Consumers Power Company

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

December 22, 1969

Regulatory File Cy.

Dr. P. A. Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, DC 20545

Re: Docket 50-155
DPR-6 ZEK
Proposed Tech Spec
Change 19

Dear Dr. Morris:

Attention: Mr. D. J. Skovolt

Transmitted herewith are three (3) executed and thirty-seven (37) conformed copies of a request for a change to the Technical Specifications of License DPR-6, Docket No 50-155, issued to Consumers Power Company on May 1, 1964 for the Big Rock Point Nuclear Plant.

This proposed change (No 19) will enable Consumers Power Company to insert into the reactor at Big Rock Point a fuel design designated as "EEI-UO_2-PuO_2", which will permit the irradiation of plutonium-uranium mixed oxide fuel. The purpose of this irradiation is to provide needed data on the operating characteristics of mixed oxide fuel with a statistically significant number of fuel rods.

It is our intention to insert "EEI-UO_2-PuO_2" fuel into the Big Rock Point Reactor during our next refueling outage which is currently scheduled for February 1970. We would, therefore, be most appreciative of an expeditious handling of this Request for a Technical Specifications Change so that we might receive approval before February 1, 1970. We recognize that this is a contracted schedule for a Technical Specifications Change. By way of explanation, we would like to point out that there are four parties involved in the various contract negotiations - USAEC, EEI and two utilities. It was easier to resolve the technical issues than the contractural issues.

Yours very truly,

Gerald J. Walke

Nuclear Fuel Management Administrator

GJW/dmb

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Consumers Power Company

Docket No. 50-155

Request for Change to the Technical Specifications

License No. DPR-6

For the reasons hereafter set forth, it is requested that the Technical Specifications of License DPR-6 issued to Consumers Power Company on May 1, 1964, for the Big Rock Point Plant be changed as follows:

A. In Section 5.1.5a, change to read as follows:

"Enrichment of Fuel", approximate weight percent U-235 from 2.6 to 5.2, inclusive. Approximate weight percent of plutonium (fissile Pu-239 and Pu-241) 1.0 to 10 in normal (0.7 w/o U-235) UO_2.

B. In Section 5.1.5b, change to read as follows:

Total nominal weight of UO_2 plus UO_2-PuO_2 in 84 bundles.

C. In Section 5.1.5 add Figure 5.9.

D. In Section 5.1.5, replace the present table of fuel bundle parameters with the following table:

5.15 (Cont'd)

General	FUEL BUNDLES								
					Research and Development				
	Original (A)	Reload (B&C)	Reload (E)	Reload (E-G)	"D" Fuel	Centermelt Intermediate	Centermelt Advanced	"Modified E-G"	EEI UO_2-PuO_2
Geometry, Fuel Rod Array	12 x 12	11 x 11	9 x 9	9 x 9	11 x 11	8 x 8	7 x 7	9 x 9	9 x 9
Rod Pitch, Inches	0.533	0.577	0.707	0.707	0.580	0.807	0.921	0.707	0.707
Standard Fuel Rods per Bundle	132	109	74	70	109	36	29	52	0
Special Fuel Rods per Bundle	12[1]	12[2]	7[3]	11[3,5]	12	28[4]	20[4]	29[7]	81[8]
Spacers per Bundle	3	5	3	3	7	5	5	3	3
Fuel Rod Cladding									
Material	304SS	Zr-2	Zr-2	Zr-2	304SS, Zr-2 Inconel 600 and/or Incoloy 800	Zr-2	Zr-2	Zr-2 with various initial mechanical properties	Zr-2
Standard Rod Tube Wall, In.	0.019	0.034	0.040	0.040	0.010 to 0.030 Inclusive	0.035	0.040	Zr-3Nb-1Sn 0.040	--
Special Rod Tube Wall, In.	0.031	0.031	0.040	0.040	0.010 to 0.030 Inclusive	0.035	0.040	0.040	0.040
Fuel Rods									
Standard Rod Diameter, In.	0.388	0.449	0.5625	0.5625	0.425	0.570	0.700	0.5625	--
Special Rod Diameter, In.	0.350	0.344	0.5625	0.5625	0.320	0.570	0.700	0.5625	0.5625
Fuel Stacked Density, Percent Theoretical	94 ± 1	94 ± 1 Pellet 85 Powdered	90-95 Pellet[7]	94 Pellet[6,7]	90-95, Inclusive	94 Pellet 85 Powder	94 Pellet 85 Powder	94 Pellet[6]	82
Active Fuel Length, Inches									
Standard Rod	70	70	69.75	70	68 to 70	66-67.3	65-66.3	70	70
Special Rod	59 (Corner)		64.6 Central	64.9 Central				64.9 central, 68.6 Removable	
Fill Gas	Helium	Helium	Helium	Helium	Helium	Helium	Helium	Helium	Helium

1 Four special fuel rods at bundle corners are segmented.
2 Reload B, C, E and E-G fuel bundles may contain (in the corner regions of the bundle) four Zr-2 tubes having encapsulated cobalt targets sealed within.
3 Reload E and E-G fuel bundles have a special central fuel rod to which the bundle spacers are fixed. In addition, two of the interior bundle fuel rods are removable and may contain UO_2-PuO_2 fuel.
4 Special rods have depleted uranium.
5 In addition to special rods for reload E, reload E-G has four gadolinia containing rods.
6 With 3% dishing on selected rods.
7 UO_2-PuO_2 fuel rod stack density will vary from 74 to 92 percent theoretical by using annular, dished, or non-dished pellets in selected rods.
8 64 UO_2-PuO_2 rods similar to standard UO_2 rods, 4 removable PuO_2 rods, 8 gadolinia containing rods, 4 cobalt corner rods and 1 empty (water-filled during operation) spacer rod.

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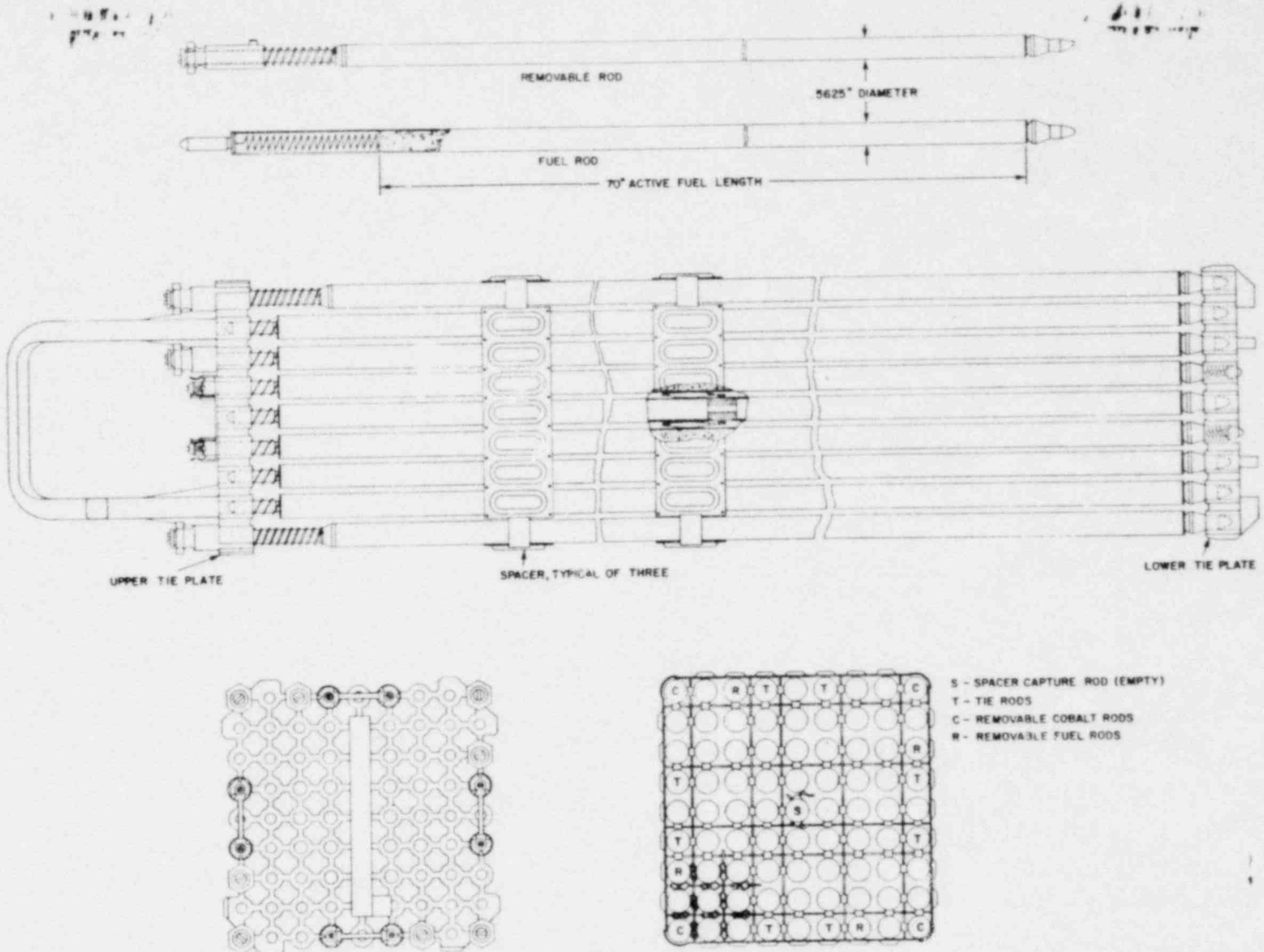

Figure 5.9 - Big Rock Point "EEI-UO_2-PuO_2" Fuel

II. Discussion - EEI UO_2-PuO_2 Bundles

A. Program Description

The EEI Program for the thermal reactor utilization of plutonium includes a test of three PuO_2-UO_2 containing, prototype bundles. The program objective is to design and test bundles which are interchangeable with regular Big Rock Point reactor bundles. These bundles are intended to demonstrate behavior and performance lifetime of PuO_2-UO_2 fuel bundles relative to UO_2 fuel.

B. Fuel Description

The EEI Bundle, like the "E-G" bundles, are designed to operate for four cycles and achieve an average burnup of 20,000 MWD/T. The design has five different types of plutonia rods. Four types are used to provide an acceptable power distribution and the fifth type provides a test of 80 percent fissile plutonium. Four cobalt rods with 35 gm Co/ft were retained for consistency with the "E-G" design. One spacer-capture tube will be filled with water at the center of the assembly. Eight UO_2-Gd_2O_3 rods augment control in a manner which matches the "E-G" design.

Eight removable rods are included in the design - four cobalt corner rods and four plutonia-containing fuel rods. The performance of the fuel will be monitored through examination of the removable fuel rods.

The bundle design is physically the same as the Reload "E" and "E-G" fuel. The only differences are:

1, four removable fuel rod positions are used (instead of two).

2, the central spacer rod contains no fuel and is perforated to permit water ingress.

3, eight gadolinia rods are included (instead of four) to match "E-G" poison reactivity control.

The enrichment distribution and local peaking factors are arranged so that established "E" and "E-G" Technical Specifications apply.

The position and number of gadolinia containing fuel rods has been changed as their reactivity worth is affected by the presence of plutonium. The gadolinia-containing rods do not contain plutonium.

Figure 5.9 and Figure 1 shows that fuel rod types, positions, and the enrichment distribution within a bundle. Four plutonium enrichments were selected to give adequate power distribution. The fifth plutonium containing fuel rod type contains plutonium of which 80 percent is fissile. The other four plutonium containing rod types contain about 90 percent fissile plutonium. The 80 percent fissile plutonium is deployed in four of the removable rod locations.

The non-plutonium containing rods, i.e., four cobalt, eight gadolinia, and the spacer rod, are mechanically identical to "E-G" design except that the spacer rod is empty and perforated.

The plutonium containing rods are also mechanically identical to "E-G" UO_2 containing rods. The PuO_2-UO_2 containing rods are identified by serial numbers on the lower end plug.

The PuO_2-UO_2 rods all contain cold pressed and sintered fuel pellets of annular design prepared from mechanically blended, ceramic grade UO_2 and PuO_2 powders. The annular hole is 0.150 inch diameter and the fuel matrix density is 92 percent. The only rod-to-rod variation is the plutonium enrichment, which is identified by varying the upper end plug diameter.
The thermal performance of this fuel will be similar to low density UO_2 fuel except that the annular feature causes lower fuel center temperatures relative to solid pellets of the same density. The plutonia fuel in all three bundles will operate well below melting at 122 percent overpower (500,000 BTU/hr-ft^2). The peak fuel temperatures at 500,000 and 410,000 BTU/hr-ft^2 are 4606°F and 3840°F, respectively. Since all fuel is 92 percent dense, the thermal conductivity integral has been reduced from the "E-G" standard. The corrected integral and equation are:

$$\int_{0^{\circ}C}^{2805^{\circ}C} K dt = 85.5 \text{ w/cm.}$$

<u>or</u>

$$K = \frac{3559.9}{602.61 + T} + 6.38 \times 10^{-12} (T + 460)^3 \text{ w/cm}$$

This corrected integral was derived for low density UO_2. Previous submittals (13) have documented the observation that UO_2-PuO_2 fuel containing small amounts of PuO_2 has essentially identical thermal performance.

C. Nuclear Design

The nuclear characteristics of the PuO_2-UO_2 bundles were calculated using standard GE nuclear methods. The enrichments were selected to give the power distribution shown in Figure 2. The peak rod power is purposely located in the removable fuel rod positions. The highest local peaking factor is 1.287 which is less than the 1.3 peaking on the plutonia rods in the "E-G" bundles. The local power distribution becomes less peaked with exposure as illustrated by Figure 2.

The reactivity values and power coefficients for the "E-Pu" design are shown in Table I. These coefficients are essentially the same with the exception of the void coefficient. Insertion of only three bundles will have an insignificant effect on the core void coefficient.

The isotopic content of the plutonium used in these bundles is as follows:

	"80%" Fissile	"90%" Fissile
Pu-238	0.268	0.104
Pu-239	75.356	86.919
Pu-240	18.238	10.162
Pu-241	4.956	2.532
Pu-242	1.182	0.283

D. Thermal Hydraulic Analysis

The thermal-hydraulic characteristics of the "E-Pu" bundles are essentially identical to the reload E-G fuel with two plutonia fuel rods in the removable rod positions. The local peaking factor was reduced to 1.287 in the "E-Pu" design as compared to 1.3 in the previous plutonia fuel rods now operating in "E-G" bundles. If necessary, these bundles will be placed in core positions that have radial power factors similar to the sixteen bundles now containing plutonia. The resultant thermal-hydraulic performance provides additional margin from the minimum critical heat flux ratio (MCHFR) limit, 1.5 at 122 percent overpower, due to the reductions of water quality in the bundle.

Core thermal-hydraulic analyses have been performed on predicted core configurations which indicate that all license limits will be met. During the refueling outage, these analyses will be performed on the finally-selected core configuration.

E. Special Handling Procedures

The three bundles will be shipped to Big Rock Point in a regular RA-1A container which is being licensed separately. Each bundle will be enclosed in a sheetmetal container which provides secondary containment during shipment. These containers will not be opened until they are inside the Big Rock Point containment vessel. Once removed, the bundles will be handled in an identical fashion to UO_2 fuel.

F. Accident Analysis

1. Reactivity Excursion Analysis

a. Postulated Reactivity Accidents

The Big Rock Point reactor operates with one specified rod withdrawal pattern. The rods are grouped in banks of two or more; all the rods in a bank are withdrawn together, with a procedural limit of two notches between any two rods in a bank. This sequencing prevents large rod worths; however, an operator error or series of errors can result in larger worths. The possible rod drop situations and rod strengths when the core is critical and at hot standby are:

Case 1: In-sequence potential of .008 Δk for drop from full-in position to drive position.

Case 2: In-sequence potential of .021 Δk for drop from full-in to full-out.

Case 3: Out-of-sequence potential of less than .021 Δk for drop from full-in to full-out.

Case 4: Maximum theoretical worst case of about .045 Δk.

Case 1 requires the following equipment malfunctions and operator error:

a) Rod becomes uncoupled from drive.

b) Drive is withdrawn (in-sequence), but blade hangs up temporarily. Operator does not notice that blade is not following.

c) Rod then unexpectedly releases and drops from full-in to position of the drive due to gravity.

Case 2 requires an additional operator error of withdrawing the drive completely rather than concurrent with the bank.

Case 3 consequences are less than those for Case 2.

C se 4 is considered hypothetical as it requires still further compounding errors beyond those enumerated above.

Case 2 at the hot standby condition was used for this analysis. These are the same conditions used by DRL for their analysis of the centermelt fuel (1).

b. Kinetics Calculations

The most important parameters in a nuclear excursion kinetics calculation are:

1) Quantity of reactivity insertion

2) Rate of reactivity insertion

3) Specific power distribution

4) Doppler coefficient

5) Resonance neutron flux distribution

6) Initial power

The only significant difference between the "current"* core and the "EEI Plutonium"** core is in the specific power distribution. The plutonium bundles have the same power producing capability as standard reload fuel and peaking factors that are very similar to the standard reload fuel. However, the plutonium fuel is of an annular design which reduces the mass of fuel that contains the energy generated during a transient. The effect is to raise the plutonium fuel energy density in any given accident by 13%. The effects on mass of fuel above given energy levels are shown below:

.021 Δk Rod Drop at Hot Standby

	"Current" Core	"EEI Plutonium" Core
Peak Enthalpy (cal/gm)	450	450
Mass of Fuel (kg) above:		
425 cal/gm	1.0	1.0
330 cal/gm	26	26
265 cal/gm	37	49
230 cal/gm	58	67

As can be seen there has been an increase in the mass of fuel above 265 cal/gm and the mass of fuel above 230 cal/gm. It should be noted that these increases will occur only if plutonium is loaded immediately adjacent to a centermelt bundle. If all of the EEI plutonium bundles are loaded next to a centermelt bundle, the figures above would still apply.

c. Primary System Integrity

As discussed at length in previous applications for this plant, the integrity of the primary system depends upon the severity of any steam explosion. The severity of a steam explosion depends upon the following factors:

1) Time of fuel failure

2) Mechanism of fuel failure

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* Currently licensed core

** Current core containing EEI plutonium bundles

3) Amount of fuel failed

4) Energy in the failed fuel

5) Heat transfer rate to the coolant

6) System geometry

As has been shown in previous applications a severe steam explosion will result only if there is a significant quantity of promptly dispersed fuel in the moderator. For material to be promptly dispersed it must attain an energy density on the order of 425 cal/gm or more. The above table demonstrates there is little material in this range for all considered conditions.

A large quantity of data has been obtained recently in the SPERT IV Capsule Driver Core (2-8). These data and earlier data indicate that fuel subjected to a transient energy deposition of 275 cal/gm or less remains intact (is not dispersed) after the transient. This also applies to fuel that has significant burnup (even though the cladding may fail). This is consistent with the latest calorimetric data for UO_2 (9-10) which indicates incipient melting occurs at an energy level of 269 cal/gm. Recent tests with physically blended mixed-oxide fuels have given no indication that this type of fuel behaves differently from conventional uranium fuels (11-12). The results of tests run at 225 and 274 cal/gm with the mixed oxide fuel were virtually identical to results obtained with uranium fuels tested at these levels.

In the previous license for plutonium fuel (13), the above information was not available and more conservative assumptions were made as to failure threshold. In light of the new test data, a conservative threshold for dispersal of mixed oxide fuels, as with urania fuels, is 265 cal/gm, as used in the supporting evidence for Change 18 to the Big Rock Point Technical Specifications, the same as uranium fuels. This analysis was based on that fact.

Even if one promptly dispersed all of the fuel above 265 cal/gm, the energy in the dispersed fuel would amount to only 61.5 MW-sec. This is below the 64 MW-sec that was considered tolerable in the DRL evaluation of the centermelt license. An evaluation of the consequences calculated by DRL for a 64 MW-sec deposition indicates that they are conservative by approximately two orders of magnitude.

As evaluated in the license application for Change 17 to the Big Rock Point Technical Specifications, the bone dose at the site boundary does not change due to the addition of plutonium to the core. This is so because plutonium is a non-volatile solid and the fuel vaporizations must occur to release non-volatile solids. However, none of the plutonium is calculated to vaporize as a result of the postulated .021 Δk rod drop accident. Nor is there calculated to be any vaporization in the case of a complete core meltdown.

d. Conclusions

It is concluded that the results of a postulated reactivity accident are slightly more severe in the "EEI Plutonium" core than in the "current" core. However, the results are still within an envelope considered acceptable in granting the license for the "Current Core". It is also concluded that there is no danger of breaching the primary system due to a credible reactivity accident with either core loading.

2. Loss of Coolant

The loss-of-coolant accident was discussed at length in conjunction with Change 14 which allowed insertion of reload "E" fuel. The addition of these bundles to the core will not increase the severity of the postulated accident. As mentioned above, in discussion of core thermal hydraulics, these assemblies will be placed in core location with lower power factors in order to readily meet thermal limits. In addition, the annular fuel will operate at a lower bulk average fuel temperature relative to solid pellet fuel for a given linear power. At full power (410,000 BTU/hr-ft^2) the peak fuel temperature in annular fuel is 3862°F compared to 4400°F for solid fuel and the fuel volume is 10 percent less so less specific heat is available in the fuel. The results of any postulated LOC accident will be less severe because of the reduced bundle stored energy.

III. Conclusions

Based on the above analyses and comparisons with E and E-G fuel, the following conclusions concerning the EEI UO_2-PuO_2 fuel bundles are made:

1. Fuel rod and bundle mechanical design is essentially identical to E/E-G.

2. The local power factor is slightly higher for some UO_2-PuO_2 rods than the UO_2 rods in the E/E-G design. The local power factor is slightly lower than UO_2-PuO_2 rods previously inserted in the core. The plutonium bundles will be located in radial positions so that the peak rod power will not exceed the design peak power for the E/E-G fuel.

3. Peak fuel temperatures in these bundles will be less than the solid and dished PuO_2-UO_2 pellet containing rods previously inserted in the core.

4. The 0.150 inch annulus selected for these pellets is conservative in that a 0.200 inch annulus has shown good structural integrity by previous analysis.(1)

5. The results of a postulated reactivity accident are slightly more severe in the core loading with these three EEI bundles. There is no danger of breaching the primary system due to a credible accident with this core loading.

6. The bone dose at the site boundary after a postulated accident does not change due to the addition of these PuO_2-UO_2 containing bundles.

7. The severity of a loss-of-coolant accident is probably less in this core because of the lower heat content in annular fuel.

Based upon the above considerations, we have concluded that the use of three E-Pu fuel bundles in the Big Rock Point reactor does not present a significant change in the hazardous considerations described or implicit in the Final Hazards Summary Report.

CONSUMERS POWER COMPANY

By HRWall
Vice President

Date: December 22, 1969

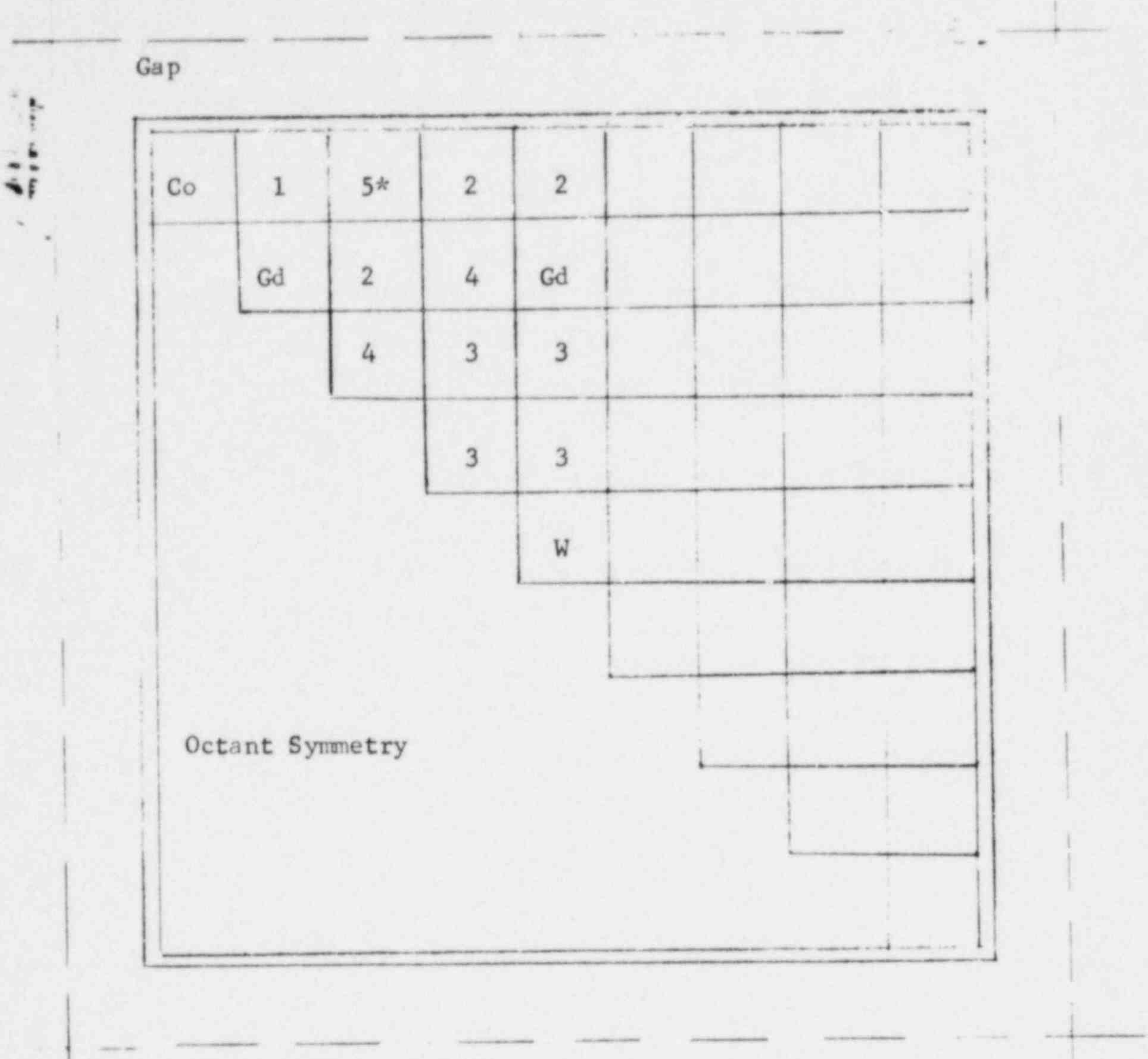
Sworn and subscribed to before me this 22nd day of December 1969.

Grace R. Warner
Notary Public, Jackson County, Michigan
My Commission Expires January 15, 1972

Figure 1

BIG ROCK POINT - EEI PHASE II

EPU BUNDLE DESIGN



*Special, removable rod in four places

# Rods	Type	w/o Pu, Total	Pu Fissile Fraction	U	Comments
8	1	1.624	.90	Natural	150 Mil I.D. Annulus
20	2	2.550	.90	Natural	150 Mil I.D. Annulus
20	3	9.072	.90	Natural	150 Mil I.D. Annulus
12	4	5.500	.90	Natural	150 Mil I.D. Annulus
8	5	2.551	.80	Natural	150 Mil I.D. Annulus
8	Gd	-----	---	3.4%	1.0 w/o Gd_2O_3
4	Co	-----	---	----	35 gm/ft Cobalt
1	W	-----	---	----	Saturated water rod

Fuel Density: 92% Theoretical

Figure 2

BIG ROCK POINT - PLUTONIUM BUNDLE DESIGN

Hot Operating (25% Voids-No Control)

Power Distribution (Norm to 76 Rods)

Top - 0 Days
Bottom - 311 Days @ Rated Power

Co	1.231 1.058	1.287 1.162	1.218 1.102	1.186 1.093				
	.431 .851	.816 .848	1.097 1.137	.341 .574				
		.929 .961	.989 1.022	.938 .981				
			.920 .915	.980 .966				
				W				

TABLE I

COMPARISON OF PRINCIPAL CALCULATED NUCLEAR CHARACTERISTICS OF "E-PU", AND RELOAD "E" AND "E-G"

Reactivity (k_∞)

	"E-Pu"	"E-G"	"E"
68°F	1.160	1.208	1.268
572°F, 0 voids	1.168	1.203	1.280
572°F, 25% voids	1.158	1.183	1.262

Temperature Coefficient; $\Delta k_{eff}/k_{eff}$ per °F @ 77°F

	"E-Pu"	"E-G"	"E"
Start of cycle	$+0.30\times10^{-4}$	$+0.27\times10^{-4}$	$+0.38\times10^{-4}$

Void Coefficient; $\Delta k/k$ per unit void within channel

	"E-Pu"	"E-G"	"E"
Cold (68°F)	-0.050	-0.08	-0.07
Hot (572°F)	-0.084	-0.12	-0.11

Doppler Coefficient $\Delta k_{eff}/k_{eff}$ per °F

Fuel Temp.	Moderator	"E-Pu"	"E-G"	"E"
68°F	68°F-0 voids	-1.35×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
1323°F	572°F-0 voids	-1.05×10^{-5}	-1×10^{-5}	-1×10^{-5}
1323°F	572°F, 25% voids	-1.25×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}

References

1. "Safety Evaluation by the Division of Reactor Licensing, Docket No. 50-155, Consumers Power Company, Proposed Amendment No. 1".

2. IDO-ITR-100, "Transient Irradiation of 1/4 Inch O.D. Stainless Steel Clad Oxide Fuel Rods to 570 cal/g UO_2", October, 1968.

3. IDO-ITR-101, "Transient Irradiation of 0.466 Inch O.D. Stainless Steel Clad Oxide Fuel Rods to 300 cal/g UO_2", November, 1968.

4. IDO-ITR-102, "Transient Irradiation of 1/4 Inch O.D. Zircaloy-2 Clad Oxide Fuel Rods to 590 cal/g UO_2", November, 1968.

5. IDO-ITR-103, "Transient Irradiation of .3125 Inch O.D. Zircaloy Clad Oxide Fuel Rods to 450 cal/g UO_2", January, 1969.

6. IDO-ITR-104, "The response of UO_2 Fuel Rods to Power Bursts, 9/16 Inch O.D., Pellet and Powder Fuel, Zircaloy Clad", April, 1969.

7. In-1302 (IDO-ITR-106), "The Response of UO_2 Fuel Rods to Power Bursts, Detailed Tests on 5/16 Inch O.D., Powder Fuel, Zircaloy Clad Rods", June, 1969.

8. In-ITR-107, "Behavior of 5 Inch Long, 1/4 Inch O.D., Zircaloy-2 Clad Oxide Fuel Rods Subjected to High Energy Power Bursts", August, 1969.

9. R. A. Hein, P. N. Flagella; "Enthalpy Measurement of UO_2 and Tungsten to 3260°K", Annual Meeting of Am. Cer. Soc., April 20-25, 1968.

10. ANL-7527 "Argonne National Laboratory, Reactor Development Program Progress Report, December, 1968", January 29, 1969.

11. D. L. Fischer, <u>et al</u>.; "Nuclear and Fast Transient Aspects of Plutonium Particle Size in Thermal Recycle Fuel", Transaction, ANS Annual Meeting, June, 1969, Volume 12, No. 1.

12. R. L. Johnson, SPERT, Personal Communication to R. W. Friis, General Electric, San Jose.

13. Change No. 17 to Technical Specification for Big Rock Point.

POOR ORIGINAL

FROM:
Consumers Power Company
Jackson, Michigan 49201
Gerald J. Walker

TO:
Dr Peter A. Morris

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CONCURRENCE ☐ COMMENT ☐

DATE ANSWERED: BY:

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DESCRIPTION: (Must Be Unclassified)
Ltr trans the following:

ENCLOSURES:
PROPOSED CHANGE REQUEST NO. 17, notarized 12-22-69 to Tech Specs to auth insertion into reactor a fuel design designated "EEI-UO_2-PuO_2 which will permit irradiation of plutonium-uranium mixed oxide fuel........
(40 cys rec'd)

REMARKS:

REFERRED TO	DATE	RECEIVED BY	DATE
Ziemann w/9 cys for ACTION	12-29-69		
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