VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 12, 1981

W. N. THOMAS
VICE PRESIDENT
FUEL RESOURCES

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No: 359
FR/KLB: plc
Docket Nos.: 50-280
50-281
50-338
50-339
License Nos.: DPR-32
DPR-37
NPF-4
NPF-7

Dear Mr. Denton:

TOPICAL REPORT VEP-FRD-33
"VEPCO REACTOR CORE THERMAL-HYDRAULIC ANALYSIS
USING THE COBRA IIIC/MIT COMPUTER CODE"

Attachment 1 provides our responses to Nuclear Regulatory Commission (NRC) Staff questions on the Vepco topical report VEP-FRD-33," "Vepco Reactor Core Thermal-Hydraulic Analysis Using The COBRA IIIC/MIT Computer Code", transmitted by the W. N. Thomas (Vepco) to H. R. Denton (NRC) letter, Serial No. 795, dated September 28, 1979. These questions on VEP-FRD-33 were sent in a letter from R. L. Tedesco (NRC) to W. N. Thomas (Vepco), dated April 9, 1981.

Should you have any further questions concerning this topical report, please contact us.

Very truly yours,

W. N. Thomas

W. N. Thomas

Attachment

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cc: Mr. R. L. Tedesco
Assistant Director
for Licensing
Division of Licensing

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ATTACHMENT 1

Response to NRC Questions on VEP-FRD-33

NRC Question 492.1

"Provide CHF predictions (plots and tables) versus the measured values for the test data (Ref. 1) using VEP-FRD-33 COBRA/W-3. Include at least two points from each set of tests in the Ref. 1 test data such that your test conditions will be similar to the limiting thermal-hydraulic operating conditions for Surry Units 1 and 2 and North Anna Units."

Response

COBRA models of the Ref. 1 3x3 and 4x4 test bundle geometries were created using code correlations and options consistent with VEP-FRD-33. Three data points from each test series were chosen such that the range of key test parameters would be maximized. The data ranges are given in Table I. The North Anna and Surry allowable operating conditions are well within these ranges.A comparison of the COBRA/W-3 DNB predictions and the experimental DNB data is given in Table II. These results are presented graphically in Figure 1. The sample mean and standard deviation of the measured-to-predicted heat flux ratio are 0.982 and 0.0638, respectively. These limited data indicate that in order to meet a 95% probability/95% confidence level reactor design criteria, a minimum DNBR of 1.19 would be required. VEPCO intends

Response to NRC Question 492.1 (continued)

to continue using the 1.30 minimum DNBR design criteria established by the original W-3 correlation data. These original W-3 correlation bounds are also indicated on Figure 1. Should we desire to use a minimum DNBR design basis other than 1.30, additional justification would be provided.

Table I: Range of Key Test Parameters

	Ranges
Pressure (psia)	1491-2433
Inlet Average Mass Velocity ($Mlbm/hr\text{-}ft^2$)	1.05-3.66
Inlet Temperature (°F)	433.0-617.0
Local Heat Flux ($MBTU/hr\text{-}ft^2$)	0.563-1.063

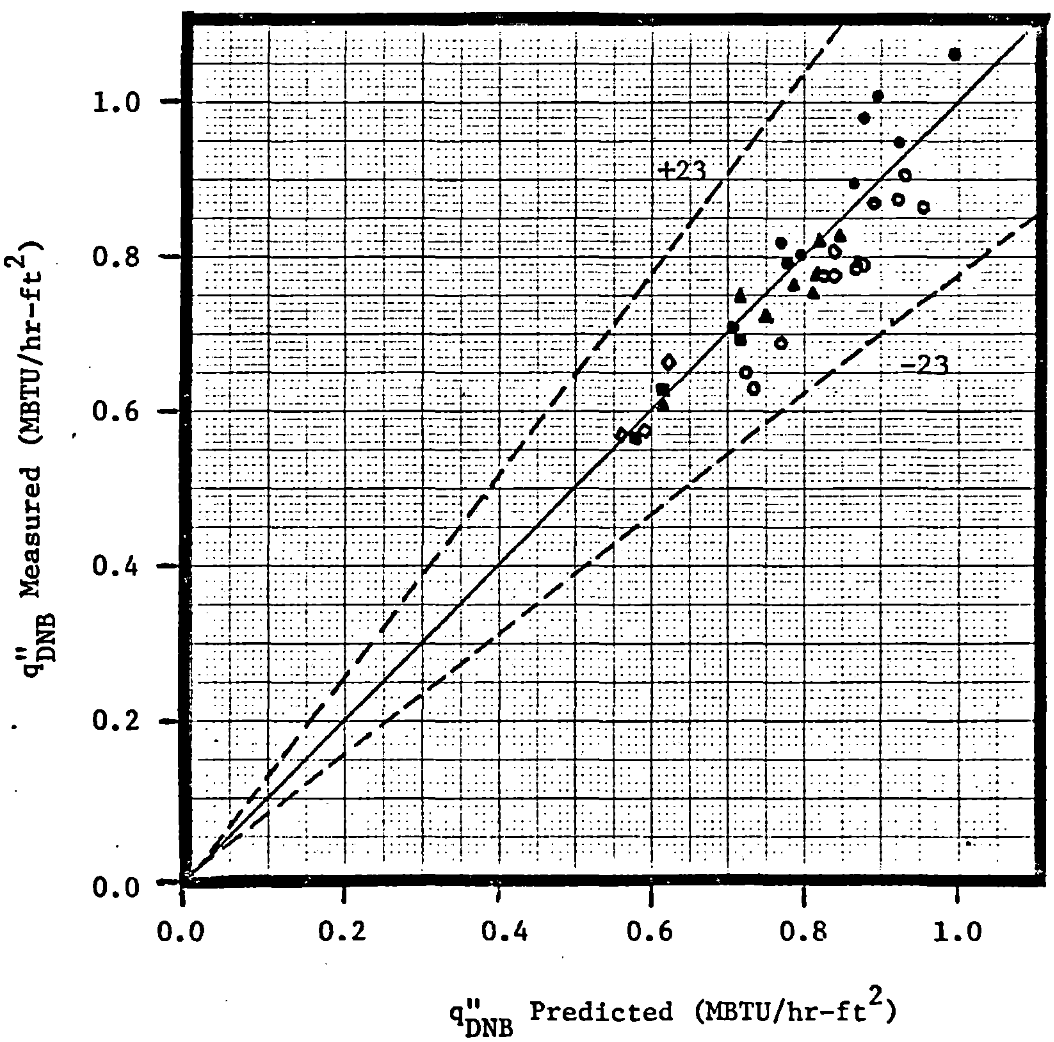
Table II: Comparison with DNB Data

Test Section	Run no.	Pressure (psia)	Inlet Temp. (°F)	Inlet Avg. Mass Vel. ($Mlb/hr-ft^2$)	Local DNB ($MBTU/hr-ft^2$) qmeas	qpred.	qmeas/qpred
I	4	1502	468.0	1.05	0.631	0.734	0.860
I	5	1502	480.0	2.03	0.803	0.839	0.957
I	6	1503	518.5	3.05	0.870	0.892	0.975
II	11	2100	567.0	2.55	0.819	0.768	1.065
II	15	1808	579.0	3.55	0.801	0.791	1.013
II	60	2115	567.5	3.06	0.893	0.865	1.032
III	21	1514	483.0	2.56	0.692	0.714	0.969
III	25	2091	544.0	2.55	0.623	0.614	1.015
III	46	1799	559.0	3.01	0.566	0.573	0.988
IV	71	1509	507.0	2.58	0.779	0.818	0.952
IV	75	1811	567.0	3.58	0.763	0.781	0.977
IV	81	2109	546.0	2.56	0.752	0.811	0.927
V	93	1502	476.0	1.57	0.751	0.711	1.056
V	99	1811	553.0	3.64	0.794	0.870	0.913
V	103	2109	560.0	2.58	0.722	0.749	0.964
VI	129	1541	540.0	3.63	0.829	0.843	0.983
VI	135	1813	560.0	3.57	0.820	0.819	1.001
VI	144	2433	615.0	3.13	0.608	0.612	0.993
VII	208	2026	536.7	2.61	0.795	0.874	0.910
VII	211	1497	478.3	2.55	0.906	0.932	0.972
VII	216	1790	501.0	2.07	0.777	0.839	0.926
VIII	219	1491	481.0	2.55	0.868	0.919	0.945
VIII	225	2105	565.3	2.55	0.690	0.769	0.897
VIII	235	2415	583.7	3.59	0.866	0.955	0.907
IX	254	1491	499.0	2.57	0.988	0.880	1.123
IX	264	2069	583.0	3.06	0.793	0.778	1.019
IX	270	2400	586.7	3.66	1.063	0.995	1.068
X	275	1497	537.7	3.59	0.949	0.918	1.034
X	277	1799	558.3	3.58	1.008	0.892	1.130
X	290	2419	617.0	3.05	0.704	0.706	0.997
XI	346	1815	561.7	3.57	0.783	0.873	0.897
XI	356	2395	604.3	3.03	0.648	0.726	0.893
XI	378	1496	433.0	1.53	0.778	0.825	0.943
XII	380	1854	568.0	3.53	0.662	0.621	1.066
XII	386	2098	578.0	3.08	0.576	0.589	0.978
XII	392	2403	584.0	2.53	0.563	0.559	1.007

FIGURE 1

COMPARISON OF DNB DATA WITH COBRA PREDICTIONS

Heated Length	Axial Flux Distribution	Grid with Vane	Grid without Vane
8'	u sin u cosine u	● ○	▲ –
14'	u sin u cosine u	■ ◇	– –



NRC Question 492.2

"Confirm that VEP-FRD-33 computer code will be used only for non-LOCA thermal-hydraulic analysis."

Response

The VEP-FRD-33 computer code will be used only for non-LOCA thermal hydraulic analysis.

NRC Question 492.3

"Confirm the applicability range for the key parameters such as temperature, quality, pressure, flow, etc., for the use of your DNB correlation."

Response

The range of the key parameters associated with the W-3 correlation, the L-grid factor, the cold wall factor and the non-uniform heat flux multiplier are included in Table III. These are supported by the references also indicated in Table III. Since these ranges bound the operating conditions present in the Condition II, III, and IV transients for which DNB is a concern, we intend to use the W-3 correlation for those transients. Any DNB calculation performed at pressures less than 1500 psia will not include the spacer factor correction because of its limited pressure range.

Table III: W-3 Correlation Limits

Correlation	Ref. No.	Pressure Range (psia)	Mass Velocity (Mlb/h-f²)	Equiv. Diameter (in)	Local Quality	Axial Height (in)	Inlet Temp (°F)
W-3	1,2	1000-2400	1.0-5.0	0.2-0.7	≤0.15	10-144	>400
F-factor	1,2	1000-2400	1.0-3.0	0.2-0.7	≤0.15	10-144	
Coldwall Factor	1,2 3,4	1000-2400	1.0-5.0		≤0.15	>10	
Spacer Factor	3,4	1490-2440	1.5-3.7		≤0.15	96-168	404-624

References

1. E.R. Rosal, J.O. Cermak, L.S. Tong, J.E.Casterline, S.Kokolis, and B. Matzer, "High Pressure Rod Bundle DNB Data with Axially Non-Uniform Heat Flux," Nuclear Engineering and Design Vol 31 (1974), No.1, pp.1-20

2. L.S. Tong, "Boiling Crisis and Critical Heat Flux," U.S. AEC Critical Review Series, 1972

3. F.F. Cadek, F.E. Motley, "Application of Modified Spacer Factor to L-Grid Typical and Cold Wall Cell DNB," WCAP-8030-A, January 1975

4. F.F. Cadek, F.E. Motley, "DNB Test Results for R-Grid Thimble Cold Wall Cells," WCAP-7958 Add. 1, January 1975