FACILITY CHANGE REQUEST NO. 4

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

Applicant hereby requests the Atomic Energy Commission to authorize:

1. The use of Oyster Creek Type IV fuel bundles, designed and fabricated by the Exxon Nuclear Company, and
2. Operation of the Oyster Creek reactor with the reload configuration described herein.

This change request includes a full description of the Type IV fuel bundles and the safety, transient, and accident analyses that demonstrate that these changes do not present significant safety hazards considerations not described or implicit in the Oyster Creek Nuclear Generating Station Facility Description and Safety Analysis Report and Amendments, thereto. Therefore, there is reasonable assurance that the health and safety of the public will not be endangered by these changes.

JERSEY CENTRAL POWER & LIGHT COMPANY

By _____
Vice President

STATE OF NEW JERSEY)
)
COUNTY OF MORRIS)

Sworn and subscribed to before me this 4th day of _____ 1973.

_____
Notary Public

MARION P. BAWIEC
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Jan. 21, 1974

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N.J. 07960 • 539-6111

April 4, 1973

The Honorable Lawrence J. McNally
Mayor, Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor McNally:

Enclosed herewith is one copy of Supplement No. 2 to Facility Change Request No. 4 for the Oyster Creek Nuclear Generating Station.

This Change Request was filed with the Atomic Energy Commission on April 4, 1973.

Very truly yours,

R. H. Sims
Vice President

mw
Enclosures

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

IN THE MATTER OF	)	
	)	DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY	)	

CERTIFICATE OF SERVICE

This is to certify that a copy of Supplement No. 2 to Facility Change Request No. 4 for the Oyster Creek Nuclear Generating Station dated April 4 , 1973, and filed with the United States Atomic Energy Commission on April 4, 1973 has this 4th day of April 1973, been served on the Mayor of Lacey Township, Ocean County, New Jersey, by deposit in the United States mail, addressed as follows:

The Honorable Lawrence J. McNally
Mayor, Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

By _____
Vice President

DATED: April 4, 1973

I INTRODUCTION

This submittal describes the design of the Exxon Nuclear PuO_2-UO_2 assemblies proposed for utilization in Jersey Central Power & Light Company's Oyster Creek Nuclear Generating Station. Loading of four lead assemblies of this design is scheduled for the Spring 1973 outage.

The design of these fuel assemblies (hereafter referred to as Type IV) is compatible with the mechanical, thermal-hydraulic, and nuclear characteristics of the first core fuel assemblies (Type I) of the Oyster Creek Nuclear Generating Station as well as the other reload fuel assemblies (156 Type II, four Type III and 144 Type III E) which have been or will be loaded into the reactor.

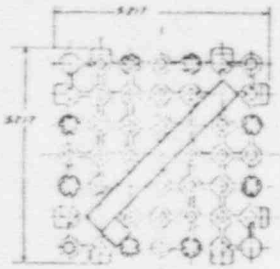
Like the previous fuel, Type III E, the Type IV fuel bundle consists of a 7 x 7 array of rods. The center rod is a spacer capture rod, filled with solid Zircaloy and four rods containing gadolinium oxide are provided for supplementary reactivity control. Type IV fuel differs from Type III E fuel only in that 12 of the 48 fueled rods contain Pu^{239} and Pu^{241} rather than U-235, as the fissile isotope.

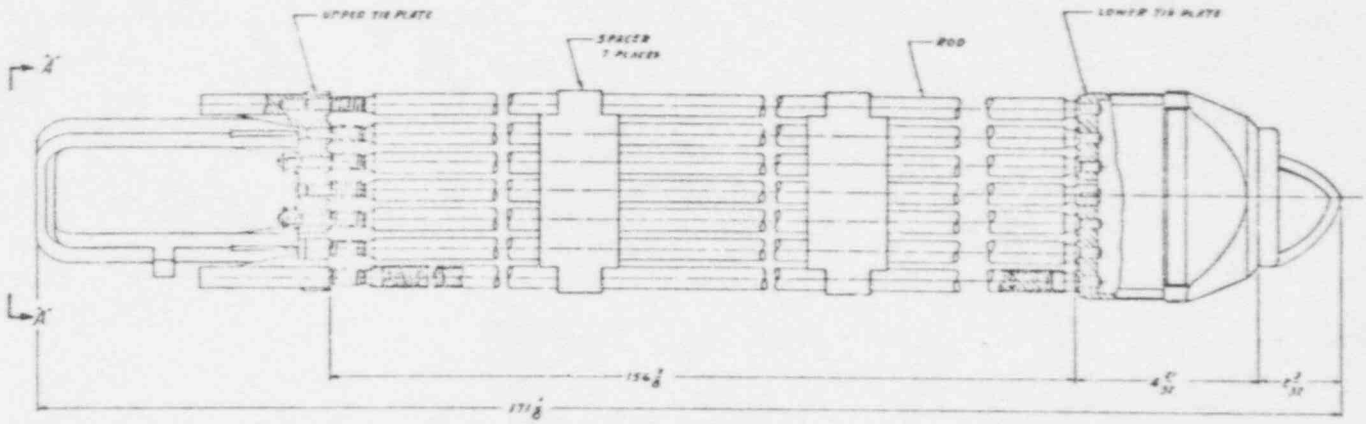
The mechanical, thermal-hydraulic, and nuclear characteristics of the Exxon Nuclear Type IV Fuel and the effects of using this fuel in the reactor core are discussed in Sections II through VII below.

II TYPE IV FUEL DESIGN AND CHARACTERISTICS

A. MECHANICAL DESIGN OF THE FUEL ASSEMBLIES

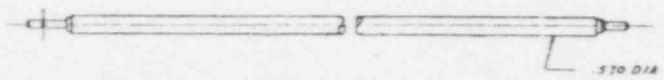
The Type IV fuel assembly, shown in Figure 1, is made up of 49 rods in a 7 x 7 square array with an active fuel length of 144 inches for all rods except for the center spacer capture rod, which is filled with solid Zircaloy.



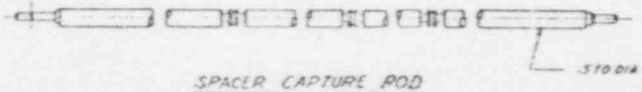


ASSEMBLY

ITEM	MATERIAL
CLADDING	ZIRCALOY
SPACER	ZIRCALOY
TIE PLATE	304 L SST
FUEL	UO_2 & PuO_2 UO_2



STANDARD ROD



SPACER CAPTURE ROD

The spacing between rods is maintained by seven Zircaloy spacers containing Inconel springs equally spaced along the length of the fuel rods and by tie plates at the upper and lower ends of the rods. A centrally-located spacer capture rod maintains the axial location of the spacers.

The ends of all rods have extensions which fit into holes in the upper and lower tie plates. Eight of the peripheral fuel rods, called tie rods, are threaded into the lower tie plate and pass through the upper tie plate where they are secured by lock washers and nuts. The remaining rods are restrained within holes in the upper and lower tie plates. Coil springs on each rod, captured by the upper fuel rod extensions (See Figure 1), act together to force the upper tie plate upward against the tie rod nuts. Each spring acts individually to assure that the fuel rod remains seated in the lower tie plate.

The level of enrichment is identified on the end surface of each fuel rod upper end cap by a letter code (See Figure 2). During and after assembly, the correct position of each rod is verified by comparison with a template which specifies the location of each rod type, and photographs are taken of the fuel rod ends as part of the quality control records. In addition notches and a unique serial number are used to identify individual fuel rods for enrichment, poison loading, and clad thickness. A detailed description of the assembly components is presented in Table I.

Spacers

The spacers are designed to maintain the correct rod-to-rod spacing but allow for differential axial expansion. The spacer uses an egg crate type design of criss-crossed narrow Zircaloy strips interlocked and welded together with a peripheral band to form a cell for each fuel rod. Each fuel rod is centered in its cell by an Inconel spring which holds the rod against support

TABLE I

COMPARISON OF MECHANICAL CHARACTERISTICS OF

TYPE III E AND IV ASSEMBLIES

Characteristics	Type III E	Type IV
FUEL MATERIALS	UO_2	UO_2 and PuO_2-UO_2
Initial Enrichment, w/o fissile isotope:		
Low	1.59	See Figure 2
Medium Low	--	
Medium	2.42	
High	2.87	
Bundle Average Fissile	2.63	2.61
Pu isotopic composition (82.2% fissile)		77% Pu^{239}, 5.2% Pu^{241}
Pellet Dish, % of Undished Volume	2.0 (in all rods)	2.0 (in all rods)
Average Pellet Density % Theoretical Density	93.5 & 94.5	93.5 & 94.5**
Number of Rods and Pellet Diameter, in.	8-0.491 11-0.489 7-1.488 22-0.468	See Figure 2
Melting Point, °F	5080	5080 - UO_2 5040 - Mixed Oxid
CLADDING MATERIAL	Zr-2	Zr-2
Outside Diameter, in.	0.570	0.570
Number of Rods and Wall Thickness, in.	27-0.0355 22-0.0455	27-0.0355** 22-0.0455
FUEL RODS		
Active Length in. Standard Rod	144	144
Gas Plenium Length, in. Standard Rod	10 5/8	10 5/8
Fill Gas	Helium	Helium

TABLE 1 (Continued)

Characteristics	Type III E	Type IV
FUEL BUNDLE		
Geometry	7 x 7	7 x 7
Standard Rods/Bundle	44	44
Spacer Capture Rods/Bundle	1	1
Poison Rods/Bundle	4	4
Total Rods/Bundle	49	49
Rod Pitch, inches	0.738	0.738
Water to Fuel Bundle Ratio (Cold)	2.48	2.48

** See Figure 2 for locations.

dimples with sufficient force to minimize flow induced vibration. Guide tabs are provided on the upper edge of the spacers to avoid possible hang-up during channeling operations.

Upper and Lower Tie Plates (See Figure 1)

The upper tie plate is a flat perforated plate of cast and machined stainless steel. It is designed to maintain the correct position of the fuel rods, provide flow passages and maintain the position of the fuel assembly within the reactor top grid assembly. A handle is attached to the plate for loading, unloading, and for general handling of the fuel assembly. A boss on the lifting bail points to the nearest control rod when the fuel assembly is correctly oriented in the reactor core.

The lower tie plate consists of a flat perforated plate and an inlet box section all cast and machined as a single unit from stainless steel. The perforated plate is designed to support and maintain the correct position of the rods and to permit coolant flow through the fuel assembly. The inlet box section distributes coolant from the assembly support section to the fuel rods.

Fuel Pellets and Rods

The fuel consists of compacted and sintered uranium dioxide or mixed PuO_2-UO_2 (natural uranium) powders formed into cylindrical pellets. The nominal density of the pellets is either 93.5 or 94.5 percent of theoretical density. The pellets are dished to remove a nominal 2 percent of the fuel volume. This volume is provided to accommodate fuel axial expansion and irradiation-induced swelling.

The fuel pellets are stacked in Zircaloy-2 cladding sealed by welding Zircaloy-2 caps into each end. The atmosphere within the rods is helium. The requirement that the cladding be free-standing is met by specifying a minimum wall thickness of .033 inches. This thickness provides substantial margin

against creep collapse of the cladding due to primary coolant pressure throughout the life of the fuel.

The design of the fuel pellets and rods considered the potential for pellet densification, fuel swelling behavior, thermal expansion and distortion of the pellets, pellet-clad interaction, and fission gas release. Consideration of these factors at the maximum design peak burnup resulted in the selection of appropriate pellet dimensions, densities, dishing requirements, diametral gap, and fuel rod plenum.

A comparison of the mechanical characteristics of the Exxon Nuclear mixed oxide assemblies (Type IV) and Exxon Nuclear UO_2 assemblies (Type III E) is given in Table I.

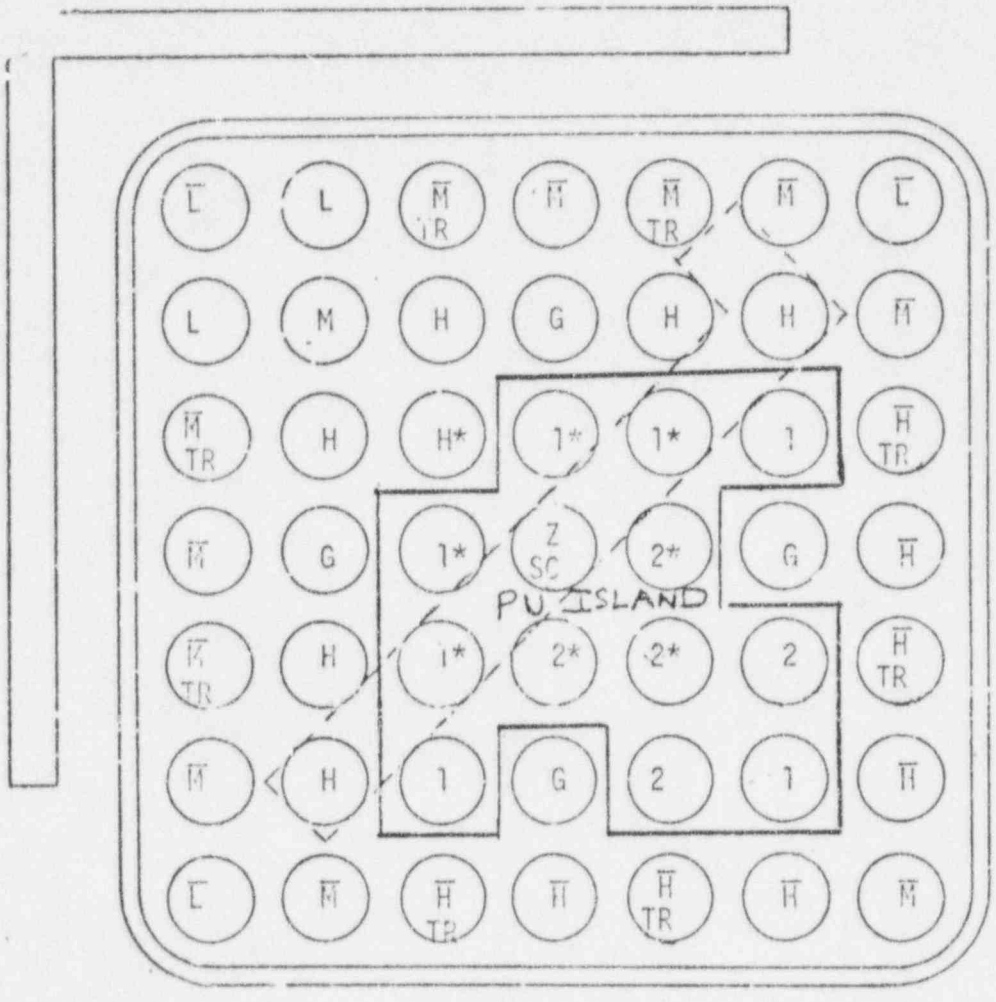
B. NUCLEAR CHARACTERISTICS OF THE FUEL ASSEMBLIES

Type IV assemblies are designed and fabricated to have a nominal average enrichment of 2.61 w/o fissile isotopes. Four of the UO_2 fuel rods in the Type IV assembly contain 1.0 w/o gadolinium as a burnable poison. The fuel rod enrichment and poison distribution are given in Figure 2. All pellets in the burnable poison rods contain gadolinium.

Calculated infinite multiplication factors and other parameters of the Type IV fuel assemblies are compared to those of Type III E assemblies in Tables II and III. As the calculations indicate, the differences between the designs are small. In the absence of gadolinium, each of the temperature and void defects is slightly more negative with the mixed oxide fuel, such that the overall cold-to-hot operating defect in k^{∞} is about 0.6% k more negative. In addition, the poison worth of the gadolinium burnable poison and its effect on the reactivity defect is less in the mixed oxide-containing fuel bundle because the relative thermal flux in the poison rod locations is depressed by the presence of the adjacent mixed oxide rods.

FIGURE 2

OYSTER CREEK TYPE IV ASSEMBLY



TR = Tie Rod SC = Spacer Capture Rod

Rod ID Mark	w/o Fissile Isotopes	No. Rods	Material	Clad Thickness, Mils	Pellet Density	Pell Diamete
L	1.59	3	UO_2	45.5	93.5	.468 ±
L	1.59	2	UO_2	35.5	93.5	.488 ±
M	2.42	11	UO_2	45.5	93.5	.468 ±
M	2.42	1	UO_2	35.5	93.5	.488 ±
H	2.87	8	UO_2	45.5	93.5	.468 ±
H	2.87	6	UO_2	35.5	94.5	.489 ±
H*	2.87	1	UO_2	35.5	94.5	.491 ±
G	2.87	4	$UO_2+Gd_2O_3$	35.5	94.5	.488 ±
1	2.50	3	UO_2+PuO_2	35.5	94.5	.488 ±
1*	2.50	4	UO_2+PuO_2	35.5	94.5	.490 ±
2	3.20	2	UO_2+PuO_2	35.5	94.5	.488 ±
2*	3.20	3	UO_2+PuO_2	35.5	94.5	.4895 ±
Z	Non-Fueled	1	Zirc	35.5	---	---

TABLE II

CALCULATED INFINITE MULTIPLICATION FACTORS

(ZERO EXPOSURE)

		TYPE E		TYPE IV	
		With Gadolinium	Without Gadolinium	With Gadolinium	Without Gadolinium
Cold -	68 F	1.1453	1.2926	1.1576	1.2849
	300 F	1.1391	1.3010	1.1558	1.2926
	549 F	1.1322	1.3067	1.1541	1.2950
Full Power -	0% Void	1.1271	1.3005	1.1485	1.2886
	32% Void	1.1112	1.2842	1.1331	1.2687
	64% Void	1.0816	1.2497	1.1071	1.2314

TABLE III

CALCULATED REACTIVITY DEFECTS
(ZERO EXPOSURE)

Defects ($\Delta k_\infty/k_\infty$)	TYPE III E		TYPE IV	
	w/Gd	wo/Gd	w/Gd	wo/Gd
Doppler Defect (Hot Standby to Full Power)	- .0045	- .0047	- .0048	- .0049
Void Defect (0-32% Voids)	- .0141	- .0125	- .0134	- .0154
Temperature Defect (68-549 F)	- .0114	+ .0109	- .0030	+ .0079
Control Rod Worth (Cold)	- .151	- .148	- .149	- .148

Fuel rod local peaking factors for Type IV fuel are compared to those for Type III E fuel in Figure 3. While the substitution of the twelve PuO_2-containing fuel rods for UO_2 fuel rods substantially increases the power in these rods, the Local Peak-to-Average power is actually calculated to be about 2 percent lower in the Type IV fuel.

C. THERMAL AND HYDRAULIC CHARACTERISTICS OF THE FUEL ASSEMBLIES

A comparison of the thermal and hydraulic characteristics of Type III E and Type IV fuel assemblies is given in Table IV. There are no difference between the heat transfer characteristics of the designs.

1. Maximum Fuel Temperature

The calculated maximum fuel temperature (at 17.2 kw/ft maximum linear heat generation rate) for Type IV fuel is 4260 F, which is the same as that calculated for Type III E fuel. This maximum temperature is found in the same location (a perimeter UO_2 fuel rod) in both designs. A PuO_2-UO_2 fuel rod operating at the same linear heat generation rate would have a maximum fuel temperature approximately 200 F lower, due to the change in the rod radial power distribution resulting from the introduction of plutonium. Since the maximum temperature is found in a UO_2 fuel rod in both cases, the discussion of the margin to centerline melting, the effects of exposure, and the effects of gadolinium are the same as given in Facility Changes Request No. 4, January 18, 1973.

2. Flow Compatibility

The Type IV fuel is hydraulically identical to Type III E fuel. The discussion of flow compatibility presented in Facility Change Request No. 4 also applies to Type IV fuel.

FIGURE 3

FUEL ROD LOCAL POWER DISTRIBUTIONS

TYPE III E AND TYPE IV FUEL

32% Void

0 Exposure

1.156 1.153	1.050 1.042	1.185 1.167	1.100 1.077	1.126 1.092	1.219 1.182	.989 .965
	1.159 1.133	1.102 1.087	.493 .467	1.039 .971	1.179 1.122	1.123 1.069
		.952 .863	.884 .998	.905 1.002	.977 1.192	1.095 1.009
			0.0 0.0	.850 .999	.454 .393	1.007 .915
		Pu ISLAND		.861 .966	.921 1.178	1.033 .909
III E IV					1.039 1.190	1.123 .996
						1.122 1.032

TABLE IV

COMPARISON OF THERMAL HYDRAULIC CHARACTERISTICS

OF THE OYSTER CREEK TYPE III E AND IV ASSEMBLIES

CORE CONDITIONS

Rated Power, MWt	1930
Operating Pressure, psia	1035
Core Inlet Enthalpy, Btu/lb	517.3
Total Core Flow Rate, 10^6 lb/hr	61.0
Core Leakage Flow, %	7
Effective Core Flow Rate, 10^6 lb/hr	56.73
Fraction of Power Generated in Fuel, %	96.7

Characteristics	Type III E	Type IV
FUEL DESCRIPTION		
Number of Active Rods	48	48
Total Fuel Length, Ft/Bundle	576	576
Heat Transfer Surface Ft^2/Bundle	85.98	85.98
Bare Rod Flow Area Ft^2/Bundle	.1057	.1057
POWER PEAKING FACTORS AND FUEL PERFORMANCE		
Technical Specification Limit-- Total Peaking Factor*	3.01	3.01
Assumed Local**	1.29	1.29
Axial X Radial	2.33	2.33

* The total peaking factor limit for Types III E and IV fuel is reduced to compensate for the introduction of the inert spacer capture rod, such that the peak linear rate generation rate is still limited to 17.2 kw/ft.

** Includes Engineering Factor of 1.05

TABLE IV (Continued)

Characteristics	Type III E	Type IV
Corresponding Maximum Heating Rate, kw/ft.	17.2	17.2
Corresponding Maximum Heat Flux at Rated Powe., Btu/hr-ft^2	393,400	393,400
Corresponding Clad Surface Temperature,F	568	568
Corresponding Critical Heat Flux Ratio	>2.0(XN-1)	>2.0(XN-1

3. Minimum Critical Heat Flux Ratio

The minimum critical heat flux ratio discussion presented in Facility Change Request No. 4 also applies to Type IV fuel.

D. FUEL QUALITY ASSURANCE AND QUALITY CONTROL PROGRAM

See Facility Change Request No. 4.

III CYCLE 3 CORE CONFIGURATION AND ANALYSIS

and

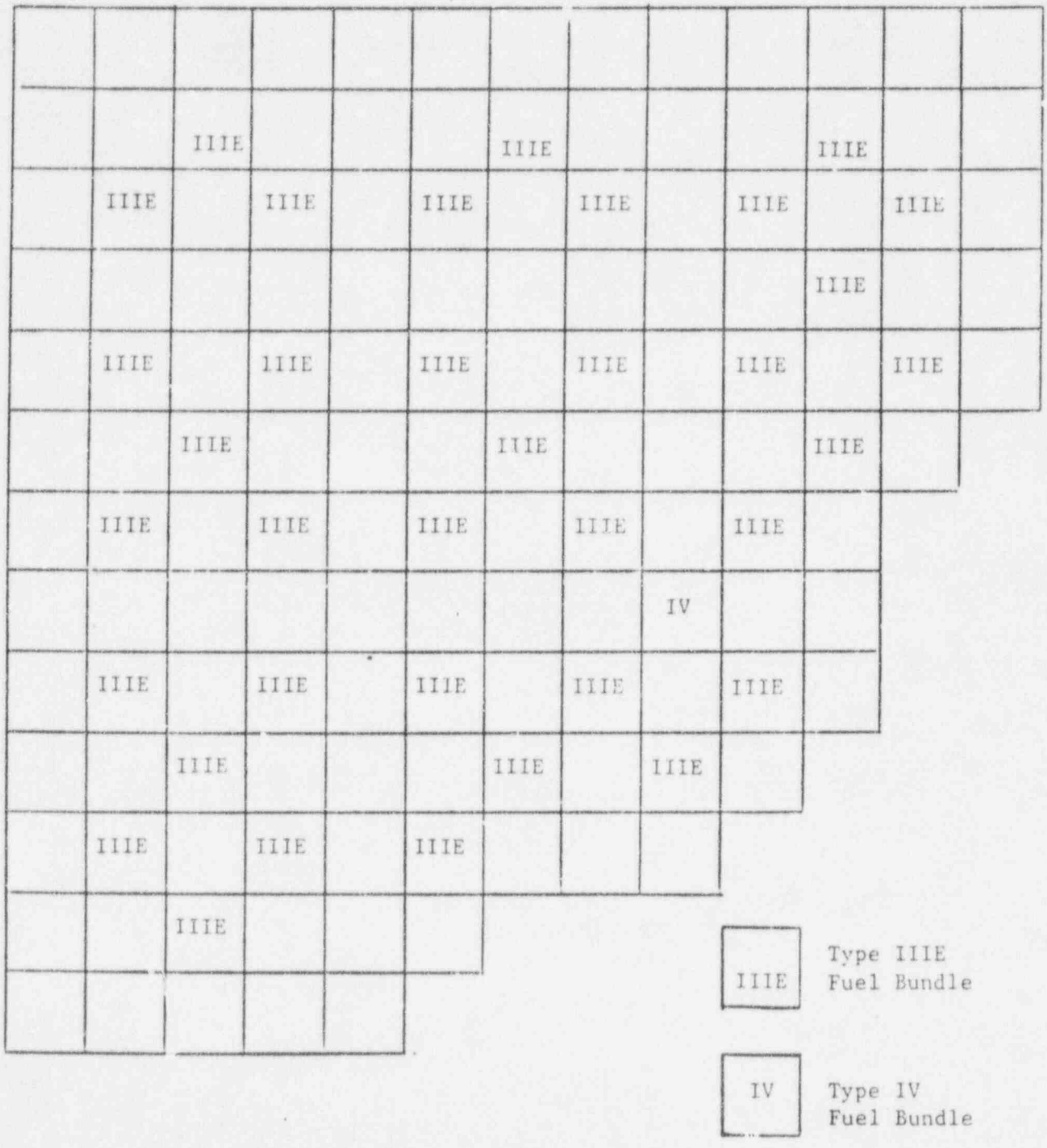
IV SAFETY ANALYSIS

Type IV fuel assemblies are designed to be substituted for Type III E fuel assemblies with an insignificant perturbation to the core neutronics and thermal-hydraulic analyses. For Cycle 3 operation, four (4) Type IV assemblies will be loaded in the core positions shown on Figure 4 or in equivalent quarter core symmetric positions where they replace four fresh Type III E assemblies.

Inspection of Tables II, III, and IV clearly shows that the core analysis performed in Facility Change Request No. 4, where reloading with 148 Type III E fuel assemblies was assumed, will be valid if four Type IV assemblie are substituted for Type III E assemblies. The small differences in assembly neutronic parameters will result in negligibly small differences in core average parameters. The maximum fuel rod power in Type IV fuel will be equal to or less than the maximum rod power in Type III E fuel in the same environment, and the two types of assemblies are hydraulically identical. Because of these features, the Safety Analyses presented in Section IV of Facility Change Request No. 4 also apply when the core loading includes the four Type IV assemblies.

OYSTER CREEK - CYCLE

LOADING PATTERN FOR LICENSING ANALYSES



One Quarter of Symmetrical Core Loading Pattern

V INCIDENT ANALYSIS

A. FUEL LOADING ERRORS

1. Rod Loading Errors

Quality control and overchecks during fuel fabrication make misplacement of fuel enrichment extremely unlikely. Additional assurance against fuel rod mislocation in the Type IV assemblies is provided by making the shanks of the mixed oxide rod end caps larger in diameter than those on UO_2 rods, and correspondingly adjusting the penetrations on both upper and lower tie plates, such that it is impossible to mislocate a mixed oxide fuel rod into a UO_2 rod location. The worst remaining case of fuel rod mislocation is the possible loading of a high enriched UO_2 rod (2.87 w/o U-235) in place of the low enriche rod (1.59 w/o U-235) in the wide-wide corner of the assembly (See Figure 2). The consequences of this loading error are the same as were discussed in Facility Change Request No. 4 for Type III E fuel.

2. Fuel Assembly Misorientation

The analysis presented in Facility Change Request No. 4 also applies to thi submittal.

B. TRANSIENT ANALYSIS

A review of the previously submitted transient analyses [1][2][3] has been conducted to ascertain the effect of utili ng the Type IV mixed oxide fuel as reload fuel. To facilitate this evaluation, the behavior of an equilibrium cycle employing mixed oxide reload fuel has been compared to the behavior of an equilibrium cycle employing UO_2 reload fuel. Note that an equilibrium mixed oxide core serves to accentuate neutronic differences between the PuO_2 and UO_2 cores for transient analysis purposes. The results of this evaluation are presented below.

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(1) Amendment No. 65, Application for an Increase In Power Level, December 30, 1970.

(2) Amendment No. 69, Additional Information in Support of Facility Change Request No. 2, May 26, 1972.

(3) Facility Change Request No. 4, January 18, 1973.

1. Parameters Significant to Transient Results

Table V summarizes the reactor parameters calculated for the equilibrium cycles of UO_2 and mixed oxide reloads, and compares these parameters to those used in previous analyses. An inspection of the tabulated data indicates that the only potential point of concern is the larger negative void coefficient of reactivity which is calculated for the mixed oxide core. Differences in the remaining parameters are either in the conservative direction or are insignificant. The effects of these parameter differences on the results of various classes of transients are discussed in paragraph 2 below.

2. Effects on Transients

a. Pressurization Transients

Pressurization transients are characterized by forced decreases in void content, which result in a power level increase which is terminated either by a rod scram or by action of the negative reactivity coefficients. The magnitude of the pressure increase is determined both by the rate at which steam flow is reduced and by the magnitude of the power level increase. Sensitivity calculations have been performed for the two pressurization transients of major interest (turbine trip without bypass and Safety Valve Sizing transients) with the following results:

- Turbine Trip Without Bypass Transient

  Analyses of this transient are performed for end-of-cycle conditions where the shape of the rod scram reactivity insertion curve maximizes the magnitude of the power excursion and accompanying power spike. The sensitivity analysis was performed using the parameters given in Table V for end of cycle conditions for the equilibrium UO_2 and mixed oxide

TABLE V

PARAMETERS FOR TRANSIENT ANALYSIS

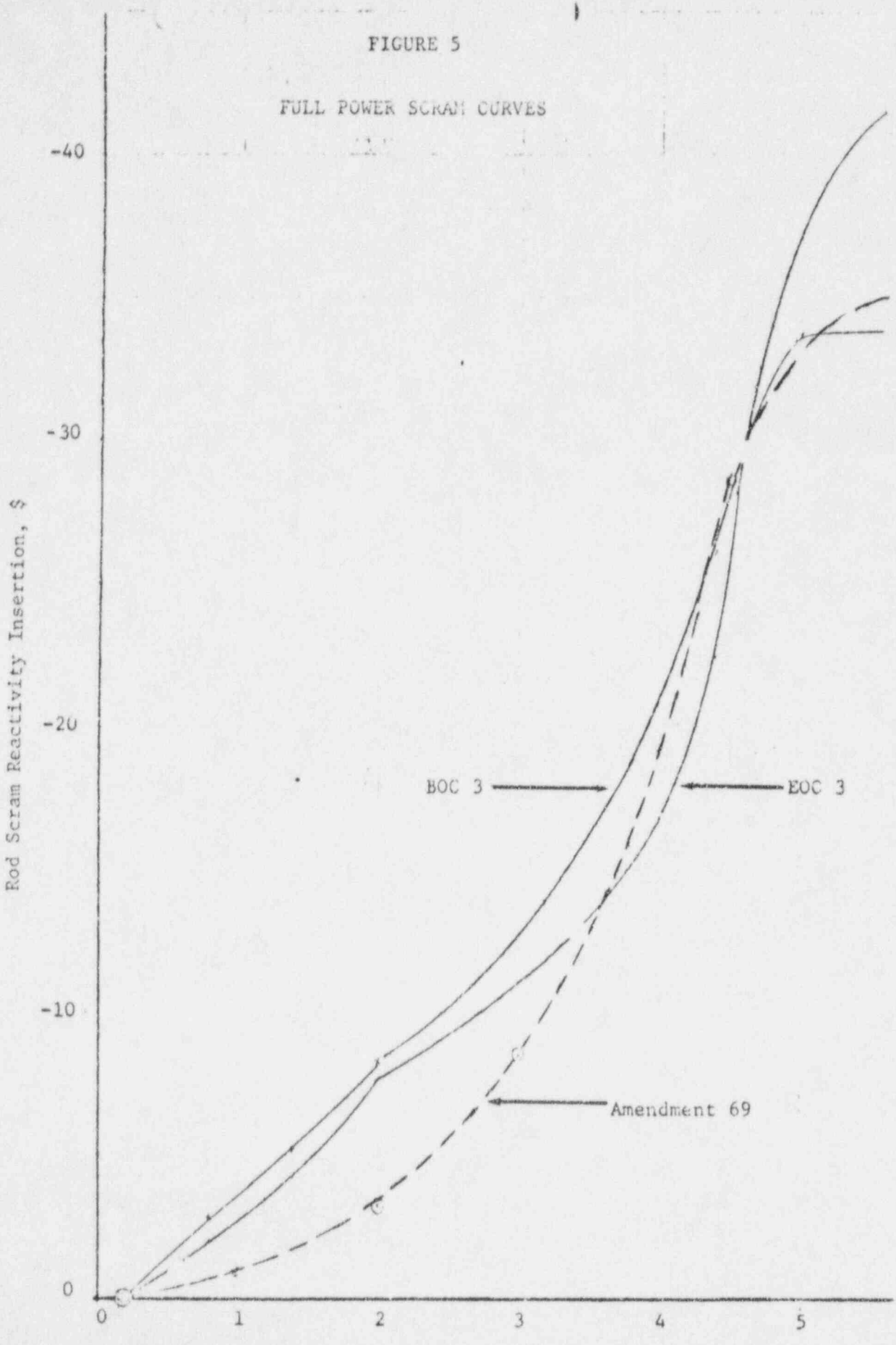
	Used In 1930 MWt Analysis, Amendment 65	Used In Pressurization Transient Re-analysis, Amendment 69	Previous Calculations				Current Calculations							
			Core I BOL	Core I @ 3000 MWD/T	Cycle 2 BOC	Cycle 2 EOC	Cycle 2 BOC	Cycle 2 EOC	Cycle 3 BOC	Cycle 3 EOC	Equilibrium UO_2 BOC	Equilibrium UO_2 EOC	Equilibrium MO BOC	Equilibrium MO EOC
$\bar{\beta}$, Delayed Neutron Fraction	.00643	.00547	(~.007)	.00643	.00605	.00547	.00560	.00493	.00563	.00492	.0053	.0047	.0045	.0036
ℓ^*, Neutron Lifetime, μsec	39.2	33.4	Not Stated	48.4	37.35	40.25	~40.6	40.2	40.6	39.7	--	--	--	--
Void Coefficient @ ~33% Voids														
$\Delta k/k/\%$ Void ($\times 10^4$)	-15.65	-11.47	-13.65	-13.4	-10.65	- 9.55	-10.1	- 9.3	-11.0	- 9.4	-11.9	-10.2	-12.8	-11
$\beta/\%$ Void/Initial Void %	- 7.96	- 6.84	- 6.84	- 6.87	- 5.81	- 5.76	- 5.95	- 6.23	- 6.45	- 6.30	- 7.41	- 7.16	- 9.39	- 8.64
Doppler Coefficient at ~33% Voids														
$\Delta k/k°F$ ($\times 10^5$)	- 1.19	- 1.39	- 1.23	N.C.	- 1.16	- 1.39	- 1.09	- 1.16	- 1.08	- 1.15	- 1.34	- 1.41	- 1.41	- 1.41
$\beta/°F$	- .166	- .254	- .176	N.C.	- .192	- .254	- .295	- .235	- .192	- .234	- .254	- .298	- .312	- .336
R-H Scram Curve						Figure 5			Figure 5	Figure	*	*	*	*

* These scram curves are as shown in Figure 5 for the EOC, but adjusted for differences in β according to:

$$\text{Reactivity Insertion } (\$)_T = \text{EOC 3 Reactivity Insertion } (\$) \times \frac{\beta_{EOC\ 3}}{\beta_T}$$

FIGURE 5

FULL POWER SCRAM CURVES



fuel cycles, including a 25% factor of conservatism on the void coefficients of reactivity. The calculated result of changing the parameters from the UO_2 EOC case to the mixed oxide EOC case is + 2 psi in the peak transient pressure, which is an insignificant difference.

- Safety Valve Sizing Transient

  Analyses of this transient are performed at beginning-of-cycle conditions. Since the incident assumptions include the absence of a control rod system scram, the result of the transient is largely controlled by the relative magnitu of the void and the Doppler coefficients of reactivity. Th coefficients are most unfavorable at beginning-of-cycle conditions. The sensitivity analysis was performed using the parameters given in Table V for beginning-of-cycle cond tions for the equilibrium UO_2 and mixed oxide fuel cycles, including a 25% factor of conservatism on the void coefficie of reactivity. The calculated result of changing the para- meters from the UO_2 BOL case to the mixed oxide BOL case is + 2 psi in the peak transient pressure, which is an insignificant difference.

b. Core Flow Reductions

Sensitivity calculations reported in Facility Change Request No. for these transients included the range of values for the Doppler coefficient of reactivity calculated for the equilibrium fuel cycles, and conservatively low values for the void coefficient of reactivity. Those results indicated an insignificant reduction in the MCHFR results from the core flow reduction transients.

c. Core Flow Increases

A core flow increase could result from the failure of a recirculation pump flow controller which results in a rapid flow increase in one loop. The transient is characterized by a reduction in the void fraction in the core, an increase in reactivity and hence power due to the decreased void volume, and a mild increase in pressure due to the power increase. The transient effectively represents shifting from one equilibrium point to another on the flow vs. power level control line. Sensitivity analyses indicate no significant difference between the UO_2 and mixed oxide fuel cycles in the results of this transient.

d. Loss of Feedwater and Excess Feedwater Flow

As indicated in Facility Change Request No. 4, the results of these transients are not significantly affected by the change in core coefficients of reactivity.

e. Rod Withdrawal Incident

An inadvertant rod withdrawal is characterized by a local insertion of reactivity through removal of a rod blade counterbalanced by a local reduction in reactivity due to increased voids and fuel temperature. As Tables III and V indicate, the negative void and temperature coefficients for both the equilibrium UO_2 and mixed oxide cores are larger than in Cycle 3, while the rod worths are essentially unchanged (all expressed in terms of $\Delta k/k$). Hence, the local power and core power transients for the equilibrium cores will be milder than for the Cycle 3 core for a given rod withdrawal increment. Hence, the results of the inadvertant rod withdrawal transient analysis presented in FCR No. 4 conservatively apply to the mixed oxide fuel cycle

VI ACCIDENT ANALYSIS

A. LOSS OF COOLANT ACCIDENT ANALYSIS

The consequences of a loss of coolant accident in the Oyster Creek reactor have been evaluated for Type IV fuel. The models used in this analysis comply with the AEC Interim Criteria. The methods and assumptions used in this analysis are described in Facility Change Request No. 4, except that the local power distribution used in the analysis has been changed to correspond to Type IV fuel at 4,000 MWD/T, which is the most limiting distribution for this analysis. This local power distribution is shown on Figure 6.

The results of the loss-of-coolant accident analysis for the design basis accident are:

Peak Clad Temperature:	2238 F
Local Peak Metal-Water Reaction	9 %
Core-Average Metal-Water Reaction:	.25 %

B. MAIN STEAM LINE BREAK ACCIDENT

The analysis of the main steam line break accident depends on the operating thermal-hydraulic parameters of the overall reactor such as the pressure, and the overall factors affecting the consequences, such as primary coolant activity. Insertion of reload fuel will not change any of these parameters so the results of the analysis discussed in Amendment 65(1) will not change.

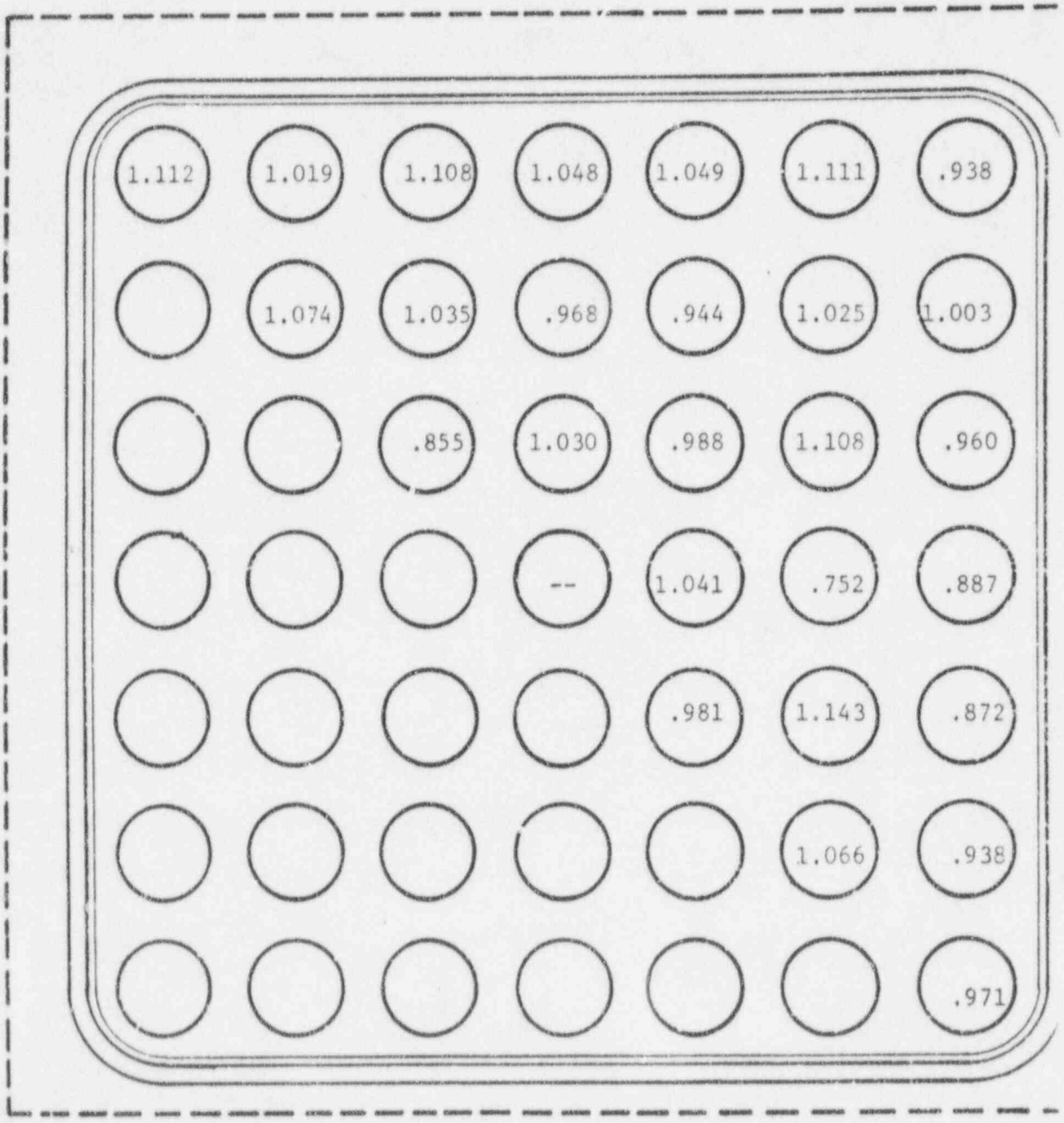
C. REFUELING ACCIDENT

The analysis of the refueling accident involves the mechanical damage caused by a fuel bundle falling back onto the top of the core while it is being removed, and the subsequent release of radioactive fission products. The severity of the consequences depends on the fission product inventory in the

FIGURE 6

LOCAL POWER DISTRIBUTION FOR LOCA CALCULATION

OYSTER CREEK TYPE IV
32% Void
4 GWD/MTM



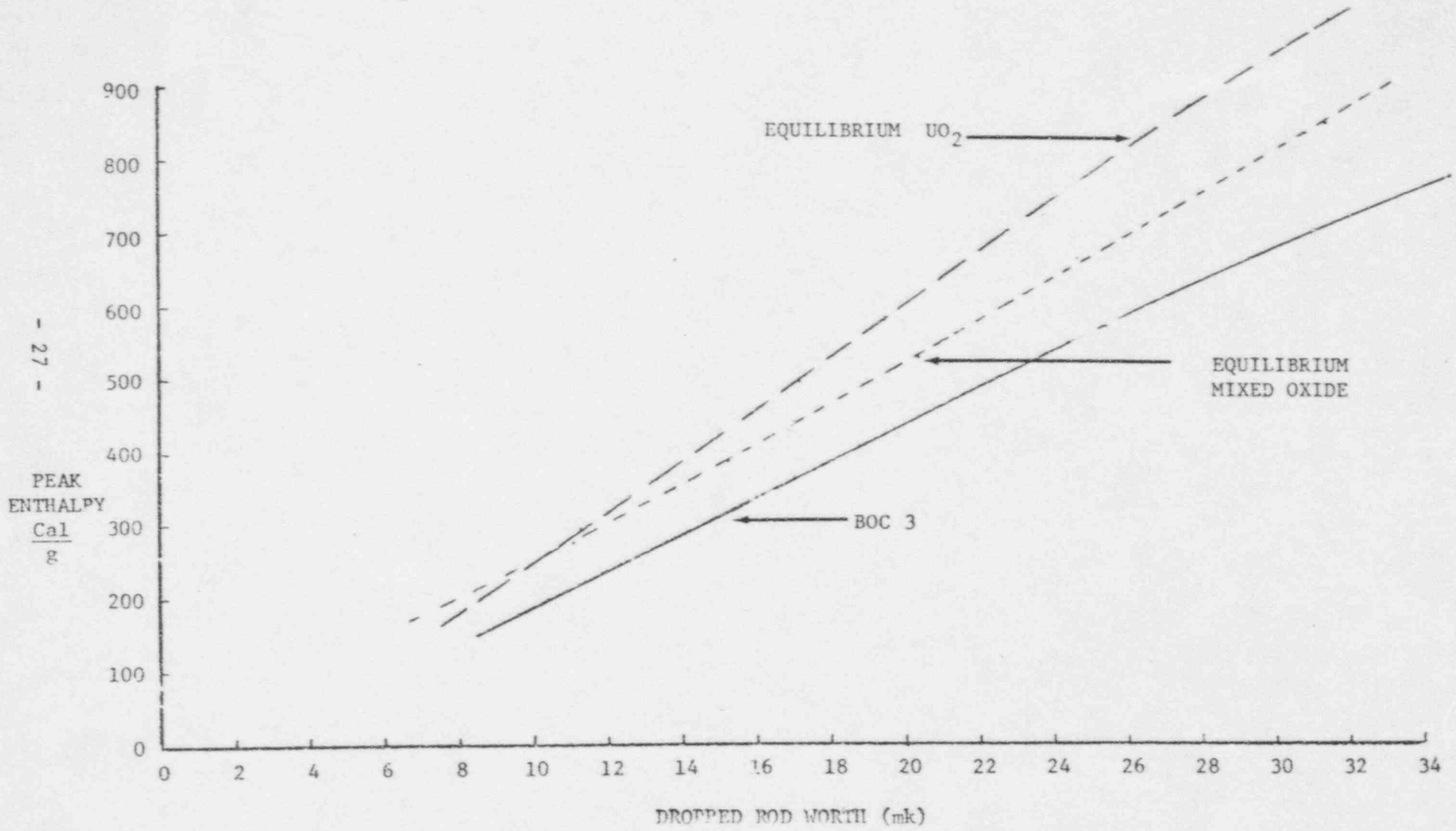
fuel and various factors affecting the amount and kind of releases to the atmosphere. The radiological effects of plutonium recycle are discussed in Section VII below. Based on those evaluations, it is apparent that utilization of mixed oxide fuel rather than UO_2 fuel would have no significant adverse affect on the consequences of a refueling accident, and that the previous assessment of this accident (FD & SAR Amendment 65) is conservatively valid for Type IV fuel.

D. ROD DROP ACCIDENT

The analysis of a rod drop accident under Cycle 3 core conditions was presented in Facility Change Request No. 4 where it was concluded that a large margin exists between the maximum in-sequence control rod worth (3.5 mk) and the dropped rod worth which would result in an enthalpy deposition of 280 calories/gram (~ 14 mk). Analyses have been performed to determine the consequences of this accident under equilibrium cycle conditions for both a UO_2 and a mixed oxide fuel cycle. The results of these analyses are shown on Figure 7 where they are compared to the results for Cycle 3 conditions reported in Facility Change Request No. 4. As the figure indicates, the calculated enthalpy deposition is higher for both equilibrium cycles than for Cycle 3. However, the conclusion reached in Facility Change Request No. 4 remains valid; the value of the dropped rod for which the peak fuel enthalpy deposition is 280 calories/gram is much higher than the maximum in-sequence rod worth (> 11 mk vs. 3.5 mk).

FIGURE 7

PEAK ENTHALPY vs. DROPPED ROD WORTH



VII OFFSITE RADIOLOGICAL EFFECTS CONSIDERATIONS

Offsite radiological effects resulting from the use of mixed oxide fuel, including differences in inventories of fission products and trans-uranium isotopes, could arise from either a severe reactor accident resulting in an overheated core (Section A, below), or from normal reactor operation with failed fuel which exposes the oxide fuel pellets to the reactor coolant (Section B, below). These effects are potentially different from those of urania fuel, due to the differences in fission product and heavy isotope inventories (Sections C and D, below). These effects are discussed in the following sections in terms of the differences resulting from the replacement of urania fuel with mixed oxide fuel. The analyses presented below are generic to Light Water Moderated and cooled power reactors. The difference in detailed reactor parameters and configurations could affect the details of the analyses, but will not alter the conclusion.

A. HYPOTHETICAL ACCIDENT EVALUATION

The offsite radiological effects of plutonium have been evaluated for a hypothetical accident where the barriers between all fuel pellets and the containment have been breached. The results of this analysis indicate that the offsite dose rates contributed by plutonium are negligible compared to the dose rates from fission products alone. Since the latter normally comply with 10 CFR 100 criteria for such an accident, it is concluded that plutonium-bearing fuel will also comply with these criteria.

These studies are based on measured values of the vapor pressure of 5 w/o PuO_2 in a PuO_2-UO_2 mixture. Table VI lists this vapor pressure and that of strontium oxide as functions of temperature.

Since core temperatures and core configurations during the hypothetical accident are not well defined, the basis chosen for an assessment of the effects of plutonium is a comparison between plutonium effects and typical fission product effects. For this purpose, plutonium is compared to iodine-131 and strontium. The relationship of the concentration of each material to the

Table VI

PuO_2 AND SrO VAPOR PRESSURES vs. TEMPERATURE

T, °C	PuO_2[4] P, atm	SrO[5] P, atm
1200	$\sim 6 \times 10^{-14}$	$\sim 2 \times 10^{-11}$
1600	6×10^{-10}	7×10^{-7}
2000	4×10^{-7}	5×10^{-4}
2200	4×10^{-6}	6.5×10^{-3}
2400	3.5×10^{-5}	6×10^{-2}
2600	2×10^{-4}	.35
2800	9×10^{-4}	> 1.0

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4 J. E. Battles, et. al., "A Mass Spectrometric Investigation of the Volatization Behavior of $(U_{0.8}Pu_{0.2})\ O_{2-x}$," Plutonium 1970 and Other Actinides, Nuclear Metallurgy, Vol. 17, Edited by W. N. Miner, October, 1970.

5 A. Classen and C. F. Veehemans, "Vapor Pressure Determination of BaO, S[illegible] and CaO and their Mixtures from Measurements of Evaporation," Z. Ph[illegible] Vol. 80, pp. 342-51 (1953).

concentration limits given in 10 CFR 20, Appendix B, Table II are used as a basis of comparison. The results of this comparative analysis are insensitive to the specific reactor design within broad limits. Conservatively, the results of an analysis of a reactor operating at 3000 MWt, with the core exposed to 35,000 MWD/MTM and a containment volume of 31,000 cubic feet, are assumed to apply and are discussed below. To estimate the quantities of I-131, strontium, and plutonium evolved, the "non-volatile" materials (Sr-90 and Pu) are assumed to reach equilibrium vapor pressures in the containment volume. Two hypothetical temperatures, 1600 C and 2300 C are chosen for illustrative purposes. To assure a large measure of conservatism in the conclusion, it is further assumed that the modest decontamination factors allowed for I-131 (plateout and rainout) also apply to the "non-volatile" materials (Sr-90 and PuO_2). This assumes that all three materials are released to the environment in the same proportions as they are present in the containment volume. Utilizin the above assumptions, the calculations indicate that:

- The entire core inventory of iodine-131 is released to the containment at both 1600 C and 2300 C.
- 0.0048% of the strontium is released to the containment at 1600 C, and all of the strontium is released at 2300 C.
- Only 2.6×10^{-7}% of the plutonium is released to the containment at 1600 C, and only 0.0034% of the plutonium at 2300 C.

The resulting activity concentration of the three materials in the containment volume are calculated to be:

Material	1600 C	2300 C.
I-131	99 ci/ℓ	99 ci/ℓ
Sr-89 & 90	7×10^{-4} ci/ℓ	14.5 ci/ℓ
Pu (all isotopes)	6.7×10^{-9} ci/ℓ	1.7×10^{-4} ci/ℓ

A measure of the biological importance of each of these materials may be obtained by relating the above concentrations to their respective 10 CFR limits and normalizing these ratios to unity for I-131. The results of this normalization are:

	Relative Biological Importance	
Material	1600 C	2300 C
I-131	1.0	1.0
Strontium	2×10^{-5}	.5
Plutonium	1×10^{-7}	3×10^{-3}

It may be seen that, for those hypothetical accidents which might approach the limiting guidelines in 10 CFR 100 with respect to doses to the thyroid from iodine-131 (300 rem), the doses from plutonium would be biologically insignificant. Further, any mechanisms which might selectively minimize the offsite radiological effects of the "non-volatile" fission products such as higher plateout and reinout factors or high efficiency filtering, would also act to further diminish the radiological effect of plutonium. Finally, the concentration of plutonium in a mixed oxide-fueled reactor core is only approximately a factor of two to three greater than that in a UO_2-fueled core; hence, in view of the already minor relative importance of the plutonium radiological effects, the impact of substituting a mixed oxide for a UO_2 fuel core is negligible.

B. PLUTONIUM RELEASE INTO PRIMARY COOLANT

Corrosion studies of UO_2 and PuO_2 fuels in high temperature water indicate little difference in the corrosion characteristics of the fuels.[6,7]

A direct experiment in the Engineering Test Reactor to measure the release of radionuclides from defected mixed oxide fuel has been performed.[8] The experiment utilized defected mixed oxide fuel (4 w/o PuO_2 in UO_2) which was pre-irradiated to about 2000 MWD/MTM, then irradiated at about 12 kw/ft while defected for over 1000 MWD/MTM (Cycle I) and then finally irradiated at about 20 kw/ft (center melted) for almost 1000 MWD/MTM (Cycle II). The 250 C coolant in the loop was sampled during the periods of defected operation and analyzed for radionuclide content. Table VII is a reproduction from the referenced report.[8] Several observations and interpretations of the data can be made:

1) The ratio of plutonium to uranium in the coolant appears to be approximately the same as in the pellets, although the plutonium concentrations were below the threshold of accurate measurement. This tends to confirm that the removal mechanism is erosion/corrosion and/or that the solubilities of the two oxides are approximately the same.

2) A comparison of selected isotopes (I-131, Cs-137, Pu-239, and U-238 to their respective 10 CFR 20 limits is shown below. It is apparent that the concentration of plutonium is very small

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6 M. Zambernard, Development of Plutonium-Bearing Fuel Materials. Progress Report, January 1 through March 31, 1963. NUMEC-P-104, Nuclear Materials and Equipment Corp., Apollo, Penn., March 31, 1963.

7 N. Zambernard, Development of Plutonium-Bearing Fuel Materials. Progress Report, April 1 through June 30, 1963. NUMEC-P-105, Nuclear Materials and Equipment Corp., Apollo, Penn., June 30, 1963.

8 M. D. Freshley, The Defect Performance of UO_2-PuO_2 Thermal Reactor Fuel, BNWL-SA-4138, Battelle Pacific Northwest Laboratories, November, 1971.

Table VII

LOOP COOLANT ANALYSES OBTAINED DURING CYCLE II AND CYCLE III OPERATION OF THE MIXED-OXIDE DEFECT EXPERIMENT (Ref. 8)

Element	Cycle II	Cycle III*
Xe-133	1.61×10^{5} d/s/ml	6×10^{4} d/s/ml
-135	7.67×10^{4} d/s/ml	2×10^{3} d/s/ml
Kr-85m	3.54×10^{3} d/s/ml	4×10^{2} d/s/ml
-87	2.71×10^{3} d/s/ml	1×10^{4} d/s/ml
-88	1.23×10^{4} d/s/ml	---
I-131	2.52×10^{3} d/s/ml	$1.6/10^{3}$ d/s/ml
-132	Present	Present
-133	4.49×10^{3} d/s/ml	2×10^{3} d/s/ml
-134	1.77×10^{3} d/s/ml	---
-135	2.58×10^{3} d/s/ml	---
Te-131	Present	----
-132	$< 1 \times 10^{3}$ d/s/ml	1×10^{3} d/s/ml
Cs136	---	$< 1 \times 10^{2}$ d/s/ml
-137	---	1×10^{2} d/s/ml
-138	1.16×10^{4} d/s/ml	5×10^{4} d/s/ml
Rb-88	---	Present
U-238	1.3×10^{-7} g/ℓ	3.5×10^{-7} g/ℓ
U-235	1.0×10^{-8} g/ℓ	2.5×10^{-8} g/ℓ
Pu-239	$< 1.0 \times 10^{-8}$ g/ℓ	$< 1 \times 10^{-8}$ g/ℓ

*The fission product analysis for the Cycle III sample is only semi-quantitative because of a malfunction of the counting equipment.

and that its relationship to its limiting concentration is far below those of the fission products. Fission products will, therefore, control the management and releases of radioactive wastes.

Isotope	Measured µCi/mℓ	10 CFR 20 Limit	Ratio
I-131	∿.05	6×10^{-5} (insol)	830
		3×10^{-7} (sol)	1.7×10^{5}
Cs-137	∿.003	4×10^{-5} (insol)	75
		2×10^{-5} (sol)	150
Pu-239	$< 6.2 \times 10^{-7}$	3×10^{-5} (insol)	< .02
		5×10^{-6} (sol)	< .12
U-238	$.84 \times 10^{-10}$	4×10^{-5}	2×10^{-6}

C. PLUTONIUM AND TRANS-PLUTONIUM ISOTOPIC CONSIDERATIONS

It is possible to assess in general terms the potential long-term effect of plutonium recycle in terms of the relative inventory of trans-uranium isotopes which would be available if plutonium were either a) stored or b) recycled. Calculations have been performed to identify the isotope buildup and burnout in both cases assuming a typical light water reactor environment. Table VIII gives the feed and discharge isotopic concentrations in typical UO_2 (3.1 w/o U-235) and mixed oxide (2.74 w/o Pu) fuel rods irradiated to an exposure of 30,000 MWD/MTM. The trans-plutonium isotopes which could contribute significantly to a long-term radiation hazard are included.

As a basis of comparison of the relative biological effect of recycling vs. not recycling plutonium, the radiological dose resulting from exposure to the trans-uranium isotopes contained in a unit quantity of each type of fuel has been computed. Values for Relative Biological Effectiveness,

TABLE VIII

FEED AND DISCHARGE ISOTOPICS

(Grams/Metric Ton Metal)

Isotope	1/2 life (yrs)	UO_2 Fuel		Mixed Oxide	
		In	Out	In	Out
Pu-238	89	--	102.7	433.1	680.9
-239	24,360	--	4,992.6	21,089.2	10,724.8
-240	6,760	--	1,986.8	8,389.7	8,262.9
-241	13	--	1,151.4	4,863.0	4,941.1
-242	3.79×10^5	--	395.5	1,669.4	2,902.1
Np-237	2.14×10^6	--	378.6		140.3
Am-241	458	--	22.7		245.2
-243	7,650	--	.36		5.59
Cm-242	.45	--	11.3		142.8
-243	32	--	.07		1.21
-244	18.1	--	15.0		335.6
U-235	7.13×10^8	31,000	8,818.1	6,849.9	3,334.0

fractional deposition in the bone (assumed to be the critical organ), and the effective half-life of each isotope were taken from ICRP Publication 2, 1959. Table IX shows the relative biological effectiveness of the quantities of trans-uranium isotopes given in Table VIII, normalized to unity for the plutonium isotopes discharged with UO_2 fuel:

TABLE IX

RELATIVE RADIOLOGICAL HAZARDS

Discharged UO_2 Fuel - Pu Isotopes	1.0
- Trans Pu Isotopes	.115
Charged Mixed Oxide Fuel - Pu Isotopes	4.224
Discharged Mixed Oxide Fuel - Pu Isotopes	4.862
- Trans-Pu Isotopes	1.917

These values must be recast to reflect the relative effect of recycling a unit quantity of plutonium, as follows:

1) If plutonium is not recycled (UO_2 fuel is charged):

The amount of plutonium stored which would otherwise be recycled would be;	4.224
The amount of trans-plutonium isotopes stored which were discharged with the above quantity of plutonium would be; (4.224 x .115)	0.460
The amount of plutonium generated in the unit quantity of UO_2 fuel at goal exposure would be;	1.000
The amount of trans-plutonium isotopes generated in the unit quantity UO_2 fuel at goal exposure would be;	0.11[illegible]
	5.799

2) If plutonium is recycled;

The amount of plutonium stored would be;	0.0
The amount of trans-plutonium isotopes stored would be;	0.460
The amount of plutonium discharged with the unit quantity of mixed oxide fuel at goal exposure would be;	4.862
The amount of trans-plutonium isotopes generated in the mixed oxide fuel at goal exposure would be;	1.917
	7.239

These results indicate that the total accumulation of plutonium and trans-plutonium isotopes, expressed in terms of their relative biological effectivenss, will be 25 percent greater if plutonium is recycled than if it is only stored.

It may be more valid to exclude from relative radiation hazards considerations those isotopes with quite short half lives (say, less than 100 years) on the basis that environmental concerns center on whether controls can be assured for several centuries, not merely years. If this revision is made, the relative radiation hazard in the recycle case is 24% less than in the non-recycle case. This reversal results from elimination of Pu-238, Pu-241, and the curium isotopes.

These results serve to place some perspective on the general effect of plutonium recycle with respect to the general risk of population exposure to the long-lived trans-uranium elements. While the specific results will vary with the actual fuel design chosen, it appears that introduction of plutonium recycle in light water power reactors does not have a significant effect in this area.

D. FISSION PRODUCT YIELD CONSIDERATIONS

The distribution of fission products is a function of the atomic mass of the fissioning isotope. The three fissile isotopes of interest are U-235, Pu-239 and Pu-241. The fission product yields for these three (and other) isotopes were measured by Lisman, et. al.[10] Their results are listed in Table X.

To provide an assessment of the importance of the changes in fission product distribution without complicating the analysis with assumptions about release mechanisms, reactor operating levels and times, fission product decay time, fuel feed isotopics, enrichment levels, and exposures, the following assumptions were used:

- The yields for each fissile isotope are considered separately.
- Biologically significant elements at each mass number are chosen for consideration without regard to half-life. Isotopes, half-lives, fractions deposited in each organ, and relative biological effectiveness are taken from ICRP Publication 2, 1959. Ingestion by inhalation of soluble compounds is assumed.
- Relative radiation doses are calculated on a unit fission basis to allow comparison of the effects of each fissile isotope.

The results of this calculation, in terms of relative lifetime dose, normalized to unity for U-235, are shown in Table XI.

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10 F. L. Lisman, et. al. Fission Yields of Over 40 Stable and Long-lived Fission Products for Thermal Neutron Fissioned ^{232}U, ^{235}U, ^{239}Pu, and ^{241}Pu, and Fast Reactor Fissioned ^{235}U and ^{239}Pu. Nuclear Science and Engineering, 42, 191-214, 1970.

Table X

FISSION PRODUCT YIELDS FROM U-235, Pu-239 and Pu-241

Mass No.	235	239	241
83	.529	.301	.200
84	1.01	.487	.353
85	1.33	.574	.387
86	1.95	.770	.601
87	2.57	1.0	.741
88	3.61	1.35	.954
90	5.93	2.09	1.53
91	5.92	2.52	1.82
92	5.98	3.02	2.23
93	6.37	3.95	2.90
94	6.45	4.50	3.33
95	6.51	4.86	3.92
96	6.26	5.12	4.33
97	5.92	5.64	4.76
98	5.83		
99	6.24		6.17
100	6.30		
101	5.08	6.50	5.94
102	4.21	6.65	6.32
104	1.83	6.61	6.80
106	.389	4.55	6.08
125	(.013)*	(.072)* .116	.042
131	2.86	3.60	3.15
132	4.27	5.09	4.64
133	6.76	7.18	6.71
134	7.73	7.20	8.06
137	6.32	6.74	6.60

*Values taken from ANL-5800, Second Edition.

Table X (Cont.)

Mass No.	235	239	241
138	6.83	5.40	6.37
140	6.35	5.61	5.86
141	5.53		
142	5.90	5.04	4.80
143	5.92	4.48	4.48
144	5.45	3.78	4.13
145	3.89	3.03	3.19
146	2.97	2.49	2.68
147	2.14	2.15	2.22
148	1.70	1.70	1.89
149	1.01	1.24	1.43
150	.64	.965	1.16
151	.409	.811	
152	.213	.581	.725
154	.056	.270	.370

Table XI

	U-235	Pu-239	Pu-241
Whole Body	1.0	.73	.74
Thyroid	1.0	1.18	1.07
Bone	1.0	.50	.47

The calculations show that the thyroid dose is dominated by the iodine isotopes. Since the mix of the iodine isotopes ingested is affected by many variables, the iodine isotope affected the most is considered as the limiting case. This isotope is I-131, whose yields from Pu-239 and Pu-241 fissions are 26% and 10% higher, respectively, than the yield from U-235 fission.

The calculations show the bone dose is dominated by the strontium isotopes and cerium-144. If the dose were due entirely to Sr-90 ingestion, the values for Pu-239 and Pu-241 would be .35 and .26 normalized to unity for U-235. If, on the other hand, the dose were due entirely to ingestion of cerium-144, the values for Pu-239 and Pu-241 would be .69 and .76.

Recognizing that the fission product inventory in an actual operating reactor core will be a mixture of yields from all the fissioning isotopes, and that the variables of power level, exposure, operating time, decay time, and release and dispersion mechanisms can result in very large variations in the relative quantities of the various fission products actually ingested, it can be concluded that any actual dose variations introduced by changing from UO_2 to mixed oxide fuel will be insignificant.