PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
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Technical Specification 6.9.1.9

LR-N14-0203

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-001

Salem Nuclear Generating Station
Renewed Facility Operating License No. DPR-70
NRC Docket No. 50-272

Subject: Salem Unit 1 Core Operating Limits Report – Cycle 24

In accordance with section 6.9.1.9 of the Salem Unit 1 Technical Specifications, PSEG Nuclear LLC submits the Core Operating Limits Report (COLR) for Salem Unit 1, Cycle 24 in Attachment 1 to this letter.

There are no commitments contained in this letter. Should you have any questions regarding this submittal, please contact David Lafleur at (856) 339-1754.

Sincerely,

John F. Perry
Site Vice President - Salem

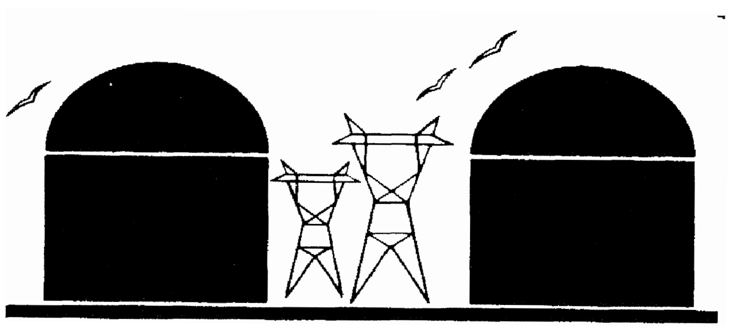
Attachments (1)

cc: Mr. D. Lew, Administrator, Region 1, NRC
Ms. C. Sanders, Licensing Project Manager - Salem, NRC
Mr. P. Finney, USNRC Senior Resident Inspector, Salem
Mr. A. Ziedonis, USNRC Resident Inspector, Salem
Mr. P. Mulligan, Manager IV, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. Cachaza, Salem Commitment Tracking Coordinator
Mr. D. Lafleur, Senior Regulatory Compliance Engineer

Attachment 1

Core Operating Limits Report for Salem Unit 1, Cycle 24

Core Operating Limits Report for Salem Unit 1, Cycle 24



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1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Salem Unit 1 Cycle 24 has been prepared in accordance with the requirements of Technical Specification 6.9.1.9. The Technical Specifications affected by this report are listed below along with the NRC-approved methodologies used to develop and/or determine COLR parameters identified in Technical Specifications.

TS Section	Technical Specifications	COLR Parameter	COLR Section	NRC Approved Methodology (Section 3.0 Number)
3.1.1.4	Moderator Temperature Coefficient	MTC	2.1	3.1, 3.6
3.1.3.5	Control Rod Insertion Limits	Control Rod Insertion Limits	2.2	3.1, 3.6
3.2.1	Axial Flux Difference	AFD	2.3	3.1, 3.2, 3.6
3.2.2	Heat Flux Hot Channel Factor - $F_Q(Z)$	$F_Q(Z)$	2.4	3.1, 3.3, 3.4, 3.5, 3.6, 3.7
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$	$F^N_{\Delta H}$	2.5	3.1, 3.5, 3.6, 3.7
3.9.1	Boron Concentration	Boron Concentration	2.6	3.1, 3.6

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.9 and in Section 3.0 of this report.

2.1 Moderator Temperature Coefficient (Specification 3.1.1.4)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less positive than or equal to 0 $\Delta k/k/°F$.

The EOL/ARO/RTP-MTC shall be less negative than or equal to -4.4×10^{-4} $\Delta k/k/°F$.

2.1.2 The MTC Surveillance limit is:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to -3.7×10^{-4} $\Delta k/k/°F$.

where: BOL stands for Beginning of Cycle Life

ARO stands for All Rods Out

HZP stands for Hot Zero THERMAL POWER

EOL stands for End of Cycle Life

RTP stands for RATED THERMAL POWER

2.2 Control Rod Insertion Limits (Specification 3.1.3.5)

2.2.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.3 Axial Flux Difference (Specification 3.2.1)

[Constant Axial Offset Control (CAOC) Methodology]

2.3.1 The Axial Flux Difference (AFD) target band shall be (+6%, -9%).

2.3.2 The AFD Acceptable Operation Limits are provided in Figure 2.

2.4 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3.2.2)

[F_{xy} Methodology]

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z) \text{ for } P \leq 0.5$$

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

2.4.1 $F_Q^{RTP} = 2.40$

2.4.2 K(Z) is provided in Figure 3.

2.4.3 $F_{xy}^{L} = F_{xy}^{RTP}[1.0 + PF_{xy}(1.0 - P)]$

where: from BOL to 12000 MWD/MTU

F_{xy}^{RTP} = *2.00* for unrodded upper core planes 1 through 6
1.91 for unrodded upper core planes 7 through 8
1.75 for unrodded upper core planes 9 through 31
1.83 for unrodded lower core planes 32 through 53
1.95 for unrodded lower core planes 54 through 55
2.00 for unrodded lower core planes 56 through 61
2.13 for the core planes containing Bank D control rods

$PF_{xy} = 0.3$

where: from 12000 MWD/MTU to EOL

F_{xy}^{RTP} = *2.00* for unrodded upper core planes 1 through 6

1.87 for unrodded upper core planes 7 through 8

1.78 for unrodded upper core planes 9 through 31

1.80 for unrodded lower core planes 32 through 53

1.83 for unrodded lower core planes 54 through 55

1.95 for unrodded lower core planes 56 through 61

2.13 for the core planes containing Bank D control rods

PF_{xy} = *0.3*

2.4.4 If the Power Distribution Monitoring System (PDMS) is used for core power distribution surveillance and is OPERABLE, as defined in Technical Specification 3.3.3.14, the uncertainty, U_{FQ}, to be applied to the Heat Flux Hot Channel Factor $F_Q(z)$ shall be calculated by the following formula:

$$U_{FQ} = \left(1.0 + \frac{U_Q}{100.0}\right) \bullet U_e$$

where:

U_Q = Uncertainty for power peaking factor as defined in equation 5-19 of Analytical Method 3.5.
U_e = Engineering uncertainty factor.
= 1.03

Note: U_{FQ}= PDMS Surveillance Report Core Monitor Fxy Uncertainty in %.

2.4.5 If the INCORE movable detectors are used for core power distribution surveillance, the uncertainty, U_{FQ}, to be applied to the Heat Flux Hot Channel Factor $F_Q(z)$ shall be calculated by the following formula:

$$U_{FQ} = U_{qu} \bullet U_e$$

where:

U_{qu} = Base F_Q measurement uncertainty.
= 1.05
U_e = Engineering uncertainty factor.
= 1.03

2.5 Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ (Specification 3.2.3)

$$F^N_{\Delta H} = F_{\Delta H}^{RTP} [1.0 + PF_{\Delta H} (1.0 - P)]$$

where: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

2.5.1 $F_{\Delta H}^{RTP} = 1.65$

2.5.2 $PF_{\Delta H} = 0.3$

2.5.3 If the Power Distribution Monitoring System (PDMS) is used for core power distribution surveillance and is OPERABLE, as defined in Technical Specification 3.3.3.14, the uncertainty, $U_{F\Delta H}$, to be applied to the Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$, shall be the greater of 1.04 or as calculated by the following formula:

$$U_{F\Delta H} = 1.0 + \frac{U_{\Delta H}}{100.0}$$

where: $U_{\Delta H}$ = Uncertainty for enthalpy rise hot channel factor as defined in equation 5-19 of Analytical Method 3.5.

2.5.4 If the INCORE movable detectors are used for core power distribution surveillance, the uncertainty, $U_{F\Delta H}$, to be applied to the Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$ shall be calculated by the following formula:

$$U_{F\Delta H} = U_{F\Delta Hm}$$

where:

$U_{F\Delta Hm}$ =Base $F_{\Delta H}$ measurement uncertainty.
= 1.04

2.6 Boron Concentration (Specification 3.9.1)

A Mode 6 boron concentration, maintained at or above 2146 ppm, in the Reactor Coolant System, the fuel storage pool, the refueling canal, and the refueling cavity ensures the most restrictive of the following reactivity conditions is met:

a) A K-effective (K_{eff}) of 0.95 or less at All Rods In (ARI), Cold Zero Power (CZP) conditions with a 1% Δk/k uncertainty added.

b) A K_{eff} of 0.99 or less at All Rods Out (ARO), CZP conditions with a 1% Δk/k uncertainty added.

c) A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

3.0 ANALYTICAL METHODS

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

3.1 WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse proprietary), Methodology for Specifications listed in 6.9.1.9.a. Approved by Safety Evaluation dated May 28, 1985.

3.2 WCAP-8385, Power Distribution Control and Load Following Procedures – Topical Report, September 1974 (Westinghouse proprietary). Methodology for Specification 3 / 4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3.3 WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985 (Westinghouse proprietary), Methodology for Specification 3 / 4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

3.4 WCAP-10266-P-A, Revision 2, The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, March 1987 (Westinghouse proprietary), Methodology for Specification 3 / 4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

3.5 WCAP-12472-P-A, BEACON - Core Monitoring and Operations Support System, August 1994 (Westinghouse proprietary). Approved by Safety Evaluation dated February 16, 1994.

3.6 CENPD-397-P-A, Revision 01, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000. Approved by Safety Evaluation dated March 20, 2000.

3.7 WCAP-12472-P-A, Addendum 1-A, BEACON Core Monitoring and Operations Support System, January 2000 (Westinghouse proprietary). Approved by Safety Evaluation dated September 30, 1999.

4.0 REFERENCES

1. Salem Nuclear Generating Station Unit No. 1, Amendment No. 305, Renewed License No. DPR-70, Docket No. 50-272.

FIGURE 1

ROD BANK INSERTION LIMITS VS. THERMAL POWER

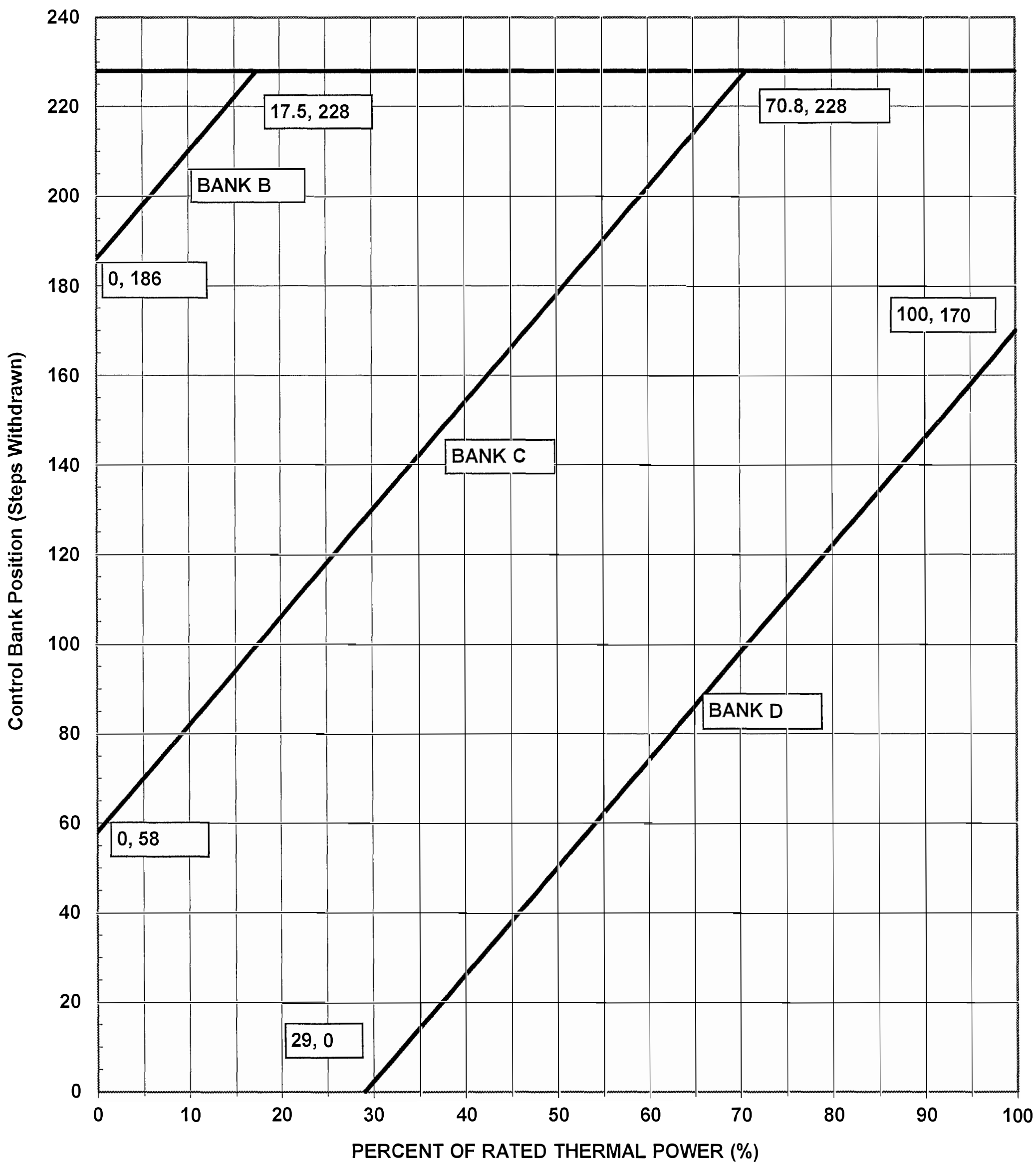


FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

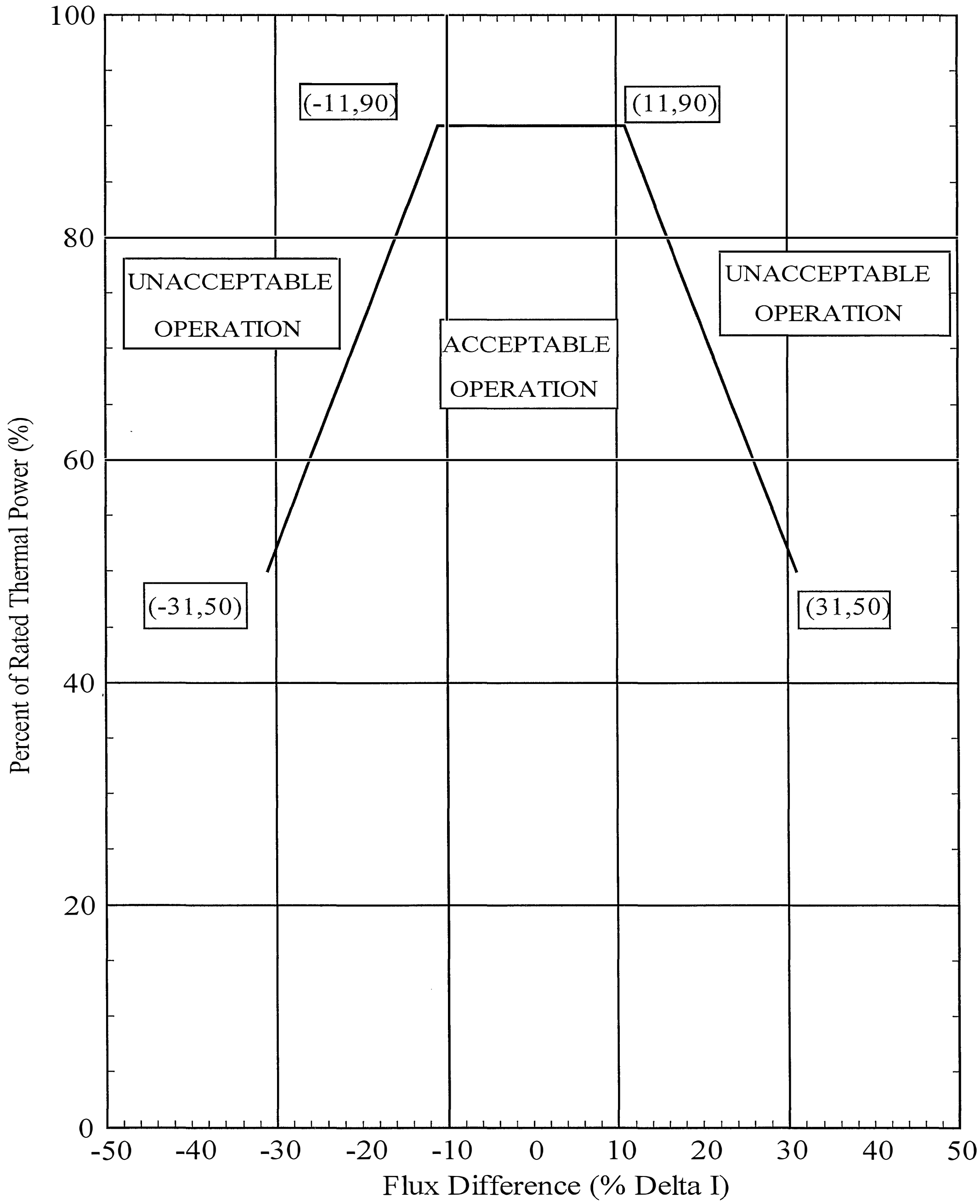
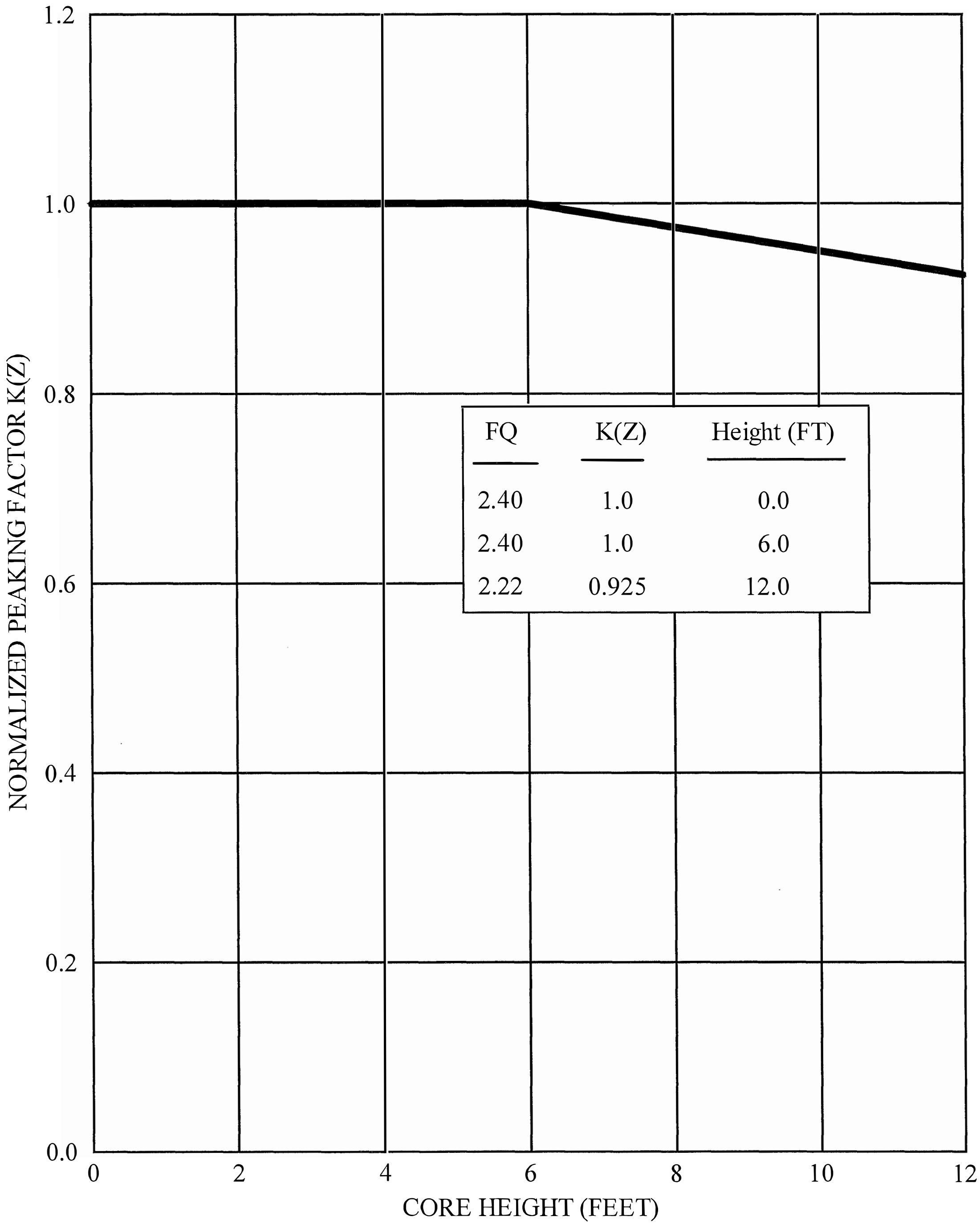


FIGURE 3

K(Z) – NORMALIZED FQ(Z) AS A FUNCTION OF CORE HEIGHT



FQ	K(Z)	Height (FT)
2.40	1.0	0.0
2.40	1.0	6.0
2.22	0.925	12.0