Consumers Power Company

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

February 12, 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 18) will enable Consumers Power Company to insert into the reactor at Big Rock Point a fuel design, designated as "Modified E-G," which will permit the irradiation of four fuel bundles with up to 24 fuel rods with various mechanical properties, in removable rod positions in the fuel bundle. The purpose of the fuel rod irradiation tests, described herein, is to evaluate the irradiation-damage-induced mechanical property changes in zirconium-base alloy cladding as a function of alloy composition and fabrication method.

It is our intent to insert "Modified E-G" fuel into the Big Rock Point Reactor during our next refueling outage which is currently scheduled for April 1969. We would, therefore, be 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/dmb

R. L. Haueter
Assistant Electric
Production Superintendent -
Nuclear

8101150478

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 No 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 Structural Components to read as follows:

"Structural Components (fuel cladding will, in addition to 304 SS and Incoloy 800, include Zr-2, Inconel 600, and Zr- 3Nb- 1Sn."

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

(c) Fuel Bundles

"The general dimensions and configuration of the seven types of fuel bundles shall be shown in Figures 5.2, 5.3, 5.4, 5.5, 5.6, 5.7, 5.8 and 8.1 of these specifications. Principal design features shall be essentially as follows:"

C. In Section 5.1.5, add Figure 5.8.

D. In Section 5.1.5, replace the present table of fuel bundle parameters with the following table (next page).

E. In Section 5.2.1(b), in column titled "Reload 'E' Fuel," change to "Reload 'E,' 'E-G' and Modified E-G fuel."

504

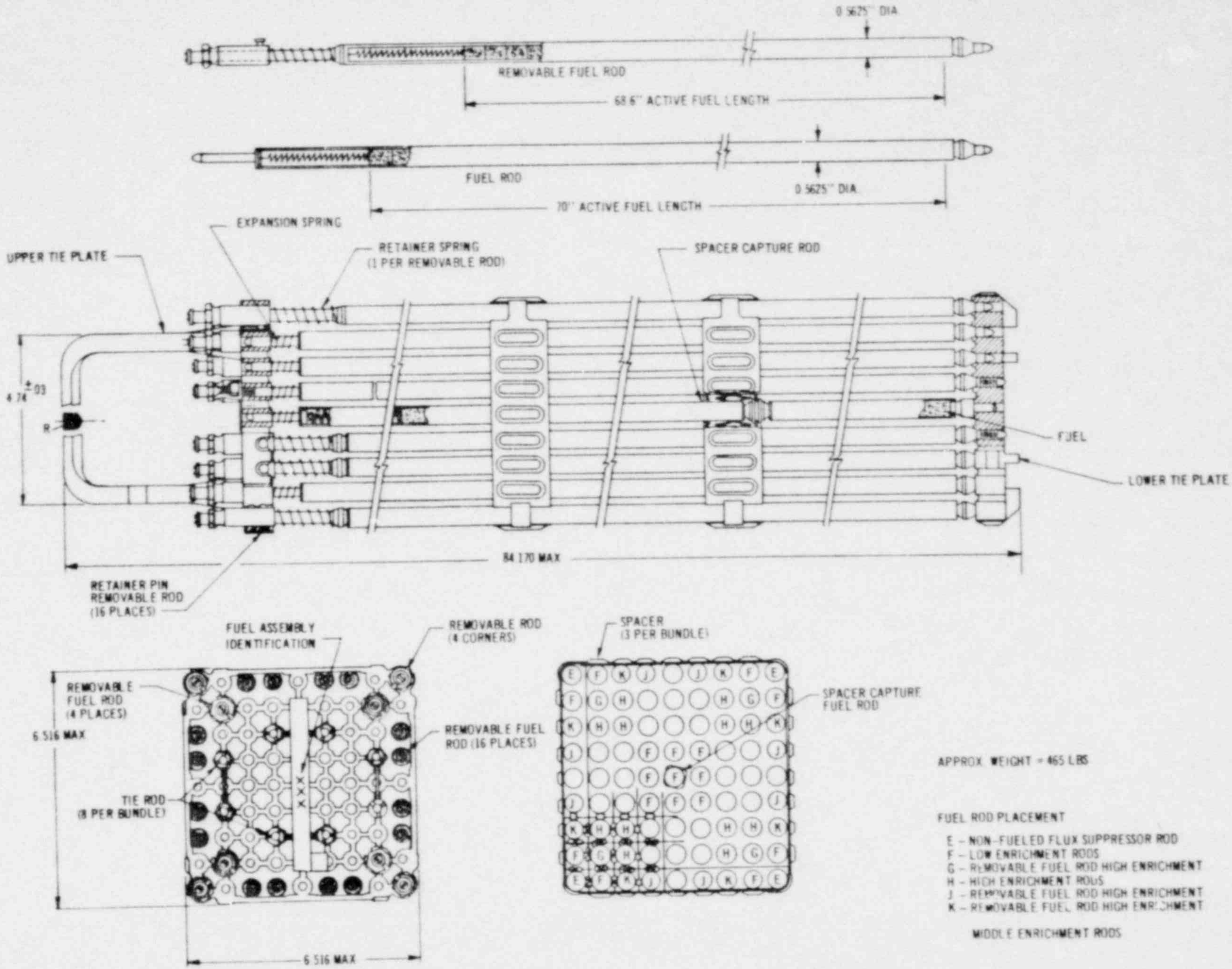

FIGURE 5.8 - BIG ROCK POINT "MODIFIED E-G" FUEL

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FUEL BUNDLES

General	Original (A)	Reload B & C	Reload E	Reload E-G	Research and Development: "D" Fuel	Research and Development: Centermelt Intermediate	Research and Development: Centermelt Advanced	Research and Development: "Modified E-G"
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
Rod Pitch, Inches	0.523	0.577	0.707	0.707	0.580	0.807	0.921	0.707
Standard Fuel Rods per Bundle	132	109	74	70	109	36	29	52
Special Fuel Rods Per Bundle	12[1]	12[2]	7[3]	11[3,5]	12	28[4]	20[4]	29[7]
Spacers Per Bundle	3	5	3	3	7	5	5	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 Zr-2 with various initial mechanical properties Zr-3Nb-1Sn
Standard Rod Tube Wall, In.	0.019	0.034	0.040	0.040	0.010 to 0.030 Inclusive	0.035	0.040	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
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
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	94 Pellet[6]
Active Fuel Length, Inches								
Standard Rod	70	70	69.75	70	68 to 70, Inclusive	66-67.3	65-66.3	70; 64.9 central, 68.6 Removable
Special Rod	59 (Corner)		64.6 Central	64.9 Central				
Fill Gas	Helium	Helium	Helium	Helium	Helium	Helium	Helium	Helium

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[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.

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.

Special rods have depleted uranium.

In addition to special rods for reload E, reload E-G has four gadolinia containing rods.

With 3% dishing on selected rods.

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 - "Modified E-G" Reload Fuel

A. Program Description

Four (4) fuel bundles designated "Modified E-G" bundles are proposed as carriers for test fuel rods.

The purpose of the fuel bundle irradiation tests is to evaluate the irradiation damage induced mechanical property changes in zirconium base alloy cladding as a function of alloy composition and fabrication method.

The initial mechanical properties of the Zr-2 tubing to be used for fuel cladding will be varied by thermomechanical treatments.

Four types of zirconium alloys (three are of basic Zr-2 with variations in thermomechanical processing; the fourth type being the Zr- 3Nb- 1Sn alloy) would be incorporated into these fuel bundles along with normal Zr-2 fuel rod cladding.

Figure 1 summarizes the proposed test matrix and position of the test fuel rods within the bundle.

The Zr- 3Nb- 1Sn alloy has the best combination of corrosion resistance and high-temperature strength of the potential alternate zirconium base water reactor cladding alloys and has been successfully tested as cladding in the KAHL-RWE boiling water reactor by Metallgesellschaft A.G., Frankfurt Am Main.

B. Fuel Description

The four demonstration fuel bundles (Figure 5-8 - outline drawing) are physically the same as the Reload "E" and "E-G" fuel. Differences are summarized below:

- Modification of the Reload "E-G" mechanical design to include:

    Addition of 16 removable peripheral fuel rod positions;

    Use of four removable interior fuel rod positions;

    Use of four corner removable thermal neutron absorber rod positions for nonfuel samples.

- The enrichment distribution and local peaking factors have been changed to enable as many of the 20 test fuel rods per assembly as possible to operate up to, but not exceeding, a maximum steady state power level of 17.7 kw/ft or 410,000 Btu/hr-ft^2 surface heat flux, which is the steady state limit set for the Reload"E" and "E-G" fuel.

- The position within the fuel bundle of the gadolinia containing fuel rods has been changed. The fuel rod design and gadolinia distribution, however, are the same as the Reload "E-G" fuel.

Figure 2 shows the tentative standard and test fuel rod positions and enrichment distribution. The configuration of the test fuel rods will be identical to the Reload "E-G" fuel rods with the exception of the removable rod feature. UO_2 pellet fuel will be utilized with no deviations from the "E-G" except for fuel enrichments. Table 1 summarizes the fuel data for all rods. The cladding operating stress criter . used for the "E-G" fuel rod design will be the same for the "Modified E-G" test rods.

The Big Rock Point Reactor is scheduled to continue to operate at a coolant pressure of 1,350 psi, giving a saturated water temperature of $\sim 585^{\circ}$ F. The reactor will be operating on 10-month to 1-year cycles starting in April 1969. The "Modified E-G" bundles will be designed to operate for three cycles and should achieve exposures of 20,000 Mwd/T average by the first quarter of 1972. Interim test fuel rod removals and inspections during outages are planned.

The design of the four "Modified E-G" bundles and the Reload "E" and "E-G" bundles includes corner positions for nonfueled rods containing thermal flux suppressors. In the "Modified E-G" bundles, these corner positions have been adapted for removable nonfueled test rods and will be used for the evaluation of Zr-2 mechanical property changes due to irradiation. These rods will include a thermal neutron absorber equivalent to that in the Reload "E-G" bundles (35 g/ft cobalt). The rods will have the same dimensions and configuration as the Reload "E-G" corner rods. The design of the nonfueled test rods will utilize thinner Zr-2 cladding (0.022-in min wall); however, the design will insure that no gross collapse or dimensional changes occur that would alter the nuclear or thermal hydraulic characteristics of the bundle.

C. Nuclear Design

The principal nuclear characteristics of the "Modified E-G" bundles have been calculated and compared to Reload "E" and "E-G" fuel and are summarized in Table 2. The reactivity values for the "Modified E-G" fuel at all conditions are lower than for "E" fuel, resulting in ample core shutdown margin. The temperature and void coefficients of the "Modified E-G" fuel are more negative than for the "E" fuel. The Doppler coefficient of the "Modified E-G" fuel is essentially the same as the "E" or "E-G" fuel at all conditions. The tentative calculated local power distribution of the test rods is shown in

Figure 3. It should be noted that all interior rods are of relative power less than average (1.0).

D. Thermal Hydraulic Data

The thermal hydraulic characteristics of the "Modified E-G" bundles are essentially identical to the Reload "E-G" fuel. The only differences are the differences in local power distribution which will be limited to a local peak-to-average ratio of less than approximately 1.4. These bundles will be placed in core positions having lower radial power factors to compensate for the higher local peaking. Consequently, the bundle steam quality will be reduced, resulting in more thermal margin to the MCHFR (1.5 at 122% overpower). Core thermal hydraulic analyses have been performed on predicted core configurations and indicate all license limits will be met. During the refueling outage, these analyses will be performed on the finally selected core configuration.

E. 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 0.045 Δk.

Case 1 required the following equipment malfunctions and operator error:

(1) Control rod becomes uncoupled from control rod drive;

(2) Control rod drive is withdrawn (in-sequence), but control rod hangs up temporarily. Operator does not notice that control rod is not following;

(3) 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 drive 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 for their analysis of the previous fuel submittals.[1]

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 "Modified E-C" bundles 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 distribution;

(4) Doppler coefficient;

(5) Resonance neutron flux distribution;

(6) Initial power.

The only significant difference between the *"Demonstration" core and **"E" core is in the specific power distribution. The "Modified E-G" fuel bundle local power factor is about 16% higher than "E-G" fuel (Figure 3). For a given reactivity excursion, this would increase the peak energy density in that bundle as well as yield more fuel mass above some stated energy levels.

0.021 Δk Rod Drop at Hot Standby	"E" Core	"Demonstration" 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	37
230 Cal/Gm	40	58

As can be seen, there has been an increase in the mass of fuel above 230 cal/gm but no increase in the mass at higher energy levels. It should be noted that the increase in mass above 230 cal/gm will occur only if a demonstration bundle is located immediately adjacent to a centermelt bundle. However, even if all four demonstration bundles are next to a centermelt bundle, it would not change the values shown above because of the transient power distribution.

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:

*"Demonstration" core would contain four "Modified E-G" bundles in the currently licensed core.

**"E" core is the currently licensed core.

(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. For material to be promptly dispersed, it must attain an energy density of 425 cal/gm or more. The above table demonstrates there is little, if any, promptly dispersed material in all the considered conditions. It is also seen that the "Demonstration" core is identical to the "E" core in this respect.

A large quantity of new transient test data has been obtained recently in the SPERT IV Capsule Driver Core.(2-5) 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 is consistent with the most recent calorimetric data for UO_2(6) which indicate incipient melting occurs at an energy level of about 269 cal/gm. Even if one promptly dispersed all of the fuel above 265 cal/gm, the energy contained in the dispersed fuel would only contain 47 Mw-sec. This is well below the 64 Mw-sec used by DRL in evaluation of other fuel submittals.(1)

2. Loss-of-Coolant Accident

The "Modified E-G" bundles have fuel rods placed such that most of the highly peaked rods are on the periphery adjacent to the bundle channel, as shown in Figure 3. The fuel rod positions that are adjacent to the bundle channel walls are the most efficiently cooled positions in the bundle in the event of a loss-of-coolant accident since the spray cooling water that enters the bundle cools the channel in a period of approximately 200 seconds, providing an excellent heat sink for the rods that are adjacent to the channel wall. To demonstrate this, an analysis of the bundle thermal performance during a loss-of-coolant accident transient of a standard "E" type fuel bundle, and a "Modified E-G" fuel bundle was made. The "Modified E-G" fuel maximum temperatures during the transient following a loss-of-coolant accident involving the maximum break size (3.5 sq ft bottom break) was the same as that of the "E" fuel. However, the percent of "E" fuel rods at the maximum temperature was 55%, while the "Modified E-G" fuel had only 5% of the rods at the maximum temperature. This type of relationship between "E" fuel behavior and "Modified E-G" fuel behavior will be consistent through the range of break sizes.

III. Conclusions

Based upon the above analyses and comparisons with "E" and "E-G" fuel, the following conclusions pertain to the "Modified E-G" fuel:

A. The mechanical design of the "Modified E-G" fuel is essentially identical to the "E" and "E-G" fuel which is a well-proven concept and has proven very satisfactory based on experience with the "E" fuel to date in the Big Rock Point Reactor. The addition of the removable rod positions is a minor mechanical modification.

B. The thermal hydraulic performance of the "Modified E-G" fuel will be within the limits set for the "E" and "E-G" fuel. The local power peaking will be higher, 1.4 compared to 1.2, than for the "E-G" fuel. Thermal hydraulic calculations show that there is ample critical heat flux margin.

C. The consequences of a loss-of-coolant accident are no more severe with "Modified E-G" fuel than with "E" or "E-G" fuel. Safe performance of "E" reload fuel was demonstrated in the "E" license submittal.

D. The consequences of a postulated reactivity accident are no more severe with "Modified E-G" fuel than with "E" or "E-G" fuel. The "Modified E-G" fuel bundles can be placed adjacent to each other or adjacent to a centermelt bundle with no additional risk. It is also concluded that there is no danger of breaching the primary system due to a credible reactivity accident with either "E," "E-G" or "Modified E-G" fuel bundles in the core.

Based upon the above considerations, we have concluded that the use of "Modified E-G" fuel 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 H R Wall
Senior Vice President

Date: February 12, 1969

Sworn and subscribed to before me this 12th day of February 1969.

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

TABLE I

DEMONSTRATION FUEL DATA (Modified "E-G")

	Standard Fuel Rods	Test Fuel Rods	Test Fuel Rods	Test Fuel Rods	Test Fuel Rods	Test Fuel Rods	Corner Rods
Fuel Pellet Diameter	0.471	0.471	0.471	0.471	0.471	0.471	-
Rod Pitch, Inches	0.707	0.707	0.707	0.707	0.707	0.707	0.707
Cladding Thickness, Inches	0.040	0.040	0.040	0.040	0.040	0.040	0.022 to 0.040
Clad Outside Diameter, Inches	0.5625	0.5625	0.5625	0.5625	0.5625	0.5625	0.5625
Active Fuel Length, Inches	70.0; Central Rod, 64.9	68.6	68.6	68.6	68.6	68.6	-
Fuel Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	-
Fuel Density, % of Theoretical	95	95	95	95	95	95	-
Cladding Material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-3Nb-1Sn	Zr-2	Zr-2
Alloy Code (See Figure 1)	-	A	B	C	D	E	-
Number of Rods per Bundle	57	4	4	4	4	4	4
Enrichment (See Figure 2)	Low - 2.5% Middle - 3.4% High - 4.5%	4.0	4.0	3.6	3.6	4.5	-
Fill Gas	Helium	Helium	Helium	Helium	Helium	Helium	-
Fuel Bundle							
Fuel Rod Array	9 x 9						
Weight UO_2 per Bundle, Pounds	~346						
Moderator-to-Fuel Volume Ratio	2.39						
Number of Spacers	3						

TABLE 2

COMPARISON OF PRINCIPAL CALCULATED NUCLEAR CHARACTERISTICS OF MODIFIED E-G FUEL WITH COBALT AND NOMINAL GADOLINIA

Reactivity (k_∞)	"E"	"E-G"	Modified E-G
68° F	1.268	1.208	1.229
572° F, 0 Voids	1.280	1.203	1.225
572° F, 25% Voids	1.262	1.183	1.206

Temperature Coefficient $\Delta k_{eff}/k_{eff}$ per °F @ 77° F	"E"	"E-G"	Modified E-G
Start of Cycle	$+ 0.38 \times 10^{-4}$	$+ 0.27 \times 10^{-4}$	$+ 0.13 \times 10^{-4}$

Void Coefficient $\Delta k/k$ per Unit Void Within Channel	"E"	"E-G"	Modified E-G
Cold (68° F)	- 0.07	- 0.08	- 0.07
Hot (572° F)	- 0.11	- 0.12	- 0.11

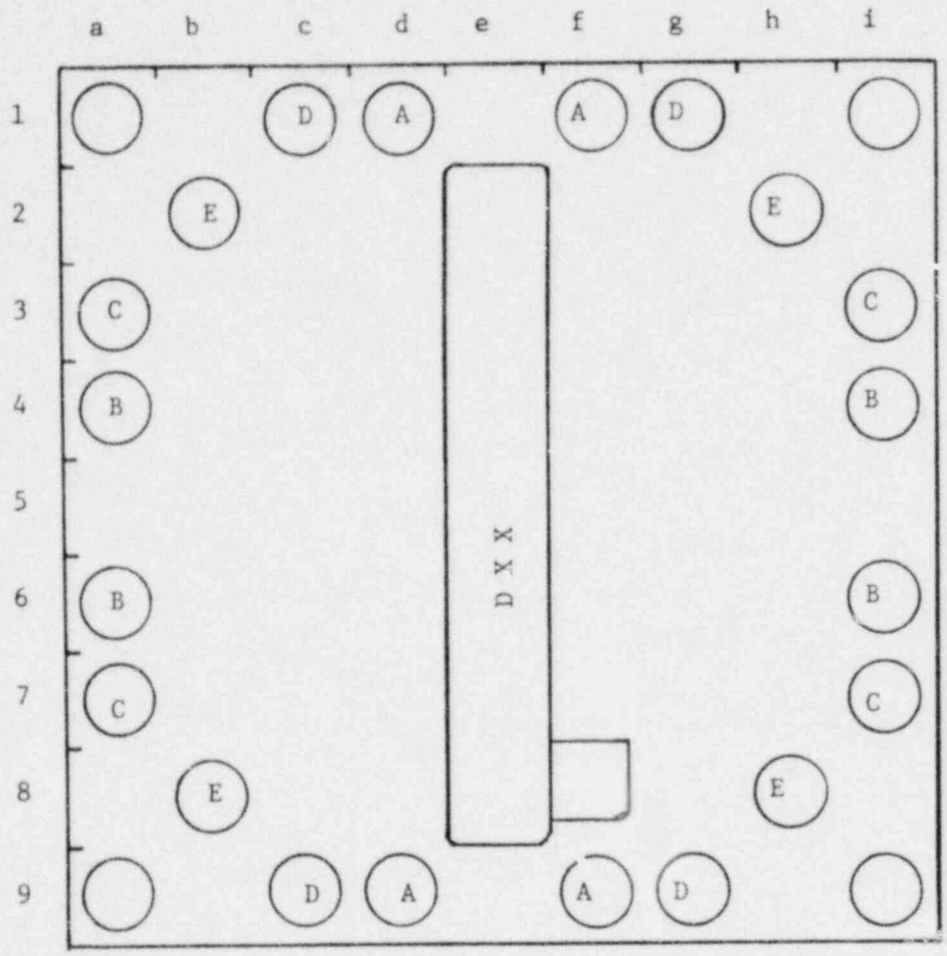
Doppler Coefficient $\Delta k_{eff}/k_{eff}$ per °F

Fuel Temp	Moderator	"E"	"E-G"	Modified E-G
68° F	68° F, 0 Voids	$- 1.3 \times 10^{-5}$	$- 1.3 \times 10^{-5}$	$- 1.3 \times 10^{-5}$
1323° F	572° F, 0 Voids	$- 1.0 \times 10^{-5}$	$- 1.0 \times 10^{-5}$	$- 1.0 \times 10^{-5}$
1323° F	572° F, 25% Voids	$- 1.2 \times 10^{-5}$	$- 1.2 \times 10^{-5}$	$- 1.2 \times 10^{-5}$

Figure 1

Modified E-G Bundle

Location of Test Fuel Rods



ALLOY CODE	ALLOY	STRENGTH LEVELS 70°F RELATIVE STRENGTH	0.2% Y.S. PSI x 1000	U.T.S. PSI x 1000
A	Zr-2	Low	40 - 50	60 - 70
B	Zr-2	Low	60 - 70	80 - 90
C	Zr-2	High	80 - 90	100 -110
D	Zr-3Nb-1Sn	High	90 -105	115 -130
E	Zr-2	Normal	70 - 80	90 -100

Figure 2

Modified E-G Enrichments

Test Rod Positions

4	1	7	6	2	6	7	1	4
1	3	3	2	2	2	3	3	1
7	3	3	2	2	2	3	3	7
6	2	2	1	1	1	2	2	6
2	2	2	1	1	1	2	2	2
6	2	2	1	1	1	2	2	6
7	3	3	2	2	2	3	3	7
1	3	3	2	2	2	3	3	1
4	1	7	6	2	6	7	1	4

Type	No.		Enrichment (wt %)
1	17		2.5
2	28		3.4
3	16		4.5
4	4	-	test locations containing absorber
6	8	-	4.0
7	8	-	3.6

Figure 3

Modified E-G Fuel

Local Fuel Rod Relative Power Distribution

300°C - 25% voids

○ = Test Rods

0	1.17	1.31	1.41	1.19				
1.17	1.37	1.17	.86					
1.31	1.17	1.02	.74					
1.41	.86	.74						
1.19								

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. IDC-ITR-1-1, "Transient Irradiation of 0.466-Inch O.D. Stainless Steel Clad Oxide Fuel Rods to 300 cal/g UO_2," Nov. 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," To be Published.

6. R. A. Hein, P. N. Flagella; "Enthalpy Measurement of UO_2 and Tungsten to 3250°K," Annual Meeting of Am. Cer. Soc., April 20 - 25, 1968.

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

TO:
Dr. Peter A. Morris

DATE OF DOCUMENT:	DATE RECEIVED:	NO.:
Feb 12, 1969	Feb 20, 1969	

LTR.: X | MEMO: | REPORT: | OTHER:

ORIG.: 1 | CC: 1 | OTHER: 37 add'l cys rec'd 3-24-69

ACTION NECESSARY ☐ | CONCURRENCE ☐ | DATE ANSWERED:
NO ACTION NECESSARY ☐ | COMMENT ☐ | BY:

CLASSIF.: U | POST OFFICE REG. NO:

FILE CODE: 50-155

DESCRIPTION: (Must Be Unclassified) Ltr trans request for Change No. 18 to the tech specs of Lic DPR-6:

ENCLOSURES: Request for Change to the Technical Specifications Lic DPR-6 to permit CPC to insert "Modified E-G" into the Big Rock Reactor......

(3 Signed 37 conf'd cys rec'd)

REMARKS:

REFERRED TO	DATE	RECEIVED BY	DATE
Ziemann	3-25		
w/9 cys for ACTION			
DISTRIBUTION:			
H. Price & Staff			
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OGC			

U.S. ATOMIC ENERGY COMMISSION MAIL CONTROL FORM FORM AEC-326S (8-60)