CONSUMERS POWER COMPANY

Docket 50-155

Request for Change to the Technical Specifications

License DPR-6

For the reasons hereinafter set forth, it is requested that the Technical Specifications contained in Provisional Operating License DPR-6, Docket 50-155, be changed as described in Section I, below.

I. <u>Changes</u>

A. Add or replace the following:

1. Table 5.1 (Page 34a).
2. Page 41b.
3. Table 1 (Page 43).
4. Table 2 (Page 43a).
5. Table 8.2 (Page 91).

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TABLE 5.1

(Additional Information)

See Page 34

General	Reload G-1U	Reload G-3
Geometry, Fuel Rod Array	11 x 11	11 x 11
Rod Pitch, Inches	0.577	0.577
UO_2 Rods	109	113
Cobalt - Bearing Corner Rods	4	0
Gadolinium - Bearing UO_2 Rods	4	4
Inert Spacer Capture Rod (Zr-2)	1	1
Zircaloy Rods	3	3
Spacers per Bundle	3	3
Fuel Rod Cladding		
Material	Zr-2	Zr-2
Wall Thickness, Inches	0.034	0.034
Fuel Rods		
Outside Rod Diameter, Inches	0.449	0.449
Fuel Stacked Density, Percent Theoretical	91.6	91.5
Active Fuel Length, Inches Standard Rod	70	70
Fill Gas	Helium $\geq$95%	Helium $\geq$95%

FUEL BUNDLE SCHEMATIC

G-3 RELOAD FUEL

(TO BE SUPPLIED)

TABLE 1

	Reload E-G and Modified E-G F, J-1 & J-2	Reload G	Reload G-1U	Reload G-3
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	395,000	407,000	392,900
Maximum Steady State Heat Flux, Btu/h-ft^2	410,000	324,000	333,600	322,100
Maximum Average Planar Linear Heat Generation Rate, Steady State, kW/ft	***	***	***	***
Stability Criterion: Maximum Measured Zero-to-Peak Flux Amplitude, Percent of Average Operating Flux	20	20	20	20
Maximum Steady State Power Level, MW_t	240	240	240	240
Maximum Value of Average Core Power Density @ 240 MW_t, kW/L	46	46	46	46
Nominal Reactor Pressure During Steady State Power Operation, psig	1335	1335	1335	1335
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	6×10^6	6×10^6

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 on correlation given in "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," by J M Healzer, J E Hench, E Janssen and S Levy, September 1966 (APED 5286 and APED 5286, Part 2).

**Based on Exxon Nuclear Corporation Synthesized Hench Levy.

***To be determined by linear extrapolation from Table 2 attached.

TABLE 2

MAPLHGR (kW/Ft)

Planar Average Exposure (MWd/STM)	Modified F	F, E-G, J-1, J-2	Reload G and NFSDA	Reload G-1U	Reload G-3
0	-	-	6.453	6.491	6.554
200	9.5	9.4	-	-	-
214	-	-	6.750	6.758	-
216	-	-	-	-	6.807
437	-	-	6.887	6.888	-
443	-	-	-	-	6.973
884	-	-	-	6.960	-
885	-	-	6.978	-	-
893	-	-	-	-	7.033
1,758	-	-	6.929	-	-
1,769	-	-	-	6.970	-
1,773	-	-	-	-	6.984
3,494	-	-	6.885	-	-
3,509	-	-	-	-	6.913
3,545	-	-	-	6.983	-
5,000	9.9	9.7	-	-	-
6,939	-	-	6.838	-	-
6,970	-	-	-	-	6.865
7,085	-	-	-	6.978	-
10,000	9.9	9.7	-	-	-
10,422	-	-	6.847	-	-
10,482	-	-	-	-	6.882
10,690	-	-	-	7.019	-
13,938	-	-	6.867	-	-
14,019	-	-	-	-	6.904
14,355	-	-	-	7.069	-
15,000	9.8	9.6	-	-	-
20,000	8.7	8.6	-	-	-
21,022	-	-	6.905	-	-
21,194	-	-	-	-	6.958
21,843	-	-	-	7.171	-
25,000	8.4	8.3	-	-	-
27,778	-	-	6.843	-	-
28,035	-	-	-	-	6.903
29,084	-	-	-	7.161	-
34,013	-	-	6.703	-	-
35,147	-	-	-	-	6.923
35,322	-	-	-	6.958	-

TABLE 8.2

	EEI UO_2-PuO_2	Centermelt Intermediate	Centermelt 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 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 Average Planar Linear Heat Generation Rate, Steady State, kW/Ft	**	**	**	(Refer to Table 2, Page 43a)
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
Nominal Reactor Pressure During Steady State Power Operation, psig	1,335	-	-	1
Minimum Recirculation Flow Rate, Lb/h (Except During Pump Trip Tests or Natural Circultion Tests as Outlined in Sec 8)	6×10^6	-	-	6×10^6
Number of Bundles:				
Pellet UO_2	-	1	3	-
Power 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.
**No longer used in reactor.

II. DISCUSSION

1.0 INTRODUCTION AND SUMMARY

The purpose of this proposed change is to allow the use of an all uranium fuel with acceptable ECCS performance characteristics in the Big Rock Point reactor and to delete the Technical Specifications limitation on the design burnup of fuel bundles.

Presently, licensed fuel types are 11 x 11 all uranium, 11 x 11 mixed-oxide and 9 x 9 all uranium. The 11 x 11 all uranium fuel (denoted G-1U) was used as reload fuel at the last reloading. The proposed 11 x 11 all uranium reload fuel (denoted G-3) is very similar to the G-1U fuel assemblies with three basic differences. For G-3 assemblies, the four corner cobalt target rods have been replaced with fueled rods. There have been changes made to the bundle enrichment distribution which reduces the overall bundle enrichment from 3.88% to 3.14%. The placement of the gadolinia poison pins has been altered for better peaking characteristics. These effects have been accounted for in the subsequent analyses presented.

This submittal contains information concerning fuel system design, nuclear design, thermal hydraulic design and accident and transient analysis as recommended by the "Guidance for Proposed License Amendments Relating to Refueling." Every feasible attempt to present the information requested by the guide has been made. In general, the major difficulty in providing this data was in formulating a suitable "reference cycle" as defined in the guide. For Sections 4 and 6, the reference cycle used was Cycle 14, specifically, the Reload G-1U fuel. For Section 7, there was no single suitable reference cycle available. Thus, the latest analysis found acceptable by the Commission was used as the reference cycle for each specific accident or transient. In many cases this dated back to the FHSR. Finally, for Section 5, since many of the parameters required by the guide were not routinely calculated or submitted in previous reload licensing submittals, no reference cycle is given. It is Consumers Power Company's intent to utilize Cycle 15 as the reference cycle for Section 5 for subsequent reload licensing submittals.

2.0 OPERATING HISTORY

Cycle 14 power production began on July 28, 1976 following a refueling outage. The core loading consisted of 38 - 9 x 9 fuel assemblies and 46 - 11 x 11 fuel assemblies, with residual fuel assemblies relocated in the core to provide adequate shutdown margin and acceptable cycle power peaking. The off-gas release rate stabilized following start-up at approximately 750 μCi/s (corrected for specific gravity). The plant has operated since that time at power levels ranging from 206 to 216 MW_t. This reduced power level resulted from reaching Technical Specifications MAPLHGR limits, primarily for the F fuel. The off-gas release rate for the latter portion of the cycle is averaging approximately 700 to 800 μCi/s. This is the lowest off-gas release rate of any cycle and is attributed primarily to the removal of copper based materials from the primary system several cycles ago, and the subsequent discharge of fuel bearing copper based crud. Cycle 14 was originally designed for an energy production of 61 GWD; however, due to the extended operating period, energy production is now expected to exceed slightly this figure with a power coastdown at the end of the cycle.

A summary of Cycle 14 start-up and tests performed at the beginning of the cycle is contained in Special Report No 24 dated November 24, 1976.

3.0 GENERAL DESCRIPTION

Cycle 15 is designed to produce a target energy of 72 GWD. This corresponds to a cycle length of approximately 325 days of power operation at 220 MW_t. The projected core loading for this cycle, including cobalt distribution, is shown in Figure 3-1. This loading scheme is subject to minor changes, depending upon the results of fuel sipping conducted during the June 1977 outage, with the goal of making effluent releases as low as reasonably achievable. Figure 3-2 details the fuel rod arrangement, the initial fuel enrichment and the gadolinium distribution and concentration for the G-3 fuel. The gadolinium is designed to burn up in a single cycle; thus, only the new assemblies contain significant amounts of burnable poison. Figure 3-3 is provided to indicate the beginning of life fuel burnup distribution for Cycle 15.

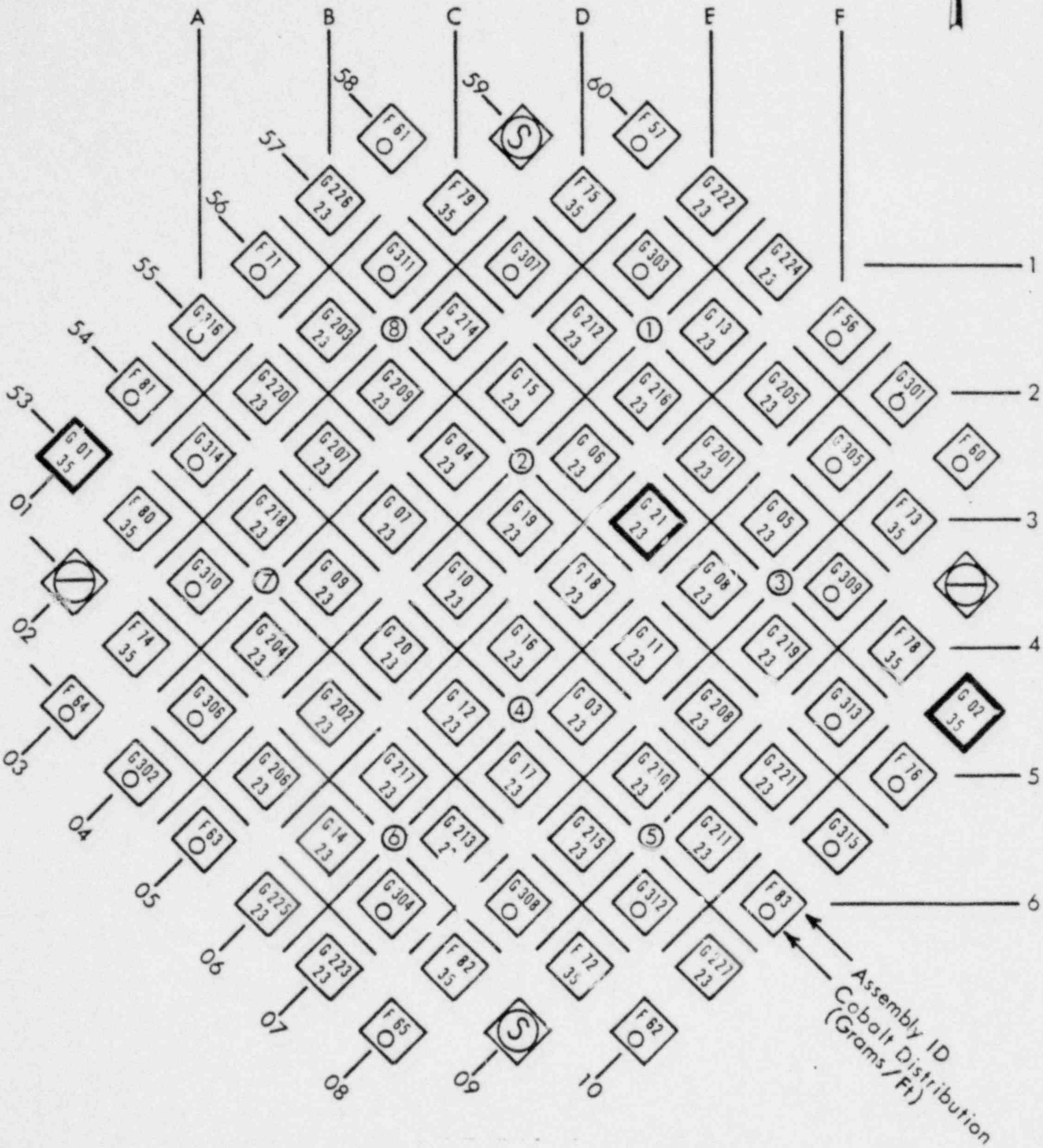
The Cycle 15 fuel loading pattern has been designed to incorporate 180° rotational symmetry throughout the core. The fuel distribution has been developed to comply with Technical Specifications limitations and safety analysis criteria. These limits and criteria include MAPLHGR, minimum critical heat flux ratios, maximum heat flux, maximum control rod worth and minimum shutdown margin among others.

Figure 3-1

CYCLE 15

BIG ROCK POINT

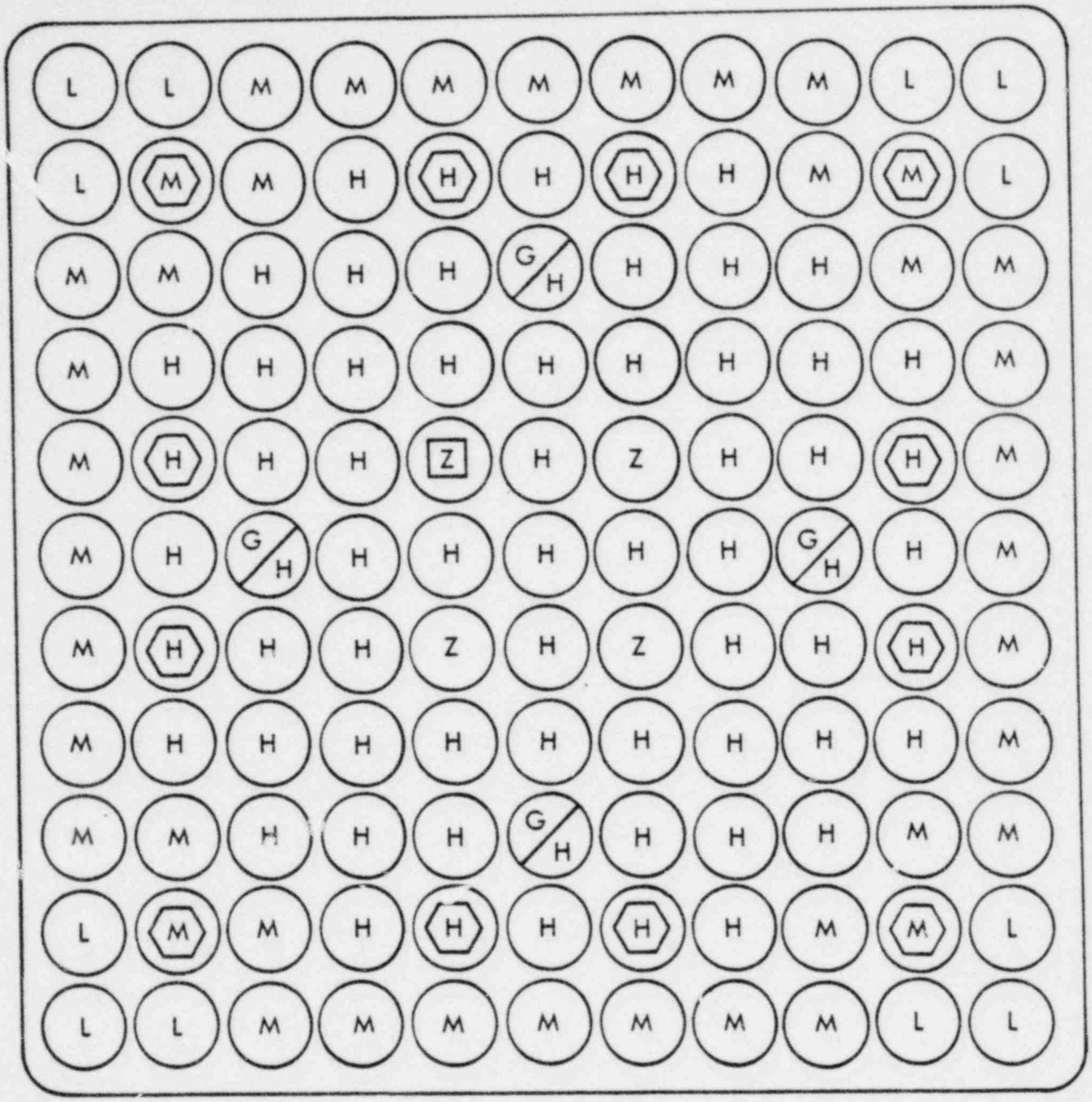
CORE CONFIGURATION



Darkened boxes indicate test assemblies.

Figure 3-2

FUEL ROD ARRANGEMENT – BIG ROCK POINT RELOAD G3



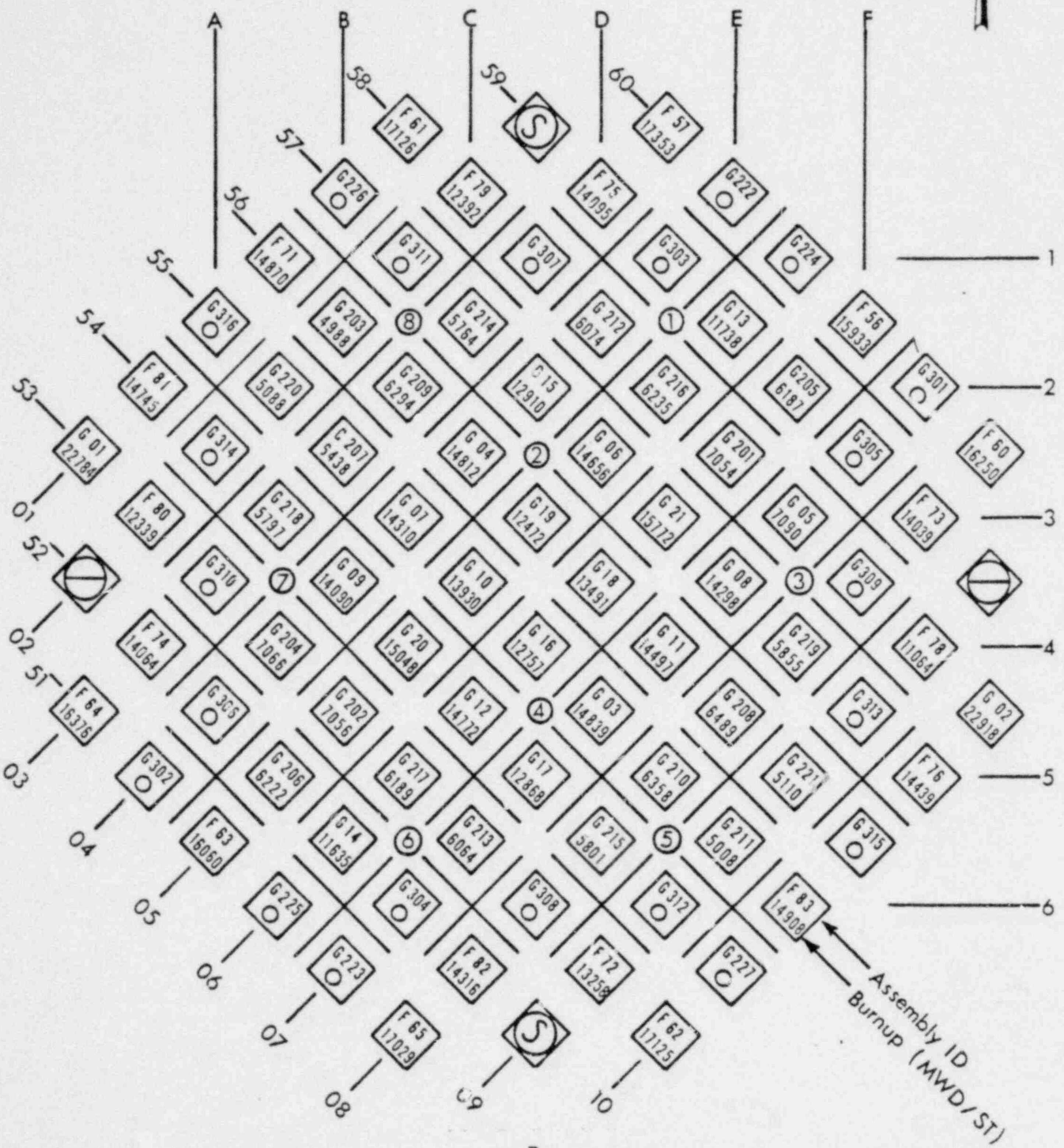
	NUMBER OF RODS	DESCRIPTION
Z	3	INERT RODS
L	12	1.50 wt% ^{235}U
M	40	2.52 wt% ^{235}U
H	61	3.82 wt% ^{235}U
G/H	4	3.82 wt% ^{235}U + 1.25 wt% Gd_2O_3
(hexagon)	12	TIE RODS
(square)	1	INERT SPACER CAPTURE ROD

Figure 3-3

CYCLE 15

BIG ROCK POINT

CORE CONFIGURATION
BOL BURNUP DISTRIBUTION



4.0 FUEL SYSTEM DESIGN

The G-3 reload fuel design for Cycle 15 has mechanical, thermal hydraulic and neutronic performance characteristics similar to the G-1U reload fuel design for Cycle 14, the reference cycle. Both G-1U and G-3 fuels employ an 11 x 11 rod matrix with four inert rods; however, G-1U fuel design incorporated four corner cobalt rods necessitating a slightly higher enrichment than G-3 fuel. A complete description of the mechanical, thermal hydraulic and neutronic characteristics of the G-1U fuel was presented in our letter dated October 13, 1975.

4.1 Fuel Design

As discussed previously, the major fuel design change for Reload G-3 fuel when compared to Reload G-1U fuel is the elimination of the four cobalt target rods and the reduction of the overall U^{235} enrichment. Table 4.1-1 lists the design parameters for both the G-3 proposed fuel and the G-1U reference fuel. Table 4.1-2 delineates the fuel inventory at BOC for Cycle 15. It consists of the fuel type, number of assemblies, number of cycles in core, and initial BOC enrichment, density and average burnup. The G-3 fuel is designed to be free-standing throughout its life in core, which is consistent with previous G reload fuels and, like other G fuels, G-3 was initially filled at nominal atmospheric pressure.

TABLE 4.1-1

DESIGN PARAMETERS FOR BIG ROCK POINT

G-3 VS G-1U All Uranium Designs

	Reload G-3	G-1U
Fuel Assembly		
Rod Array	11 x 11	11 x 11
Rod Pitch, In	0.577	0.577
Water-to-Fuel Volume Ratio of Lattice	2.60	2.69
Heat Transfer Area, Ft^2	80.23	77.48
Number of Spacer Grids	3	3
Rods per Bundle		
Cobalt Target	0	4
Low Enrichment Urania	12	16
Intermediate Enrichment Urania (Nonpoison)	40 (Includes 4 Tie Rods)	32 (Includes 4 Tie Rods)
High Enrichment Urania	61 (Includes 8 Tie Rods)	61 (Includes 8 Tie Rods)
Poison (Urania-Gadolinia)	4	4
Spacer Capture (Nonfueled)	1	1
Inert Rod (Nonfueled)	3	3
	121	121
Fuel Rod		
Diametral Pellet-to-Clad Gap, In	0.0095	0.0095
Overall Fuel Rod Length, In	78.501	78.501
Active Fuel Length, In	70.00	70.00
Plenum Volume Length	3.901	3.901
Fill Gas	Helium	Helium

TABLE 4.1-1 (Contd)

	Reload G-3	G-1U
Fuel Rod Weights		
UO_2 Rods (Total Ceramic Grams)	1248	1248
Poison Rods (Total Ceramic Grams)	1242	1242
Fuel Pellet		
Material	Sintered UO_2	Sintered UO_2
Diameter, In	0.3715	0.3715
Length, In	0.300	0.300
Density, % Theoretical (TD = 10.96 gm/cm^3)	93.5	93.5
Initial Enrichment		
Low Enrichment Rods (Wt% U-235)	1.50	2.30
Intermediate Enrichment Rods (Wt% U-235)	2.52	3.20
High Enrichment Rods (Wt% U-235)	3.82	4.60
UO_2-1.20 Wt% Gd_2O_3 Poison (Wt% U-235)	3.82	4.60
Average for Bundle	3.14	3.88
Dishing	Both Ends	Both Ends
Dish Volume, % of Undished Pellet Volume	2	2
*EBC of Impurities, ppm	< 4	< 4
Poison Pellet		
Material	1.20 Wt% Gd_2O_3 in UO_2	1.20 Wt% Gd_2O_3 in UO_2
Diameter, In	0.3715	0.3715
Length, In	0.300	0.300

*EBC = Equivalent Boron Content

TABLE 4.1-1 (Contd)

	Reload G-3	G-1U
Poison Pellet (Contd)		
Density, % of Theoretical ($TD = 10.91\ gm/cm^3$)	93.5	93.5
Dishing	Both Ends	Both Ends
Dish Volume, % of Undished Pellet Volume	2	2
EBC of Impurities, ppm	< 7	< 7
Cladding		
Material	Zircaloy-2 Cold Worked and Stress Relieved	Zircaloy-2, Cold Worked and Stress Relieved
Outside Diameter, In (After Etching)	0.449	0.449
Inside Diameter, In	0.3810	0.3810
Nominal Wall Thickness, In (After Etching)	0.034	0.034
Minimum Wall Thickness, In (After Etching)	0.032	0.032
EBC, Total, Including Impurities	< 40	< 40
Insulator Pellet		
Material	Alumina (Al_2O_3)	Alumina (Al_2O_3)
Diameter, In	0.365	0.365
Length, In	0.200	0.200
EBC, Total, Including Impurities, Ppm	< 76	< 76

TABLE 4.1-2

FUEL INVENTORY TABLE

Cycles in Core	No of Assemblies	Fuel Type	Initial Enrichment (w/o) U	Initial Enrichment (w/o) Pu	Initial Fuel Stacked Density (% Theoretical)	Average BOC Burnup (MWD/ST)
5	6	F	3.52	0	94	16,641
5	2	F-Modified	3.51	0	94	16,703
5	2	G	3.08	0.90	91.5*	22,851
4	12	F-Modified	3.51	0	94	13,711
3	18	G	3.08	0.90	91.5*	13,828
2	8	G	3.08	0.90	91.5*	6,518
2	14	G-1U	3.88	0	91.6*	5,863
1	6	G-1U	3.88	0	91.6*	0
1	16	G-3	3.14	0	91.5*	0

*Pellets 2% Dished

4.2 Mechanical Design

The mechanical design of the Reload G-3 fuel is consistent with the reference G-1U fuel with the exception of the upper tie plate. This plate was modified slightly in the Reload G-3 design to provide standard fuel rod location holes in each corner of the plate replacing the locking slots utilized by the cobalt target rods. By letters dated June 16, 1972 and October 13, 1975, mechanical design analyses for Reload G and Reload G-1U fuels were submitted; these analyses are applicable for Reload G-3 fuel. Table 4.2-1 describes the G-3 fuel assembly components, their purpose and composition.

TABLE 4.2-1

DESCRIPTION OF TYPE G-3 FUEL ASSEMBLY COMPONENTS

Item	Purpose	Material/Rationale
Upper Tie Plate and Handle	Maintains fuel rod array. Provides lifting fixture.	Cast SS, Grade CF-3 - Strength - Corrosion Resistance
Compression Springs	Accommodates differential fuel rod lengths and supports upper tie plate.	Inconel X-750 - Corrosion Resistance - Strength at Operating Conditions. - Springs loaded high enough to minimize fretting and low enough not to cause excessive rod bowing.
Fuel Rod End Cap Welds	Provides high quality seal of fuel rods.	TIG - Fillet Head - Excellent penetration. - Extremely low porosity. - High strength integrity.
Plenum Spring	Maintains compact fuel column during handling and shipping.	Inconel X-750 Wire - Withstand autoclave treatment. - Maintain spring load during reactor operation.
Plenum Chamber	Collects fission gases. Provides space for axial expansion of fuel.	- Assures that gas pressure will not overstress cladding.
Cladding	Contains fission gases and keeps water from contacting fuel.	Zircaloy-2 - Minimize neutron absorption. - Cladding is autoclaved for prefilming for corrosion resistance and to provide a corrosion-proof test.
Pellet Cladding Gap	Provides clearance between fuel and cladding.	- Designed to maximize fuel rod fissile content and to minimize pellet-clad interaction from swelling expected at high burnup.

TABLE 4.2-1 (Contd)

Item	Purpose	Material/Rationale
Insulator Pellet	Reduces pellet-nonpellet interface temperature.	Al_2O_3 - Maintains temperature below those causing excessive stress levels and below those of concern with metal-fuel reaction. - Controls hydride preceiptation.
Atmosphere	Heat transfer medium between pellet and clad.	Helium - Good heat transfer characteristics. - Provides an easy and reliable leak detection monitoring means.
Spacers	Maintains correct rod-to-rod spacing.	Zircaloy-4 Frame, Inconel 718 Springs - Corrosion minimized. - Mechanical stability. - Spring loads on cladding must be sufficient to minimize lateral and rotational movement of fuel rod but must not cause excessive cladding or spring stress. - Spacer must not cause excessive coolant flow resistance.
Inert Rods	Displaces the highest peak clad temperature rods under LOCA conditions and provide a radiation sink.	Zircaloy-2 Cladding End Caps Filler - Corrosion resistance. - Low absorption cross section.
Bottom Tie Plate	Maintains fuel rod array and distributes coolant to fuel rods.	Cast SS, Grade CF-3 - Strength. - Corrosion resistance.
Spacer Capture Rod	Maintains correct longitudinal position of spacers.	- Continuous clad and formed Zircaloy sheet stock connectors.
Tie Rod	Provides structural skeleton of assembly by securing the upper and lower tie plates.	Zr-2 clad fuel rods with end fittings for attachment to tie plates.

4.3 Thermal Design

The design basis for the thermal performance of Reload G-3 fuel is identical to that described in our submittals dated June 16, 1972 and October 13, 1975 for Reload G and G-1U, respectively.

4.4 Chemical Design

The adequacy of materials selected for the chemical fuel design has been demonstrated through the excellent performance of Exxon Nuclear Fuels to date. Past irradiation tests for assemblies similar to G-3 have produced no fuel failures or degradation due to incompatibility with the reactor water chemistry. Results of the post-irradiation examinations of fuel assemblies, including those of the G design, are contained in Special Report No 24 dated November 24, 1976.

5.0 NUCLEAR DESIGN

The fresh fuel to be used for Cycle 15 is Exxon Nuclear's Type G-1U and Type G-3. Fourteen bundles of Type G-1U fuel are currently used in Cycle 14.

Important differences between the reload G-1U and reload G-3 fuel bundle designs are the replacement of the four corner cobalt target rods with low enrichment fuel rods, a change in the gadolinia poison pin locations and changes in the bundle enrichment distribution which reduce the bundle average enrichment from 3.88% for G-1U fuel to 3.14% for G-3 fuel. The effects of these changes on local peaking factor and fuel bundle reactivity have been accounted for in computing the core physics characteristics.

5.1 Physics Characteristics

As discussed earlier, previous reload licensing submittals for the Big Rock Point Plant did not include many of the physics parameters requested in the "Guidance for Proposed License Amendments Relating to Refueling," thus these parameters are not available for previous cycles and consequently no reference cycle, meeting the criteria of "reference cycle" as defined in the guide, is available. It is the intent of Consumers Power Company to calculate the parameters required by the guide for Cycle 15 and to utilize them in subsequent licensing submittals as the reference cycle for physics parameters. These are included as Table 5.1-1 and Figure 5.1-1.

Table 5.1-1 includes the full power doppler coefficient, delayed neutron fraction, void coefficient and total peaking factors for both the Beginning of Cycle and end of Cycle Conditions The maximum reactivity for in sequence rod drop worth is 2.29% $\Delta k/k$ for BOC and 1.74% $\Delta k/k$ for EOC, both cases well below the Technical Specification limit of 2.5% $\Delta k/k$.

The Cycle 15 core can be maintained subcritical in the most reactive condition throughout the operating cycle with the most reactive rod fully withdrawn and all other rods fully inserted. The Technical Specification concerning shutdown margin is 0.3% $\Delta k/k$ which is significantly less than the BOC and EOC values for shutdown margin listed in Table 5.1-1. Figure 5.1-1 is the full shutdown margin curve for Cycle 15.

The limiting scram reactivity curve for Cycle 15 is provided in Figure 5.1-2. This curve is most limiting at EOC when the control rod density in the critical rod configuration is the lowest. Figure 5.1-2 is a comparison of Cycle 15 scram reactivity to the bounding curve of Cycle 11. The Cycle 11 curve is considered limiting since it was utilized in the reference analysis of the rod drop accident, the accident most sensitive to scram insertion characteristics.

Two physics parameters requested by the guide are not provided in this submittal. The moderator temperature coefficients are not considered in any accident or transient analysis, and essentially have no meaning for a boiling water reactor other than in its initial heatup prior to power operation. However, these will be calculated prior to power operation for Cycle 15. The other parameter not provided is the worth of the standby liquid control system. The worth of the standby liquid control system is being evaluated for this cycle.

TABLE 5.1-1

Parameter	BOC	EOC
Doppler Coefficient ($\Delta k/k/\%$ Power)	-7.06×10^{-5}	-7.65×10^{-5}
Maximum Radial x Axial Peaking Factor	1.697	1.668
Maximum Radial x Axial x Local Peaking Factor	2.479	2.306
Maximum Rod Worth ($\% \Delta k/k$)	2.29	1.74
Delayed Neutron Fraction	.00606	.00588
Shutdown Margin ($\% \Delta k/k$)	2.21	6.21
Void Coefficient ($\Delta k/k$/Unit Void)	-.1663	-.1127

5.2 Analytical Input

Reactor power distributions, reactivities, reactivity coefficients, fuel burnup and margin to thermal limits are calculated with the GROK computer program. GROK is a three-dimensional coarse mesh reactor simulator with thermal hydraulic feedback and is a derivative of the FLARE program. (D L Delp, et al, "FLARE, A THREE-DIMENSIONAL BOILING WATER REACTOR SIMULATOR," GEAP-4598, July 16, 1964.) The neutronics parameters k_∞, M^2 and local peaking factor, as a function of local operating state as computed by the fuel designer, are the major inputs. Algorithms have been included which calculate peak heat flux, MCHFR, MAPLHGR and theoretical flux wire traces for comparison with reactor measurements.

5.3 <u>Changes in Nuclear Design</u>

There are no changes in core design features, calculational methods, data or information relevant to determining important nuclear design parameters, other than those mentioned above, for Cycle 15.

FIGURE 5.1-1
SHUTDOWN MARGIN VS EXPOSURE
CYCLE 15

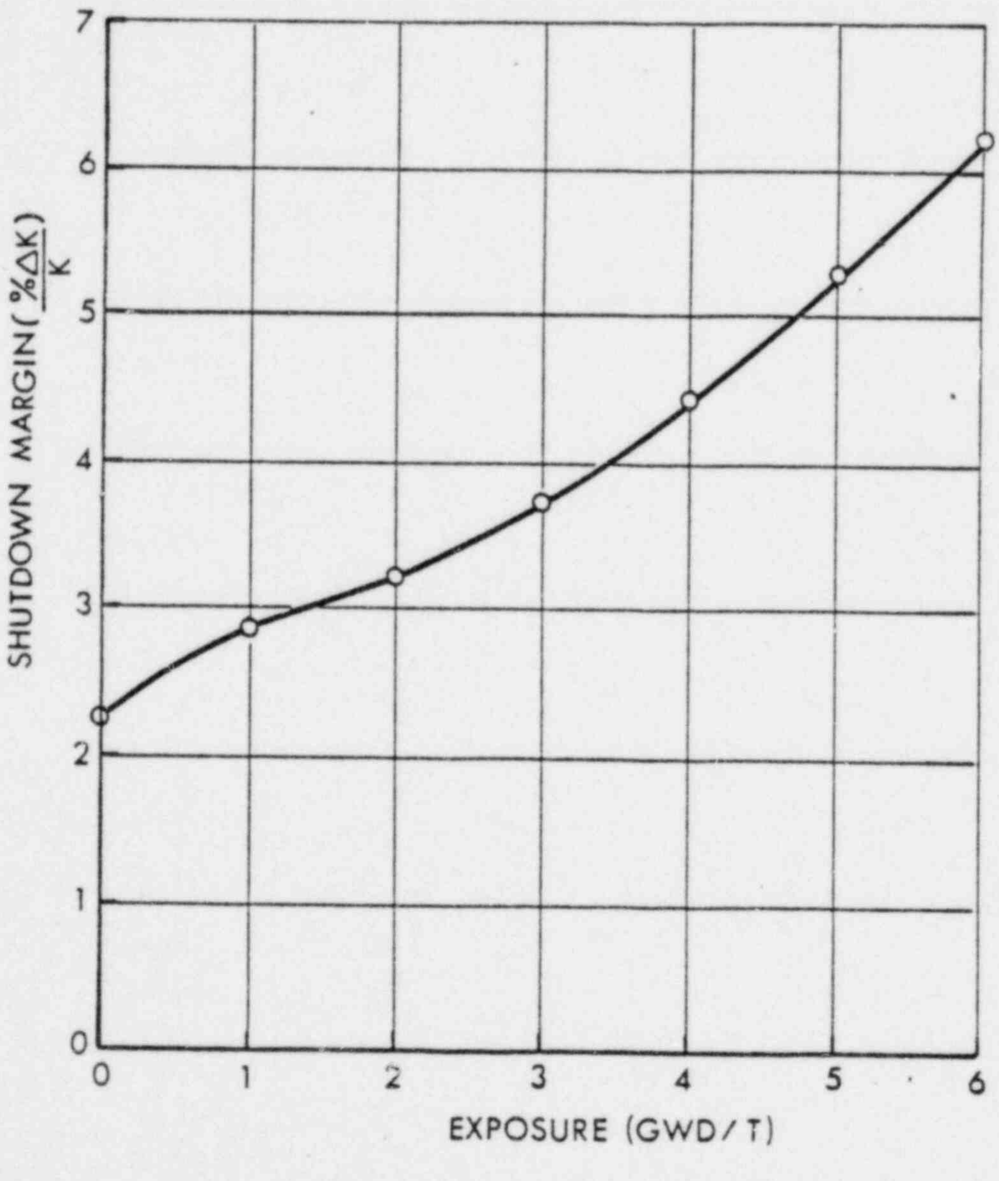
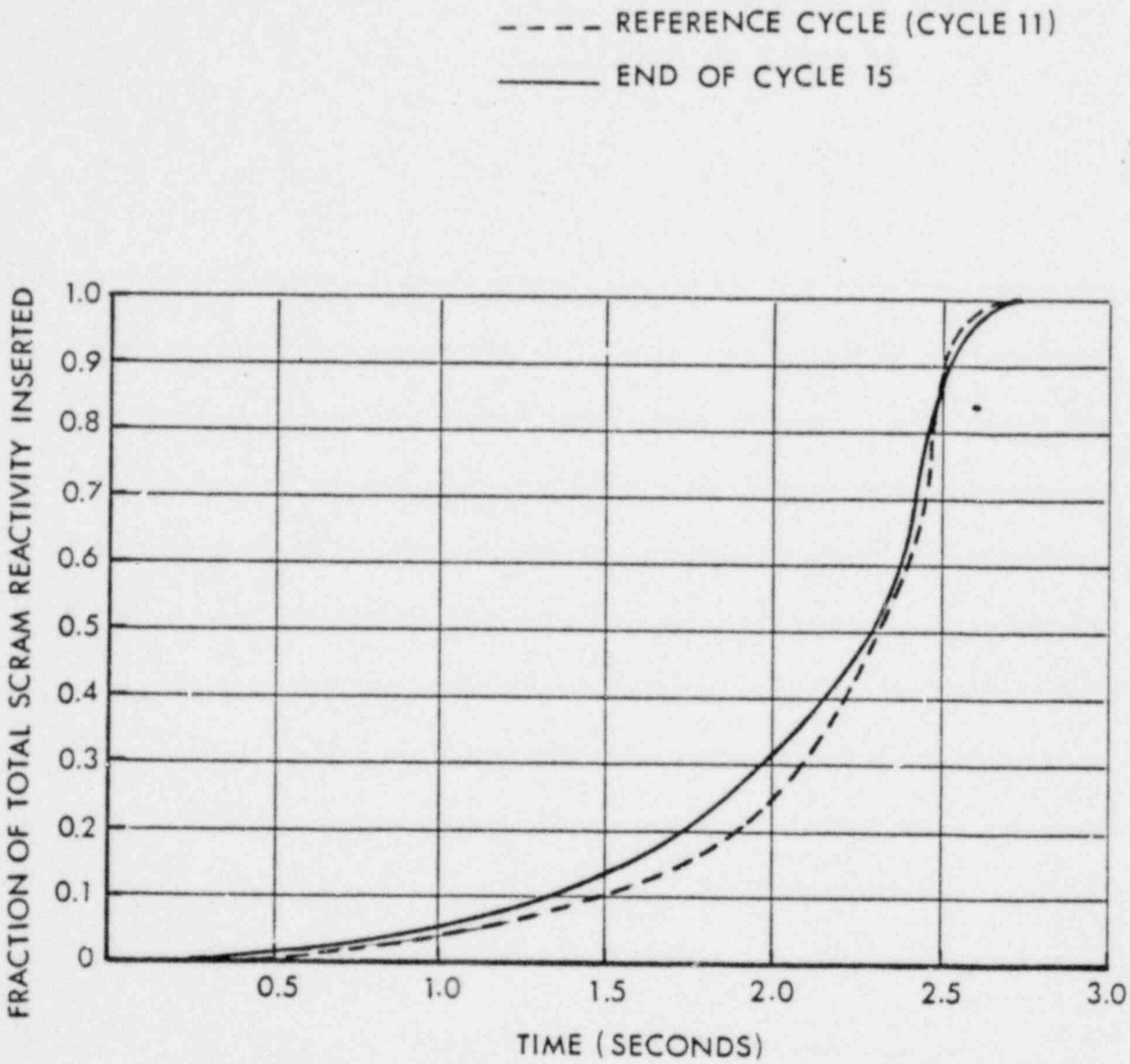


Figure 5.1-2

BIG ROCK POINT PLANT
SCRAM CURVE



6.0 THERMAL HYDRAULIC DESIGN

The hydraulic design of the reload G-3 assemblies is identical to that of the reload G-1U assemblies. The thermal performance of reload G-3 fuel differs slightly from reload G-1U in that the average rod power has changed since the four cobalt corner rods have been replaced with low enrichment UO_2 fuel rods. This has led to a decrease in average rod power, consequently it has also led to a decrease in maximum overpower clad and fuel temperatures and an increase in the overpower minimum critical heat flux ratio. A comparison of the thermal hydraulic parameters for complete cores of reload G-3 and reload G-1U fuel assemblies is contained in Table 6-1.

Although Cycle 15 will be predominantly 11 x 11 reload G-type fuel assemblies, approximately 25% of the core will be 9 x 9 reload F fuel assemblies. However, the maximum radial peaking factor for the reload F fuel for Cycle 15 is expected to be 0.84. This will result in a 122% rated power minimum critical heat flux ratio of greater than 2.0 for small and large orifice channel locations. Therefore, minimum critical heat flux ratio units for reload F fuels will not limit core power operations for Cycle 15.

TABLE 6-1
Thermal Hydraulic Parameters
(Core Contains All of Each Type Fuel)

	Reload G-3	Reload G-1U
Core Conditions		
Reference Design Thermal Output, $(MW_t)/(Btu/h)$	$240/8.191 \times 10^8$	
Total Flow Rate, Lb/h	12.3×10^6	
Effective Flow Rate for Heat Transfer, Lb/h	9.9×10^6	
System Pressure, Nominal in Steam Dome, Psia	1350	
Assembly Description		
Rod Diameter, Inches	0.449	0.449
Rod Pitch, Inches	0.577	0.577
Number of Active Rods	117	113
Total Fuel Length per Assembly, Feet	632.5	659.2
Heat Transfer Area, Ft^2.	80.23	77.48
Flow Area, Ft^2/In^2	0.163/23.44	0.163/23.44
Design Power Peaking Factors		
Fraction Generated in Fuel, %	97.0	96.6
Fuel Assembly Power Factor	1.45	1.45
Local Peaking Factor	1.20	1.20
Axial Peaking Factor	1.51	1.51
Engineering Heat Flux Factor	1.04	1.04
Assembly Thermal Performance		
Maximum Heating Rate, kW/ft, at 22% Overpower	13.53	14.0
Maximum Heating Rate, kW/ft, at Rated Power	11.09	11.44
Average Heating Rate, kW/ft	4.06	4.20
Maximum Heat Flux, $Btu/h\text{-}Ft^2$ at 22% Overpower	392,900	407,000
Maximum Heat Flux, $Btu/h\text{-}Ft^2$ at Rated Power	322,100	333,600
Average Heat Flux, $Btu/h\text{-}Ft^2$ at Rated Power	117,900	127,100
Maximum UO_2 Temperature, °F, at 22% Overpower	3879	3900
*Maximum Clad Temperature, °F, at Overpower	745	754
MCHFR at Overpower Conditions Axial Peak at X/L = .45	1.68	1.63
Coolant Subcooling at Core Inlet, Btu/Lb	22.8	22.8
Assembly Hydraulic Performance		
Average Assembly Flow		
Inner Orifice Zone, Lb/h	132,900	132,900
Outer Orifice Zone, Lb/h (24 Assemblies on Periphery of Core)	80,500	80,500
Active Core Flow at Design Power Lb/h	9.9×10^6	9.9×10^6
Hot Assembly Flow at 122% Design Power (Reference Design Flow)	123,000	123,000
Assembly ΔP at Average Design Power (Includes Orifice ΔP)	5.37 Psi	5.37 Psi
Hot Assembly Engineering Enthalpy Rise Factor	1.10	1.10

*Crud-Free Surface

7.0 TRANSIENT AND ACCIDENT ANALYSIS

In order to update this section, the NRC Standard Review Plans, Regulatory Guide 1.70, and the General Electric Standard Safety Analysis Report were thoroughly researched to determine what accidents, transients and limiting design criteria were necessary for a proper review. The "reference cycle" for accident and transient analysis consists of the latest analysis run for each. In many cases these date back to the FHSR, but wherever analyses have been run subsequent to this, they have been used as the reference cycle. The references for this section are contained in Subsection 7.3.

7.1 Transient Analysis

7.1.1 Significant Reactor Kinetics and Fuel Thermal Hydraulic Design Parameters

For each reactor transient considered in previous licensing submittals for the Big Rock Point Plant, the reactor kinetics parameters which control the reactor transient response are shown in Table 7-1. Also shown in Table 7-1 are the reference value and the corresponding Cycle 15 value for each significant parameter. Below is a discussion of the effects that the Cycle 15 values are expected to have on the reactor transient response.

Important to the analysis of the reactor transients is the thermal and hydraulic design of the various fuel bundles comprising the reactor core. All Big Rock Point fuel bundles up to and including the Reload G-3 fuel bundles have been designed to meet the following constraints. (Refer to Section 6)

(1) Minimum critical heat flux ratio (MCHFR) at design overpower (122%) and design peaking factors must be greater than 1.50.

(2) Maximum fuel temperature at design overpower and design peaking factors must be less than the fuel melting temperature.

Given that these constraints are met, the thermal response of each fuel type (ie, MCHFR, peak fuel temperature, peak clad temperature) to a given transient will be as previously predicted or better.

TABLE 7-1

Significant Reactor Kinetics Parameters

Event	Latest Analysis	Important Kinetics Parameter(s)	Reference Value	Nominal Cycle 15 Value
• Loss of External Load With and Without Turbine Bypass (Bounds Main Steam Line Isolation Valve Closure and Loss of Condensor Vacuum)	Reference 1 Page 13	Void Coefficient	BOC: -.20* EOC: -.10* ($\Delta k/k$/Unit Void)	-.1663 -.1127
		Doppler Coefficient	-5.42×10^{-5} $\Delta k/k$/% Power	-7.06×10^{-5}
• Steam Pressure Regulator Failure Resulting in Reduced Steam Flow	Reference 1 Page 10	Void Coefficient	BOC: -.20* EOC: -.10* ($\Delta k/k$/Unit Void)	-.1663 -.1127
		Doppler Coefficient	-5.42×10^{-5} $\Delta k/k$/% Power	-7.06×10^{-5}
• Uncontrolled Rod Withdrawal From Subcritical				
• Cold Start-Up	Reference 1 Page 6	Doppler Coefficient	-1.47×10^{-5} $\Delta k/k/^{\circ}F$	$-.95 \times 10^{-5}$
		Maximum Reactivity Addition	3.9% $\Delta k/k$	1.77%**
		β/ℓ*	175	183
• Hot Start-Up	Reference 1 Page 6	Doppler Coefficient	-1.37×10^{-5} $\Delta k/k/^{\circ}F$	$-.95 \times 10^{-5}$
		Maximum Reactivity Addition	4.2% $\Delta k/k$	2.29%**
		β/ℓ*	175	183
• Uncontrolled Rod With-Drawal at Power	Reference 1 Page 12	Void Coefficient	BOC: -.20* EOC: -.10* ($\Delta k/k$/Unit Void)	-.1663 -.1127
		Doppler Coefficient	-5.42×10^{-5} $\Delta k/k$/% Power	-7.06×10^{-5}

TABLE 7-1 (Contd)

Event	Latest Analysis	Important Kinetics Parameter(s)	Reference Value	Nominal Cycle 15 Value
Inactive Recirculation Pump Start-Up	Reference 1 Page 22	Void Coefficient	BOC: -.20* EOC: -.10* (Δk/k/Unit Void)	-.1663 -.1127
		Doppler Coefficient	-5.42×10^{-5} Δk/k/% Power	-7.06×10^{-5}
Loss of Recirculation Pumps		(This event is reevaluated for each new core loading using the methods described in Appendix B of Reference 2.)		

*Reference 7.

**Reference value is maximum worth for out-of-sequence rod. Cycle 15 value is maximum worth for in-sequence rod.

7.1.2 Pressurization Events (Loss of Load, Turbine Trip, Main Steam Line Isolation Valve Closure, Steam Pressure Regultor Failure)

Pressurization events are characterized by a decrease in voids resulting in a power increase. The more negative void coefficient for BOC conditions tends to maximize the system pressures and powers reached in these events. Since the reference cycle BOC void coefficient is more negative than expected at any time during Cycle 15, the reference cycle analysis conservatively bounds the upcoming cycle.

7.1.3 Inactive Recirculation Pump Start-up or Cold Water Event

Like the pressurization events, this event is characterized by a decrease in voids and a power increase; therefore, the void coefficient is again the important kinetics parameter for this event. Because the reference cycle BOC void coefficient is more negative than expected at any time during Cycle 15, the reference analysis bounds Cycle 15.

7.1.4 Loss of Recirculation Pumps

This event is reevaluated for every core reloading using the method described in Appendix B of Reference 2. Results of this analysis, although presently incomplete, are expected to show, as they have shown for previous cores, that the minimum critical heat flux ratio never falls below 1.5 and, in fact, monotonically increase throughout the critical portion of the transient. Therefore, this event is not limiting for Big Rock Point.

7.1.5 Rod Withdrawal at Power

This event is characterized by increases in core power level and core voids. The void and doppler coefficients are the important kinetics parameters for this event. As noted in Table 7-1 the Cycle 15 doppler coefficient is more negative (ie, more conservative) than was assumed in the reference analysis. In addition, the Cycle 15 void coefficient is more negative than the worst case (EOC) void coefficient for the reference cycle. Therefore, it is concluded that the reference cycle analysis bounds Cycle 15 for this event.

7.1.6 Start-Up Event

The start-up event (or the uncontrolled rod withdrawal from subcritical) was analyzed in Reference 1 for both the cold and hot standby initial conditions. The start-up event is characterized by an extremely rapid increase in nuclear power (to approximately 100 times rated power) followed by an equally rapid power reduction due to doppler feedback. The important kinetics parameters for this event are the doppler coefficient, the ratio of BETA/ℓ*, and the maximum reactivity addition due to the withdrawal of a control rod while subcritical. As noted in Table 7-1 only the Core 15 doppler coefficient is significantly nonconservative as compared to the values assumed in the reference analysis. The maximum reactivity addition is much less than assumed for the reference analysis, and the BETA/ℓ* ratio is nearly the same as assumed in the reference analysis. If, however, the event were reanalyzed assuming the Cycle 15 value for the doppler coefficient and assuming the same rod worths and BETA/ℓ* ratios as in the reference analysis, the consequences of this accident would still not be severe. Assuming a linear relationship between doppler coefficient and fuel effective temperature rise, the reduced Core 15 doppler coefficient would result in full effective temperature rises of 900°F and 850°F (as compared to 580°F and 590°F in the reference analysis) for the cold and hot start-up events, respectively.

Thus, assuming a hot spot peaking factor of 3.0, the peak fuel temperatures of 2800°F and 3100°F for the cold and hot start-up events, respectively, would still be significantly less than the fuel melting temperature. This is still extremely conservative since a hot spot peaking factor of 3.0 is significantly greater than will be allowed during Cycle 15 based on ECCS limitations.

7.2 <u>Accident Analysis</u>

7.2.1 Loss of Coolant Accident (LOCA)

The Big Rock Point Loss of Coolant Accident analysis for Exxon Nuclear Company (ENC) fuel was performed with ENC calculational models which are consistent with the requirements of Appendix K of 10 CFR 50. The appropriate assumptions and results of the ECCS analysis for Reload G-3 all-uranium fuel were documented in Reference 3. This report was submitted to the Director of Nuclear Reactor Regulation on February 18, 1977 in support of a proposed Technical Specifications change dated December 17, 1976 for updating MAPLHGR limits for Exxon Reload G and Reload G-1U fuel. The same report also included a reanalysis of Loss of Coolant Accident for Reload G and Reload G-1U fuel. The limiting break size for all three fuel types (Reload G-3, Reload G and Reload G-1U) was identified to be a 0.25 ft^2 small recirculation line break. MAPLHGR limits as a function of burnup were also provided in this report. Limits for all other fuel types (General Electric F and Modified F) which will be reloaded into the core for Cycle 15 will remain unchanged from values approved by the NRC for previous cycles (Reference 4), with one exception discussed in Section 8.0.

7.2.2 Rod Drop Accident

The control rod drop accident has been previously analyzed in Reference 5. The worst case (hot standby) was analyzed for both an all-uranium core and a mixed-oxide core. The important kinetics parameters for the control rod drop accident are listed below along with the values assumed in the analysis and the Cycle 15 values.

	Assumed Value		
Parameter	Uranium Core	Mixed-Oxide Core	Core 15 Value
Effective Delayed Neutron Fraction	.00591	.00529	.00588
Doppler Coefficient	$-.916 \times 10^{-5}$ Δk/k/°F	$-.96 \times 10^{-5}$ Δk/k/°F	$-.95 \times 10^{-5}$ Δk/k/°F
Maximum Worth of a Single Control Rod	[illegible].5% Δk/k	2.5% Δk/k	2.29% Δk/k

The Cycle 15 values for the important kinetics parameters are very similar to the values assumed for both cores analyzed in Reference 5. The Cycle 15 values of doppler coefficient and BETA are bounded by the values assumed in the two analyses. Based on this comparison, the reference analysis is considered conservative for the upcoming cycle.

7.2.3 Anticipated Transient Without Scram (ATWS)

The consequences of the most limiting ATWS event, the loss of load without turbine bypass, were previously evaluated in References 1 and 6. These analyses assumed a void coefficient much more negative (-.20 $\Delta k/k$/unit void) than expected at any time during Cycle 15, and a doppler coefficient much less negative (-5.42×10^{-5} $\Delta k/k/\%$) than expected during Cycle 15. Thus, the previous analyses are considered conservative for the upcoming cycle.

7.3 References

1. APED-4093, "Transient Analysis, Consumers Power Company Big Rock Point Plant," October 1962, and/or Big Rock Point Final Hazards Summary Report.
2. GEAP-4496, "Core Performance and Transient Flow Testing - Big Rock Point Boiling Water Reactor," July 1965.
3. XN-NF-76-55, Revision 1, "ECCS Analysis for Exxon Nuclear Company G-3 All Uranium No Cobalt Fuel for Big Rock Point (Including Reanalysis of Reload G and G-1U Designs)," February 1977.
4. "Big Rock Point Plant Loss-of-Coolant Accident Analysis for General Electric Fuel in Conformance With 10CFR50 Appendix K," July 11, 1975. (Submitted as Appendix A to a Technical Specifications change request from Consumers Power Company to the NRC dated July 25, 1975.)
5. Technical Specifications change request from R B Sewell (CP Co) to J F O'Leary (USAEC) dated June 20, 1974.
6. NEDE-21065, "Anticipated Transients Without Scram Study for Big Rock Point Power Plant," October 1975.
7. Proposed Technical Specifications change dated January 17, 1964.

8.0 PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The proposed Technical Specifications are contained under Section I of this submittal. In general, the changes consist of proposing specific parameters and drawings for the Reload G-3 fuel for the Big Rock Point Technical Specifications with justification presented in Sections 1 through 7. The MAPLHGR limits, as proposed, are consistent with the proposed MAPLHGR limits contained in the Technical Specifications change request dated December 17, 1976. Justification for these limits is contained in Exxon Report XN-NF-76-55, Revision 1, forwarded to the Commission on February 18, 1977. There is also a minor correction to the MAPLHGR for modified F fuel at a burnup of 25,000. By letter dated July 25, 1975, we proposed a MAPLHGR limit of 8.4. In Amendment 10, dated June 4, 1976, this value was incorrectly transposed to 8.7. We propose to correct this back to 8.4.

One proposed change to the Technical Specifications has not been addressed up to this point. That change is the deletion, from Section 5.2.1(b) and Table 8.2, of the limitation for the "Maximum MWd/T of Contained Uranium for an Individual Bundle." This limitation first appeared in Consumers Power proposed Technical Specifications for the Big Rock Point Plant dated June 1, 1962. It was contained in Section 5.2.2, "Principal Calculated Nuclear Characteristics of the Core." This section contained specific parameters relevant to Cycle 1 for the Big Rock Point reactor (eg, moderator and void coefficients, dopplers, reactivity balance, average MWd/Ton of contained uranium, etc). It is apparent that although the other design parameters associated with the original core composition were updated or deleted as necessary to account for the different reload fuels, the limitation on maximum fuel burnout was maintained intact for each subsequent reload licensing submittal.

By letter dated January 20, 1977, Consumers Power Company responded to a letter from Mr D L Ziemann concerning fission gas release from fuel pellets with high burnup. In our response, we calculated the relevant parameters for all G series reload fuels for burnups ranging from 30,620 to 38,935 MWd/MT. This is well above the design burnup of the fuels. The results of this analysis indicated that the fission gas release model burnup had very little effect on the peak clad temperature prediction (less than 1% reduction in MAPLHGR limits at end of life) and consequently insignificant effect on the Big Rock Point LOCA analysis and, therefore, was of no safety concern. We further indicated that these results would be consistent for the 9 x 9 fuels.

Attached as Appendix 1 to this report is an analysis of the fission product inventory change with increased burnup for the Big Rock Point core. This analysis was conducted assuming a core average burnup of 30,000 MWD/STU, which is also higher than the design average burnup of the Reload G fuel. The results of this analysis indicated that the radiation dose increases due to increased fission product inventory were insignificant and would remain insignificant (less than 1% of total dose) until the fuel burnups reached approximately 15×10^6 MWd/T.

Further justification for deletion of the maximum burnup limitation exists in the Standard Review Plans. Section 4.2 states in part, "The cladding design should be such as to accommodate the fission gas evolved in operation, so that the fuel can reach design burnup without exceeding the cladding structural design criteria." By virtue of the preceding discussion and analyses, Consumers Power Company concludes that it adequately meets the Standard Review Plan criteria and therefore no arbitrary burnup limitation is necessary. Also, the General Electric Standard Technical Specifications for Boiling Water Reactors makes no mention of the maximum fuel burnup allowable. Since Big Rock Point is in the process of conversion to the standard format, we feel that consistency would also dictate deleting this arbitrary limitation.

Thus, based on safety analyses performed concerning fuel burnup and on guidance developed from the Nuclear Regulatory Commission in the form of the Standard Review Plans and Standard Technical Specifications, Consumers Power Company concludes that the limitation on maximum fuel burnup is unnecessary and should be deleted from the Big Rock Point Technical Specifications.

9.0 START-UP PROGRAM

The testing and start-up program planned for the next refueling outage and subsequent start-up will include:

(1) Control rod drive testing, as required by the Technical Specifications, including scram times.

(2) Core shutdown margin verification with the most reactive rod withdrawn.

(3) Critical control rod pattern.

(4) Measurement of flux shapes during power escalation and comparison to computer predictions.

A brief explanation of these tests is contained below.

Shutdown Margin Verification:

Core shutdown margin is verified at the beginning of each cycle and during the first cold shutdown after 35,000 MWd_t generation. The Technical Specifications require subcriticality to be demonstrated with the most reactive rod withdrawn from the core as well as an immediately adjacent rod known to contribute .003 k_{eff} or more. An analytical determination of the highest worth rod is made, as well as the number of notches of an immediately adjacent rod required to contribute .6% $\Delta k/k$. .3% $\Delta k/k$ is added to account for a reactivity increase at temperatures higher than the ambient temperature at which the test is performed (normally 10¢ to 20¢). Plant procedures call for the individual withdrawal of each control rod in the core, plus at least the number of notches specified in the physics analysis on an adjacent rod to verify the analytical determination. Core monitoring is provided by gas-filled Boron 10 lined proportional counters and may be supplemented by portable fission chambers positioned above the core when available count rate is low.

Rod Drive Scram Time Testing:

Technical Specifications requirements state the maximum control rod drive scram time from the fully withdrawn position to 90% of insertion shall not exceed 2.5 seconds. This requirement is verified at the beginning of each cycle by attaching leads from a strip chart recorder traveling at a predetermined rate to the position indication for the drive to be tested and to the 26 volt d-c signal from the high reactor pressure input on one of the two safety channels. The control rod is fully withdrawn and a high reactor pressure trip is simulated by removing power from the reactor pressure input to the safety system logic circuitry. Scram time is measured on the strip chart from the time of safety channel trip to full insertion. The test is repeated for all control rods using both safety channels.

Critical Rod Pattern:

Control rod withdrawal sequences and initial critical rod patterns are analytically determined at the beginning of each cycle. On completion of core reconstitution and shutdown margin verification, an initial criticality at ambient conditions is performed. Integrated control rod worth curves and shutdown margin are adjusted based on the conditions of actual critical rod pattern. On attaining steady state equilibrium conditions at rated power, a reactivity balance is performed and the actual critical rod pattern is verified to be within 1% $\Delta k/k$ of expected.

Figure 9.1-1 shows the cold critical control rod pattern at BOC and EOC.

Power Distribution Measurement:

Flux distribution measurements are made during the escalation to rated power after the beginning of a new cycle by inserting of copper-titanium alloy wires into the core and counting the copper activation. Predictions of this flux wire activation are made with GROK, a three-dimensional one group diffusion theory code, using actual operating conditions as input. (See Section 5.2.)

Power distribution calculations are then adjusted based on the flux wire, GROK calculation comparison. In-core instrumentation is calibrated to conform with flux wire measurements. Thermal hydraulic analysis based on the flux wire corrected power distribution calculations are compared to MAPLHGR, MCHFR and heat flux limits to insure conformance with the Technical Specifications.

Figure 9.1-1

COLD CRITICAL ROD PATTERNS

BOC

	A	B	C	D	E	F
1		4	6	4	6	
2	6	0	0	0	0	4
3	4	0	0	0	0	6
4	6	0	0	0	0	4
5	4	0	0	0	0	6
6		6	4	6	4	

N

BRP CONTROL ROD POSITIONS

EOC

	A	B	C	D	E	F
1		23	9	23	9	
2	9	0	0	0	0	23
3	23	0	0	0	0	9
4	9	0	0	0	0	23
5	23	0	0	0	0	9
6		9	23	9	23	

N

BRP CONTROL ROD POSITIONS

0 is all in

23 is all out

III. CONCLUSIONS

Based on the foregoing, Big Rock Point Plant Review Committee has concluded that this change does not involve an unreviewed safety question.

CONSUMERS POWER COMPANY

By C R Bilby

C R Bilby, Vice President
Production & Transmission

Sworn and subscribed to before me this 15th day of April 1977.

Linda R Thayer

Linda R Thayer, Notary Public
Jackson County, Michigan
My commission expires July 9, 1979.

FISSION PRODUCT INVENTORY CHANGE WITH INCREASED BURNUP - BIG ROCK POINT

BACKGROUND

Radiation dose analysis for the "MCA" in the FHSR assumes 1/3 core at 5000 MWd/T, 1/3 at 10,000 MWd/T and 1/3 at 15,000 MWd/T. All nuclides which are significant in radiation dose contribution (Table A-1), with the exception of Kr-85 and I-129, have achieved steady state equilibrium at the lowest burnup utilized in the FHSR. Consequently, only Kr-85 and I-129 continue to increase as burnup increases.

CALCULATIONS

The FHSR does not list specific quantities for core inventory but activities for the 10% release case were determined for an earlier analysis (Table A-1) to be 1.41×10^{-1} Curies of I-129 (10% of core inventory) and 1.88×10^{4} Curies of Kr-85 (30% of core inventory) after two years of full power operation. Two years were chosen to provide conservative inventories of Kr-85 and I-129 equal to a full core at 15,000 MWd/T. Activities are calculated for 30,000 MWd/T by Equation I:

$$\text{Activity} = \left(\frac{8.435 \times 10^{11}\ \mu\text{Ci-Sec}}{MW_t}\right) (Y)\ (\lambda)\ (240\ MW_t)\ (GR)\ (1-e^{-\lambda T}) \quad \text{Equation I}$$

where: Y = Fission Yield

λ = Decay Constant (Sec^{-1})

GR = 0.1 for I-129, 0.3 for Kr-85

T = 4 Years Irradiation (1.26×10^{8} Sec)

Kr-85 Activity = 3.53×10^{10} µCi Released to Containment

I-129 Activity = 2.82×10^{5} µCi Released to Containment

Contribution to off-site dose from leakage of the above quantities at maximum containment leak rate (FHSR Figures 13.2 and 13.3) of 4.3×10^{-9}/sec is performed as follows:

$$\text{Kr-85:}\quad \text{Rad/h} = 0.35\ \overline{E}_{\gamma}\ (X/Q)\ (Q/\text{Sec})\ \left(\frac{3600\ \text{Sec}}{h}\right) \quad \text{Equation II}$$

where: X/Q = Diffusion Constant for Ground Level Release at 842m, From Regulatory Guide 1.3 (4.5×10^{-4} Sec/m^3)*

NOTE:

*8.6×10^{-4} from Figure 3.A divided by a wake correction factor of 1.9 from Figure 2.

Q/Sec = Release Rate = $(4.3 \times 10^{-9}/Sec)\ (1.88 \times 10^{4} Ci) = 8.1 \times 10^{-5}\ Ci/Sec$

$\overline{E}_{\gamma}$ = Average Energy per Disintegration (0.002 Mev)

From Equation II, Kr-85 dose rate equals 6.6×10^{-8} rad/h to the total body. In comparison, FHSR Section 13.11.4 indicates the maximum dose rate from the plume (all components) is 5×10^{-3} rad/h. Thus, the percentage due to Kr-85 is insignificant at $(6.6 \times 10^{-8}/5 \times 10^{-3})\ (100\%) = 0.0013\%$.

I-129: Rad/h = (K) (X/Q) (Q/Sec) (B) Equation III

where: K = Dose Conversion Factor per Regulatory Guide 1.109 (5.55 Rem/μCi Inhaled)

X/Q = Diffusion Constant $(4.5 \times 10^{-4}\ Sec/m^3)$

Q/Sec = Release Rate = $(4.3 \times 10^{-9}/Sec)\ (1.41 \times 10^{5}\ \mu Ci) = 6.1 \times 10^{-4}\ \mu Ci/Sec$

B = Breathing Rate per Regulatory Guide 1.3 $(1.25\ m^3/h)$

From Equation III, I-129 thyroid dose commitment from one hour of inhalation equals 1.9×10^{-6} rem, or 3.8×10^{-6} rem from two hours of exposure. In comparison, FHSR Section 13.14.3 indicates a 2-hour thyroid dose of 2 rem is expected from the total halogen mixture. Thus, I-129 is insignificant at $(3.8 \times 10^{-6}/2)\ (100\%) = 0.00019\%$.

CONCLUSION

Dose contributions from nuclides affected by increased fuel burnup are negligible relative to total "MCA" doses. Radiation dose increases due to increased production of I-129 and Kr-85 will remain insignificant (less than 1% of total dose) up to core burnups of approximately 15×10^{6} MWd/T.

TABLE A-1
Full Core Fuel Pin Gap
Radioactivity Released

Isotope	$\lambda(Min^{-1})$	Gap Activity (μCi): Big Rock Point Plant Initial Design Calculations From GE - Modified for 80/20 U-235 Pu-238 Fission Mixture	Gap Activity (μCi): Quanicassee Design Data Ratioed to Big Rock Point Power Levels
I-129	7.76E-14	1.41E+05	-
I-131	5.98E-05	5.62E+12	5.92E+12
I-132	5.11E-03	8.35E+12	9.02E+12
I-133	5.69E-04	1.37E+13	1.33E+13
I-134	1.33E-02	1.45E+13	1.56E+13
I-135	1.73E-03	1.29E+13	1.21E+13
Xe-138	4.88E-02	1.21E+12	1.21E+12
Kr-87	9.12E-03	4.2E+11	5.11E+11
Kr-88	4.14E-03	6.43E+11	7.27E+11
Kr-85m	2.62E-02	2.39E+11	2.66E+11
Xe-135	1.26E-03	1.37E+12	3.74E+11
Xe-133	9.12E-05	1.36E+12	1.37E+12
Xe-143	4.33E+01	8.11E+09	
Kr-94	4.16E+01	1.80E+10	
Kr-93	3.22E+01	9.34E+10	
Xe-141	2.42E+01	2.23E+11	
Kr-92	2.26E+01	3.24E+11	
Kr-91	4.84E+00	5.81E+11	
Xe-140	3.06E+00	6.89E+11	
Kr-90	1.29E+00	8.52E+11	
Xe-139	1.04E+00	9.74E+11	
Kr-89	2.18E-01	8.09E+11	
Xe-137	1.81E-01	1.20E+12	
Xe-135m	4.42E-02	2.12E+11	3.68E+11
Kr-83m	6.18E-03	9.82E+10	
Xe-133m	2.13E-04	3.82E+10	3.47E+10
Xe-131m	4.03E-05	3.62E+09	4.51E+09
Kr-85	1.22E-07	1.88E+10	2.02E+10

Note: For I-129 and Kr-85 equilibrium is not obtained. Hence, the activity available is the equilibrium value at full power operation for two years times $(1 - e^{-\lambda t})$ where t is two years.

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PROJECT MANAGEMENT	REACTOR SAFETY	OPERATING TECH.	GAMMILL
BOYD	ROSS	X EISENHUT	STEPP
P. COLLINS	NOVAK	X SHAO	HULMAN
HOUSTON	ROSZTOCZY	X BAER	
PETERSON	CHECK	X BUTLER	SITE ANALYSIS
MELTZ		X GRIMES	VOLLMER
HELTEMES	AT & I		BUNCH
SKOVHOLT	SALTZMAN		X J. COLLINS
	RUTBERG		KREGER

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