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VASSALLO,D.B. Operating Reactors Branch 2

SUBJECT: Forwards response to 830908 request for addl info re critically analysis,supporting application for amend spent fuel storage capacity expansion.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

October 5, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Vassallo:

Our letter of June 24, 1983 provided information on the criticality analysis supporting our application for amendment to expand our spent fuel pool storage capacity. Your September 8, 1983 letter requested additional information regarding this criticality analysis. Attached is our response to your request. Information verbally requested by members of your staff is also included.

Sincerely,

CVMangan

C. V. Mangan
Vice President
Nuclear Engineering and Licensing

CVM/MTG:djm
Attachment

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RESPONSE TO QUESTIONS ON NINE MILE POINT UNIT 1
SPENT FUEL STORAGE CAPACITY EXPANSION

Question No. 1

It appears that only the eight poison high density racks (1710 spaces) in the south half of the pool have been analyzed for a maximum enrichment of 3.75 weight percent U-235. Therefore, the reference to 18.13 grams of U-235 per axial centimeter in Technical Specification 5.5 should apply only to these 1710 poison high density spaces. Unless reanalyzed for the higher enrichment, the maximum loading in the 1066 flux trap storage spaces should remain at 15.6 grams of U-235 per axial centimeter.

Response

Flux trap design spent fuel storage racks were installed in the north half of the pool in 1978. The criticality analysis for these racks was performed using a U-235 loading of 15.6 grams per axial centimeter. This is equivalent to a fuel assembly with a lattice enrichment of approximately 3 weight percent U-235. The criticality analysis for the poison design spent fuel storage racks to be installed in the south half of the pool (1710 spaces), was performed using a U-235 loading of 18.13 grams per axial centimeter. This is equivalent to a fuel assembly lattice enrichment of approximately 3.75 weight percent U-235. Only bundles with an average lattice enrichment of less than 3 weight percent U-235 will be stored in the north half of the pool. Bundles with an average lattice enrichment of less than 3.75 weight percent will be stored in the south half of the pool. It should be noted that following the next refueling outage, the flux trap design racks in the north half of the pool will be full.

Question No. 2

Since 3.75 weight percent U-235 is higher than any present fuel assembly design, how is the calculational bias which corrects for the explicit multi-enrichment pin distribution derived?

Response

The basic rack cell used for the criticality analysis included a fuel bundle wherein the enrichment of each of the 62 contained fuel rods was 3.75 weight percent U-235. The k_{∞} of this basic rack cell was 0.9105. Since this uniform enrichment bundle could never be used in the reactor, a calculation was also performed for a more realistic bundle with a specific distribution of enrichments, such that the average of those enrichments would also be 3.75 weight percent U-235. The k_{∞} of this latter rack cell was 0.8997. Therefore, the perturbations to the basic rack cell resulting from a typical realistic enrichment distribution was -0.0108 Δk.

The particular enrichment distribution selected to represent a typical bundle was based on a reference bundle design with a maximum average planar enrichment of 3.01 weight percent U-235 (for example fuel bundle -P8DRB282 of NEDO-24195). The enrichment of each fuel rod type was increased by the ratio of 3.75/3.01 to obtain the distribution shown below, which was used in the calculation.

Rod Lattice

7	6	5	5	4	4	4	5
6	4	4	3	2	3	2	4
5	4	3	1	1	1	3	2
5	3	1	1	W	1	1	2
4	2	1	W	1	1	1	2
4	3	1	1	1	1	3	2
4	2	3	1	1	3	1	3
5	4	2	2	2	2	3	4

ROD I.D.	ENRICHMENT w/o U-235
1	4.74
2	4.11
3	3.74
4	2.99
5	2.49
6	2.12
7	1.62
W	Water Rod

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Question No. 3

Provide information on a) K_∞ calculations, b) computer code benchmarking.

Response

a. See attached Table 1, Summary of Perturbations to the Multiplication Factor of the Basic Cell and Figure 1, Basic Rack Cell Geometry and Dimensions for the Minimum Pitch Between Bundles.

b. The criticality safety analysis model consists of a proprietary version of the LEOPARD computer program (for generating four energy group neutron cross sections) and the PDQ-7 computer program (for calculating the neutron multiplication factor of the specific geometry and materials comprising the spent fuel storage racks). The resulting analysis model has been extensively benchmarked against the reliable and applicable critical experiments to determine both the bias and uncertainty in the analysis model.

In order to test the accuracy of the LEOPARD program for generating neutron cross sections, 27 particularly applicable experiments were analyzed using the Pickard, Lowe & Garrick version of the LEOPARD Program together with reported measured bucklings. As shown in Table 2, the average calculated k_{eff} is 0.9979 and the standard deviation is 0.0080 Δk. Since the resulting standard deviation is of the same order as the uncertainty in the measured bucklings, these experiments serve only to qualitatively verify the accuracy of the LEOPARD program for generating neutron cross sections.

To provide a quantitative evaluation of the accuracy of the combined LEOPARD/PDQ-7 model, two applicable and completely independent sets of critical experiments have been analyzed. The results summarized in Table 2 show that for 14 high leakage critical arrays with a large variation in hydrogen to uranium ratios, the average of the calculated k_{eff} was 0.9928 with a standard deviation about this value of 0.0012 Δk. Also, as shown in Table 2 for 12 high leakage critical experiments which included self shielded neutron absorbers, the average of the calculated k_{eff}'s was 0.9931 with a standard deviation of 0.0011 Δk. The agreement of the bias' and standard deviations for these two independent sets of experiments is excellent. Therefore, the bias and standard deviation derived from the 12 Battelle critical experiments (Reference 11) was adopted for the final evaluation of criticality safety.

As a result of the method of analysis and extent of benchmarking, any bias in the model resulting from the use of diffusion theory in PDQ-7 is implicitly included in the bias and standard deviation determined from those benchmarking results. Therefore, the "so called" transport theory correction factor is zero in the application of the criticality safety analysis model used for the analysis of the Nine Mile Point Unit 1 spent fuel storage racks.

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TABLE 1

Summary of Perturbations to the Multiplication Factor of the Basic Cell

Description	k effect	k
Basic Cell at 68°F, 3.75 w/o U-235 in enriched center section of bundle, 0.0217 gm B-10/cm^2 in 0.110 inch thick Boraflex		.9105
Calculational Biases		
Explicit multi-enrichment pin distribution	-.0108	
Mesh spacing effect	-.0003	
Axial Leakage net effect	.0024	
LEOPARD/PDQ Model Bias	+.0069	
Spacer grids	-.0029	
Space between grids	+.0252	
Basic Cell including Biases		.9262
Tolerances and Uncertainties		
Minimum pitch (i.e., box to box)	+.0035	
Tolerance on SS box thickness	+.0014	
Maximum pellet density	+.0012	
Fuel position uncertainty	+.0002	
Calculational uncertainty (2σ)	+.0022	
Total Uncertainty (statistical)	+.0045	
Maximum, including biases & uncertainties		.9307
Design Conservatism	-.0300	
Maximum, including design conservatism		.9007

BASIC RACK CELL GEOMETRY AND DIMENSIONS FOR THE MINIMUM PITCH BETWEEN BUNDLES

(ALL DIMENSIONS IN INCHES)

MATERIALS

1. Homogenized Fuel Pin Cells
2. Explicit Water
3. Explicit Stainless Steel
4. Homogenized Water Rod Cells
5. Explicit Boraflex

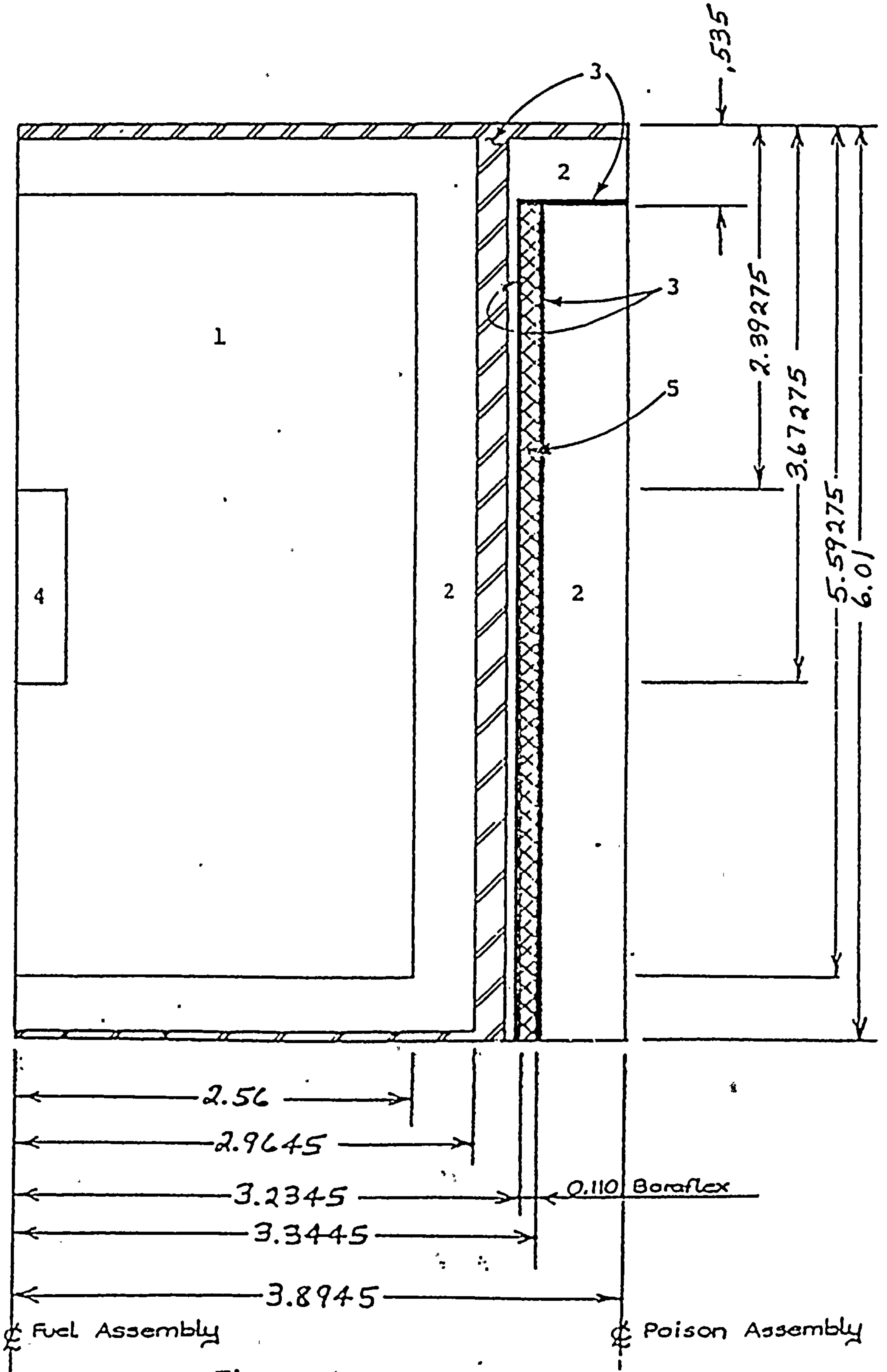

Figure 1

TABLE 2

Benchmarking Summary Results

1. LEOPARD CROSS SECTION VERIFICATION

27 Measured Criticals with No Soluble Boron (References 1-8)

Averaged of Calculated K_{eff}'s = 0.9979 σ = 0.0080 Δk

2. LEOPARD, PDQ-7 CRITICALITY SAFETY MODEL BENCHMARKING

A. 14 Westinghouse UO_2 Zr-4 Clad Cylindrical Core Criticals (References 9,10)

Average of Calculated K_{eff}'s = 0.9928 σ = 0.0012 Δk

B. 12 Battelle UO_2 Zirconium Clad Fixed Neutron Poison Criticals - Including Boral, Stainless Steel and Borated Stainless Steel Absorbers (Reference 11)

Average of Calculated k_{eff}'s = 0.9931 σ = 0.0011 Δk

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