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DESCRIPTION:
Ltr notarized 9-14-73, furnishing info re scheduled refueling outage at the Nine Mile Point Unit # 1......& advising that the planned replacement of up to 120 of the 532 bundles, may reflect a generic design change & may require a change to the Tech Specs............

PLANT NAME: Nine Mile Point Unit # 1

ENCLOSURES:
ATTACHMENTS - Figs 1, 2 & 3 & Tables 1 thru 5

ACKNOWLEDGED
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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD WEST
SYRACUSE, N. Y. 13202

September 14, 1973

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Re: Nine Mile Point Unit 1
Docket No. 50-220
License DPR-17

The second major refueling outage at Nine Mile Point Unit 1 is tentatively scheduled for March 1974. At this outage it is planned to replace up to 120 of the 532 bundles which comprise the core with new Type 5 and Type 6 fuel bundles. These reflect a generic design change for reload fuel on the part of our fuel supplier, and are expected to enhance overall fuel reliability. This refueling will provide reactivity augmentation for continued high load factor operation through to the refueling planned for Spring 1975.

Pursuant to Paragraph 50.59 of 10 CFR 50, Niagara Mohawk requests authorization to use these Type 5 and Type 6 fuel bundles and to operate the reactor with the core design characteristics described herein. Additional information on the mechanical, thermal-hydraulic and nuclear characteristics of these fuel bundles has been submitted to the Commission by General Electric in the report, "Generic Design Information for General Electric Reload Fuel," NEDO-20103, transmitted by General Electric's letter from Mr. J. L. Benson to Mr. V. Stello, dated September 3, 1973. In this report, the Type 5 and Type 6 fuel bundles are identified as 8D250 and 8D262. Results of analyses on the response of the refueled core to anticipated transients and postulated accidents will be reported to the Commission by October 15, 1973, as indicated in our June 6, 1973, letter.

7001

Mechanical Design

As described in the General Electric report NEDO-20103, the Type 5 and Type 6 fuel bundles both contain 63 fuel rods and one spacer capture water rod which are spaced and supported in a square (8 x 8) array. Figure 1 is a schematic diagram of Type 5 and Type 6 fuel bundles. Figures 2 and 3 show the location of four enrichments within the Type 5 and Type 6 fuel bundles respectively, as well as the location of gadolinia-urania rods, tie rods and the spacer capture water rods. Table 1 summarizes the significant characteristics of all fuel types available for use in Unit 1.

Most aspects of the 8 x 8 bundle design are similar to the current 7 x 7 design. Specifically, the upper and lower tie plate, the fuel rod spacers, the upperaand lower end plugs, and other associated bundle hardware are the same as the 7 x 7 except for modeling down in size to be compatible with the increased number of rods per bundle and the reduced diametrical wall thickness.

Type 5 and Type 6 bundles have been designed with the same envelope dimensions as the initial core and previous reload fuel. Accordingly, this 8 x 8 fuel is physically compatible with all 7 x 7 fuel and all internal core structures.

The primary result of the different mechanical design is the significant increase in heat transfer area of 8 x 8 bundles relative to the 7 x 7 bundles. The increase in heat transfer area leads to substantially lower lineal heat generation rates (LHGR) and, therefore, lower fuel temperatures during normal operation and anticipated transients.

The mechanical design bases for Type 5 and Type 6 bundles are described in detail in NEDO-20103. It should be noted that these generic analyses have shown that the design stress intensity limits are not exceeded during steady state operation with LHGR's up to 13.4 kw/ft nor for short term transients up to 15.6 kw/ft.

The design LHGR is 13.4 kw/ft for the Type 5 and Type 6 reload fuel as compared with 17.5 kw/ft LHGR for the 7 x 7 fuel. The reduction in operating lineal heat generation rate for the 8 x 8 fuel results in a substantial margin between the maximum steady state LHGR and the LHGR corresponding to incipient center melting or one percent plastic strain.

Thermal-Hydraulic Characteristics

The steady state thermal-hydraulic design bases of the Type 5 and Type 6 fuel are a minimum critical heat flux ratio (MCHFR) of 1.9 using the Hench-Levy correlation and a maximum lineal heat generation rate of 13.4 kw/ft.

Analyses were performed for a variety of core loadings to fully assess the effect of 8 x 8 reload fuel on the core thermal-hydraulic characteristics. Table 2 presents a tabulation of significant thermal-hydraulic characteristics calculated for a spectrum of core loadings ranging from all 7 x 7 fuel to all 8 x 8 fuel. The results show that, irrespective of the number of 8 x 8 fuel assemblies loaded in the core, both the 7 x 7 and 8 x 8 fuel assemblies receive adequate coolant flow. The margin-to-CHF for the limiting assembly in an 8 x 8 core, or in a mixed 7 x 7 - 8 x 8 core, is always equal to or greater than the margin-to-CHF for the limiting assembly in an all 7 x 7 core. Furthermore, due to the increased heat transfer area and correspondingly lower operating heat flux of the 8 x 8 assembly relative to the 7 x 7 assembly, the 8 x 8 fuel has greater margin-to-CHF than does the 7 x 7 fuel.

The thermal-hydraulic analyses were performed for the following reactor conditions:

Reactor power..............1850 MWt
Reactor pressure...........1030 psig (steam dome)
Recirculation flow rate....67.5 x 10^6 lb/hr
Inlet enthalpy.............526 Btu/lb
Bypass flow................10% of total core flow

The power peaking factors used in these analyses are as follows:

Power Peaking Factor	7 x 7	8 x 8
Radial	1.50	1.50
Axial	1.57	1.57
Local	1.30	1.22

Nuclear Characteristics

The nuclear characteristics of the Type 5 and Type 6 reload fuel bundles are not substantially different than for initial core fuel and previous reload fuel used at Nine Mile Point Unit 1. A summary of the bundle nuclear characteristics are provided in the General Electric report NEDO-20103.

Reported herein are the results of analyses of the nuclear characteristics of the refueled core with up to 120 8 x 8 reload fuel bundles. The analyses assume a scatter (as opposed to zoned) loading of reload fuel as shown in Figure 4.

A. Core Effective Multiplication, Control System Worth and Reactivity Coefficients

A tabulation of the typical nuclear characteristics of the pre and post outage cores is provided in Table 3. Since the nuclear characteristics of the new 8 x 8 reload bundles are similar to the nuclear characteristics of 7 x 7 bundles, the reload core does not differ

significantly in temperature and void dependent characteristics from the values previously reported.

B. Reactor Shutdown Margin

The refueled core fully meets established criteria of the Technical Specifications in that it may be maintained subcritical by at least one-quarter of one percent with the most reactive control rod in its full-out position and all other rods fully inserted.

An analysis assuming use of as many as 120 new 8 x 8 fuel bundles at an incremental exposure of 4000 MWD/t from the Spring 1973 refueling outage shows that minimum shutdown margin of 0.012 ΔKeff would result for the typical refueling pattern given in Figure 4. This minimum shutdown margin occurs at the beginning of cycle, and the shutdown margin increases throughout the cycle as shown in Figure 5. Thus, the "R" factor referred to in the Technical Specifications and Bases will be zero for this cycle.

For an incremental exposure (from the Spring 1973 refueling outage) as low as 3800 MWD/t, this same loading pattern will also satisfy the design criterion of 0.01 ΔKeff on shutdown margin. This latter calculation assumes the discharge of the highest exposure, least reactive bundles in the core. If the outage were to occur several hundred MWD/t less than the 3800 MWD/t value, shutdown margin could readily be maintained within the 0.01 ΔKeff design value by discharge of some limited quantity of lower exposure, more highly reactive fuel bundles. In such a case, sufficient analyses would be done to ensure compliance with shutdown margin requirements.

For any higher value of incremental core exposure above 3800 MWD/t this same core loading plan may be used without modification.

C. Liquid Poison System

The liquid poison system requirements have been examined, and the present Technical Specifications are deemed adequate.

D. Reactivity of Fuel in Storage

Reactivity conditions in the spent fuel storage pool and the new fuel storage rack configuration have not been re-analyzed since the k_∞ of the replacement fuel in its most reactive condition is less than k_∞ of the initial fuel assemblies without curtains.

Conclusions

Type 5 and Type 6 8 x 8 reload fuel have been designed to be compatible with initial core and previous reload fuel used at Nine Mile Point Unit 1. Based on the results of analyses described in General

Electric's report NEDO-20103 and in this letter it is tentatively concluded that (1) this reloading does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazards considerations not described or implicit in the Final Safety Analysis Report, the Petition to Increase Power Level and Supplements and Addendums thereto; and (2) there is reasonable assurance that the health and safety of the public will not be endangered. This proposed design change may require a change to the Technical Specifications. If required, this change will be proposed with the additional analyses to be submitted by October 15, 1973. These tentative conclusions will be confirmed when the analyses on anticipated transients and postulated accidents are submitted.

This matter has been reviewed by the Site Operations Review Committee and the Safety Review and Audit Board which concur with these tentative findings.

Very truly yours,

Philip D Raymond

Philip D. Raymond
Vice President-Engineering

KDV/vk
Attachments

Subscribed and sworn to before me this 14th day of September 1973.

Valerie N. Kelly
Notary Public

VALERIE N. KELLY
Notary Public in the State of New York
Qualified in Onon. Co. No. 34-4504729
My Commission Expires March 30, 1975

TABLE 1

INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS

	Initial Core Fuel	Type 2 Fuel	Type 3 Fuel	Type 4 Fuel	Type 5 Fuel	Type 6 Fuel
FUEL ASSEMBLY						
Geometry	7 x 7	7 x 7	7 x 7	7 x 7	8 x 8	8 x 8
High Enrichment Rods	30	32	32	32	44	44
Medium High Enrichment Rods	16	10	10	10	14	14
Medium Low Enrichment Rods	3	6	6	6	4	4
Low Enrichment Rods	0	1	1	1	1	1
Poison Rods	0	4	3	4	4	4
Water-Spacer Capture Rods	0	0	0	0	1	1
Rod Pitch (in.)	0.738	0.738	0.738	0.738	0.640	0.640
Water to Fuel Volume Ratio	2.38	2.43	2.43	2.53	2.60	2.60
Heat Transfer Area (ft^2)	87.6	86.5	86.5	86.5	97.6	97.6
FUEL RODS						
Active Fuel Length (in.)	144.0	144.0	144.0	144.0	144.0	144.0
Gas Plenum Length (in.)	11.25	11.25	11.25	11.0	11.24	11.24
Fill Gas	helium	helium	helium	helium	helium	helium
Getter	no	no	no	yes	yes	yes
FUEL						
Material	sintered UO_2	sintered UO_2	sintered UO_2	sintered UO_2	sintered UO_2	sintered UO_2
INITIAL ENRICHMENT, WT/% U-235						
Average for Bundle	2.11	2.50	2.30	2.50	2.62	2.50
High	2.42	2.79	2.56	2.79	2.87	2.73
Medium High	1.67	2.10	1.94	2.09	2.14	2.06
Medium Low	1.19	1.80	1.69	1.80	1.87	1.80
Low	-	1.40	1.33	1.40	1.45	1.40
Pellet Diameter (in.)	0.488	0.487	0.487	0.477	0.416	0.416
Pellet Immersion Density (% TD)	95.0	95.0	95.0	95.0	95.0	95.0
CLADDING						
Material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Thickness	0.0355	0.032	0.032	0.037	0.034	0.034
Outside Diameter (in.)	0.570	0.563	0.563	0.563	0.493	0.493
FUEL CHANNEL						
Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Outside Dimension (in.)	5.438	5.438	5.438	5.438	5.438	5.438
Wall Thickness (in.)	0.080	0.080	0.080	0.080	0.080	0.080
Channel Length (in.)	162 1/8	162 1/8	162 1/8	162 1/8	162 1/8	162 1/8
SPACERS						
Material	stainless steel with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs
Number per Bundle	7	7	7	7	7	7

Table 2

RESULTS OF THERMAL-HYDRAULIC ANALYSES

Case Number	1	2	3		4		5		6
Core Average Void Fraction, %	27.3	27.2	27.3		27.3		27.3		27.3
Core Pressure Drop, psi	17.1	16.9	16.9		17.0		17.2		17.6
Water Rod Flow, % of Total Core Flow	NA	NA	0.0007		0.08		0.16		0.3
Assembly Type	7 x 7	7 x 7	7 x 7	8 x 8	7 x 7	8 x 8	7 x 7	8 x 8	8 x 8
Number	532	532	531	1	399	133	266	266	532
Hot Channel Coolant Flow, 10^3 lb/hr	110	113	113	105	114	106	115	107	109
Hot Channel MCHFR	1.90	1.92	1.92	2.17	1.93	2.18	1.95	2.20	2.23

Case Description:

Case 1: Full core loading (532 assemblies) of initial core fuel-0.570-inch diameter fuel rods, 7 x 7 array.
Case 2: Mixture of 7 x 7 initial core (328 assemblies-0.570-inch rod diameter) and 7 x 7 reload fuel (204 assemblies-0.563-inch rod diameter).
Case 3: Same as Case 2 with one 7 x 7 initial core assembly replaced with an 8 x 8 assembly.
Case 4: One-quarter core load of 8 x 8 reload assemblies with a mixture of 7 x 7 initial core and reload assemblies.
Case 5: One-half core load of 8 x 8 reload assemblies with a mixture of 7 x 7 initial core and reload assemblies.
Case 6: Full core loading of 8 x 8 reload fuel.

Table 3

NINE MILE POINT UNIT I NUCLEAR CHARACTERISTICS OF CORE

Core Effective Multiplication and Control System Worth (0% Voids, 68°F)	Pre-Outage Core*	Reloaded Core
Keff Uncontrolled	1.11	1.13
Δk Control Rods	-0.16	-0.17
Keff Fully Controlled	0.95	0.96
Keff Strongest Single Rod Withdrawn	0.98	0.99
Reactivity Coefficients		
Steam Void Coefficient at 1850 MWt, 526 BTU/lb Inlet Enthalpy, 30.5% Voids $(1/k)(\Delta k/\Delta V)$, 1/% Void, Range of Values	-9.3×10^{-4}	-9.7×10^{-4} to -8.5×10^{-4}
Power Coefficient at 1850 MWt, 526 BTU/lb Inlet Enthalpy, $(\Delta k/k)/(\Delta P/P)$, Range of Values	-0.049	-0.057 to -0.042
Fuel Temperature Coefficient, $(1/k)(\Delta k/\Delta T)$, 1/°F Fuel, Range of Values	-1.13×10^{-5}	-1.04×10^{-5} to -1.14×10^{-5}

* Analysis for refueling at a core exposure 10700 MWD/t, 4000 MWD/t from spring 1973 refueling outage.

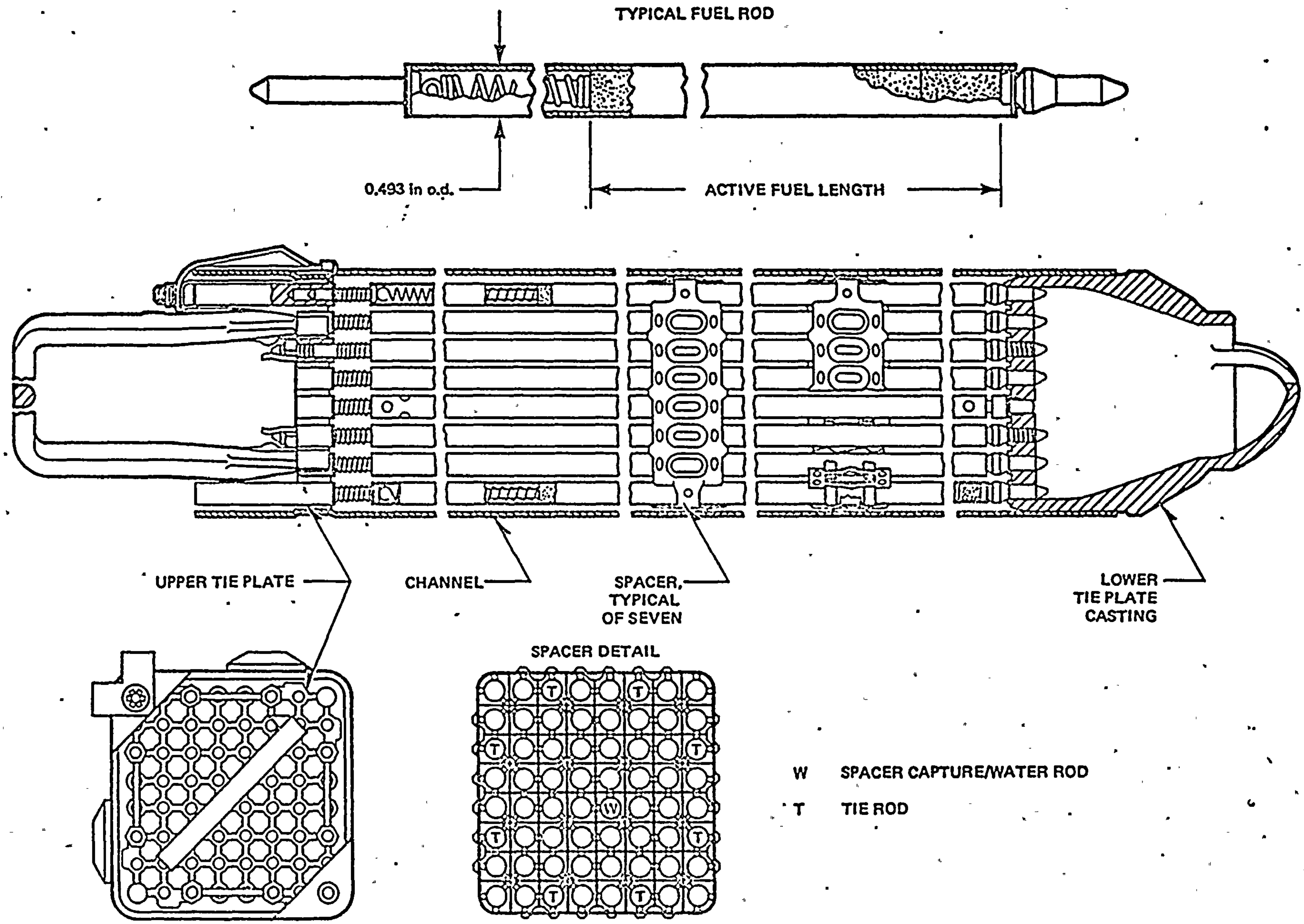

Figure 1. 8 x 8 Type 5 & Type 6 Reload Fuel Assembly

WIDE-WIDE CORNER

4	3	2 T	2	2	2 T	2	3
3	2	1	1	1	1	1	2
2 T	1	5 G	1	1	1	5 G	1 T
2	1	1	1	1	1	1	1
2	1	1	1	WS	1	1	1
2 T	1	1	1	1	1	1	1 T
2	1	5 G	1	1	1	5 G	1
3	2	1 T	1	1	1 T	1	2

ROD TYPE	ENRICHMENT wt % U-235	NUMBER OF RODS
1	2.87	40
2	2.14	14
3	1.87	4
4	1.45	1
5	2.87	4
WS	–	1

WS – SPACER CAPTURE WATER ROD
T – TIE RODS
G – GADOLINIUM RODS

Figure 2. *Nine Mile Point* Type 5 *Reload Fuel Lattice*

WIDE-WIDE CORNER

4	3	2 T	2	2	2 T	2	3
3	2	1	1	1	1	1	2
2 T	1	5 G	1	1	1	5 G	1 T
2	1	1	1	1	1	1	1
2	1	1	1	WS	1	1	1
2 T	1	1	1	1	1	1	1 T
2	1	5 G	1	1	1	5 G	1
3	2	1 T	1	1	1 T	1	2

ROD TYPE	ENRICHMENT wt % U-235	NUMBER OF RODS
1	2.73	40
2	2.06	14
3	1.80	4
4	1.40	1
5	2.73	4
WS	–	1

WS – SPACER CAPTURE WATER ROD
T – TIE RODS
G – GADOLINIUM RODS

Figure 3. Nine Mile Point Type 6 Reload Fuel Lattice

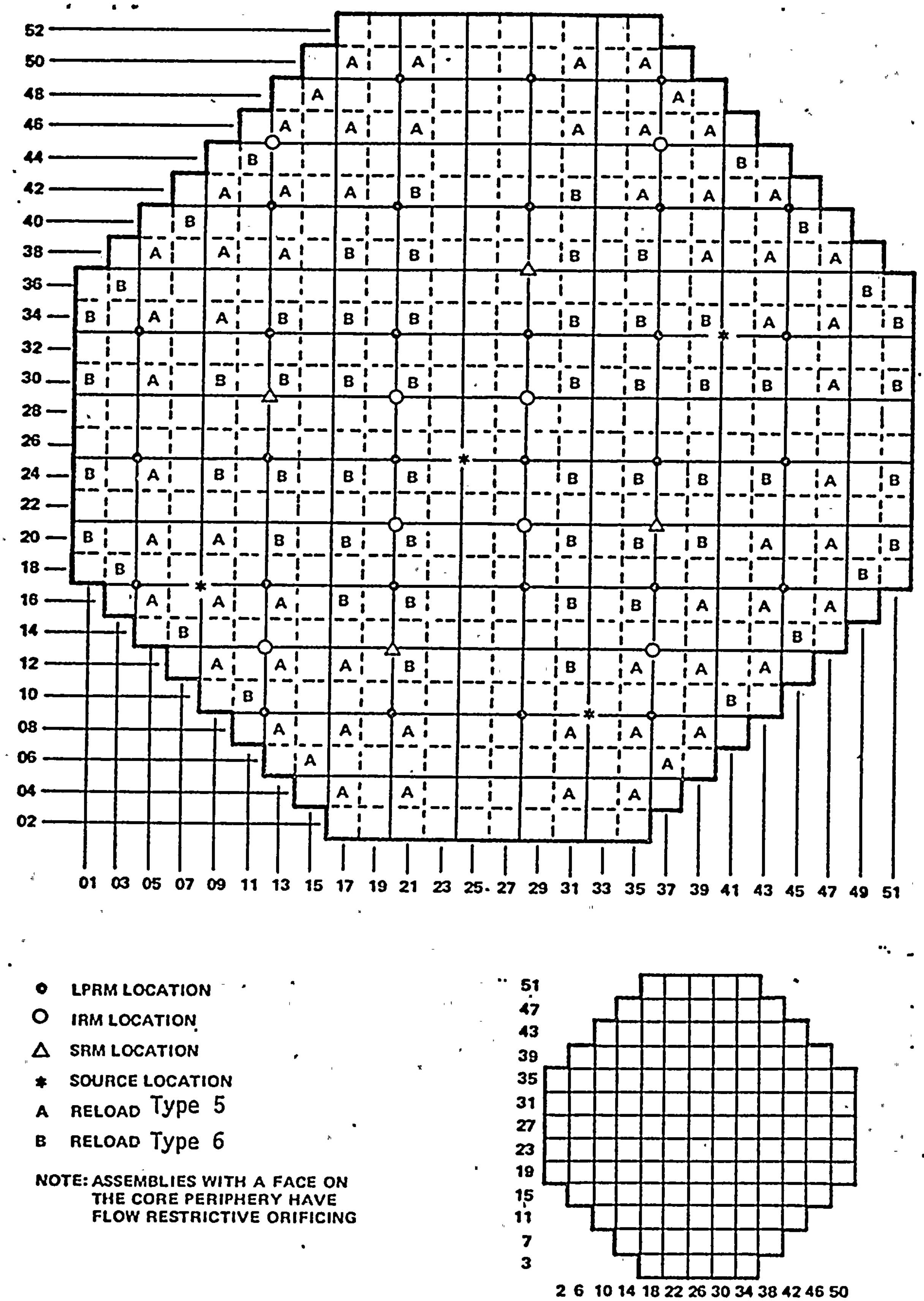

Figure 4.: Design Reference Core Loading

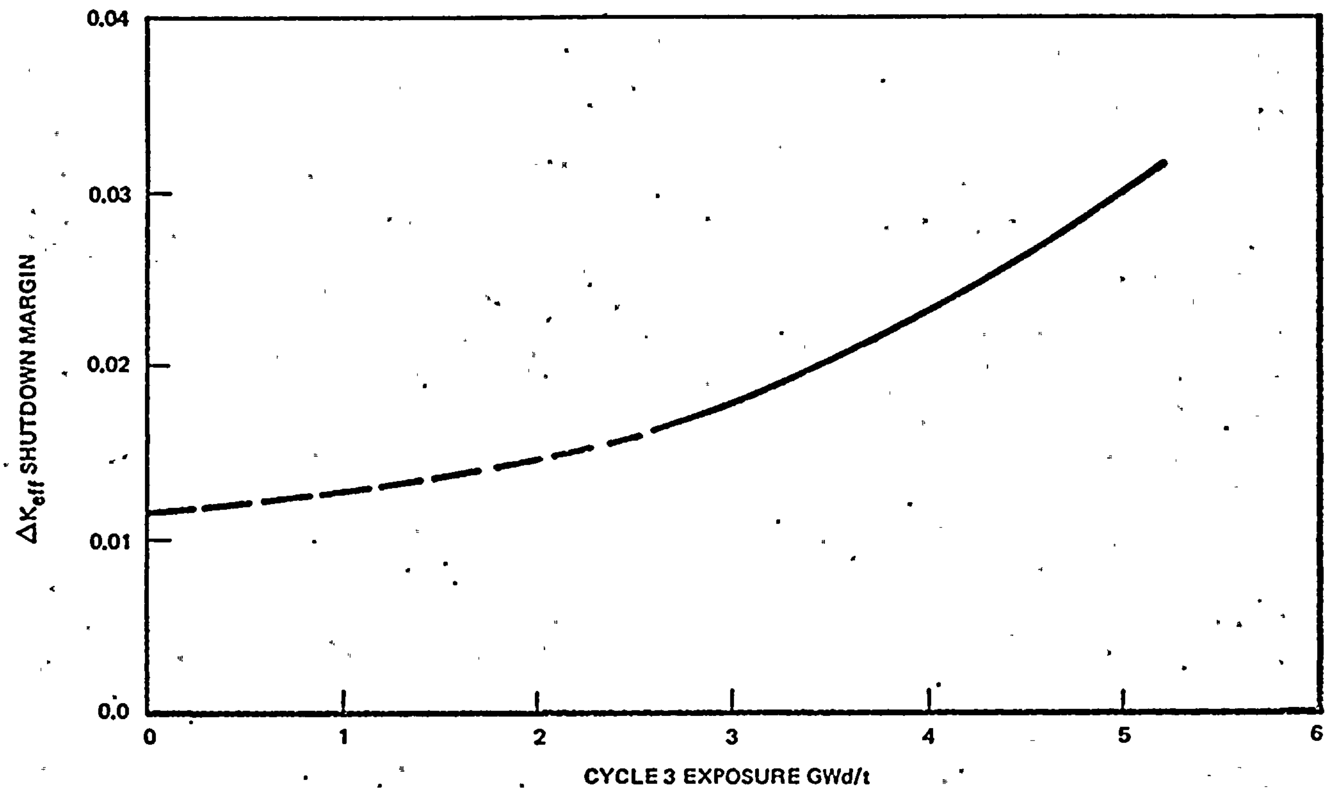

Figure 5. *Nine Mile Point Cycle 3 ΔK_{eff} Shutdown Margin versus Exposure Strong Rod Out, 20°C*