Consumers
Power
Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

January 12, 1979

Director, Nuclear Reactor Regulation
Att Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - PROPOSED TECHNICAL SPECIFICATIONS
CHANGE REQUEST - FUEL STORAGE

The following documents are provided to update our submittal dated December 18, 1978 requesting Technical Specifications changes to allow the storage of higher enriched fuel in both new and spent fuel storage locations.

1. Page 18 of the NUS Report entitled "Criticality Analysis for 3.27 Weight Percent or W/O Enriched Fuel Palisades High Density Fuel Rack."

2. Exxon Report XN-309 - "Palisades New Fuel Storage Array - Criticality Safety Analysis."

3. Example calculations of U^{235} linear density.

4. Corrected Technical Specifications page changes incorporating final Batch H design data. These changes have been discussed with the NRC staff.

David P Hoffman (Signed)

David P Hoffman
Assistant Nuclear Licensing Administrator

CC JGKeppler, USNRC

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ATTACHMENT 1

TABLE 3

SUMMARY OF CRITICALITY ANALYSIS RESULTS

	Original File G-RA-06						This Work		
	Main Pool Rack k_∞			Tilt Pool Rack k_∞			Most Reactive Rack k_∞		
		Worst Tolerances	Accident		Worst Tolerances	Accident		Worst Tolerances	Accident
Nominal, 68°F		0.8720	0.8787		0.8832	0.8899		.8843	See Section 11.4, G-RA-06 Addendum 1
2σ Δk_∞ =	.0086			.0090			.0084		
B_4C Particle Self-Shielding	.0040			.0040			.0040		
KENO Benchmark	.0086			.0086			.0086		
	.0212			.0216			.0210	.9053	
Nominal (Adjusted), 68°F		0.8932	0.8999		0.9048	0.9115			
Variation of Water Temp	.0031	0.8963	0.9030	.0031	0.9079	0.9146	.0028	.9081	
Worst Tolerances									
Enrichment Variation	.0036			.0036			.0037		
Fuel Rod	.0083			.0083			.0006		
B_4C Slab Width	.0006			.0006			.0008		
B_4C Slab Thickness and B_4C Loading	–			–			.0038		
Variation of Spacing	.0187			.0084			.0081		
Can Dimensions	.0101			.0107			.0097		
B_4C Slot Thickness	.0043			.0043			.0045		
B_4C Slab Missing (1/100)	.0018			.0018			.0018		
Bow and Twist	–			–			.0165		
Can Wall Thickness	–			–			.0061		
(rms)	.0236			.0170			.0228		
Worst Possible k_∞		0.9199	0.9266		0.9249	0.9316		.9309	

ATTACHMENT 2

XN-309

PALISADES NEW FUEL STORAGE ARRAY CRITICALITY SAFETY ANALYSIS

JULY, 1975

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

XN-309

PALISADES NEW FUEL STORAGE ARRAY

CRITICALITY SAFETY ANALYSIS

July, 1975

EXXON NUCLEAR COMPANY, INC.

XN-309

PALISADES NEW FUEL STORAGE ARRAY

CRITICALITY SAFETY ANALYSIS

Prepared by: L E Hansen — 7/14/75

L. E. Hansen, Specialist
Criticality Safety — Date

Approved by: W S Nechodom — 7/14/75

W. S. Nechodom, Manager
Licensing and Compliance — Date

W. E. Niemuth — 7/14/75

W. E. Niemuth, Manager
Contract Performance — Date

R. Nilson — 7/14/75

R. Nilson, Manager
Quality Assurance and
Licensing — Date

TABLE OF CONTENTS

INTRODUCTION

American Nuclear Society Standard ANS-N18.2 stipulates that new fuel storage arrays be designed to limit the reactivity of such arrays to a value of $\leq$ 0.98 for all credible conditions of moderation. Further, it is suggested that such conditions of moderation include full flooding or the envelopment of the array in a uniform low density aqueous foam as could credibly exist as a result of fire fighting.

As originally designed, the Palisades new fuel storage array location and design was felt to preclude the addition of moderation. The 3 x 24 array of fuel assemblies, therefore, was not designed to remain subcritical in the event of water flooding or the addition of low density hydrogeneous materials within and between the stored fuel assemblies.

To comply with suggested limits established in ANS-N18.2, the Consumers Power Company has proposed a reduction in capacity of the array from 72 to 36 fuel assemblies located in a checkerboard array with alternate locations occupied by structural steel box beams. This document describes the criticality safety analysis of that proposed storage array and demonstrates compliance with the suggested limits established in ANS-N18.2.

SUMMARY

Criticality safety analyses of the proposed new fuel storage array containing steel box beams in alternate storage locations demonstrate the safety of the array for all credible conditions of moderation. Array

reactivities were computed using the KENO-2 Monte Carlo code assuming fully flooded conditions and the presence of low density aqueous materials uniformly distributed both within and between the stored fuel assemblies. Calculated reactivities are well below the suggested limit established in ANS-N18.2 for new fuel storage arrays. The highest reactivity (k_{eff} = 0.896 $\pm$ .007) occurs when the array is fully flooded by water.

FUEL ASSEMBLY DESCRIPTION

A typical Palisades fuel assembly design (Reload E[1]) is depicted in Figure 1. As indicated, this arrangement includes a single zirconium instrument guide tube located in the center of the assembly and eight zirconium guide bars positioned on the exterior of the assembly. UO_2 fuel rods within the assembly contain uranium of three different ^{235}U enrichments.

Actual "Reload E" fuel assembly specifications and two different assumed conditions evaluated as part of this analysis are given in Table I. These conditions include existing and bundle averaged cell parameters. The bundle averaged cell parameters were calculated by including the zirconium associated with the guide bars and instrument guide tube in the zirconium clad thickness of each rod. Water within the assembly was included by increasing the unit cell dimensions (lattice pitch). Such assumptions permit an estimation of the effect of the extra zirconium and water within the fuel assembly.

It should be noted that the analysis discussed herein assumed a single enrichment of 3.2 wt % ^{235}U for all rods. The nominal "Reload E"

FIGURE 1

RELOAD E ASSEMBLY

℄ WG

℄ NG

℄ WG

L	L	M	M	G	M	M	M	M	M	G	M	M	L	L
L	M	M	M	M	M	M	M	M	M	M	M	M	M	L
M	M	M	H	H	H	H	H	H	H	H	H	M	M	M
M	M	H	H	H	H	H	H	H	H	H	H	H	M	M
G	M	H	H	H	H	H	H	H	H	H	H	H	M	G
M	M	H	H	H	H	H	H	H	H	H	H	H	M	M
M	M	H	H	H	H	H	H	H	H	H	H	H	M	M
M	M	H	H	H	H	H	I	H	H	H	H	H	M	M
M	M	H	H	H	H	H	H	H	H	H	H	H	M	M
M	M	H	H	H	H	H	H	H	H	H	H	H	M	M
G	M	H	H	H	H	H	H	H	H	H	H	H	M	G
M	M	H	H	H	H	H	H	H	H	H	H	H	M	M
M	M	M	H	H	H	H	H	H	H	H	H	M	M	M
L	M	M	M	M	M	M	M	M	M	M	M	M	M	L
L	L	M	M	G	M	M	M	M	M	G	M	M	L	L

℄ NG

L = 2.40 w/o U-235 Fuel

M = 2.81 w/o U-235 Fuel

H = 3.28 w/o U-235 Fuel

G Guide Bar

I Instrument Tube

℄ Identifiers: WG Wide Interassembly Water Gap

NG = Narrow Interassembly Water Gap

TABLE I

PALISADES FUEL ASSEMBLY

PARAMETERS

	Actual[1] (Nominal)	Lattice Cell Parameters	Bundle Averaged Cell Parameters
Lattice Pitch	0.5500"	0.5500"	0.5613"
Clad OD	0.4150"	0.4150"	0.4241"
Clad Thickness	0.0285"	0.0285"	0.0331"
UO_2 Pellet Diameter	0.3505"	0.3505"	0.3505"
Pellet Density (%TD)	94±1.5	96%	96%
Percent Dish	1.0	0%	0%
Avg. Enrichment (wt % U-235)	3.04	3.20*	3.20*
Rod Array	15x15	15x15	15x15

* Design base enrichment specified by Consumers Power Company[2].

bundle averaged enrichment, however, is 3.04 wt % ^{235}U. Consequently, for the water to fuel volume ratio of these assemblies, this analysis is valid for bundle averaged enrichments of ∿ 3.2 wt % ^{235}U.

STORAGE ARRAY DESCRIPTION

The Palisades new fuel storage rack has been measured to determine actual "as built" dimensions. Figure 2 is an arrangement drawing giving those measured dimensions. This information was supplied by Consumers Power Company(2). Subsequent information(3) established that the dimensions indicated are actually measured center-to-center distances between adjacent top bands. Such dimensions, therefore, represent nominal center-to-center spacings between adjacent storage locations. It should be noted that the nominal center-to-center separation between assemblies was designed to be 9.5 inches. Measurements of the installed rack, however, show that a maximum negative tolerance of 1/8" exists on the design value (i.e., a minimum nominal center-to-center separation of 9-3/8" was measured). It should also be noted that the plates connecting adjacent angle irons are 5" wide and 3/16" thick. These plates establish a minimum edge-to-edge separation between adjacent storage locations (i.e., 3/16").

Concrete walls are adjacent to three sides of the storage array and are separated from the fuel by 0.5 to 1.5 inches. For the purpose of this analysis, a 16" thick concrete reflector were assumed to be touching three edges of the storage rack. The fourth side of the array was assumed to be reflected by 4" of water, which is effectively an infinitely thick water reflector.

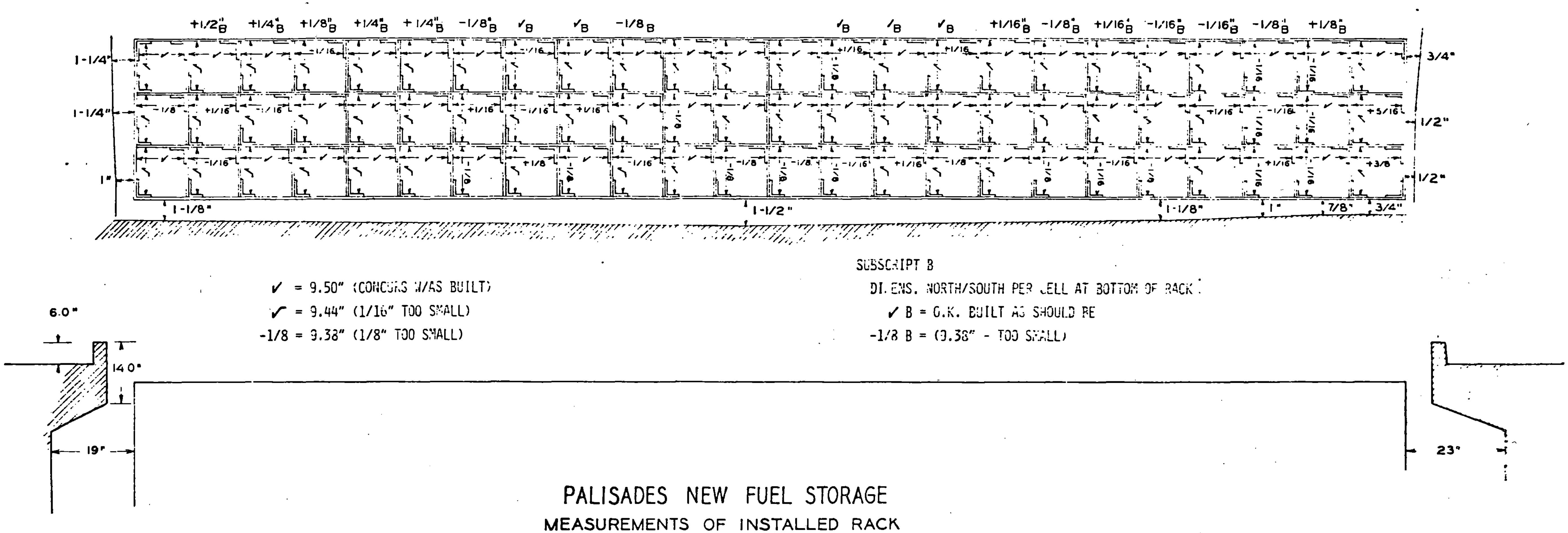

+1/2"B +1/4"B +1/8"B +1/4"B +1/4"B -1/8"B ✓B ✓B -1/8B ✓B ✓B ✓B +1/16"B -1/8"B +1/16"B -1/16"B -1/16"B -1/8"B +1/8"B
1-1/4"
1-1/4"
1"
3/4"
1/2"
1/2"
1-1/8"
1-1/2"
1-1/8"
1"
7/8"
3/4"
✓ = 9.50" (CONCURS WAS BUILT)
✓ = 9.44" (1/16" TOO SMALL)
-1/8 = 9.38" (1/8" TOO SMALL)
SUBSCRIPT B
DIMENS. NORTH/SOUTH PER CELL AT BOTTOM OF RACK.
✓ B = O.K. BUILT AS SHOULD BE
-1/8 B = (9.38" - TOO SMALL)
6.0"
14.0"
19"
23"
PALISADES NEW FUEL STORAGE
MEASUREMENTS OF INSTALLED RACK

FIGURE 2

The proposed arrangement for storing Palisades fuel assemblies is indicated in Figure 3. Calculations for this arrangement result in the assumption of a 3 x 25 array of storage locations. This "checkerboard" array contains fuel bundles with alternate positions occupied by 8"x8" structural steel box beams having a nominal wall thickness of 5/16"[2]. A minimum wall thickness of 0.25" was established for this analysis[4].

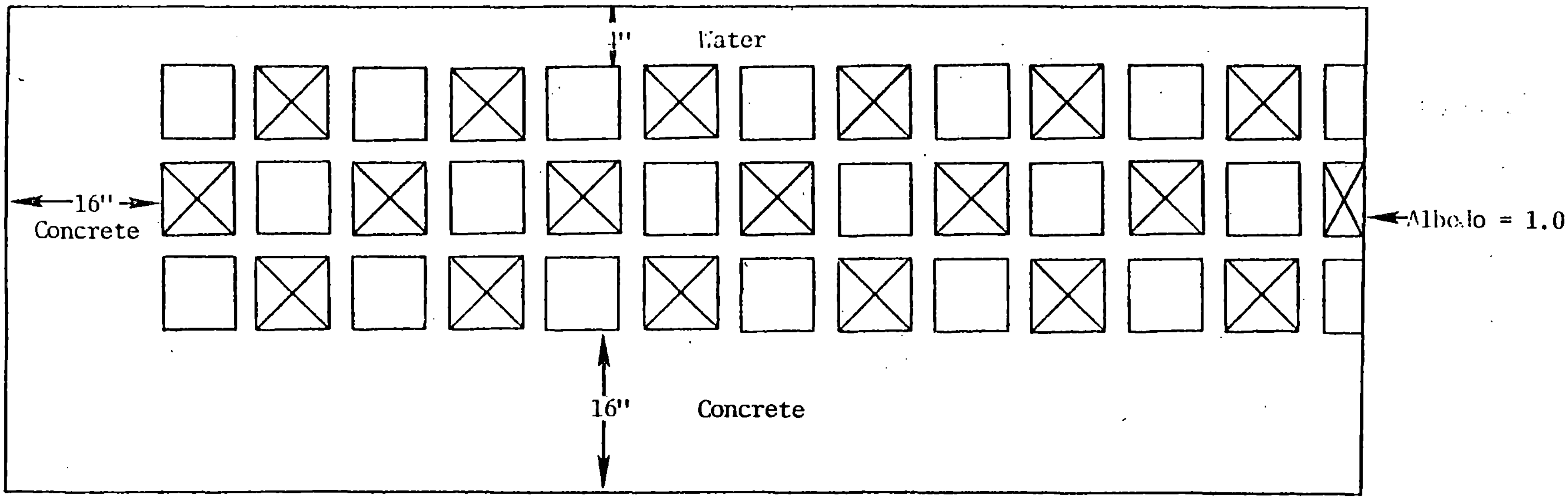
In addition to the nominally spaced array of fuel assemblies shown in Figure 3, it is quite possible that the fuel assemblies will not be centered in each storage location. As previously noted, however, each storage location is bounded by a 3/16" plate. Hence, a minimum edge-to-edge separation between adjacent storage locations is assured. A section of the array considered to evaluate the effect of abnormal arrangements of fuel assemblies is indicated in Figure 4.

CALCULATIONAL METHODS

The KENO-II Monte Carlo code[5] was utilized to calculate the reactivity of the Palisades new fuel storage array. Multigroup cross section data (18 energy groups) utilized in these calculations were averaged using the CCELL[6], BRT-1[7], and GAMTEC-II[8] codes. Specifically, the cross section data for various regions within the storage array were averaged as follows:

- CCELL - Utilized to obtain cell averaged multigroup cross section data for fuel rod-water lattices. Such calculations included both the bundle averaged cell parameters and the actual lattice cell parameters (See Table I).

PALISADES NEW FUEL STORAGE ARRAY (NOMINAL SPACING)



Fuel Bundle Location (8.25"x8.25")

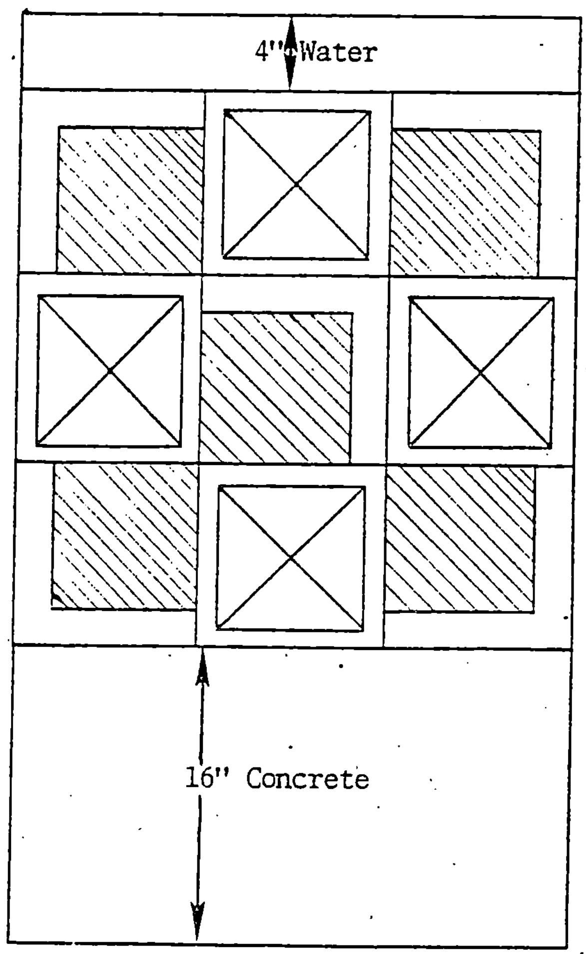
Steel Box Beam (8"x8" Outside Dimension)

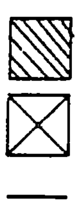
Center-to-Center Spacing of Units = 9-3/8"

FIGURE 3

FIGURE 4

ABNORMAL ARRAY ARRANGEMENT





Minumum Center-to-Center Distance Between Adjacent Aluminum Straps = 9-3/8"

- BRT-1 - Thermal group ($\leq$ 0.683 ev) cross section data for the structural steel box beams were averaged using the Battelle Revised THERMOS code. Such data were averaged assuming a 0.25" thick region of iron separated from the rod-water lattice by the thickness of interspersed water (at varying densities) that would exist in the nominally spaced storage array (0.4375"). Epithermal multigroup data were averaged over a slowing down neutron energy distribution in water.

- GAMTEC-II - Multigroup cross section data for water and concrete were averaged over neutron energy spectra characteristic of infinite media of these respective materials. The concrete was assumed to have constituents as specified by the Consumers Power Company[9]. (Due to the fact that cross section data for sulfur were not readily available, and since the atom density of sulfur represents only 0.06% of the total constituents, the effects of sulfur in the concrete were ignored.)

In addition to the codes identified above, the XMC Monte Carlo code was utilized to verify the accuracy of CCELL calculations for undermoderated rod-water lattices. The XMC code is a puesdo-point energy Monte Carlo code ($\sim$ 90 energy groups) which permits the discrete representation of the entire fuel assembly.

RESULTS

Values of k∞ were computed using the CCELL code for 3.2 wt % ^{235}U rods as a function of water density within the unit cell. Both lattice cell and bundle averaged cell parameters (See Table I) were assumed to gain an insight into the reactivity effects of the zirconium guide bars and the water associated with guide bars and the instrument guide tube. The results of those calculations are given in Table II.

Comparison of the calculated values of k∞ indicates that the uncertainty associated with the zirconium and water extraneous to the actual lattice cells is quite small. Indeed a maximum effect of 5 mk was calculated assuming these materials were uniformly distributed throughout the bundle.

The CCELL code has been used extensively at Exxon Nuclear for averaging multigroup constants in rod-water lattice configurations. Theory-experiment correlations[10] indicate that the reactivity of measured critical UO_2 rod water lattices can be calculated to within 18 mk. Biases associated with such calculations appear to be conservative in nature.

No experimental data are known to exist at water densities consistent with those assumed in this analysis. As a consequence, in an effort to determine the validity of the CCELL code for water densities of < 0.6 g/cm^3, the XMC Monte Carlo code was used to calculate k∞ of the "Reload E" bundle with all rods assumed to contain uranium enriched to 3.2 wt % ^{235}U. Water interspersed between the rods was assumed to have a density

TABLE II

Water Density (g/cm^3)	Infinite Media Multiplication Factors	
	Lattice Cell Parameters	Bundle Averaged Cell Parameters
1.00	1.401	1.402
0.75	1.369	1.371
0.50	1.299	1.302
0.30	1.187	1.190
0.20	1.090	1.093
0.15	1.023	1.023
0.10	0.936	0.934
0.05	0.816	0.811

of 0.2 g/cm^3. The value of k∞ calculated for this case was 1.079 ± .005. This value compares favorably to that calculated by the CCELL code using bundle averaged cell parameters (1.093) or lattice cell parameters (1.090). These calculations indicate that the CCELL code calculates, with reasonable accuracy, the spatial and energy distribution of neutrons within the lattice cells at water densities far below the values where experimental data have been utilized to confirm the validity of this calculational method. It appears, therefore, that the calculational biases will result in the calculation of array reactivities which are conservative.

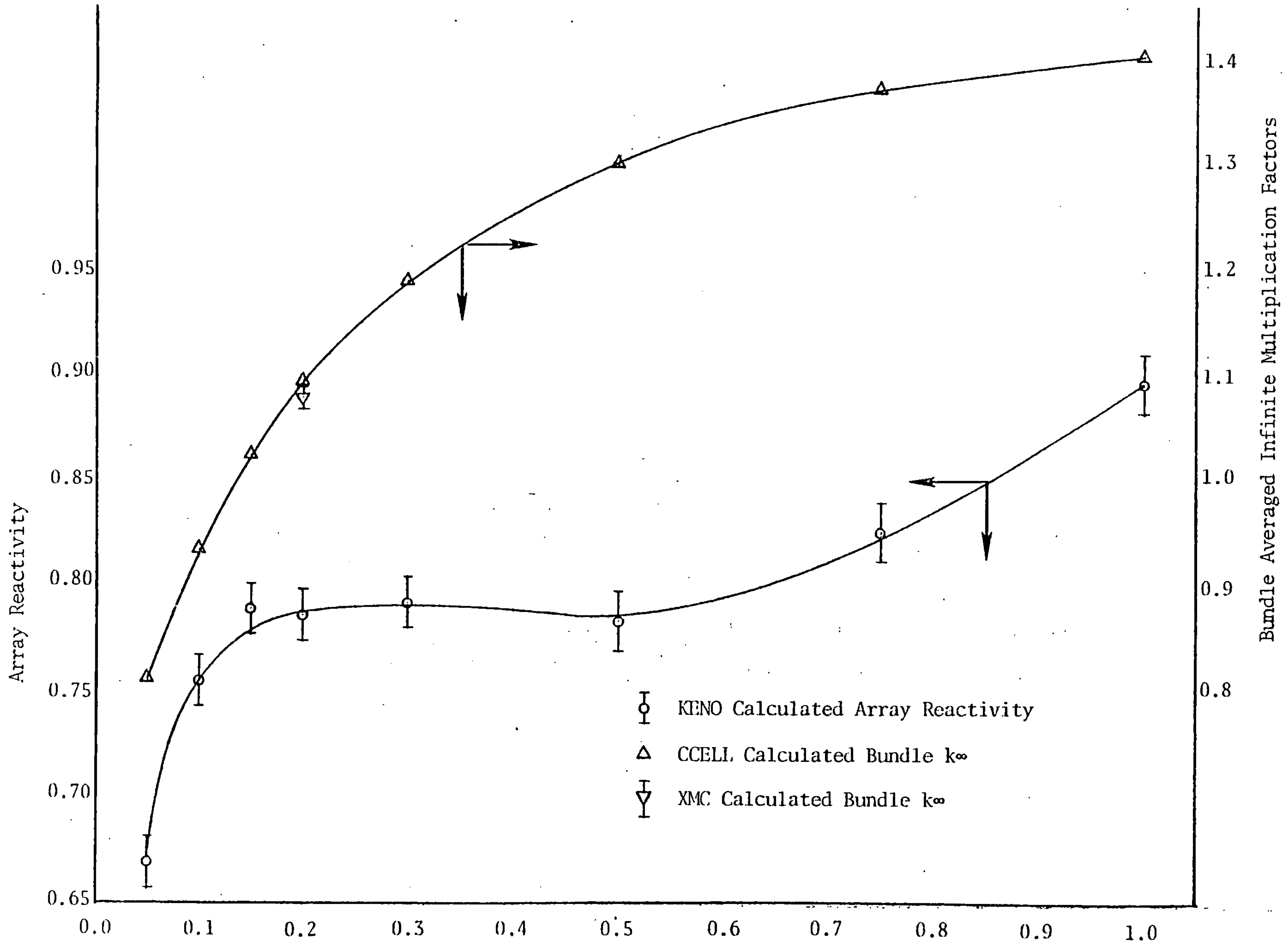
For the nominally spaced "checkerboard" array shown in Figure 3, reactivities were computed as a function of water density with such water uniformly distributed both within and between the fuel assemblies. The results of those calculations are tabulated in Table III and shown graphically in Figure 5. These reactivities were calculated assuming bundle averaged cell parameters. Figure 5 also includes bundle-averaged values of k∞ calculated using the CCELL code and, for comparison, indicates the single check calculation performed using the XMC Monte Carlo code.

The highest array reactivity occurs in the fully flooded condition. For all water densities, however, the array reactivity is well below the limiting value of 0.98 established in ANS-N18.2. The highest calculated reactivity (cited at the 95% confidence level) is 0.910. All available checks of the validity of this calculated value indicate a probable conservative bias of 1 to 2% k.

TABLE III

Fractional Water Density	KENO-II Calculated Array Reactivity
1.00	0.896 ± .007
0.75	0.825 ± .007
0.50	0.783 ± .007
0.30	0.792 ± .006
0.20	0.786 ± .006
0.15	0.789 ± .006
0.10	0.755 ± .006
0.05	0.669 ± .006

PALISADES NEW FUEL STORAGE ARRAY CALCULATED REACTIVITIES



Fractional Water Density

FIGURE 5

In addition to the nominally spaced array, the array of fuel assemblies spaced as indicated in Figure 4 was also evaluated. For this abnormal arrangement, which assumes the minimum possible spacing between bundles, water densities of 1.0 and 0.2 g/cm^3 were assumed and the resultant calculated reactivities for such conditions were 0.914 $\pm$ .007 and 0.812 $\pm$ .005, respectively. Hence, even for abnormal conditions, the reactivity of the array remains well below established limits.

CONCLUSIONS

This analysis conservatively demonstrates that the proposed Palisades new fuel storage array will remain subcritical under all conditions of moderation as could credibly exist through the uniform addition of water at any density both within and between the stored fuel assemblies. It is important to note, however, that the array might not be critically safe if moderators are added to individual stored fuel assemblies without the addition of such materials between the assemblies. It is believed valid to assume that this is not a credible moderator arrangement which could result from any accidental addition of water.

The analytical efforts described herein were reviewed by an independent second party knowledgeable in the performance of criticality safety evaluations. This independent assessment of the adequacy of this analysis is discussed in Appendix A.

REFERENCES

1) F. D. Lang, G. R. Correll, and K. P. Galbraith, "Final Design Report for Palisades Fuel," XN-74-32, Exxon Nuclear Company, Inc., October, 1974.

2) Letter, W. J. Beckius to W. E. Niemuth, May 13, 1975.

3) W. J. Beckius, Personal Communication, Consumers Power Company, June 10, 1975.

4) B. Webb, Personal Communication, Consumers Power Company, June 10, 1975.

5) G. E. Whitesides and N. F. Cross, "Keno - A Multigroup Monte Carlo Criticality Program," CTC-5, Union Carbide Corporation Nuclear Division, September, 1969.

6) W. W. Porath, "CCELL Users Guide," BNW/JN-86, Pacific Northwest Laboratories, February, 1972.

7) C. L. Bennett and W. L. Purcell, "BRT-1: Battelle Revised THERMOS," BNWL-1434, Pacific Northwest Laboratories, June, 1970.

8) L. L. Carter, C. R. Richey, and L. E. Hushey, "GAMTEC-II: A Code for Generating Consistent Multigroup Constants Utilized in Diffusion and Transport Theory Calculations," BNWL-35, Pacific Northwest Laboratories, March, 1965.

9) Letter, W. J. Beckius to W. E. Niemuth, June 3, 1975.

10) U. P. Jenquin and D. R. Oden, "Verification of Neutronic Design Methods Using Lattice Criticals as Benchmarks," BNW/JN-118, Pacific Northwest Laboratories, December, 1972.

APPENDIX A

Battelle
Pacific Northwest Laboratories

Date July 11, 1975

To L. E. Hansen

From D. R. Oden *D.R. Oden*

Subject Q.A. Review of Criticality Safety Calculations For The Palisades Plant New Fuel Storage Rack

We have completed the Q.A. review of the criticality safety calculations for the Palisades Plant New Fuel Storage Rack, and have found no significant errors or omissions. We concur with your conclusion that k_{eff} of the loaded rack, under the conditions specified, is less than 0.98 at a 95% confidence level for all degrees of neutron moderation.

Information provided by Exxon Nuclear and reviewed by BNW is summarized in Attachments I and II. This consists of letters of correspondence, a copy of a drawing of the storage rack, and several computer printouts of Exxon Nuclear calculations. All of this information is being returned to you at this time.

Items checked in the Q.A. review included the following:

- Code input parameters including nuclei densities for all materials
- Computer code options selected
- Assumptions made in problem set up and analysis versus problem definition in correspondence
- Lattice and array geometry in CCELL and KENO
- Method of homogenization to obtain bundle average parameters
- Region/material correspondence in CCELL and KENO
- Results of analysis
- Adequacy of computer codes used.

Concerning this last item of code validation, there is a degree of uncertainty. The CCELL code has gained acceptance as a reliable method of generating cross sections for UO_2-water lattice through its

use at Exxon Nuclear for criticality safety and fuel design work. The KENO code is used throughout the industry for the calculation of arrays of fissile material in criticality safety applications. However, as you are aware, the lack of experimental data on lattices with low density water moderation makes it impossible to completely validate methods of calculation in this area. With no evidence to the contrary, one must thus take the position that if a model is validated for the full water density UO_2 lattice criticals it will adequately handle the low water density cases, which may not be tenable.

Although it is not a validation of the array k_{eff} calculation, the fact that the XMC calculated bundle average k_∞ agrees well with that from CCELL for the 20% water density case, lends a degree of credibility to the CCELL cross section generation technique. In the absence of experimental data this is about all that can be done short of setting up the whole array calculation in XMC. The cost of this approach must however be weighed relative to the benefit resulting from having yet another (although more sophisticated) calculated result.

ATTACHMENT I

REFERENCE LIST OF CORRESPONDENCE AND DRAWINGS PROVIDED BY EXXON NUCLEAR FOR PALISADES STORAGE RACK Q.A. REVIEW

1. Letter, W. J. Beckius to Wayne Niemuth, 5/13/75.
2. Letter, G. R. Correll to L. E. Hansen, "Criticality Analysis of Palisades New Fuel Storage Racks", 5/19/75.
3. Letter, Wayne Niemuth to W. J. Beckius, 5/27/75.
4. Letter, W. J. Beckius to W. E. Niemuth, 6/3/75.
5. Figure 2.4.2, Reload E Assembly.
6. Drawing, Palisades New Fuel Storage - Measurements of Installed Rack.
7. Sketch, Palisades New Fuel Storage Array (Nominal Spacing)
8. Sketch, Abnormal Array Arrangement.
9. Tables, Two Pages of Exxon Calculated Results.
10. Figure, Array Reactivity and Bundle Averaged k_∞ Versus Fractional Water Density

ATTACHMENT II

LIST OF COMPUTER PRINTOUTS PROVIDED BY EXXON NUCLEAR
FOR PALISADES STORAGE RACK Q.A. REVIEW

Reference Number	Code	Date	Title
1	GAMTEC	6/9/75	Palisades Concrete (No Sulfur)
2	THERMOS	6/24/75	Fe thermal data 0% void 25% void 50% void 70% void 85% void 90% void 95% void
3	THERMOS	6/23/75	Fe thermal data 80% void
4 (5 parts)	XMC	6/13/75	Bundle k_∞, 80% void
5	CCELL	6/9/75	Lattice Avg. Cell 0% void
6	CCELL	6/9/75	Bundle Avg. Cell 0% void
7	CCELL	6/12/75	Bundle Avg. Cell 70% void 50% void 25% void
8	CCELL	6/9/75	Bundle Avg. Cell 80% void 85% void 90% void 95% void
9	CCELL	6/9/75	Lattice Avg. Cell 80% void 85% void 90% void 95% void
10	KENO	9/26/75	Worst Spacing, 80% void
11	KENO	9/25/75	Worst Spacing, 0% void
12	KENO	9/25/75	0% Void Het. Fe
13	KENO	9/25/75	25% void Het. Fe
14	KENO	9/25/75	50% void Het. Fe
15	KENO	9/25/75	70% void Het. Fe
16	KENO	9/25/75	80% void Het. Fe
17	KENO	9/25/75	85% void Het. Fe
18	KENO	9/25/75	90% void Het. Fe
19	KENO	9/25/75	95% void Het. Fe

ATTACHMENT 3

EXAMPLE

CALCULATION OF U^{235} LINEAR DENSITY
Based on January, 1979, Exxon Data

Active Fuel Length	131.8 ± 25 in
Weight of UO_2 per rod	2117.48 ± 42.35 grams
Average enrichment (Batch H)	3.27 ± .05
High enrichment	3.43 ± .05

I. Using Average Enrichment (3.27 ± .05 = 3.32%)

Weight of Uranium only ($U^{235} + U^{238}$)

$$= (2117.48 + 42.35)g \frac{(.0332)(235)+(.9668)(238)}{(.0332)(235)+(.9668)(238)+(2)(16)}$$
$$= (2159.83)g(.8814)$$
$$= 1903.67 \text{ g}$$

Weight of U^{235} only

$$= (1903.67)(.0332)g$$
$$= 63.2g$$

Linear density (g/cm)

$$= \frac{(63.2g \text{ of } U^{235}/rod)(208 \text{ rods})}{(131.8 - .25 \text{ in})(2.54/cm/in)}$$
$$= 39.34 \text{ g/cm } U^{235} \text{ per Bundle}$$

II. Using High Enrichment (3.43 + .05 = 3.48)

Weight of Uranium only ($U^{235} + U^{238}$)

$$= (2117.48 + 42.35)g \frac{(.0348)(235)+(.9652)(238)}{(.0348)(235)+(.9652)(238)+(2)(16)}$$
$$= (2159.83)g(.8814)$$
$$= 1903.67g$$

Weight of U^{235} only

$$= (1903.67g)(.0348)$$
$$= 66.25g$$

Linear Density g/cm

$$= \frac{(66.25 \frac{gU^{235}}{rod})(208 \text{ rods})}{(131.8-.25 \text{ in})(2.54 \text{ cm/in})}$$
$$= 41.24 \text{ g/cm } U^{235} \text{ per bundle}$$

Note: Values for active fuel length and weight of UO_2 per rod was obtained by phone from Exxon Nuclear on 1/8/79. These are to be the final design parameters.

CALCULATION OF U^{235} LINEAR DENSITY
Based on October, 1978, Exxon Data

Active Fuel Length	131.8 ± .25 in
Weight of UO_2 per rod	2101.2 gms
Average Enrichment	3.27 ± .05
High Enrichment	3.43 ± .05

I. For Average Enrichment

Weight of Uranium only ($U^{235} + U^{238}$)

$$= (2101.2\text{g}) \frac{(.0332)(235)+(.9668)(238)}{(.0332)(235)+(.9668)(238)+(2)(16)}$$

$$= (2101.2\text{g})\ (.8814)$$

$$= 1852.00\text{g}$$

Weight of Uranium - 235 only

$$= (1851.99\text{g})(.0332)$$

$$= 61.49\text{g}$$

Linear Density (g/cm)

$$= \frac{(61.49\text{ g } U^{235}/\text{rod})(208\text{ rods})}{(131.8 - .25\text{ in})(2.54\text{ cm/in})}$$

$$= 38.28\text{ g/cm } U^{235} \text{ per bundle}$$

II. For High Enrichment

Weight of Uranium only ($U^{235} + U^{238}$)

$$= (2101.2\text{g}) \frac{(.0348)(235)+(.9652)(238)}{(.0348)(235)+(9652)(238)+(2)(16)}$$

$$= (2101.2\text{g})\ (.8814)$$

$$= 1852.00\text{ g}$$

Weight of U^{235} only

$$= (1852.0\text{g})\ (.0348)$$

$$= 64.45\text{g}$$

Linear Density g/cm

$$= \frac{(64.45\text{g-}U^{235}/\text{rod})(208\text{ rods})}{(131.8-.25\text{ in})(2.54\text{ cm/in})}$$

$$= 40.12\text{ g/cm } U^{235} \text{ bundle}$$

CALCULATION OF U^{235} LINEAR DENSITY
Based on Batch G Fuel Data

Active Fuel Length (in)	$131.8 \pm .25$
Weight of UO_2 (cms)	$2104.0 \pm 2\%$
Average Enrichment (Batch G)	$3.00 \pm .05$
High Enrichment	$3.20 \pm .05$

I. Weight of Uranium Only ($U^{235} + U^{238}$)

$$= (1.02)(2104.0)\,\frac{(.0325)(235)+(.9675)(238)}{(.0325)(235)+(.9675)(258)+(2)(16)}$$

$$= (2146.08)(.8814)$$

$$= 1891.56\text{g}$$

Weight of U^{235} only

$$= (1891.56\text{g})(.0325)$$

$$= 61.48\text{g}$$

Linear Density (g/cm)

$$= \frac{(61.48\text{g } U^{235}/\text{rod})(208 \text{ rods})}{(131.8 - .25 \text{ in})(2.54 \text{ cm/in})}$$

$$= 38.27 \text{ g/cm } U^{235} \text{ per bundle}$$

Our present Technical Specifications (for Batch G Fuel) assume the highest enrichment and limit us to a maximum linear density of 38.3 g/cm U^{235}.

ATTACHMENT 4

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) (Contd)

5.3.2 Reactor Core and Control

a. The reactor core shall approximate a right circular cylinder with an equivalent diameter of about 136 inches and an active height of about 132 inches.

b. The reactor core shall consist of approximately 43,000 Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall be grouped into 204 assemblies.

A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution.

c. The fully loaded core shall contain approximately 211,000 pounds UO_2 and approximately 56,000 pounds of Zircaloy-4.

Poison may be placed in the fuel bundles for long-term reactivity control.

d. The core excess reactivity shall be controlled by a combination of boric acid chemical shim, cruciform control rods, and mechanically fixed boron rods where required. Forty-five control rods shall be distributed throughout the core as shown in Figure 3-5 of the FSAR. Four of these control rods may consist of part-length absorbers.

5.3.3 Emergency Core Cooling System

An emergency core cooling system shall be installed consisting of various subsystems each with internal redundancy. These subsystems shall include four safety injection tanks, three high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in Section 6 of the FSAR.

5.4 FUEL STORAGE

5.4.1 New Fuel Storage

a. Unirradiated fuel may be stored in the new fuel storage rack which is designed to ensure an effective multiplication factor of less than 0.98 under the worst credible conditions for fuel enriched to 3.30 weight percent U-235.

5.4 FUEL STORAGE (Contd)

b. New fuel may be stored in shipping containers.

c. New fuel enriched to 3.27 weight percent U-235 may be stored in the poisoned high capacity racks which are designed to ensure an effective multiplication factor of less than 0.95 when flooded with unborated water.

d. The new fuel storage racks are designed as a Class I structure.

5.4.2 Spent Fuel Storage

a. Irradiated fuel bundles will be stored, prior to off-site shipment in the stainless steel-lined spent fuel pool.

b. The spent fuel racks are designed to maintain fuel in a geometry which insures an effective multiplication factor of 0.95 or less with new fuel flooded with unborated water.

c. The spent fuel pool water boron concentration shall be verified at least once monthly to be equal to or greater than 1720 ppm.

d. The spent fuel racks are designed as a Class I structure.

e. The fuel placed in the spent fuel pool and stored in the poisoned high capacity storage racks shall not contain more than 41.24 grams of U-235 per axial centimeter of active fuel assembly subject to a maximum assembly average loading of 3.27 weight percent U-235. The fuel placed in the spent fuel pool and stored in the unpoisoned lower capacity racks shall not contain more than 38.3 grams of U-235 per axial centimeter of active fuel assembly, subject to a maximum assembly average loading of 3.05 weight percent U-235.

f. Spent fuel shipping casks shall not be moved in the fuel storage building until such time as the NRC has reviewed and approved the spent fuel cask drop evaluation.

g. Fuel stored in the higher capacity storage racks as described in the SER supporting Amendment No. 28, shall have decayed for a minimum of 12 months if the storage racks are not supported by similarly designed, adjacent racks and the spent fuel pool wall or the cask anti-tipping device.[1]

References

[1] Until needed for fuel storage, two A-type racks in the northeast corner of the spent fuel pool will be removed and replaced with the cask anti-tipping device to provide necessary seismic restraint.

FSAR, Appendix A.

FSAR, Appendix B.