CONSUMERS POWER COMPANY

Docket No 50-155

Application for Amendment No 4 to Operating
License DPR-6

and

Request for Change to the Technical Specifications

Change No 34

License DPR-6

For the reasons hereinafter set forth, the following changes 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 are requested:

I. Section 8

   A. Change the first paragraph of Section 8.1 to read as follows:
      "8.1 Developmental Fuel Design Features
      The general dimensions and configurations of the developmental fuel designs shall be as shown in Figures 8.1 through 8.8. Principal design features shall be essentially as on Table 8.1."

   B. Section 8 - Figures:
      Add Figure 8.8 - Nuclear Fuel Services Inc, Demonstration Assembly.

   C. Delete present Table 8.1 and insert the attached Table 8.1.

   D. Delete present Table 8.2 and insert the attached Table 8.2.

8101100278

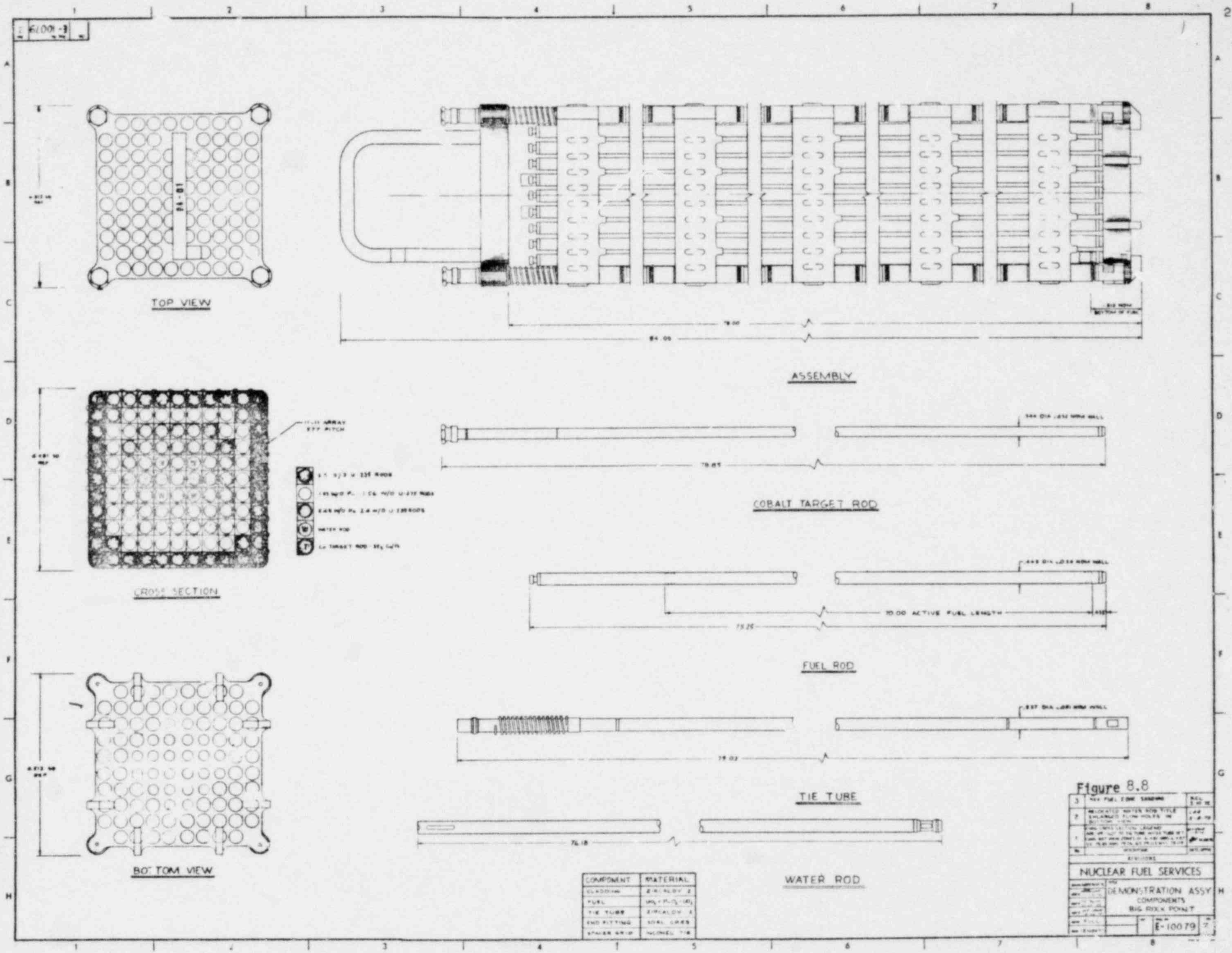


Figure 8.8

COMPONENT	MATERIAL
CLADDING	ZIRCALOY 2
FUEL	$UO_2 + PuO_2 - UO_2$
TIE TUBE	ZIRCALOY 2
END FITTING	304L CRES
SPACER GRID	INCONEL 718

POOR ORIGINAL

TABLE 8.1

	Research and Development Fuel Types						
General	Centermelt Intermediate	Centermelt Advanced	"Modified E-C"	EI UO_2-PuO_2	New Centermelt Intermediate	New Centermelt Advanced	NFS DA
Geometry, Fuel Rod Array	8 x 8	7 x 7	9 x 9	9 x 9	8 x 8	7 x 7	11 x 11
Rod Pitch, Inch	0.807	0.921	0.707	0.707	0.807	0.921	0.577
Standard Fuel Rods per Bundle	36	29	52	0	60	45	40(12)
Special Fuel Rods per Bundle	28(3)	20(3)	29(1, 2, 4)	81(6, 7)	4(1)	4(1)	81(9)
Spacers per Bundle	5	5	3	3	5	5	5
Fuel Rod Cladding							
Material	Zr-2	Zr-2	Zr-2 With Various Initial Mechanical Properties Zr-3Nb-1Sn	Zr-2	Zr-2	Zr-2	Zr-2(10)
Standard Rod Tube Wall, Inch	0.035	0.040	0.040	-	0.035	0.040	0.034
Special Rod Tube Wall, Inch	0.035	0.040	0.040	0.040	0.031	0.031	*(11)
Fuel Rods							
Standard Rod Diameter, Inch	0.570	0.700	0.5625	-	0.570	0.700	0.449
Special Rod Diameter, Inch	0.570	0.700	0.5625	0.5625	0.347(8)	0.347(8)	0.449(11)
Fuel Stacked Density, Percent Theoretical	94 Pellet 85 Powder	94 Pellet 85 Powder	94 Pellet(5)	82 Powder	92-93 Pellet	92-93 Pellet	91.5%
Active Fuel Length, Inches							
Standard Rod	66-67.3	65-66.3	70	70	67.3	66.3	70
Special Rod	-	-	64.9 Central, 68.6 Removable	-	-	-	-
Fill Gas	Helium	Helium	Helium	Helium	Helium	Helium	Helium

See attached page for footnotes.

(Revised with Change No 34 issued 7/24/72.)

FOOTNOTES TO TABLE 8.1

(1) Modified E-G and EEI UO_2-PuO_2 and new centermelt fuel bundles may contain (in the corner regions of the bundle) four Zr-2 tubes having encapsulated cobalt targets sealed within.

(2) Modified E-G and EEI UO_2-PuO_2 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.

(3) Special rods have depleted uranium.

(4) Also has four gadolinia-containing rods.

(5) With 3% dishing on selected rods.

(6) UO_2-PuO_2 fuel rod stack density will vary from 74 to 92% theoretical by using annular, dished, or nondished pellets in selected rods.

(7) Sixty-four UO_2-PuO_2 rods similar to standard UO_2 rods, four removable PuO_2 rods, eight gadolinia-containing rods, four cobalt corner rods and one empty (water-filled during operation) spacer rod.

(8) Diameter of cobalt targets inside SS corner tubes.

(9) The 81 special fuel rods are composed of 73 mixed oxide rods, 4 cobalt-bearing rods and 4 empty rods (water filled during operation); and include 12 developmental rods.

(10) The 12 developmental rods and 4 cobalt-bearing rods are Zr-4 clad.

(11) The corner tie tubes which will accept cobalt-bearing rods are 0.537" diameter, 0.081" nominal wall. The cobalt-bearing rods are 0.344" diameter, 0.032" nominal wall. The tie tubes are structural members of the assembly and fix the spacer grids in position.

(12) Some of the 12 developmental rods may replace the standard rods.

TABLE 8.2

	"EEI UO_2-PuO_2, and 'Modified E-G' Fuels"	Centermelt		
		Intermediate	Advanced	NFS-DA
Minimum Core Burnout Ratio at Overpower	1.5*	1.5*	1.5*	>1.5
Transient Minimum Burnout Ratio in Event of Loss of Recirculation Pumps From Rated Power	1.5	1.5	1.5	>1.5
Maximum Heat Flux at Overpower, Btu/h-Ft^2	500,000	-	-	402,000
Maximum Steady State Heat Flux, Btu/h-Ft^2	410,000	500,000	500,000	329,000
Maximum Fuel Rod Power at Overpower, kW/Ft	21.6	-	-	13.8
Maximum Steady State Fuel Rod Power, kW/Ft	17.7	21.8	26.8	11.3
Stability Criterion: Maximum Measured Zero-to-Peak Flux Amplitude, Percent of Average Operating Flux	20	-	-	20
Maximum Steady State Power Level, MW_t	240	-	-	240
Maximum Value of Average Core Power Density at 240 MW_t, kW/L	46	-	-	46
Maximum Reactor Pressure During Power Operation, Psig	1,485	-	-	1,485
Minimum Recirculation Flow Rate, Lb/h (Except During Pump Trip Tests or Natural Circulation Tests as Outlined in Section 8)	6×10^6	-	-	6×10^6
Maximum MWd/T of Contained Uranium for an Individual Bundle	23,500	-	-	23,500
Number of Bundles				
Pellet UO_2	-	1	3	-
Powder UO_2	-	1	2	-

Rate of Change of Reactor Power During Power Operation:

Control rod withdrawal during power operation shall be such that the average rate of change of reactor power is less than 50 MW_t per minute when power is less than 120 MW_t, less than 20 MW_t per minute when power is between 120 and 200 MW_t, and 10 MW_t per minute when power is between 200 and 240 MW_t.

---

*Based upon critical heat flux correlation, APED-5286.

II. Discussion - NFS Demonstration Assemblies

A. Fuel Description

The NFS-DA fuel has been designed within the concept of direct interchangeability with incumbent fuel so that no change in operations, instructions or limits will be required.

Most of the features of the NFS-DA fuel have been employed previously in Big Rock Point fuel assemblies. Table 1 (attached) compares the principal characteristics with Types B&C, EEI, F and J-2 fuel assemblies. Like the Type B&C fuel, the NFS-DA employs an 11 x 11 array of rods, utilizes a fuel rod outer diameter of 0.449 inch and a cladding thickness of 0.034 inch. It has cobalt target rods in the four corners of the assembly, is designed with sufficient reactivity to achieve an average burnup of 15,000 MWd/MTM, and does not use gadolinia burnable poison. Like the EEI and J-2 fuel, it uses UO_2 and mixed oxide (PuO_2-UO_2) sintered pellets of approximately the same density, with the mixed oxide rods arranged in the inner regions of the assembly and the UO_2 fuel rods on the outer rows. Nonfueled water rods, similar to that of the EEI fuel, are positioned within the assembly.

The principal differences between the NFS-DA fuel and the previous fuel are in mechanical construction as described below.

1. Mechanical Design

The NFS-DA fuel fits the same physical envelope as other Big Rock Point fuel assemblies and is fully interchangeable with them. However, the NFS-DA utilizes a mechanical design that is unique (see Figure 8.8). The tubes in the corner positions serve as the structural members to tie the assembly together, as the spacer grid capture rods, and as the cobalt target rod housings.

The fuel rods are positioned by five spacer grids. Previous Big Rock Point fuel designs accomplished this by providing three intermediate spacer grids and employing fuel rods with end plug shanks that fit into holes in the upper and lower end fittings.

The lower end plugs of the NFS-DA fuel rods rest on the lower end fitting and the upper end plugs stop short of the upper end fitting. Each rod is free to expand axially, independent of other rods in the assembly.

The lower end fitting is similar in appearance to those of earlier Big Rock Point fuel designs (ie, Type F), except that it has its coolant flow openings graded in size from the center outward to redistribute the jet formed by the existing channel support tube orifice located beneath the fuel and is not drilled to position fuel rod end shanks. The coolant flow openings in this lower end fitting were designed to be compatible with the existing support tube orifices as well as with the new orifices which are being used to replace a portion of the existing orifices. The upper end fitting is completely removable so that any fuel rod may be removed from the assembly and examined or replaced, if desired.

Four nonfueled tubes are positioned in the inner regions of the fuel assembly. They are made of the same material, diameter, wall thickness and overall length as the fuel rod cladding. The upper end of these tubes is fully open but the lower end has a special end plug with side openings to admit coolant to the inside of the tube. These end plugs rest on the lower end fitting like those of the fuel rods.

2. Nuclear Design

The NFS-DA fuel is designed to achieve an average burnup of 15,000 MWd/MTM during three (3) cycles in the reactor. The assembly average enrichment is 3.31 w/o fissile (Pu-239, Pu-241 and U-235), distributed as shown in Figure 1 (attached) and in accordance with the detailed mass balance given in Table 2 (attached). Beginning-of-life (BOL), normalized rod power distributions within a fuel assembly are shown in Figures 2 through 4 (attached) for uncontrolled, hot operating conditions at 0%, 25% and 40% void planes. The lack of rod power symmetry in the assemblies is due to nonuniform burnups assumed for three surrounding assemblies in a two-by-two unit assembly group. This arrangement of fuel, with varying degrees of depletion, around a fresh fuel assembly, was employed for the rod power factor evaluations rather than one that assumes the assembly to be inserted in an infinite array of identical fresh fuel because the former represents a realistic evaluation of the environment these bundles will experience.

A comparison of rod power peaking factors for the NFS-DA fuel with Types F, EEI and J-2 fuel is included in Table 4 (attached).

The infinite multiplication factor, the reactivity deficit from cold to hot standby, the reactivity responses and the control rod worth for the NFS-DA fuel are compared to those of previous fuels in Table 3 (attached).

The use of mixed oxide in the NFS-DA produces a slightly reduced control rod worth (cold). However, with the presence of four DA and the lower k_∞ (cold) of the fuel, shutdown and hold-down capability of the core with one control rod adjacent to an NFS-DA fuel assembly fully withdrawn is assured.

Lattice characteristics of the NFS-DA fuel produce a large cold-to-hot reactivity swing. The result is a more negative moderator temperature coefficient than that of the Type J-2 fuel. Similarly, the void response of the NFS-DA fuel is more negative than that of the Type J-2 fuel but somewhat less negative than that of the Type F fuel. Finally, the increased resonance absorption of the Pu-240 creates a Doppler response for the NFS-DA fuel that is more negative than any other fuel shown in Table 3 (attached). The combination of the moderator temperature, the void, and the Doppler characteristics produces a power response to reactivity insertion events equivalent to or less severe than that of previously licensed fuel.

3. Thermal and Hydraulic Design

The thermal performance design of the NFS-DA is such that they may be considered operationally interchangeable with the present principal resident fuel (Type F). The hydraulic characteristics of the upper and lower end fittings and the spacer grids have been carefully adjusted to produce an overall assembly hydraulic characteristic similar to that of the Type F fuel. The purpose of this tailoring of the hydraulic characteristics is to permit the assemblies to pass the same flow as that of Type F fuel. This assures that they will not detract from the hydraulic performance of the Type F or other resident fuel. By matching flow of the NFS-DA, the critical heat flux characteristic is affected mainly by the differences of heat transfer surface area and local peaking factors. Table 4 (attached) compares the principal thermal-hydraulic characteristics of the Types B, E, E-G, F, EEI, J-2 and NFS-DA fuel. Individual

NFS-DA have 17% more heat transfer surface area than a Type F fuel assembly and slightly more than this when compared to the other mixed oxide (UO_2-PuO_2) fuel assemblies in the core.

The local peaking factor depends upon the interaction of the effects of fissile material distribution in the fuel rods, the assembly internal water to fuel ratios, the void fraction within the assembly and the exposure. Figures 2, 3 and 4 (attached) show the local power distributions at 0%, 25% and 40% void planes at beginning of life. These are comparable to or slightly below what would be expected from previously licensed fuel in the BRP reactor. Accordingly, with comparable fuel assembly pin power peaking factors, the same coolant flow rate, 17% more heat transfer surface area, and an equivalent or less severe power response to reactivity insertion events, the critical heat flux margins in the NFS-DA fuel will be greater than that of 9 x 9 array fuel under both steady state and transient operation.

As stated earlier, the NFS-DA fuel is designed to be operationally interchangeable with the principal resident fuel (Type F) at the time of its insertion into the reactor. Accordingly, it will be operated at the same assembly power limits applied to the existing fuels. In this mode the NFS-DA linear and surface flux limits are less than those of previously licensed 9 x 9 array fuel. If an NFS-DA was operated at that power level and power distribution that would produce a surface heat flux in a Type F fuel assembly of 500,000 Btu/h-ft^2 and a linear heat flux of 21.6 kW/ft, the corresponding values in the NFS-DA fuel would be approximately 402,000 Btu/h-ft^2 and 13.8 kW/ft. At this condition the NFS-DA fuel central temperature would only be 3650°F. The thermal conductivity characteristics of UO_2 were used in these calculations. Because the content of PuO_2 in the mixed oxide rods is low, the thermal conductivity of the fuel material will be essentially that of UO_2. Furthermore, the greater neutron self-shielding of the PuO_2 will cause a greater fraction of the power to be generated away from the pellet center line, with the result that temperature calculations using UO_2 characteristics will indicate higher than actual values.

4. Developmental

Of the 113 fuel rods in each assembly, 12 rods are test or developmental rods. These rods are identical to the standard fuel rods in all respects except that their cladding material will be Zr-4 instead of Zr-2 and 6 of them will not be prefilmed (autoclaved). The use of Zr-4 cladding material in 12 rods and the nonprefilming of 6 rods in each assembly is not expected to interfere with the performance of the assemblies in any way.

5. PuO_2 Particle Size

Manufacturing specifications require that 95% of the PuO_2 particles in the mixed oxide pellets be less than 100 microns in diameter at a 95% confidence level. At this level, 95% of all particles measured by the alpha-autoradiographic technique must be 60 microns or less. Manufacturing procedures will assure that only a very few particles will exceed 100 microns in diameter.

The significance of particle size is amply discussed in the "Additional Information for Proposed Technical Specification Change No 19," dated January 28, 1970. It is shown that the presence of ~60 micron plutonium particles does not adversely affect the negative Doppler reactivity coefficient because of prompt energy transfer from the particles to the surrounding UO_2 matrix during a transient. Also, Change No 19 shows that if 15% of the plutonium is in particles greater than 50 microns, the fraction of energy that is available for conversion to mechanical energy for isolated particles is less than 0.3% of the total energy available, and is therefore rather insignificant.

Accordingly, the discussion of the significance of particle size presented in Change No 19, including results and conclusions stated therein, applies to the NFS-DA fuel.

B. Accident Analysis

1. Reactivity Insertion Accident

The reactivity response data of Table 3 (attached) indicate that the characteristics of the NFS-DA fuel are all acceptable. The table reveals that the Doppler coefficient is more negative than either the J-2 or the EEI-Pu fuel and that the void response is also more negative than that of either the J-2 or the EEI-Pu fuel and that the void response is also more negative than that of either the J-2 or EEI-Pu fuel. The delayed neutron fraction is comparable to that of the J-2 fuel and greater than that of the EEI-Pu fuel. The rod worth is less. Accordingly, the consequences of a rod drop accident would be expected to be less for the NFS-DA fuel than for the J-2 fuel. Therefore, the reactivity insertion accident discussion presented in the application for Change No 19, dated December 27, 1969 and additional information dated January 20, 1970 and February 13, 1970 will conservatively apply to the NFS-DA fuel.

2. Loss-of-Coolant Accident

Analysis of the Design Basis Accident for the NFS Demonstration Assemblies has been performed in accordance with the Interim Acceptance Criteria. The results of this analysis were submitted to the Directorate of Licensing by letter dated May 18, 1972. The analysis yields a maximum clad temperature of 2129°F and a core average metal-to-water reaction of 0.3%.

The reanalysis was performed assuming that the NFS-DA fuel was operating at a bundle power equivalent to the licensed limit of Type F fuel. The limits proposed for Section 8.2 of the specifications are consistent with values used in the reanalysis.

Reanalysis of small break loss-of-coolant accidents are in progress. The results of this reanalysis will be submitted to the Directorate of Licensing when they are completed.

3. Primary System Integrity

The reactivity insertion associated with a rod drop accident will be less than that for the J-2 fuel, as discussed earlier. Therefore, the discussion on Primary System Integrity presented in the application for Change No 27 will conservatively apply to the NFS-DA fuel.

4. Fuel Handling and Criticality

The NFS-DA fuel will be shipped to Big Rock Point in a fresh fuel shipping container, licensed separately for mixed oxide fuel shipment. Inside the Big Rock Point containment vessel they will be handled in the same fashion as previous fuel.

The NFS-DA fuel exhibits reactivity characteristics which are similar to other BRP fuel. Therefore, they may be handled and stored under the criticality and control limits which apply to BRP fuels.

III. Conclusions

Based on the data and discussion presented above, it is concluded that:

1. The NFS-DA fuel may be operated interchangeably with present resident fuel. All present limits and operating philosophy may be applied.
2. The fuel rod power peaking factors of the NFS-DA fuel are comparable to or lower than those of previously licensed fuels. In addition, the relative or radial power of these assemblies will be lower than those of the Types E, E-G, F and EEI-Pu fuels due to lower k_∞.
3. The greater heat transfer surface area and the same overall hydraulic characteristics as the Types E, E-G and F fuels, together with comparable or lower power peaking factors, assure a greater critical heat flux ratio for equivalent assembly power performance under both steady state and transient conditions.
4. Peak fuel temperatures of the NFS-DA fuel will be considerably lower than those of previously licensed 9 x 9 array fuel due primarily to the lower linear heat flux that occurs as a result of an 11 x 11 array design.
5. The consequences of a rod drop reactivity insertion accident will be less severe than for the J-2 fuel because of lower rod worth.
6. The NFS-DA fuel meets the Interim Acceptance Criteria for a Design Basis Accident.

Based on the above considerations, we have concluded that the use of NFS-DA 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 R. B. Sewell
Vice President

Date: July 24, 1972

Sworn and subscribed to before me this 24th day of July 1972.

Marion Y. Van Aken
Notary Public, Jackson County, Michigan
My commission expires October 14, 1974.

TABLE 1

	B&C	EEI	F	J-2	NFS-DA
General					
Geometry, Fuel Rod Array	11x11	9x9	9x9	9x9	11x11
Rod Pitch, Inch	0.577	0.707	0.707	0.707	0.577
Standard Fuel Rods per Bundle	109	--	70	4[illegible]	40 $^{(11)}$
Special Fuel Rods per Bundle	12$^{(1)}$	81$^{(2)}$	11$^{(1,3,4)}$	32$^{(6)}$	81 $^{(8,11)}$
Spacers per Bundle	5	3	3	3	5
Fuel Rod Cladding					
Material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2$^{(9)}$
Standard Rod Tube Wall, Inch	0.034	---	0.040	(7)	0.034
Special Rod Tube Wall, Inch	0.031	0.040	0.040	(7)	0.034$^{(10)}$
Fuel Rods					
Standard Rod Diameter, Inch	0.449	---	0.5625	0.5625	0.449
Special Rod Diameter, Inch	0.344	0.5625	0.5625	0.5625	0.449$^{(10)}$
Fuel Stacked Density, % Theoretical	94 + 1 pellet 85 powdered	82	94 pel-	90.7$^{(5)}$	91.5
Active Fuel Length, Inches					
Standard Rod	70	--	70	68	70
Special Rod	--	70	64.9	62.2	70
Fill Gas	Helium	Helium	Central Helium	Central Helium ≥ 95%	Helium ≥ 90%

(1) Reload B, C, E, E-G and F fuel bundles may contain (in the corner regions of the bundle) four Zr-2 tubes having encapsulated cobalt targets sealed within.

(2) 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.

(3) Fuel bundles have a special central fuel rod to which the bundle spacers are fixed.

(4) In addition to special rods there are 4 gadolinia containing rods.

(5) With 3% dishing on selected rods.

(6) This includes 24 mixed oxide (PuO_2-UO_2) rods, 4 cobalt bearing corner rods and 4 gadolinia bearing rods.

(7) This includes 44 fuel rods of 0.050", 33 fuel rods of 0.040"; and 4 cobalt rods of 0.035".

(8) The 81 special rods are comprised of 4 cobalt bearing rods, 12 experimental rods, 4 empty rods (water filled during operation), with the remainder UO_2-PuO_2 rods similar to the standard UO_2 rods.

(9) The 12 experimental rods and cobalt bearing have Zr-4 cladding, the empty rods are of Zr-2.

(10) The corner tubes for loading the cobalt target rods are 0.537" OD and have a tube wall 0.081".

(11) Some of the 12 experimental rods may replace standard fuel rods.

TABLE 2

NFS-DA FUEL MASS BALANCE (AT BOL)

		Inner Region 2.45 Pu-2.4 U	Intermediate Region 1.03 Pu-1.56 U	Outer Region 2.4 U	Assembly Average* 1.18 Pu-2.16 U
		(grams/rod)	(grams/rod)	(grams/rod)	(Kgs/Assy)
1.	U-235	25.767	17.043	26.545	2.663
2.	U-236	0.000	1.747	0.000	0.056
3.	Uranium Total	1073.679	1092.508	1106.098	123.225
4.	Pu-238	0.184	0.078	0.000	0.010
5.	Pu-239	24.864	10.501	0.000	1.355
6.	Pu-240	6.062	2.560	0.000	0.330
7.	Pu-241	2.251	0.951	0.000	0.123
8.	Pu-242	0.554	0.234	0.000	0.031
9.	Fissile Plutonium (5 + 7)	27.115	11.452	0.000	1.478
10.	Plutonium Total	33.915	14.324	0.000	1.849
11.	Total Fissile (1 + 9)	52.882	28.495	26.545	4.141
12.	(U + Pu) Total	1107.594	1106.832	1106.098	125.074
13.	Average w/o Fissile	4.77	2.57	2.40	3.31

* Assembly average data is based upon the following combinations:

Inner Region - 41 rods of 2.45 Pu - 2.4 U (Fresh)
Intermediate Region - 32 rods of 1.03 Pu - 1.56 U (Reprocessed)
Outer Region - 40 rods of 2.4 U (Fresh)

TABLE 3

	E	E-G	F	EEI-Pu	J-2	MFS-DA
Fuel Loading						
No. of Gd_2O_3 Rods	---	4	4	8	4	---
Kg of Uranium	136.4	138.4	138.2	117.8	123.3	123.2
Kg of Plutonium	---	---	---	5.33	1.47	1.85
Kg of U + Pu	136.4	138.4	138.2	123.1	124.8	125.1
Total Fissile Content (w/o fissile)	2.979	3.55	3.233	4.84	3.522	3.31
Relative Fissile Content in Pu	---	---	---	88.9	82.06	79.95
Reactivity (Uncontrolled K_∞)						
Cold (68°F)	1.268	1.208	1.179	1.160	1.148	1.199
Hot Standby	NA	NA	NA	NA	1.144	1.184
Full Power, 0% Void	1.280	1.203	1.183	1.168	1.138	1.176
Full Power, 25% Void	1.262	1.183	1.171	1.158	1.118	1.144
Temperature Deficit						
68° - 77°F ($\Delta K_\infty/K_\infty$ per °F)	NA	NA	NA	NA	NA	$+0.32\times10^{-4}$
Start of Cycle ($\Delta K_{eff}/K_{eff}$ per °F)	$+0.38\times10^{-4}$	$+0.27\times10^{-4}$	$+0.31\times10^{-4}$	$+0.30\times10^{-4}$	NA	NA
Cold - Hot Standby (ΔK_∞)	NA	NA	NA	NA	-0.0036	-0.0146
Void Response						
0 - 25% Void (ΔK_∞)	NA	NA	-0.012*	-0.010*	-0.0202	-0.0325
Full Power, 25% Void ($\Delta K/K$ per unit void)	-0.11	-0.12	-0.133	-0.084	-0.0710*	-0.1107
Doppler Response						
Hot Standby ($\Delta K_\infty/K_\infty$ per °F)	NA	NA	NA	NA	NA	-1.33×10^{-5}
Full Power, 25% Void ($\Delta K_{eff}/K_{eff}$ per °F)	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	-1.25×10^{-5}	NA	NA
Hot Standby - Full Power ($\Delta K_\infty/K_\infty$ per °F)	NA	NA	NA	-1.05×10^{-5}	-0.90×10^{-5}	-1.48×10^{-5}
Control Rod Worth						
Cold (ΔK_∞, one adjacent rod)	NA	NA	NA	NA	0.1126	0.1048
Delay Neutron Fraction	NA	NA	NA	0.00385	0.0057	~0.0050

* Inferred Values

NA = Not Available

TABLE 4

		B	E	E-G	F	EET-Pu	J-2	NFS-DA
Number of Fuel Rods per Bundle		117	77	77	77	76	77	113
Fuel Rod Diameter	(in)	0.449	0.5625	0.5625	0.5625	0.5625	0.5625	0.449
Active Fuel Length	(in)	70	69.75[1]	70[2]	70[2]	70	68[3]	70
Heat Transfer Area per Bundle	(sq. ft)	78.94	65.84	66.08	66.08	65.29	64.18	77.48
Coolant Flow Area per Bundle	(sq. in)	24.23	22.47	22.47	22.47	22.47	22.47	23.17
Average Surface Heat Flux[4]	(Btu/hr-sq. ft)	119,830	144,000	143,140	143,140	144,870	151,915	122,080
Maximum Surface Heat Flux[5] [6]	(Btu/hr-sq. ft)	530,000	500,000	500,000	500,000	500,000	500,000	500,000
Average Linear Heat Flux[4]	(Kw/ft)	4.06	6.26	6.18	6.18	6.25	6.55	4.20
Maximum Linear Heat Flux[5] [6]	(Kw/ft)	17.2	21.6	21.6	21.6	21.6	21.6	21.6
Maximum Fuel Temperature[6]	(°F)	NA	NA	5040	5203	4606	5060	4600
Minimum Critical Heat Flux Ratio[5]		≥ 1.5	≥ 1.5	≥ 1.5	≥ 1.5	≥ 1.5	≥ 1.5	≥ 1.5
Pin Power Factor of Highest Power Rod[7]		1.53	1.21	1.3	~1.3	1.287[8]	1.121[8]	1.163[8]

[1] One spacer capture rod of 64.6 inches active fuel length

[2] One spacer capture rod of 64.9 inches active fuel length

[3] One spacer capture rod of 62.2 inches active fuel length

[4] At rated reactor power

[5] License limit

[6] At 122% of rated reactor power

[7] Beginning of life, uncontrolled

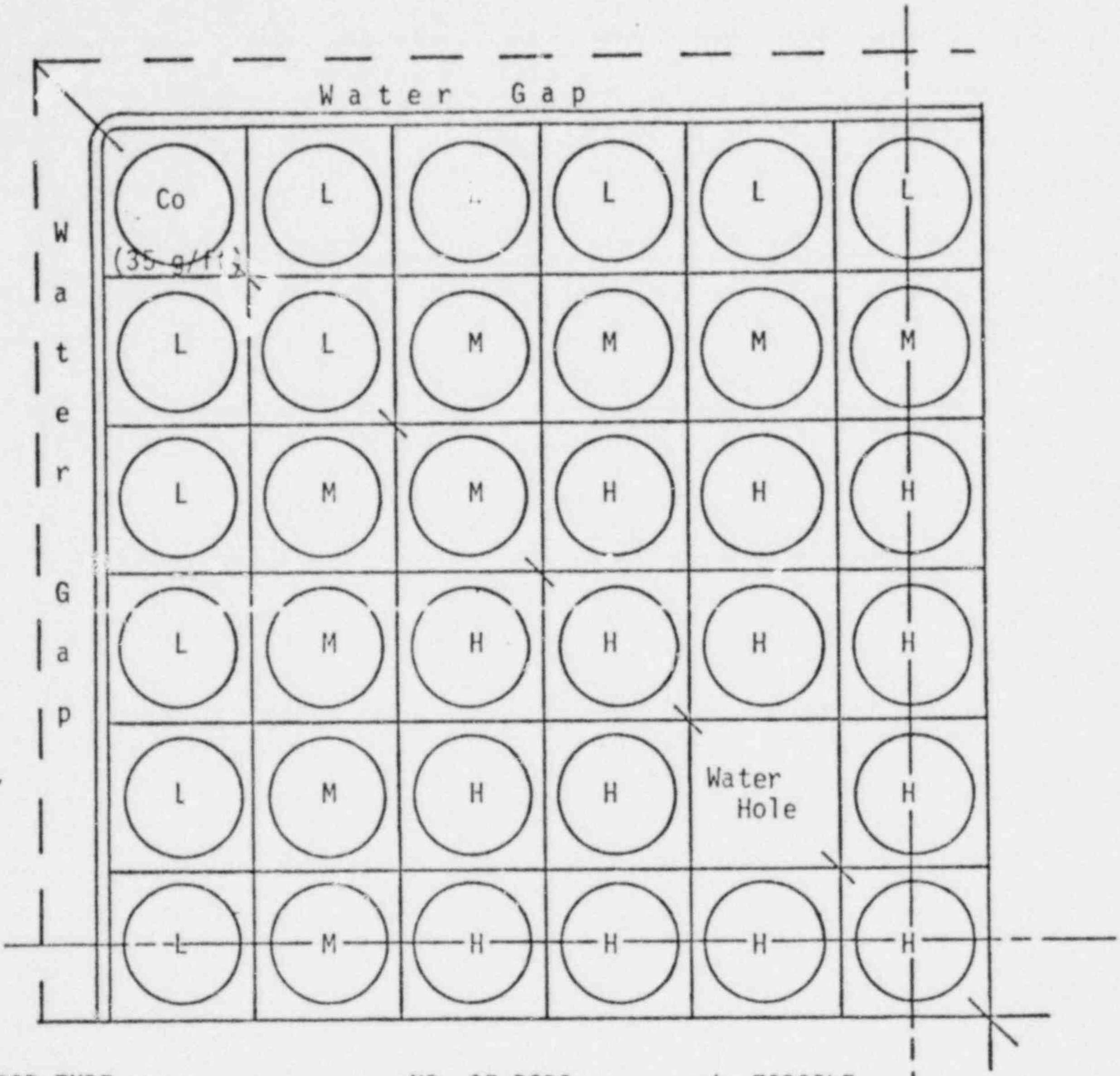
[8] At 25% Void

FIGURE 1

DISTRIBUTION OF FUEL ROD TYPES

NFS MIXED OXIDE (PuO_2-UO_2) DEMONSTRATION ASSEMBLY

FOR BIG ROCK POINT



FUEL ROD TYPE	NO. OF RODS	w/o FISSILE
L - 2.40 w/o U-235 (Fresh)	40	2.40
M - 1.03 w/o Puf + 1.56 w/o U-235	32	2.57
H - 2.45 w/o Puf + 2.40 w/o U-235 (Fresh)	41	4.77
	113	3.31 (Average)

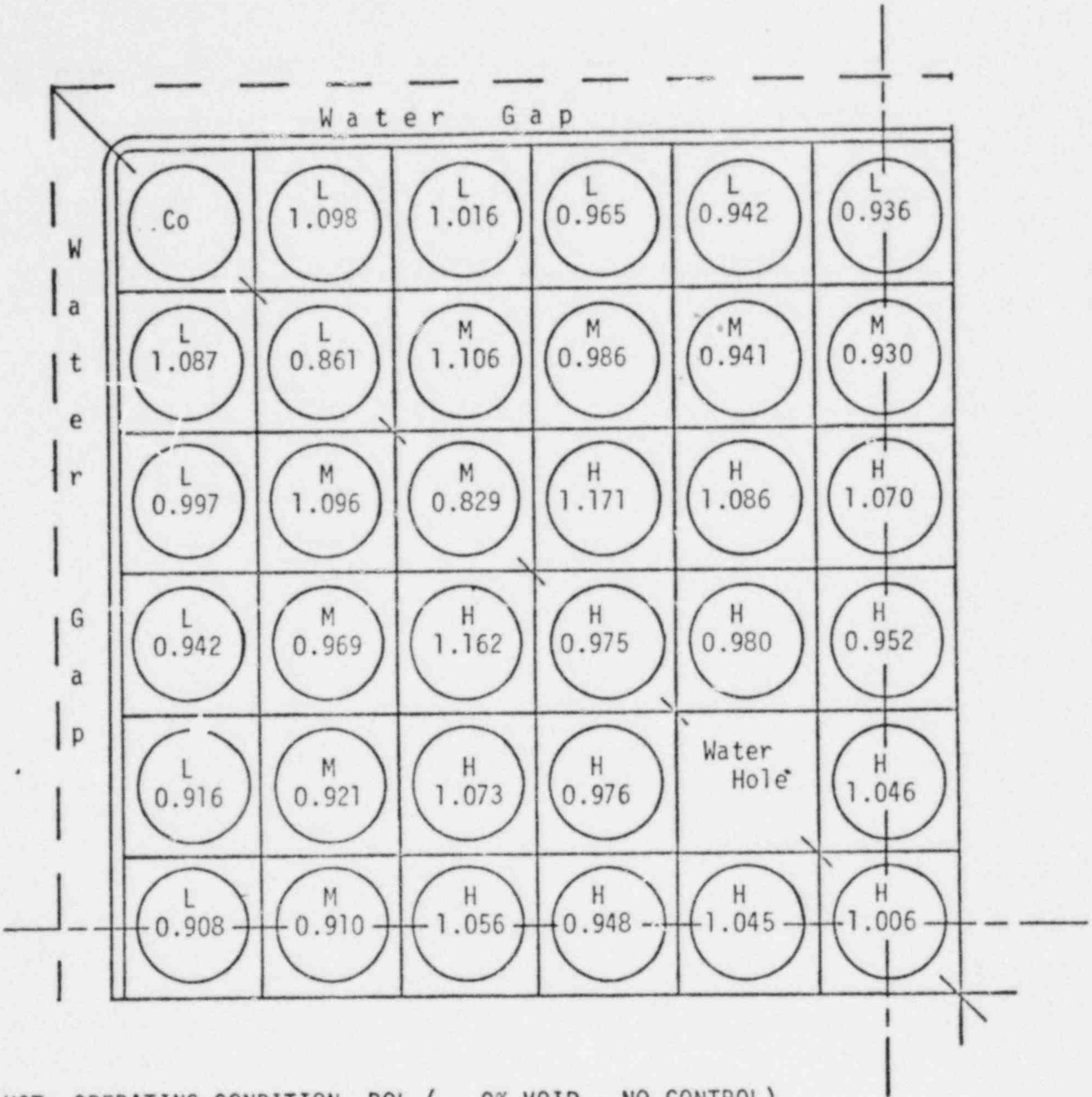
POOR ORIGINAL

FIGURE 2

LOCAL POWER DISTRIBUTION* (NORM TO 113 RODS)

NFS MIXED OXIDE (P_uO_2-UO_2) DEMONSTRATION ASSEMBLY

FOR BIG ROCK POINT



Co	L 1.098	L 1.016	L 0.965	L 0.942	L 0.936
L 1.087	L 0.861	M 1.106	M 0.986	M 0.941	M 0.930
L 0.997	M 1.096	M 0.829	H 1.171	H 1.086	H 1.070
L 0.942	M 0.969	H 1.162	H 0.975	H 0.980	H 0.952
L 0.916	M 0.921	H 1.073	H 0.976	Water Hole	H 1.046
L 0.908	M 0.910	H 1.056	H 0.948	H 1.045	H 1.006

* @ HOT, OPERATING CONDITION, BOL (0% VOID - NO CONTROL)

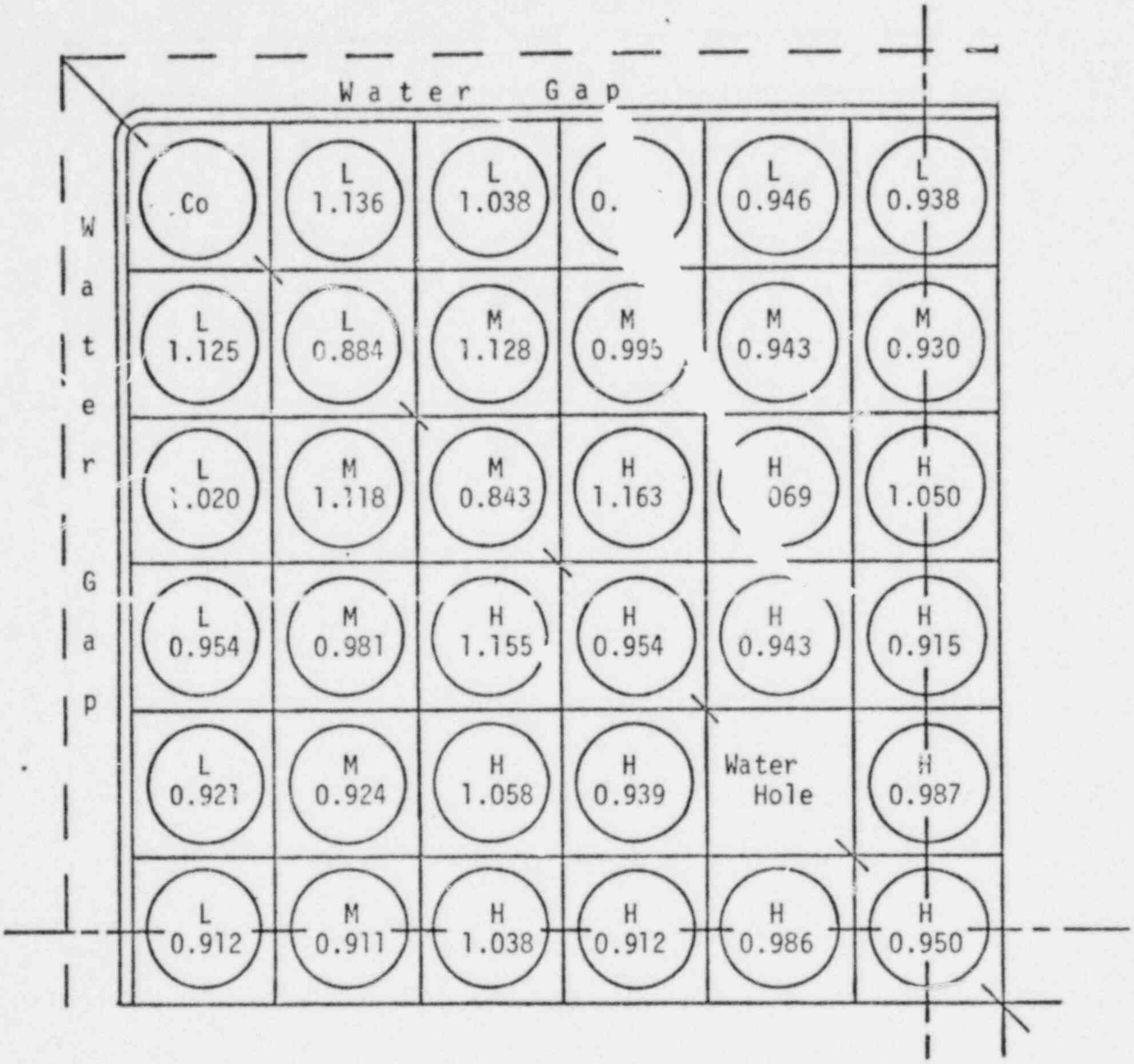
ASSEMBLY AVERAGE BURNUP = 0.0

FIGURE 3

LOCAL POWER DISTRIBUTION* (NORM TO 113 RODS)

NFS MIXED OXIDE (P_uO_2-UO_2) DEMONSTRATION ASSEMBLY

FOR BIG ROCK POINT



Co	L 1.136	L 1.038	0.	L 0.946	L 0.938
L 1.125	L 0.884	M 1.128	M 0.995	M 0.943	M 0.930
L 1.020	M 1.118	M 0.843	H 1.163	H 069	H 1.050
L 0.954	M 0.981	H 1.155	H 0.954	H 0.943	H 0.915
L 0.921	M 0.924	H 1.058	H 0.939	Water Hole	H 0.987
L 0.912	M 0.911	H 1.038	H 0.912	H 0.986	H 0.950

* HOT, OPERATING CONDITION (25% VOID - NO CONTROL)

ASSEMBLY AVERAGE BURNUP = 0.0

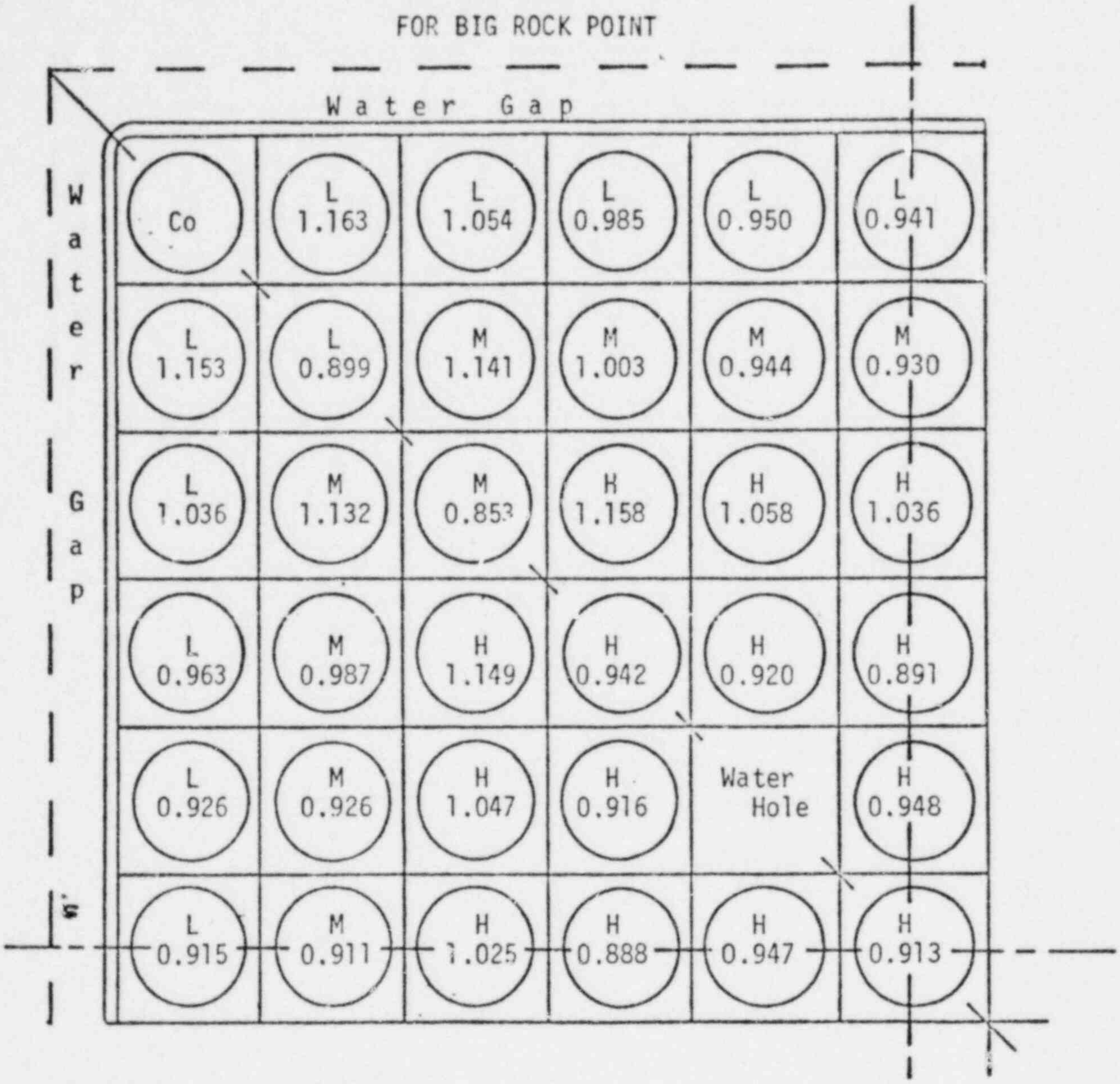
POOR ORIGINAL

FIGURE 4

LOCAL POWER DISTRIBUTION* (NORM TO 113 RODS)

NFS MIXED OXIDE (P_uO_2-UO_2) DEMONSTRATION ASSEMBLY

FOR BIG ROCK POINT



Co	L 1.163	L 1.054	L 0.985	L 0.950	L 0.941
L 1.153	L 0.899	M 1.141	M 1.003	M 0.944	M 0.930
L 1.036	M 1.132	M 0.853	H 1.158	H 1.058	H 1.036
L 0.963	M 0.987	H 1.149	H 0.942	H 0.920	H 0.891
L 0.926	M 0.926	H 1.047	H 0.916	Water Hole	H 0.948
L 0.915	M 0.911	H 1.025	H 0.888	H 0.947	H 0.913

* @ HOT, OPERATING CONDITION, BOL (40% VOID - NO CONTROL)

ASSEMBLY AVERAGE BURNUP = 0.0

POOR ORIGINAL