Range and Intermediate Power Range Neutron Flux high trip setpoint to $\leq$ 108% of the fraction of THERMAL POWER allowable for the main coolant pump combination.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The peak LHGR shall be determined to be within the limits of Figure 3.2-1 using incore instrumentation to obtain a power distribution map:

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

b. At least once per 1,000 EFPH.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The below factors shall be included in the calculation of peak full power LHGR:

a. Heat flux power peaking factor, F_q^N, measured using incore instrumentation at a power $\geq$ 10%.

b. Effect of inserting the control group from its position at the time of measurement to its insertion limit, F_I as shown in Figure 3.2-2. The rod insertion limit is shown in Figure 3.1-1.

c. The multiplier for xenon redistribution is a function of core lifetime as given in Figure 3.2-3. In addition, if control rod Group A is inserted below 75 inches, allowable power may not be regained until power has been at a reduced level defined below for at least twenty four hours with control rod Group A between 75 and 90 inches.

Reduced power = allowable fraction of full power times multiplier given in Figure 3.2-4.

Exception: If the rods are inserted below 75 inches and power does not go below the reduced power calculated above, hold at the lowest attained power level for at least twenty four hours with control rod Group A between 75 and 90 inches before returning to allowable power.

d. Shortened stack height factor, 1.009.

e. Measurement uncertainty, 1.05.

f. Power level uncertainty, 1.03.

g. Heat flux engineering factor, F_q^E, 1.04.

h. Core average linear heat generation rate at full power, 4.34 kw/ft.

4.2.1.3 At least once per 1000 EFPH the following limits shall be determined by calculation not to be exceeded at RATED THERMAL POWER:

a. Hottest channel exit coolant temperature $\leq$ 624°F, and

b. Maximum clad surface temperature in hottest channel $\leq$ 647°F.

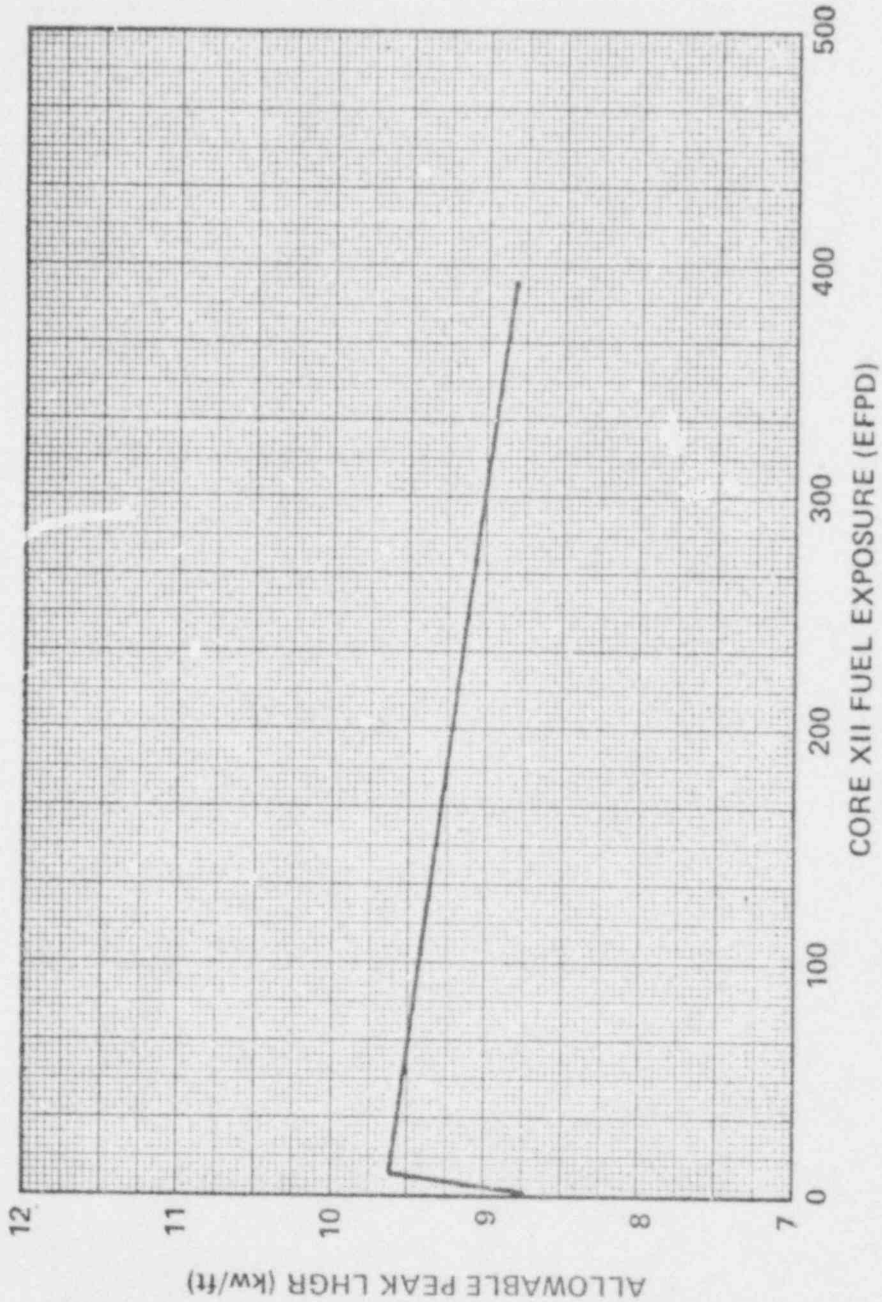
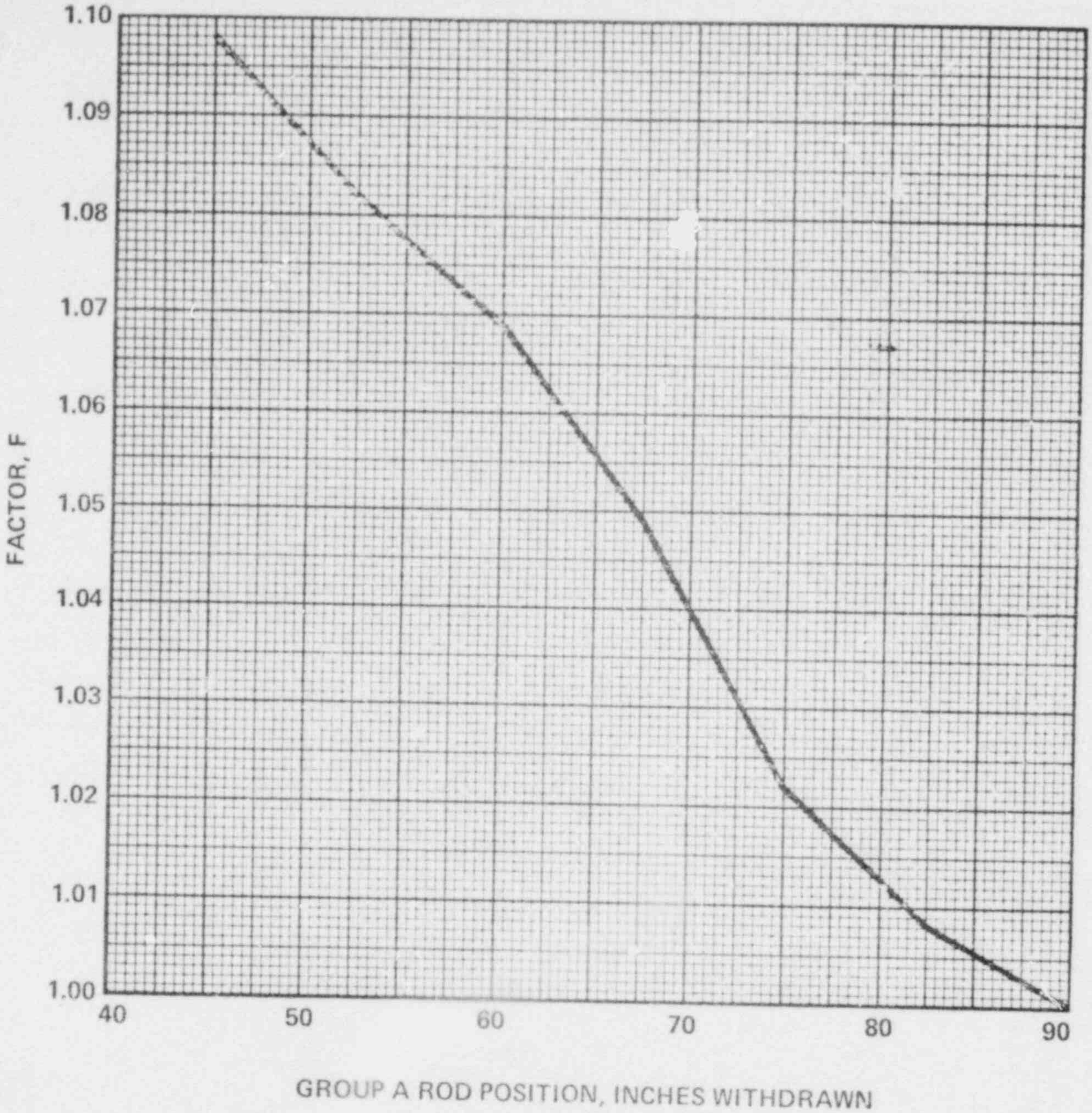


FIGURE 3.2-1

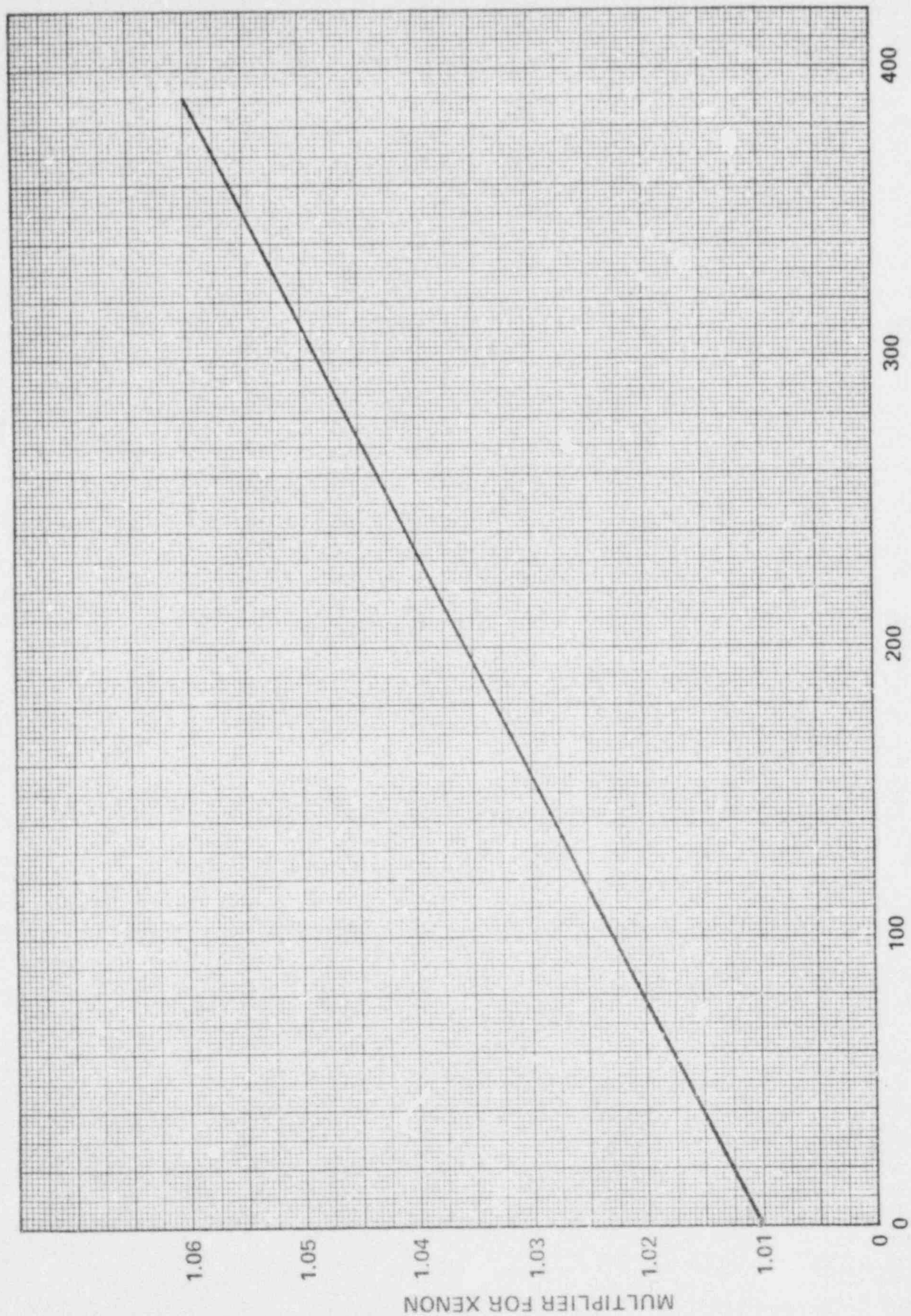
Core XII Allowable Peak LHGR Versus Exposure



$$F_I = \frac{F \text{ @ Limit}}{F \text{ @ Measurement}}$$

FIGURE 3.2-2

Factor F as a Function of Rod Insertion



Core XII Fuel Exposure (EFPD)

Figure 3.2-3

Multiplier for Xenon Redistribution As a Function of Exposure

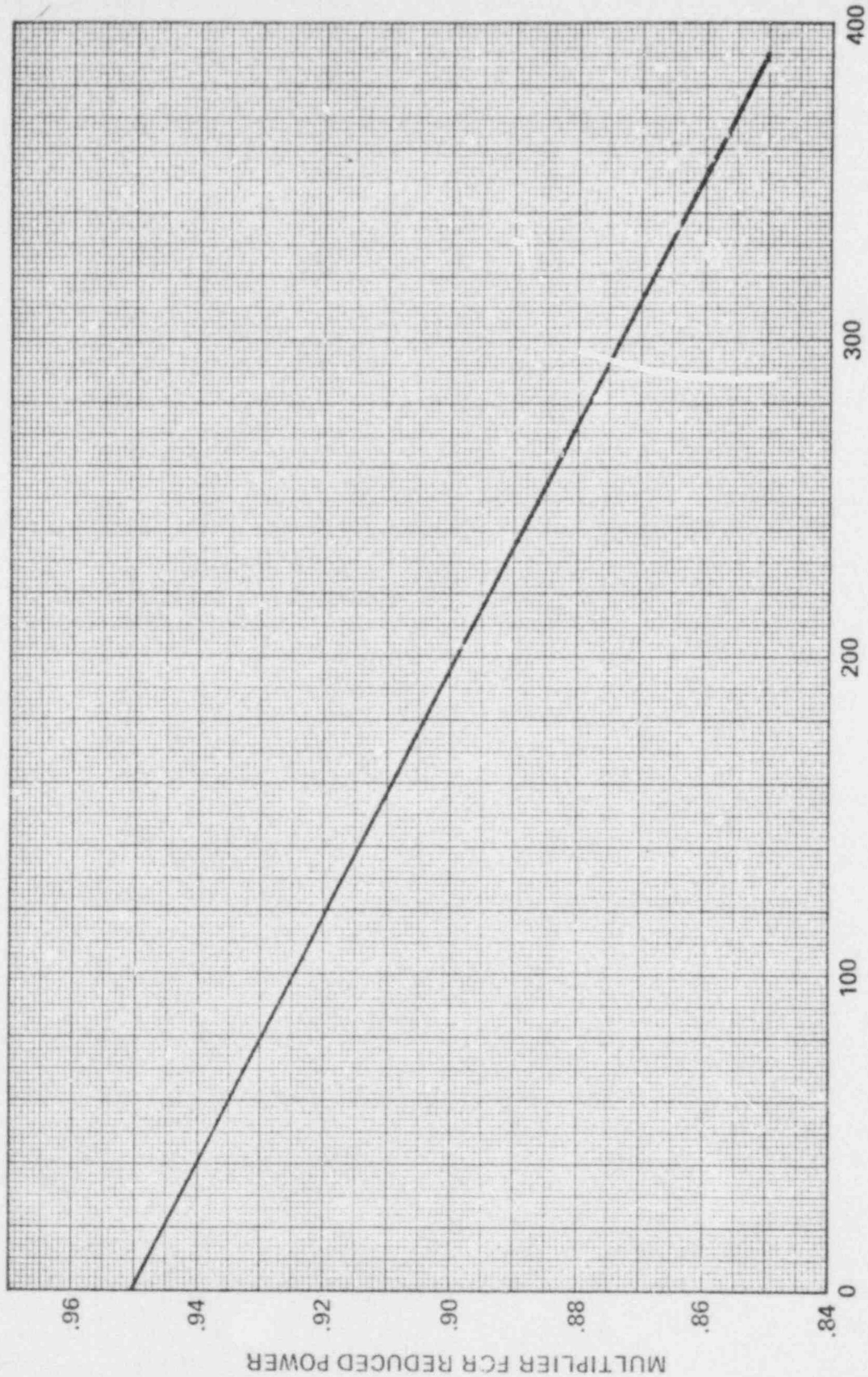


Figure 3.2-4

Multiplier for Reduced Power As a Function of Exposure

POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR-F_q^N

LIMITING CONDITION FOR OPERATION

3.2.2 F_q^N shall not exceed (____).

APPLICABILITY: MODE 1

ACTION:

With F_q^N exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each (1)% F_q^N exceeds the limit within 15 minutes and similarly reduce the Power Range and Intermediate Power Range Neutron Flux-high trip setpoints within the next 4 hours.

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^N is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_{xy} shall be evaluated to determine if F_q^N is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map:

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

2. At least once per 1000 Effective Full Power Hours.

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

c. Comparing the computed F_{xy} obtained in b, above, to the appropriate values of Figure 4.3-X of the FSAR or the reload safety

SURVEILLANCE REQUIREMENTS (Continued)

evaluation for the current fuel cycle. This comparison shall be limited to core planes between (15)% and (85)% of full core height inclusive and shall exclude regions influenced by grid effects.

d. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if F_Q^N is within its limit in the event the computed F_{xy} exceeds the appropriate value of c, above.

4.2.2.2 When F_q^N is measured pursuant to Specification (4.10.2.2), an overall measured F_q^N shall be obtained from a power distribution map and increased by (3)% to account for manufacturing tolerances and further increased by (5)% to account for measurement uncertainty.

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

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall not exceed 1.86 for Gulf United fuel and 1.75 for EXXON Nuclear Fuel

APPLICABILITY: MODE 1

ACTION:

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

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range and Intermediate Power Range Neutron Flux-high trip setpoints to $\leq$ 55% 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% 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% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 1000 Effective Full Power Hours.

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

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Main Coolant System Inlet Temperature.

b. Pressurizer Pressure

c. Main Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation	 3 Loops in Operation
Main Coolant System Inlet Temperature	(510 ± 4)°F	(510 ± 4)°F
Pressurizer Pressure	2015 ± 75 psia*	2015 ± 75 psia*
Main Coolant Sytstem Total Flow Rate	$\geq 38.3 \times 10^6$ lb/hr	$\geq 29.9 \times 10^6$ lb/hr

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective system instrumentation channels and reactor permissive functions of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.2 The logic for the Reactor Permissive Circuit shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by permissive circuit operation. The total permissive function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by permissive circuit operation.

TABLE 3.3-1

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	3	1	3	1, 2 and *	1
2. Power Range, Neutron Flux and Intermediate Power Range, Neutron Flux	6	2	4	1, 2 and *(1)	2
3. Intermediate Range, Neutron Flux, High Startup Rate	2	1	2	1, 2(2) and *	3
4. Source Range, Neutron Flux					
a. Startup##	2	NA	2	2# and *(5)	4
b. Shutdown	2	NA	1	3, 4, 5(5)	5
5. Low Main Coolant Flow (SGΔP)	4	2	3	1(3)	6
6. Low Main Coolant Flow (MC Pump Current)					
a. System A	4	2	3	1(3)	7
b. System B	4	2	3	1(3)	7
7. Low Pressurizer Pressure	1	1	1	1, 2(4)	8
8. Low Main Coolant System Pressure	1	1	1	1, 2(4)	8
9. High Pressurizer Water Level	1	1	1	1, 2(4)	8
10. Low Steam Generator Water Level	4	2	3	1(3)	6

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Turbine Trip	1	1	1	1(3)(6)	8
12. Generator Trip	1	1	1	1(3)(7)	8
13. Reactor Trip Breaker	2	1	2	1, 2 and *	9
14. Automatic Trip Logic	2	1	2	1, 2 and *	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

#High voltage to detector is automatically de-energized above 5×10^{-9} Amperes on the Intermediate Range.

##Or when other activities might increase reactivity.

(1) Power Range, Neutron Flux, Low Setpoint Trip may be manually bypassed at $\geq$ 15 MWe. Bypass shall be manually removed at $\leq$ 15 MWe.

(2) Intermediate Range, Neutron Flux, High Startup Rate Trip is automatically bypassed $\geq$ 15 MWe. Bypass is automatically removed at $\leq$ 15 MWe.

(3) Trip may be manually bypassed $\leq$ 15 MWe. Bypass is automatically removed at $\geq$ 15 MWe.

(4) Trip may be manually bypassed when the reactor is not critical.

(5) Startup rate alarm setpoint $\leq$ 1.1 decade/minute.

(6) Turbine shall be protected by at least the following protective trips: rotor excessive axial movement, low bearing oil pressure; low condenser vacuum; and overspeed.

(7) Generator shall be protected by at least the following protective trips: overcurrent; differential; and loss of field.

ACTION STATEMENTS

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 2 -

a. With two Power Range and/or Intermediate Power Range channels inoperable and with the THERMAL POWER $\leq$ 2% of RATED THERMAL POWER, set the power range high neutron level scram logic for single channel operation for both the Power Range and Intermediate Power Range instrumentation.

b. With one Power Range or one Intermediate Power Range channel inoperable and with the THERMAL POWER level > 2% of RATED THERMAL POWER, POWER OPERATION may continue at the existing THERMAL POWER level provided the power range high neutron level scram logic is set for single channel operation for the RPS instrumentation group containing the inoperable channel.

c. With two Power Range channels, or two Intermediate Power Range channels, or one Power Range and one Intermediate Power Range channels inoperable and with the THERMAL POWER level > 2% of RATED THERMAL POWER, POWER OPERATION may continue at the existing THERMAL POWER level provided the power range high neutron level scram logic is set for single channel operation for both the Power Range and Intermediate Power Range instrumentation.

d. One power range high neutron level scram logic may be reset for coincidence operation in a, b and c above for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 3 - With the number of OPERABLE channels one less than the Total Number of Channels and with the power level:

a. Less than or equal to 15 MWe, place the inoperable channel in the bypassed condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours unless reactor power level has been increased to > 15 MWe; otherwise be in at least HOT STANDBY within the following 6 hours with the reactor trip breakers open.

b. Above 15 MWe, 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:

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

a. Reactor not critical, restore the inoperable channel to OPERABLE status prior to increasing reactor reactivity,

b. Reactor critical, operation may continue.

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

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels and with the power level:

a. Less than or equal to 15 MWe, operation may continue.

b. Above 15 MWe, operation may continue provided both 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.

ACTION 7 -

a. With the number of OPERABLE channels in either System A or B one less than the Total Number of Channels, operation may continue provided all channels in the other system are OPERABLE.

b. With the number of OPERABLE channels in both Systems A and B one less than the Total Number of Channels each system and with the power level:

1. Less than or equal to 15 MWe, operation may continue.

2. Above 15 MWe, operation may continue provided both of the following conditions are satisfied:

a) The inoperable channels are placed in the tripped condition within 1 hour.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 7 (Continued) -

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

ACTION 8 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours.

ACTION 9 - With the number of channels OPERABLE one less than required by the Minimum Channels Operable requirement, be in HOT STANDBY within 6 hours with reactor trip breakers open.

TABLE 4.3-1

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	NA	NA	S/U(1)	NA
2. Power Range, Neutron Flux and Intermediate Power Range, Neutron Flux	S	D(2), Q	M	1, 2 and*
3. Intermediate Range, Neutron Flux, High Startup Rate	S	R	M	1, 2 and*
4. Source Range, Neutron Flux	S	R	S/U(1)	2,3,4,5 and*
5. Low Main Coolant Flow (SGΔP)	S	R(4)	M(3)	1
6. Low Main Coolant Flow, Systems A and B (MC Pump Current)	S	R	M	1
7. Low Pressurizer Pressure	S	R(4)	M(3)	1, 2
8. Low Main Coolant System Pressure	S	R(4)	M(3)	1, 2
9. High Pressurizer Water Level	S	R(4)	M(3)	1, 2
10. Low Steam Generator Water Level	S	R(4)	M	1

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11. Turbine Trip	NA	NA	S/U(1)	1
12. Generator Trip	NA	NA	S/U(1)	1
13. Reactor Trip Breaker	NA	NA	S/U(1)	1, 2 and *
14. Automatic Trip Logic	NA	NA	S/U(1)	1, 2 and *

TABLE 4.3-1 (Continued)

NOTATION

*	-	With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.
(1)	-	If not performed in the previous 7 days.
(2)	-	Heat balance only, above 15% of RATED THERMAL POWER, at least 3 times per week with a maximum time interval of 72 hours.
(3)	-	When shutdown longer than 24 hours, if not performed in the previous 31 days.
(4)	-	Known pressure applied to sensor.

3/4.3.2 ENGINEERED SAFEGUARDS SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safeguards System (ESS) instrumentation channels and sensors shown in Table 3.3-2 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESS instrumentatic channel sensor trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.3-3, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-2 until the channel is restored to OPERABLE status with the sensor trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESS instrumentation channel inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESS instrumentation channel and sensor 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 total bypass function of all bypasses shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

TABLE 3.3-2

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS AND SENSORS	CHANNELS AND SENSORS TO TRIP	MINIMUM CHANNELS AND SENSORS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. Actuation Channel #1	1	1	1	1, 2, 3(2)(3)	10
1) RPS Low Main Coolant Pressure Channel	1	1	1	1, 2, 3(2)(3)	10
2) High Containment Pressure Sensor	1	1	1	1, 2, 3(2)(3)	10
3) Manual Initiation	1	1	1	1, 2, 3, 4, 5(1)	10
b. Actuation Channel #2	1	1	1	1, 2, 3(2)	10
1) Low Pressurizer Pressure Sensor	1	1	1	1, 2, 3(2)	10
2) High Containment Pressure Sensor	1	1	1	1, 2, 3(2)	10
3) Manual Initiation	1	1	1	1, 2, 3, 4, 5(1)	10
2. CONTAINMENT ISOLATION					
a. Manual Initiation	1	1	1	1, 2, 3, 4, 5(1)	10
b. Actuation Channel	1	1	1	1, 2, 3, 4, 5(1)	10
1) High Containment Pressure Sensor	2	1	2	1, 2, 3, 4, 5(1)	10

TABLE 3.3-2 (Continued)

TABLE NOTATION

(1) Trip function may be bypassed in this MODE with main coolant pressure < 300 PSIG.

(2) Trip function may be bypassed in this MODE with main coolant pressure < 1800 PSIG.

(3) Automatic initiation of Actuation Channel #1 may be bypassed in this MODE during functional test of the Main Coolant System pressure channel.

ACTION STATEMENTS

ACTION 10 - With the number of OPERABLE channels or sensors one less than the Total Number of Channels or sensors, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one safety injection channel high containment pressure sensor may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. SAFETY INJECTION	
a. Actuation Channel #1	
1) RPS Low Main Coolant Pressure Channel	$\geq$ 1700 psig
2) High Containment Pressure Sensor	$\leq$ 5 psig
3) Manual Initiation	Not Applicable
b. Actuation Channel #2	
1) Low Pressurizer Pressure Sensor	$\geq$ 1700 psig
2) High Containment Pressure Sensor	$\leq$ 5 psig
3) Manual Initiation	Not Applicable
2. CONTAINMENT ISOLATION	
a. Manual Initiation	Not Applicable
b. Actuation Channel	
1) High Containment Pressure Sensor	$\leq$ 5 psig

TABLE 4.3-2

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION				
a. Actuation Channel #1	S	R	M(2)	1, 2, 3#
1) RPS Low Main Coolant Pressure Channel	S	R(3)	M(2)	1, 2, 3#
2) High Containment Pressure Sensor	S	R(3)	M(3)	1, 2, 3#
3) Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4, 5*
b. Actuation Channel #2	S	R	M(2)	1, 2, 3#
1) Low Pressurizer Pressure Sensor	S	R(3)	M(2)	1, 2, 3#
2) High Containment Pressure Sensor	S	R(3)	M(3)	1, 2, 3#
3) Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4, 5*
2. CONTAINMENT ISOLATION				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4, 5*
b. Actuation Channel	S	R	M(2)	1, 2, 3, 4, 5*
1) High Containment Pressure Sensor	S	R(3)	M(2)	1, 2, 3, 4, 5*

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown.

(2) When Shutdown longer than 24 hours, if not performed in the previous 31 days.

(3) The test shall include exercising the sensor by applying either a vacuum or pressure to the appropriate side of the sensor.

* Not required in this MODE with main coolant pressure < 300 PSIG.

\# Not required in this Mode with main coolant pressure < 1700 PSIG.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-4 shall be OPERABLE with their alarm setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With a radiation monitoring channel alarm setpoint exceeding the Alarm Setpoint shown in Table 3.3-4, 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 3.3-4.

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 3.3-4

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pit Area					
1) Fuel Manipulator Gamma Guard	1	*	$\leq$ 5 mr/hr or 2 x background, whichever is greater	0.5 - 50 mr	20
b. Containment					
1) Fuel Manipulator Gamma Guard	1	*	$\leq$ 10 mr/hr or 2 x background, whichever is greater	0 - 1000 mr	21
2. PROCESS MONITORS					
a. Containment					
1) Main Coolant System Leakage Air Particulate Monitor	1	1, 2, 3, & 4	$\leq$ 30 CPS above background	$10 - 10^4$ CPS	22
b. Radioactive Gaseous Waste Monitor					
1) Loop Seal Discharge Monitor	1	**	$\leq$ 2000 cpm or $\overline{2}$ x background whichever is greater	$10 - 10^4$ cpm or $10 - 10^6$ cpm	23
2) Primary Vent Stack Monitor					
a) Particulate Monitor	1	At all times	$\leq$ 900 cpm	$10 - 10^6$ cpm	24

TABLE 3.3-4 (Continued)

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
b) Iodine Monitor	1	At all times	$\leq$ 700 cpm	$10 - 10^6$ cpm	24
c) Noble Gas Monitor	1	At all times	$\leq$ 3500 cpm	$10 - 10^6$ cpm	24
c. Radioactive Liquid Monitors					
1) Steam Generator Blowdown Monitor	1[(1)]	1, 2, 3 & 4	$\leq$ 80 cps or 2 x background, whichever is greater	1×10^3 cps	23
3. ACCIDENT-EMERGENCY MONITORS					
a. High Level Radiation Monitor	1	At all times	$\leq$ 5 R/hr	0.01 - 1000 R/hr	23
b. Environmental Monitor (MAP-1)	2	At all times*	$\leq$ 2 x background	0 - 200 cpm 0 - 2000 cpm 0 - 20,000 cpm	23

* In standby, not normally operating.

TABLE 3.3-4 (Continued)

TABLE NOTATION

* When handling irradiated fuel, control rods, or sources.

** With radioactive effluent in the waste gas surge drum.

(1) Per steam generator in a non-isolated loop.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving CORE ALTERATIONS.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, provide an OPERABLE temporary continuous monitor within 8 hours.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend all planned releases and releases from the evaporator to the atmosphere through the primary vent stack.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pit Area (1) Fuel Manipulator Gamma Guard	S	R	M	*
b. Containment (1) Fuel Manipulator Gamma Guard	S	R	M	*
2. PROCESS MONITORS				
a. Containment Main Coolant System Leakage Air Particulate Monitor	S	R	M	1, 2, 3, 4
b. Radioactive Gaseous Waste Monitors				
(1) Loop Seal Discharge Monitor	S	R	M	**
(2) Primary Vent Stack Monitors	S	R	M	At all times
c. Radioactive Liquid Monitors				
(1) Steam Generator Blow-Down Monitor	S	R	M	1, 2, 3, 4

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. ACCIDENT-EMERGENCY MONITORS				
a. High Level Radiation Monitor	S	R	M	At all times
b. Environmental Monitors (MAP-1)	M	R	M	At all times

*When handling irradiated fuel, control rods or sources.

**With radioactive effluent in the waste gas surge drum.

INCORE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of the neutron detector thimbles OPERABLE,

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

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

d. At least ten OPERABLE radial position thermocouples with an OPERABLE thermocouple in at least one of the two calculated hottest instrumented fuel assemblies.

APPLICABILITY: When the incore detection system is used for core power distribution measurements.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE by normalizing each detector output to be used within 24 hours prior to its use for core power distribution measurements.

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The meteorological monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE meteorological monitoring channels less than required by Table 3.3-5, 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 6.9.6 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

4.3.3.3 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 4.3-4.

TABLE 3.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM OPERABLE
1. WIND SPEED			
	Nominal Elev. 30 feet	$\pm$ 0.5 mph*	1
2. WIND DIRECTION			
	Nominal Elev. 30 feet	$\pm$ 5°	1
3. AIR TEMPERATURE - DELTA T			
	Nominal Elev. 140-30 feet	$\pm$ 0.5° F	1

* Starting speed of anemometer shall be < 1 mph.

TABLE 4.3-4

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 30 feet	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 30 feet	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 140-30 feet	D	SA

3/4.4 MAIN COOLANT SYSTEM

3/4.4.1 MAIN COOLANT LOOPS

NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All main coolant loops shall be in operation with all loop isolation valves, in a non-isolated loop, OPERABLE with power removed per Specification 4.5.2.b.4.

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

ACTION:

a. MODE 1: With one main coolant loop and associated pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to less than 75% of RATED THERMAL POWER and the following Reactor Protective System instrumentation channels associated with the loop not in operation, are placed in their tripped condition within 1 hour:

1. Low Main Coolant Flow (Steam generator ΔP)
2. Low Main Coolant Flow (Main coolant pump current)
3. Low Steam Generator Water Level

b. MODES 2,3,4, and 5:

1. With $K_{eff} \geq 1.0$ operation may proceed provided at least two main coolant loops and associated pumps are in operation.
2. With $K_{eff} < 1.0$, operation may proceed provided at least one main coolant loop is in operation with an associated main coolant pump or the Shutdown Cooling System is in operation, except when there is a substantial amount of decay heat in the reactor fuel, then at least two main coolant loops shall be tied to the reactor vessel (main coolant pumps may be off), or the Shutdown Cooling System shall be in operation.

c. MODES 3,4, and 5: For purposes of cold leak testing only, the reactor vessel and connecting pressurizer system may be isolated, by closing the loop isolation valves, from the heat removal system provided the coolant temperature in the reactor vessel does not increase at a rate exceeding 50°F per hour, the maximum temperature increase during the test period does not exceed 100°F, and pressurizer pressure does not exceed 2485 psig.

LIMITING CONDITION FOR OPERATION (Continued)

d. MODES 1,2,3, and 4: With one or more main coolant system loop isolation valves in non-isolated loop inoperable, restore the inoperable valve(s) to operable status prior to exceeding 200°F T_{avg}.

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

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 In MODE 1, with one reactor coolant loop and associated pump not in operation, at least once per 7 days determine that the applicable Reactor Protection System instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions.

4.4.1.1.2 In MODES 2,3,4 or 5, determine that the steam generators associated with the main coolant loops in operation are capable of decay heat removal by verifying at least once per 24 hours that:

a. The Main Coolant System is closed and pressurized to $\geq$ 100 psi above saturation pressure.

b. The Main Coolant System loop cold and hot leg stop valves are fully open, with the bypass valve closed.

c. The steam generator water level is above the top of the tube bundle.

d. An inventory of over 85,000 gallons of primary grade feedwater is available.

e. A boiler feed pump is OPERABLE.

4.4.1.1.3 Verify that the Shutdown Cooling System isolation valves are locked closed within one hour prior to increasing Main Coolant System pressure above 300 PSIG.

4.4.1.1.4 At least once per 18 months, during shutdown, demonstrate main coolant loop isolation valve operability by cycling each valve through at least one complete cycle of full travel from the control room.

ISOLATED LOOP

LIMITING CONDITION FOR OPERATION

3.4.1.2 The boron concentration of an isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops.

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

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loop's 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 Main Coolant System borated to a SHUTDOWN MARGIN (all rods inserted) equivalent to at least 5% $\Delta k/k$ at 200°F.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 The boron concentration of an isolated loop shall be determined to be greater than or equal to the boron concentration of the operating loops at least once per 24 hours and within 30 minutes prior to opening either the hot leg or cold leg stop valves of an isolated loop. With Main Coolant System loop isolation valve controls removed from service and mechanically prevented from operation by lock and key, the frequency of determination that the boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops may be reduced to 3 times per week with a maximum of 72 hours between analyses.

4.4.1.2.2 An isolated loop shall be determined to be borated to at least 5% $\Delta K/K$ at 200°F before loop temperature is reduced $\geq$ 30°F below the highest cold leg temperature of the operating loops. With Main Coolant System loop isolation valve controls, main steam line isolation valves and main coolant pump controls removed from service and mechanically prevented from operation by lock and key, the boron concentration of the isolated loop may be reduced to that existing in the Main Coolant System at that time.

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.3 A main coolant loop shall remain isolated until:

a. The temperature of the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops.

b. The boron concentration of the isolated loop is not less than the main coolant system boron concentration. and

c. 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

4.4.1.3.1 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.

4.4.1.3.2 The isolated loop boron concentration shall be determined to be not less than the Main Coolant System boron concentration within 4 hours prior to opening the cold leg stop valve.

4.4.1.3.3 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.

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 +0, -3% or 2560 PSIG +0, -3%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place the Shutdown Cooling System into operation.

SURVEILLANCE REQUIREMENTS

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

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 Two pressurizer code safety valves shall be OPERABLE with lift settings of 2485 PSIG +0, -3% and 2560 PSIG +0, -3%, respectively.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.3 Each pressurizer code safety valve shall be demonstrated OPERABLE with lift settings of 2485 PSIG +0, -3% and 2560 PSIG +0, -3%, respectively, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, and Addenda through Summer 1975.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble.

APPLICABILITY: MODES 1 and 2

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer shall be determined operable:

a. At least once per 18 months by determining that;

1. Pressurizer circulation spray flow is sufficient to require 500-700 KW per 8 hours pressurizer heater power consumption.

2. Sufficient pressurizer surge spray flow exists by opening the spray valve and observing the resultant pressurizer pressure decrease and heater cycling.

b. At least once per 18 months, and after completion of any maintenance on the solenoid relief valve, by determining that the solenoid relief valve opens at $\leq 2400 \pm 30$ psig and closes at $\leq 2350 \pm 30$ psig on a test signal from the RPS pressurizer pressure channel.

3/4.4.5 MAIN COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Main Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment drain tank level monitoring system.

c. The incore detection system thimble leak alarm system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required radioactivity monitoring leakage detection system inoperable, operation may continue for up to 7 days provided:

1. Main Coolant System water inventory balance is performed at least once per 24 hours.

2. The other above required leakage detection systems are OPERABLE, and

3. Appropriate grab samples are obtained and analyzed at least once per hour:

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

b. With the containment drain tank level monitoring system inoperable, restore the 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. With the incore detection system thimble leak alarm system inoperable, restore the leak alarm system to OPERABLE status within 7 days or close all thimble isolation valves; restore the leak alarm system to OPERABLE status within 31 days or be in at least HOT STANDBY within the the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

b. Containment drain tank level monitoring system-performance of CHANNEL FUNCTIONAL TEST at least once per 31 days and CHANNEL CALIBRATION TEST at least once per 18 months.

c. Incore detection system thimble leak alarm system-performance of CHANNEL FUNCTIONAL TEST at least once per 31 days.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Main 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 Main Coolant System,

d. 4 GPM IDENTIFIED LEAKAGE from the Main Coolant System, and

e A maximum of two leaking incore detection system thimbles which are valved off and not plugged, when in MODE 1.

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 Main 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.

c. With > 6 gpm IDENTIFIED LEAKAGE from the Main Coolant System, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

d. With no Main Coolant System water inventory balance within the previous 24 hours, and with the containment atmosphere particulate radioactivity monitor indicating an unexplained increase of:

1. $\geq$ 30 CPS, immediately initiate an investigation within the containment vessel to locate the source of the high radioactivity level, and

LIMITING CONDITION FOR OPERATION (Continued)

2. $\geq$ 45 CPS, be in at least HOT STANDBY within the next 6 hours.

e. With more than two leaking incore detection system thimbles which are valved off but not plugged when in MODE 1, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.2 Main Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours except when a Main Coolant System water inventory has not been performed within the previous 24 hours, then monitor the containment atmosphere particulate radioactivity monitor at least once per hour.

b. Monitoring the containment drain tank monitoring system at least once per 12 hours.

c. Performance of a Main Coolant System water inventory balance at least once per 24 hours during steady state operation,

d. Monitoring the reactor head flange leakoff system at least once per 24 hours, and

e. Visual inspection of the accessible portions of the Main Coolant System in the containment vessel at least 2 times per week, with a minimum interval of 72 hours between inspections.

CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4

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

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

At all other times

With the concentration of either chloride or fluoride in the Main 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 Main Coolant System; determine that the Main 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

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

TABLE 3.4-1

MAIN 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 4.4-1

MAIN COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	MINIMUM ANALYSIS FREQUENCIES	MAXIMUM TIME BETWEEN ANALYSES
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 $T_{avg} \leq 250°F$

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

b. $\leq 100/\bar{E}$ μCi/gram.

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

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 514°F within 6 hours.

c. With the specific activity of the primary coolant > $100/\bar{E}$ μCi/gram, be in HOT STANDBY with T_{avg} < 514°F within 6 hours.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4a of Table 4.4-2 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.4. This report shall contain the results of the specific activity analyses together with the following information:

*With $T_{avg} \geq$ 514°F.

LIMITING CONDITION FOR OPERATION (Continued)

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

2. Fuel burnup by core region,

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

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

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

SURVEILLANCE REQUIREMENTS

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

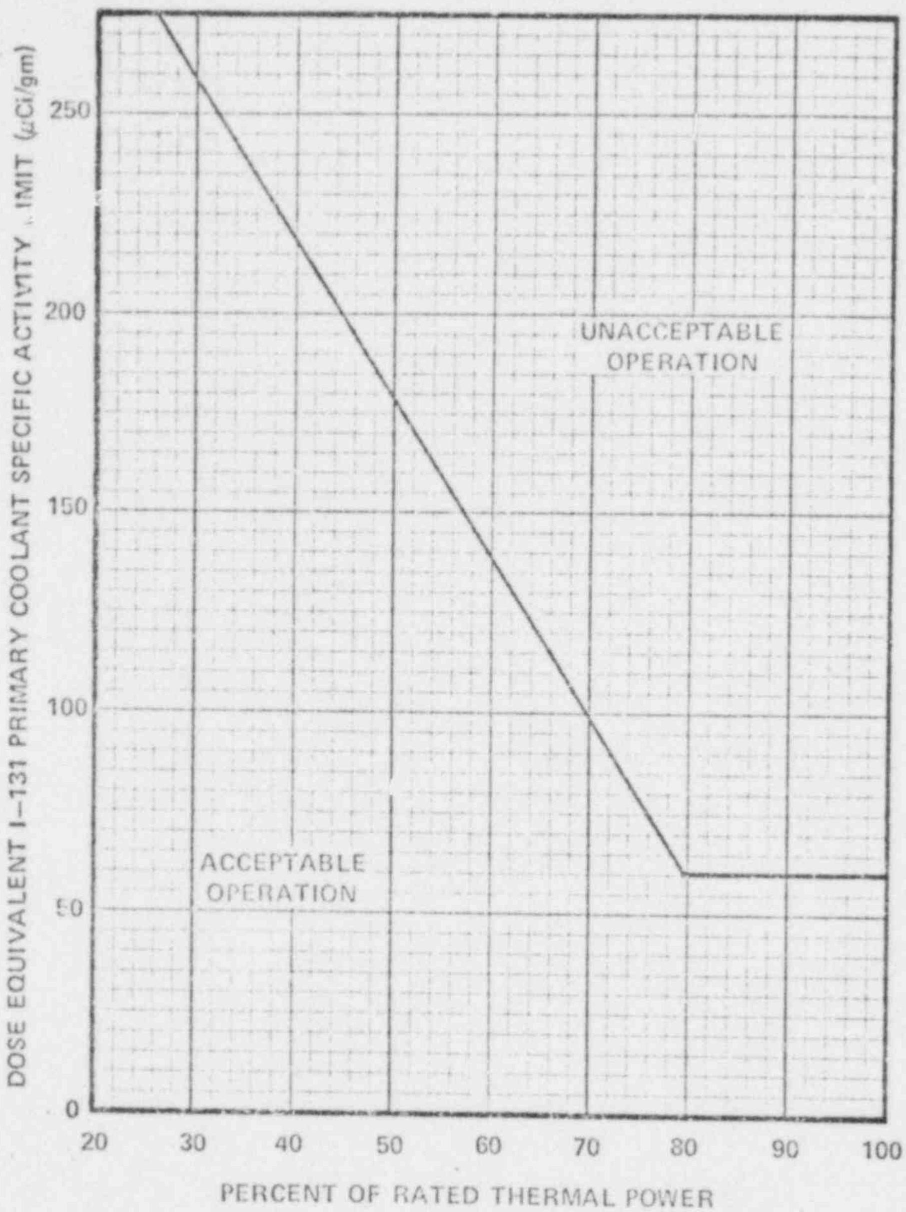
TABLE 4.4-2

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SAMPLING AND ANALYSIS 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 µCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

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

*After at least 2 EFPD and 20 days since any shutdown longer than 48 hours.



DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram DOSE EQUIVALENT I-131.

FIGURE 3.4-1

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

MAIN COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Main Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 3.4-4 and 3.4-5 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup or cooldown of 50°F in any one hour period,

b. A stop at a nominal 250°F for at least two hours during heatup until the reactor pressure vessel and the main coolant are in an isothermal condition, if heatup started with either the vessel or coolant less than 150°F, and

c. Reactor criticality prohibited when the reactor coolant is less than the minimum permissible temperature for the inservice system hydrostatic pressure test as shown by X on Figure 3.4-4 or less than 40°F above that temperature required by Figure 3.4.2 during heatup.

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

APPLICABILITY: At all times.*

ACTION:

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

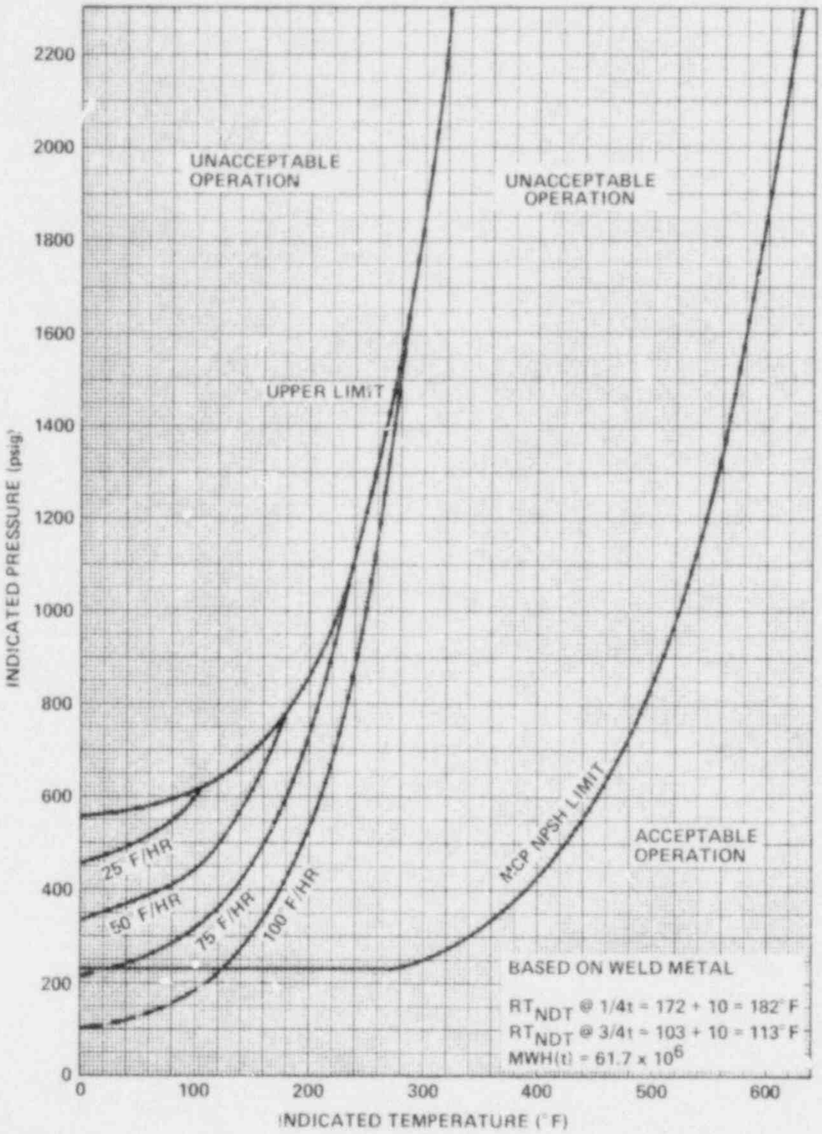
*See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.8.1

a. The Main 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.

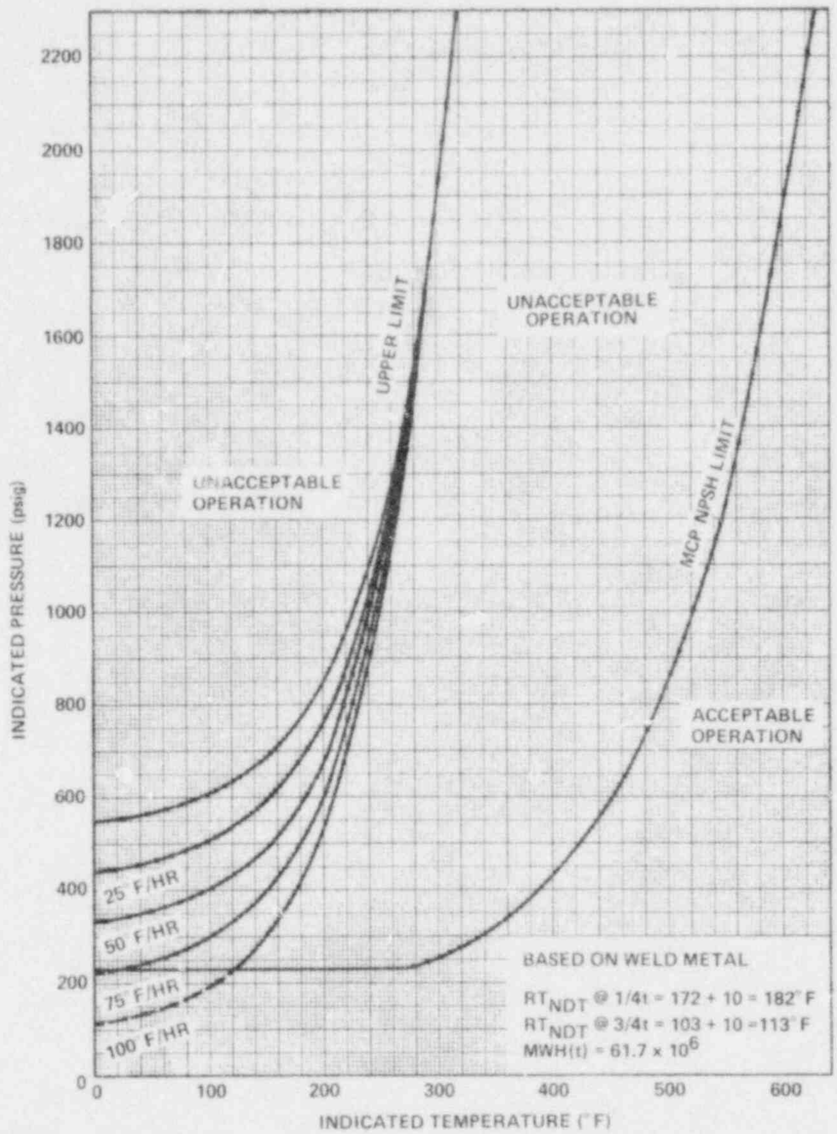
b. The Main Coolant System temperature and pressure shall be determined to be within limits within 15 minutes prior to achieving reactor criticality.



Allowable Pressure-Temperature Limitations
During Heatup

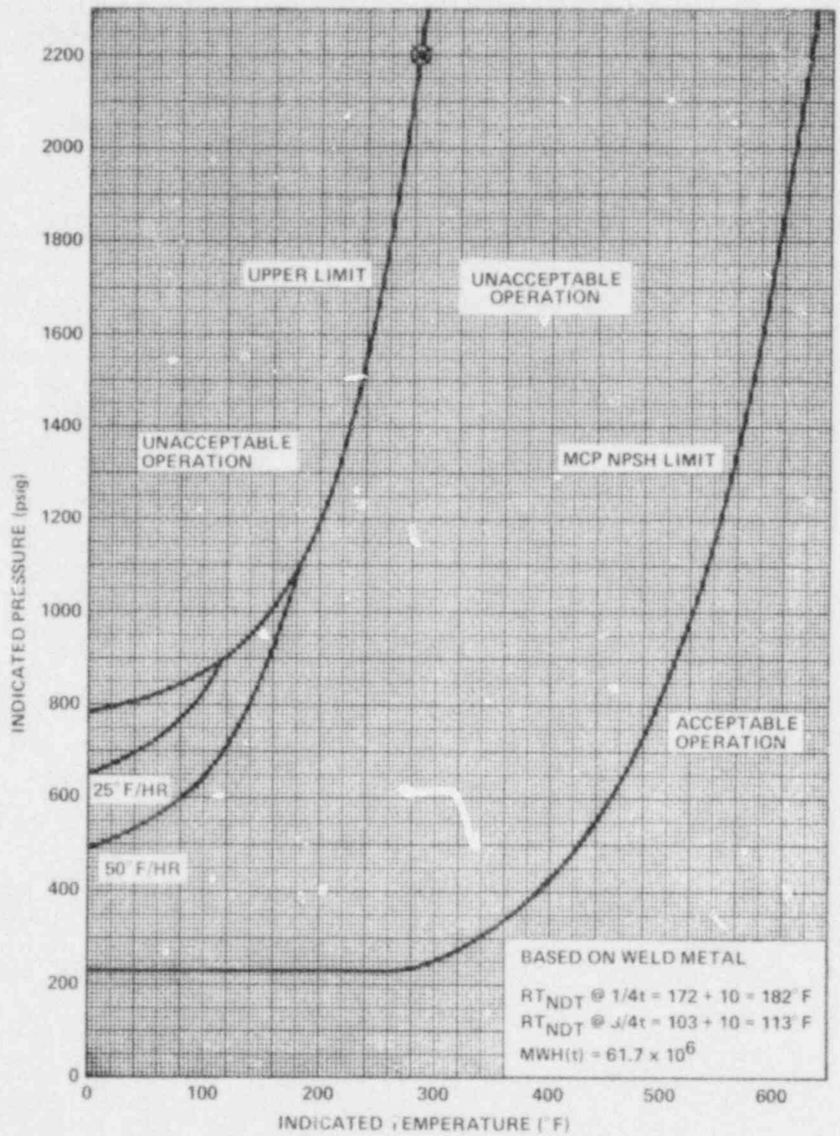
FIGURE 3.4-2

POOR ORIGINAL



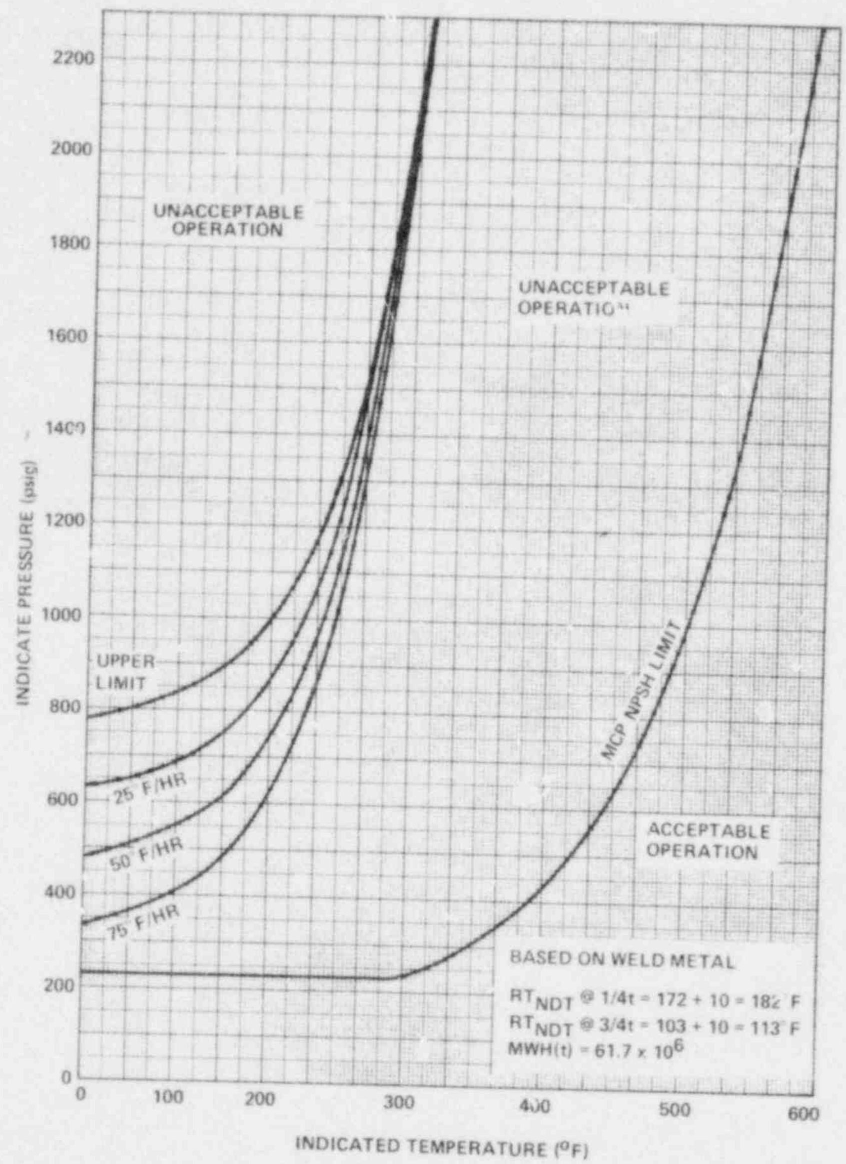
Allowable Pressure-Temperature Limitations
During Cooldown

FIGURE 3.4-3



Allowable Pressure-Temperature Limitations
During Heatup for Inservice Test

FIGURE 3.4-4



Allowable Pressure-Temperature Limitations
During Cooldown for Inservice Test

FIGURE 3.4-5

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be:

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

b. Within 225°F of the Main Coolant System temperature, and

c. Greater than 70°F whenever pressurizer pressure exceeds 500 psig.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature outside 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

4.4.8.2 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 ste-dy state operation.

STRUCTURAL INTEGRITY

CLASS I COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of Main Coolant System components (except steam generator tubes) identified in Table 4.4-3 as Class 1 components shall be maintained at a level consistent with the acceptance criteria in Specifications 4.4.9.1, 4.4.9.2, 4.4.9.3 and 4.4.9.4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of any of the above components not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component prior to increasing the Main Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.1 The following inspection program shall be performed during shutdown:

a. Inservice Inspections The structural integrity of the Class 1 components shall be demonstrated by verifying their acceptability when inspected per the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1970 Edition, and Addenda through Winter 1970, as outlined by the inspection program shown in Table 4.4-3.

   An initial report of any abnormal degradation of the structural integrity of the Safety Class 1 components detected during the

SURVEILLANCE REQUIREMENTS (Continued)

above required inspections shall be made within 10 days after detection and the detailed report shall be submitted pursuant to Specification 6.9.4 within 90 days after completion of the surveillance requirements of this specification.

The Inservice Inspection Program shall be reviewed every 5 years to assure that the equipment, techniques and procedures being utilized are current and applicable. The results of these reviews shall be reported in Special Reports to the Commission pursuant to Specification 6.9.6 within 90 days of completion.

b. Inspections Following Repairs or Replacements The structural integrity of the Main Coolant System shall be demonstrated after completion of all repairs and/or replacements to the system by verifying the repairs and/or replacements meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1970 Edition, and Addenda through Winter 1970. When repairs and/or replacements are made which involve new strength welds on components greater than 2 inch diameter, the new welds shall receive a surface and 100 percent volumetric examination and meet applicable code requirements. When repairs and/or replacements are made which involve new strength welds on components 2 inch diameter or smaller, the new welds shall receive a surface examination and meet applicable code requirements.

c. Inspections Following System Opening The structural integrity of the Main Coolant System shall be demonstrated after each closing by performing a leak test, with the system pressurized to at least 2200 psig, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1970 Edition, and Addenda through Winter 1970, and the Pressure/Temperature limits of Specification 3.4.8.1.

4.4.9.2 The following inspection program shall be performed at least once per 18 months during shutdown on at least one shroud tube per quadrant .

a. Inspect the integrity of the bolts and locking devices in the lower flange at the bottom of the shroud tubes.

b. Inspect the interface between the shroud tube lower flange and the tie plate for separation.

c. Inspect the interface between the shroud tube upper flange and the top shroud tube support plate for separation

d. Inspect the interface between the top shroud tube support plate and the lower core support plate for separation.

e. Inspect for abnormalities one of each of the types of bolts per quadrant.

4.4.9.3 The pressurizer interior shall be inspected at least once per 18 months during shutdown using the best available techniques to determine if any change has occurred in the cladding cracks that exist and whether any further cracking of the cladding has taken place.

4.4.9.4 The 2" charging line between CH-MOV-524 and CH-611A shall be dye penetrant tested at least once per 18 months during shutdown.

TABLE 4.4-3

INSERVICE INSPECTION PROGRAM - CLASS I COMPONENTS

Section XI Examination Category (1)	Components and Parts to be Examined	Methods	Percent of Welds or Components to be Examined
	Reactor Vessel and Closure Head		
C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	100%
E-2	Head penetrations	Visual	25%
G-1	Closure studs and nuts	Volumetric and Visual or Surface	100%
G-1	Ligaments between threaded stud holes	Volumetric	100%
G-1	Closure washers, bushings	Visual	100%
G-2	Pressure retaining bolting	Visual	100%
I-1	Closure head cladding	Visual and Surface or Volumetric	6 Patches (36 sq. inches each)
	Pressurizer		
B	Longitudinal and circumferential welds	Visual and Volumetric	10% of Longitudinal 5% of Circumferential
D	Nozzle-to-vessel welds	Visual and Volumetric	100%
E-1	Heater nozzle to shell welds	Visual and Volumetric	25%

TABLE 4.4-3 (Continued)

INSERVICE INSPECTION PROGRAM - CLASS I COMPONENTS

Section XI Examination Category (1)	Components and Parts to be Examined	Methods	Percent of Welds or Components to be Examined
	Pressurizer (cont'd)		
G-2	Pressure retaining bolting	Visual	100%
H	Integrally welded vessel supports	Visual and Volumetric	50%
I-2	Vessel cladding	Visual	1 Patch (36 sq. inches)
	Steam Generators		
B	Tube sheet-to-head weld on the primary side	Visual	5% of circumferential weld each unit
D	Primary nozzle-to-vessel head welds	Visual	100%
G-2	Pressure retaining bolting	Visual	100%
I-2	Vessel cladding	Visual	1 Patch (36 sq. inches each unit if opened for maintenance)
	Piping Pressure Boundary		
G-2	Pressure retaining bolting	Visual	100%
J	Circumferential pipe welds	Visual and Volumetric	4 loops - 25% each
K-2	Piping Support and hanger	Visual	100%

TABLE 4.4-3 (Continued)

INSERVICE INSPECTION PROGRAM - CLASS I COMPONENTS

Section XI Examination Category (1)	Components and Parts to be Examined	Methods	Percent of Welds or Components to be Examined
	Pump Pressure Boundary		
G-2	Pressure retaining bolting	Visual	100%
K-2	Supports and hangers	Visual	100%
L-2	Pump casing	Visual	One pump (if opened for maintenance)
	Valve Pressure Boundary		
G-2	Pressure retaining bolting	Visual	100%
K-2	Supports and hangers	Visual	100%
M-2	Valve bodies	Visual	100% of one valve of each class (if opened for maintenance)

(1) Those examination categories not applicable or accessible are not listed.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATOR

LIMITING CONDITION FOR OPERATION

3.5.1 The low pressure safety injection accumulator shall be OPERABLE with:

a. The isolation valves open,

b. A minimum useable volume of 700 cubic feet of borated water, equivalent to an indicated level of 87%.

c. A minimum boron concentration of 2200 PPM, and

d. A nitrogen cover-pressure of between 410 and 435 psig.

e. The nitrogen supply system with two supply pressure regulating valves and at least:

1. Ten 48 cubic foot nitrogen bottles $\geq$ 1420 psig, or

2. Eleven 48 cubic foot nitrogen bottles $\geq$ 1305 psig, or

3. Twelve 48 cubic foot nitrogen bottles $\geq$ 1210 psig.

f. Two OPERABLE low level venting systems.

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

ACTION:

a. With the accumulator inoperable, except as a result of a closed isolation valve or as a result of one inoperable supply pressure regulating valve or one inoperable low level venting system, restore the inoperable accumulator to OPERABLE status within 15 minutes or be in at least HOT SHUTDOWN with main coolant pressure < 1000 psig within the next 8 hours.

b. With the accumulator inoperable due to one isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in at least HOT SHUTDOWN with main coolant pressure < 1000 psig within the next 8 hours.

*Main coolant pressure $\geq$ 1000 psig.

LIMITING CONDITION FOR OPERATION (Continued)

c. With one inoperable supply pressure regulating valve or with one inoperable low level venting system, restore the inoperable regulating valve or venting system to OPERABLE status within 8 hours or be in at least HOT STANDBY within one hour and be in at least HOT SHUTDOWN with main coolant pressure < 1000 psig within the following 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 The accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2. Verifying that each accumulator isolation valve is open.

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

c. At least once per 31 days by verifying:

1. Power to the isolation valve SI-MOV-1 operator is disconnected by removal of the breaker from the motor control center.

2. The nitrogen supply bottle pressures.

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

1. Verifying OPERABILITY of the nitrogen supply system by observing operation of each regulating valve while venting accumulator nitrogen pressure.

2. Verifying OPERABILITY of each low level venting system when the level switch column water level is lowered.

ECCS SUBSYSTEMS

LIMITING CONDITION FOR OPERATION

3.5.2 The ECCS subsystems shall be OPERABLE with:

a. Three OPERABLE independent ECCS safety injection subsystems with each subsystem comprised of:

1. One OPERABLE high pressure safety injection pump,

2. One OPERABLE low pressure safety injection pump,

3. An OPERABLE flow path capable of taking suction from the safety injection tank on a safety injection signal.

b. An OPERABLE ECCS recirculation subsystem with two OPERABLE purification pumps and an OPERABLE flow path capable of taking suction from the containment sump and recirculating to the safety injection header.

c. An OPERABLE ECCS long term hot leg injection subsystem with at least two OPERABLE fixed speed charging pumps and an OPERABLE flow path capable of taking suction from the ECCS recirculation subsystem and discharging to the Main Coolant System #4 hot leg.

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

ACTION:

a. With one ECCS safety injection subsystem or one purification pump or one fixed speed charging pump inoperable, restore the inoperable subsystem or pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN with Main Coolant pressure $<$ 1000 psig within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Main Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.6 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The provisions of Specification 3.0.3 are not applicable.

*Main coolant pressure $\geq$ 1000 psig.

SURVELLANCE REQUIREMENTS

4.5.2 Each ECCS safety injection subsystem, the recirculation subsystem, and the long term hot leg injection subsystem shall be demonstrated OPERABLE:

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

1. Verifying that each high pressure safety injection pump:

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

b) Develops a discharge pressure of $\geq$ 850 psig on recirculation flow.

c) Operates for at least 15 minutes.

2. Verifying that each low pressure safety injection pump:

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

b) Develops a discharge pressure of $\geq$ 250 psig on recirculation flow through CS-MOV-532.

c) Operates for at least 15 minutes.

3. Verifying that each purification pump:

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

b) Develops a discharge pressure $\geq$ 80 psig in normal purification system service.

c) Operates for at least 15 minutes.

4. Verifying that each fixed speed charging pump:

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

b) Develops a discharge pressure of $\geq$ 30 psig above Main Coolant System pressure at a flow $\geq$ 26 gpm, and

c) Operates for at least 15 minutes.

SURVEILLANCE REQUIREMENTS (continued)

b. At least once per 31 days by:

1. Verifying that the following valves are in the indicated positions with power to the valve operators removed by opening at least two breakers in series:

Valve Number	Valve Function	Valve Position
a. CH-MOV-522	a. Charging Header/LPSI Isolation	a. Closed
b. CH-MOV-523	b.&c. Charging Header/Loop 4 Hot Leg Injection Long-Term Recirculation	b. Open
c. CH-MOV-524		c. Open

2. Cycling each of the following valves, except PU-V-651, through at least one complete cycle of full travel and verifying that the valves are in the indicated normal positions:

Valve No.	Valve Position
PU-MOV-541	Open
PU-MOV-542	Open
PU-MOV-543	Closed
PU-MOV-544	Closed
PU-MOV-545	Open
PU-MOV-546	Open
PU-MOV-547	Closed
PU-MOV-548	Closed
PU-V-651	Open

3. Verifying that the following valves are in their normally opened positions with power to the valve operators removed by removal of the circuit breaker from the motor control center:

Valve Number	Valve Function
a. SI-MOV-4	a. LPSI Pump Cross Over to HPSI Pump
b. SI-MOV-22	b. SI Header Isolation to Cold Leg
c. SI-MOV-23	c. SI Header Isolation to Cold Leg
d. SI-MOV-24	d. SI Header Isolation to Cold Leg
e. SI-MOV-25	e. SI Header Isolation to Cold Leg
f. SI-MOV-46	f. HPSI Flow Control
g. SI-MOV-49	g. HPSI Recirculation Test Line

SURVEILLANCE REQUIREMENTS (continued)

4. Verifying that power to the valve operators is removed by disconnecting the power cables as they leave the motor starters:

Valve Number	Valve Function
a. CS-MOV-533	a. LPSI Pump Header Isolation
b. CS-MOV-535	b. LPSI Pump Header Isolation
c. CS-MOV-536	c. SI Header Isolation to Cold Leg
d. CS-MOV-537	d. SI Header Isolation to Cold Leg
e. CS-MOV-538	e. SI Header Isolation to Cold Leg
f. CS-MOV-539	f. SI Header Isolation to Cold Leg
g. MC-MOV-301	g. MCS Loop Isolation
h. MC-MOV-302	h. MCS Loop Isolation
i. MC-MOV-309	i. MCS Loop Isolation
j. MC-MOV-310	j. MCS Loop Isolation
k. MC-MOV-318	k. MCS Loop Isolation
l. MC-MOV-319	l. MCS Loop Isolation
m. MC-MOV-325	m. MCS Loop Isolation
n. MC-MOV-326	n. MCS Loop Isolation

5. Verifying that the following valve is in its normally closed position with power to the valve operator removed by disconnecting the power cables as they leave the motor starter:

Valve Number	Valve Function
a. CS-MOV-532	a. LPSI Recirculation Line

Note: This valve may be opened for $\leq$ 30 minutes once per week for safety injection tank mixing or low pressure safety injection pump testing after restoring power to the valve operator. Insure that power to the valve operator is properly removed after closing the valve.

6. Verifying that each ECCS safety injection subsystem is aligned to receive electrical power from an OPERABLE emergency bus.

SURVEILLANCE REQUIREMENTS (Continued)

7. Verifying that ECCS recirculation subsystem each pair of redundant valves and purification pumps are aligned to receive electrical power from separate OPERABLE busses.

8. Verifying that each ECCS long term hot leg injection subsystem charging pump is aligned to receive electrical power from an OPERABLE bus.

9. Verifying that the long term hot leg injection flow metering instrument is OPERABLE by observing charging flow rate at least once per 12 hours.

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 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 through at least one complete cycle of full travel.

   2. Verifying that valve CS-MOV-532 actuates to its correct position on a safety injection signal.

SURVEILLANCE REQUIREMENTS (Continued)

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

    a) High pressure safety injection (HPSI) pump

    b) Low pressure safety injection (LPSI) pump

4. Verifying that two high pressure safety injection pumps develop a combined flow $\geq$ 500 gpm. Test every HPSI pump at least once per 36 months.

5. Verifying that two low pressure safety injection pumps develop a combined flow $\geq$ 2180 gpm. Test every LPSI pump at least once per 36 months.

6. Verifying that each charging pump stops automatically upon receipt of a safety injection signal.

7. Verifying that the long term hot leg injection flow metering instrument is OPERABLE by performing a CHANNEL CALIBRATION.

8. Verifying that each valve listed in Specification 4.5.2.b.4 is in its normally open position.

LIMITING CONDITION FOR OPERATION

3.5.3 The ECCS subsystems shall be OPERABLE with:

a. As a minimum, one OPERABLE ECCS safety injection subsystem comprised of the following:

1. One OPERABLE high pressure safety injection pump,

2. One OPERABLE low pressure safety injection pump, and

3.. An OPERABLE flow path capable of taking suction from the safety injection tank.

b. An OPERABLE ECCS recirculation subsystem with at least one OPERABLE purification pump and an OPERABLE flow path capable of taking suction from the containment sump and recirculating to the safety injection header.

c. An OPERABLE ECCS long term hot leg injection subsystem with at least one OPERABLE fixed speed charging pump and an OPERABLE flow path capable of taking suction from the ECCS recirculation subsystem and discharging to the Main Coolant System hot legs.

APPLICABILITY: MODE 4.*

ACTION:

a. With the ECCS safety injection subsystem, the recirculation subsystem, or the long term hot leg injection subsystem inoperable, restore all subsystems to OPERABLE status within 1 hour or be in COLD SHUTDOWN with Main Coolant pressure < 1000 psig within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Main Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.6 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

*Main coolant pressure < 1000 psig.

ECCS SUBSYSTEMS

SURVEILLANCE REQUIREMENTS

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

SAFETY INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The safety injection tank (SIT) shall be OPERABLE with:

a. A minimum contained volume of 117,000 gallons of borated water, equivalent to a level of $\geq$ 25.5 feet.

b. A minimum boron concentration of 2200 ppm, and

c. A minimum water temperature of 40°F.

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

ACTION:

With the safety injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN with Main Coolant pressure < 1000 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The SIT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

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

*Main coolant pressure $\geq$ 1000 psig.

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, 4 and *

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 with Main Coolant pressure < 300 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

   1. All penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.2, and

   2. All equipment hatches are closed and sealed.

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

c. By verifying that the containment continuous leak monitoring system is OPERABLE per Specification 3.6.1.7.

d. By calculating and plotting the containment air mass as determined by the containment continuous leak monitoring system at least once per 24 hours.

*Main coolant pressure $\geq$ 300 PSIG

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. $\leq L_a$, 0.20 percent by weight of the containment air per 24 hours at P_a, 31.6 psig, or

2. $\leq L_t$, 0.1123 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 16 psig.

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

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

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.70 L_a or 0.70 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Main Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a, 31.6 psig, or at P_t, 16 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10 year plant inservice inspection

*Main Coolant pressure $\geq$ 300 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, 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 either 0.70 L_a or 0.70 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.70 L_a or 0.70 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which requires the metered mass of gas injected into the containment or bled from the containment for the supplemental test to be equivalent to between 50 and 100% of the allowable 24 hour mass loss. The acceptability is demonstrated if the mass change, as measured by the Type A instrumentation, agrees with the mass change as metered by the flow meter to within 25% of the allowable 24 hour mass loss.

d. Type B and C tests shall be conducted with gas at P_a, 31.6 psig, at intervals no greater than 24 months except for tests involving the air lock, the equipment hatch and the emergency hatch.

e. The air lock shall be tested and demonstrated OPERABLE per Specification 4.6.1.3.

f. The equipment and emergency hatch seals and seating surfaces shall be inspected before each hatch closure.

g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to provide a maximum expected error.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, 31.6 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 The containment air lock shall be demonstrated OPERABLE:

a. By verifying that the containment continuous leak monitoring system is OPERABLE per Specification 3.6.1.7.

b. By calculating and plotting the containment atmosphere mass as determined by the containment continuous leak monitoring system at least once per 24 hours.

c. By inspecting the door seals and seating surface after each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours.

d. At least once per 6 months by conducting an overall air lock leakage test at P_a, 31.6 psig, and by verifying that the overall air lock leakage rate is within its limit, and

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

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall not exceed 3.0 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the weighted average of the temperatures of at least eight of the following nine locations and shall be determined at least once per 24 hours:

Location

a. Each main coolant loop - 4

b. Charging floor - 1

c. North and south hemisphere, low - 2

d. North and south hemisphere, fan - 2

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation. An initial report of any abnormal degradation of the containment vessel detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted to the Commission pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

CONTINUOUS LEAK MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 The continuous leak monitoring system shall be OPERABLE with:

a. Containment internal pressure $\geq$ 0.75 psig,

b. At least eight containment temperature detectors,

c. The containment pressure manometer,

d. Two relative humidity detectors.

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

ACTION:

With the continuous leak monitoring system inoperable, restore the system to OPERABLE status within 72 hours or within 24 hours after closing any containment air lock door, whichever is sooner, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN with Main Coolant pressure < 300 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The continuous leak monitoring system shall be demonstrated OPERABLE by:

a. Verifying containment internal pressure to be $\geq$ 0.75 psig at least once per 12 hours.

b. Calibrating the temperature detectors, the manometer, and the relative humidity detectors at least once per 18 months.

*Main coolant pressure $\geq$ 300 psig.

3/4.6.2 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 92 days by cycling each OPERABLE power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

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 cycling test, above, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.2.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

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

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

d. 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 ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
A. AUTOMATIC ISOLATION VALVE			
TV-401A	No. 1 SG Blowdown	Yes	30
TV-401B	No. 2 SG Blowdown	Yes	30
TV-401C	No. 3 SG Blowdown	Yes	30
TV-401D	No. 4 SG Blowdown	Yes	30
TV-408	Containment Cooling Water Return	Yes	30
TV-409	Containment Heater Condensate Return	Yes	30
VD-SOV-301	Air Particulate Monitor-in	Yes	30
VD-SOV-302	Air Particulate Monitor-out	Yes	30
HV-SOV-1	Hydrogen Vent System	Yes	30
HV-SOV-2	Hydrogen Vent System	Yes	30
TV-202	Main Coolant Drain	Yes	30
TV-203	Main Coolant Vent	Yes	30
TV-204	Valve Stem Leakoff	Yes	30
TV-205	Component Cooling Return	No	30
TV-206	Main Coolant Sample	Yes	30
TV-207	Neutren Shield Tank Sample	Yes	30
TV-209	Containment Drain	Yes	30
TV-211	Containment Pressure Sensing	Yes	30
TV-212	Containment Pressure Sensing	Yes	30
TV-213	LP Sample	Yes	30

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
A. AUTOMATIC ISOLATION VALVE (Continued)			
PCV-402*	Steam Dump to Condenser	No	30
TV-404*	Main Steam Drain to Condenser	No	30
TV-405*	Auxiliary Steam to Air Ejector	No	30
TV-406*	Main Steam Drain to Condenser	No	30
TV-410*	Auxiliary Steam to #1 Feedwater Heater	No	30
TV-411*	Atmospheric Steam Dump	Yes	30
______*	Turbine Generator Throttle Valves	Yes	2
B. CHECK VALVES			
SI-V-14*	Safety Injection (HP)	NA	NA
CS-V-621*	Safety Injection (LP)	NA	NA
CH-V-611*	MC Feed to Loop #4	NA	NA
CC-V-667*	Component Cooling to MCP #1	NA	NA
CC-V-663*	Component Cooling to MCP #2	NA	NA
CC-V-671*	Component Cooling to MCP #3	NA	NA
CC-V-675*	Component Cooling to MCP #4	NA	NA
CC-V-649*	Component Cooling to Sample Cooler	NA	NA
CC-V-653*	Component Cooling to Neutron Shield Tank Coolers	NA	NA
CC-V-660*	Neutron Shield tank Fill	NA	NA

*Not subject to Type C tests

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
B.	CHECK VALVES (Continued)			
	SW-V-820*	Service Water to Containment Cooler #1	NA	NA
	SW-V-821*	Service Water to Containment Cooler #2	NA	NA
	SW-V-822*	Service Water to Containment Cooler #3	NA	NA
	SW-V-823*	Service Water to Containment Cooler #4	NA	NA
	HC-V-1199*	Steam Supply to Containment Heaters	NA	NA
C.	Manual Valves			NA
	SC-MOV-551+553*	Shutdown Cooling - In	No	NA
	SC-MOV-552+554*	Shutdown Cooling - Out	No	NA
	CH-MOV-522*	MC Feed to Loop Fill Header	NA	NA
	CS-V-601	Shield Tank Cavity Fill	NA	NA
	CA-V-746*	Containment Air Charge	NA	NA
	HV-V-5*	Containment H2 Vent System	NA	NA
	HV-V-6*	Containment H2 Vent System	NA	NA
	CA-V-688	Containment H2 Vent System Air Supply	NA	NA
	CS-MOV-500	Fuel Chute Lock Valve	No	NA

*Not subject to Type C tests

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
C. Manual Valves (Cont'd)			
CS-CV-215	Fuel Chute Equalizing	NA	NA
CS-CV-216	Fuel Chute Dewatering Pump Discharging	NA	NA
VD-V-752*	Neutron Shield Tank-Outer Test	NA	NA
VD-V-754*	Neutron Shield Tank-Inner Test	NA	NA
BF-V-4-1	Air Purge Inlet	NA	NA
BF-V-4-2	Air Purge Outlet	NA	NA
HC-V-602	Air Purge Bypass	NA	NA
PU-MOV-543	ECCS Recirculation	Yes	50
PU-MOV-544	ECCS Recirculation	Yes	50
BF-CV-1000*	SG#1 Feedwater Regulator	No	30
BF-CV-1100*	SG#2 Feedwater Regulator	No	30
BF-CV-1200*	SG#3 Feedwater Regulator	No	30
BF-CV-1300*	SG#4 Feedwater Regulator	No	30
____ ______	Main Coolant Heise Pressure Gauge	NA	NA

*Not subject to Type C tests

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
D. Other			
18" Bolted Manway**		NA	NA
Demineralized Water Supply (Blank flanged)		NA	NA
Cavity Purification (Blank flanged)		NA	NA
LP Vent Header (Blank flanged)		NA	NA
Personnel Airlock		NA	NA
Electrical Penetrations		NA	NA
Equipment Hatch**		NA	NA
Containment Leg Expansion Joints**		NA	NA
Fuel Chute Expansion Joints**		NA	NA

**Not subject to Type B tests

CONTAINMENT SYSTEMS

3/4.6.3 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZER

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment hydrogen analyzer shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the hydrogen analyzer inoperable, restore the analyzer to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days by performing a CHANNEL CALIBRATION using sample gases containing a nominal:

a. Zero volume percent hydrogen, balance nitrogen, and

b. Five volume percent hydrogen, balance nitrogen.

CONTAINMENT SYSTEMS

HYDROGEN VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.2 A containment hydrogen vent system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the containment hydrogen vent system inoperable, restore the hydrogen vent system to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.2 The hydrogen vent system shall be demonstrated OPERABLE at least once per 31 days by verifying that the purge system operates for at least 15 minutes.

CONTAINMENT SYSTEMS

ATMOSPHERE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.3 The atmosphere recirculation system shall be OPERABLE with three OPERABLE fans.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.3.3 The atmosphere recirculation system shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by:

   1. Verifying that each fan can be started on operator action in the control room, and

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a flow rate of approximately 6000 cfm for each fan.

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 of an unisolated main coolant loop shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 main 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 and Intermediate 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 3 main 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 and Intermediate 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, 1974 Edition, and Addenda through Summer, 1975.

TABLE 3.7-1

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

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	27

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE AND INTERMEDIATE 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 Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	20

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

TABLE 4.7-1

STEAM LINE SAFETY VALVESPER LOOP

VALVE NUMBER	LIFT SETTING ($\pm$ 3%)	ORIFICE SIZE*
a. SV-409 E, F, G or H	935 psig	K
b. SV-409 A, B, C or D	985 psig	K_2
c. SV-409 I, J, K or L	1035 psig	Q

*K = 1.838 square inches
K_2 = 2.545 square inches
Q = 11.05 square inches

EMERGENCY BOILER FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 The emergency boiler feedwater system shall be OPERABLE with the emergency boiler feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the emergency boiler feedwater system inoperable, be in at least HOT STANDBY within one hour and in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The emergency boiler feedwater system shall be demonstrated OPERABLE:

a. At least once per 15 days by:

1. Starting the pump,
2. Verifying that, on recirculation flow, the steam turbine driven pump develops a discharge pressure of $\geq$ 950 psig when the secondary steam pressure is greater than 100 psig,
3. Verifying that the pump operates for at least 15 minutes,
4. Cycling each testable manual valve in the flow path through at least one complete cycle of full travel, and
5. Verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

1. Cycling each manual valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.
2. Verifying that the steam turbine driven pump develops a discharge pressure of $\geq$ 950 psig at a flow of $\geq$ 80 gpm while feeding a steam generator.
3. Cycling each main feed control valve manually through at least one complete cycle of full travel.

PLANT SYSTEMS

PRIMARY AND DEMINERALIZED WATER STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.7.1.3 The primary (PWST) and demineralized (DWST) water storage tanks shall be OPERABLE with a minimum combined contained volume of 85,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION.

With the PWST and DWST inoperable, be in at least HOT STANDBY within one hour and in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The PWST and DWST shall be demonstrated OPERABLE at least once per 12 hours by verifying the water volume is within its limits.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ 0.20 μCi/ml I-131.

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

ACTION:

With the specific activity of the secondary coolant system > 0.20 μCi/ml I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for 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.

TURBINE GENERATOR THROTTLE AND CONTROL VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each turbine generator throttle and control valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one turbine generator throttle or control 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 turbine generator throttle or control valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed after:

a. The inoperable valve is restored to OPERABLE status, or

b. The inoperable valve is maintained closed;

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

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each turbine generator throttle and control valve that is open shall be demonstrated OPERABLE by:

a. Cycling each valve through at least one complete cycle of full travel at least once per month, and

b. Verifying full closure within 2 seconds on any closure actuation signal whenever shutdown longer than 24 hours, if not performed in the previous 92 days.

PLANT SYSTEMS

SECONDARY WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the secondary water chemistry parameter outside of its limit, restore the parameter to within its limit within 3 days; otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 96 hours.

b. The provisions of Specification 3.0.4 are not applicable for up to 72 hours provided:

   1. The chloride concentration is < 2.0 ppm,

   2. The THERMAL POWER is < 30% of RATED THERMAL POWER, and

   3. Corrective measures have been implemented to restore chloride concentrations to $\leq$ 0.5 ppm.

SURVEILLANCE REQUIREMENTS

4.7.1.6 The secondary water chemistry shall be determined to be within the limit by analysis of the parameter at the frequency specified in Table 4.7-3.

TABLE 3.7-3

SECONDARY WATER CHEMISTRY LIMITS

Water Sample Location	Parameters	Limit
Steam generator blowdown	Chloride	$\leq$ 0.5 ppm

TABLE 4.7-3

SECONDARY WATER CHEMISTRY SURVEILLANCE REQUIREMENT

Water sample Location	Parameter	Minimum Analysis Frequency	Maximum Time Between Analyses
Steam generator blow down	Chloride	3 times per 7 days*	72 hours

*With chloride concentration > 0.5 ppm, sampling and analysis shall be performed at least once per 24 hours.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The steam generator temperature and pressure shall be limited as required below:

a. The temperature difference across the tube sheet shall not exceed 100°F.

b. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

c. The maximum heatup and cooldown rate for the steam generators shall not exceed 100°F in any one hour period.

d. The primary side of the steam generator must not be pressurized above 500 psig unless the temperature of the Main Coolant System is above 70°F.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure or temperature to within the limits 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

4.7.2.1 The pressure in the primary and secondary sides of the steam generator shall be determined to be < 500 psig or < 200 psig, respectively, at least once per hour when the temperature of either the primary or secondary coolant in the steam generator is < 70°F.

4.7.2.2 The steam generator shall be determined to be within limits at least once per 30 minutes during heatup, cooldown, and inservice leak and hydrostatic testing operations.

3/4.7.3 PRIMARY PUMP SEAL WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The primary pump seal water system shall be OPERABLE with:

a. The seal water tank level between 32" and 47" and a tank nitrogen cover pressure $\geq$ 105 psig.

b. Two OPERABLE seal tank fill pumps.

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

ACTION:

a. With only one seal tank fill pump OPERABLE, restore at least two pumps to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the seal water tank inoperable, restore the seal water tank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with Main Coolant Pressure < 1000 psig within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The primary pump seal water system shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying seal water tank level and cover pressure.

b. At least once per 31 days by:

1. Starting (unless already operating) each seal tank fill pump from the control room.
2. Verifying that each pump operates for at least 15 minutes.
3. Verifying that each pump is aligned to receive electrical power from an OPERABLE bus.
4. Verifying that each isolation valve in the flow path between the primary water storage tank and the purification pumps seals is open.

*With main coolant pressure > 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

5. Verifying that each valve servicing the purification pumps that is not locked, sealed, or otherwise secured in position, is in its correct position.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The service water system shall be OPERABLE with at least two OPERABLE service water pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water pump OPERABLE, restore at least two pumps 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 The service water system shall be demonstrated OPERABLE:

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

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

   2. Verifying that each above required pump develops at least a discharge pressure of 80 psig at shutoff flow conditions.

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

   4. Verifying that each above required pump is aligned to receive electrical power from separate OPERABLE busses.

b. At least once per 31 days by:

   1. Cycling power operated valve SW-MOV-603 through at least one complete cycle of full travel.

   2. Verifying that each valve servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that the standby service water pump starts automatically on a discharge header pressure $\leq$ 40 psig.

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM EMERGENCY SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.5 The control room ventilation system emergency shutdown shall be OPERABLE.

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

ACTION:

With the control room ventilation system emergency shutdown inoperable, restore the system to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The control room ventilation system emergency shutdown shall be demonstrated OPERABLE at least once per 31 days by verifying that the control room ventilation system can be shutdown manually from the control room.

3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

 1. Either decontaminated and repaired, or

 2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.6.1 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.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

2. In any form other than gas.

b. Stored sources not in use - Each sealed source 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 and fission detector shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.7.6.3 Reports - A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.6 annually if sealed source leakage tests reveal the presence of greater than .005 microcuries of removable contamination.

3/4.7.7 WASTE EFFLUENTS

RADIOACTIVE SOLID WASTE

LIMITING CONDITION FOR OPERATION

3.7.7 Radioactive solid waste shall not be disposed of at the site.

APPLICABILITY: At all times

ACTION:

With any radioactive solid waste disposed of at the site, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.6 within 90 days describing the circumstances of the disposal and outlining plans for removal of the waste. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7 Not applicable.

RADIOACTIVE LIQUID WASTE

LIMITING CONDITION FOR OPERATION

3.7.7.2 Radioactive liquid waste shall be discharged only when the activity of the waste, together with the activity being released from steam generator blowdown, is less than the maximum permissible concentration established in 10 CFR Part 20.

APPLICABILITY: At all times

ACTION:

With discharge of radioactive liquid waste in excess of the limits, immediately suspend the discharge. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.2.1 Radioactive liquid waste shall be determined to be within the above limits by radioactivity analysis prior to discharge.

4.7.7.2.2 Steam generator blowdown radioactivity shall be analyzed at least once every 7 days.

PLANT SYSTEMS

RADIOACTIVE GASEOUS WASTE

LIMITING CONDITION FOR OPERATION

3.7.7.3 Concentration of radioactive gaseous wastes discharged, as determined at the point of discharge from the primary vent stack and averaged over a period not exceeding one year, shall not exceed 1000 times the limits specified in 10 CFR Part 20, Appendix B, Table II, except that, for isotopes of Iodines and particulates with half-lives > 8 days, the MPC values of Table II shall be reduced by a factor of 1/700.

APPLICABILITY: At all times

ACTION:

With discharge of radioactive gaseous waste in excess of limits, immediately suspend the discharge. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.3 Radioactive gaseous waste discharge concentrations shall be determined to be within the limits, by continuously monitoring the primary vent stack effluent and analyzing for tritium, particulates, iodines, and fission, and activation gases at least once per 31 days.

3/4.7.8 ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The environmental monitoring program shall be performed in accordance with Table 4.7-4.

APPLICABILITY: At all times

ACTION:

With the sampling and analysis program specified in Table 3.7-4 not satisfied, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.5 within 90 days describing the circumstances of the violation and outlining plans to prevent re-occurrence of the violation. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 The environmental monitoring samples shall be collected and analyzed in accordance with the requirements of Table 4.7-4.

Table 4.7-4

ENVIRONMENTAL MONITORING PROGRAM

Environs Parameter	Number of Sample Points	Sample Frequency	Extent of Analyses
Air	5	Continuous	Analyze weekly for gross α* & β; I^{131}. Composit of filters analyzed monthly for quantitative γ spectrum; and Sr90
Water	7	Monthly	Gross α* & β; tritium; quantitive spectrum; & Sr90.
Food	2	Annually	Maple syrup for quantitative γ spectrum; and Sr90.
Radiation - External direct	22	Monthly	Integrated dose(s) at boundary of Restricted Area and in the surrounding environs
Soil	9	At least once per 92 days, April through December	Gross α* & β; quantitative γ spectrum Sr90.
Vegetation	9	At least once per 92 days, April through December	Gross α* & β; quantitative γ spectrum Sr90.
River sediments	13	At least once per 92 days, April through December	Gross α* & ∞; quantitive γ spectrum; Sr90.
Milk	2	Monthly	I-131 quantitative γ spectrum; Sr90.

Table 4.7-4 (cont'd)

ENVIRONMENTAL MONITORING PROGRAM (cont'd)

Environs Parameter	Number of Sample Points	Sample Frequency	Extent of Analyses
Fish	3	At least once per 92 days, April through December	Tritium, quantitative γ spectrum; Sr90.
Aquatic Plants	3	At least once per 92 days, April through December	Quantitative γ spectrum; Sr90.

NOTE: *Gross alpha analyses are to be made in cases where alpha emitters are found to be present in releases

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators:

1. Each with separate day fuel tank containing a minimum of 210 gallons of fuel, equivalent to a 3/4 full tank, and

2. With a fuel storage system containing a minimum of 8000 gallons of fuel, equivalent to a tank level of 4'4".

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 three diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit inoperable, reduce THERMAL POWER to $\leq$ 75% of RATED THERMAL POWER within 2 hours.

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

LIMITING CONDITION FOR OPERATION (Continued)

hours. Restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.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 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With less than two of the above required diesel generators OPERABLE demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least two of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore three diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system 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 during shutdown by manually transferring unit power supply from one independent circuit to the second independent circuit

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying the fuel level in the day fuel tank,

2. Verifying gravity flow from the storage system to the day tanks,

3. Verifying the diesel starts from ambient condition,

4. Verifying the generator is synchronized, loaded to $\geq$ 200 KW, and operates for $\geq$ 2 hours, and

5. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days by:

1. Verifying the fuel level in the fuel storage tank,

2. 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,

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

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of $\geq$ 275 Amperes 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.

SURVEILLANCE REQUIREMENTS (Continued)

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 $\geq$ 5 minutes while its generator is loaded with the emergency loads.

4. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 400 KW.

5. Verifying that the high pressure safety injection pump breakers on each emergency bus delay 10 $\pm$ 3 seconds in closing on the bus.

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 fuel tank containing a minimum of 210 gallons of fuel, equivalent to a 3/4 full tank, and

   2. A fuel storage system containing a minimum of 4000 gallons of fuel, equivalent to a tank level of 2'2".

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.4.

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:

a. Redundant busses:

1. 2400 volt bus #1*
2. 2400 volt bus #2
3. 2400 volt bus #3
4. 480 volt bus #4-1*
5. 480 volt bus #5-2
6. 480 volt bus #6-3
7. 480 volt Emergency Bus #1
8. 480 volt Emergency Bus #2
9. 480 volt Emergency Bus #3

b. Non-Redundant Busses:

1. 480 volt Emergency MCC #1
2. 480 volt Emergency MCC #2
3. 120 volt Vital Bus.

APPLICABILITY: Modes 1, 2, 3 and 4

ACTION:

a. With less than the above complement of redundant A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be

* One tie breaker may be closed $\leq$ 30 MWe.

LIMITING CONDITION FOR OPERATION (Continued)

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

b. With any non-redundant bus inoperable, be in at least HOT STANDBY within one hour and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 Each specified A.C. bus and emergency MCC shall be determined OPERABLE and energized from A.C. sources other than the diesel generators with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.1.2 Emergency MCC #1 and #2 busses shall be determined OPERABLE by verifying:

a. At least once per 31 days that the alternate power supply is disconnected by racking out and locking out the breakers, and

b. At least once per 18 months that the interlocks preventing the normal and alternate breakers from simultaneously being in the closed position are OPERABLE.

4.8.2.1.3 The 120 volt vital bus shall be determined OPERABLE at least once per 6 months by verifying the OPERABILITY of the following alternate power sources:

a. 480 volt Emergency MCC #1.

b. 480 volt MCC #1 bus 2.

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.

a. 1 - 2400 volt bus #2 or #3

b. 2 - 480 volt buses #4-1 and #5-2

c. 1 - 480 volt Emergency Buses #1, 2 or 3

d. 2 - 480 volt buses, Emergency MCC #1 and Emergency MCC #2

e. 1 - 120 volt Vital Bus

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. distribution system shall be energized and OPERABLE with tie breakers between t[illegible] open:

TRAIN "1" consisting of 125[illegible] distribution switchboard No. 1, 125-volt D.C. [illegible]nk No. 1 and a full capacity charger.

TRAIN "2" consisting of 12[illegible]. distribution switchboard No. 2, 125-volt D.C. battery bank No. 2 and a full capacity charger.

TRAIN "3" consisting of 125-volt D.C. distribution switchboards No. 3 and 3A, 125-volt D.C. battery bank No. 3 and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 125-volt D.C. distribution switchboard inoperable, restore the inoperable switchboard to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery and/or its charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. distribution system train shall be determined OPERABLE and energized with tie breakers open 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 and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

2. The pilot cell specific gravity, corrected to 77°F, is $\geq$ 1.200.

3. The pilot cell voltage is $\geq$ 2.1 volts, and

4. The overall battery voltage is $\geq$ 125 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is $\geq$ 2.1 volts under float charge,

2. The specific gravity, corrected to 77°F, of each connected cell is $\geq$ 1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The battery chargers, #1, and 2 will supply at least 140 amperes at 140 volts and #3 will supply at least 7[illegible] amperes at 132 volts for at least 2 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 2 hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment shall be energized and OPERABLE:

2 - 125-volt D.C. distribution switchboards, and

2 - 125-volt battery banks and chargers associated with the above d.c. distribution switchboards.

APPLICABILITY: Modes 5 and 6.

ACTION:

With less than the above complement of D.C. equipment OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. distribution switchboards 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 banks and chargers shall be demonstrated OPERABLE per Surveillance Requirements 4.8.2.3.2.

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Main Coolant System and the shield tank cavity shall be maintained uniform and sufficient to ensure a k_{eff} of 0.93 or less, which includes a 2% $\Delta k/k$ conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

a. With the boron concentration requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 26 gpm of 2200 ppm boric acid solution or its equivalent until K_{eff} is reduced to $\leq$ 0.93.

b. With a significant unexpected increase in the count rate on any channel, or an unexpected increase in the count rate by a factor of two after addition of a new fuel assembly or removal of a control rod, suspend CORE ALTERATIONS until the situation is reviewed by plant technical supervisory personnel.

c. The provisions of Specification 3.0.3 are not applicable.

*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed with fuel in the vessel.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The above required reactivity condition shall be determined:

a. Prior to removing or unbolting the reactor vessel head, and

b. Prior to withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

c. At least after the insertion or removal of each 5 fuel assemblies by withdrawing a single control rod using the manipulator crane to obtain a plot of control rod position versus inverse countrate multiplication. Using the inverse countrate data obtained, the SHUTDOWN MARGIN shall be calculated. If these calculations indicate that the SHUTDOWN MARGIN will be less than 5% Δ k/k (without the 2% Δ k/k conservative allowance for uncertainties) with all control rods inserted in the fully loaded core, the boron concentration will be increased to provide the required 5% Δ k/k calculated SHUTDOWN MARGIN.

4.9.1.2 Equipment which would make possible inadvertent reactivity increases shall be made inoperable and tagged out of service.

4.9.1.3 The boron concentration of the Main Coolant System and the shield tank cavity shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

4.9.1.4 Before flooding the shield tank cavity with borated water, it shall be determined that the boron concentration of the water in the Safety Injection Tank is not less than the required boron concentration of the Main Coolant System as specified above.

4.9.1.5 A record will be made of the neutron count rate before and after any change in core geometry.

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 in the control room.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

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

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 120 hours.

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

ACTION:

With the reactor subscritical for less than 120 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 120 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment access hatch (or temporary cover) closed and held in place by a minimum of four bolts,

b. A minimum of one door in the airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE containment isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE containment isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Verifying that the containment isolation valves are capable of being closed.

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.

SHIELD TANK CAVITY

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 Control rods and fuel assemblies shall be handled one-by-one with an OPERABLE shield tank cavity manipulator crane and universal handling tool with:

a. A minimum capacity of 900 pounds, and

b. An overload cut off limit $\leq$ 4800 pounds above base load.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and handling tool OPERABILITY not satisfied, suspend use of the inoperable manipulator crane or handling tool from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 The manipulator crane and handling tool used for movement of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 900 pounds, demonstrating an automatic load cut off when the crane load exceeds 4800 pounds above base load, and verifying proper operation of the handling tool.

CRANE TRAVEL - SPENT FUEL PIT

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 900 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pit.

APPLICABILITY: With fuel assemblies in the spent fuel pit.

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 Loads in excess of 900 pounds shall be prevented from traveling over fuel essemblies by administrative control.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 The Shutdown Cooling System shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With the Shutdown Cooling System not 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 Main Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The Shutdown Cooling System may be removed from operation for up to 6 hours in any 24 hour period during the performance of CORE ALTERATIONS.

. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The Shutdown Cooling System shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 950 gpm at least once per 24 hours.

CONTAINMENT PURGE FAN SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge fan shutdown system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

With the containment purge fan shutdown system inoperable, remove power to the purge fan. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge fan shutdown system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge fan shutdown occurs on manual initiation from the primary auxiliary building fan room.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least, 32 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

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

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.

SPENT FUEL PIT WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pit.

ACTION:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pit areas and restore water level to within its limit within 4 hours. The provision of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pit shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pit.

SPENT FUEL PIT BUILDING ISOLATION

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel pit building exhaust ventilation shutdown system and both building exit doors shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pit.

ACTION:

a. With the spent fuel pit building exhaust ventilation shutdown system or an exit door inoperable, suspend all operations involving movement of fuel within the spent fuel pit or crane operation with loads over the spent fuel pit until the exhaust ventilation shutdown system and exit door are restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required spent fuel pit building exhaust ventilation shutdown system and exit doors shall be demonstrated OPERABLE within 100 hours prior to the start and at least once per 7 days during handling of irradiated fuel in the spent fuel pit by:

a. Verifying that spent fuel exhaust ventilation shutdown occurs on manual initiation from the primary auxiliary building fan room and

b. Verifying that each exit door can be closed.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for low power physics tests provided 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 $\geq$ 26 gpm of 2200 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 $\geq$ 26 gpm of 2200 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 control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each control rod not fully inserted shall be demonstrated OPERABLE by verifying its rod drop time to be $\leq$ 2.5 seconds within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

CONTROL ROD OPERABILITY AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height and insertion limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained $\leq$ 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 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 Specification 3.2.1 being exceeded while the requirements of Specification 3.1.3.1, 3.1.3.4, or 3.1.3.5 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specification 3.2.1 or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be $\leq$ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specification 4.2.1 shall be performed at least once per 12 hours.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specification 3.4.8.1 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 Power Range and Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER, and

c. The Main Coolant System temperature and pressure are maintained $\geq$ 250°F and $\geq$ 300 psig, respectively.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Main Coolant System temperature and pressure < 250°F or 300 psig, immediately open the reactor trip breakers and restore the temperature-pressure to within its limit within 30 minutes; perform the analysis required by Specification 3.4.8.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Main Coolant System temperature and pressure shall be verified to be $\geq$ 250°F and 300 psig at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour.

SURVEILLANCE REQUIREMENTS (Continued)

4.10.3.3 Each Intermediate Power Range and Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.1.1.4, 3.1.3.1, 3.1.3.4, and 3.1.3.5, may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate Power Range and Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate Power Range and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.1.2.6 calls for two charging pumps to be OPERABLE and provides explicit ACTION requirements when only one charging pump is OPERABLE. Under the terms of Specification 3.0.3, if no charging pumps are inoperable, the facility is required to be in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, Main Coolant System boron concentration, and Main Coolant System 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 Main Coolant System cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 4.72% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with accident analysis assumptions. With $T_{avg} \leq 330°F$, the reactivity transients resulting from a postulated steam line break cooldown are minimal. 5% $\Delta k/k$ SHUTDOWN MARGIN (with all rods inserted) provides adequate protection to preclude criticality for all postulated accidents with the reactor vessel head in place.

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. Normally, when full power is reached after each refueling, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted steady state curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction and the reactivity status of the core can be continuously evaluated, and any deviation would be thoroughly investigated and evaluated.

BASES

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 950 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Main Coolant System. A flow rate of at least 950 GPM will circulate an equivalent Main Coolant System volume of 2,940 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control. Restriction of boron dilution with Main Coolant System temperature < 250°F prevents inadvertent criticality due to excess dilution below the temperature limit for criticality.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT

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 requirements 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 Main Coolant System boron concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

BASES

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths from the borated water sources to the charging pumps, 4) boric acid gravity flow connection, 5) associated heat tracing systems, and 6) power supply from OPERABLE 480 volt busses. flow paths.

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 is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 5.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 776 gallons of 22,000 ppm borated water from the boric acid mix tank or 9192 gallons of 2200 ppm borated water from the safety injection tank.

With the Main Coolant System temperature below 200°F, except during refueling, one charging pump is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting positive reactivity change in the event the single charging pump becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 7% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 1059 gallons of 22,000 ppm borated water from the boric acid mix tank or 13,218 gallons of 2200 ppm borated water from the safety injection tank.

3/4.1.3 MOVABLE CONTROL RODS

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, (3) the individual control rod worth and hot channel factors used for the accident analyses are not exceeded, and (4) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

For control rod misalignment up to 8 inches from every other control rod in its group and for all control rod groups within their insertion limits, the hot channel factors will be well within the design limits of:

$F_q^N \leq$ 2.70 Gulf United fuel
2.71 EXXON Nuclear fuel

$F_{\Delta H}^N \leq$ 1.86 Gulf United fuel
1.75 EXXON Nuclear fuel.

The limit applies to three loop operation, in which case the power coordinate is rescaled to 100% of rated three loop power. This ensures that the induced peaking will not lead to worse thermal conditions than for four loop operation since the flow to power ratio is greater for three loop operation.

If a control rod is misaligned the hot channel factors, potential ejected rod worth, and SHUTDOWN MARGIN will be shown to be within design limits and reactor power will be reduced. The requirements that no more than one inoperable control rod is allowed and that the shutdown margin is maintained ensures that the reactor can be brought to a safe shutdown condition at any time.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires ejected rod worth and SHUTDOWN MARGIN to be within limits and a restriction in THERMAL POWER; either of these restrictions provide

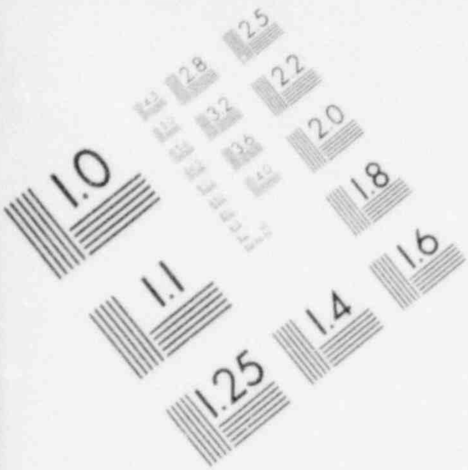
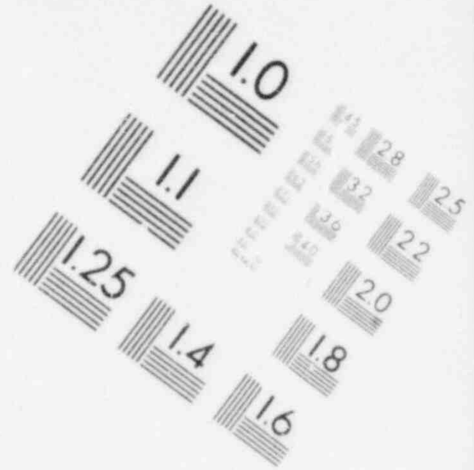
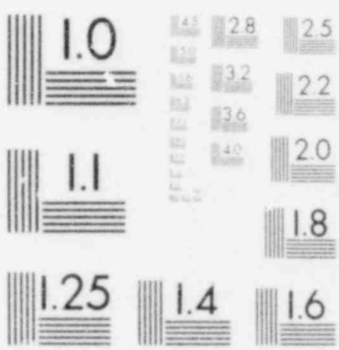



IMAGE EVALUATION TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART

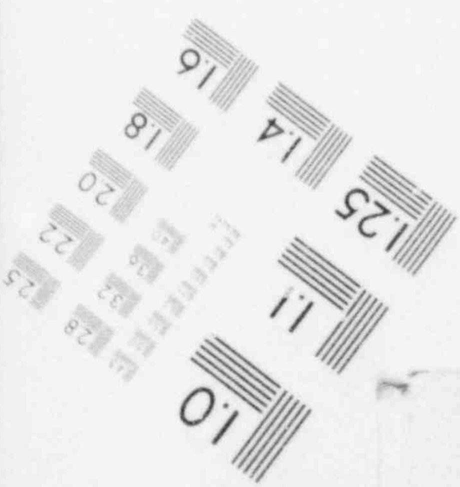
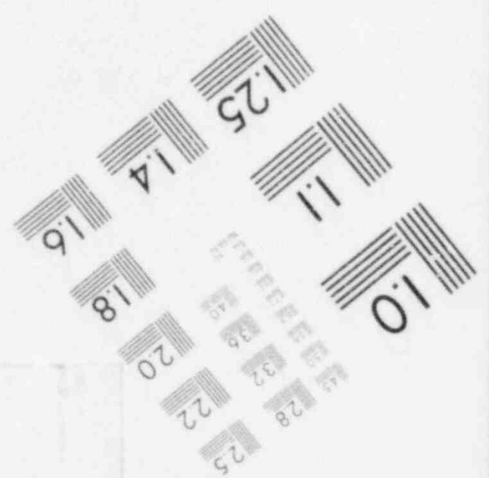



IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

3/4.1.3 MOVABLE CONTROL RODS (Continued)

assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis for a rod ejection accident.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 511°F$ and with all main 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 4 hours. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria.

3/4.2.1 PEAK LINEAR HEAT GENERATION RATE

Limiting the peak linear heat generation rate (LHGR) during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

When operating at constant power, all rods out, with equilibrium xenon, power peaking in the Yankee Rowe core decreases monotonically as a function of cycle burnup. This has been verified by both calculation and measurement on Yankee cores and is in accord with the expected behavior in a core that does not contain burnable poison. The all-rods-out power peaking measured prior to exceeding 75% of RATED THERMAL POWER after each fuel loading thus provides an upper bound on all-rods-out power peaking for the remainder of that cycle. Thereafter the measured power peaking shall be checked every 1000 equivalent full power hours and the latest measured value shall be used in the computation. The only effects which can increase peaking beyond this value would be control rod insertion and xenon transients and these are accounted for in calculating peak LHGR.

The core is stable with respect to xenon, and any xenon transients which may be excited are rapidly damped.

The xenon multiplier in Figure 3.2-3 was selected to conservatively account for transients which can result from control rod motion at full power.

The limits on power level and control rod position following control rod insertion were selected to prevent exceeding the maximum allowable linear heat generation rate limits in Figure 3.2-1 within the first few hours following return to power after the insertion. With Yankee's highly damped core, the 24 hour hold allows sufficient time for the initial xenon maldistribution to accommodate itself to the new power distribution. The restriction on control rod location during these 24 hours assures that the return to allowable fraction of full power will not cause additional redistribution due to rod motion.

BASES (Continued)

These conclusions are based on plant tests and on calculations performed with the SIMULATE three dimensional nodal code used in the analysis of Core XI (reference cycle) described in Proposed Change No. 115, dated March 29, 1974.

3/4.2.2 and 3/4.2.3 NUCLEAR HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux and ethalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specification 4.2.2.1 and 4.2.3.1. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 8 inches from any other rod in the group.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specification 3.1.3.5 is maintained.

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 with its limits provided conditions a thru c above, are maintained.

When an F^N_q measurement is taken, both experimental error and manufacturing tolerance must be allowed for 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 4% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.86$ for Gulf United fuel and 1.75 for EXXON Nuclear fuel. The 8% allowance is based on the following considerations:

BASES (Continued)

a. Abnormal perturbation in the radial power shape, such as from rod misalighment, effect $F^N_{\Delta H}$ more directly than F^N_q,

b. Although rod movement has a direct influence upon limiting F^N_q to within its limits, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F^N_q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient & accident analyses. The limits are consistent with the accident analysis 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 & other expected transient operation. The 18 month periodic measurement of the Main Coolant System total flow rate is adequate to detect flow degradation & ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE SYSTEM (RPS) AND ENGINEERED SAFEGUARDS SYSTEM (ESS) INSTRUMENTATION

The OPERABILITY of the RPS and ESS instrumentation systems and permissive circuits ensure that 1) the associated ESS action and/or RPS trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained, and 4) sufficient system functional capability is available for RPS and ESS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTION SYSTEM

The OPERABILITY of the incore detection system with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the reactor core power distribution.

3/4.3.3.3 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public (and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs").

3/4.4 MAIN COOLANT SYSTEM

BASES

3/4.4.1 MAIN COOLANT LOOPS

The plant is designed to operate with all main coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one main coolant loop not in operation, THERMAL POWER is restricted to < 75 percent of RATED THERMAL POWER. With four loops operating, a loss of flow or low SG water level in two loops will cause a reactor trip. A loss of flow or low SG water level in one loop will cause a reactor trip with three loops operating.

Adequate main coolant loops are required to provide sufficient heat removal capability for removing core decay heat. Single failure considerations require placing the Shutdown Cooling System into operation if the required main coolant loops are not OPERABLE. A steam generator is capable of removing core decay heat by natural or forced circulation provided the conditions specified in 4.4.1.1.2 are met.

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 isolated loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. Startup of an isolated loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying 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

The pressurizer code safety valves operate to prevent the Main Coolant System from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 92,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating Shutdown Cooling System connected to the Main Coolant System provides overpressure relief capability and will prevent Main Coolant System overpressurization during shutdown.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the Main Coolant System from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code, 1974 Edition, and Addenda through Summer, 1975.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the Main Coolant System is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve Main Coolant System pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

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 Main Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Main Coolant System since isolation removes the source of potential failure.

Primary system leakage may be identified by one or more of the following methods:

1. Reactor containment air particulate monitoring.
2. Tritium balance within the containment.
3. Relative humidity within the containment.
4. Containment sump level.
5. Periodic visual examinations of all systems for evidence of leakage.
6. Water inventories and balances.
7. Vapor container sound system.

Techniques used to identify primary system to secondary system leakage include the following:

1. Radiochemical analyses for tritium.
2. Analyses for boron.
3. Radiochemical analyses for beta-gamma activity in the steam generator blowdown and/or air ejector discharge.

Primary system leakage will be maintained at the lowest practicable value to facilitate detection and identification of change in leak rate.

3/4.4.6 CHEMISTRY

The limitations on Main Coolant System chemistry ensure that corrosion of the Main Coolant System is minimized and reduces the potential for Main 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 Main 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 Main 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.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-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 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 514°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.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Main 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. 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.

For the following cases the heatup and cooldown rates may be increased to as high as 100°F in any one hour period. The vessel is limited to 200 cycles total at this rate.

a) When starting from a nominal ambient temperature which is less than 150°F, the maximum rate of heatup shall not exceed 50°F in any one hour period up to 250°F and 100°F in any one hour period from 250°F to operating temperature. No isothermal wait at 250°F is required.

b) When starting from an isothermal condition of 150°F the maximum heatup rate shall not exceed 100°F in any one hour period.

c) The maximum cooldown rate shall not exceed 100°F in any one hour period down to 150°F and 50°F in any one hour period from there to ambient conditions.

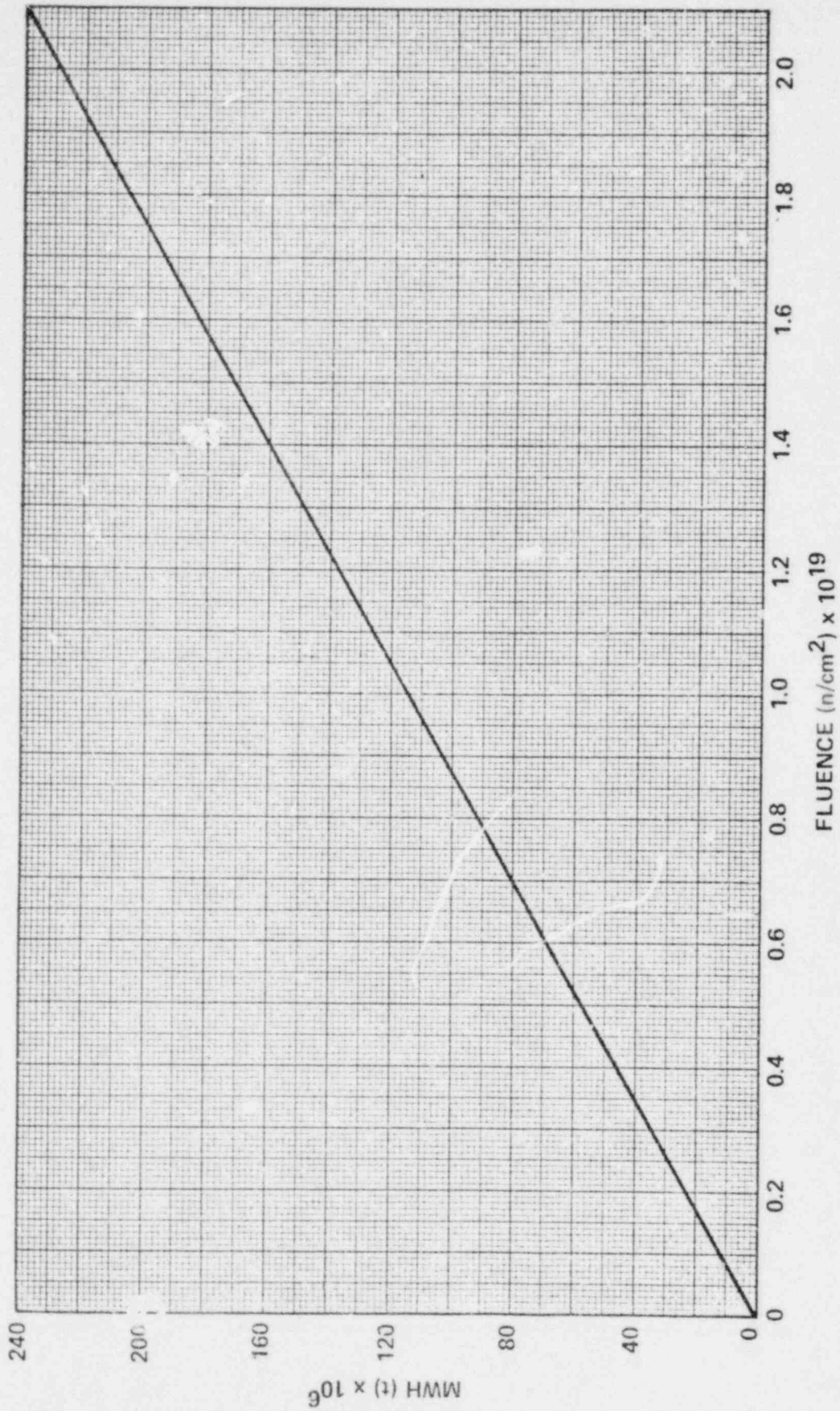
During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

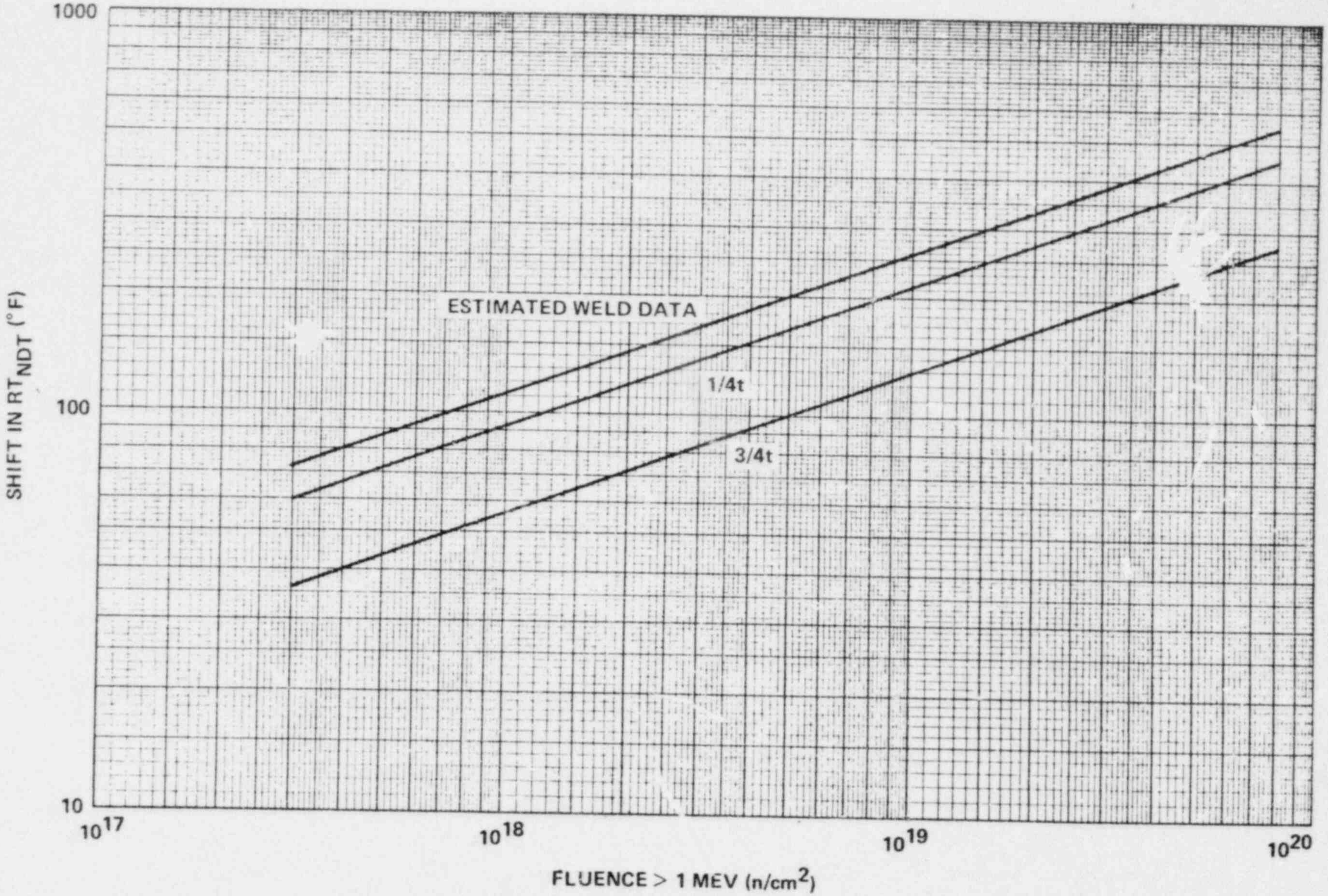
The heatup limit curves, Figures 3.4-2, and 3.4-4, are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour. The cooldown limit curves, Figures, 3.4-3, and 3.4-5, are composite curves which were prepared based upon the same type analysis. The heatup and cooldown curves were prepared based upon a beginning of life RT_{NDT} + 10°F. These limitations are derived by using the rules contained in Section III of the ASME Code including Appendix G, Protection Against Nonductile Failure and the rules contained in 10CFR50, Appendix G, Fracture Toughness Requirements.

Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in RT_{NDT}. The weld metal is assumed to be the controlling vessel material throughout the remainder of vessel life. The shift in RT_{NDT} can be pridicted by use of Bases Figure B 3/4 4-1 and Bases Figure B 3/4 4-2. The latter figure provides a shift curve for the surveillance specimens, and calculated shift curves for both the 1/4t and 3/4t locations in the vessel plate. The heatup and cooldown limit curves in Figures 3.4-2, 3.4-3, 3.4-4 and 3.4-5 will be adjusted to include the predicted shift in RT_{NDT} at the end of next core cycle. Adjustments for possible errors in the pressure and temperature sensing instruments have been included.

Figures 3.4-2, 3.4-3, 3.4-4 and 3.4-5 shall be revised (without submittal of a tech spec change) at each refueling outage by estimating the MWH(t) on the reactor and by indexing through Bases Figures B 3/4.4-1 and B3/4.4-2 to provide shifts in RT_{NDT} at the 1/4t and 3/4t locations in the reactor vessel beltline plate. The heatup and cooldown curves shall be shifted parallel to the temperature axis (horizontal) in the



BASES FIGURE B 3/4.4-1 REACTOR VESSEL INSIDE WALL FLUENCE EXPOSURE AS A FUNCTION OF POWER GENERATION



SHIFT IN RT_{NDT} AS A FUNCTION OF VESSEL FLUENCE FOR WELD METAL, 1/4t AND 3/4t

BASES FIGURE B3/4 4-2

direction of increasing temperature a distance equivalent to the shift in RT_{NDT} at the 1/4t or 3/4t as applicable during the period since the curves were last constructed. The following table provides the appropriate shift parameter to be applied.

CURVE	SHIFT POSITION
Heatup, upper limit	1/4t
Heatup, other rate limits	3/4t
Cooldown, all limits	1/4t

The pressure-temperature limit lines shown on Figures 3.4-4 and 3.4-5 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The requirement that the reactor is not to be made critical outside the limits specified provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained relative to the RT_{NDT} of the reactor coolant system. The limits for reactor criticality have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. Heatup to these temperatures will be accomplished by utilizing decay heat and by operating the reactor coolant pumps.

Temperature requirements for pressurization of the pressurizer correspond with DTT measured for the material of each component. The DTT is defined as the initial Nil Ductility Transition Temperature (NDTT) plus 60°F.

A temperature difference of 225°F between the pressurizer and reactor coolant system is specified to maintain thermal stresses within the surge line below design limits.

3/4.4.9 STRUCTURAL INTEGRITY

Following issuance of Section XI of the ASME Boiler and Pressure Vessel Code on January 1, 1970, YANKEE-ROWE adopted an Inservice Inspection Program consistent with the requirements of the Code and compatible with the "as constructed" plant. The adopted program, Table 4.4-3, will complete an inspection cycle within the Code accepted 10 year completion period.

Table 4.4-3 does not list those examination categories not applicable or accessible in YANKEE-ROWE. The Code does not apply intoto to plants constructed prior to its adoption, since provision for component access have not been incorporated into all.

The required inspection programs for the Main Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for the Main Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems", dated 1970 and Addenda through Winter 1970.

To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most main coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels and the requirement for remote underwater accessibility.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

The nondestructive testing for repairs on components greater than 2 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 2 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Main Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2200 psig following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing of Figure 3.4-4 and 3.4-5.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATOR

The OPERABILITY of the accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the Main Coolant System pressure falls below the pressure of the accumulator. This initial surge of water into the core provides the initial cooling mechanism during large Main Coolant System 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. A minimum useable water volume of 700 cubic feet require accumulator water volume to be at least 800 cubic feet equivalent to a level of $\geq$ 227 inches or 87% indicated level.

The accumulator power operated isolation valve fails to meet single failure criteria and removal of power to the valve is required.

The limits for operation with the accumulator inoperable for any reason except an isolation valve closed or pressurization system inoperable minimizes the time exposure of the plant to a LOCA event occurring concurrently which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of the accumulator is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of three independent ECCS safety injection subsystems, the recirculation subsystem, and the long term hot leg injection subsystem ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one safety injection subsystem, one purification pump and one fixed speed charging pump through any single failure consideration. Two safety injection subsystems operating in conjunction with the accumulator are capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest Main Coolant System cold leg pipe downward. In addition, the recirculation and long term hot leg injection subsystems provide long term core cooling and boron mixing capability during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the Main Coolant System temperature and pressure below 330°F, and 1000 psig, respectively, one OPERABLE ECCS safety injection subsystem, a recirculation subsystem and a long term hot leg injection subsystem with only one pump per subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor, the decreased probability of a LOCA and the limited core cooling requirements because of the negligible energy stored in the primary coolant under these conditions.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

Complete system tests cannot be performed when the reactor is operating because of their inter-relation with operating systems. The method of assuring operability of these systems is a combination of complete system tests performed during refueling shutdowns and monthly tests of active system components (pumps and valves) during reactor operation. The test interval is based on the judgement that more frequent testing would not significantly increase reliability.

The subsystems power operated valves fail to meet single failure criteria and removal of power to the valves is required.

3/4.5.4 SAFETY INJECTION TANK

The OPERABILITY of the Safety Injection Tank (SIT) 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 SIT 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 SIT and the Main Coolant System water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses, which is based on allowing a minimum of 77,000 gallons to be injected by the safety injection subsystems before the recirculation is manually established. LOCA analyses show that an injection of 77,000 gallons is sufficient to limit core temperatures and containment pressure for the full spectrum of pipe ruptures. This leaves up to 40,000 gallons in the SIT as reserve. The boron concentration of 2200 ppm is the highest value assumed in any accident analysis.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

Containment integrity is not required with Main Coolant System temperature < 200°F or pressure < 300 psig because no steam will be generated in the unlikely event of a Main Coolant System rupture and hence, no driving force to release fission products from the containment will exist.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.70\ L_a$ 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 except for the exemption granted by letter dated January 14, 1974 which waived the requirement for local leak testing of equipment and emergency hatch seals providing the containment continuous leak monitoring system is OPERABLE. The licensee has requested an exemption to Appendix J for containment electrical penetration testing.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air lock are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. The containment continuous leak monitoring system provides some assurance that the overall air lock leakage will

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

not become excessive due to door seal damage during the intervals between air lock leakage tests. The surveillance testing requirements are consistent with the requirements of Appendix "J" to 10 CFR 50 except for the licensee's reliance on the containment continuous leak monitoring system to detect excessive airlock door seal leakage between airlock leakage tests. The licensee was granted an exemption by letter dated January 14, 1974 to use this monitoring system rather than leak test the airlock door seal after each opening.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that the containment peak pressure does not exceed the design pressure of 34.5 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 31.6 psig. The limit of 3.0 psig for initial positive containment pressure will limit the total pressure to 31.6 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that:

a. The overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA, and

b. The ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The average air temperature is determined in accordance with the following equation:

Average air temperature = (each loop temperature x 6) plus (charging floor temperature x 20) plus (each hemisphere, low, temperature x 8) plus (each hemisphere, fan, temperature x 20) divided by 100.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 31.6 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTINUOUS LEAK MONITORING SYSTEM

The OPERABILITY of the continuous leak monitoring system provides some assurance that the containment equipment hatch seal, emergency hatch seal, and airlock seal are not leaking excessively between Type A leak tests.

3/4.6.2 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.3 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. The purge system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment.

The hydrogen recirculating system is provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1035 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Code, 1956 Edition. The total relieving capacity for all valves on all of the steam lines is 3.1×10^6 lbs/hr which is 129 percent of the total secondary steam flow of 2.4×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Tables 3.7-1 and 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 and Intermediate Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (108)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (81)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam generator

U = maximum number of inoperable safety valves per operating steam generator

(108) = Power Range and Intermediate Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

(81) = Maximum percent of RATED THERMAL POWER permissible for 3 loop operation.

X = Total relieving capacity of all safety valves per steam generator in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 EMERGENCY BOILER FEEDWATER SYSTEM

The OPERABILITY of the emergency boiler feedwater system ensures that the Main Coolant System can be cooled down to less than 330°F from normal operating conditions in the event of a total loss of off-site power.

The steam driven emergency boiler feedwater pump is capable of delivering a total feedwater flow of 80 gpm at a pressure of 950 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 Main Coolant System temperature to less than 330°F when the Shutdown Cooling System may be placed into operation.

The monthly testing interval of the steam generator emergency boiler feed pump verifies its operability by recirculating water to the demineralized water tank. Proper functioning of the steam turbine and the emergency boiler feed pump will be made by direct visual observation.

3/4.7.1.3 PRIMARY AND DEMINERALIZED WATER STORAGE TANK

The OPERABILITY of the primary and demineralized water storage tanks with the minimum combined water volume ensures that sufficient water is available to maintain the Main Coolant System at HOT STANDBY in excess of 24 hours with steam discharge to the atmosphere concurrent with total loss of off-site power.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

The steam break accident is based upon a postulated release of the entire contents of the secondary system to the atmosphere using a site boundary dose limit of 1.31 rem for thyroid dose.

The limiting dose for this accident results from iodine in the secondary coolant. The reactor distribution of iodine isotopes with 1% failed fuel was used for this calculation. I-131 is the dominant isotope because of its low MPC in air and because the other iodine isotopes have shorter half-lives and therefore cannot build up to significant concentrations in the secondary coolant, given the limitations on primary system leak rate and activity. The entire secondary system contains approximately 132m^3 of water at standard conditions. One-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets.

3/4.7.1.5 TURBINE GENERATOR THROTTLE AND CONTROL VALVES

The OPERABILITY of the turbine generator throttle and control valves ensures that each steam generator will be isolated at the instant of scram for a LOCA with an area smaller than 0.1 ft^2, as assumed in the accident analysis. The OPERABILITY of the turbine generator throttle and control valves within the closure times of the surveillance requirements is consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

The limitations on secondary water chemistry assures that the potential for stress corrosion is reduced. The time interval during which operation is permited to continue provides time for taking corrective action to restore the contaminant concentration to within limits. The effects of exceeding the chloride limit are time dependent and corrosion studies show that operation may be continued without a significant effect on structural integrity for the specified limited time.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations are based on a steam generator initial RT_{NDT} plus 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 PRIMARY PUMP SEAL WATER SYSTEM

The OPERABILITY of the primary pump seal water system ensures that sufficient seal water is available for continued operation of the purification pumps during normal and accident conditions.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM EMERGENCY SHUTDOWN

The OPERABILITY of the control room ventilation system emergency shutdown enhances the opportunity for the control room to remain habitable for operations personnel during and following accident conditions.

3/4.7.6 SEALED SOURCE CONTAMINATION

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.

3/4.7.7 WASTE EFFLUENTS

3/4.7.7.1 RADIOACTIVE SOLID WASTES

Radioactive solid wastes are permitted to be disposed of only at licensed offsite disposal areas.

3/4.7.7.2 RADIOACTIVE LIQUID WASTE

Releases of radioactive materials in effluents will be kept at small fractions of the limits specified in 10 CFR 20 except under unusual operating conditions which may temporarily result in releases higher than such small fractions, but still within the limits specified in 10 CFR 20. The Licensee will exert his best efforts to keep levels of radioactive materials in effluents as low as reasonably achievable.

Radioactive liquid wastes are diluted in the plant discharge system prior to release up to a maximum dilution multiple of 7.24×10^{-6} minutes per gallon times the radioactive release rate in gallons per minute.

Radioactive liquid wastes are collected in tanks until a quantity sufficient for processing has accumulated. The processed liquids are directed to separate tanks where they are sampled and analyzed for radioactivity content prior to release. Discharges from the waste disposal system are made through a recorder controller which provides a measure and control of the volume released. The volume released and the radioactivity analyses provides the basis for record keeping.

3/4.7.7.3 RADIOACTIVE GASEOUS WASTE

Releases of radioactive materials in airborne waste will not exceed the concentration limits specified. The limit for iodine isotopes and particulate isotopes with half lives greater than 8 days was further reduced by a factor of 700 to account for reconcentration in the food chain.

Gaseous wastes flow from the low pressure surge tank through the gas compressor(s) which discharge to the waste gas surge drum. The drum is pressurized and then allowed to decay before discharge to the primary vent stack. The gaseous discharge is continuously monitored in the primary vent stack. Coincident with the loss of exhaust fan suction, the vent line flow control valve will close, terminating the release.

3/4.7.8 ENVIRONMENTAL MONITORING

The operational environmental monitoring program is designed to demonstrate the adequacy of safeguards inherent in plant design and the effectiveness of established control measures for the releases of radioactive materials.

Special attention is given to gamma spectrum analysis of samples in order to identify and reference nuclides present in plant effluents. Spectra of environmental samples are observed for the ingrowth of plant produced radionuclides.

3/4.8 ELECTRICAL POWER 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 8,000 gallons is the amount of fuel which would be required by two diesels when operating under accident conditions for a period of 7 days.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least two of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

The test specified for the batteries will demonstrate that they are in a constant state of readiness. The frequencies of the tests will allow detecting any tendency towards battery deterioration and subsequent loss of capacity.

BASES

3/4.9.1 REACTIVITY

The limitations on reactivity ensure that: 1) the reactor will remain substantially 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.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core during CORE ALTERATIONS. One monitor is acceptable whenever CORE ALTERATIONS are not in progress on the basis of the core stable reactivity condition.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 SHIELD TANK CAVITY MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) the manipulator crane and universal handling tool will be used for movement of control rods and fuel assemblies, 2) each crane and tool has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over 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 analyses.

3/4.9.8 COOLANT CIRCULATION

The requirement that the Shutdown Cooling System be in operation is consistent with the assumptions in the analysis of the boron dilution accident and prevents local variations in boron concentrations, thus minimizing the effects of an inadvertent boron dilution. It also assures that 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.

3/4.9.9 CONTAINMENT PURGE FAN SHUTDOWN SYSTEM

The OPERABILITY of this system ensures that the containment purge fan can be shutdown upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND SPENT FUEL PIT

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

BASES

3/4.9.12 SPENT FUEL PIT BUILDING ISOLATION

The requirements for spent fuel pit building isolation ensure that radioactive material released from an irradiated fuel assembly will be remain within the spent fuel pit area. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when physics tests are performed to measure such parameters as control rod worth. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 CONTROL ROD CPERABILITY AND INSERTION UNITS

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

This special test exception permits the reactor to be critical at less than or equal to 5% of RATED THERMAL POWER during low temperature PHYSICS TESTS required to measure such parameters as control rod worth and SHUTDOWN MARGIN.

3/4.10.4 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

5.2 CONTAINMENT

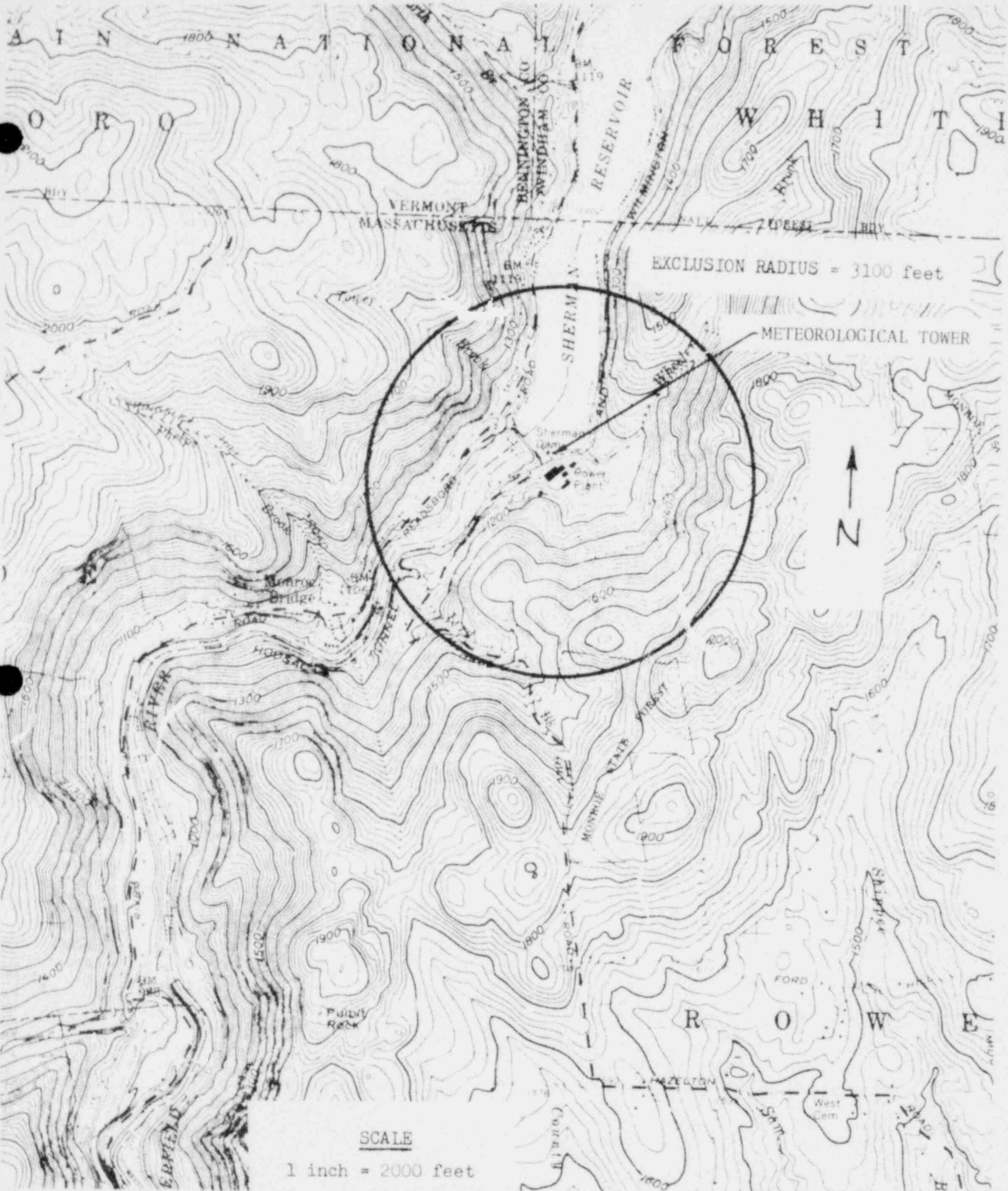
CONFIGURATION

5.2.1 The reactor containment building is a steel spherical shell having the following design features:

a. Nominal inside diameter = 125 feet.

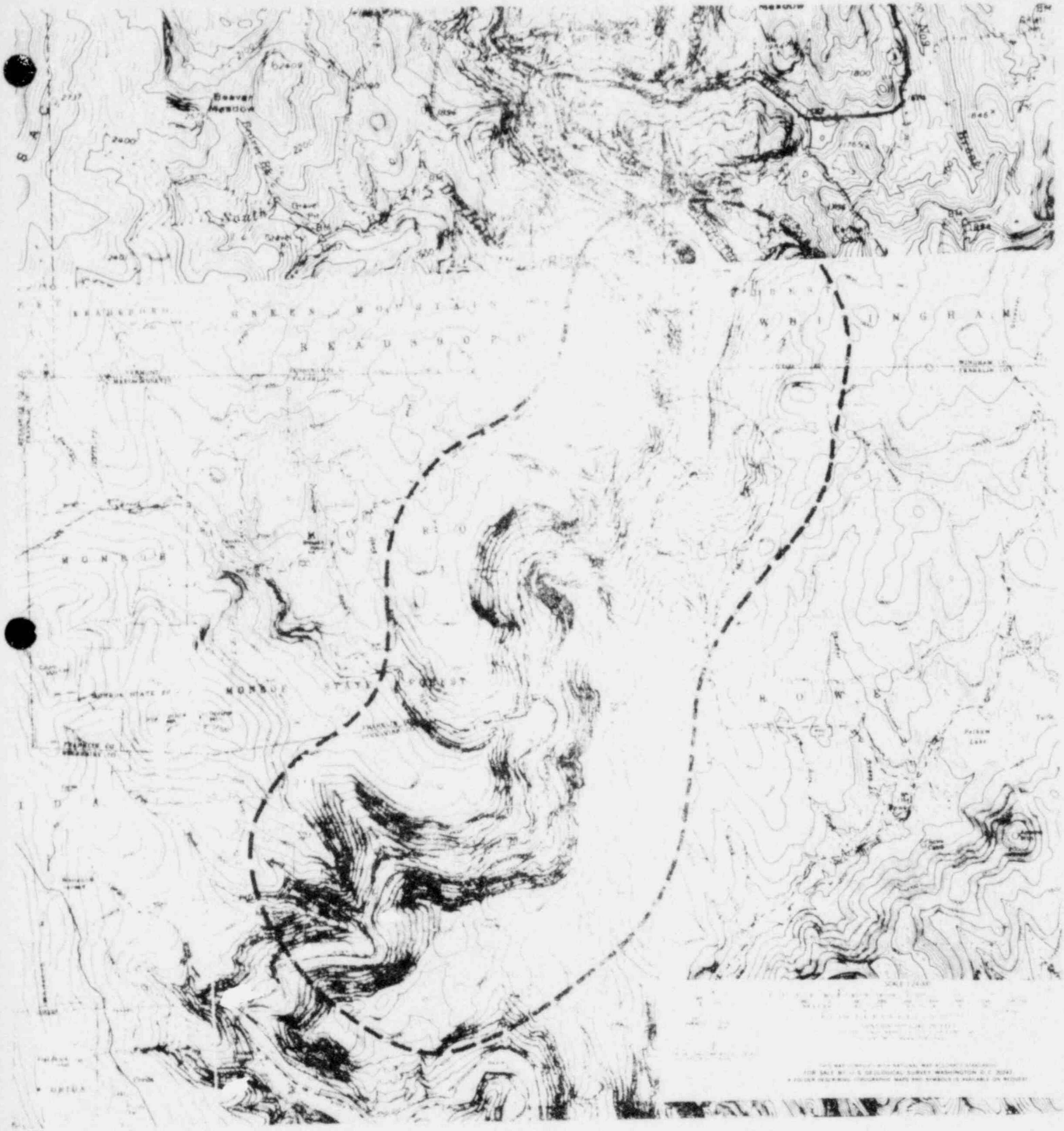
b. Minimum thickness of steel shell = 7/8 inches.

c. Net free volume = 860,000 cubic feet.



EXCLUSION AREA

YANKEE-ROWE FIGURE 5.1-1 April 1, 1976



LOW POPULATION ZONE

YANKEE-ROWE FIGURE 5.1 April 1, 1976

a. In accordance with the code requirements specified in ASME Boiler and Pressure Vessel Code, Section VIII, including all addenda through 1956, and the ANSI (formerly ASI) Standards, Power Piping Code, B31.1, 1955 Edition, and B16.5, 1957 Edition, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 668°F.

VOLUME

5.4.2 The total water and steam volume of the Main Coolant System is 2940 cubic feet.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the design provisions, contained in proposal changes #96 and #117 as approved by change #97 and Amendment #21, respectively, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained as described in the Final Hazards Summary Report section 218 with a center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 with the spent fuel pit filled with unborated water. The k_{eff} of ≤ 0.95 include a conservative allowance of 3% $\Delta k/k$ for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool.

CAPACITY

5.6.3 The spent fuel pit is designed and shall be maintained with a storage capacity limited to no more than 170 fuel assemblies.

5.7 METEOROLOGICAL TOWER LOCATION

5.7.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.8 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.8.1 The components identified in Table 5.8-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.8-1.

TABLE 5.8.1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Pressure Vessel	200 heatup and cooldown cycles > 50°F/hour and ≤ 100°F/hour.	70°F to 534°F to 70°F

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

Administrative controls are the written rules, orders, instructions, procedures, policies, practices, and the designation of authorities and responsibilities by the management to obtain assurance or safety and quality of operation and maintenance of a nuclear power reactor. These controls shall be adhered to.

6.1 RESPONSIBILITY

6.1.1 The Plant Superintendent shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 In all matters relating to the operation of the plant and to these Technical Specifications the Plant Superintendent shall report to and be directly responsible to the Manager of Operations in the Yankee Atomic Electric Company.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. Operating crew personnel trained in radiation protection procedures fill this requirement.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

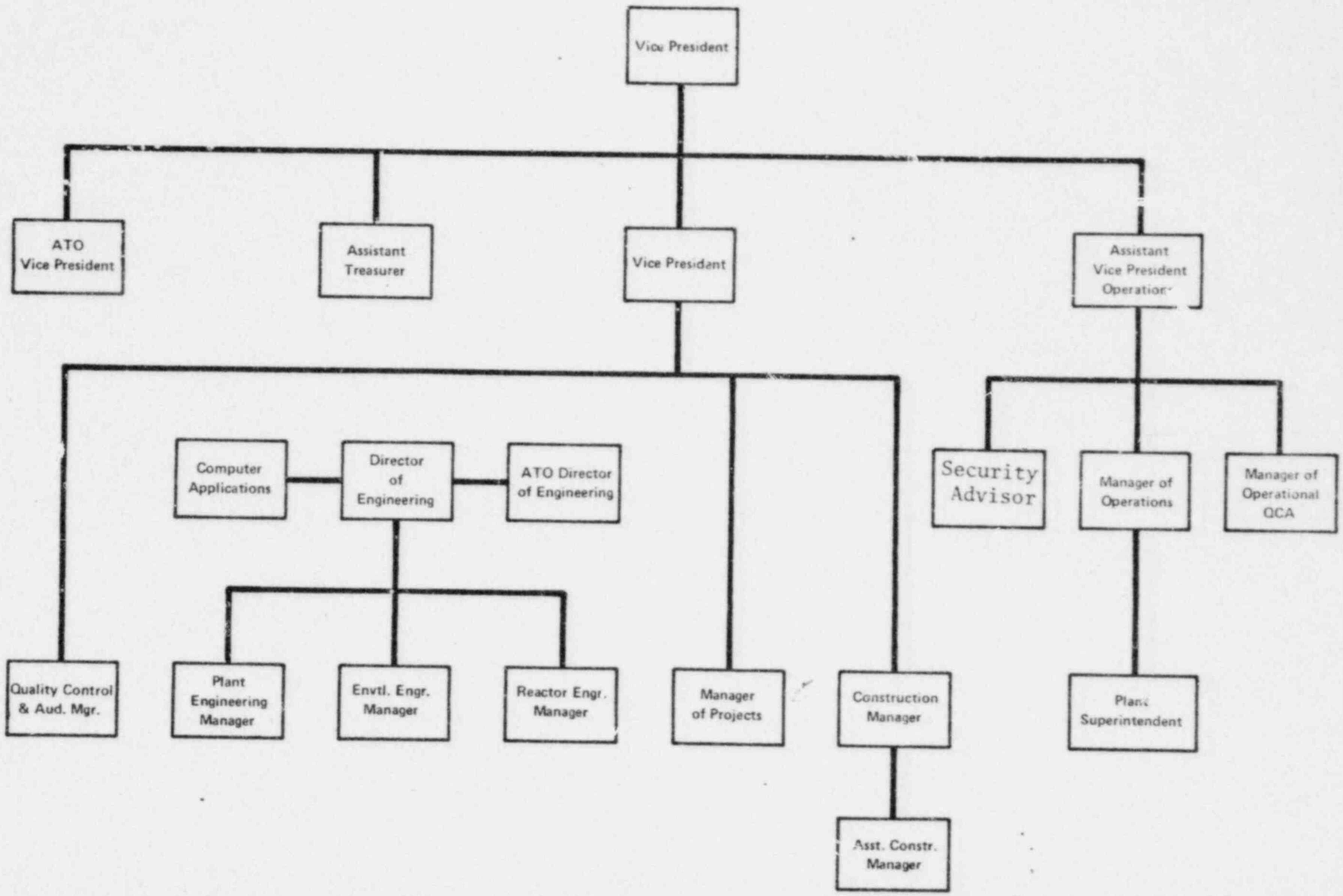


Figure 6.2-i

OFFSITE ORGANIZATION

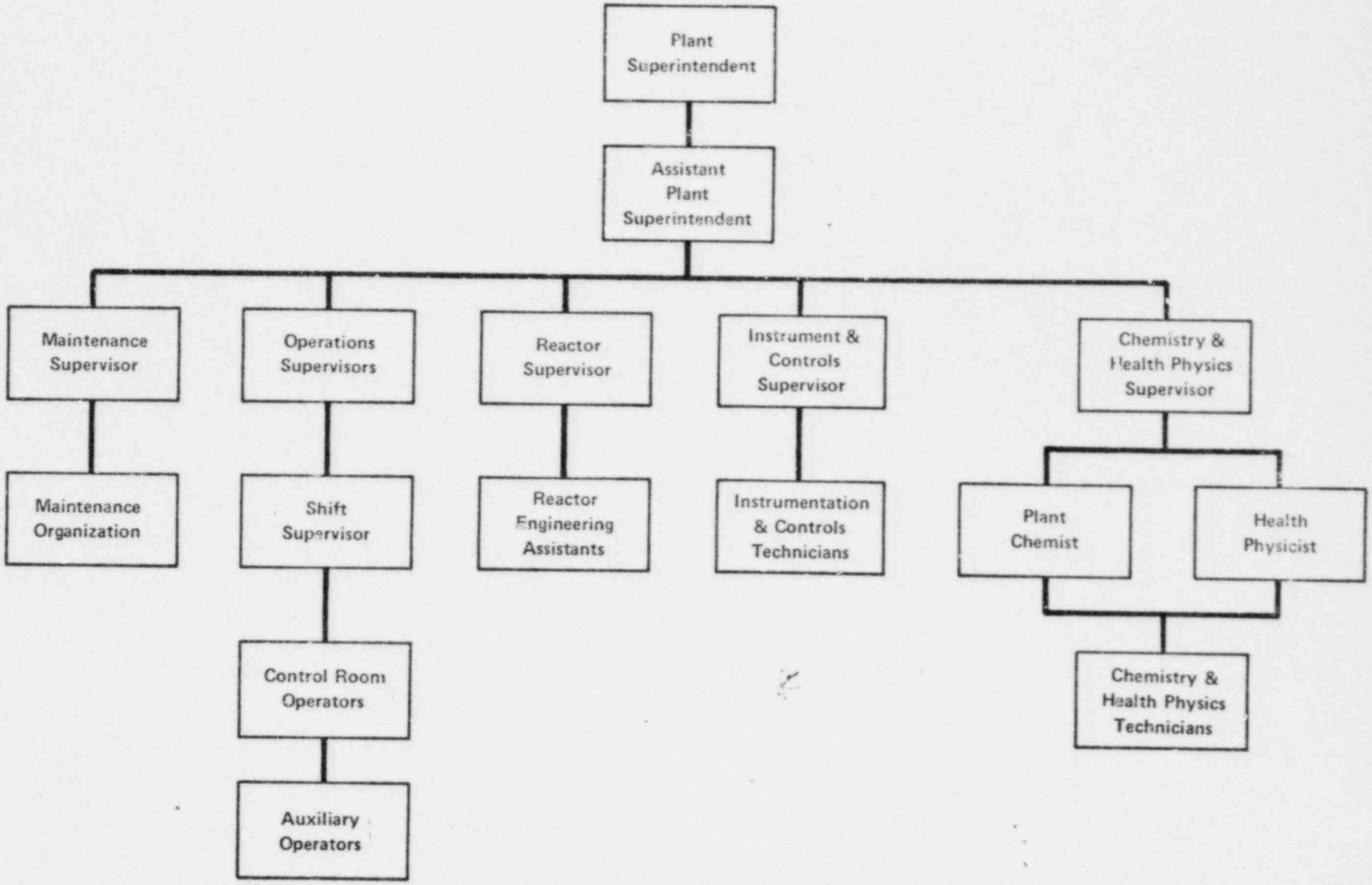


Figure 6.2-2

FACILITY ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2**	1
NON-Licensed	2	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising Core ALTERNATIONS after the initial fuel loading.

**One additional licensed operator (OL) is required for MODE 2 except when restarting within four hours of a shutdown for which the cause has been clearly established.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff listed below shall meet or exceed the minimum qualifications of ANSI N18.1971 for comparable position.

a. Plant Superintendent

b. Assistant Plant Superintendnet

c. Chemistry and Health Physics Supervisor

d. Operations Supervisor

e. Reactor Engineer

f. Maintenance Supervisor

g. Instrument and Controls Supervisors

h. Shift Supervisors

i. Health Physicist

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATION REVIEW COMMITTEE

FUNCTION

6.5.1.1 The Plant Operation Review Committee (PORC) shall function to advise the Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Operation Review Committee shall be composed of the:

Chairman: Plant Superintendent

Vice Chairman: Assistant Plant Superintendent

Member: Operations Supervisor

Member: Maintenance Supervisor

Member: Reactor Engineer

Member: Chemistry and Health Physics Supervisor

Member: Instrument and Control Supervisor

Member: Health Physicist

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities or count toward a PORC quorum at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or Vice Chairman.

QUORUM

6.5.1.5 A quorum of the PORC shall consist of a minimum of five people as follows:

a. The Chairman or Vice Chairman plus four members, or

b. The Chairman and Vice Chairman plus three members.

RESPONSIBILITIES

6.5.1.6 The Plant Operation Review Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the 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 the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager of Operations and to the Chairman of the Nuclear Safety Audit and Review Committee.

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews, investigations of analyses and reports thereon as requested by the Chairman of the Nuclear Safety Audit and Review Committee.

h. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Security Advisor.

i. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Radiation Protection Manager.

AUTHORITY

6.5.1.7 The Plant Operation Review Committee 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 Operations of disagreement between the PORC 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 Plant Operation Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Manager of Operations and Chairman of the Nuclear Safety Audit and Review Committee.

6.5.2 NUCLEAR SAFETY AUDIT AND REVIEW COMMITTEE

FUNCTION

6.5.2.1 The Nuclear Safety Audit and Review (NSAR) Committee shall function to provide independent review and audit of all aspects of plant safety. Adequacy of this review and audit is assured by the cross section of disciplines required of the Committee membership as described in Section 6.5.2.3.

COMPOSITION

6.5.2.2 The NSAR Committee shall be composed of at least six persons with the Committee membership and its Chairman and Vice Chairman appointed by the Yankee Atomic Electric Company Vice President or such person as he shall designate.

a. Chairman

b. Vice Chairman

c. Four technically qualified persons who are not members of the plant staff.

QUALIFICATION

6.5.2.3 Membership to the NSAR Committee requires that an individual meet one of the below academic and/or experience requirements:

a. Bachelor Degree or equivalent, plus five (5) years total experience in the below listed disciplines.

b. Nine (9) years total experience in the below listed disciplines:

(a) Nuclear Power Plant Technology

(b) Reactor Operations

(c) Utility Operations

(d) Power Plant Design

(e) Reactor Engineering

(f) Radiation Safety

(g) Safety Analysis

(h) Instrumentation and Control

(i) Me.allurgy

(j) Quality Assurance

ALTERNATES

6.5.2.4 All NSAR Committee alternate members shall be apppointed in writing by the Yankee Atomic Electric Company Vice President, or such person as he may designate, to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSAR Committee activities at any one time.

CONSULTANTS

6.5.2.5 Consultants may be utilized as determined by the NSAR Committee Chairman to provide expert advice, when needed, to the NSAR Comittee.

MEETING FREQUENCY

6.5.2.5 The NSAR Committee shall meet at least once pe months, $\pm$ 25%. Special meetings may be held when deemed necessa Company management or by the Chairman of the NSAR Committee, or, i.. the absence of the Chairman, by the Vice Chairman.

QUORUM

6.5.2.7 A NSAR Committee quorum shall consist of a minimum of five members as follows:

a. The Chairman or Vice Chairman plus four members (or two members plus two alternates).

b. The Chairman and Vice Chairman plus three members (or one member plus two alternates).

c. Those personnel from the organization reporting to the Manager of Operations shall always be in the minority.

REVIEW

6.5.2.8 The NSAR Commit'ee shall review:

a. The safety evaluations for 1) changes to equipment or systems and 2) tests or experiments completed under the provisions of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to equipment or systems which involve an unreviewed safety question as defined in Sc tion 50.59, 10 CFR.

c. Proposed test 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. Sign' icant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety, defined as Plant Information Reports.

g. REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

h. Reports and meetings minutes of the Plant Operation Review Committee.

i. Peform special reviews and investigations and render reports thereon as requested by the Assistant Vice President of Operations.

AUDITS

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the NSAR Committee. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months, ± 25%.

b. The performance, training and qualification of those members of the facility staff who have a direct relationship to operation, maintanance or technical aspects of the plant, at least once per 12 months, ± 25%.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months, ± 25%.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendis "B", 10 CFR 50, at least once pre 24 months, ± 25%.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months, ± 25%.

f. The Facility Security Plan and implementing procedures at least once per 24 months, ± 25%.

g. Any other area of facility operation considered appropriate by the NSAR Committee or the Vice President.

AUTHORITY

6.5.2.10 The NSAR Committee shall report to and advise the Vice President on those areas of responsibility specified in Sections 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Minutes of each NSAR Committee meeting shall be prepared and forwarded to the Vice President and each member of the Committee for review within 20 working days following each meeting. The meeting minutes shall include, where applicable, reports of reviews encompassed by Section 6.5.2.8; and reports of audits encompassed by Section 6.5.2.9. The review of the minutes shall be completed within 60 days of the date of their distribution.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NSAR Committee and the Manager of Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Manager of Operations and to the NSAR Committee within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Operation Review Committee. 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 NSAR Committee and the Manager of Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of Regulatory Guide 1.33, November 1972, except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the Plant Operations Review Committee and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in each document.

6.8.3 Procedures that have been developed as a result of changes defined in 10 CFR 50.59(a)(2) shall be independently reviewed to verify that the implementing actions do not constitute an unreviewed safety question. Those reviews shall be performed by Nuclear Service Division personnel having qualifications at least equivalent to those specified for NSAR Committee membership in 6.5.2.3. The procedures shall be approved by the Manager of Operations, NSD.

6.8.4 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License.

c. The change is documented, reviewed by the PORC and approved by the Plant Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted. The reporting requirements of Specifications 6.9.1, 6.9.2, 6.9.3 and 6.9.4 are in accordance with Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

ROUTINE REPORTS

6.9.1 Startup Report A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving and planned increase in power level, (3) installation of fuel that has a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FHSR and shall in general include a description of 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 required in license conditions based on other committments shall be included in this report.

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 commerical power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

6.9.2 Annual Operating Report. Routine operating reports covering the operation of the unit during 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.

The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

a. A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in item b(5) below.

b. For each outage or forced reduction in power* of over twenty percent of design power level where the reduction extends for greater than four hours:

(1) The proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

(2) A brief discussion of (or reference to reports of) any REPORTABLE OCCURRENCES pertaining to the outage or power reduction;

(3) Corrective action taken to reduce the probability of recurrence, if appropriate;

(4) Operating time lost as a result of the outage or power reduction (for scheduled or forced outages,** use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

(5) A description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

(6) A report of any single release of radioactivity or radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

c. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,*** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the whole body dose received from external sources shall be assigned to specific major work functions.

d. Indications of failed fuel resulting from irradiated fuel examinations, including eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

6.9.3 Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the tenth of each month following the calendar month covered by the report.

REPORTABLE OCCURENCES

6.9.4 REPORTABLE OCCURRENCES, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully described final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification With Wr.tten Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours, by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regional Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide a complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the Technical Specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items a(5), a(6), or b(1) below.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the Technical Specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item b(2) below.

(3) Abnormal degration discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for IDENTIFIED LEAKAGE set forth in Technical Specifications need not be reported under this item.

(4) Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the FHSR.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the FHSR.

Note: For items a(5) and a(6), reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items b(2) and b(3) below.

(7) Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described the FHSR or in the Bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the FHSR or Technical Specifications Bases; or discovery during plant life of conditions not specifically considered in the FHSR or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of unsafe conditions.

Note: This item is intended to provide for reporting of potentially generic problems.

b. Thirty Day Written Reports. The REPORTABLE OCCURRENCES discussed below shall be the subject of written reports to the Director of the appropirate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by assitional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items b(1) and b(2) need not be reported except where test results them selves reveal a degraded mode as described above.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for IDENTIFIED LEAKAGE set forth in Technical Specifications need not be reported under this item.

6.9.5 Unique Reporting Requirements

a. Environmental Radiological Monitoring

A report on the Environmental Radiological Monitoring Program for the previous twelve months of operation shall be submitted as a separate document within ninety days after January 1 of each year.

(1) For each medium sampled during the reporting period, e.g., air, baybottom, surface water, soil, fish; include:

(a) Number of sampling locations,

(b) Total number of samples,

(c) Number of locations at which levels are found to be significantly above local backgrounds, and

(d) Highest, lowest, and average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive materials in environmental media as determined by an environmental monitoring program indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

b. Semiannual Effluent Release Report

Within 60 days after January 1 and July 1 of each year a report shall be submitted covering the radioactive content of effluents released to unrestricted areas during the previous six months of operation. The data shall be summarized on a monthly basis and included as a minimum:

(1) Gaseous Effluents

(a) Gross Radioactivity Releases

(i) Total gross radioactivity (in curies), primarily noble and activation gases.

(ii) Maximum gross radioactivity release rate during any one-hour period.

(iii) Total gross radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(iv) Percent of Technical Specification limit.

(b) Iodine Release

(i) Total iodine radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(ii) Percent of Technical Specification limit for I-131 released.

(c) Particulate Releases

(i) Total gross radioactivity (β,γ) released (in curies) excluding background radioactivity.

(ii) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(iii) Total gross radioactivity (in curies) of nuclides with half-lives greater than 8 days.

(iv) Percent of Technical Specification limits for particulate radioactivity with half-lives greater than 8 days.

(2) Liquid Effluents

(a) Total gross radioactivity (β,γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

(b) The maximum concentration of gross radioactivity (β,γ) released to the unrestricted area (averaged over the period of release).

(c) Total tritium and total alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(d) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(e) Total volume (in liters) of liquid waste released.

(f) Total volume (in liters) of dilution water used prior to release from the restricted area.

(g) Total gross radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(h) Percent of Technical Specification limit for total radioactivity.

(3) Solid Wastes

(a) The total amount of solid waste shipped (in cubic feet).

(b) The total estimated radioactivity (in curies) involved.

(c) Disposition including date and destination.

6.9.6 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inservice Inspection Program Reviews, Specification 4.4.9.1.

b. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.3.

d. Sealed Source leakage in excess of limits, Specification 4.7.6.3.

e. Radioactive Solid Waste Disposal, Specification 3.7.9.1.

6.9 REPORTING REQUIREMENTS (Continued)

NOTES

* The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

** The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

*** This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacement of principla items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCE reports submitted to the COMMISSION.

d. Records of surviellance 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 Hazards Summary Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.8-1.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the NSAR Committee.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposures.

6.12 RESPIRATORY PROTECTION PROGRAM

ALLOWANCE

6.12.1 Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table 1 Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table 1, Column I, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in §20.101. These materials shall be subject to applicable process and other engineering controls.

PROTECTION PROGRAM

6.12.2 In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in 6.12.1 above, are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

   1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protection equipment.

   2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

   3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for OPERABILITY immediately prior to use.

   4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

   5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

   6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U.S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U.S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

REVOCATION

6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1.a above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the (Shift Supervisor) on duty.

TABLE 6.12-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION (7)	MODES(1)	PROTECTION FACTORS(2) PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE(3)	GUIDES TO SELECTION OF EQUIPMENT* BUREAU OF MINES/NATIONAL INSTITUTE FOR OCCUPATIONAL SAFETY AND HEALTH APPROVALS (*or schedule superseding for equipment type listed.)
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask(4),	NP	5	30 CFR Part 11 Subpart K
Facepiece, full	NP	100	30 CFR Part 11 Subpart K
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	30 CFR Part 11 Subpart J
Facepiece, full	CF	1,000	30 CFR Part 11 Subpart J
Facepiece, full	D	100	30 CFR Part 11 Subpart J
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart J
Hood	CF	(5)	30 CFR Part 11 Subpart K
Suit	CF	(5)	(6)
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full	D	100	30 CFR Part 11 Subpart H
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart H
Facepiece, full	R	100	30 CFR Part 11 Subpart H
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	30 CFR Part 11 § 11.63(b)

TABLE 6.12-1 (Continued)

TABLE NOTATION

(1) See the following symbols:

CF: continuous flow
D: demand
NP: negative pressure (i.e., negative phase during inhalation)
PD: pressure demand (i.e., always positive pressure)
R: recirculating (closed circuit)

(2) (a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency [above 99.9% removal efficiency by U.S. Bureau of Mines type dioctyl phthalate (DOP) test] particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

(3) Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5, below, concerning supplied-air suits and hoods.

TABLE 6.12-1 (Continued)

TABLE NOTATION

(4) Under chin only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

(5) Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

(6) No approval schedules current available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

(7) Only for shaven faces and where nothing interferes with the seal of tight fitting facepieces against the skin. Hoods and suits are excepted.

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines or the National Institute for Occupational Safety and Health according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Bureau of Mines or the National Institute for Occupational Safety and Health in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.