for

Rochester Gas & Electric Corporation

- Ginna Plant -

CRITICALITY ANALYSIS FOR THE SPENT FUEL STORAGE RACKS

by

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CRITICALITY ANALYSIS FOR THE SPENT FUEL STORAGE RACKS

1.0 INTRODUCTION

The following discussion summarizes the evaluation of the spent fuel racks with respect to criticality safety. The analytical techniques described here are the same as those used to license the existing spent fuel racks for Ginna.

2.0 ANALYTICAL TECHNIQUE

The LEOPARD[1] computer program was used to generate macroscopic cross sections for input to four energy group diffusion theory calculations which are performed with the PDQ-7[2] program. LEOPARD calculates the neutron energy spectrum over the entire energy range from thermal up to 10 Mev and determines averaged cross sections over appropriate energy groups. The fundamental methods used in the LEOPARD program are those used in the MUFT[3] and SOFOCATE[4] programs which were developed under the Naval Reactor Program and thus are well founded and extensively tested techniques. In addition, Westinghouse Electric Corporation, the developers of the original LEOPARD program, demonstrated the accuracy of these methods by extensive analysis of measured critical assemblies consisting of slightly enriched UO_2 fuel rods.[5]

In addition, Pickard, Lowe and Garrick, Inc. (PLG) has made a number of improvements to the LEOPARD program to increase its accuracy for the calculation of reactivities in systems which contain significant amounts of plutonium mixed with UO_2. PLG has tested the accuracy of these modifi-

modifications by analyzing a series of UO_2 and PuO_2-UO_2 critical experiments. These benchmarking analyses not only demonstrate the improvements obtained for the analysis of PuO_2-UO_2 systems but also demonstrate that these modifications have not adversely affected the accuracy of the PLG-modified LEOPARD program for calculations of slightly enriched UO_2 systems.

The UO_2 critical experiments chosen for benchmarking include variations in H_2O/UO_2 volume ratios, U-235 enrichments, pellet diameters and cladding materials. Although the LEOPARD model also accurately calculates the reactivity effects of soluble boron, these experiments have not been included in the LEOPARD benchmarking criticals since the spent fuel pool calculations do not involve soluble boron.

Neutron leakage was represented by using measured buckling input to infinite lattice LEOPARD calculations to represent the critical assembly. A summary of the results is shown in Table 1 for the 27 measured criticals chosen as being directly applicable for benchmarking the LEOPARD model for generating group average cross sections for spent fuel rack criticality calculations. The average calculated k_{eff} is 0.9979 and the standard deviation from this average is 0.0080 Δk. Reference 5 raised questions concerning the accuracy of the measured buckling reported for the experiments number 12 through 19. If these data are excluded, the average calculated k_{eff} for the remaining 19 experiments is 1.0006 with a standard deviation from this value of 0.0063 Δk. In all of these experiments, there are significant uncertainties in the measured bucklings which are necessary inputs to the LEOPARD analysis. These uncertainties are the same order of magnitude as the indicated errors in

the LEOPARD results, and therefore a more definitive set of experimental data is used to establish the accuracy of the combined LEOPARD/PDQ-7 model used for the criticality analysis of the spent fuel racks.

The PDQ series of programs have been extensively developed and tested over a period of 20 years, and the current version, PDQ-7, is an accurate and reliable model for calculating the subcritical margin of the proposed spent fuel rack arrangement. This code or a mathematically equivalent method is used by all the U.S. suppliers of light water reactor cores and reload fuel. In addition, this code has received extensive utilization in the U.S. Naval Reactor Program.

As a specific demonstration of the accuracy of the calculational model used for the spent fuel rack calculations, the combined LEOPARD/PDQ-7 model has been used to calculate fourteen measured just critical assemblies. The criticals are high neutron leakage systems with a large variation in H_2O/U_2O volume ratio and include parameters in the same range as those applicable to the spent fuel rack design. Experiments including soluble boron are included in this demonstration since the ability of PDQ-7 to calculate neutron leakage effects is of primary interest. The use of soluble boron allows changes in the neutron leakage of the assembly while maintaining a uniform lattice and thus allows a better test of the accuracy of the model. Furthermore, it eliminates the error associated with the measured bucklings, which is inherent in the LEOPARD benchmarks, thus permitting determinations of the actual calculational uncertainty which much be accounted for in the spent fuel rack criticality analysis.

These combination LEOPARD/PDQ-7 calculations result in a calculated average k_{eff} of 0.9928 with a standard deviation about this value of 0.0012 Δk. These results, as shown in Table 2 demonstrate that the proposed LEOPARD/PDQ-7 calculational model can calculate the reactivity of the proposed spent fuel rack arrangements with an accuracy of better than 0.010 Δk at the 95 percent confidence level.

In addition to the fourteen critical assemblies in Table 2, the LEOPARD/PDQ model was used to calculate the k_{eff} for seven additional critical assemblies, two of which incorporated thin stainless steel plates in an array which is geometrically similar to a section of the spent fuel racks.

These seven criticals were performed by Battelle Pacific Northwest Laboratories specifically for the purpose of providing benchmark critical experiments in support of spent fuel criticality analysis. They are described in detail in Reference 17. The results of these critical experiments are summarized in Table 3. The overall average k_{eff} calculated for these seven just critical assemblies was 0.9933, with a standard deviation around this value of 0.0013 Δk.

Combining the results of benchmarking against both the Westinghouse (Table 2) and the Battelle (Table 3) critical experiments results in a mean calculated k_{eff} for 21 experiments of 0.9929 with a standard deviation of .0012 as shown in Table 3. Thus, the final bias to be applied to the combined LEOPARD/PDQ-7 model is +.0071, and the 2σ uncertainty to be applied corresponding to a 95 percent probability at a 95 percent confidence level is .0024.

As a result of this approach to separately benchmark both the cross sections and the diffusion theory calculations against applicable critical assemblies, the "transport theory correction factor" is implicitly included in the derived calculational uncertainty factor.

3.0 EVALUATION OF CRITICALITY SAFETY

The PDQ-7 program is used in the final predictions of the multiplication factor of the spent fuel storage pool. The calculations are performed in four energy groups and take into account all the significant geometric details of the fuel bundles, fuel boxes and major structural components. The geometry used for most of the calculations is a basic cell representing one quarter of the area of a repeating array of two identical stainless steel boxes. The specific geometry and dimensions of this basic cell are shown in Figure 1, and the fuel assembly characteristics are listed in Table 4.

The calculational approach is to use the basic cell to calculate the reactivity of an infinite array of uniform spent fuel racks and to account for any deviations of the actual spent fuel rack array from this assumed infinite array as perturbations on the calculated reactivity of the basic cell. The effects of mechanical tolerances are also treated as perturbations on the calculated reactivity of the basic cell. The fuel assemblies were assumed to be unirradiated with a U-235 enrichment of 4.25 w/o in the enriched center section which is higher than any anticipated reload enrichment for the Ginna core. This corresponds to a U-235 loading of 41.9 gm U-235 per axial cm of fuel assembly. Most of the calculations were performed at a uniform pool temperature of 68°F, but the reactivity effects of pool temperature are also taken into account as a perturbation on the basic cell calculations.

The reference basic cell calculation is performed with the minimum dimension on all the stainless steel boxes which results in a $k_\infty = 0.9305$. Other tolerances on the geometric array representing the racks are treated as perturbations on this reference basic cell calculation.

The calculated variation of the basic cell k_∞ as a function of the initial enrichment of the fuel assembly is shown in Figure 2.

Most of the calculations with the basic cell geometry utilized a 40x20 two-dimensional array of mesh points. To test the adequacy of this mesh description, a calculation was run with a 80x40 mesh size and the resulting k_∞ was .9296. Thus, the perturbation on the basic cell due to mesh spacing effects is $-.0009\ \Delta k$.

Based on the results of the benchmarking of the combined LEOPARD/PDQ-7 analysis model, the bias in the calculated multiplication factor compared to measured just critical arrays is .0071, and this bias must be added to the calculated basic cell reactivity.

The k_∞ of the basic cell as a function of temperature is shown in Figure 3. With a maximum pool temperature of 200°F, under the worst possible conditions the k_∞ is .9383, which results in a perturbation due to temperature effects of $+0.0078\ \Delta k$. Although the overall steady state reactivity temperature coefficient of the spent fuel pool is positive, the temperature coefficient of the fuel assemblies is negative.

The sensitivity of the spent fuel storage rack multiplication factor to the simultaneous and uniform variation of water density in both the fuel box and water box is illustrated in Figure 4. Simultaneous variation of water density in both boxes is conservative and unrealistic since the water density in the fuel box will always be less than or equal to the water density in the water box. Furthermore, it has been established that due to the large flow area provided by this rack design it is not possible to trap steam or air in the water boxes when the fuel boxes are filled with water. As shown in Figure 4, there is no increase in reactivity until the relative density drops below about 70% of full water density and since there is no mechanism by which the water density within the pool could be reduced to a value anywhere near this, the effect of water density variation on spent fuel rack reactivity is considered to be zero.

The basic cell was also used to evaluate the reactivity effect of axial neutron leakage. A one-dimensional axial model, using flux weighted cross sections from two-dimensional radial planar cell calculations, was used to simulate the center enriched section and the short natural uranium sections located at the top and bottom of the enriched section. The resulting reactivity perturbation due to axial neutron leakage is -.0026 Δk.

A summary of the perturbations and biases to the basic cell reactivity calculation is shown in Table 5. Thus, the calculated reactivity of the spent fuel pool with 595 unirradiated bundles with 4.25 w/o U-235 is .9419 for a pool temperature of 200°F.

4.0 TOLERANCES AND UNCERTAINTIES

There are a number of tolerances and uncertainties which result in perturbations which must be considered in the criticality analysis. The reactivity effect of all such positive perturbations is then combined statistically in accordance with Reference 18 to determine a single reactivity perturbation which is added to the calculated basic cell multiplication factor (including biases) to determine the final conservative evaluation of the spent fuel rack maximum possible multiplication factor.

Based on the results of the calculational model benchmarking described in Section 2, the 2σ uncertainty in the model, which corresponds to a 95/95 confidence statement, is .0024.

The stainless steel fuel and water boxes are nominally .090 inches thick with a tolerance of ±.004 inches. Assuming a worst case in which all boxes were at the minimum thickness of .086 inches, the k_∞ of the basic cell is .9333. Therefore, the maximum perturbation on the reactivity of the basic cell due to variations in the stainless steel box thickness is +.0028 Δk.

With the fuel bundles located in their most reactive positions inside the stainless steel boxes, the k_∞ of the basic cell is .9327. Thus, the perturbation on the basic cell reactivity due to positioning uncertainties is +.0022 Δk.

The nominal density of the pellets contained in the fuel assemblies is 95% of theoretical density. Increasing this to the maximum possible theoretical density of UO_2 pellets of 96% results in a positive reactivity perturbation of .0012 Δk.

As shown in Table 5, the total reactivity perturbation to be added to the biased basic cell reactivity to account for tolerances and uncertainties is .0044 Δk. This results in a final conservatively calculated spent fuel rack multiplication factor of .9463.

Realistically, the spent fuel pool coolant contains a minimum concentration of 2000 ppm boron at all times. When the reactivity effect of this minimum boron concentration is included, the actual spent fuel pool multiplication factor including all biases, tolerances and uncertainties is .6622 as shown in Table 5.

5.0 ACCIDENT ANALYSIS

The fuel racks are designed to prevent any criticality threat caused by a dropped fuel bundle which penetrates and occupies a position other than a normal fuel storage location. The only positive effect of such a bundle on the reactivity of the rack would be by virtue of reduction in axial neutron leakage from the rack. Since the calculations reported here show the total axial neutron leakage effect to be only .0026 Δk, a dropped fuel bundle lying on top of the rack would not have any significant effect on the reported maximum possible reactivity of the spent fuel storage rack.

The lattice of the fuel assemblies results in an under-moderated configuration so that any crushing or compaction of the fuel assemblies would tend to reduce the multiplication factor of the spent fuel pool. Therefore, the dropping of

heavy objects into the fuel pool or deformations from the effects of earthquakes or tornadoes could not produce a criticality accident.

Figure 5 depicts a hypothetical accident condition which involves an extra bare fuel assembly inadvertently placed in a side water channel and next to the fuel rack fully loaded with unirradiated Westinghouse 14 x 14 optimized fuel assemblies. When such an assembly is located in a non-fuel storage location, it is considered to be an abnormal condition and appropriate credit is taken for the soluble boron that is present in the pool water. PDQ-07 calculations show that 2000 ppm of soluble boron is worth 0.268 Δk which is more than enough to compensate for the positive reactivity effect caused by the extra fuel assembly. The neutron multiplication factor of the rack system in this accident condition is 0.725 which is substantially less than the basic cell k_∞ of 0.9305 (see Table 5) at 68°, 4.25 w/o U-235, and no soluble boron.

Because of the well founded, conservative technique used for determination of the infinite multiplication factor, there is more than reasonable assurance that this spent fuel rack design will not cause undue risk to the public health and safety resulting from criticality considerations.

6.0 ANALYSIS CONSERVATISMS

As discussed previously, the basic cell calculations make the conservative assumption that the fuel pool water contains no boric acid when in fact the minimum boron content in the pool water is 2000 ppm. As shown in Table 5, a realistic evaluation of the maximum multiplication factor of the spent fuel racks, even when completely filled with unirradiated fuel, is .6622.

REFERENCES

1. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," SCAP-3269, September 1963.

2. W. R. Caldwell, "PDQ-7 Reference Manual," WAPD-TM-678, January 1967.

3. H. Bohl, E. Gelbard and G. Ryan, "MUFT-4 -- Fast Neutron Spectrum Code for the IBM-740, "WADP-TM-72, July 1957.

4. H. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged Over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code," WAPD-TM-39, January 1957.

5. L. E. Strawbridge and R. F. Barry, "Criticality Calculations for Uniform Water-Moderated Lattices," Nuclear Science and Engineering, 23, 59, 1965.

6. "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period January 1 - March 31, 1965," WCAP-3269-12.

7. "List of Equipment and Apparatus at WREC," Westinghouse Reactor Evaluation Center, February 1967.

8. W. L. Orr, H. I. Sternbert, P. Deramaix, R. H. Chastain, L. Binder and A. J. Impink, "Saxton Plutonium Program, Nuclear Design of the Saxton Partial Plutonium Core," WCAP-3385-51, December 1965. (Also EURAEC-1490.)

9. R. D. Leamer, W. L. Orr, R. L. Stover, E. G. Taylor, J. P. Tobin and A. Bukmir, "PuO_2-UO_2 Fueled Critical Experiments," WCAP-3726-1, July 1967.

10. A. F. Henry, "A Theoretical Method for Determining the Worth of Control Rods," WAPD-218, August 1959.

11. P. W. Davison, et al., "Yankee Critical Experiments Measurements on Lattices of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water," YAEC-94, Westinghouse Atomic Power Division (1959).

12. V. E. Grob and P. W. Davison, et al., "Multi-Region Reactor Lattice Studies - Results of Critical Experiments in Loose Lattices of UO_2 Rods in H_2O," WCAP-1412, Westinghouse Atomic Power Division (1960).

REFERENCES (cont:)

13. W. J. Eich and W. P. Kovacik, "Reactivity and Neutron Flux Studies in Multiregion Loaded Core," WCAP-1433, Westinghouse Atomic Power Division (1961).

14. W. J. Eich, Personal Communication (1963).

15. T. C. Engelder, et al., Measurement and Analysis of Uniform Lattices of Slightly Enriched HO_2 Moderated by D_2O-H_2 Mixtures," BAW-1273, the Babcock & Wilcox Company (1963).

16. A. L. MacKinney and R. M. Ball, "Reactivity Measurements on Unperturbed, Slightly Enriched Uranium Dioxide Lattices," BAW-1199, the Babcock & Wilcox Company (1960).

17. Battelle Pacific Northwest Laboratories, "Critical Separation Between Subcritical Clusters of 2.35 Wt% 235-U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," PNL-2438.

18. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," U.S. NRC, April 14, 1978.

TABLE 1

SUMMARY OF LEOPARD RESULTS FOR MEASURED CRITICALS

Case** Number	Reference Number	Enrichment (atom %)	H_2O/U Volume	Fuel Density (g/cm^3)	Pellet Diameter (cm)	Clad Diameter (cm)	Clad Thickness (cm)	Lattice Pitch (cm)	Critical Buckling m^{-2}	Calculated k_{eff}
1	11	2.734	2.18	10.18	0.7620	0.8594	0.04085	1.0287	40.75	1.0015
2	11	2.734	2.93	10.18	0.7620	0.8594	0.04085	1.1049	53.23	1.0052
3	11	2.734	3.80	10.18	0.7620	0.8594	0.04085	1.1938	63.28	1.0043
4	12	2.734	7.02	10.18	0.7620	0.8594	0.04085	1.4554	65.64	1.0098
5	12	2.734	8.49	10.18	0.7620	0.8594	0.04085	1.5621	60.07	1.0118
6	12	2.734	10.13	10.18	0.7620	0.8594	0.04085	1.6891	52.92	1.0072
7	13	2.734	2.50	10.18	0.7620	0.8594	0.04085	1.0617	47.5	1.0008
8	13	2.734	4.51	10.18	0.7620	0.8594	0.04085	1.2522	68.8	0.9987
9	13	3.745	2.50	10.37	0.7544	0.8600	0.0406	1.0617	68.3	1.0010
10	13	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.1	1.0025
11	14	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.68	1.0009
12	15	4.099	2.55	9.46	1.1278	1.2090	0.0406	1.5113	88.0	0.9889
13	15	4.099	2.14	9.46	1.1278	1.2090	0.0406	1.450	79.0	0.9830
14	16	4.099	2.59	9.45	1.1268	1.2701	0.07163	1.555	69.25	0.9999
15	16	4.069	3.53	9.45	1.1268	1.2701	0.07163	1.684	85.52	0.9958
16	16	4.069	8.02	9.45	1.1268	1.2701	0.07163	2.198	92.84	1.0040
17	16	4.069	9.90	9.45	1.1268	1.2701	0.07163	2.381	91.79	0.9872
18	16	3.037	2.64	9.28	1.1268	1.2701	0.07163	1.555	50.75	0.9946
19	16	3.037	8.10	9.28	1.1268	1.2701	0.07163	2.198	68.81	0.9809
20	8	0.714*	1.68	9.52	0.8570	0.9931	0.0592	1.3208	108.8	0.9912
21	8	0.714*	2.17	9.52	0.8570	0.9931	0.0592	1.4224	121.5	1.0029
22	8	0.714*	4.70	9.52	0.8570	0.9931	0.0592	1.8669	159.6	0.9944
23	6	0.714*	10.76	9.52	0.8570	0.9931	0.0592	2.6416	128.4	1.0008
24	9	0.729*	1.11	9.35	1.2827	1.4427	0.0800	1.7526	89.1	0.9902
25	9	0.729*	3.49	9.35	1.2827	1.4427	0.0800	2.4785	104.72	1.0055
26	9	0.729*	3.49	9.35	1.2827	1.4427	0.0800	2.4785	79.5	0.9948
27	9	0.729*	1.54	9.35	1.2827	1.4427	0.0800	1.9050	90.0	0.9878

*These are PuO_2 in Natural UO_2.

** Cases 1 through 19 are with stainless steel clad, Cases 20 through 27 are zircaloy.

TABLE 2

WESTINGHOUSE UO_2 Zr-4 CLAD CYLINDRICAL CORE CRITICAL EXPERIMENTS

EXPERIMENT	PITCH (IN.)	BORON CONCENTRATION (ppm)	MATERIAL BUCKING FOR LEOPARD CM-2	CRITICAL NO. OF PINS	RADIUS OF FUEL REGION (cm)	k_{eff} (LEOPARD/PDQ-)
1	0.600	0	.008793	489.4	19.021	0.9912
2	0.690	0	.009725	317.0	17.605	0.9941
3	0.848	0	.008637	251.6	19.276	0.9927
4	0.976	0	.006458	293.0	23.935	0.9935
5	0.600	306.	.007177	659.9	22.088	0.9927
6	0.600	536.4	.006244	807.2	24.429	0.9937
7	0.600	727.7	.005572	950.2	26.504	0.9940
8	0.600	104.	.008165	546.3	20.097	0.9919
9	0.600	218.	.007599	607.1	21.186	0.9917
10	0.600	330.	.007106	669.5	22.248	0.9916
11	0.600	446.	.006661	735.3	23.315	0.9909
12	0.600	657.1	.005809	895.3	25.727	0.9944
13	0.848	104.	.007320	321.0	21.772	0.9938
14	0.848	218.	.006073	420.5	24.919	0.9925
						0.9928 Mean 0.0012 Std.

Fuel Region Data

Enrichment = 2.719 w/o U-235
Fuel Density = 10.41 g/cm^3
Pellet Radius = 0.20 in
Clad IR = 0.2027 in
Clad OR = 0.23415 in

(b) Thickness of water reflector is that required to attain total radius of 50 cm for model.

(c) $B_z^{2(PDQ-7)} = .000527\ cm^{-2}$

TABLE 3

BATTELLE CRITICALS[17]

Case	Length Times Width	No. of Assemblies In Array	Absorber Type	Absorber Thickness	Distance To Fuel Cluster	Critical Separation of Clusters	k_{eff} LEOPARD/PDQ
028	20x16	3	S.S.	.485 cm	.645 cm	6.88 cm	0.9922
027	20x16	3	S.S.	.302	.645	7.43	0.9919
002B	20x18.075	1	None	-	-	-	0.9956
015	20x17	3	"	-	-	11.94 cm	0.9932
013	20x16	3	"	-	-	8.42	0.9921
022	20x15	3	"	-	-	6.39	0.9933
021	20x16	3	"	-	-	4.46	0.9946

Statistical Summary:

Experiments	Number	Mean k_{eff}	σ
Battelle	7	.9933	.0013
Battelle and Westinghouse (Table 2)	21	.9929	.0012

Fuel region data: Enrichment = 2.35 w/o, Pellet radius = 0.5588 cm, Clad OR = .635 cm, Wall thickness = .0762 cm, Pitch = 2.032 cm

TABLE 4

FUEL ASSEMBLY CHARACTERISTICS

Number of rods containing UO_2	179
Rod pitch (in)	0.5560
Overall envelope dimensions (in)	7.763
Weight of U (KgU)	350.5
Active fuel length (in)	141.4
Enriched uranium region	
Length (in)	128.98
Enrichment (w/o)	4.25
Natural uranium blanket region	
Lenght (in)	12.42
Enrichment (w/o)	0.711
Instrument tube	
Material	Zr-4
O.D. (in)	0.4015
I.D. (in)	0.3499
Guide tubes	
Material	Zr-4
O.D. (in), above dashpot	0.5280
O.D. (in), in dashpot	0.4825
I.D. (in), above dashpot	0.4900
I.D. (in), in dashpot	0.4425
Fuel pellet	
Material	UO_2
Density (% theoretical)	$95^{+1.0}_{-1.5}$
O.D. (in)	0.3444 ± 0.0019
Cladding	
O.D. (in)	.400
I.D. (in)	.3514
Spacer Grids	
Number	9
Location	(see outline drawing)
Weights of materials*	
Inconel grids (2), lbs total	3.00
Zircaloy grids (7), lbs total	19.46

*Does not include weight of the stainless steel sleeves or inserts.

TABLE 5

SUMMARY OF PERTURBATIONS TO THE MULTIPLICATION FACTOR OF THE BASIC CELL

Description	Δk effect	k
Basic cell at 68°F, 4.25 w/o U-235 in the enriched center section of the assembly		.9305
Calculational Biases		
Increase in temperature from 68°F to 200°F	+.0078	
LEOPARD/PDQ Model Bias	+.0071	
Mesh spacing effect	-.0009	
Net axial leakage effect	-.0026	
Basic Cell including biases		.9419
Tolerances and Uncertainties		
Tolerance on SS box thickness	+.0028	
Fuel position uncertainty	+.0022	
Maximum fuel pellet density	+.0012	
Calculational uncertainty	+.0024	
Total Uncertainty (statistical)	+.0044	
Maximum k including Biases and Uncertainties		.9463
Basic Cell @ 68°F, 4.25 w/o with 2000 ppm Boron		.6622

FIGURE 1

NOTE: Boundary Condition at the Top of this Figure is 180° Rotational Symmetry

RGE RACK REFERENCE DESIGN

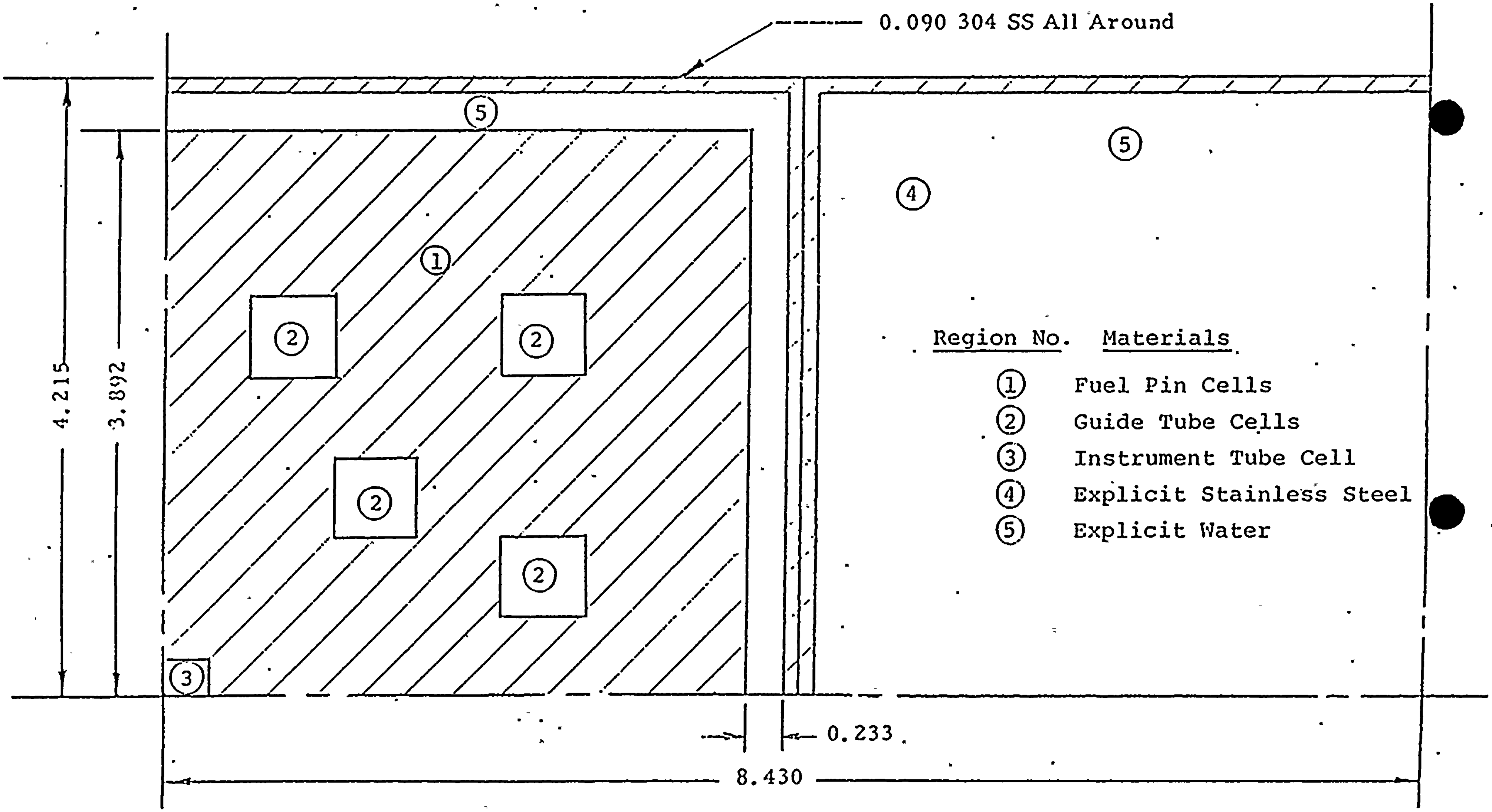


Figure 2

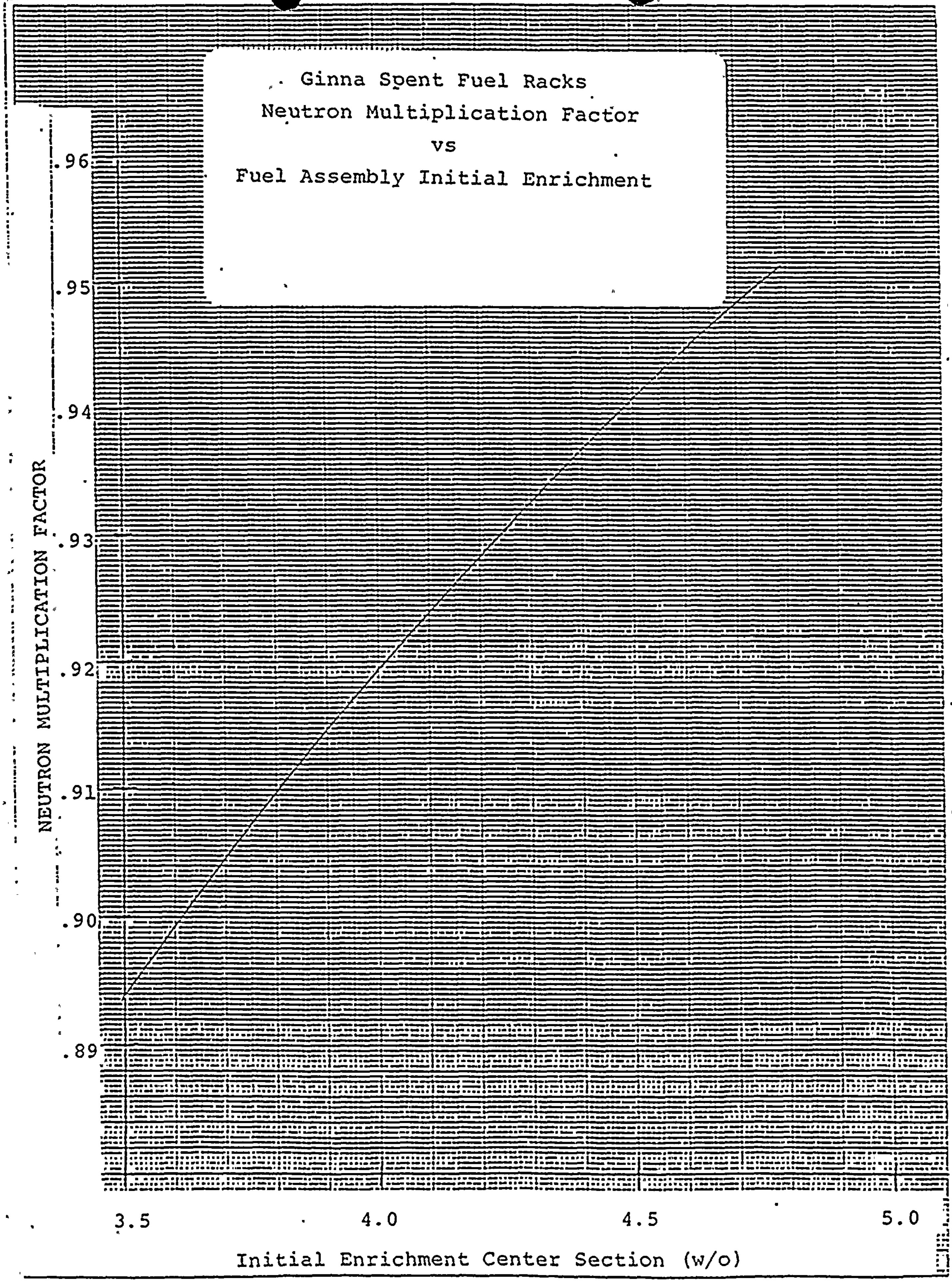

Ginna Spent Fuel Racks
Neutron Multiplication Factor
vs
Fuel Assembly Initial Enrichment
NEUTRON MULTIPLICATION FACTOR
.96
.95
.94
.93
.92
.91
.90
.89
3.5
4.0
4.5
5.0
Initial Enrichment Center Section (w/o)

Figure 3

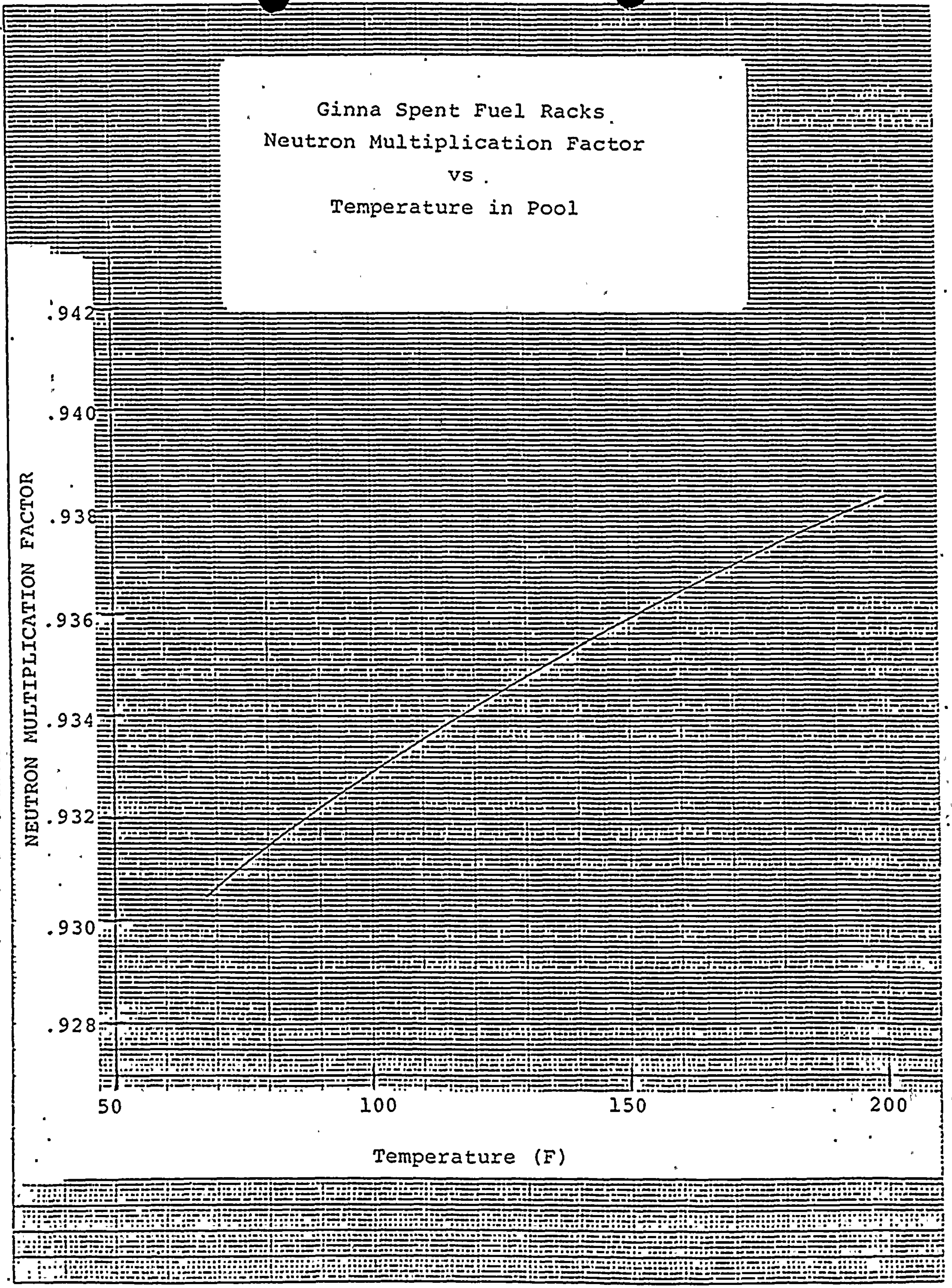
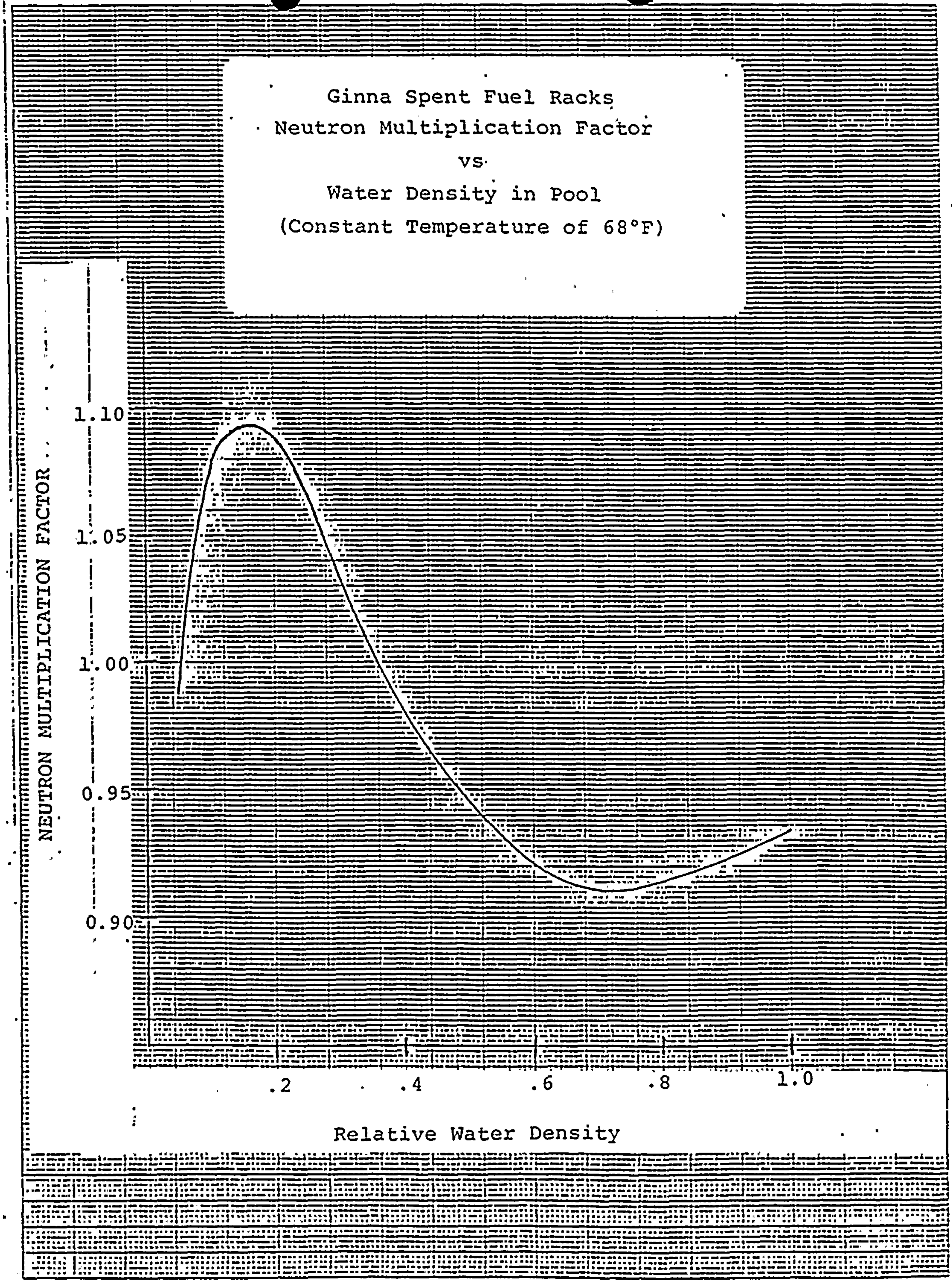
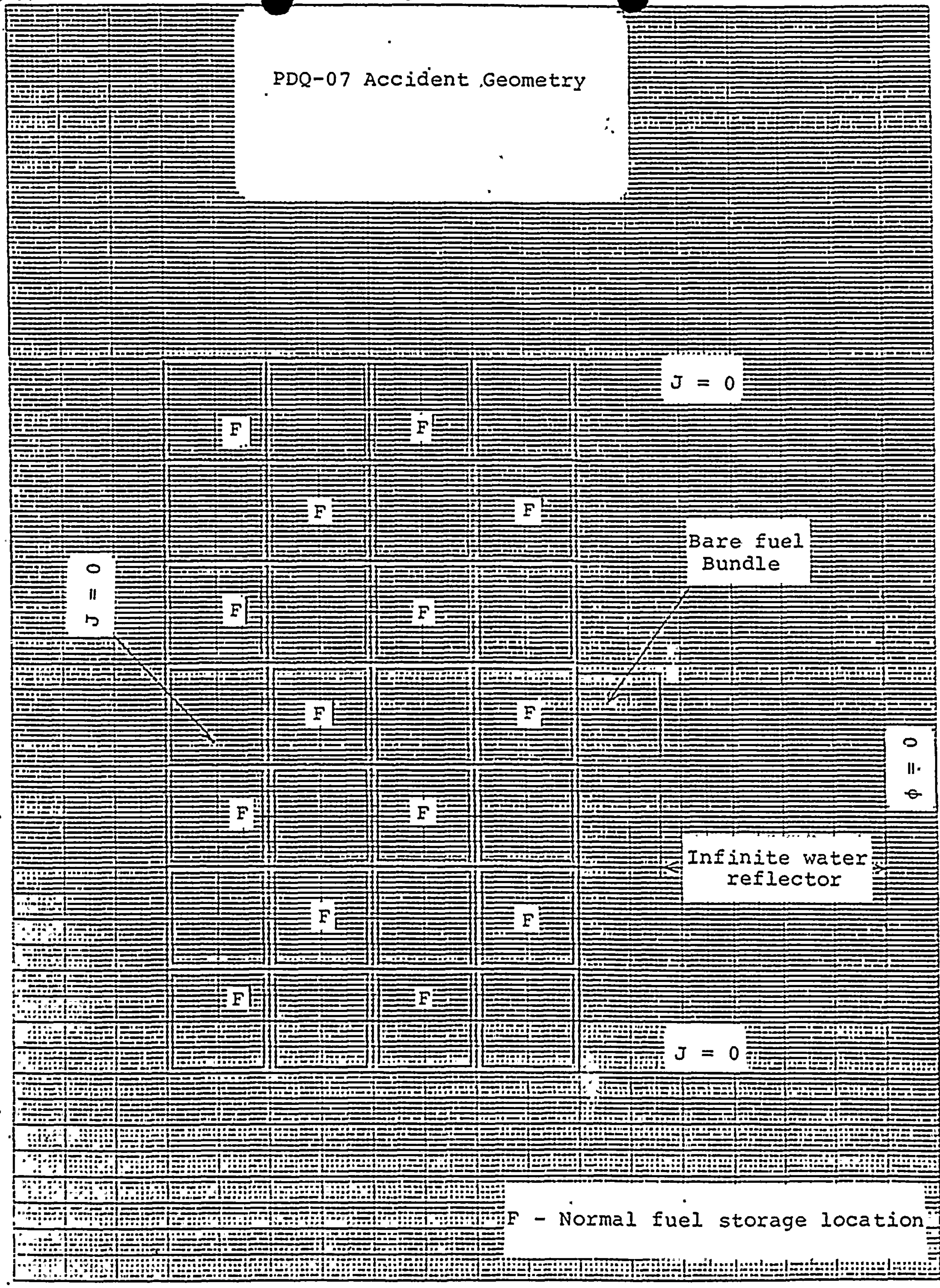

Ginna Spent Fuel Racks
Neutron Multiplication Factor
vs
Temperature in Pool
NEUTRON MULTIPLICATION FACTOR
.942
.940
.938
.936
.934
.932
.930
.928
50
100
150
200
Temperature (F)

Figure 4



Ginna Spent Fuel Racks
Neutron Multiplication Factor
vs
Water Density in Pool
(Constant Temperature of 68°F)
NEUTRON MULTIPLICATION FACTOR
1.10
1.05
1.00
0.95
0.90
.2
.4
.6
.8
1.0
Relative Water Density

Figure 5


PDQ-07 Accident Geometry
J = 0
F
F
F
F
Bare fuel
Bundle
J = 0
F
F
F
F
φ = 0
F
F
Infinite water
reflector
F
F
F
F
J = 0
F - Normal fuel storage location