ATTACHMENT TO LICENSE AMENDMENT NO. 61

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages contain the captioned amendment number and marginal lines which indicate the area of changes.

REMOVE	INSERT
page 1-1	page 1-1
page 2.1-2	page 2.1-2
page 2.1-4	page 2.1-4
Figure 2.1-1	Figure 2.1-1
pages 2.3-2 through 2.3-3	pages 2.3-2 through 2.3-3
page 3.1-4b	page 3.1-4b
pages 3.1-17 through 3.1-19	pages 3.1-17 through 3.1-19
page 3.8-3 through 3.8-4	page 3.8-3 through 3.8-4
pages 3.10-3 through 3.10-4	pages 3.10-3 through 3.10-4
Figures 3.10-2, 3.10-3	Figures 3.10-2, 3.10-3

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

1.1 Thermal Power

The rate that the thermal energy generated by the fuel is accumulated by the coolant as it passes through the reactor vessel.

1.2 Reactor Operating Modes

Mode	Reactivity $\Delta k/k\%$	Coolant Temperature (°F)
Refueling	≤ -5	$T_{avg} \leq 140$
Cold Shutdown	≤ -1	$T_{avg} \leq 200$
Hot Shutdown	≤ -1	$T_{avg} \geq 540$
Operating	≥ 0	$T_{avg} \sim 580$

1.3 Refueling

Any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted.

1.4 Operable

Capable of performing all intended functions in the intended manner.

boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and/or WRB-1 DNB correlation. These DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. A minimum value of the DNB ratio, MDNBR, is specified so that during steady state operation, normal operational transients and anticipated transients, there is a 95% probability at a 95% confidence level that DNB will not occur.[1] The curves of Figure 2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which this minimum DNB value is satisfied. The area of safe operation is below these lines.

Since it is possible to have somewhat greater enthalpy rise hot channel factors at part power than at full power due to the deeper control bank insertion which is permitted at part power, a conservative allowance has been made in obtaining the curves in Figure 2.1-1 for an increase in $F^N_{\Delta H}$ with decreasing power levels. Rod withdrawal block and load runback occurs before reactor trip set points are reached.

The Reactor Control and Protective 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 there being less than a 95% probability at a 95% confidence level that DNB would not occur.[3]

(1) FSAR, Section 3.2.2

(2) FSAR, Section 3.2.1

(3) FSAR, Section 14.1.1

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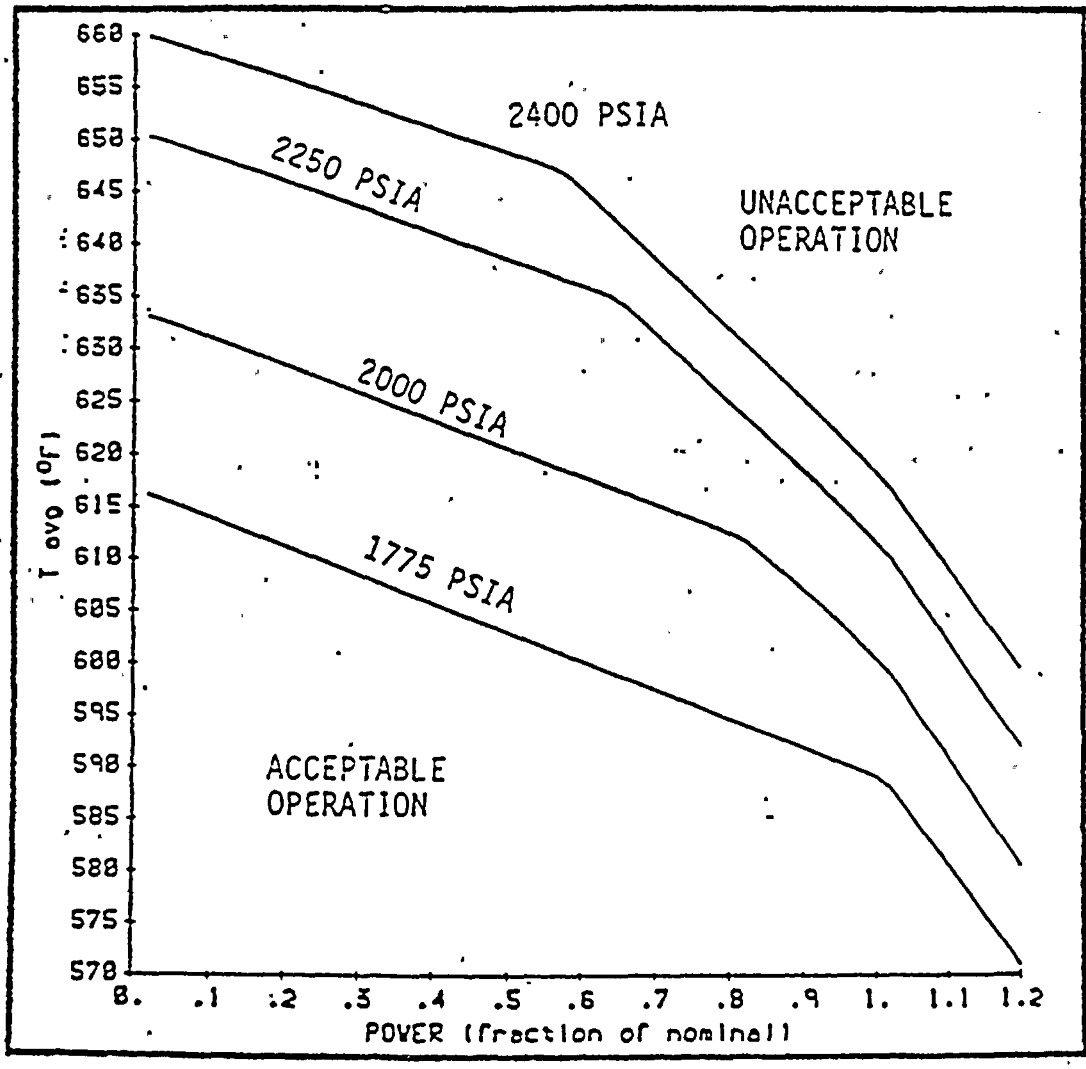
FIGURE 2.1-1

CORE DNB SAFETY LIMITS

2 LOOP OPERATION



d. Overtemperature ΔT

$$\leq \Delta T_o \left[K_1 + K_2(P-P^1) - K_3(T-T^1) \left(\frac{1 + \tau 1S}{1 + \tau 2S}\right)\right] - f(\Delta I)$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T^1 = 573.5°F

P = pressurizer pressure, psig

P^1 = 2235 psig

K_1 = 1.20

K_2 = .000900

K_3 = .0209

$\tau 1$ = 25 sec

$\tau 2$ = 5 sec

and f (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 where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power such that:

(i) for $q_t - q_b$ less than +21 percent, f (ΔI) = 0

(ii) for each percent that the magnitude of $q_t - q_b$ is more positve than +21 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 1.6 percent of rated power.

e. Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5(T-T^1) - K_6 \frac{\tau 3\ ST}{\tau 3S + 1}\right] - f(\Delta I)$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T^1 = indicated T avg at nominal conditions at rated power, °F

K_4 = 1.077

K_5 = 0.0 for $T<T^1$

= 0.0011 for $T \geq T^1$

K_6 = 0.0262 for increasing T

= 0.0 for decreasing T

$\tau 3$ = 10 sec

$f(\Delta I)$ = as defined in 2.3.1.2.d.

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3.1.1.5 Pressurizer

Whenever the reactor is at hot shutdown or critical the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hrs. or have the RHR system in operation within an additional 6 hrs.

Bases

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal

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3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at a temperature below 500°F, and if the moderate temperature coefficient is more positive than

a. 5 pcm/°F (below 70 percent of rated thermal power)

b. 0 pcm/°F (at or above 70 percent of rated thermal power)

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit line shown on Figure 3.1-1 of these specifications.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

Basis

Previous safety analyses have assumed that for Design Basis Events (DBE) initiated from the hot zero power or higher power condition, the moderator temperature coefficient (MTC) was either zero or negative.(1)(2) Beginning in Cycle 14, the safety analyses have assumed that a maximum MTC of +5 pcm/°F can exist up to 70% power. Analyses have shown that the design criteria can be satisfied for the DBE's with this assumption.(3) At greater than 70% power the MTC must be zero or negative.

The limitations on MTC are waived for low power physics tests to permit measurement of the MTC and other physics design parameters of interest. During these tests special operating precautions will be taken.

The requirement that the reactor is not to be made critical above and to the left of the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of an increase in moderator temperature or a decrease of coolant pressure.

<u>Reference</u>

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-8

(3) Safety Evaluation for R. E. Ginna Transition to 14 x 14 Optimized Fuel Assemblies, Westinghouse Electric Corporation, November 1983.

to public health and safety.[1] Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin as indicated will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 230,000 gallons of borated water. The boron concentration of this water at 2000 ppm boron is sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold condition with all rods inserted (best estimate of 10% subcritical), and will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Periodic checks of refueling water boron concentration insure the proper shutdown margin. Communication requirements allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to insure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time. In addition interlocks on the auxiliary building crane will prevent the trolley from being moved over storage racks containing spent fuel.

The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.

References:

(1) FSAR - Section 9.5.2

(2) Reload Transition Safety Report, Cycle 14

(3) FSAR - Section 9.3.1

average power tilt ratio shall be determined once a day by at least one of the following means:

a. Movable detectors

b. Core-exit thermocouples

3.10.2.2 Power distribution limits are expressed as hot channel factors. At all times, except during low power physics tests the hot channel factors must meet the following limits:

$F_Q(Z) = (2.32/P)*K(Z)$ for $P \geq .5$

$F_Q(Z) = 4.64*K(Z)$ for $P \leq .5$

$F^N_{\Delta H} = 1.66\ [1 + .3(1-P)]$ for $0 \leq P \leq 1.00$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given by Figure 3.10-3, and Z is the height in the core. The measured F_Q^N shall be increased by three percent to yield F_Q. If the measured F_Q or $F\Delta_H^N$ exceeds the limiting value, with due allowance for measurement error, the maximum allowable reactor power level and the Nuclear Overpower Trip set point shall be reduced on percent for each percent which $F\Delta_H^N$ or F_Q exceeds the limiting value, whichever is more restrictive. If the hot channel factors cannot be reduced below the limiting values within one day, the Overpower ΔT trip setpoint and the Overtemperature ΔT trip setpoint shall be similarly reduced.

3.10.2.3 Except for physics tests, if the quadrant to average power tilt ratio, exceeds 1.02 but is less than 1.12, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors, and apply Specification 3.10.2.2, or

c. Limit power to 75% of rated power.

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

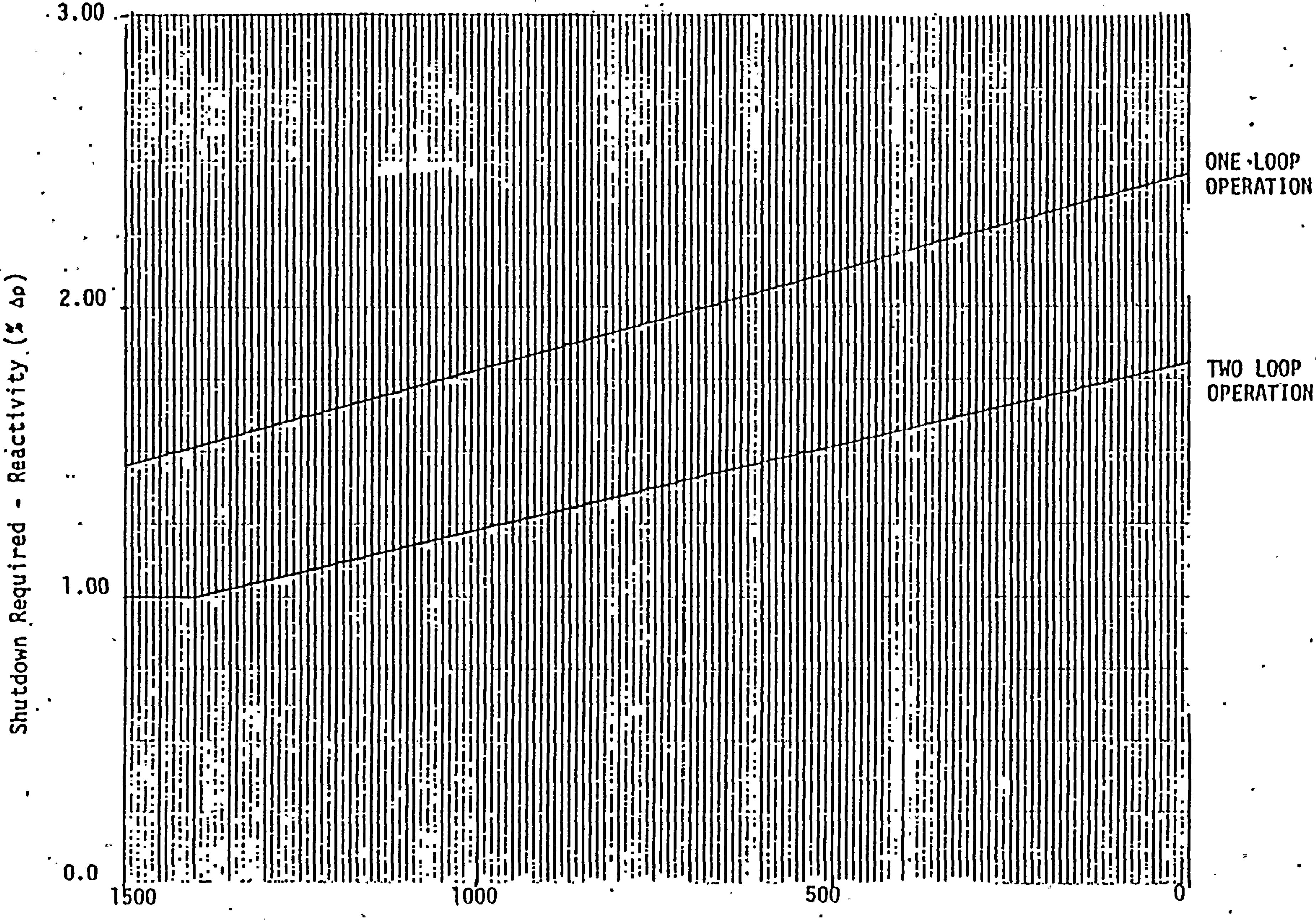
3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip "set point is reduced by 50%".

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by linear interpolation using the most recent measured value and the predicted value at the end of the cycle life.

3.10.2.8 Except during physics tests, control rod exercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within ±5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

3.10.2.9 Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.



COOLANT BORON CONCENTRATION (PPM)

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REQUIRED SHUTDOWN MARGIN

FIGURE 3.10-2

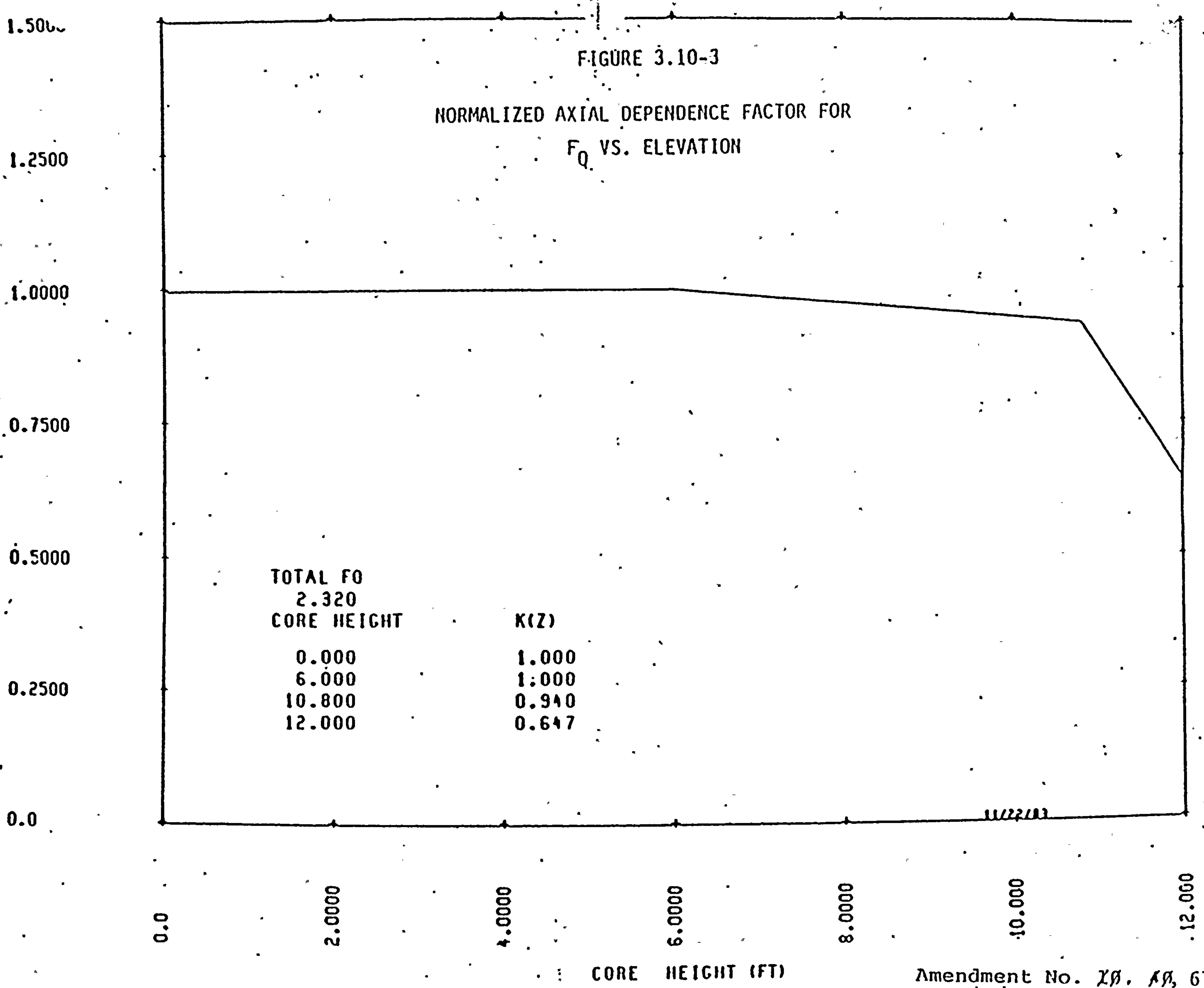

FIGURE 3.10-3

NORMALIZED AXIAL DEPENDENCE FACTOR FOR

F_Q VS. ELEVATION

TOTAL FQ
2.320

CORE HEIGHT	K(Z)
0.000	1.000
6.000	1.000
10.800	0.940
12.000	0.647

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