Consumers Power Company

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

February 11, 1969

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

Re: Docket 50-155
DPR-6 ZEK

Dear Dr. Morris:

Attention: Mr. D. J. Skovholt

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.

The proposed change (No 17) will enable Consumers Power Company to insert into the reactor at Big Rock Point fuel bundles having one or two removable fuel rods containing plutonium oxide. The plutonium fuel will be irradiated under a joint program sponsored by the Edison Electric Institute, General Electric Company, the USAEC and Consumers Power Company. This specific irradiation is one phase of a broader program which has, as its main objective, the demonstration of the economic and technical feasibility of utilizing plutonium in light-water reactors. This specific experimental program includes the design, fabrication and irradiation of a significant number of fuel rods and bundles in an operating reactor. The program is aimed at obtaining data on the performance of a reference plutonium fuel form to provide design information for a reload core program in a commercial reactor.

The Joint Committee on Atomic Energy and the Atomic Energy Commission, along with the whole nuclear power industry, have expressed great interest in an experimental plutonium recycle program. This program is intended to be directly responsive to that interest and was brought to this point with concerted effort over a contracted time schedule. It is our intent to insert approximately sixteen (16) plutonium bearing fuel bundles into the Big Rock Point reactor during our next refueling outage, which is currently scheduled for April, 1969. We would, therefore, be

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most appreciative of an expeditious handling of this Request for a Technical Specification Change so that we might receive approval before April 1, 1969.

Yours very truly,

R. L. Haueter

GJW/map

R. L. Haueter
Assistant Electric Production
Superintendent - Nuclear

CONSUMERS POWER COMPANY

Docket No 50-155

Request for Change to the Technical Specifications

License No DPR-6 ZEK

For the reasons hereinafter set forth, it is requested that the Technical Specifications of License DPR-6, Docket 50-155, issued to Consumers Power Company on May 1, 1964, for the Big Rock Point Nuclear Plant be changed as follows:

I. Section 5

A. In Section 5.1.1, change to read as follows:

"Fuel (Sintered Pellets or Compressed Powder)	UO_2 or UO_2-PuO_2"

B. In Section 5.1.5(a), change to read as follows:

"Enrichment of Fuel approximate weight percent of U-235 from 2.6 to 5.2 inclusive. Approximate weight percent of Pu (fissile - Pu-239 and Pu-241) 1.0 to 4.0 in normal (0.7 w/o U-235) UO_2."

C. In Section 5.1.5, revise Figure 5.7 under E-G Fuel heading, Note R, to read:

"R - Removable fuel rods - low enrichment or UO_2-PuO_2 rods."

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

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

1 Four special fuel rods at bundle corners are segmented.

2 Reload B,C,E, and EG 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 EG 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 82 - 92 percent theoretical by using annular, dished, or nondished pellets in selected rods.

II. Discussion - Reload "E" or "E-G" Fuel With UO_2-PuO_2 Fuel

A. Program Description

One or both of the two removable rod positions in approximately 16 of the Reload "E" or "E-G" bundles will contain UO_2-PuO_2 sintered pellet fuel rods instead of UO_2 fuel rods. The purpose of this modification is to utilize the removable rod positions to further demonstrate recycling of plutonium in thermal power reactors.

B. Description of Fuel

The cladding and mechanical design of the mixed oxide fuel rods is identical to the design of the removable UO_2 rods they will replace (Table 1). The fuel rods contain cold pressed and sintered UO_2-PuO_2 pellets. The pellets are prepared from mechanically blended ceramic grade UO_2 and PuO_2 powder. Four types of mixed oxide pellets are used:

1. Low density, approximately 92% theoretical without dish.

2. High density, 95% theoretical, with 3% dish.

3. Low density, approximately 92% theoretical, annular pellets. The central annulus will remove approximately 5.5% of the pellet.

4. Low density, approximately 92% theoretical, annular pellets. The central annulus will remove approximately 18% of the pellet.

The weight of plutonium in each rod is constant and this is done by adjusting the weight fraction of PuO_2 in normal UO_2 in each type of pellet to achieve a constant plutonium loading.

The UO_2-PuO_2 fuel rods will be loaded in specified bundles. The removable UO_2-PuO_2 rods are identified by crossed grooves on the top end plug and by serial numbers on the top retainer and lower end plug.

Previous experience with UO_2-PuO_2 fuel containing small amounts of PuO_2 indicates that the thermal performance is essentially identical to UO_2 fuel.(1,2,3,4,5) Therefore, for the solid dished and nondished UO_2-PuO_2 pellets, the thermal performance will be identical to the Reload "E" and "E-G" fuel.

The two types of annular pellet fuel rods will both operate well below melting at 122% overpower 500,000 Btu/hr-ft^2 (Table II) and will have lower peak temperatures at normal operating conditions than the standard dished or nondished pellet rods.

C. Nuclear Design

The plutonium concentration was set to achieve a local power factor of 1.3 the removable rod position. The bundle array is shown on Figure I. The local power distribution is shown in Table III along with local power factors of "E-G" fuel without plutonium rods. Both designs have corrections for water gap effects. The calculations were made with standard GE nuclear methods. However, it should be noted that the plutonium rods will burn down almost twice as fast as the neighboring uranium rods or the 2.5 A/o U-235 rod that it replaced.

The addition of two plutonium rods per bundle is calculated to make the void, Doppler, and temperature coefficients slightly more negative. The bundle reactivity is essentially unchanged.

D. Thermal Hydraulic Analysis

As stated above, the nuclear design of the PuO_2-UO_2 rod increased the local peaking (maximum rod power) factor in the fuel bundle from 1.2 to a maximum of 1.3. Consequently, these bundles will be placed in core positions that have lower radial power factors. The resultant thermal hydraulic performance provides additional margin from the minimum critical heat flux ratio (MCHFR) limit, 1.5 at 122% overpower, due to the reductions of water quality in the bundle.

The effect of these assemblies on reactor thermal hydraulic performance has been evaluated with a predicted core configuration and the results indicate that the desired performance can be achieved within all reactor limits. During the refueling outages, after fuel inspection and prior to start-up, core analysis will be performed on the selected core configurations.

E. Related Experience

Approximately 470 Zr-2 clad fuel rods containing cold pressed and sintered UO_2-PuO_2 pellets have been irradiated in the Plutonium Recycle Test Reactor (PRTR) and SAXTON (Table IV). The irradiation of approximately 560 full-length mixed oxide fuel rods was initiated in October 1968 in the Garigliano Nuclear Power Station (SENN - Table IV). The PRTR and SAXTON experience with sintered pellet mixed oxide fuel is summarized below:

1. PRTR Experience (Ref 1, 2)

As part of the Batch Core Experiment in PRTR, two sintered pellet rods of $(Pu, U)O_2$ are being irradiated in a 19-rod PRTR type bundle. (See Table IV.) One additional rod, removed from a bundle at ~1,400 Mwd/MT peak burnup, was post-irradiation examined. Although operated at a peak rating of 17.8 kw/ft, there was only recrystallization and equiaxed grain growth in the central region of the cold pressed and sintered pellet. Such a microstructure is consistent with fuel operating temperatures of 1500° C - 1600° C maximum. Some discrepancy does exist, however, since temperatures of ~2000° C maximum were expected. The O/M ratio of the $(Pu, U)O_2$ fuel, determined post-irradiation, was 2.008. Fission gas release was 1.4% which is reasonable for equiaxed grain growth in the fuel.

No evidence was uncovered which would indicate that the low enrichment (Pu, U)O_2 rod was significantly different from a UO_2 rod operated at a comparable rating.

2. SAXTON Plutonium Project (Ref 3, 4, 5)

As part of the SAXTON Plutonium Experiment, some low 4% enrichment (Pu, U)O_2 rods, have been irradiated without any known failures. (See Table IV.) Peak power levels of 9 - ~16 kw/ft were experienced by some of these rods to peak burnups of ~22,200 Mwd/T.

One fuel rod which had operated at a maximum power rating of 10.7 kw/ft was post-irradiation examined. The fuel microstructure at the maximum power location showed only equiaxed grain growth at the pellet central region. Such a microstructure is consistent with the calculated heat rating which would predict a maximum central temperature of ~1500° C at 10.6 kw/ft.

Zircaloy-2 clad fuel rods containing hot pressed pellets are under test in the PRTR, SENN, and Dresden reactors. The PRTR test rods have operated satisfactorily at peak linear power ratings up to 21 kw/ft to a peak exposure of 10,000 Mwd/T.

The only known experience with annular mixed oxide pellets has been in the liquid metal fast reactor programs in the US and UK. As far as is known, the results of these irradiations have been satisfactory since, as a result of these tests, the UKAEA has selected annular pellets for the reference design for their liquid metal fast breeder demonstration reactor (Ref 8, 9).

Tests of fuel rods containing annular UO_2 pellets are summarized in Table V. The annular pellet integrity was good and there was no gross redistribution of fuel in the central hole due to pellet cracking and falling into the void in any of these tests. Although failures were encountered in these tests, the failures were caused either by fission product swelling(11) or by fuel swelling due to an excessively rapid power increase to allow fuel redistribution in previously molten core fuel rods(12). None of the failures were intrinsic to annular pellet fuel. In fact, the annular pellets were slightly superior to solid pellets in accommodating fission products, and the use of annular pellets made it possible to operate molten core pellet fuel rods under proper operating conditions.

Annular pellets have been previously approved for use in the Big Rock Point reactor in the centermelt bundles.

F. Accident Analyses

1. Reactivity Excursion Analysis

a. Postulated Reactivity Accidents

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

Case 1: In-sequence potential of 0.008 Δk for drop from full-in position to drive position.

Case 2: In-sequence potential of 0.021 Δk for drop from full-in to full-out.

Case 3: Out-of-sequence potential of less than 0.021 Δk for drop from full-in to full-out.

Case 4: Maximum theoretical worst case of about 0.04 Δk.

Case 1 requires the following equipment malfunctions and operator error:

1) Control rod becomes uncoupled from drive.
2) Control rod drive is withdrawn (in-sequence), but control rod hangs up temporarily.
3) Operator does not notice that control rod is not following.
4) Control 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 control rod completely rather than concurrent with the bank.

Case 3 consequences are less than those for Case 2.

Case 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 in a previous analysis[13].

At the present time, the core is licensed to contain six centermelt fuel bundles. Analysis is performed for a core of "E/E-G" fuel with the centermelt bundles and plutonium rods included. To prevent a large amount of centermelt fuel from being in the peak neutron flux during a reactivity accident, the six centermelt bundles are to be loaded in the core in a dispersed array with a minimum center-to-center distance of 42 cm. This restriction means that the closest centermelt bundle spacing will be no closer than two bundles in the x-direction and one in the y-direction.

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 dist-'bution.
4) Doppler coefficient.
5) Resonance neutron flux distribution.
6) Initial power.

The only significant difference between the *"plutonium" core and **"E" core is in the specific power distribution. The "plutonium" fuel bundle local power factor is about 8% higher than 'E" fuel. For a given reactivity excursion, this would increase the peak energy density in that assembly as well as yield more fuel mass above some energy levels. Also, the heterogeneity of the plutonium fuel might slightly affect the Doppler coefficient under rapid transient conditions. The best estimate of this effect indicates a reduction in the Doppler coefficient of 0.01%. Even extremely conservative considerations would only indicate a reduction of 0.5%. Both of these errors are within the uncertainties associated with the Doppler coefficient. Therefore, this effect was not considered in this analysis.

c. Primary System Integrity

As discussed at length in previous license 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.
3) Amount of fuel failed.
4) Energy in the failed fuel.
5) Heat transfer rate to 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. There is little or no information available on the effects of plutonium heterogeneity on prompt fuel failure. Because of this lack of information, the most conservative assumption is that all of the plutonium fuel is promptly dispersed. This would lead to a maximum energy release of 16.5 Mw-sec in the core. This energy would be quite diluted as it is contained in 15 bundles scattered about the core. Choosing a more reasonable but still conservative failure threshold such as 150 cal/gm would lead to an energy release of 10.5 Mw-sec in the core.

*"Plutonium" core contains the currently licensed core with 30 plutonium fuel rods distributed with two rods in each of 15 standard bundles.
**"E" core is the currently licensed core.

Taking into account the conservative figures for energy release from the "E" core, or the "E" core with demonstration bundles, (calculated to be 47 Mw-sec for an 0.021 Δk rod drop at hot standby with all fuel above 265 cal/gm being promptly dispersed in the moderator) and the energy in the plutonium fuel, the total prompt energy release in the core would be 63.5 Mw-sec. This is slightly less than the 64 Mw-sec used by DRL in previous analyses[13].

d. Conclusions

It is concluded that the results of a postulated reactivity accident are slightly more severe in the "plutonium" core than in the "E" core. However, the results are still within an envelope considered acceptable in granting the license for the "E" fuel. 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 30 UO_2-PuO_2 rods 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 locations with lower power factors in order to readily meet thermal limits. Thus, the result of any postulated LOC accident will be less severe because of the reduced bundle stored energy. For equal bundle powers in an "E-G" bundle with and without the UO_2-PuO_2 rods, the peak clad temperatures are only slightly different. For example, for an average bundle (ie, thermal power of 2.68 Mwt), the peak clad temperature changes from 1755° F to 1790° F.

III. Conclusions

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

1. Fuel rod mechanical design is identical to "E/E-G."

2. The local power factor is slightly higher for the UO_2-PuO_2 rods than the UO_2 rods in the "E/E-G" design but 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. The local power coefficients for the UO_2 rods are essentially unchanged.

4. The data available for low enrichment UO_2-PuO_2 fuel indicate that the performance is essentially identical to UO_2 fuel; therefore, the peak fuel temperatures in the solid pellet UO_2-PuO_2 rods are identical to the UO_2 fuel rods. In the case of annular pellet UO_2-PuO_2 fuel rods, the peak fuel temperature is lower than the UO_2 rods. (See Table II.)

5. Annular pellets have shown good structural integrity during tests. Annular pellets also are capable of higher thermal ratings without the fuel becoming molten.

6. The results of a postulated reactivity accident are slightly more severe in the "plutonium" core than the "E/E-G" reload. However, there is no danger of breaching the primary system due to a credible accident with either core loading. The severity of a loss-of-coolant accident is essentially unchanged from "E/E-G" fuel without UO_2-PuO_2 rods.

Based upon the above considerations, we have concluded that the use of Reload "E" or "E-G" fuel bundles containing one or two plutonium bearing rods in the Big Rock Point reactor does not present a significant change in the hazards considerations described or implicit in the Final Hazards Summary Report.

CONSUMERS POWER COMPANY

By: HRWall
Senior Vice President

Date: February 11, 1969

Sworn and subscribed to before me this 11th day of February 1969.

Grace R Warner
Notary Public, Jackson County, Michigan
My commission expires January 15, 1972

Figure I

Bundle Array

E-G FUEL ENRICHMENTS + 2 Pu RODS

In the 9 x 9 fuel array the following distribution is to be used:

Type 1	18 - 2.5 wt %	
2	32 - 3.4 wt %	
3	25 - 4.5 wt %	
4	4 - 35 g/ft cobalt	
Pu	2 - Pu Rods (Nat'l Uranium, plutonium)	

4	1	1	2	2	2	1	1	4
1	Pu	2	2	2	2	2	1	1
1	2	3	3	3	3	3	2	1
2	2	3	3	3	3	3	2	2
2	2	3	3	3	3	3	2	2
2	2	3	3	3	3	3	2	2
1	2	3	3	3	3	3	2	1
1	1	2	2	2	2	2	Pu	1
4	1	1	2	2	2	1	1	4

TABLE I

RELOAD "E" AND "E-G" FUEL DATA

	"EG" Fuel Rods	"E" Fuel Rods	Cobalt Rods	UO_2-PuO_2 Rods
Fuel Pellet Diameter	0.471	0.471	- -	0.471
Rod Pitch, Inches	0.707	0.707	0.707	0.707
Cladding Thickness, Inches	0.040	0.040	0.040	0.040
Clad Outside Diameter, Inches	0.5625	0.5625	0.5625	0.5625
Active Fuel Length, Inches	70.0; Central Rod 64.9	69.75; Central Rod, 64.62	- -	68.62
Fuel Material	UO_2	UO_2	- -	UO_2-PuO_2
Fuel Density, % of Theoretical	95	90-95	- -	92-95
Cladding Material	Zr-2	Zr-2	Zr-2	Zr-2
Number of Rods per Bundle	77	77	4	2
Enrichment (See Figures 5.7 & 5.8)*	Low-2.5%	Low-2.35%	- -	Nondished 1.30
	Middle-3.4%	Middle-2.93%	- -	Dished- 1.30
	High-4.5%	High-3.55%	- -	Annular 0.1" hole 1.36 Annular 0.[illegible]" hole 1.59
Fill Gas	Helium	Helium	- -	Helium

Fuel Bundle

Fuel Rod Array	9 x 9
Weight UO_2 and UO_2-PuO_2 per Bundle, Pounds	346.
Moderator-to-Fuel Volume Ratio	2.39
Number of Spacers	3

*UO_2-PuO_2 Enrichment is percent fissile Pu (Pu-239 and Pu 241) in normal (0.7 w/o) UO_2.

TABLE II

THERMAL PERFORMANCE CHARACTERISTICS OF

	RELOAD "E" & "E-G" FUEL		UO_2-PuO_2 FUEL			
Fuel Pellet Diameter, Inches	0.471		0.471	0.471	0.471	0.471
Fuel Pellet Inside Diameter, Inches	0.0		0.0	0.0	0.1	0.2
Cladding Thickness, Inches	0.040		0.040	0.040	0.040	0.040
Cladding Outside Diameter, Inches	0.5625		0.5625	0.5625	0.5625	0.5625
Incipient Melting Temperature of UO_2, °F	5080		5080	5080	5080	5080
Fuel Density, % Theoretical	95	90-92	92	95	92	92
Fuel Center Line Temperature at 500,000 Btu/Hr-ft^2, °F	5040	>5080	>5080	5040	4600	3800
Fuel Center Line Temperature at 410,000 Btu/Hr-Ft^2, °F	4250	4400	4400	4250	3900	3300
Heat Flux for Incipient Melting, Btu/Hr-Ft^2	>500,000	477,000	477,000	>500,000	540,000	680,000
Area Fraction Molten at Peak Heat Flux	0	0.04	0.04	0	0	0

Table III

Plutonium Isotopics

Pu-239	89.0 A/O
Pu-241	1.0 A/O
Pu-240	10.0 A/O

Local P/A-Hot (25% In-Channel Voids)

	1	2	3	4	5	
1	C_o	1.214 1.217	1.050 1.059	1.217 1.238	1.148 1.178	← E-G Fuel ← E-G Fuel + 2 Pu Rods
2		.901 1.297	.969 .968	.837 .852		
3			1.034 1.043			

TABLE IV - MIXED-OXIDE Pu RECYCLE EXPERIENCE - SINTERED PELLET FUEL

Reactor	Clad			Fuel*			No. of Rods	Peak Heat Rating Kw/ft	Peak Burnup MWD/T	Reference
	Type	Thickness (mils)	O.D. Inches	den %T.D.	Enrichment	Active Fuel Length				
PRTR	Zr-2	30	0.566	~91.5	1.94% PuO_2	75.7	3	18	~ 3,600	1, 2
Saxton	Zr-4	23.3	0.391	94	6.6% PuO_2	36.6	470	9-16	~22,200	3,4,5
SENN	Zr-2	37	0.593	91.5	2.0-3.2% Pu	104.3	60	<17	**	6
SENN	Zr-2	37	0.593	94.5	1.4-2.8% Pu	106.2	504	<17	**	7

*All fuel, mechanically blended powders of UO_2 & PuO_2.
**Irradiation started in October, 1968.

Table V

ANNULAR UO_2 PELLET TESTS

Reactor	Coolant	Type	O.D. In.	% T.D.	V/o In Hole	No. of Rods	Peak Heat Rating, KW/ft	Peak Burnup MWD/T	Ref.
ORR	NaK	Stainless Steel	0.755	95	~ 25	4	9	1500	10
ETR	2,000 psi water	Zr-4	0.28 & 0.56	95	~ 15	4	12	40000	11
GETR	1,000 psi water	Zr-2	0.566	95	8	13	56	12000	12

References

1. BNWL-SA-1204 "Operating Experience with Plutonium Fuels in PRTR" by M. D. Freshley & S. Goldsmith 8/25/67.

2. BNWL-739 "Plutonium Utilization Program Technical Activities Quarterly Report 12/67, 1/68, 2/68" 4/68.

3. WCAP-3385-52 Saxton Plutonium Program Mechanical, Thermal and Hydraulic Design of Saxton Partial Plutonium Core" by E. A. Bassler, et al 12/65.

4. WACP-3385-8 "Saxton Plutonium Program Semi-Annual Progress Report Ending 6/30/66" by N. R. Nelson 7/61.

5. WCAP-3385-12 "Saxton Plutonium Program Semi-Annual Progress Report for the Period Ending 6/30/67" by R. S. Miller & J. B. Roll 8/67.

6. Cooperative Program on Pu Recycle between GE and ENEL.

7. Cooperative Program on Pu Recycle between U.K., GE, and ENEL.

8. J. Simmons, et al, "Visit of AEC Fast Reactor Fuels Team to Installations in the U.K. and West Germany," May 23 - 31, 1968; August 20, 1968.

9. K. W. Swanson, J. K. Butler & J. A. L. Robertson, "Mark II Subassembly Report on Examination of DFR-114 at 7.3% Maximum Burnup," TRG-Memo-4073(D); July, 1967.

10. R. F. Boyle, "Post-Irradiation Examination of ORNL Group II ORR Capsules," GEAP-3813, Oct. 1961.

11. E. Duncombe, et al," Comparisons with Experiment of Calculated Dimensional Changes and Failure Analysis of Irradiated Bulk Oxide Fuel Test Rods Using the CYGRO-1 Computer Program," WAPD-TM-583, Sept. 1966.

12. M. F. Lyons, et al, "Molten Fuel Rod Operation to High Burnup," GEAP-5100-2.

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

FROM:
Consumers Power Company
Jackson, Michigan
(R. L. Haueter)

TO:
Dr. Peter A. Morris

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