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SUBJECT: Forwards list of Exxon Nuclear Co computer codes (MODELS) used in current util ECCS calculations to further corroborate acceptability of Exxon audit.Power reduction to maintain compliance w/height dependent Fq limit unwarranted.

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NRC-85-69

WISCONSIN PUBLIC SERVICE CORPORATION
P.O. Box 1200, Green Bay, WI 54305

April 3, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. S. A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Exemption Request from 10 CFR 50.46

References:
1) Letter from D. C. Hintz (WPSC) to S. A. Varga (NRC) dated March 27, 1985 addressing errors in Exxon ECCS analyses

2) Letter from D. C. Hintz (WPSC) to S. A. Varga (NRC) dated February 26, 1985 providing additional information on Proposed T.S. Amendment 64 to KNPP Technical Specifications.

3) Meeting with NRC and Exxon Nuclear Co., 10 A.M. EST, March 29, 1985, Room P-422 Phillips Building, Bethesda, MD concerning topical report XN-NF-82-20(P) Rev. 1

4) Letter from C. W. Giesler (WPSC) to H. R. Denton (NRC) dated March 29, 1985 requesting exemption from 10 CFR 50.46

Reference 1 transmitted a summary of an audit performed at Exxon Nuclear by WPSC personnel to resolve NRC concerns regarding recent ECCS analyses. In summary, the audit concluded that:

1) The heat transfer augmentation factors were removed from the latest reanalysis for Kewaunee Nuclear Power Plant (KNPP) which was submitted in February, 1985 (Ref. 2).

2) The computer coding error was likewise corrected in the latest ECCS submittal (Ref. 2).

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3) There is reasonable assurance that the input data used in these calculations is correct.

During the meeting (reference 3) in which these concerns were further discussed, NRC requested that WPSC also provide a list of Exxon Nuclear Company Computer Codes (MODELS) used in the current KNPP ECCS calculations to further corroborate the acceptability documented in reference 1. Exxon Nuclear has prepared this information which is enclosed as attachment 1 to this letter.

In reference 4, WPSC requested an exemption from 10 CFR 50.46 and Appendix K as it relates to K(Z) in order to allow KNPP to operate at full power. NRC further requested during the same meeting (reference 3) that WPSC provide additional information related to the exemption request including:

1) The K(Z) resulting from the application of the currently approved Exxon ECCS models including the axial sensitivity studies, and

2) the analyses demonstrating the impact on plant operations.

The resulting K(Z) is shown in figure 1 of attachment 2. The supporting documentation supplied by Exxon Nuclear is also provided in attachment 2. Power distributions enveloped by this K(Z) function result in Peak Clad Temperatures (PCT) < 2200 °F using approved Exxon Evaluation models which comply with 10 CFR 50.46 and Appendix K.

The result of imposing this K(Z) on plant operations is evident in the figures of attachment 3. The predicted power distributions shown in figures 1 through 3 are for full power operation and are displayed along with the current K(Z) from the Technical Specifications and the proposed K(Z). The power reduction inferred from these calculations is 15%. Actual power reduction is expected to be slightly greater than this since reducing power will tend to accentuate top skew and require further power reduction.

To further substantiate this analysis, power distributions from the previous cycle of operation (Cycle 10) are also displayed in figures 4 through 6 along with the current K(Z) from the Technical Specifications and the proposed K(Z). It is clear that KNPP could not have been operated at full power during Cycle 10 with this restrictive K(Z) profile. The Cycle 11 power distributions are expected to be nearly identical to those seen in Cycle 10.

Unfortunately, there are no effective compensatory actions that can be taken to ameliorate this effect. The use of control rods is precluded for various reasons: 1) control rod insertion moves the limiting Fq down the core axially but the severity of the proposed K(Z) continues to prevail, 2) the resulting non-uniform burnup will exacerbate the problem when control rods are eventually withdrawn, and 3) additional administrative burdens on the operators when control rods are inserted during power operation make rod insertion undesirable.

Time is of the essence as the current refueling/maintenance outage is nearing completion. Initial criticality is currently scheduled for Sunday, April 7, with power production beginning on April 9. Based on our discussions with the staff, we understand that the single remaining concern related to our proposed

amendment 64 is the K(Z) issue. Therefore, we believe that startup and operation of KNPP can proceed without violating current Technical Specifications. However, WPSC intends to limit operation of the plant to ensure that the assumptions of the most recent analyses remain inviolate and thereby demonstrate compliance with 10 CFR 50.46 until your reviews are completed. This will be accomplished by limiting power level, as necessary, to maintain peaking factors below the most restrictive analysis.

In summary, we project that a power reduction will be necessary to maintain compliance with the height dependent Fq limit determined by Exxon Nuclear. We strongly believe that any power reduction regarding this issue is unwarranted, especially in light of our mutual agreement that this is a modelling problem and not representative of the significant margins that actually exist to the limits of 10 CFR 50.46. As you know, WPSC has responded promptly and completely to each of your concerns on this issue. We have taken extraordinary actions to quickly provide you the information you need to allow as much time as possible for your reviews. We request that you continue with your review expeditiously in order to minimize the impact on KNPP operation.

As always, we stand ready to discuss or meet with you on this issue.

Very truly yours,

D. C. Hintz
Manager - Nuclear Power

DR/js

cc - Mr. Robert Nelson, US NRC

Attachment 1

EXXON COMPUTER CODE LIST

Table 1 Kewaunee LOCA/ECCS Model Summary

		Prior Analysis	Prior Analysis	Current Analysis
1)	Fission Gas Release Model	WREM-II	EXEM	EXEM
2)	Stored Energy Model	WREM	EXEM	EXEM
3)	Blowdown Model	WREM	WREM	WREM
4)	Containment Model	WREM	WREM	WREM
5)	Clad Swelling and Rupture Model	WREM-II	EXEM-NUREG	EXEM-NUREG
6)	Reflood Model			
	a) Carryout and Quench Correlation	WREM	WREM	WREM
	b) Downcomer/Upper Plenum Leakage	WREM: No Leakage	WREM: No Leakage	WREM: No Leakage
	c) Break Model	CD = 0.4 Guillotine	CD = 0.4 Guillotine	CD = 0.4 Guillotine
	d) Core Outlet Enthalpy Model	OFF	EXEM	EXEM
	e) Z-Equivalent Model	OFF	OFF	OFF
7)	Heatup Model			
	a) Steam Cooling Model	WREM Swedish	EXEM	EXEM
	b) Heat Transfer Correlation	WREM-II	WREM-II	WREM-II
	c) Mixing Vane Multiplier	1.00	1.02	1.00
	d) Local Peaking Multiplier	1.00	1.129	1.00
	e) Z-Equivalent Model	WREM	WREM	EXEM
	f) Radiation Model	OFF	ON	ON
8)	Documentation of Results	XN-NF-79-72	XN-NF-84-31, Revision 1	Letter, RAC:009:85

Table 2 Exxon Nuclear Company ECCS Models

	Model	Reference
1. Fission Gas Release Model		
a. GAPEX	WREM	1
b. GAPEX with Uncertainties	WREM-II	1, 6
c. RODEX2	EXEM	13
2. Stored Energy Model		
a. GAPEX	WREM	1
b. RODEX2	EXEM	13
c. RODEX2 in RELAP4	EXEM	11c, 13
3. Blowdown Model	WREM	3, 5, 7
4. Containment Pressure Model		
a. Dry Containment	WREM	3
b. Ice Condenser Containment	WREM-II	6, 14
5. Clad Swelling and Rupture Model		
a. Exxon Model	WREM	3, 4
b. Revised Exxon Model	WREM-II	3, 4, 6
c. Exxon Model including NUREG-0630	EXEM	10
6. Reflood Model		
a. RELAP4	WREM	3
b. REFLEX	EXEM	8
c. Carryout and Quench Correlations		
1) 15x15 FLECHT	WREM	2
2) 17x17 FLECHT	EXEM	11a
3) 15x15/17x17 FLECHT	EXEM	11b
d. Downcomer/Upper Plenum Leakage	EXEM	11a
e. Break Model		
1) Split Break	EXEM	11a
2) Guillotine Break	EXEM	11a
f. Core Outlet Enthalpy Model	EXEM	11a
g. Z-Equivalent Model		
1) WREM	WREM	3
2) EXEM	EXEM	11d

Table 2 (Continued)

7.	Fuel Rod Heatup Model		
	a. TOODEE2	WREM	3
	b. Steam Cooling Model		
	1) WREM	WREM	3, 9
	2) WREM-II	WREM-II	6, 9
	3) EXEM	EXEM	11
	c. Heat Transfer Correlation		
	1) 15x15	WREM	3
	2) 15x15	WREM-II	6
	3) 17x17	EXEM	11a
	4) 15x15/17x17	EXEM	11b
	d. Mixing Vane HTC Multipliers		
	1) Off	WREM	3
	2) EXEM	EXEM	11a
	e. Local Peaking HTC Multipliers	EXEM	11
	1) Off	WREM	3
	2) EXEM	EXEM	11a
	3) D.C. Cook 2	EXEM	16
	f. Z-Equivalent Model		
	1) WREM	WREM	3
	2) EXEM	EXEM	11d
	g. Radiation Model		
	1) WREM	WREM	3
	2) WREM Expanded New Geometries	EXEM	11d
8.	Core Wide Metal-Water Reaction	WREM	15

REFERENCES

1. XN-73-25, "GAPEXX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," Exxon Nuclear Company, Inc., Richland, WA 99352, August 1973.

2. XN-75-19, "Carryout Rate Fraction Correlation for Pressurized Water Reactors," Exxon Nuclear Company, Inc., Richland, WA 99352; (a) March 24, 1975; (b) Supplement 1, "Statistical Evaluation of the Carryout Rate Fraction," June 1975.

3. XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA 99352.

   a. Volume 1, July 25, 1975.

   b. Volume 2, "Model Justification," August 1, 1975.

   c. Supplement 1, "Further Definitions and Justifications to Reflood Heat Transfer Models," August 14, 1975.

   d. Supplement 2, "Supplementary Information Related to Blowdown and Reflood Analysis," August 14, 1975.

   e. Supplement 3, "Supplementary Information Related to Blowdown and Heatup Analysis," August 16, 1975.

   f. Supplement 4, "Supplementary Information Related to Blowdown and Heatup Analysis," August 20, 1975.

   g. Supplement 5, "Supplementary Information Related to Blowdown and Heatup Analysis," October 3, 1975.

   h. Supplement 6, "Supplementary Information Related to Blowdown and Heatup Analysis," October 27, 1975.

   i. Supplement 7, "Supplementary Information," November 9, 1975.

   j. Volume II Appendix A and B, "3-Loop Westinghouse Sample Problem," August 1, 1975.

   k. Volume II Appendix C, "Yankee Rowe Example Problem," August 22, 1975.

   l. Volume II Appendix D, "3-Loop Westinghouse Large Break Example Problem (Using September 26, 1975 Model)," October 2, 1975.

   m. Volume III Revision 2, "Small Break Model," August 20, 1975.

REFERENCES (Continued)

4. XN-75-6, "Flow Blockage Model for LOCA Analysis," Exxon Nuclear Company, Inc., Richland, WA 99352, April 1, 1975.

5. XN-75-43, "Core Physics Methods and Data Used as Input to LOCA Analysis," Exxon Nuclear Company, Inc., Richland, WA 99352, August 1975.

6. XN-76-27(A), "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II," Exxon Nuclear Company, Inc., Richland, WA 99352; (a) March 1977; (b) Supplement 1(A), "Supplementary Information Relating to," March 1977; and (c) Supplement 2(A), "Supplementary Information Relating to," March 1977.

7. XN-76-44, "Revised Nucleate Boiling Lockout for ENC WREM-Based ECCS Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA 99352, September 1976.

8. XN-NF-78-30(A), "Exxon Nuclear Company WREM-Based Generic PWR ECCS Model Updates ENC WREM-IIA," Exxon Nuclear Company, Inc., Richland, WA 99352, May 1979.

9. Letter, G.F. Owsley (Exxon Nuclear Company) to D.F. Ross (USNRC), Subject: TOODEE2 Updates; Letter No. GFO:077:80 dated April 1, 1980.

10. XN-NF-82-07(P)(A), Rev. 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., Richland, WA 99352, November 1982.

11. XN-NF-82-20(P), "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Inc., Richland, WA 99352; (a) Revision 1, August 1982; (b) Revision 1, Supp. 1, "Revised FLECHT-Based Reflood Carryover and Heat Transfer Correlations," June 1983; (c) Revision 1, Supp. 2(A), February 1985; (d) Revision 1, Supp. 3, "Response to NRC Request for Additional Information," Draft; (e) Revision 1, Supp. 4(A), "Adjustments to FLECHT-Based Heat Transfer Correlations," July 1984.

12. XN-NF-82-49(P), "Exxon Nuclear Company Evaluation Model - EXEM/PWR Small Break Model," Exxon Nuclear Company, Inc., Richland, WA 99352; (a) June 1982; (b) Supp. 1, "Supplement 1: Responses to NRC Questions," March 1985.

13. XN-NF-81-58(P)(A), Revision 2, Supps. 1 & 2, "RODEX2 Fuel Rod Thermal Response Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA 99352, March 1984.

14. XN-CC-39, Rev. 1, "ICECON: A Computer Program Used to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)," Exxon Nuclear Company, Inc., Richland, WA 99352, November 1978.

REFERENCES (Continued)

15. XN-CC-36, "Exxon Nuclear Procedure for Calculating Core-Wide Metal-Water Reaction During a Loss-of-Coolant Accident," Exxon Nuclear Company, Inc., Richland, WA 99352, December 1975.

16. Letter, J.C. Chandler (Exxon Nuclear Company) to H.R. Denton (USNRC), Subject: Local Peaking Multiplier for Reflood Heat Transfer Coefficient; Letter No. JCC:076:84 dated May 7, 1984.

Attachment 2

RESULTS OF KEWAUNEE K(z)
ECCS ANALYSIS

Attachment 2: Results of Kewaunee K(z) ECCS Analysis

In order to determine the axially dependent F_q limits for Kewaunee which can be supported by currently approved ENC LOCA-ECCS methodology, ENC performed a LOCA-ECCS analysis for upskewed axial power profiles. The analysis was performed for the identified limiting LOCA break for Westinghouse 2-loop PWRs, the double-end guillotine break of the cold leg pipe with a discharge coefficient of 0.4 (0.4 DECLG). Previous ENC experience has shown that an analysis of the axial profile having the power peak at the highest elevation in the core can be used along with the center-peaked axial profile results to define a conservative axially dependent F_q relationship which will meet the NRC 10 CFR 50.46 criteria. The power distribution used in the analysis was developed as follows: (1) ENC neutronics and fuel management provided the axial power shape having the highest elevation upskew for the Kewaunee reactor. This axial power shape was adjusted to correspond to the maximum allowed axial offset. (2) A fuel assembly radial power peaking was chosen which was estimated to give a PCT near 2200°F with the above axial power profile. A LOCA-ECCS calculation was performed using this power distribution and PWR LOCA Evaluation Model codes currently accepted by the USNRC. The limiting break calculation used system blowdown, refill, and containment results from previous Kewaunee analyses since these results will not change significantly for a revised core power distribution. Calculations were performed for initial stored energy using RODEX2, blowdown core behavior using RELAP4-EM, system reflood using REFLEX, and refill and reflood hot rod temperatures using TOODEE2. The change in PCT for UPI effects was computed using the current interim model. Final results are given in Table 1.

The large break $F_q(z)$ limit is derived by plotting the center peaked power shape and the upskewed power shape for which the PCT is less than 2200°F. A line tangent to these power shapes is drawn to represent the $F_q(z)$ limit in the upper portion of the core. A line of constant F_q equal to the maximum F_q in the center peaked power shape is drawn for the bottom portion of the core. The large break $F_q(z)$ limit which results from this analysis is shown in Figure 1.

Attachment 2

Table 1 Kewaunee LOCA-ECCS K(z) Analysis Results

Analysis Results	2500 MWD/MTM Peak Rod Average Exposure
Peak Clad Temperature (PCT), °F***	2100
ΔPCT for UPI, °F	-26
Time of PCT, sec	264
Peak Clad Temperature Location, ft	10.75
Local Zr/H_2O Reaction (max), %*	11.2
Local Zr/H_2O Location, ft from bottom	10.0
Total H_2 Generation, % of total Zr reacted	<1.0
Hot Rod Burst Time, sec	79
Hot Rod Burst Location, ft	10.0
Calculation Basis	
License Core Power, MWt	1650
Power Used for Analysis, MWt**	1683
Peak Linear Power for Analysis, kW/ft**	10.29
Total Peaking Factor, F^T_Q	1.59
Enthalpy Rise, Nuclear, $F^T_{\Delta H}$	1.41
Steam Generator Tube Plugging	5.00

* Computed value at 380 seconds
** Including 1.02 factor for power uncertainties
*** Includes PCT for UPI

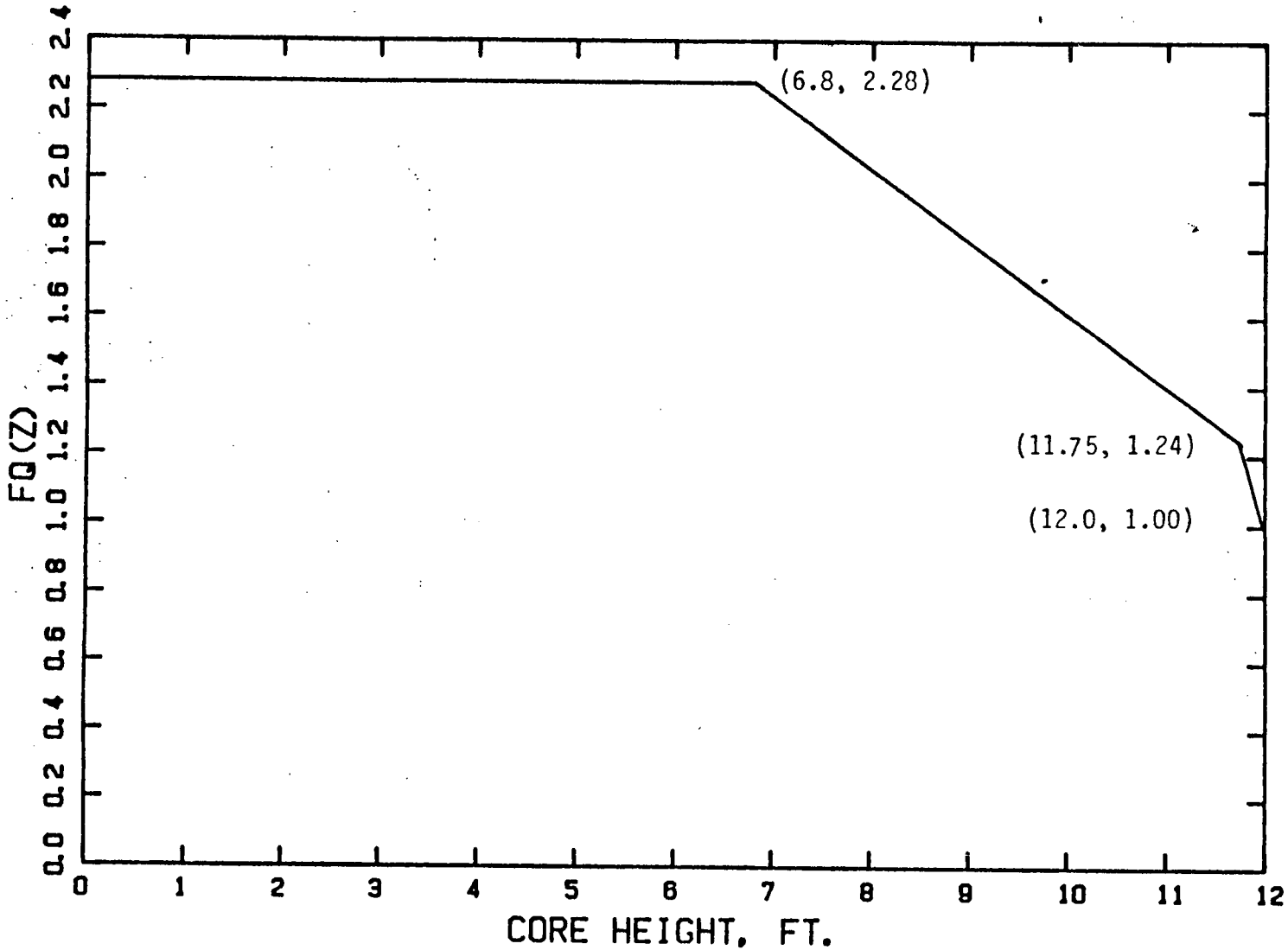
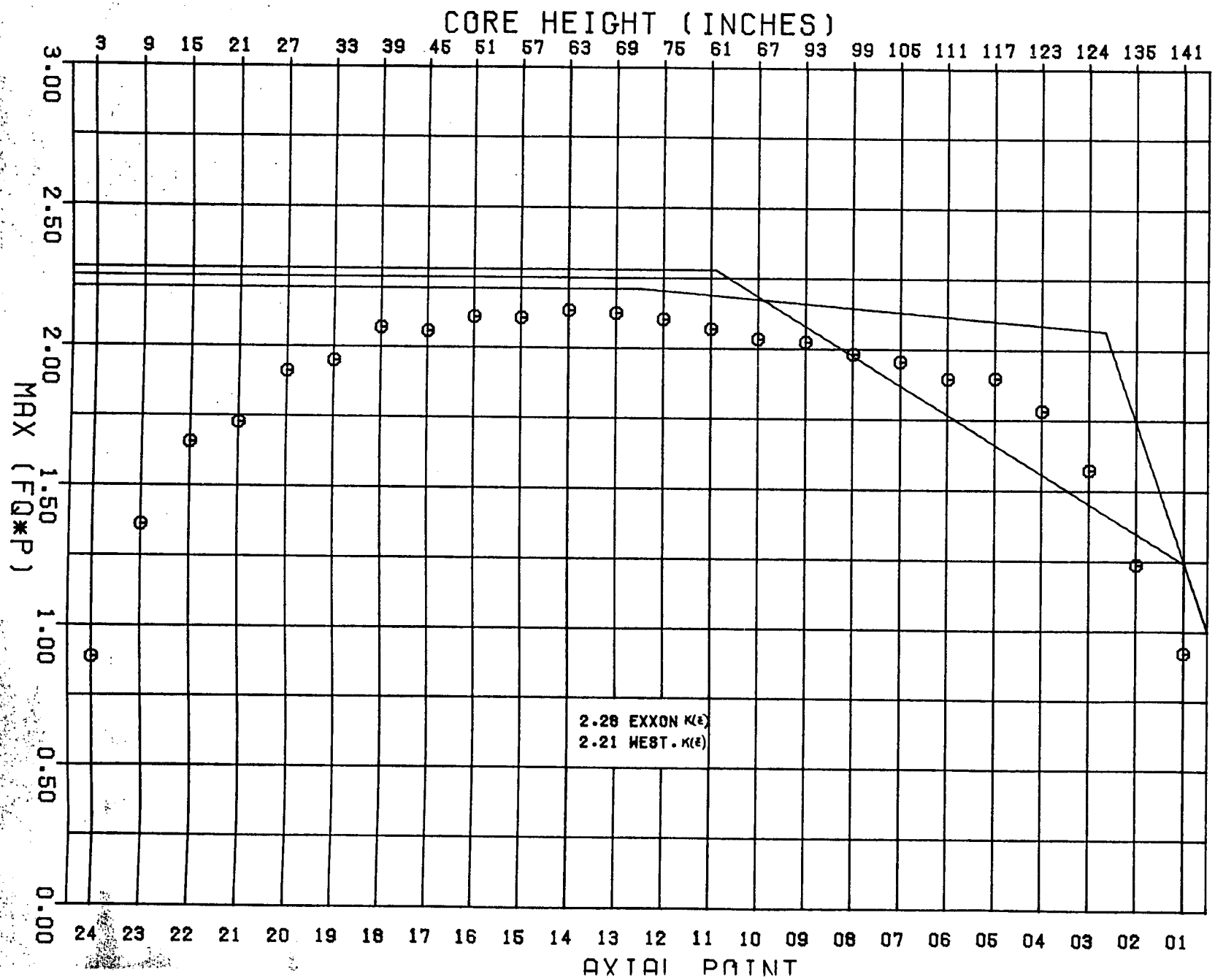


Figure 1 Kewaunee $F_q(z)$

Attachment 3

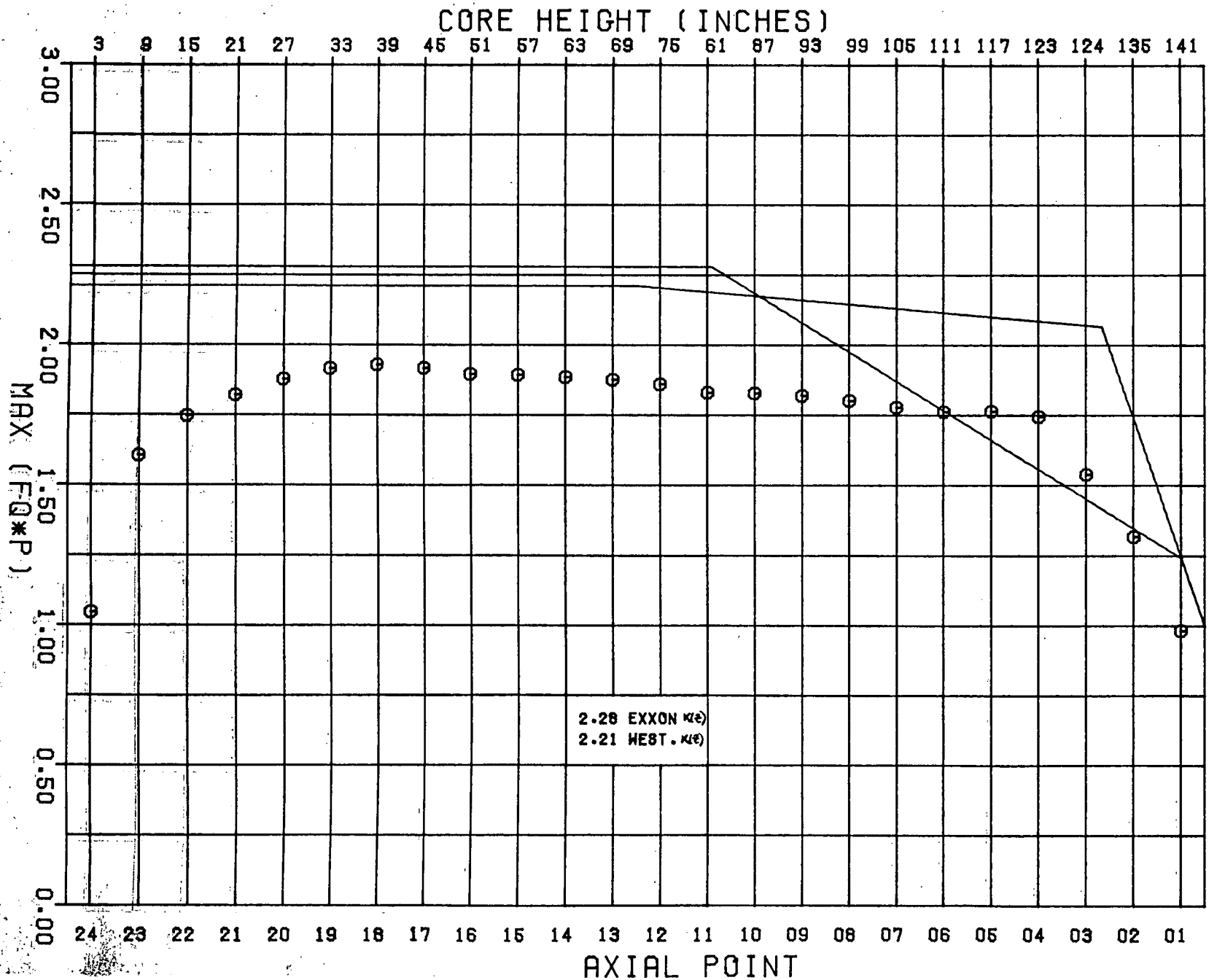
POWER DISTRIBUTIONS PLOTS FOR KNPP

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 11 BOL
S3D 85050.1029



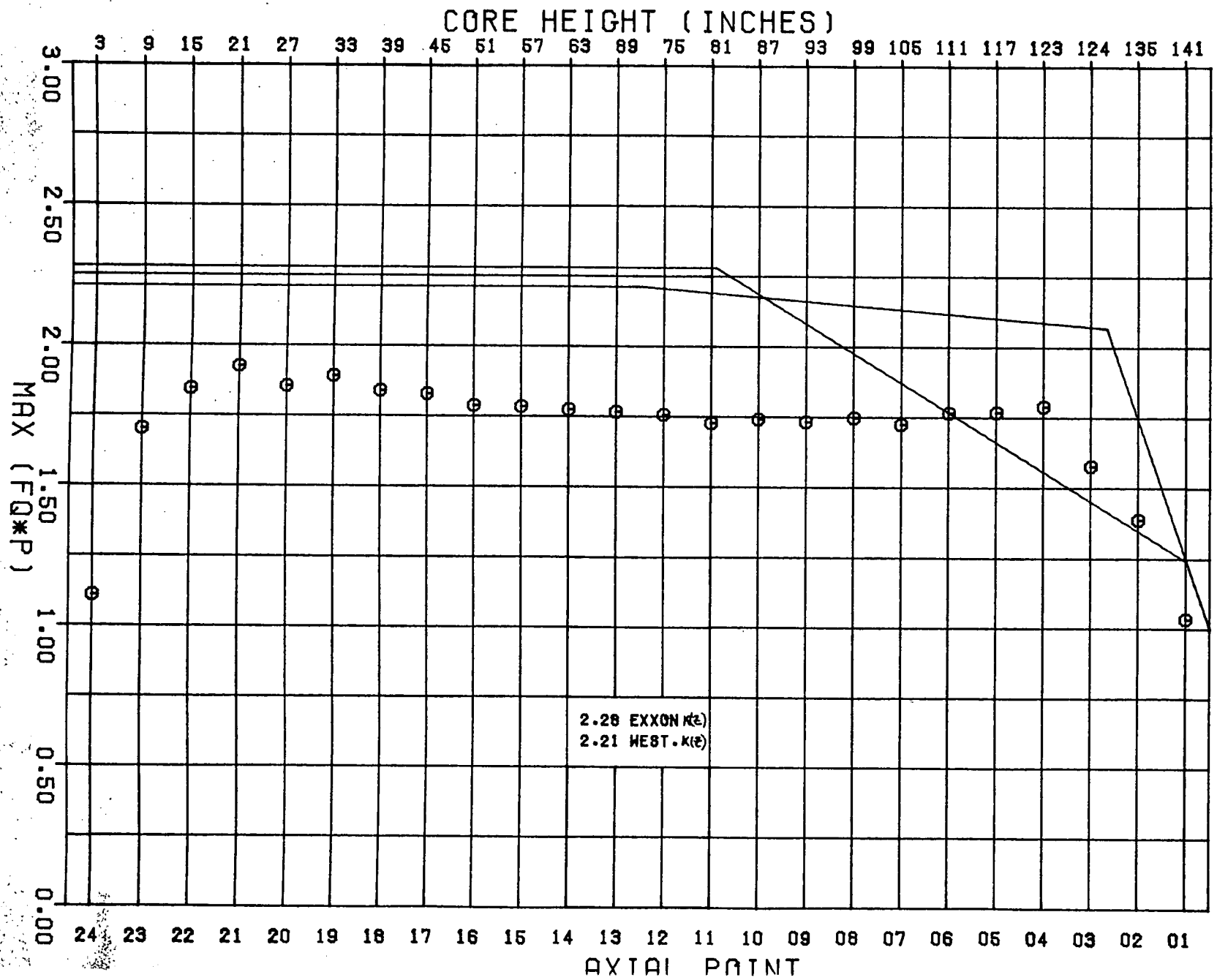
Attachment 3
Figure 1

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 11 MOL
S3D 85050.1029



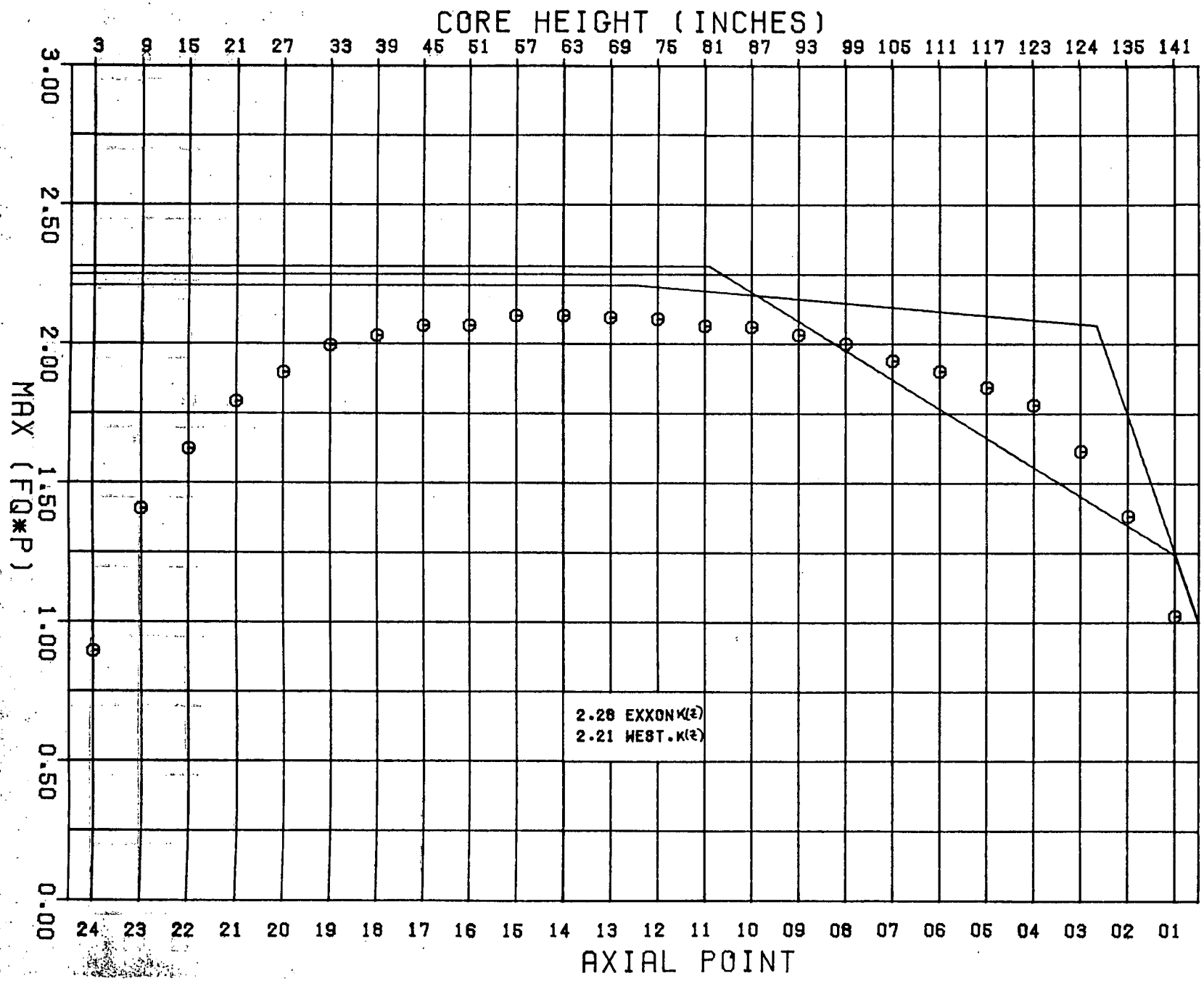
Attachment 3
Figure 2

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 11 EOL
S3D 85050.1029



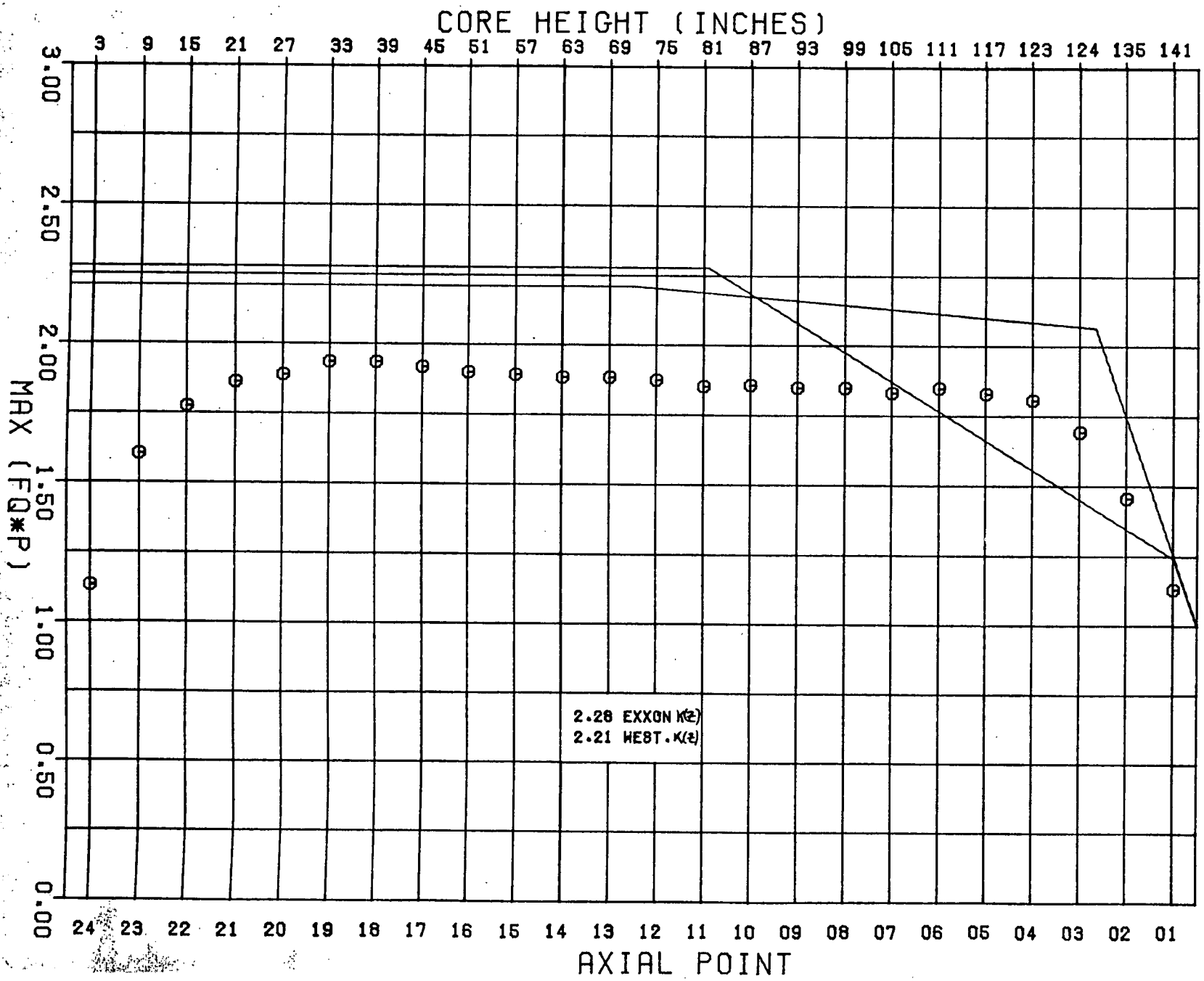
Attachment 3
Figure 3

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 10 BOL
FM1009



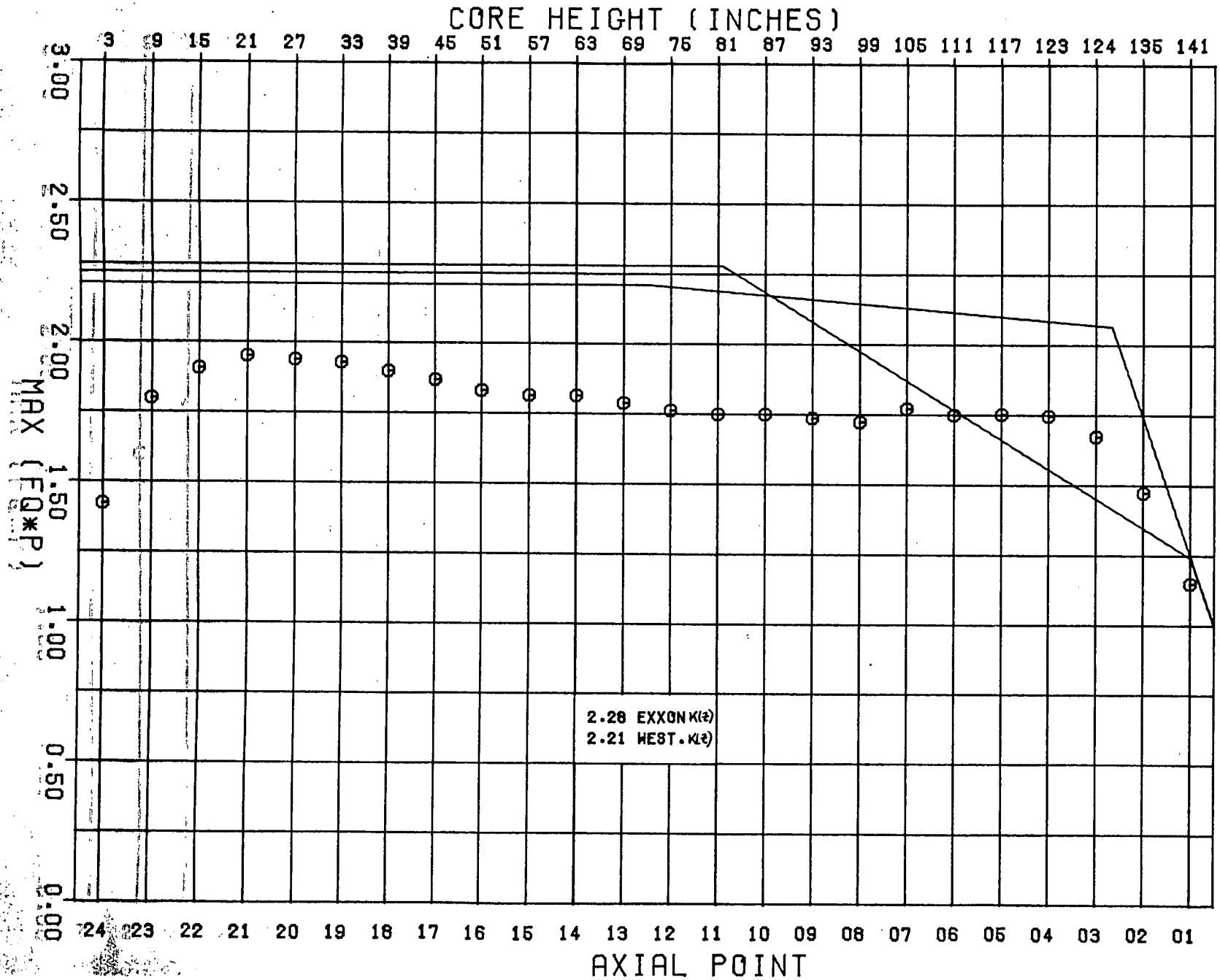
Attachment 3
Figure 4

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 10 MOL
FM1024



Attachment 3
Figure 5

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 10 EOL
FM1043



Attachment 3
Figure 6