



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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170  
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated July 8, 1991, as supplemented April 15, 1992, complies 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.

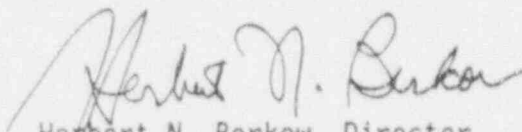
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 170, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Herbert N. Berkow, Director  
Project Directorate II-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 1, 1992



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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169  
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated July 8, 1991, as supplemented April 15, 1992, complies 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.

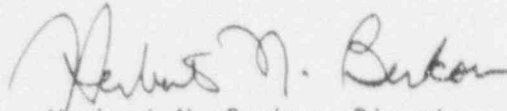
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 169, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Herbert N. Berkow, Director  
Project Directorate II-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 1, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages

TS 2.1-4  
TS 2.1-5  
TS 2.1-6  
TS Figure 2.1-1  
TS 3.12-3  
TS 3.12-11  
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TS 3.12-14  
TS 3.12-15  
TS 3.12-16  
TS 3.12-19  
TS Figure 3.12-8  
TS 4.1-9d

Insert Pages

TS 2.1-4  
TS 2.1-5  
TS 2.1-6  
TS Figure 2.1-1  
TS 3.12-3  
TS 3.12-11  
TS 3.12-11a  
TS 3.12-14  
TS 3.12-15  
TS 3.12-16  
TS 3.12-19  
TS Figure 3.12-8  
TS 4.1-9d

conservative, than the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the calculated DNBR is equal to the design DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the calculated DNBR reaches the design DNBR limit and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figure 2.1-1 is based on an  $F\Delta H(N)$  of 1.62, a 1.55 cosine axial flux shape, and a deterministic DNB analysis procedure including margin to accommodate rod bowing<sup>(1)</sup>. TS Figure 2.1-1 is also bounding for a statistical treatment of key DNBR analysis parameter uncertainties including an enthalpy rise hot channel factor which follows the following functional form:  $F\Delta H(N) = 1.56 [1 + 0.3(1-P)]$  where P is the fraction of rated power. TS Figures 2.1-2 and 2.1-3 are based on an  $F\Delta H(N)$  of 1.55, a deterministic treatment of key DNB analysis parameter uncertainties, and include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies

full; withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figures 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit<sup>(3)</sup> based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also

ensures that at least 99.9% of the core avoids the onset of DNB when the limiting rod is at the DNER limit.

The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

#### References

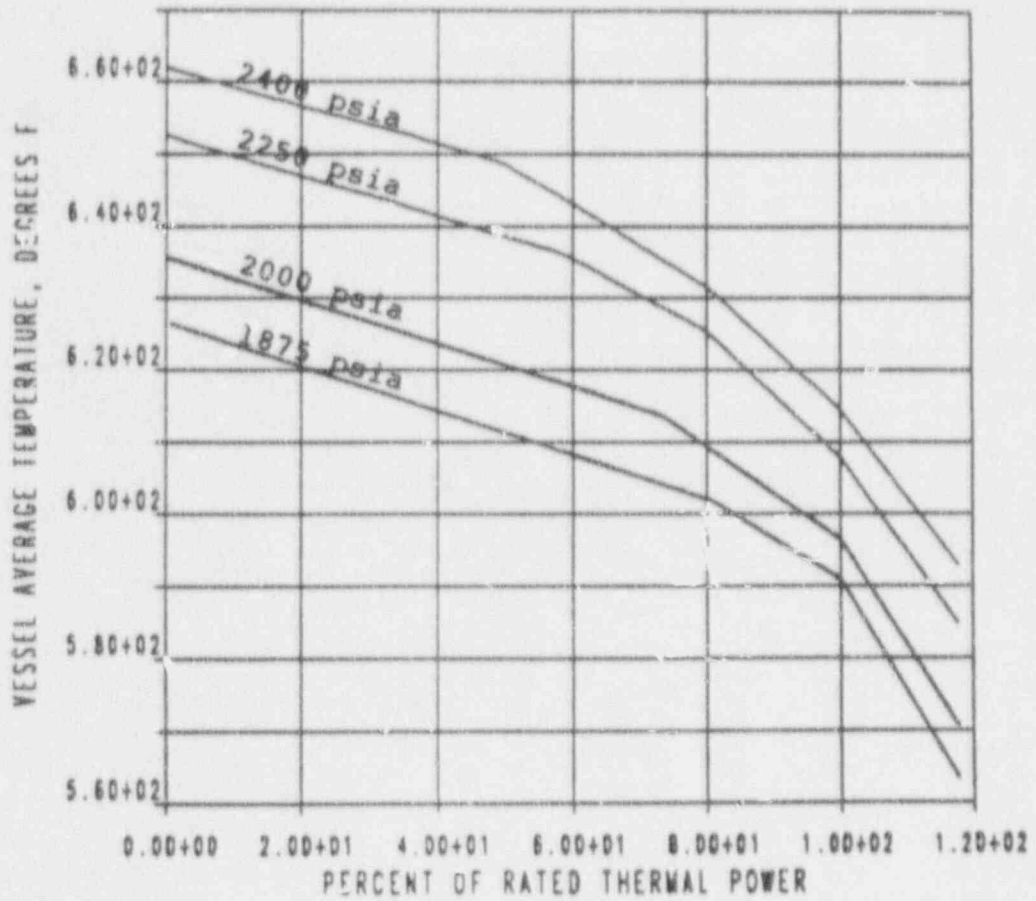
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- 1) FSAR Section 3.4
- 2) FSAR Section 3.3
- 3) FSAR Section 14.2



TS FIGURE 2.1-1

REACTOR CORE THERMAL AND  
HYDRAULIC SAFETY LIMITS -  
THREE LOOP OPERATION, 100% FLOW



B. Power Distribution Limits

1. At all times except during low power physics tests, the hot channel factors defined in the basis meet the following limits:

$$F_Q(Z) \leq 2.32/P \times K(Z) \text{ for } P > 0.5$$

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

$$F_{\Delta H}^N \leq 1.56 [1 + 0.3 (1-P)] \text{ for three loop operation}$$

$$\leq 1.55 [1 + 0.2 (1-P)] \text{ for two loop operation}$$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, and Z is the core height location of  $F_Q$ .

2. Prior to exceeding 75% power following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:
  - a. The measurement of total peaking factor  $F_Q^{\text{Meas}}$  shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor  $F_{\Delta H}$  shall be compared directly to the limit specified in Specification 3.12.B.1. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits of  $F_Q(Z) \leq 2.32/P \times K(Z)$  and  $F_{\Delta H}^N \leq 1.56$  within 24 hours, the Overpower  $\Delta T$  and Overtemperature  $\Delta T$  trip setpoints shall be similarly reduced.

- c. In hot, intermediate and cold shutdown conditions, the step demand counters shall be operable and capable of determining the group demand positions to within  $\pm 2$  steps. The rod position indicators shall be available to verify rod movement upon demand.
2. If a rod position indicator channel is out of service, then:
    - a. For operation above 50% of rated power, the position of the RCC shall be checked indirectly using the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod exceeding 24 steps, or
    - b. Reduce power to less than 50% of rated power within 8 hours. During operations below 50% of rated power, no special monitoring is required.
  3. If more than one rod position (RPI) indicator channel per group or two RPI channels per bank are inoperable during control bank motion to achieve criticality or power operations, then the requirements of Specification 3.0.1 will be followed.

F. DNB PARAMETERS

1. The following DNB related parameters shall be maintained within their limits during power operation:

Reactor Coolant System  $T_{avg} \leq 578.4^{\circ}\text{F}$

Pressurizer Pressure  $\geq 2205$  psig

Reactor Coolant System Total Flow Rate  $\geq 273,000$  gpm

- a. The Reactor Coolant System  $T_{avg}$  and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

- b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.
2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce reactor power to less than 5% of rated thermal power within the next 4 hours.
3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a thermal power ramp increase in excess of 5% of rated thermal power per minute or a thermal power step increase in excess of 10% of rated thermal power.

#### Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition. The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit

In addition to the above, the peak linear power density and the nuclear enthalpy rise hot channel factor must not exceed their limiting values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits of power distribution, the following hot channel factors are defined:

$F_Q(Z)$ , Height Dependent 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 tolerance on fuel pellets and rods.

$F_Q^E$ , Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux for non-statistical applications.

$F_{\Delta H}^N$ , 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 for both LOCA and non-LOCA considerations.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using the upper bound  $F_Q(Z)$  times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

When an  $F_Q$  measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowance for measurement error for a full core map (greater than or equal to 38 thimbles, including a minimum of 2 thimbles per core quadrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of  $F_Q$ .

In the specified limit of  $F_{\Delta H}^N$ , there is a four percent error allowance, which means that normal operation of the core is expected to result in  $F_{\Delta H}^N \leq 1.56 [1 + 0.3 (1-P)]/1.04$ . The 4% allowance is based on the considerations that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect  $F_{\Delta H}^N$ , in most cases without necessarily affecting  $F_Q$ , (b) the operator has a direct influence on  $F_Q$  through movement of rods and can limit it to the desired value; he has no direct control over  $F_{\Delta H}^N$ , and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence  $F_Q$ , can be compensated for by tighter axial control. An appropriate allowance for the measurement uncertainty

for  $F_{\Delta H}^N$  obtained from a full core map ( $\geq 38$  thimbles, including a minimum of 2 detectors per core quadrant, monitored) taken with the movable incore detector flux mapping system has been incorporated in the statistical DNBR limit. Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor  $F_{\Delta H}^N$  limit will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.
2. Control rod banks are sequenced with overlapping banks as shown in TS Figures 3.12-1A, 3.12-1B.
3. The full length control bank insertion limits are not violated.
4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux differences refers to the difference

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range  $\pm 13.8$  percent ( $\pm 10.8$  percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod and an error allowance. No increase in  $F_Q$  occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum  $F_Q$  occurs.

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The measurement of the RCS total flow rate once per refueling cycle is adequate to detect flow degradation.



TS FIGURE 3.12-8

HOT CHANNEL FACTOR NORMALIZED  
OPERATING ENVELOPE

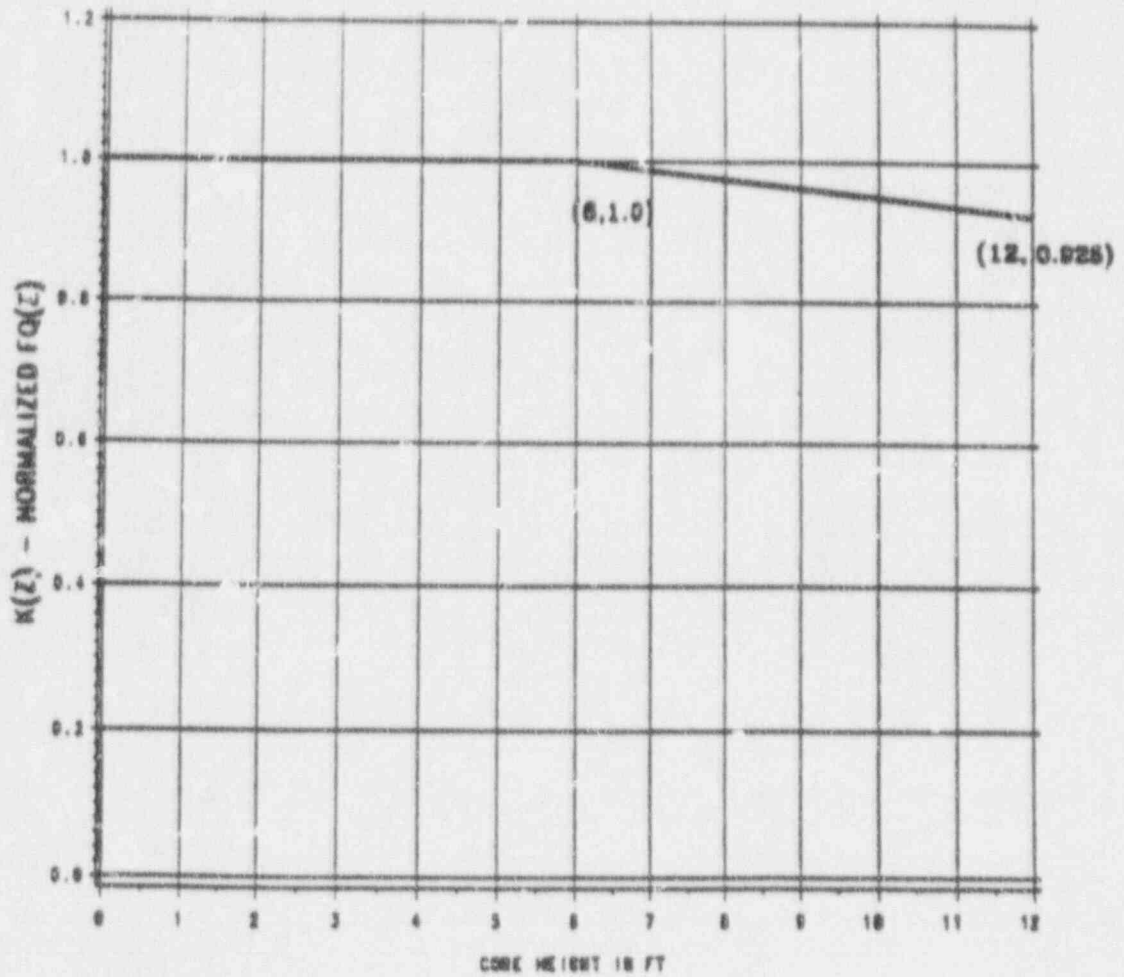


TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
18. Primary Coolant System	Functional	1. Periodic leakage testing (a) on each valve listed in Specification 3.1.C.7a shall be accomplished prior to entering power operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
19. Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval (c)	
20. Containment Hydrogen Analyzers	a. Channel Check b. Channel Functional Test c. Channel Calibration using sample gas containing: 1. One volume percent ( $\pm 0.25\%$ ) hydrogen, balance nitrogen; 2. Four volume percent ( $\pm 0.25\%$ ) hydrogen, balance nitrogen; 3. Channel calibration test will include startup and operation of the Heat Tracing System	Once per 12 hours Once per 31 days Once per 92 days on staggered basis	
21. RCS Flow	Flow $\geq 273,000$ gpm	Once per refueling cycle	14

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

\*See Specification 4.1.D.