Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

April 4, 1985
LIC-85-141

Mr. James R. Miller, Chief
Office of Nuclear Reactor Regulation
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: 1) Docket No. 50-285

2) Letter from OPPD (R. L. Andrews) to NRC (James R. Miller) dated March 25, 1985, LIC-85-122

Dear Mr. Miller:

Error in LOCA-ECCS Analysis for Fort Calhoun Station

On March 25, 1985, Omaha Public Power District provided Reference 2, its initial response, to the error in Exxon Nuclear Company's (ENC) LOCA coding. Subsequently, a meeting was held on Friday, March 29, with the NRC staff, ENC representatives, and utilities using ENC methodology for LOCA analysis in attendance. Based on discussions during this meeting, the Omaha Public Power District is providing the following information relative to the ENC LOCA analysis applicable to the Fort Calhoun Station.

A description of models used by ENC in the Fort Calhoun LOCA analysis for Cycle 8 and a description of models used in the reanalysis after the ENC coding misformulation are contained in Table 1. Table 2 lists the ENC models and their appropriate references.

Non-uniform Fort Calhoun specific axial power distributions were generated for use in the Fort Calhoun LOCA analysis performed by ENC as discussed in Section 14.15 of the USAR. These axial power distributions were used by ENC in the generation of the enclosed F_Q^T versus core height curve.

The F_Q^T value of 2.53 is equivalent to a Peak Linear Heat Generation Rate (PLHGR) limit of 15.22 kw/ft. The PLHGR limit is a Limiting Condition of Operation (LCO) and is included as Technical Specification Figure 2-5. This LCO is monitored in accordance with Technical Specification 2.10.4(1). The linear heat rate limit is normally monitored by the incore detectors which are 40 cm long and their midpoints are nominally located at 20, 40, 60, and 80 percent of core height. This incore detector monitoring system assures that the F_Q^T limit is maintained at or below 80

8504100447 850404
PDR ADOCK 05000285
P PDR

A001
1/1

percent of core height. The LCO for F_{xy}^T (Technical Specification Figure 2-9) in conjunction with the full power DNBR LCO axial shape limit (Technical Specification Figure 2-7) preclude the total peaking factor above the 80 percent core height level from reaching the limits prescribed in the enclosed figure. As a normal part of the generation or verification of the above-mentioned LCO, calculations are performed for reload cores to verify that total peaking above the 80 percent core height will not exceed the limits defined in the enclosed figure.

Based on Reference 2 and the enclosed information, the District finds that Fort Calhoun Station remains in compliance with 10 CFR 50.46.

Sincerely,

J K Gasper

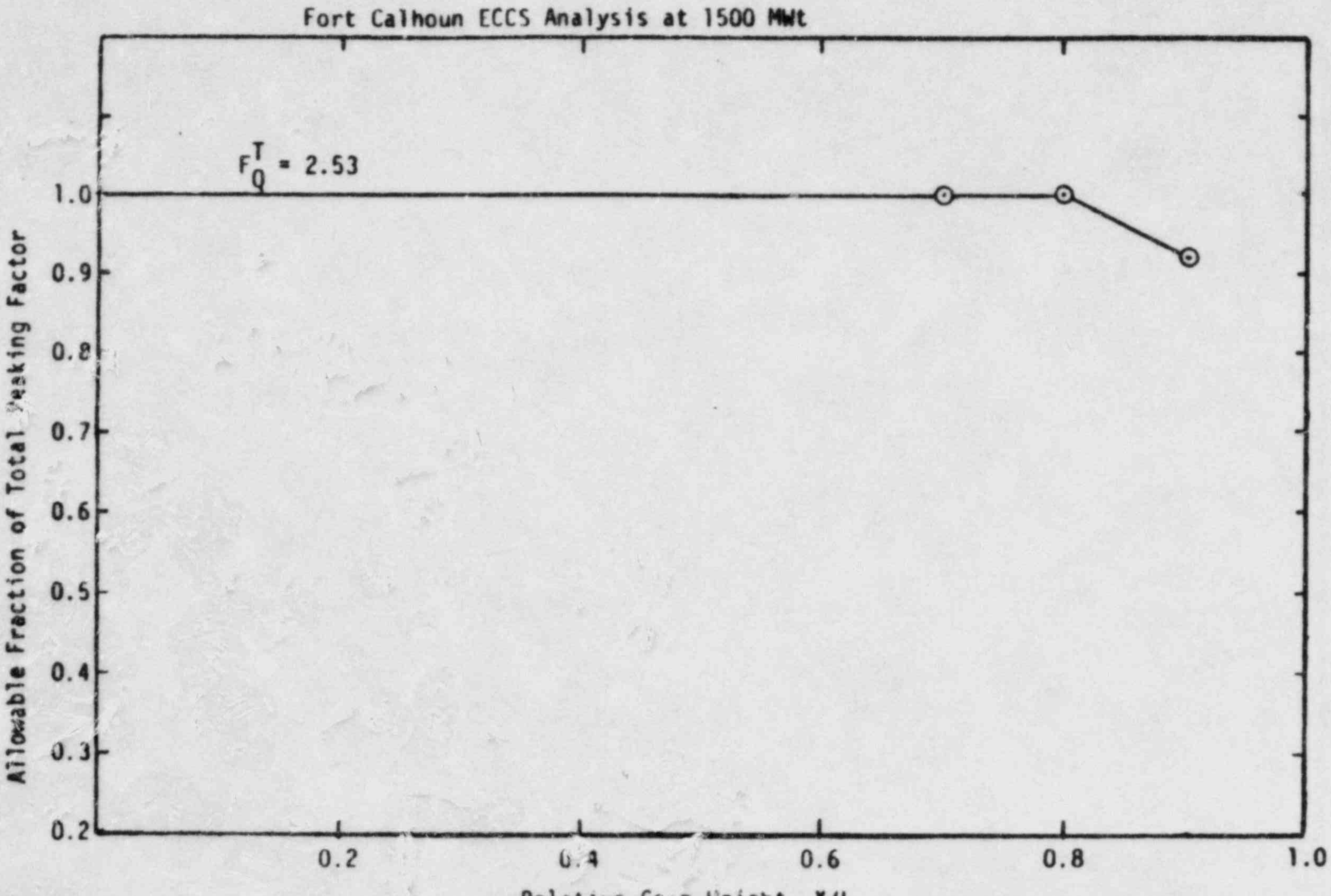
R. L. Andrews for
Division Manager
Nuclear Production

RLA/JDK/lp

Enclosures

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

Mr. E. G. Tourigny, NRC Project Manager
Mr. L. A. Yandell, NRC Senior Resident Inspector



Limits On Total Peaking Versus Axial Power Peak Location

Table 1 Fort Calhoun LOCA/ECCS Model Summary

			Current* Analysis
1)	Fission Gas Release Model	WREM	WREM
2)	Stored Energy Model	WREM	WREM
3)	Blowdown Model	WREM	WREM
4)	Containment Model	WREM	WREM
5)	Clad Swelling and Rupture Model	EXEM-NUREG	EXEM-NUREG
6)	Reflood Model		
	a) Carryout and Quench Correlation	WREM	WREM
	b) Downcomer/Upper Plenum Leakage	WREM: No Leakage	WREM: No Leakage
	c) Break Model	CD = 1.0 Guillotine	CD = 1.0 Guillotine
	d) Core Outlet Enthalpy Model	EXEM	EXEM
	e) Z-Equivalent Model	OFF	OFF
7)	Heatup Model		
	a) Steam Cooling Model	EXEM	EXEM
	b) Heat Transfer Correlation	WREM	WREM
	c) Mixing Vane Multiplier	1.0	1.0
	d) Local Peaking Multiplier	1.045	1.0
	e) Z-Equivalent Model	WREM	WREM
	f) Radiation Model	ON	ON
8)	Documentation of Results	USAR 14.15 Loss of Coolant Accidents	Letter LIC-85-122

* Current analysis performed to correct error in code TOODEE2.

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.