UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by the Virginia Electric and Power Company (the licensee) dated May 1, June 2, July 14, September 12 and October 24, 1978, January 12, April 17, April 23, August 6, September 4, September 13, September 20, October 10, October 15, October 23, October 24, November 2, November 9, November 29, December 6 and December 10, 1979, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

1751 233

8001160 293

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   2.D.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, the following paragraphs of Facility Operating License No. NPF-4 are hereby deleted:

   Paragraph 2.D.(3).b
   Paragraph 2.D.(3).h
   Paragraph 2.D.(3).i
   Paragraph 2.D.(3).k
   Paragraph 2.D.(3).l
   Paragraph 2.D.(3).m

4. Additionally, paragraph 2.D.(3).j of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   2.D.(3).j The Virginia Electric and Power Company shall modify or replace the presently installed Barton Models No. 763 and No. 764 Lot 1 Transmitters used in safety-related circuits inside containment with transmitters that have been demonstrated to provide a greater tolerance to harsh environments. The modifications or replacement of these transmitters shall be completed prior to startup after the second refueling outage.

5. Also, new paragraph 2.D.(3).o is hereby added to Facility Operating License No. NPF-4 to read as follows:

   2.D.(3).o The Virginia Electric and Power Company shall perform a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

   1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

   2. Identification of the procedures used to quantify parameters that are critical to control points;

   3. Identification of process sampling points;

   4. Procedure for the recording and management of data;

   5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

6. The license amendment is effective as of the date of issuance except for License Paragraph 2.D.(3).j.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 28, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

III
V
VIII
XIII
XV
1-1
1-2
1-3
2-9
3/4 1-9
3/4 1-11
3/4 1-12
3/4 1-15
3/4 1-16
3/4 1-18
3/4 1-19
3/4 1-20
3/4 1-21
3/4 1-22
3/4 1-28
3/4 2-3
3/4 2-5
3/4 2-7
3/4 2-8
3/4 2-11
3/4 2-12
3/4 2-13
3/4 2-16
3/4 3-10
3/4 3-16
3/4 3-20a (added)
3/4 3-22
3/4 3-23
3/4 3-24
3/4 3-26
3/4 3-26a

Pages

3/4 3-28
3/4 3-29a (added)
3/4 3-30
3/4 3-31
3/4 3-32
3/4 3-33
3/4 3-52
3/4 4-3
3/4 4-26
3/4 4-27
3/4 4-28
3/4 4-29
3/4 4-31 (added)
3/4 4-32 (added)
3/4 4-33 (was 3/4 4-31)
3/4 4-34 (was 3/4 4-31a)
3/4 4-35 (was 3/4 4-32)
3/4 5-3
3/4 5-6
3/4 5-6a (added)
3/4 5-9
3/4 6-23
3/4 6-25
3/4 6-29
3/4 6-33
3/4 6-35
3/4 6-36
3/4 7-11
3/4 7-12
3/4 7-13
3/4 7-14
3/4 7-15
3/4 7-20
3/4 7-21
3/4 7-22
3/4 7-23

Pages

3/4 7-24
3/4 7-25
3/4 7-53a
3/4 7-53b
3/4 7-68
3/4 7-69
3/4 7-79
3/4 9-10
3/4 10-1
3/4 10-2
3/4 10-3
3/4 10-5
B 3/4 1-3
B 3/4 1-4
B 3/4 2-1
B 3/4 2-4
B 3/4 4-1
B 3/4 4-10
B 3/4 4-11
B 3/4 4-12
B 3/4 5-2
B 3/4 5-3 (added)
B 3/4 7-3
B 3/4 7-4
B 3/4 7-7
B 3/4 7-7a (added)
5-4
6-1
6-10
6-13
6-21
6-22

1751 236

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

1751 237

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

1751 238

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

1751 239

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

1751 240

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

1751 243

BASES

SECTION PAGE

1751 244

BASES

SECTION		PAGE

1751 245

BASES

1751 246

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 2775 MWt.

OPERATIONAL MODE - MODE

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

ACTION

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

OPERABLE - OPERABILITY

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

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.8 and 6.9.1.9.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

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

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

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

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

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

CORE ALTERATION

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

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

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

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

UNIDENTIFIED LEAKAGE

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

1751 249

1751 250

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

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

QUADRANT POWER TILT RATIO

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

DOSE EQUIVALENT I-131

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

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

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

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.141	K_1 = ( )	K_1 = ( )
K_2 = 0.0128	K_2 = ( )	K_2 = ( )
K_3 = 0.000608	K_3 = ( )	K_3 = ( )

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

(i) for $q_t - q_b$ between - 34 percent and + 10 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 34 percent, the ΔT trip setpoint shall be automatically reduced by 3 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 10 percent, the ΔT trip setpoint shall be automatically reduced by 1.25 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

1751 251

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4-K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T'')-f_2(\Delta I)]$

where:

ΔT_o	=	Indicated ΔT at rated power
T	=	Average temperature, °F
T''	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 580.3°F
K_4	=	1.086
K_5	=	0.02/°F for increasing average temperature
K_5	=	0 for decreasing average temperatures
K_6	=	0.00116 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	0 for all ΔI

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% Δk/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

1751 253

SURVEILLANCE REQUIREMENTS (Continued)

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

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the switches in the Control Room have been placed in the pull to lock position.

1751 255

1751 256

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour following heatup above 320°F or prior to cooldown below 320°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The above required charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of $\geq$ 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F by verifying that the switches in the Control Room have been placed in the pull to lock position.

---

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

1751 256

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

1. A minimum contained borated water volume of 835 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,000 gallons,
2. Between 2000 and 2100 ppm of boron, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

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

1751 257

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

1. A contained borated water volume of between 4450 and 16,280 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A contained borated water volume of between 475,058 and 487,000 gallons,

2. Between 2000 and 2100 ppm of boron, and

3. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

1751 258

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*

ACTION:

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

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

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

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

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

a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions, and

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

e) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod within the 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 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Rupture (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

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

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

1751 263

POSITION INDICATOR CHANNELS-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one rod postion indicator channel (excluding demand position indication) shall be OPERABLE for each shutdown or control rod not fully inserted.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip system breakers in the closed position.
#See Special Test Exception 3.10.5.

1751 264

This page left blank pending NRC approval
of ECCS evaluation for two loop operation.

FIGURE 3.1-2 ROD GROUP INSERTION LIMITS VERSUS
THERMAL POWER TWO LOOP OPERATION

1751 265

THIS PAGE INTENTIONALLY LEFT BLANK

SURVEILLANCE REQUIREMENTS (Continued)

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

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

1751 267

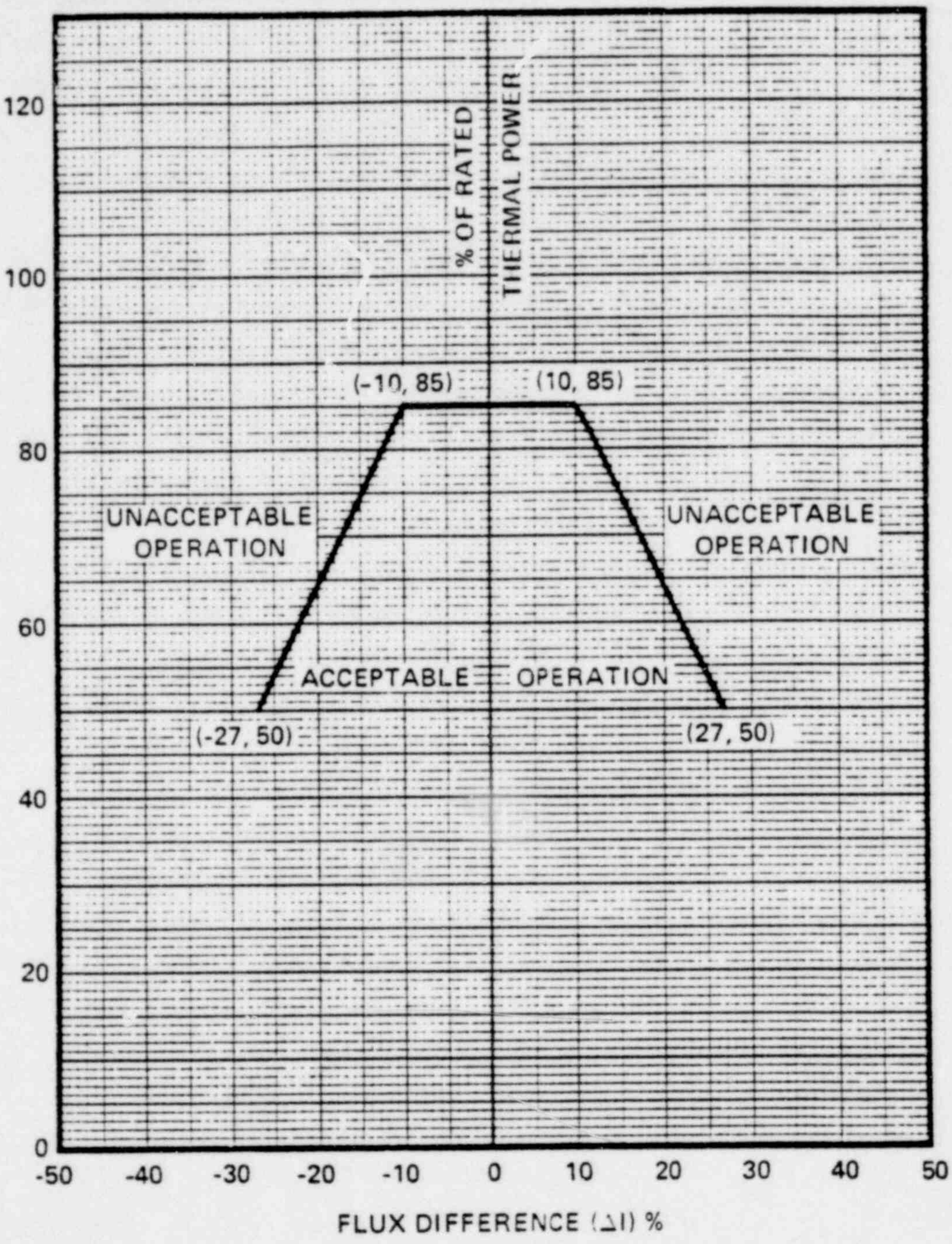


FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \left[\frac{2.10}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.20] [K(Z)] \text{ for } P \leq 0.5$$

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

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

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

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

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\overline{R}$.

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

1751 26

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

2. The relationship:

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

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

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

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

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

1751 270

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods,

2. $F_{xy}^{RTP} \leq 1.65$ for all unrodded core planes from 0 to 65% of core height, and

3. $F_{xy}^{RTF} \leq 1.57$ for all unrodded core planes above 65% of core height.

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

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

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

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

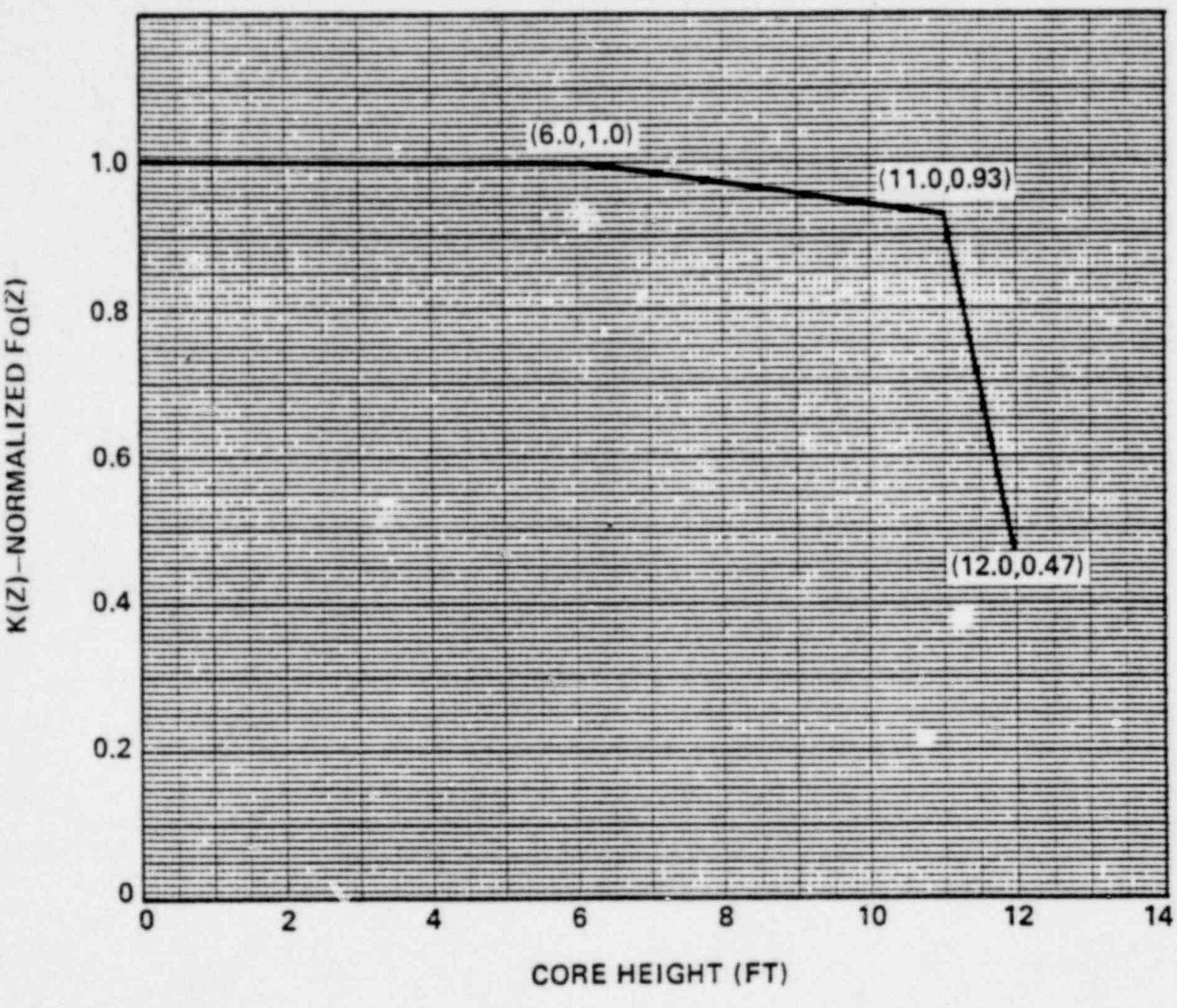


Figure 3.2-2 K(Z)–Normalized $F_Q(Z)$ as a Function of Core Height

1751 272

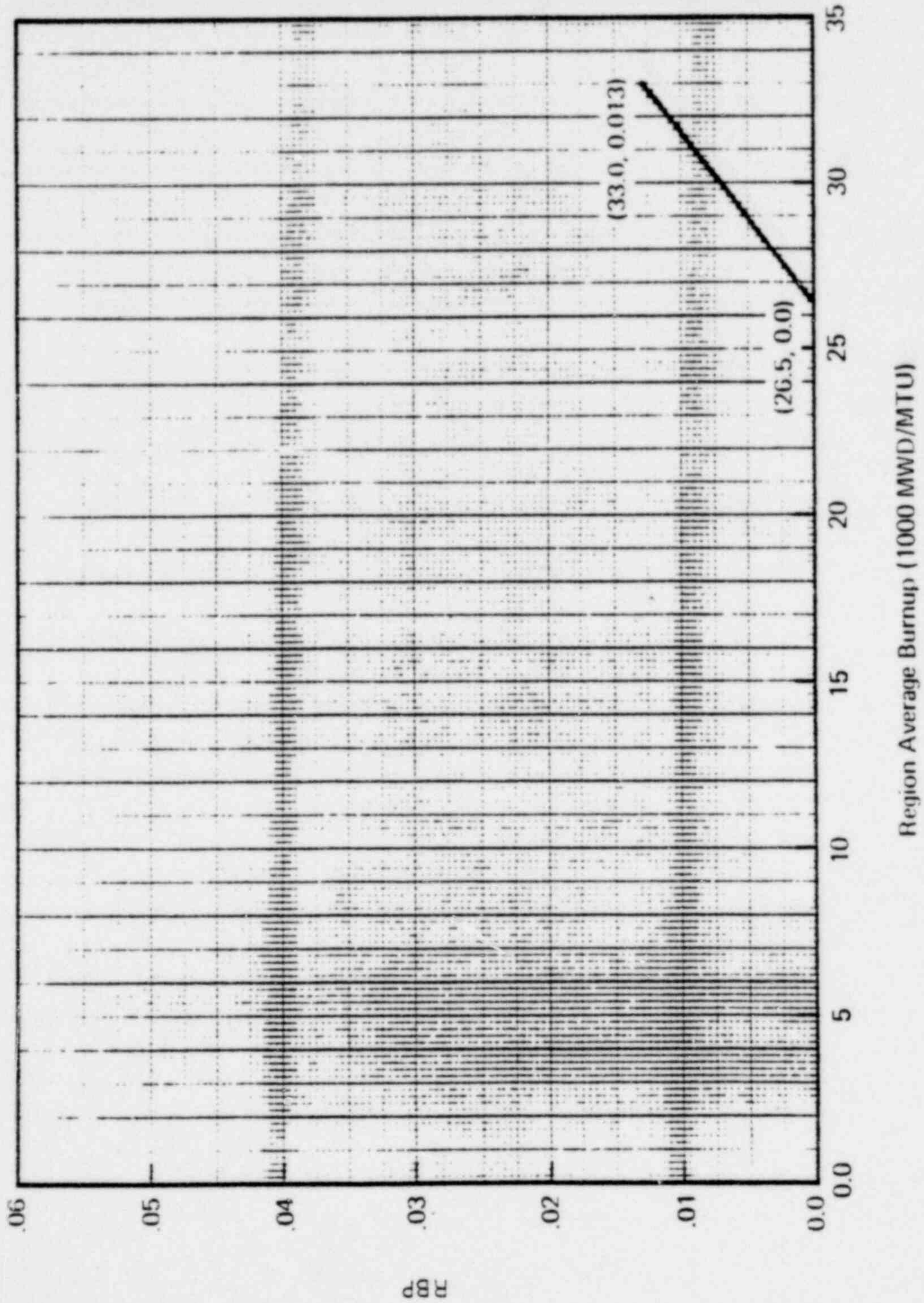


Figure 3.2-3 Rod Bow Penalty Fraction Versus Region Average Burnup

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

   1. Within 2 hours:

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

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

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

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

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

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

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

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

1751 276

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops In Operation	2 Loops In Operation** & Loop Stop Valves Open	2 Loops In Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 585°F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 278,400 gpm		

*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.

**Values dependent on NRC approval of ECCS evaluation for these conditions

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.10]\,[K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

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

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns above 95% of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}\,(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

1751 278

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and	8%	>7%	Prevents reactor trip on: Low flow or reactor coolant pump breakers open in more than one loop, Undervoltage (RCP busses), Underfrequency (RCP busses), Turbine Trip, Pressurizer low pressure, and Pressurizer high level.
	2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8%	>7%	
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Permit reactor trip on low flow or reactor coolant pump breaker open in a single loop.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Blocks reactor trip on low flow or reactor coolant pump breaker open in a single loop.

1751 279

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 4.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

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

1751 280

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

1751 281

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14*
d. Pressurizer Pressure-Low-Low	3	2	2	1, 2, 3#	14*
e. Differential Pressure Between Steam Lines - High				1, 2, 3##	
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		14*
Two Loops Operating	3/operating steam line	2###/steam line twice in either operating steam line	2/operating steam line		15
f. Steam Flow in Two Steam Lines-High				1, 2, 3##	

1751 282

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	19*
b. 4.16 Kv Emergency Bus Undervoltage (Grid Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	19*

1751 283

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

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

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

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

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

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

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

1751 284

TABLE 3.3-3 (Continued)

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour and the Minimum Channels OPERABLE requirement is met, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

1751 285

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels above setpoint	2000 psig	$\leq$ 2010 psig	P-11 prevents manual block of safety injection actuation on low-low pressurizer pressure.
	With 2 of 3 pressurizer pressure channels below setpoint	1980 psig	$\leq$ 1990 psig	P-11 allows manual block of safety injection actuation on low-low pressurizer pressure.
P-12	With 2 of 3 T_{avg} channels above setpoint	543°F (Nominal)	$\leq$ 545°F	P-12 prevents manual block of safety injection actuation on high steam line flow.
	With 2 of 3 T_{avg} channels below setpoint	543°F (Nominal)	$\geq$ 541°F	P-12 allows manual block of safety injection actuation on high steam line flow.

1751 286

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	$\leq$ 17 psia	$\leq$ 19 psia
d. Pressurizer Pressure -- Low-Low	$\geq$ 1765 psig	$\geq$ 1755 psig
e. Differential Pressure Between Steam Lines--High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	$\leq$ A function defined as follows: a Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load $T_{avg} \geq 543°F$ $\geq$ 600 psig steam line pressure	$\leq$ A function defined as follows: a Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load $T_{avg} \geq 541°F$ $\geq$ 580 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 25 psia	$\leq$ 27 psia
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High	$\leq$ 25 psia	$\leq$ 27 psia

1751 288

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--Intermediate High-High	$\leq$ 20 psia	$\leq$ 22 psia
d. Steam Flow in Two Steam lines--High Coincident with T_{avg}--Low-Low Or Steam Line Pressure--Low	$\leq$ A function defined as follows: a Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: a Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
	$T_{avg} \geq 543°F$	$T_{avg} \geq 541°F$
	$\geq$ 600 psig steam line pressure	$\geq$ 580 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water level--High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator

1751 289

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUES
6.	AUXILIARY FEEDWATER PUMP START		
a.	Steam Generator Water Level Low-Low	≥ 5% of narrow range instrument span each steam generator	≥ 4% of narrow range instrument span each steam generator
b.	S.I.	See 1 above (All S.I. Setpoints)	
c.	Station Blackout	≥ 57.5% Transfer Bus Voltage	≥ 52.5% Transfer Bus Voltage
d.	Trip of Main Feed Pump	N.A.	N.A.
7.	LOSS OF POWER		
a.	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2999 ± 60 volts with a 2.2 ± 0.03 second time delay	2912 ± 60 volts with a 3 ± 0.03 second time delay
b.	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	3744 ± 1.4 volts with a 60 ± 3 second time delay	3619 ± 1.4 volts with a 75 ± 3 second time delay

1751 290

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0*
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{\#}/28.0^{\#\#}$
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable

1751 291

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure -- Low-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/13.0#
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq$ 15.0#/25.0##
b. Reactor Trip (from SI)	$\leq$ 5.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq$ 20.0#/30.0##
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	$\leq$ 10.0

1751 292

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Quench Spray	$\leq$ 60.0
b. Containment Isolation-Phase "B"	$\leq$ 60.0
8. Containment Pressure-Intermediate High-High	
a. Steam Line Isolation	$\leq$ 7.0
9. Steam Generator Water Level Low-Low	
a. Auxiliary Feedwater Pumps	$\leq$ 60.0
10. Station Blackout	
a. Auxiliary Feedwater Pumps	Not Applicable
11. Main Feedwater Pump Trip	
a. Auxiliary Feedwater Pumps	Not Applicable
12. Steam Generator Water Level--High High	
a. Turbine Trip - Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0

POOR ORIGINAL

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
13. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$ 13.3###
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	$\leq$ 11.5### with SI signal $\leq$ 64.0### with no SI signal

1751 294

TABLE 3.3-5 (Continued)

TABLE NOTATION

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

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

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

The response times shown are based on the time from when the signal reaches the trip setting until the diesel generator is supplying the emergency bus.

1751 295

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

1751 296

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M(3)	1, 2, 3
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--Intermediate High-High	S	R	M	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low-Low or Steam Line Pressure--Low	S	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3, 4
b. S.I.	See 1 above (all S.I. Surveillance Requirements)			
c. Station Blackout	N.A.	R	N.A.	1, 2, 3, 4
d. Main Feedwater Pump Trip	N.A.	N.A.	R	1, 2
7. LOSS OF POWER 4.16 KV Emergency Bus				
a. Loss of Voltage	N.A.	R	M	1, 2, 3
b. Degraded Voltage	N.A.	R	M	1, 2, 3

1751 298

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

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

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

1751 299

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature-T_{hot} (wide range)	M	R
3. Reactor Coolant Inlet Temperature-T_{cold} (wide range)	M	R
4. Reactor Coolant Pressure-Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level-Narrow Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Boric Acid Tank Solution Level	M	R

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

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

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11 inoperable:

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

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

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

4.3.3.7.3 The non-supervised circuits between the local panels in Specification 4.3.3.7.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

1751 301

ACTION (Continued)

Below P-7#

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. The P-8 interlock setpoint is within the following limits if the P-8 interlock was reset for 2 loop operation:

1. $\leq$ 71% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed, or

2. $\leq$ 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open.

4.4.1.1.2 At least once per 31 days, with the reactor coolant loops in operation verify that power is removed from the loop stop valve operators.

*All reactor coolant pumps and residual heat removal pumps may be deenergized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

#A reactor coolant pump shall not be started with one or more the RCS cold leg temperatures less than or equal to 320°F unless 1) the pressurizer water volume is less than 457 cubic feet or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

1751 302

REACTOR COOLANT SYSTEM

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, unless the loop has been drained for maintenance.

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

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated 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 RCS borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F.

SURVEILLANCE REQUIREMENTS

4.4.1.2 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.

1751 303

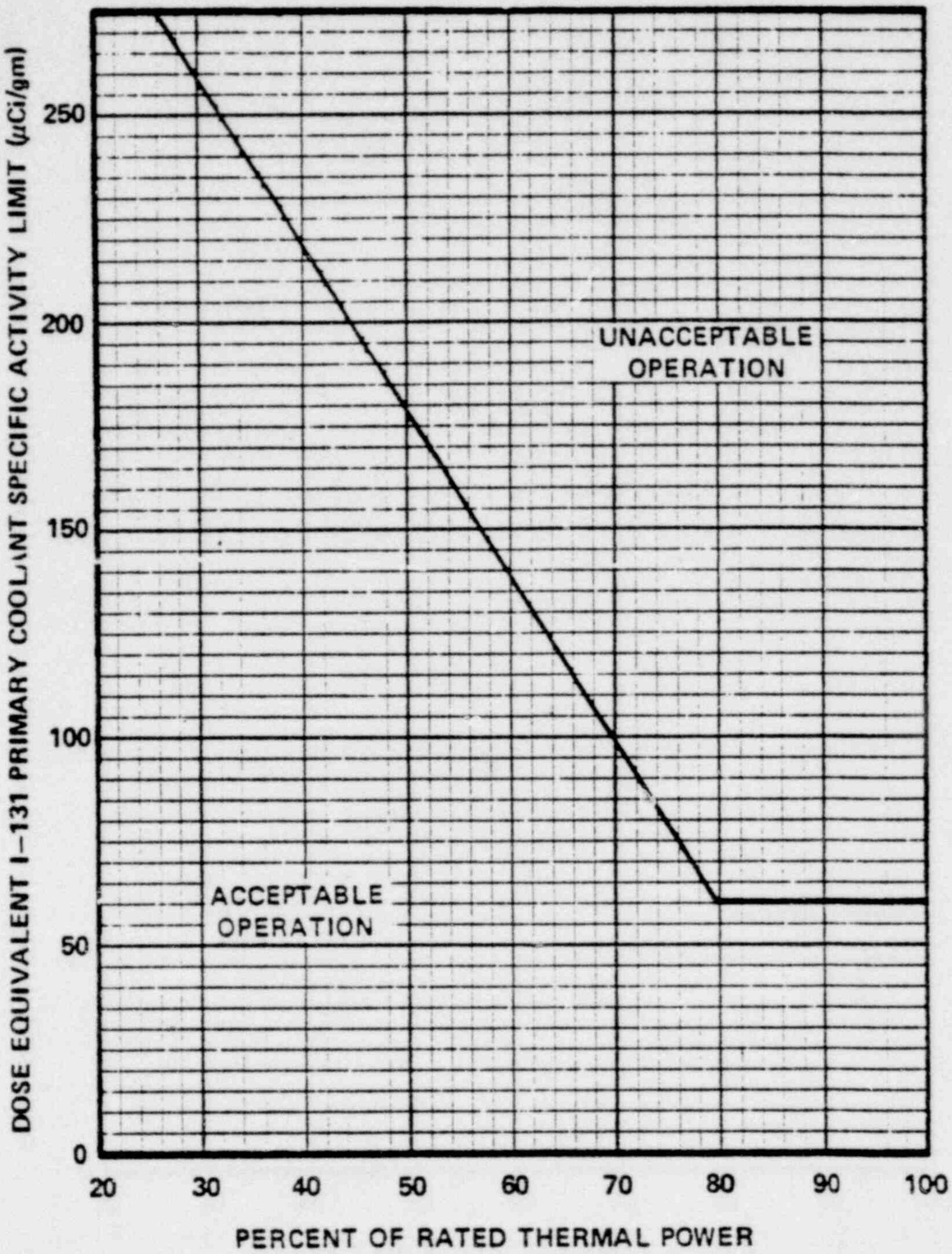


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity $> 1.0 \mu$Ci/gram Dose Equivalent I-131

1751 304

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

1751 305

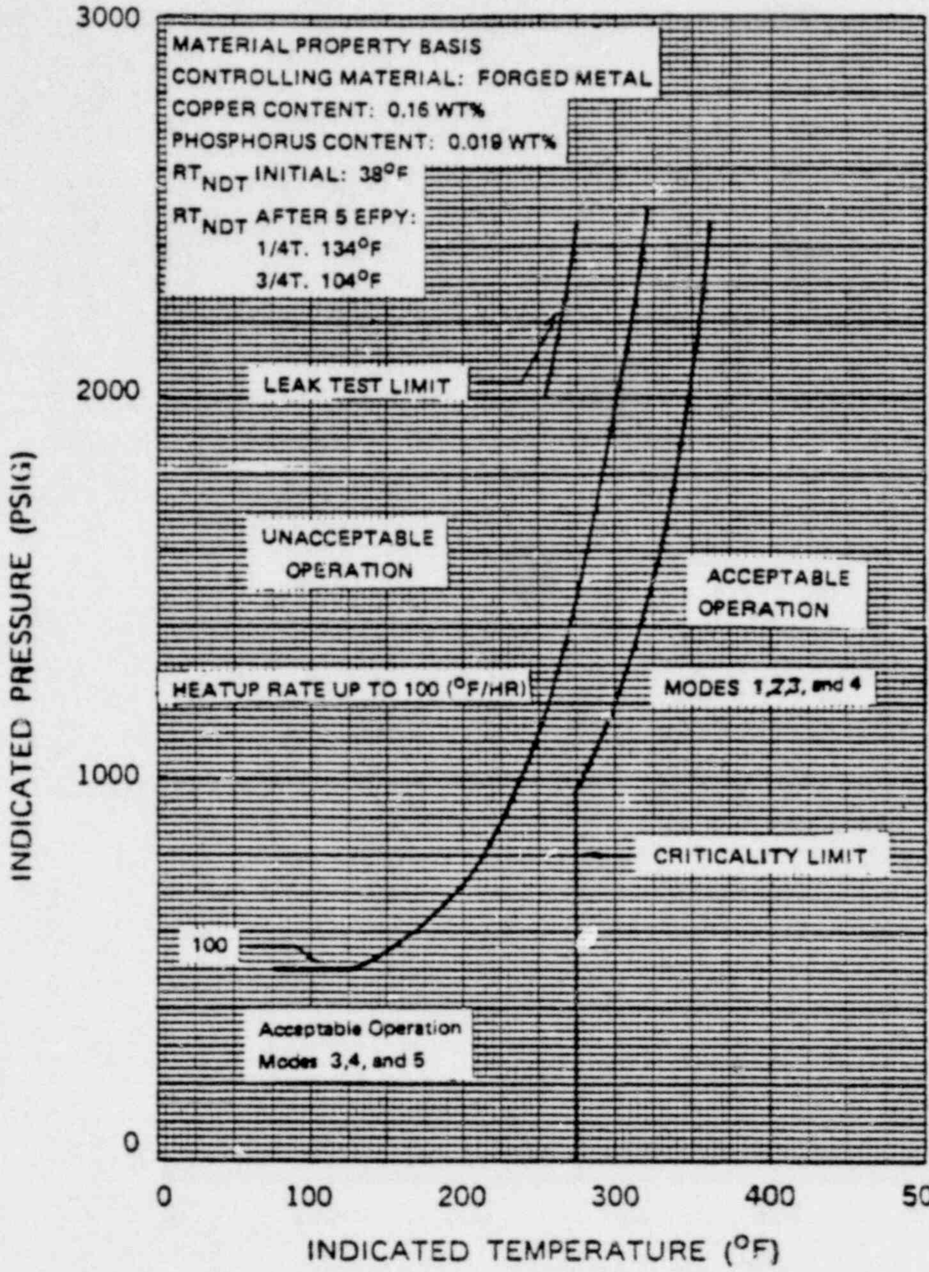


Figure 3.4-2 Reactor Coolant System Temperature-Pressure Heatup Limitations

1751 306

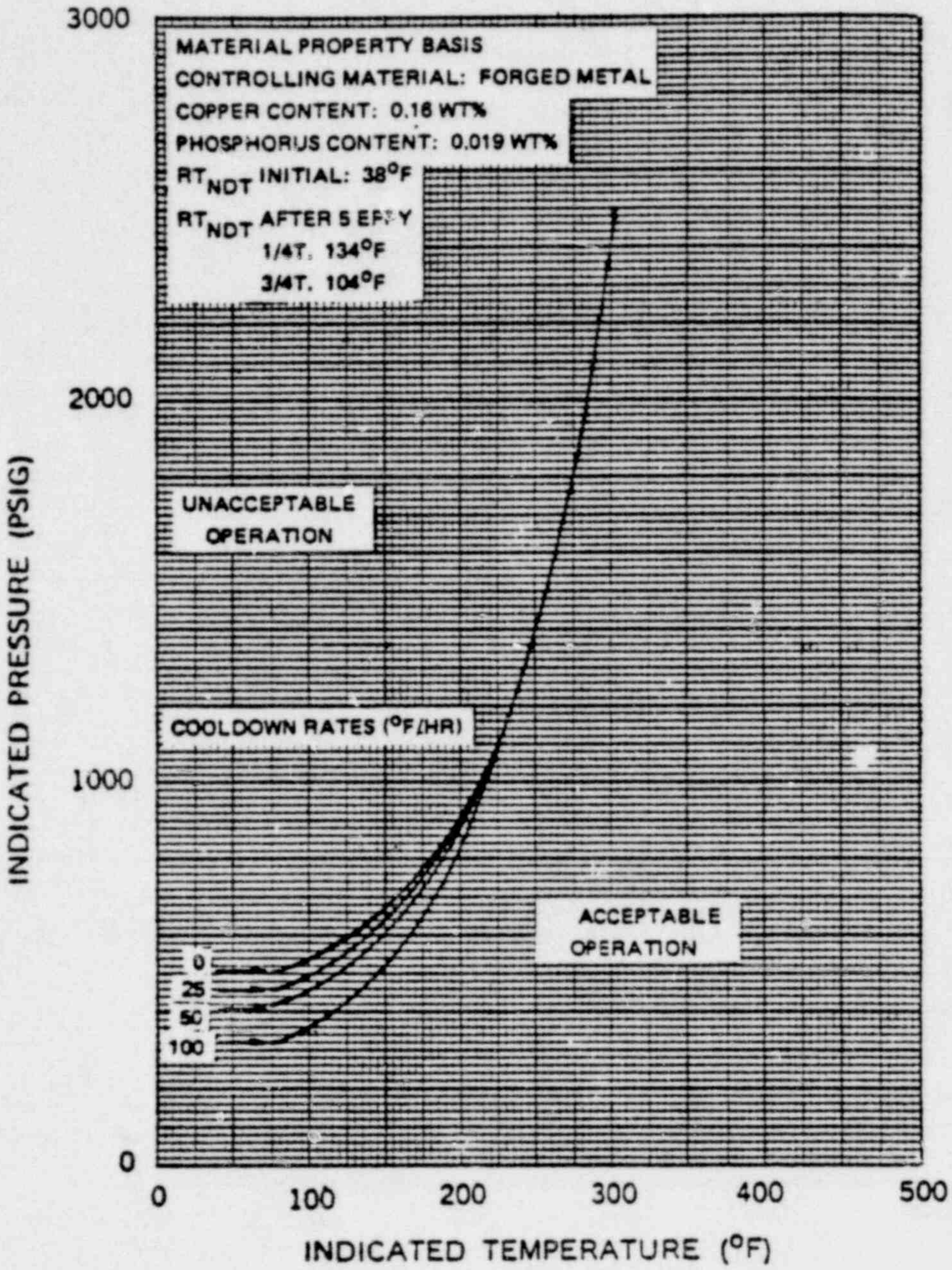


Figure 3.4-3 North Anna Power Station No. 1 Reactor Coolant System Cooldown Limitations

1751 307

THIS PAGE INTENTIONALLY LEFT BLANK

1751 308

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F or cooldown of 200°F, in any one hour period, and

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

1751 309

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of: 1) less than or equal to 505 psig whenever any RCS cold leg temperature is less than or equal to 320°F, and 2) less than or equal to 430 psig whenever any RCS cold leg temperature is less than 140°F, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 320°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

1751 310

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

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

c. Verifying the PORV keyswitch is in the Auto position and the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Testing in accordance with the in service test requirements for ASME Category C valves pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

1751 311

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 & 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: ALL MODES.

ACTION:

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

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

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

d. With any RCP shaft deflection indication greater than 20 mils, the reactor shall be placed in at least HOT STANDBY within 1 hour, the affected RCP(s) tripped and then affected flow straightener plate(s) ultrasonically examined.

e. The provisions of Specification 3.0.4 are not applicable.

1751 312

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5, 1) the Reactor Coolant pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 and 2) the flow straighteners in each steam generator-to-RCP elbow shall be ultrasonically examined whenever a RCP shaft deflection of greater than 20 mils is indicated and at least once per 18 months.

1751 313

LIMITING CONDITION FOR OPERATION

3.4.10.2 The temperature of the steam generator supports shall be maintained:

a. > 225°F for A572 material monitored at a middle level corner during operation and at a top level corner during heatup of the supports.

b. < 355°F at the monitored top level corner.

c. > 85°F for A36 material monitored at a bottom level corner during heatup.

APPLICABILITY: With pressurizer pressure > 1000 psig.

ACTION: With the temperature of any steam generator support outside the above limits, restore the temperature to within the limit within 4 hours or be below 1000 psig within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.2.1 The steam generator support temperatures for A572 material shall be verified to be within the specified limits at least once per 12 hours.

4.4.10.2.2 The steam generator support temperatures for A36 material shall be verified to be within the specified limit prior to exceeding a pressurizer pressure of > 1000 psig.

4.4.10.2.3 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumlated actuation cycles to date.

c. The provisions of Specification 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 320°F or prior to cooldown below 320°F.

1751 315

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
a. MOV-1890A	a. LHSI to hot leg	a. closed
b. MOV-1890B	b. LHSI to hot leg	b. closed
c. MOV-1836	c. Ch pump to cold leg	c. closed
d. MOV-1869A	d. Ch pump to hot leg	d. closed
e. MOV-1869B	e. Ch pump to hot leg	e. closed

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

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

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

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

d. At least once per 18 months by:

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

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

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

1751 316

SURVEILLANCE REQUIREMENTS (Continued)

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

   a) Centrifugal charging pump, and

   b) Low head safety injection pump.

f. By verifying that each of the following pumps develop the indicated discharge pressure (after subtracting suction pressure) on recirculation flow when tested pursuant to Specification 4.0.5.

   1. Centrifugal charging pump $\geq$ 2410 psig.

   2. Low head safety injection pump $\geq$ 156 psig

g. At least once per 18 months, during reactor shutdown, verify that the following manual valves requiring adjustment to prevent pump "runout" and subsequent component damage are locked and tagged in the proper position for injection:

1.	1-SI-188	Loop A Cold Leg
2.	1-SI-191	Loop B Cold Leg
3.	1-SI-193	Loop C Cold Leg
4.	1-SI-203	Loop A Hot Leg
5.	1-SI-204	Loop B Hot Leg
6.	1-SI-205	Loop C Hot Leg

1751 317

ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank upon being manually realigned or from the containment sump when the suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODE 4.

ACTION:

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

1751 318

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F by verifying that the switches in the Control Room are in the pull to lock position.

1751 319

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F by verifying that the switches in the Control Room are in the pull to lock position.

1751 320

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
26.	TV-IA102B	Instrument Air to Reactor Containment	60
C.	CONTAINMENT PURGE AND EXHAUST (VENTILATION DUCTS)		
1.	MOV-HV100A*	Purge Supply	N.A.
2.	MOV-HV100B*	Purge Supply	N.A.
3.	MOV-HV102*	Alternate Supply	N.A.
4.	MOV-HV100C*	Purge Exhaust	N.A.
5.	MOV-HV100D*	Purge Exhaust	N.A.
6.	MOV-HV101*	Bypass	N.A.

1751 521

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
D.	MANUAL		
1.	1-SI-58*	Safety Injection Accumulator Make Up	NA
2.	1-RH-36*	Residual Heat Removal System to Refueling Water Storage Tank	NA
3.	1-RH-37*	Residual Heat Removal System to Refueling Water Storage Tank	NA
4.	1-HC-12*	Discharge From Atmosphere Clean-up System (Hydrogen Recombiner)	NA
5.	1-HC-31*	Discharge From Atmosphere Clean-up System (Hydrogen Recombiner)	NA
6.	1-HC-16*	Discharge From Atomsphere Clean-up System (Hydrogen Recombiner)	NA
7.	1-HC-28*	Discharge From Atmosphere Clean-up System (Hydrogen Rcombiner)	NA
8.	1-DA-39*	Primary Vent Pot Vent	NA
9.	1-DA-41*	Primary Vent Pot Vent	NA
10.	1-CH-310#*	Reactor Coolant Pump Seal Water Supply	NA
11.	1-CH-318#*	Reactor Coolant Pump Seal Water Supply	NA

1751 322

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
12.	1-CH-314#*	Reactor Coolant Pump Seal Water Supply	NA
13.	1-SA-29*	Service Air	NA
14.	1-SA-2*	Service Air	NA
15.		(Deleted)	
16.	NA*	Fuel Transfer (Tube Penetration #65)	NA
17.	1-CV-4*	Air Ejector Suction	NA
18.	1-RC-176*	Dead Weight Pressure Calibrator	NA
19.	1-RC-178*	Dead Weight Pressure Calibrator	NA
20.	1-RP-26*	Refueling Purification Inlet	NA
21.	1-RP-28*	Refueling Purification Inlet	NA
22.	1-RP-6*	Refueling Purification Inlet	NA
23.	1-RP-8*	Refueling Purification Inlet	NA
24.	1-WT-354#*	Chemical Feed Lines	NA
25.	1-WT-357#*	Chemical Feed Lines	NA
26.	1-WT-351#*	Chemical Feed Lines	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC.)
E. REMOTE MANUAL		
1. MOV-QS101A*	Quench Spray Pump Discharge	NA
2. MOV-QS101B*	Quench Spray Pump Discharge	NA
3. MOV-RS155A#*	Recirc. Spray Pump Suction	NA
4. MOV-RS155B#*	Recirc. Spray Pump Suction	NA
5. MOV-1860A#*	LHSI Pump Suction From Containment Sump	NA
6. MOV-1860B#*	LHSI Pump Suction From Containment Sump	NA
7. MOV-RS156A*	Recirculation Spray Pump Discharge	NA
8. MOV-RS156B*	Recirculation Spray Pump Discharge	NA
9. MOV-SW103A*	Service Water to Recirculation Spray Coolers	NA
10. MOV-SW103B*	Service Water to Recirculation Spray Coolers	NA
11. MOV-SW103C*	Service Water to Recircualtion Spray Coolers	NA
12. MOV-SW103D*	Service Water to Recirculation Spray Coolers	NA
13. MOV-SW104A*	Service Water to Recirculation Spray Coolers	NA
14. MOV-SW104B*	Service Water to Recirculation Spray Coolers	NA
15. MOV-SW104C*	Service Water to Recirculation Spray Coolers	NA
16. MOV-SW104D*	Service Water to Recirculation Spray Coolers	NA

1751 324

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
14.	1-HC-18	Discharge From Containment Atmosphere Cleanup System (Hydrogen Recombiner)	NA
15.	1-HC-14	Discharge From Containment Atmosphere Cleanup System (Hydrogen Recombiner)	NA
16.	1-CH-380#	Reactor Coolant Pump Seal Water Supply	NA
17.	1-CH-336#	Reactor Coolant Pump Seal Water Supply	NA
18.	1-CH-358#	Reactor Coolant Pump Seal Water Supply	NA
19.	1-IA-149	Air Radiation Monitor Return	NA
20.	1-RC-149	Primary Grade Water to Pressurizer Relief Tank	NA
21.	1-CH-330	Loop Fill Header	NA
22.	1-IA-55	Instrument Air Line	NA
23.	1-SI-106	Nitrogen to Pressurizer Relief Tank and SI Accumulators	NA
24.	1-SI-206	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
25.	1-SI-207	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
26.	1-SI-195	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
27.	1-SI-197	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
28.	1-SI-199	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
29.	1-QS-19**	Quench Spray Pump Discharge	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	ISOLATION TIME (SEC.)
30.	1-QS-11**	Quench Spray Pump Discharge	NA
31.	1-RS-27**	Recirculation Spray Pump Discharge	NA
32.	1-RS-18**	Reciruclation Spray Pump Discharge	NA
33.	1-VP-12	Air Ejector Vent	NA
34.	1-SI-90	High Head Safety Injection to RCS Except Boron Injection Line	NA
35.	1-SI-201	High Head Safety Injection to RCS Except Boron Injection Line	NA
36.	1-SI-85	High Head Safety Injection to RCS Except Boron Injection Line	NA
37.	1-FW-47#	Feedwater to Steam Generators	NA
38.	1-FW-111#	Feedwater to Steam Generators	NA
39.	1-FW-79#	Feedwater to Steam Generators	NA
40.	1-WT-50#	Chemical Feed Lines	NA
41.	1-WT-66#	Chemical Feed Lines	NA
42.	1-WT-38#	Chemical Feed Lines	NA
43.	1-FW-68#	Auxiliary Feedwater to Steam Generator	NA
44.	1-FW-100#	Auxiliary Feedwater to Steam Generator	NA
45.	1-FW-132#	Auxiliary Feedwater to Steam Generator	NA

1751 326

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers (shared with Unit 2) shall be CPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLLANCE REQUIREMENTS

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

a. One volume percent ($\pm$ .25%) hydrogen, balance nitrogen, and

b. Four volume percent ($\pm$ .25%) hydrogen, balance nitrogen.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two separate and independent containment hydrogen recombiner systems (shared with Unit 2) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to $\geq$ 700°F within 90 minutes and is maintained for at least 2 hours and that each purge blower operates for 15 minutes.

b. At least once per 18 months by:

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

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

3. Verifying during a recombiner system functional test using containment atmospheric air at a flow rate of $\geq$ 50 scfm, that the heater temperature increases to $\geq$ 1100°F within 5 hours and is maintained for at least 4 hours.

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

1751 328

WASTE GAS CHARCOAL FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A waste gas charcoal filter system (shared with Unit 2) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the waste gas charcoal filter system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The waste gas charcoal filter system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Initiating flow through the HEPA filter and charcoal adsorber train using the process vent blowers and verifying that the purge system operates for at least 15 minutes,

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

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a., C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 300 cfm $\pm$ 10% (except as shown in Specifications 4.6.4.3e. and f.).

1751 329

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b. of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a. of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 300 cfm $\pm$10% during system operation when tested in accordance with ANSI N 510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is < 8.5 inches Water Gauge while operating the filter train at a flow rate of 300 cfm $\pm$ 10%.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 300 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 300 cfm $\pm$ 10%.

1751 330

PLANT SYSTEMS

SECONDARY WATER CHEMISTRY

THIS SPECIFICATION DELETED

1751 331

THIS PAGE INTENTIONALLY LEFT BLANK

1751 332

THIS PAGE INTENTIONALLY LEFT BLANK

1751 333

STEAM TURBINE ASSEMBLY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The structural integrity of the steam turbine assembly shall be maintained.

APPLICABILITY: MODES 1 and 2

ACTION: With the structural integrity of the steam turbine assembly not conforming to the above requirement restore the structural integrity of the steam turbine prior to placing it in service.

SURVEILLANCE REQUIREMENTS

4.7.1.6 The structural integrity of the steam turbine assembly shall be demonstrated;

a. At least once per 40 months, during shutdown, by a visual and surface inspection of the steam turbine assembly at all accessible locations, and

b. At least once per 10 years, during shutdown, by disassembly of the turbine and performing a visual, surface and volumetric inspection of all normally inaccessible parts.

TURBINE OVERSPEED

LIMITING CONDITION FOR OPERATION

3.7.1.7 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODE 1, 2 and 3

ACTION: With the above required turbine overspeed protection system inoperable, within 6 hours either restore the system to OPERABLE status or isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENT

4.7.1.7.1 The provisions of Specification 4.0.4 are not applicable.

4.7.1.7.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE;

a. At least once per 7 days by cycling each of the following valves through one complete cycle.

   1. 4 Turbine Throttle valves

   2. 4 Turbine Governor valves

   3. 4 Turbine Reheat Stop valves

   4. 4 Turbine Reheat Intercept valves

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle.

c. At least once per 18 months, by performance of CHANNEL CALIBRATION on the turbine overspeed protection instruments.

d. At least once per 40 months, by disassembly of at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

1751 335

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

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

1751 336

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sinks shall be OPERABLE:

a. Service Water Reservoir with:

1. A minimum water level at or above elevation 313 Mean Sea Level, USGS datum, and

2. An average water temperature of $\leq$ 95°F as measured at the service water pump outlet.

b. The North Anna Reservoir with:

1. A minimum water level at or above elevation 244 Mean Sea Level, USGS datum, and

2. An average water temperature of $\leq$ 95°F as measured at the condenser inlet.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIRMENTS

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

4.7.5.2 Data for calculating the leakage from the service water reservoir shall be obtained and recorded at least once per 6 months.

1751 337

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the North Anna Reservoir exceeds 256 feet Mean Sea Level USGS datum, at the main reservoir spillway.

APPLICABILITY: At all times.

ACTION:

With the water level at the main reservoir spillway above elevation 256 feet Mean Sea Level USGS datum:

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

b. Initiate and complete within 36 hours, the following flood protection measures:

    1. Stop the circulating water pumps, and

    2. Close the condenser isolation valves.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at the main reservoir spillway shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 255 feet Mean Sea Level USGS datum.

b. Measurement at least once per 2 hours when the water level is equal to or above 255 feet Mean Sea Level USGS datum.

1751 338

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. The emergency ventilation system.

b. The bottled air pressurization system.

c. Two air conditioning systems.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either the emergency ventilation system or the bottled air presurization 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 at least COLD SHUTDOWN within the following 30 hours.

b. With both the emergency ventilation system and the bottled air pressurization system inoperable, restore at least one of these systems to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

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

d. With both air conditioning systems inoperable, restore at least one system to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

1751 339

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

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

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e. and f.).

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

   3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is < 6 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

1751 340

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of $\geq$ 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm $\pm$ 10%.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove $\geq$ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm $\pm$ 10%.

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 84 bottles of air (shared with Unit 2) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of $\geq$ 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is $\leq$ 120 F.

1751 341

3/4.7.8 SAFEGUARDS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two safeguards area ventilation systems (SAVS) shall be OPERABLE with:

a. one SAVS exhaust fan

b. one auxiliary building HEPA filter and charcoal adsorber assembly (shared with Unit 2)

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one SAVS 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.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each SAVS system shall be demonstrated OPERABLE:

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

1. Initiating, from the control room, flow through the auxiliary building HEPA filter and charcoal adsorber assembly and verifying that the SAVS operates for at least 10 hours with the heater on.

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

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6,300 cfm ± 10% (except as shown in Specifications 4.7.8.1e. and f.).

SURVEILLANCE REQUIREMENTS (Cont'd)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 6,300 cfm ± 10% during operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:
   1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is < 6 inches Water Gauge while operating the ventilation system at a flow rate of 6,300 cfm ± 10%.

   2. Verifying that on a Containment Hi-Hi Test Signal, the system automatically diverts its exhaust flow through the auxiliary building HEPA filter and charcoal adsorber assembly.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 6,300 cfm ±10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 6,300 cfm ± 10%.

1751 343

3/4.7.9 RESIDUAL HEAT REMOVAL SYSTEM

RHR OPERATING

LIMITING CONDITION FOR OPERATION

3.7.9.1 Two residual heat removal subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one residual heat removal subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each residual heat removal subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months by verifying automatic isolation of the RHR system from the Reactor Coolant System when the RCS pressure is above 660 psig.

b. At least once per 18 months during shutdown by cycling each of the valves in the subsystem flow path not testable during plant operation through one complete cycle of full travel.

c. At least once per 18 months by verifying that each residual heat removal pump develops a differential pressure of $\geq$ 123 psi.

1751 344

Table 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER No.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
221B	SHP-300-SB	A	No	No
222A	SHP-300-SB	A	No	Yes
222B	SHP-300-SB	A	No	No
224	SHP-300-SB	A	No	No
225A	SHP-300-SB	A	No	No
225B	SHP-300-SB	A	No	No
226	SHP-300-SB	A	No	No
227A	SHP-300-SB	A	No	No
227B	SHP-300-SB	A	No	Yes
228A	SHP-300-SB	A	No	Yes
228A	SHP-300-SB	A	No	No
229	SHP-300-SB	A	No	No

Table 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER No.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
217A	SHP-297-MSVH	A	No	Yes
217B	SHP-297-MSVH	A	No	Yes
218A	SHP-297-MSVH	A	No	Yes
218B	SHP-297-MSVH	A	No	Yes
219A	SHP-297-MSVH	A	No	Yes
219B	SHP-297-MSVH	A	No	Yes
209A	SHP-291-RCA	I	Yes	Yes
209B	[illegible]P-291-RCA	I	Yes	Yes
223A	SHP-291-RCA	I	Yes	Yes
223B	SHP-291-RCA	I	Yes	Yes
23A	SHP-290-MSVH	A	No	No
23B	SHP-290-MSVH	A	No	No

1751 346

Table 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER No.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
400A	SI-258-13	I	No	Yes
400B	SI-258-13	I	No	Yes
400C	SI-258-13	I	No	Yes
400D	SI-258-13	I	No	Yes
100A	SI-256-SG	A	No	No
100B	SI-256-SG	A	No	No
101A	SI-256-SG	A	No	No
101B	SI-256-SG	A	No	No
102A	SI-256-SG	A	No	No
102B	SI-256-SG	A	No	No
103	SI-256-SG	A	No	No
104A	SI-256-SG	A	No	NO

Table 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER No.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
30A	SHP-290-MSVH	A	No	No
30B	SHP-290-MSVH	A	No	No
38A	SHP-290-MSVH	A	No	No
38B	SHP-290-MSVH	A	No	No
203	SHP-291-RCA	I	Yes	Yes
204	SHP-291-RCA	I	Yes	Yes
205	SHP-291-RCA	I	Yes	Yes
206	SHP-291-RCA	I	Yes	Yes
207	SHP-291-RCA	I	Yes	Yes
208	SHP-291-RCA	I	Yes	Yes

TABLE 4.7-3

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
≥ 8	31 days ± 25%

*Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

**The required inspection interval shall not be lenghtened more than one step at a time, and the provisions of specification 4.0.2 are not applicable.

1751 349

3/4.7.11 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.11.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of greater than or equal to 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

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

a. The licensee, or

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

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

4.7.11.1.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive materials.

SURVEILLANCE REQUIREMENTS (Continued)

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

2. In any form other than gas.

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

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.11.1.3 Reports - A Special Report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of ≥ 0.005 microcuries of removable contamination.

1751 351

3/4.7.12 SETTLEMENT OF CLASS 1 STRUCTURES

LIMITING CONDITION FOR OPERATION

3.7.12.1 The total settlement of each Class 1 structure or the differential settlement between Class 1 structures shall not exceed the allowable values of Table 3.7-5.

APPLICABILITY: All MODES

ACTION:

a. With either the total settlement of any structure or the differential settlement of any structures exceeding 75 percent of the allowable settlement, conduct an engineering review of field conditions and evaluate the consequences of additional settlement. Submit a special report to the Commission pursuant to Specification 6.9.2 within 60 days, containing the results of the investigation, the evaluation of existing and possible continued settlement and the remedial action to be taken if any, including the date of the next survey.

b. With the total settlement of any structure or the differential settlement of any two structures exceeding the allowable settlement value of Table 3.7-5, be in at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.12.1 The total settlement of each Class 1 structure or the differential settlement between Class 1 structures listed in Table 3.7-5 shall be determined to the nearest 0.01 foot by measurement and calculation at least once per 6 months. Measurements on settlement points SM-7, 8, 9, 10, 15, 16, 17, 18, H-569, and H-584 shall be made at least once per 31 days for the time period following 5 years from the date of issuance of the Operating License.

LOW PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.2 The following low pressure CO_2 systems shall be OPERABLE with a minimum of 3.5 tons in the storage tank at a minimum pressure of 275 psig.

a. Cable tunnels and vaults
b. Charcoal filters
c. Emergency diesel generator rooms

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.7.14.2. Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying CO_2 storage tank level and pressure, and

b. At least once per 18 months by verifying:

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

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

1751 353

PLANT SYSTEMS

HIGH PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.3 The following high pressure CO_2 systems shall be OPERABLE with the storage tanks having at least 90% of full charge weight.

a. Fuel oil pump rooms

APPLICABILITY: Whenever equipment in the high pressure CO_2 protected areas is required to be OPERABLE.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.7.14.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying CO_2 storage tank weight.

b. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

1751 354

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provision of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from the containment gaseous and particulate radiation monitoring instrumentation channels.

1751 355

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During CORE ALTERATIONS while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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 measurement of control rod worth and SHUTDOWN MARGIN provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, initiate and continue boration at $\geq$ 10 gpm of at least 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 10 gpm of at least 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod that is not fully inserted shall be demonstrated capable of full insertion when tripped from at least 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

1751 357

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

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

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

1751 358

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

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

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

APPLICABILITY: MODE 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

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

1751 359

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set ≤ 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

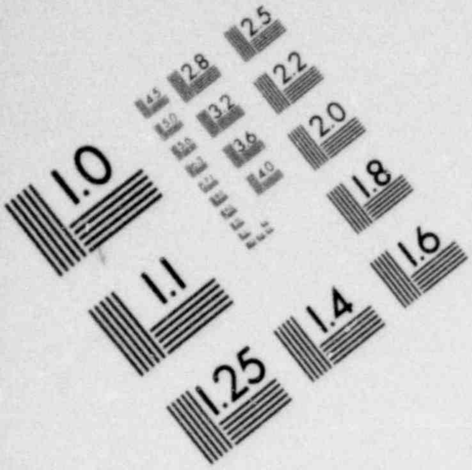
4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

1751 360

IMAGE EVALUATION
TEST TARGET (MT-3)

6"



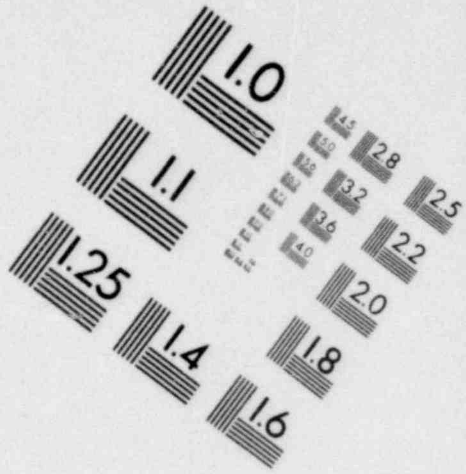
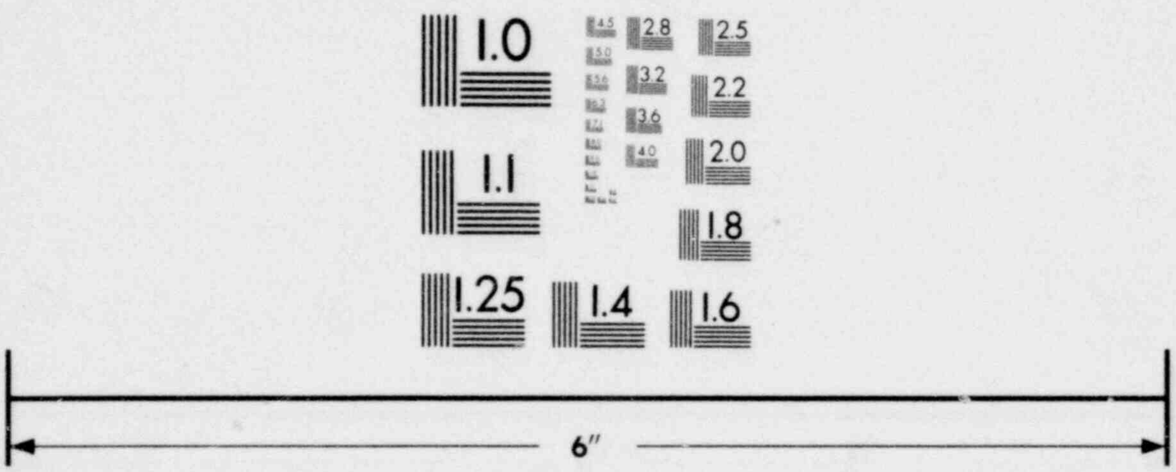
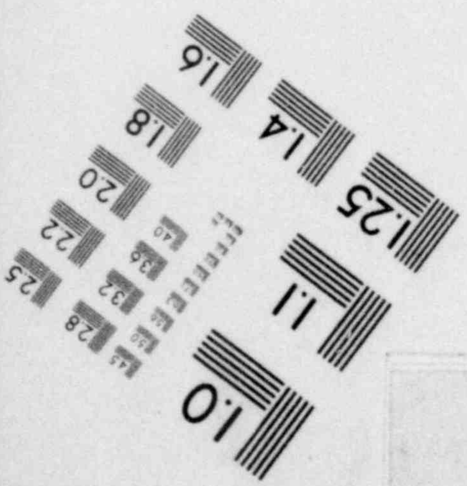
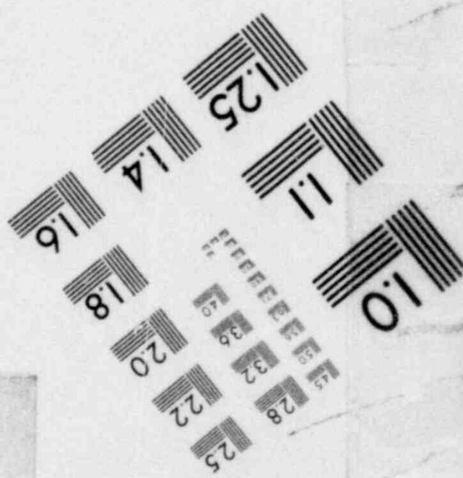
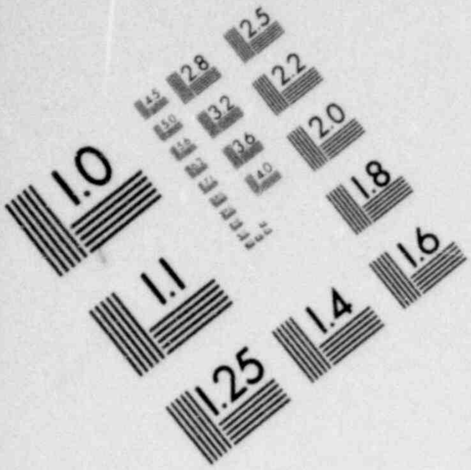


IMAGE EVALUATION TEST TARGET (MT-3)









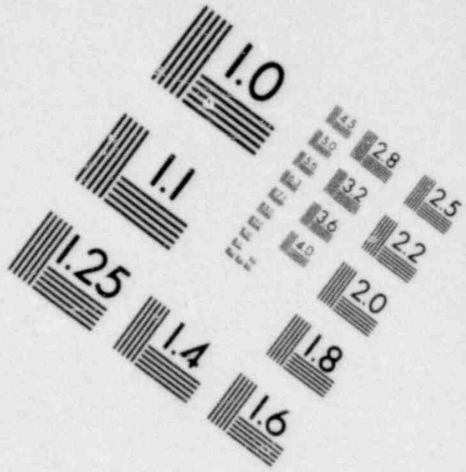
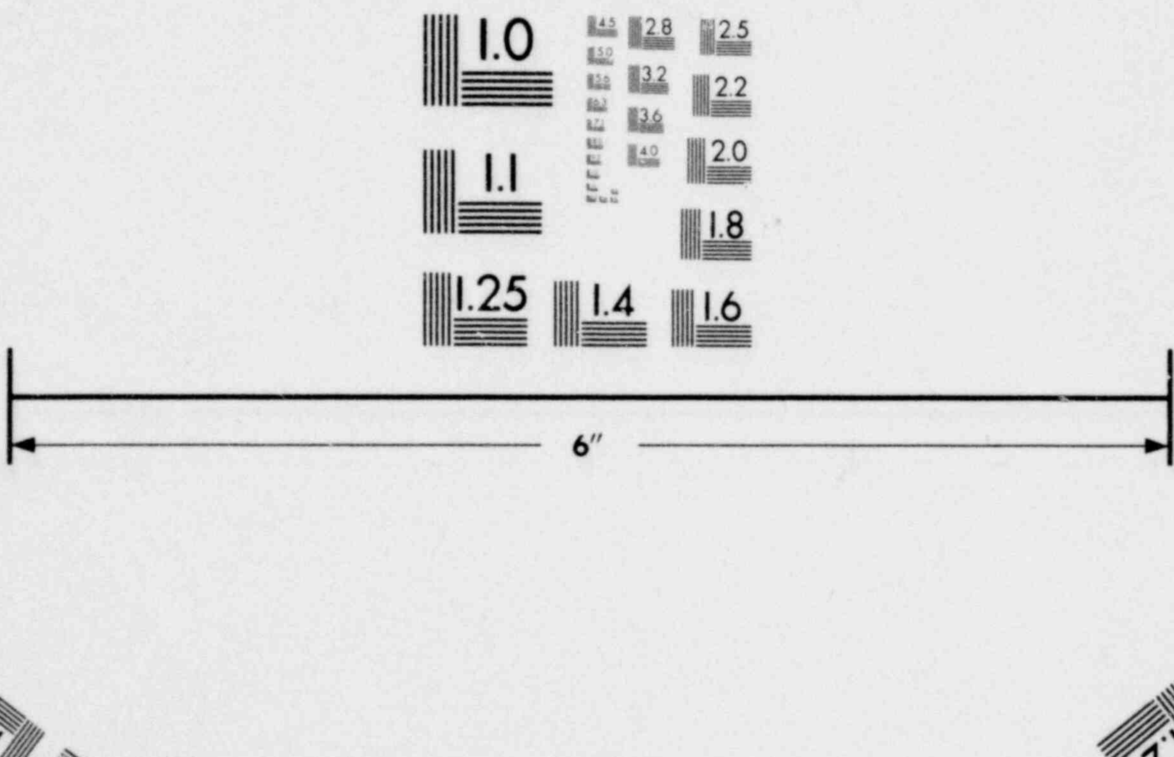
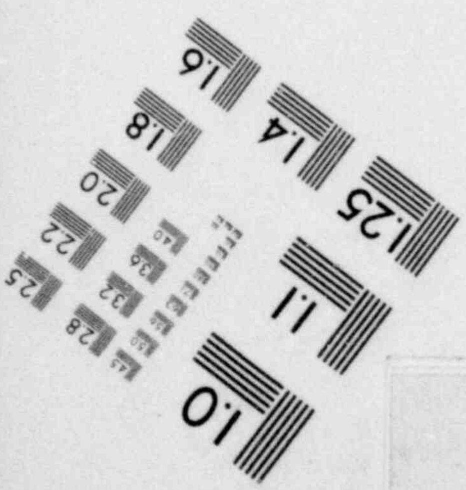


IMAGE EVALUATION TEST TARGET (MT-3)





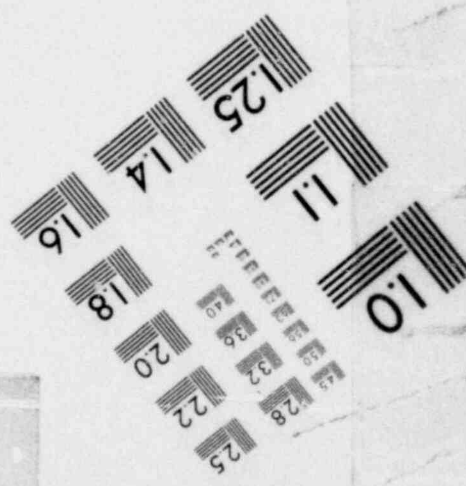


IMAGE EVALUATION
TEST TARGET (MT-3)

6″

SPECIAL TEST EXCEPTIONS

POSITION INDICATOR CHANNELS-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indicator channels inoperable, or more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indicator channels shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication channels agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one control rod bank is withdrawn from the fully inserted position at one time.

1752 001

3/4.1.2 BORATION SYSTEMS (Continued)

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.77% $\Delta k/k$ after xenon decay and cooldown to 200°F. This expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 4450 gallons of 20,000 ppm borated water from the boric acid storage tanks or 70,000 gallons of 2000 ppm borated water from the refueling water storage tank.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 320°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING insures that this system is available for reactivity control while in MODE 6.

1752 002

3/4.1.2 BORATION SYSTEMS (Continued)

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

1752 003

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. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.10 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

1752 004

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ± 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 85% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 85% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 85% and 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

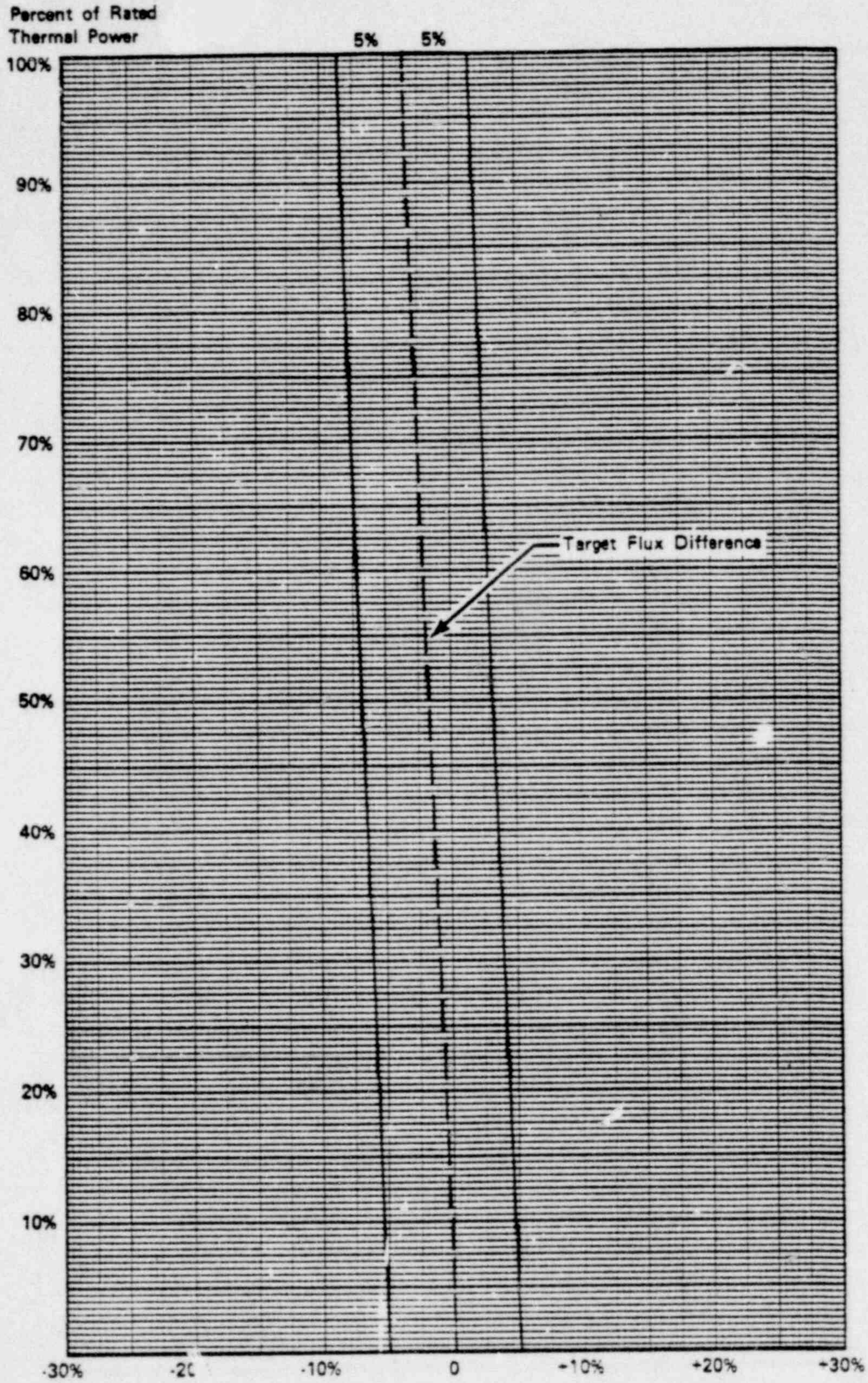


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

1752 006

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS-

$F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

1752 007

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to below the P-8 setpoint, 31 percent of RATED THERMAL POWER, until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.30. A loss of flow in two loops will cause a reactor trip if operating above P-7 while a loss of flow in one loop will cause a reactor trip if operating above P-8.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 320°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by 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 RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve

BASES

is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam buble is formed and thus the RCS is not a hydraulically solid system.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83,

1752 009

POOR ORIGINAL

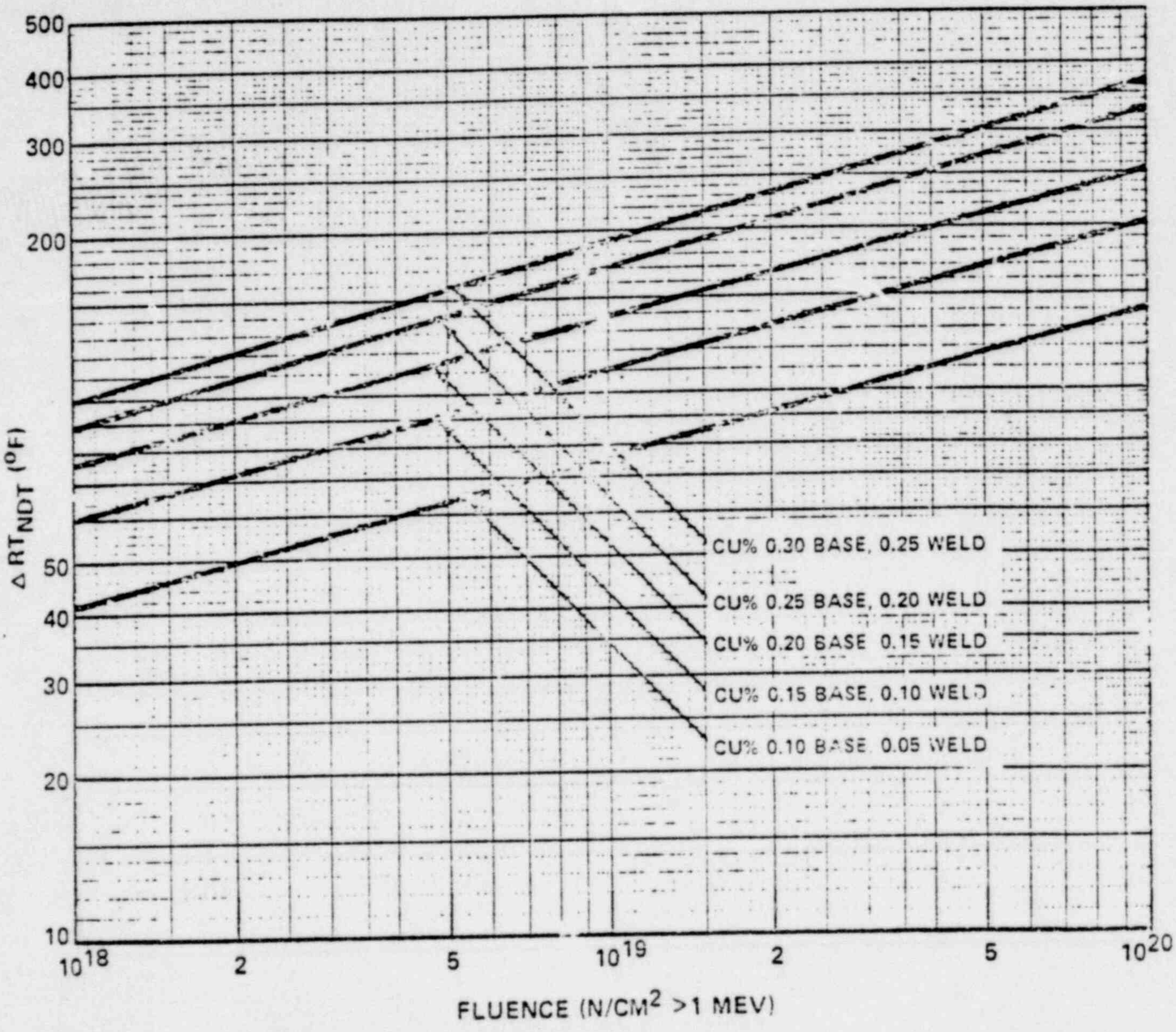


Figure B 3/4.4-2

Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessels Exposed to 550° F Temperature

TABLE B 3/4 4-1

REACTOR VESSEL TOUGHNESS TABLE (UNIT 1)

Component	Heat No.	Material Type	(%)	Cu (%)	P (°F)	NDTT	Min.mum 50 ft-lb/ 35 mil Temp. (°F) Parallel to Major Working Direction	Min.mum 50 ft-lb/ 35 mil Temp. (°F) Normal to Major Working Direction	RT_{NDT} (°F)	Average Upper Shelf (ft-lb) Parallel to Major Working Direction	Average Upper Shelf (ft-lb) Normal to Major Working Direction
Cl. Hd. Dome	53565-1	A533,B,Cl.1				-31	14	34*	-26	87[a]	
Cl. Hd. Flange	4984	A508,Cl.2				-40	<-76	-56*	-40	141[a]	
Ves. Flange	4982	A508,Cl.2				-22	<-70	-50*	-22	161[a]	
Inlet Nozzle	4964	A508,Cl.2				-31	14	34*	-26	108[a]	
Inlet Nozzle	4966	A508,Cl.2				-22	10	30*	-22	88[a]	
Inlet Nozzle	4968	A508,Cl.2				-22	43	63*	3	80[a]	
Outlet Nozzle	4961	A508,Cl.2				-13	43	63*	3	100[a]	
Outlet Nozzle	4965	A508,Cl.2				-22	14	34*	-22	90[a]	
Outlet Nozzle	4967	A508,Cl.2				-4	34	54*	-4	90[a]	
Upper Shell	4952	A508,Cl.2				2	46	66*	6	60[a]	
Inter. Shell	4958	A508,Cl.2	0.12		0.009	-31	20	77[s]	17[s]	94[a]	92[s]
Lower Shell	4979	A508,Cl.2	0.16		0.019	-13	40	98[s]	38[s]	74[a]	85[s]
Bot. Hd. Seg.	53647-3	A533,B,Cl.1				-31	27	47*	-13	66[a]	
Bot. Hd. Seg.	53648-4	A533,B,Cl.1				-13	27	47*	-13	72[a]	
Bot. Hd. Dome	53774	A533,B,Cl.1				-22	32	52*	-8	67[a]	
Weld		Weld	0.086		0.020	-13		79	19		102
Haz		Haz				-22		39	-21		142

*Estimated temperature per NRC Standard Review Plan 5.3.2.

(a) Minimum energy at highest test temperature (≤ 68°F) - % shear not reported.

(s) Average tranverse data obtained from surveillance program.

POOR ORIGINAL

1752 011

vess... inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 320°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water solid RCS.

1752 012

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

The inspection programs for ASME Code Class 1, 2 and 3 the Reactor 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 components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.10.2 STEAM GENERATOR SUPPORTS

For the A572 material, operation above 225° provides a conservative temperature limit and thus toughness level in the steel. This assures the safety of the A572 material even under the worst postulated accident conditions. The points to be monitored were determined during hot functional testing, which indicated the top level corner lags the middle level corner during heatup; however, once the material achieved 225°F the top level corner exceeded the temperature of the middle level corner. The latter thus becomes the controlling zone during operation.

For the monitored top level corner of the steam generator supports, operation below 355°F provides assurance that no other region of the supports will exceed this temperature. The monitored top level corner is the highest temperature region in the supports. With the temperature of the supports less than 355°F all materials will be within allowable stress limits even, under the worst postulated accident conditions.

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

1752 014

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one low head safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and low head safety injection pumps except the required OPERABLE pump to be inoperable below 320°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam li e rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 137°F at 22,500 ppm boron.

1752 015

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

1752 016

available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 EMERGENCY CONDENSATE STORAGE TANK

The OPERABILITY of the emergency condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usuable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM TRIP VALVES

The OPERABILITY of the main steam trip valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam trip valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

1752 017

3/4.7.1.6 and 3/4.7.1.7 STEAM TURBINE and OVERSPEED PROTECTION

The turbine generator at the North Anna facility is arranged in a nonpeninsular orientation. Analysis has shown that this arrangement is such that if a turbine failure occurs as a result of destructive overspeed, potentially damaging missles could impact the auxiliary building, containment, control room and other structures housing safety related equipment. The requirements of these two specifications provide additional assurance that the facility will not be operated with degraded valve performance and/or flawed turbine material which are the major contributors to turbine failures.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SUBSYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

POOR ORIGINAL

3/4.7.11 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. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

1752 019

3/4.7.12 SETTLEMENT OF CLASS 1 STRUCTURES

In order to assure that settlement does not exceed predicted and allowable settlement values, a program has been established to conduct a survey of a specified number of points at the site on a semi-annual basis. The first survey was conducted in May 1976 to establish base-line elevations for most of the points. Where applicable, the base-line elevations of points established subsequent to the May 1976 survey have been adjusted to the May 1976 survey by an evaluation of the settlement records of settlement points on the same structure or on nearby structures. Baseline elevations for some points were established on dates other than May 1976 as indicated in Table 3.7-5. Additional surveys will be performed semiannually. The determination of the elevation of the points shall be by precise leveling with second order Class II accuracy as defined by the U. S. Department of Commerce, National Oceanic and Atmospheric Administration, National Ocean Survey, 1974. When any settlement point listed in Table 3.7-5 is inaccessible during a survey, comparison to allowable settlement shall be based on settlement of other points on the same structure or on nearby structures having similar foundation conditions. When any settlement point listed in Table 3.7-5 is dislocated or replaced, a documented review of the settlement records of points on the same structure and additionally points on nearby structures having similar foundation conditions shall provide a new reference elevation for the point that reflects the estimated settlement since the base-line survey. If the total settlement or differential settlement exceeds 75 percent of the allowable value, the frequency of surveillance shall be increased as dictated by the engineering review.

Allowable differential movement is controlled by pipe deflections permitted by fixation points in buildings. The items limiting differential settlement are as follows:

1752 020

Reference	Monitoring Points	Limiting Item
Containment Unit 1	Fuel Building	Fuel Transfer Tube
Containment Unit 1	Auxiliary Building	3"-WGCB-14
Containment Unit 1	Unit 1 Safeguards Area	12"-SI-1
Containment Unit 1	Unit 1 Main Steam Valve House	6"-SI-16
Containment Unit 1	Service Building (E-3)	32"-SHP-2
Safeguards Area Unit 1	Unit 1 Main Steam Valve House	8"-Q5-3
Auxiliary Building	Unit 1 Main Steam Valve House	8"-WS-113
Auxiliary Building	Fuel Building	4"-RP-28
Auxiliary Building	Service Building Tunnel	24"-WS-102
Service Water Pump House	Service Water Piping @ SWPH	Expansion Joint
Service Water Pump House	Pipe Hanger in Reservoir	24" -WS-11-151-Q3
Service Water Pump House	Service Water Pump House	Mat
Service Bldg. (E-5,E-6)	Unit 1 Main Steam Valve House	24"-WS-26-151-Q3
Service Bldg. (E-14)	Unit 2 Main Steam Valve House	24"-WS-426-151-Q3
Auxiliary Feedwater Pump House - Unit 1	Pipe Tunnel	3"-WAPD-9-601-Q3
Decontamination Bldg.	Pipe Tunnel	3"-CC-90-151-Q3
Fuel Building	Waste Gas Decay Tk. Enclosure	4"-GW-30-154-Q3
Safeguards Area Unit 1	Unit 1 Casing Cooling Building	6"-RS-E15-153A-Q3

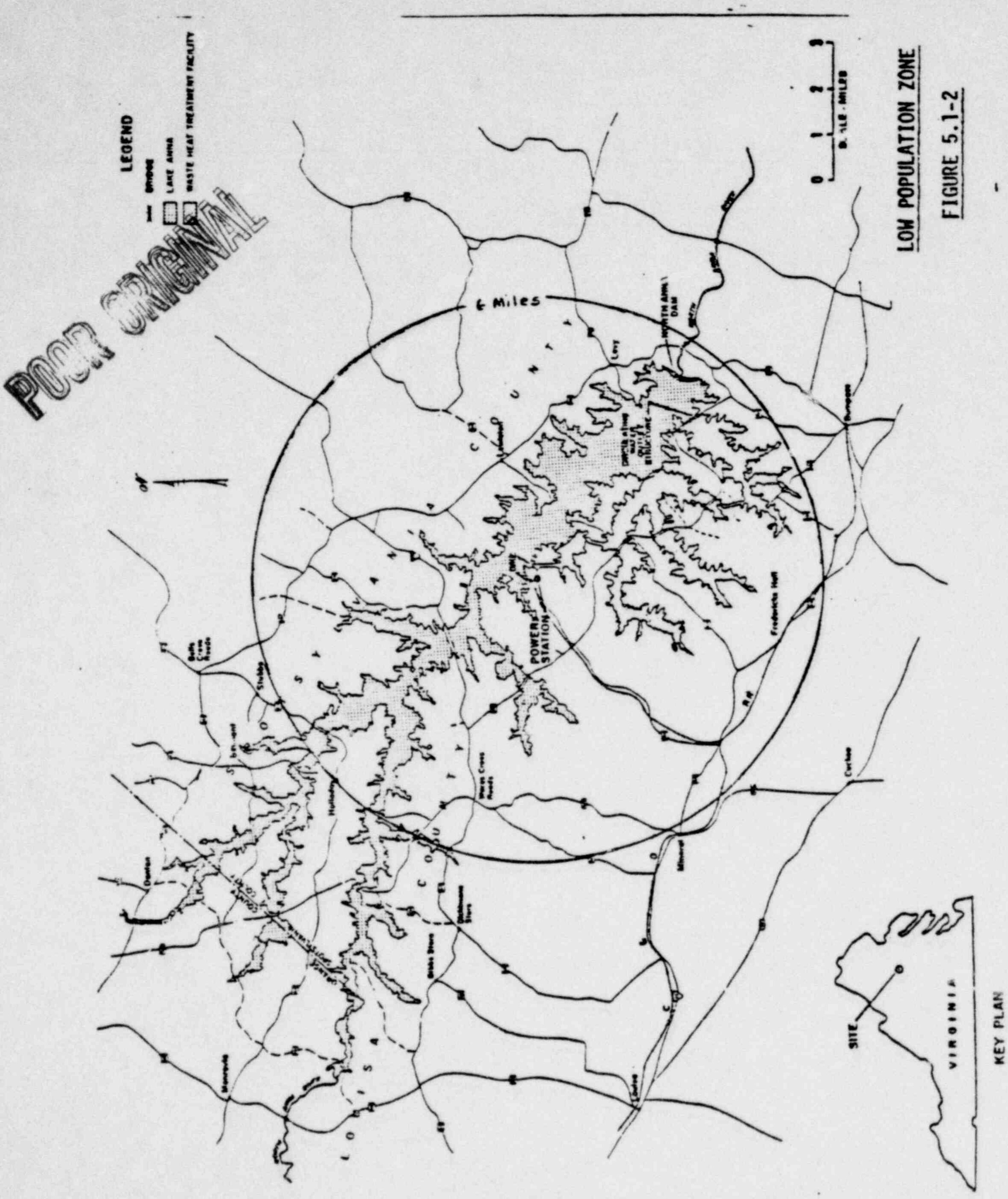
The items limiting total settlement of structures are as follows:

Monitoring Points	Limiting Items
Service Water Piping at SWPH	36"-WS-1-151-Q3
Service Building (E-17)	36"-WS-1-151-Q3
Turbine Building (B-9 1/2)	24"-WS-25-151-Q3
Fuel Oil Pump House	2 1/2"-FOF-151-S
Boron Recovery Tank Dike	NOTE (1)
Circulating Water Intake Structure	Service Water Piping expansion joint

Note (1) No settlement expected; settlement in excess of 0.03 ft. would indicate an abnormality.

(2) Measurements of certain points are required to be performed at least once per 31 days for the first five years of facility operation to provide additional settlement information.

1752 021



LOW POPULATION ZONE

FIGURE 5.1-2

1752 022

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

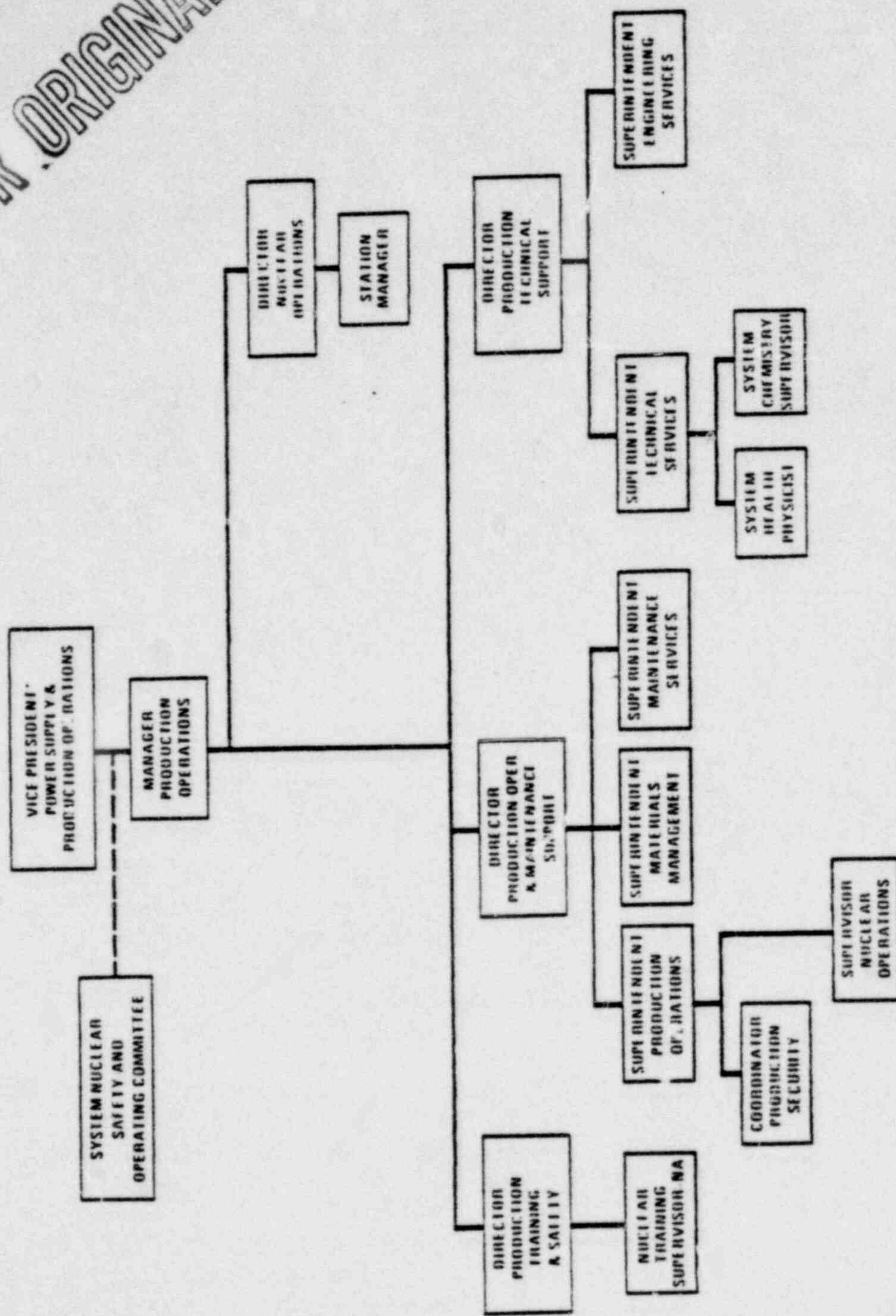
d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include the minimum shift crew shown in Table 6.2-1 or any personnel required for other essential functions during a fire emergency.

---

*Fire Brigade composition may be less than the minimum requirement for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirement.



*Responsible for Corporate Fire Protection Program

Figure 6.2-1 Offsite Organization for Facility Management and Technical Support

MEETING FREQUENCY

6.5.2.5 The SyNSOC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of the SyNSOC shall consist of not less than a majority of the members or duly appointed alternates and shall be subject to the following constraints:

1. The Chairman or Vice Chairman shall be present for all meetings.

2. No more than a minority of the quorum shall have line responsibility for operation of the stations.

3. A motion-carrying vote must consist of no less than three (3) votes.

4. No more than a minority of a quorum may be alternates.

REVIEW

6.5.2.7 The following subjects shall be reviewed by the SyNSOC:

a. Written safety evaluations of changes in the stations as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2) and is accomplished by review of minutes of the Station Nuclear Safety and Operating Committee and the design change program.

b. Proposed changes in procedures, proposed changes in the station, or proposed tests or experiments, any of which may involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2). Matters of this kind shall be referred to the SyNSOC by the Station Nuclear Safety and Operating Committee following its review prior to implementation.

c. Changes in the technical specifications or license amendments relating to nuclear safety prior to implementation except in those cases where the change is identical to a previously reviewed proposed change.

d. Violations and reportable occurrences such as:

1752 026

POOR ORIGINAL

REVIEW (Cont'd)

1. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements or internal procedures or instructions having safety significance;

2. Significant operating abnormalities or deviations from normal or expected performance of station safety-related structures, systems, or components; and

3. Reportable occurrences as defined in the station Technical Specifications.

Review of events covered under this paragraph shall include the results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Any other matter involving safe operation of the nuclear power stations which a duly appointed subcommittee or committee member deems appropriate for consideration, or which is referred to the SyNSOC by the Station Nuclear Safety and Operating Committee.

AUDITS

6.5.2.8 Audits of station activities shall be performed under the cognizance of the SyNSOC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Station Emergency Plan and implementing procedures at least once per 24 months.

f. The Station Security Plan and implementing procedures at least once per 24 months.

1752 027

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program Implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the SNSOC and approved by the Station Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

POOR ORIGINAL

1752 028

STARTUP REPORTS

6.9.1.1 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 a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

POOR ORIGINAL

1752 029

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.9-1.

g. Records of reactor tests and experiments

h. Records of training and qualification for current members of the plant staff.

i. Records of in-service inspections performed pursuant to these Technical Specifications.

j. Records of Quality Assurance activities required by the QA Manual.

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

l. Records of meetings of the SNSOC and the SyNSOC.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

POOR ORIGINAL

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/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 requiring issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist.

*Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.