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Response to NRC questions concerning XN-75-27, Suppl #1 "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors".....

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INDIANA & MICHIGAN POWER COMPANY
P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

December 9, 1976

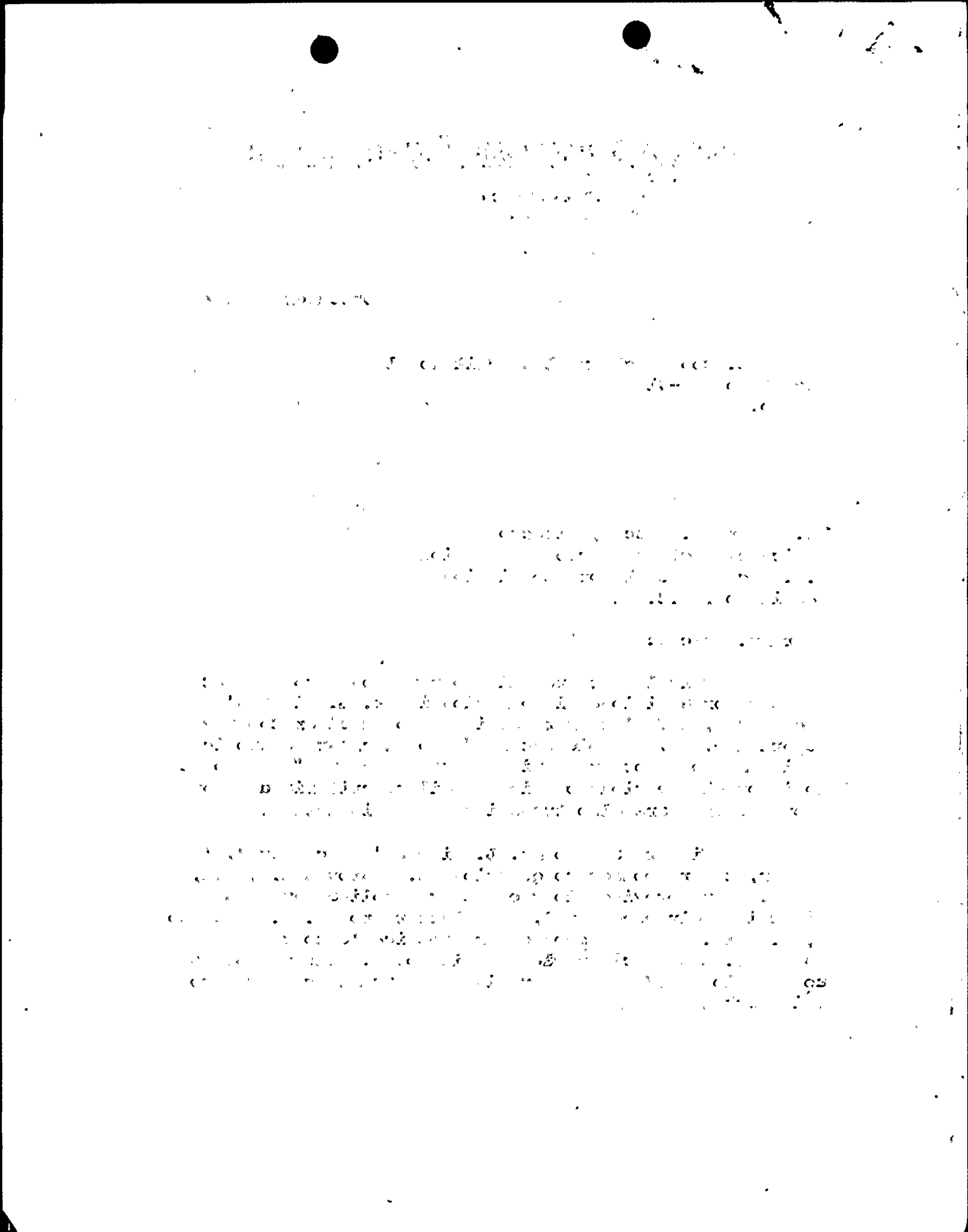
Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
DPR No. 58

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Rusche:

This letter transmits our response to the NRC request for additional information in Mr. D. Ziemann's December 2, 1976 letter regarding Exxon Nuclear Company Report XN-75-27, Supplement 1, "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors." Responses to informal questions on single failure criteria and core stored energy are also transmitted by this letter.

With regard to Mr. D. Ziemann's December 2, 1976 letter, the responses to questions A.1 through A.6, B.1, and B.2 were provided to the NRC by Carolina Power and Light in their December 2, 1976 letter from E. E. Utley to R. W. Reid. These responses are applicable to the Donald C. Cook Nuclear Plant, Unit No. 1. The responses to questions A.7 and A.8 are included as Attachment A to this letter.



Our responses to the informal NRC staff questions are included as Attachment B to this letter.

Very truly yours,

John Tillinghast
Vice President

JT:mam
Attachment
Sworn and subscribed to before
me this 9 th day of December 1976
in New York County, New York

David G. Hume
Notary Public

DAVID G. HUME
NOTARY PUBLIC, State of New York
No. 31-4608113
Qualified in New York County
Commission Expires March 30, 1977.

cc: G. Charnoff
R. C. Callen
R. J. Vollen
P. W. Steketee
R. Walsh
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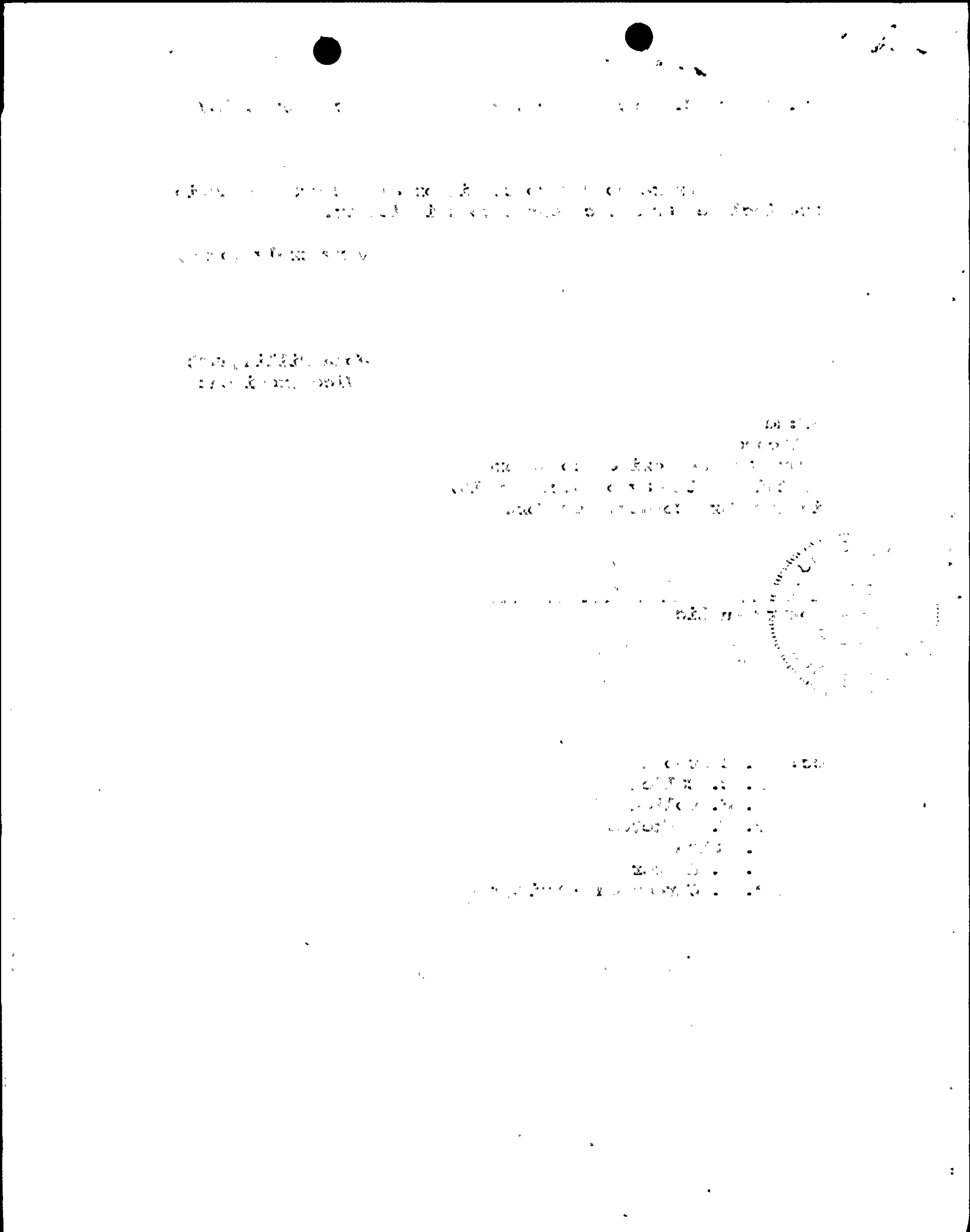


Attachment A

Question A.7

Provide a plan to verify the analytical model used for 193 assembly plants beyond Cycle 1 to assure that an empirical correction is not required as exposure accumulates.

Response to Question A.7

A power map will be taken at least once for every 31 days* of operation for which comparisons between calculations and measurements of power distribution will be made. Comparison will be made between calculated and measured power distribution during Cycle 2 startup which is scheduled in early 1977. The initial results for Cycle 2 will be reported to the NRC 90 days following the startup.

Question A.8

Asymmetries in the "measured" assembly powers up to -7% (fig. 3-5, assemblies G-11 and E-9) are reported. What is the cause of these variations?

Response to Question A.8

The asymmetries cited were caused by measurement errors due to drift in certain detectors. The results reported on map 30 reflect this fact which was particularly associated with detectors A and B. At the time map 30 was taken, (April 4, 1975), AEP personnel became acutely aware of the problem, and remedial steps were taken to correct this condition. The detector's manufacturer was also notified of this problem.

As a result of these efforts, detectors A, B and F were replaced on April 21, 1975. The improvements from this replacement can be seen in the attached map 34 (taken on April 29, 1976), where detector drift was much smaller. Following this, a constant monitoring of detector behavior has been carried out at the Donald C. Cook Nuclear Plant Unit No. 1. The intent of this program is that if any uncorrectable abnormal detector behavior is observed, the anomalous detectors will be replaced.

*Effective full power days.

Attachment B

ADDITIONAL QUESTIONS ASKED BY NRC STAFF

Question 1.

Describe the single failure assumed in the ECCS accident analysis as reported in XN-76-51.

Response to Question 1.

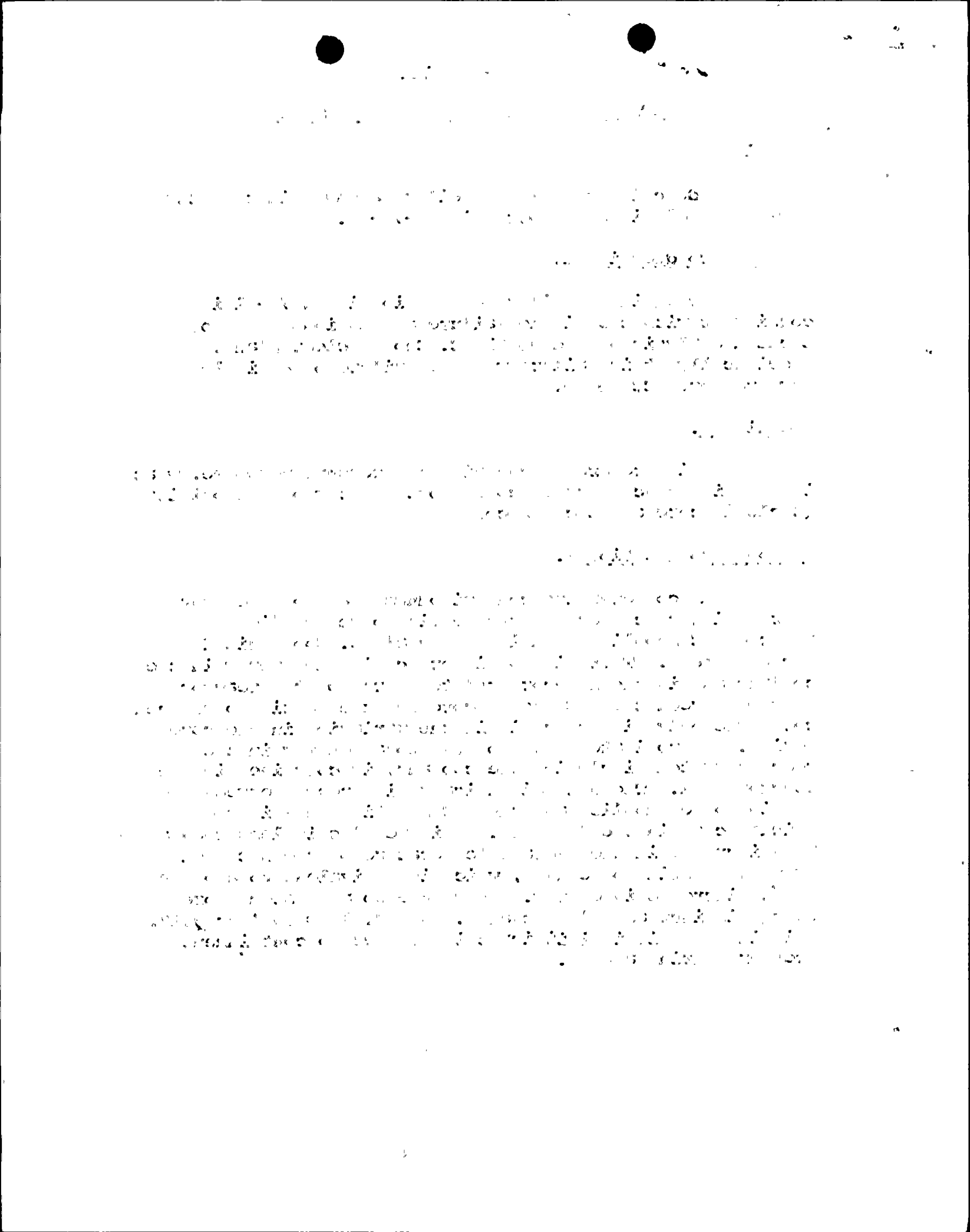
The single failure assumption in XN-76-51 is consistent with the single failure assumptions used for the Cycle 1 analysis of the Donald C. Cook Nuclear Plant. Specifically, this failure was the failure of a single low pressure ECCS pump.

Question 2.

With regard to containment pressure response, what is the difference between core stored energy of the original (Cycle 1) core and Cycle 2 core?

Response to Question 2.

A conservative comparison was made of the stored energy difference between the Westinghouse supplied fuel and the ENC supplied fuel in the Donald C. Cook Unit I Nuclear Plant. There is no difference in the energy in the coolant and in the non-core related parts of the reactor system between the ENC fueled core and the Westinghouse fueled core; the only difference is in the variation in the core design. Approximately 6.5% of the total energy in the reactor system, including the coolant, is contained in the reactor fuel. The only significant difference between the Westinghouse supplied and the ENC supplied fuel is the thicker clad in the ENC fuel. This results in less than a 20°F increase in the average temperature of the ENC fuel over the Westinghouse fuel, which is equivalent to about a 1-1/2% increase in energy. Thus, the total system stored energy is increased less than $0.065 \times 0.015 = .001$ or 0.1%. This increase is insignificant in respect to containment pressure during a LOCA.



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A					0.598	0.677	0.801	0.717	0.801	0.670	0.595				
B			0.468	0.841	0.985	1.031	0.980	1.030	0.970	1.002	0.969	0.837	0.466		
C		0.466	0.971	0.852	1.091	1.150	1.224	1.248	1.193	1.118	1.073	0.849	0.969	0.466	
D		0.836	0.852	1.222	1.013	1.214	1.160	1.267	1.134	1.165	0.980	1.217	0.859	0.852	
E	0.594	0.980	1.087	0.999	1.167	1.100	1.215	1.121	1.170	1.085	1.156	0.994	1.104	0.991	0.598
F	0.689	1.025	1.171	1.196	1.092	1.159	1.020	1.185	1.004	1.134	1.090	1.186	1.158	1.031	0.689
G	0.831	0.997	1.219	1.131	1.184	1.011	1.147	1.090	1.172	1.033	1.237	1.114	1.207	1.001	0.834
H	0.744	1.068	1.216	1.254	1.123	1.164	1.093	1.184	1.099	1.187	1.136	1.224	1.186	1.074	0.748
I	0.831	0.997	1.219	1.158	1.193	1.006	1.155	1.090	1.164	1.018	1.195	1.140	1.194	0.999	0.836
J	0.665	0.995	1.191	1.201	1.093	1.150	1.010	1.155	1.013	1.169	1.103	1.217	1.143	1.036	0.693
K	0.577	0.950	1.096	0.997	1.143	1.088	1.192	1.106	1.184	1.093	1.100	0.955	1.071	1.027	0.624
L		0.829	0.858	1.218	0.989	1.195	1.142	1.254	1.148	1.176	0.955	1.184	0.855	0.677	
M		0.471	0.979	0.857	1.095	1.159	1.192	1.171	1.208	1.161	1.080	0.837	0.985	0.488	
N			0.471	0.845	0.989	1.019	0.979	1.056	1.002	1.041	0.979	0.824	0.468		
O					0.600	0.676	0.821	0.740	0.831	0.696	0.594				

MAP 34

COOK UNIT I, BOL, HZP, ARO, D IN, D+C IN, D+C+B IN, SET 1

AEP - THIMBLE DATA

NUCLEAR PEAKING FACTORS FOR ENTHALPY RISE FOR ASSEMBLAGES IN THE POWER NORMALIZATION