ADDENDUM TO THE CRITICALITY ANALYSIS FOR THE GINNA NUCLEAR PLANT FUEL STORAGE RACKS TO ADDRESS THE STORAGE OF MIXED OXIDE FUEL ASSEMBLIES

By

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The following discussion summarizes the criticality safety of the Ginna fuel storage racks when loaded with mixed oxide fuel. The analytical techniques described here are identical to those previously used to license the Ginna spent fuel racks.

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 modifications by analyzing a series of UO_2 and PuO_2-UO_2 critical experiments. These bench marking 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 benchmarking criticals since the fuel storage rack calculations do not take credit for 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 LEOPARD results is shown in Table 1 for the 27 measured criticals chosen as being directly applicable for benchmarking the model for spent fuel pool 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 reamining 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 analysis of the fuel storage racks.

The PDQ series of programs have been extensively developed and tested over a period of 20 years, and there is no question that the current version, PDQ-7, is an accurate and reliable model for calculating the subcritical margin of the proposed fuel storage rack arrangements. 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 fuel storage 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 U/H_2O volume ratio and include parameters in the same range as those applicable to the proposed fuel pool design.[6,7] 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 must 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 fuel storage rack arrangements with an accuracy of better than 0.010 Δk at the 95 percent confidence level.

The above methods may also be used to calculate the subcritical margin of the spent fuel storage rack designs when mixed oxide fuel is used. Table 3 shows a comparison of LEOPARD results with a set of five Saxton PuO_2-UO_2 critical experiments. This set of critical experiments is described in detail in Reference 8. The average k_{eff} calculated for these just critical assemblies was 0.9995, with a standard deviation around this value of 0.0068 Δk. A similar comparison is provided in Table 4 for a set of six ESADA PuO_2-UO_2 critical experiments. This set of critical experiments is described in detail in Reference 9. The average k_{eff} calculated for these just critical assemblies was 0.9946, with a standard deviation around this value of 0.0061 Δk. These predicted results are in excellent agreement with the measured critical data in view of the large variation in H_2O/UO_2 volume ratios and the additional complexities introduced by the mixed PuO_2-UO_2 fuel. Based on these two sets of critical experiments an uncertainty of 0.0163 Δk has been established for the PuO_2-UO_2 fuel at the 95 percent confidence level. This is analogous to the .0096 Δk value that will be used for the UO_2 fuel.

The PDQ-7 program is used in the final predictions of the reactivity of the fuel storage racks. The calculations are performed in four energy groups and take into account all the significant geometric details of the fuel assemblies, fuel boxes, and major structural components. The geometry used for most of the calculations is the basic cell, representing a repeating array of stainless-steel boxes. The specific geometry of this basic cell is shown in Figure 1 for the Ginna spent fuel storage rack with the mixed oxide fuel assemblies in place.

CALCULATIONAL APPROACH AND RESULTS

The calculational approach is to use the basic cell as illustrated in Figure 1 to calculate the reactivity of an infinite array of uniform spent fuel racks loaded with the mixed oxide fuel assemblies. These fuel assemblies are identical in mechanical design to the Westinghouse 14 x 14 fuel assemblies which were the basis for the earlier spent fuel storage rack criticality analysis. Table 5 provides the relevant data on the four mixed oxide fuel assemblies, including the enrichment range. Figure 1 illustrates the loading pattern to be used for the mixed oxide fuel assemblies.

The base case for the mixed oxide fuel assumes the minimum spacing on the spent fuel rack as did the earlier analysis for the uranium oxide fuel. No credit was taken for the axial or radial leakage, soluble boron, lumped burnable poison, or inconel spacer grids. The resulting base case multiplication factor for an infinite array of mixed oxide fuel of the reference design was calculated conservatively to be 0.8672 at 68°F. This can be compared with the original base case for the 3.5 w/o U-235 uranium oxide fuel for which the calculated multiplication factor was 0.8779 at 80°F (or 0.8770 at 68°F). Even when the incremental difference in the calculational uncertainty of .0067 Δk (= .0163 Δk-.0096Δk) due to the differences between the UO_2 and (UO_2-PuO_2) benchmark results is added to the base case for the mixed oxide fuel, the resulting multiplication factor 0.8739 (=.8672+.0067) is still less than the multiplication factor for the uranium oxide fuel of 0.8770. The other perturbations would be essentially the same as those previously determined for the uranium oxide fuel. The resulting worst case multiplication factor would therefore be less for the mixed oxide fuel than for the 3.5 w/o uranium oxide fuel for which these racks were originally licensed which was 0.8871.

Because of the well founded, conservative techniques 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 when loaded with mixed oxide fuel assemblies.

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TABLE 1

SUMMARY OF LEOPARD RESULTS [illegible] 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	1.0003
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	8	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 zircalloy

TABLE 2

WESTINGHOUSE UO_2 Zr-4 CLAD CYLINDRICAL CORE CRITICAL EXPERIMENTS

EXPERIMENT	PITCH (IN)	BORON CON-CONCENTRATION (ppm)	MATERIAL BUCKLING (FOR LEOPARD) CM-2	CRITICAL NO. OF PINS	RADIUS OF FUEL REGION (cm)	k_{eff} (LEOPARD/PDQ-7)
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

SAXTON PuO_2-UO_2 CRITICAL EXPERIMENTS[8]

Experiment	Boron (ppm)	H_2O/UO_2 (Volume)	Pitch (In)	k_{eff} LEOPARD	
1	0	1.68	.520	.9912	
2	0	2.17	.560	1.0029	
3	337	2.17	.560	1.0084	
4	0	4.70	.735	.9944	
5	0	10.76	1.040	1.0008	
				.9995	Mean
				.0068	Std. Dev.

TABLE 4

ESADA PuO_2-UO_2 CRITICAL EXPERIMENTS[9]

Experiment	Boron (ppm)	H_2O/UO_2 (Volume)	Pitch (In)	k_{eff} LEOPARD	
1	0	1.11	.690	.9902	
2	0	3.49	.9758	1.0055	
3	526	3.49	.9758	.9949	
4	0	3.49	.9758	.9948	
5	0	1.54	.750	.9878	
6	526	1.11	.690	.9945	
				.9946	Mean
				.0061	Std. Dev.

TABLE 5

MIXED OXIDE FUEL ASSEMBLY DATA
FOR
THE GINNA NUCLEAR PLANT

Item	Material	Dimensions (in.)
Fuel Assembly		
Overall Cross Section		7.784 x 7.784*
Overall Length		160.1
Control Rod Guide Tube		
Number per Assembly		16
Material	304 S.S.	
OD, Upper Section		0.5375
ID, Upper Section		0.5075
OD, Dashpot		0.4765
ID, Dashpot		0.4455
Dashpot Length		26.297
Instrumentation Tube		
Number per Assembly		1
Material	304 S.S.	
OD		0.422
ID		0.3455
Fuel Rod		
Number per Assembly		179
Active Length, inches		141.4
Overall Rod Length		148.6
Rod Pitch		0.556
Pre-pressurized		Yes
Cladding		
Material	Zirc-4	
Outside Diameter		0.422
Wall Thickness		0.0243
Inside Diameter		0.3734
Fuel Pellets		
Material	(UO_2+PuO_2) Sintered Pellets	
Pellet Diameter		0.3659
Pellet Density		95%
Dilutent	Natural UO_2	

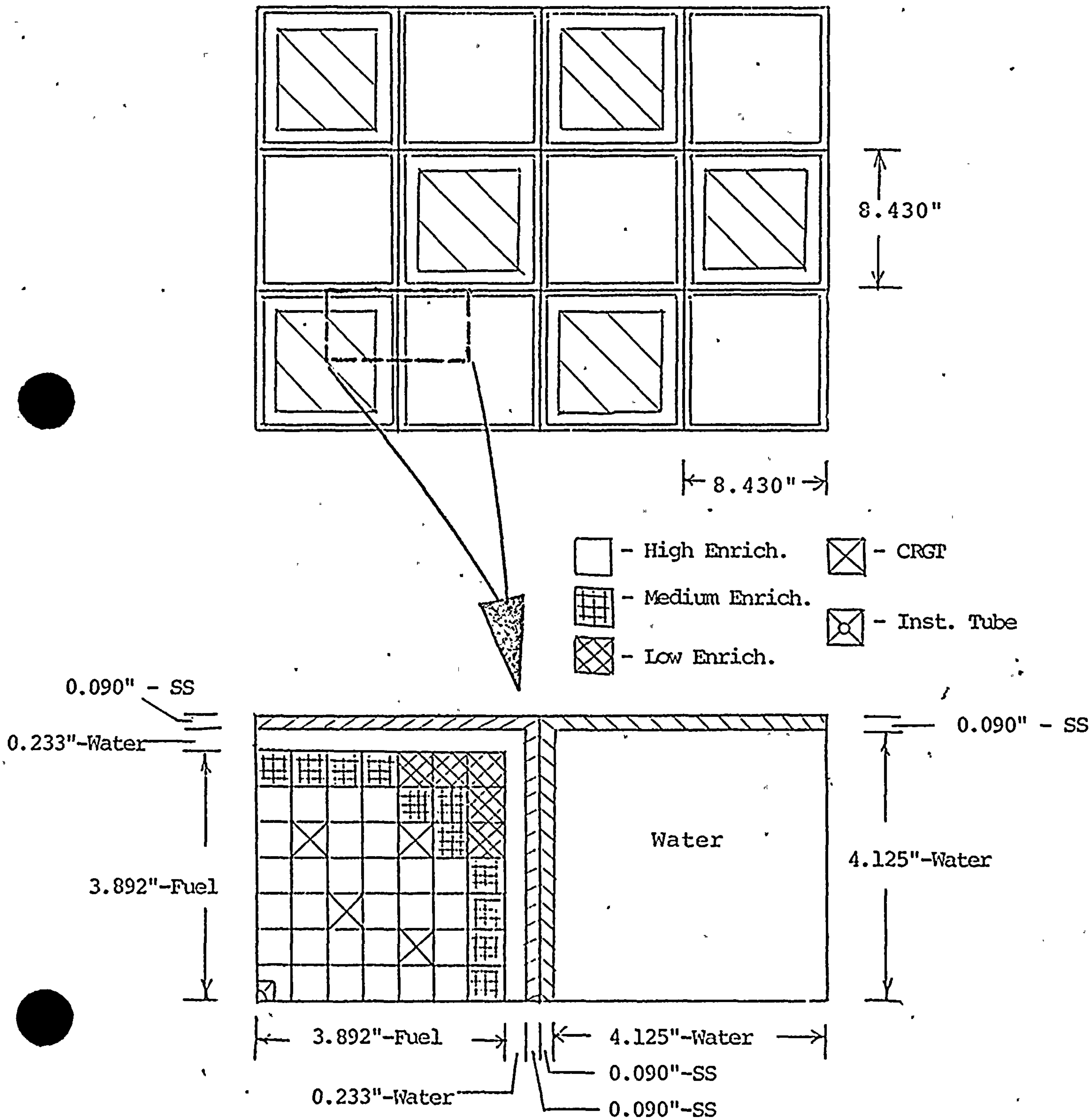
Enrichments (See Figure 1 for assembly fuel rod loading pattern)	w/o (U+Pu) Fis.	w/o Pu. tot
High enrichment (115 per assembly)	3.279	3.110
Medium enrichment (44 per assembly)	3.090	2.883
Low enrichment (20 per assembly)	2.736	2.452

Note: Pu fissile/Pu total = .8329

*(14 x .556") x (14 x .556") = 7.784" x 7.784"

FIGURE 1

PDQ CALCULATIONAL MODEL
FOR THE
GINNA SPENT FUEL STORAGE RACK
WITH MIXED OXIDE FUEL



Note: Boundary condition at top of detailed figure is 180° Rotational Symmetry

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Radiological Impact of Mixed Oxide Fuel Assemblies

I. Summary

An assessment is performed which addresses the radiological impact of the use of four mixed oxide fuel assemblies in the Ginna Station reactor core. Normal operation and accidental effluent releases are evaluated by comparing the relative quantities of radioisotopes generated for uranium-only and mixed oxide fuel.

II. Method of Evaluation

A substantial amount of information pertaining to the use and impact of mixed oxide fuel was developed in NUREG-0002, the Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors or GESMO Report. In that report, a model LWR using MOX fuel was devised for comparative impact assessment with LWR's fueled only with uranium. The model MOX-fueled LWR reactor is assumed to be charged with fuel having an average plutonium content of 1.8 weight percent of the heavy metal (Pu and U) in the charged fuel. Furthermore, as many as 40 percent of the rods in the model LWR may be MOX rods. The planned Ginna core reload with 4 MOX fuel assemblies will comprise less than a 0.5 weight percent average Pu content of the total heavy metal being added and less than 4 percent of the rods in the reactor will be MOX rods. Therefore, radiological impacts calculated for the GESMO model reactor will envelop those for the Ginna case.

A radiological assessment was then performed using the radioactive source terms calculated in the GESMO Report for the model MOX-fueled plant and an equivalent-sized reactor unit utilizing UO_2 fuel. Relative inventories and release quantities of key dose-contributing radionuclides could then be directly used in determining the net effect upon resultant whole body and thyroid doses.

III. Accidental Releases

1. Spent Fuel Assembly Drop:

   The radiological impact of a postulated fuel handling accident involving a dropped MOX fuel assembly was derived by comparing the calculated GESMO source terms with the results of an evaluation which considered the potential consequences of a refueling accident inside the Ginna Station containment building (submitted to A. Schwencer, Nuclear Regulatory Commission, March 18, 1977). The limiting accident dose pathway identified in the March 18, 1977 evaluation was the 0-2 hour thyroid dose from inhalation, which was calculated to be 103 rem and within the guidelines of 10 CFR Part 100. The associated maximum

[illegible]

whole body dose was approximately 2 rads from cloud immersion. Although containment isolation would occur in the event of such an accident, no credit was taken for isolation.

Table IV C-35 illustrates the results of the GESMO analysis of MOX and uranium fuel source terms for calculating thyroid and whole body dose. In general, whole body dose due to released quantities of noble gas and iodine from a MOX assembly having a burnup history similar to that assumed in the 1977 Ginna evaluation would not be expected to exceed the dose from a uranium assembly. For thyroid dose, GESMO showed that the iodine thyroid dose source term may increase 3-14 percent depending upon the Pu characteristics and degree of burnup. More typically, at high burnups, which is the limiting case for a fuel handling accident, the increase in the thyroid dose source term is at the lower end of the range.

The resulting impact upon the fuel handling accident with a MOX fuel assembly will therefore be a modest increase in the maximum offsite thyroid dose and the thyroid dose remains well within the site boundary dose guidelines of 10 CFR Part 100.

2. Loss-of-Coolant Accident:

The design basis loss-of-coolant accident was analyzed by the Commission Staff in the January 20, 1972 <u>Safety Evaluation for the R. E. Ginna Nuclear Power Plant Increase</u>; in Section 14 of the R. E. Ginna FSAR, in Section 7 of the R. E. Ginna Final Environmental Statement, and in Section 6 of the R. E. Ginna Environmental Report. In each evaluation the offsite consequences of a postulated accident were shown to be well within the 10 CFR Part 100 guidelines.

The 0-2 hour site boundary thyroid inhalation dose was calculated to be 155 rem in the Commission's 1972 Safety Evaluation, and was more limiting than the associated dose to the whole body. The total increase in iodine core inventory available for release contributed by the addition of 4 MOX fuel assemblies will necessarily be well below the 3-14 percent mentioned above due to the presence of 117 other uranium fuel assemblies. Thus, the potential offsite thyroid dose will increase by only a small amount and will remain below 10 CFR Part 100 by a considerable margin.

IV. Routine Releases

The June 4, 1976 evaluation entitled <u>Dose Calculations to Conform with Appendix I Requirements - Ginna Station</u> demonstrated that calculated effluent releases were well within the Appendix I

design objectives. In Tables IV C-18 and IV C-19 of the GESMO report, a comparison is made between calculated liquid and gaseous radioactive release quantities for PWR's utilizing MOX and UO_2 fuel, respectively. The GESMO tables indicate that differences in the relative quantities of radionuclides releases are insignificant, except where modest increases result in I-131 and tritium source terms in the MOX case. The percentage increases are 8 percent and 9 percent, respectively. The total increase in normal effluents will be less because only 4 MOX assemblies will be loaded. Therefore the Appendix I objectives will still be met.

V. Conclusion

The radiological impacts caused by the addition of 4 mixed oxide fuel assemblies have been conservatively analyzed for accidents and routine operations. The incremental radiological doses attributed to the presence of the MOX assemblies have been shown to be small for the most potentially significant dose pathways and all applicable guidelines for routine and accidental radiation exposure continue to be met.

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Reg File

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	Docket Nos.:
	)	50-237, 50-249, 50-254, 50-265
COMMONWEALTH EDISON COMPANY	)	
	)	Amendments to Facility
(Dresden Station, Units 2 & 3, and	)	Operating License Nos.:
Quad Cities Station, Units 1 & 2)	)	DPR-19, DPR-25, DPR-29, DPR-30

ORDER

This Board, by an Order docketed on December 4, 1978, gave notice that a Special Prehearing Conference in the above proceeding would be held on January 11, 1979, in Chicago, Illinois. In response to this notice the parties to this proceeding and the persons seeking to intervene in this proceeding (petitioners) jointly arranged and participated in a telephone conference on December 11, 1978, with the Chairman of the Board in order to discuss the Special Prehearing Conference.

During the telephone conference the parties and petitioners stated their belief that the business of the Special Prehearing Conference could be conducted more effectively if the Conference were postponed. The parties and petitioners requested additional time to discuss possible contentions, and they proposed a schedule for filing and responding to contentions which would enable the Board to have received substantially all argument on the contentions by the date of the Conference. According to the schedule proposed, petitioners' contentions

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shall be filed on December 29, 1978, responses to those contentions shall be filed on January 12, 1979, and responses to the responses shall be filed on January 26, 1979. Oral argument at the Conference could be focused on the precise issues which then remain.

For good cause shown, the Special Prehearing Conference scheduled for January 11, 1979, is hereby cancelled and notice is hereby given that the Conference will be held at 10:00 a.m. on Thursday, February 1, 1979, in Room 2502, United States Courthouse and Federal Building, 219 South Dearborn Street, Chicago, Illinois.

In light of the change in the date of the Conference, the parties and petitioners are excused from the requirement that they report to the Board by December 15, 1978, the progress of their negotiations.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY AND LICENSING
BOARD DESIGNATED TO RULE ON
PETITIONS FOR LEAVE TO INTERVENE

Gary J. Milhollin

Gary L. Milhollin, Chairman

Dated at Madison, Wisconsin,

December 13, 1978.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of COMMONWEALTH EDISON COMPANY (Dresden Nuclear Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2)	)	Docket No.(s) 50-237 50-249 50-254 50-265

CERTIFICATE OF SERVICE

I hereby certify that I have this day served the foregoing document(s) upon each person designated on the official service list compiled by the Office of the Secretary of the Commission in this proceeding in accordance with the requirements of Section 2.712 of 10 CFR Part 2 - Rules of Practice, of the Nuclear Regulatory Commission's Rules and Regulations.

Dated at Washington, D.C. this
14th day of Dec 1978.

Peggy T. Bowring
Office of the Secretary of the Commission

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket No.(s) 50-237
	)	50-249
(Dresden Nuclear Power Station,	)	50-254
Units 2 and 3; Quad Cities	)	50-265
Nuclear Power Station, Units 1	)	
and 2)	)	

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

3/8/78

In the Matter of ROCHESTER GAS & ELECTRIC CORPORATION (R. E. Ginna Nuclear Power Plant, Unit No. 1)	Docket No. 50-244

NOTICE OF WITHDRAWAL OF APPEARANCE

Notice is hereby given that effective March 10, 1978, I will withdraw my appearance in the above captioned proceeding. All mail and service lists should be amended to delete my name after that date.

Respectfully submitted,

Auburn L. Mitchell

Auburn L. Mitchell
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 8th day of March, 1978

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ROCHESTER GAS & ELECTRIC CORPORATION	)	Docket No. 50-244
	)	
(R. E. Ginna Nuclear Power Plant, Unit No. 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NOTICE OF WITHDRAWAL OF APPEARANCE" of Auburn L. Mitchell in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or air mail, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 10th day of March, 1978:

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Auburn L. Mitchell

Auburn L. Mitchell
Counsel for NRC Staff

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of ROCHESTER GAS AND ELECTRIC CORPORATION (R. E. Ginna Nuclear Power Plant, Unit No. 1)	)	Docket No. 50-244

ORDER

The Regulatory Staff and Intervenor Michael R. Slade have agreed upon a statement of contentions to be asserted by the Intervenor in this case. The Applicant opposes the intervention and has moved to strike all of the Intervenor's contentions. The agreement between the Staff and Intervenor Slade states the following: "Upon approval of these stipulated contentions by the Board, all contentions previously submitted by intervenor shall be deemed withdrawn". The Applicant correctly points out that such an attempted reservation by the Intervenor is somewhat ambiguous. In the Board's view, however, the only contentions presently being asserted are those stated in the Intervenor's written agreement with the Regulatory Staff. All other statements of contentions are deemed to be withdrawn.

The Applicant's motion to dismiss the petition is denied. The Board's ruling on each of the contentions follows.

Contentions C, G, and H are rejected as issues in controversy because each of them is vague and lacking in particularity.

The remaining contentions are admitted, as follows:

Contention A

The Applicant's quality assurance program is inadequate and/or fails to conform to 10 CFR Part 50, Appendix B criteria because:

a) it has not corrected malfunctions of electric type valve operators;

b) the main steamline isolation valves do not meet minimum code requirements for wall thicknesses; and

c) criterion X is not met in that there is an inadequate operations program for inspection of activities affecting quality.

Contention B

Applicant has not demonstrated conformance with the amended ECCS criteria as determined by the AEC in Docket RM-50-1.

Contention D

The Applicant is in violation of applicable Federal and New York State water quality standards in that it does not possess an exemption for the discharge of water at

temperatures of 23.4°F above ambient as described in the FES, pp. 3-7, sec. 3.4.1.

Contention E

The NEPA analysis for the facility is inadequate because it fails to adequately consider the effect of cold shock on lake biota resulting from emergency shutdown of the facility, and because it fails to adequately consider the effect of cold shock on lake biota as a result of recirculation of discharge water into the intake water during the winter when lake ambient temperature falls below 37°F.

Contention F

The FES is inadequate because it fails to treat the following energy conservation alternatives:

a) ending special discounts for large volume electrical use;

b) increasing electrical pricing in order to decrease demand;

c) implementation by the Applicant of maximum lighting levels per square foot by its customers;

d) setting insulation standards for new and old customers;

e) promoting energy efficiency labeling;

f) discouraging electric space heating and air conditioning (in climatic conditions that do not require it); and

g) peak or demand load flattening techniques including time of day metering charges, load staggering and/or selective load shedding.

Contention I

Applicant has failed to submit an adequate site contingency plan because the Applicant has failed to apprise the population of the existence of the site contingency plan and what would be required of the surrounding population if the plan had to be implemented.

Contention J

Applicant has failed to provide flood protection against maximum high water levels shown to have occurred or to have been projected for Lake Ontario.

Contention K

Applicant's radwaste systems management program is inadequate because it does not keep releases to a level as low as reasonably achievable.

The State of New York shall participate in this proceeding as an interested State pursuant to 10 CFR §2.715(c).

SO ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

Edward Luton, Chairman

Dated at Bethesda, Maryland
this 25th day of March 1977.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of ROCHESTER GAS AND ELECTRIC COMPANY (R. E. Ginna Nuclear Power Plant, Unit No. 1)	)))))))))	Docket No.(s) 50-244

CERTIFICATE OF SERVICE

I hereby certify that I have this day served the foregoing document(s) upon each person designated on the official service list compiled by the Office of the Secretary of the Commission in this proceeding in accordance with the requirements of Section 2.712 of 10 CFR Part 2 - Rules of Practice, of the Nuclear Regulatory Commission's Rules and Regulations.

Dated at Washington, D.C. this
28th day of Mar 1977.

PA Downing
Office of the Secretary of the Commission

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of ROCHESTER GAS AND ELECTRIC CORPORATION (R. E. Ginna Nuclear Power Plant, Unit No. 1)	)	Docket No.(s) 50-244

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UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION	)	Docket No. 50-244
	)	
(R. E. Ginna Nuclear Power Plant, Unit 1)	)	

ORDER

Following discussion with all parties to this proceeding by conference telephone call on February 15, 1974, a "Motion for Postponement of Prehearing Conference" made by Intervenor Michael Slade was granted orally by the Board. That conference had been scheduled to take place on February 20, 1974.

The prehearing conference in this matter is hereby rescheduled and will take place on March 12, 1974, at 9:30 a.m., local time, in the East Courtroom, 2nd Floor, U.S. District Court, 100 State Street, Rochester, New York.

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Office of the Secretary
Public Proceedings
Branch

IT IS SO ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

Edward Luton, Chairman

Issued at Washington, D. C.,
this 19th day of February, 1974.

HEARING

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of)
)
ROCHESTER GAS & ELECTRIC COMPANY) Docket No. 50-244
(R.E. Ginna Nuclear Power Plant,)
Unit No. 1))

CERTIFICATE OF SERVICE

I hereby certify that I have this day served the foregoing document upon each person designated on the official service list compiled by the Office of the Secretary of the Commission in this proceeding in accordance with the requirements of Section 2.712 of 10 CFR Part 2 - Rules of Practice, of the Atomic Energy Commission's Rules and Regulations.

Dated at Washington, D. C.
this 19th day of 4/25 1974.

P. A. Downing
Office of the Secretary of the Commission

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION	)	Docket No. 50-244
	)	
(R. E. Ginna Nuclear Power Plant Unit No. 1)	)	

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