FROM:

Niagra Mohawk Power Corporation
Syracuse, New York 13202
T. J. Brosnan

TO:

Dr. Peter A. Morris

DATE OF DOCUMENT:	DATE RECEIVED	NO.:
2-24-72	2-28-72	

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Ltr notarized 2-24-72 pursuant to Para 50.59 request authorization to make changes to Unit #1 of Nine Mile Point Nuclear Station & trans:

ENCLOSURES:

TABLES 1, 2 & 3
FIGURES 1, 2, 3, 4 & 5

(71 cys ea encl rec'd)

REMARKS:

(Holding 16 cys for ACRS)

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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

SYRACUSE, NEW YORK 13202

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February 24, 1972

RECEIVED FEB 28 1972 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Dr. Peter A. Morris
Division of Reactor Licensing
United States Atomic Energy
Commission
Washington, D. C. 20545

RE: Docket No. 50-220; License DPR-17

Dear Dr. Morris:

Pursuant to Paragraph 50.59 of 10 CFR Part 50, Niagara Mohawk Power Corporation hereby requests authorization to make changes to Unit No. 1 of the Nine Mile Point Nuclear Station.

Introduction

The outage previously described[(1)] for removal of temporary control curtains is scheduled to begin April 2, 1972. At this same outage, it is planned to replace a number of initial core fuel bundles. These replacements will be for the purpose of maintaining activity releases as low as practicable and to provide some reactivity augmentation for continued high load factor operation through to the first major refueling now anticipated for the Spring, 1973.

This letter describes the insertion of up to 32 Type 2 fuel bundles previously reviewed[(2)] and 40 new Type 3 fuel bundles not previously reported to the Commission for the Nine Mile Point Nuclear Station. The 32 Type 2 bundles are the balance of the 56 bundles on hand for the September, 1971, outage but which were not used at that time. In addition, several reconstituted bundles may also be reinserted as described herein.

[(1)] Docket No. 50-220; Final Safety Analysis Report, Volume I, Section IV, p. IV-72

[(2)] Docket No. 50-220; Letter: Niagara Mohawk Power Corporation (T. J. Brosnan) to United States Atomic Energy Commission (Peter A. Morris), dated July 27, 1971

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Mechanical Design

The mechanical design has been previously described for the initial core[3] and Type 2[4] fuel. Type 3 fuel bundles are dimensionally the same as Type 2 fuel and have the same envelope dimensions as the initial core fuel. They can be inserted in all locations within the reactor core. Figure 1 is a schematic diagram of the Type 3 fuel assembly; Table 1 summarizes the significant characteristics of all three fuel types.

Basic lattice arrangement of Type 3 fuel is 49 rods in a 7 x 7 array. It has a centrally located spacer capture rod, and eight tie rods located on the periphery. Figure 2 shows the location of four enrichments within the Type 3 fuel bundle as well as the location of dished rods, gadolinia rods, tie rods and the spacer capture rod.

The principal changes in the Type 3 reload fuel as compared with the Type 2 fuel are the use of three instead of four gadolinia bearing rods and a lower average enrichment. The average enrichment of the Type 3 reload fuel is 2.3 w/o as compared with 2.5 w/o for the Type 2 reload fuel.

The peak power gadolinia rod in Type 3 fuel will operate at a power no greater than 0.86 of the peak UO_2 rod. This, then, limits the gadolinia rods to a lineal heat generation rate (LHGR) of 15.1 Kw/ft which compares with a LHGR of 17.5 Kw/ft for the limiting UO_2 rods. The LHGR's corresponding to normal operation, onset of fuel melting and clad damage (one percent plastic strain) are shown below for the UO_2 and gadolinia bearing rods for Type 3 fuel.

	UO_2 Rods	UO_2/Gd_2O_3 Rods
Maximum Operating LHGR (Kw/ft)	17.5	15.1
Onset of Fuel Melting (Kw/ft)	21.5	19.2
Clad Damage (Kw/ft)	28.0	27.2

From these data it can be seen that more margin exists between maximum operating LHGR level and either the onset of fuel melting or clad damage limit for the gadolinia rods than for the UO_2 rods.

(3) Docket No. 50-220; Final Safety Analysis Report, Volume I, Section IV

(4) Docket No. 50-220; Letter: Niagara Mohawk Power Corporation (T. J. Brosnan) to United States Atomic Energy Commission (Peter A. Morris), dated July 27, 1971

Thermal-Hydraulic Characteristics

The refueled core is designed to operate with a maximum lineal heat generation rate $\leq$ 17.5 Kw/ft and a minimum critical flux ratio $\geq$ 1.90. The thermal-hydraulic characteristics of Type 3 fuel are essentially the same as for the initial core fuel and identical to the Type 2 fuel.

As shown in Figure 3, the reloaded core will have up to 40 Type 3 and an additional 32 Type 2 fuel bundles. A total of 24 Type 2 fuel bundles had been previously inserted into the periphery of the core. The operating parameters, which are monitored periodically during power operation, are shown in Table 2 for the design power distribution for the reloaded core at full power and full recirculation flow.

Nuclear Characteristics

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

The nuclear characteristics of the reloaded core are compared in Table 3 with the core characteristics prior to the outage. Since the geometry and materials used in construction of the replacement fuel are nearly identical to those of the initial fuel, the nuclear characteristics of the reloaded core will not differ significantly from the previous loading condition, except as noted below.

B. Reactor Shutdown Margin

The reloaded core may be maintained subcritical in the most reactive condition throughout the subsequent operating cycle with the most reactive control rod in its full-out position and all other rods fully inserted. A minimum shutdown margin of 0.010 Δk was calculated for the reloaded core as shown in Figure 3.

This core loading assumes removal of all of the remaining temporary control curtains, insertion of an additional 32 Type 2 reload bundles, and insertion of the 40 Type 3 reload bundles. The shutdown margin of 0.010 Δk was calculated for the reloaded core at an average exposure of 6500 MWD/T. For a reloading at exposures greater than 6500 MWD/T, the shutdown margin will increase. A shutdown margin of 0.013 Δk has been calculated for a reloading at 7000 MWD/T.

The shutdown margin will increase throughout the balance of this cycle as shown in Figure 4. Thus, the value of R referred to in the Technical Specifications remains at zero for this cycle because the shutdown margin does not decrease below the initial value.

[illegible]

[illegible]

[illegible]

[illegible] [illegible]

C. Liquid Poison System

The liquid poison system requirements have been examined and are found to be adequate as licensed power level of 1850 MWt has not changed and the core reactivity effects have not been significantly altered by the introduction of reload fuel.

D. Reactivity of Fuel in Storage

The spent fuel storage pool and new fuel storage rack configuration can accommodate the Type 3 fuel since the k_∞ of the replacement fuel in its most reactive condition is less than k_∞ of the initial fuel assemblies without curtains.

Reconstitution of Fuel Bundles

Our letters of July 27, 1971, and August 20, 1971, indicated that reconstitution of fuel bundles removed from the core was planned. Seven of these reconstituted bundles are presently available and reconstitution of additional fuel removed from the core during the forthcoming outage may be conducted. The reconstituted bundles will not differ significantly in local power distribution, reactivity or exposure capability from the original fuel bundle.

Records of all fuel rod movements will be kept. The mechanical design of the fuel bundle precludes the possibility of a higher enrichment rod being inserted into a low enrichment rod location, because the end plug shank diameter is larger for higher enrichment rods (highest enrichment has largest diameter; lowest enrichment has smallest diameter). Once the bundle is reconstituted, the same procedures and safeguards for proper loading and positioning as described in our letter of August 30, 1971, are also applicable for loading this fuel.

Specific criteria used in development of bundle reconstitution and location rules are given below:

A. The maximum local power peaking will be less than the design value of 1.30.

B. The maximum reactivity change of any reconstituted bundle will be within $\pm$ 0.003 Δk of that existing for the original bundle.

C. Movements of rods will be restricted with regards to enrichment. Replacement rods must be of the same initial enrichment level as the replaced rod.

D. Replacement of failed rods and, where required, bundle location in the core, will be done in such a manner which would result in LOCA-peak clad temperatures being less than or equal to that calculated for the bundle prior to reconstitution. Replacement rods having an exposure equal to or greater than the replaced rod will result in a reconstituted bundle of equal or lower reactivity (compared to the original bundle). Reconstituted bundles having replacement rods of a lower exposure than the original will be placed in a lower power region of the core near the periphery to compensate for any increase in local power peaking and bundle reactivity.

Safety Analysis

The safety analyses described in the Petition to Increase Power Level[5] and its Addendums have been reviewed to see which of the transients and/or accidents, if any, may be affected by the introduction of Type 3 reload fuel.

The operating and damage limits applicable to the refueled core are the same as those assumed in the analyses reported in the Petition to Increase Power Level.

There are no significant changes to core transients which are occasioned by the insertion of the reload fuel.

Postulated accidents which might result in the release of fission products as analyzed in the Petition to Increase Power Level, have been reviewed to determine if the proposed refueling would affect the results. The control rod worths, thermal-hydraulic parameters, and the fission products inventories used in previous safety analyses are not significantly affected by the use of Type 3 and additional Type 2 fuel. Therefore, the analyses for those accidents dependent on these parameters (Control Rod Drop), Refueling, Main Steam Line Break Accidents) remain unchanged.

The presence of gadolinia in Type 3 fuel rods lower the threshold for failure. However, as described in the previous section on Mechanical Design, the lower operating power of these rods assures margins to failure equal to or greater than that for the limiting UO_2 rods. Below, the peak enthalpies in the limiting UO_2 rod and maximum powered gadolinia bearing rod are compared for the Control Rod Drop accident resulting in the highest fuel enthalpy.

	Enthalpy (cal/gm)	
	UO_2 Rod	GD_2O_3/UO_2Rod
Design Limit	280	271
Calculated Peak	250	215
Margin to Limit	30	56

[5] Docket No. 50-220; Technical Supplement to Petition to Increase Power Level, Section II-xv

Previously reported[6] calculations have demonstrated that the initial core and Type 2 fuel comply with the AEC Interim Acceptance Criteria for the performance of Emergency Core Cooling Systems.

The enrichment levels of Type 3 fuel are slightly different than those of initial core and Type 2 reload fuel. The distribution of local peaking factors and changes in peaking factors with exposure are, therefore, different for Type 3 fuel. However, the maximum design local peaking factor is not increased. As these differences directly affect the results of the loss-of-coolant accident analyses, the analyses have been re-done for the use of Type 3 fuel at any position in the core, including the most reactive position.

The same single failure events described in our letter of August 20, 1971, to the Commission for initial core fuel were used in the analysis of Type 3 fuel. The peak clad temperatures calculated are shown in Figure 5 for the design basis accident. The calculations were not repeated for the smaller break cases, because these cases would result in lower peak clad temperatures as has been demonstrated for initial core and Type 2 fuel.

The results of these analyses are summarized below and show that the refueled core will continue to meet the AEC Interim Acceptance Criteria.

	Peak Clad Temperature (F)	Total Metal Water Reactor (%)
AEC Criteria	<2300	< 1
Initial Core Fuel	2237	< 0.25
Type 2 Reload Fuel	2265	< 0.25
Type 3 Reload Fuel	2280	< 0.25

Loading Errors

The possibility of loading errors consisting of misplaced pellets in a fuel rod, misplaced rods in a fuel bundle, and misplaced bundles in the core have been considered. The worse case loading error for the reference core configuration occurs when a Type 2 (2.5 wt %) bundle is rotated 180 degrees in location in the center of the core. Analysis of this loading error resulted in MLHGR < 20.3 and MCHFR > 1.55. These are less than the damage limits previously established for this fuel.

[6] Docket No. 50-220; Letters (2): Niagara Mohawk Power Corporation (T. J. Brosnan) to United States Atomic Energy Commission (P. A. Morris), dated August 20, 1972

Conclusions

Type 2 and Type 3 reload fuel has been designed to be compatible with and closely match the mechanical, nuclear and thermal-hydraulic characteristics of the initial core with curtain removal, and no changes to previous safety analyses are occasioned by use of this fuel. Consequently, it is concluded that (1) no changes are required to the Station's Technical Specifications to accommodate this fuel; (2) 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 Analyses Report, the Petition to Increase Power Level and Supplements and Addendums thereto; and (3) there is reasonable assurance that the health and safety of the public will not be endangered. This matter has been reviewed by the Safety Review and Audit Board which concurs with these findings.

Very truly yours,

T J Brosnan

T. J. Brosnan
Vice President and Chief Engineer

Attachments

State of New York, County
of Onondaga, Sworn to before
me this 24th day of February, 1972

Carolyn F. Robertson

Notary Public

CAROLYN F. ROBERTSON
Notary Public in the State of New York
Qualified in Onon. Co. No. 34-8399125
My Commission Expires March 30, 1972

TABLE 1

INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS

FUEL Material	CORE FUEL SINTERED UO_2 PELLETS	TYPE 2 FUEL SINTERED UO_2 PELLETS	TYPE 3 FUEL SINTERED UO_2 PELLETS
Initial Enrichment, w/o U^{235}			
High	2.42	2.79	2.56
Medium	1.67	2.10	1.94
Medium Low	1.19	1.80	1.69
Low		1.40	1.33
Average for Bundle	2.11	2.50	2.30
Pellets, Undished	26 rods	28 rods	32 rods
Pellets, Dished	23 rods	21 rods	17 rods
Pellet Diameter, inches	0.488	0.487	0.487
Density, % of Theoretical	[illegible]	[illegible]	[illegible]
Undished Rods	94.3	94.3	94.3
Dished Rods	91.5	91.5	92.4
Melting Point, °F	5080	5080	5080
CLADDING			
Material	Zr-2	Zr-2	Zr-2
Thickness, inches	0.0355	0.032	0.032
Fuel Rod O.D., inches	0.570	0.563	0.563
FUEL RODS			
Active length, inches	144.0	144.0	144.0
Gas Plenum Length, inches	11-¼	11-¼	11-¼
Fill Gas	Helium	Helium	Helium

TABLE 1

INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS (Cont.)

FUEL BUNDLE	INITIAL CORE FUEL SINTERED UO_2 PELLETS	TYPE 2 FUEL SINTERED UO_2 PELLETS	TYPE 3 FUEL SINTERED UO_2 PELLETS
Geometry	7 x 7	7 x 7	7 x 7
High Enrichment Rods	30	32	32
Medium High Enrichment Rods	16	10	10
Medium Low Enrichment Rods	3	6	6
Low Enrichment Rods		1	1
Poison Rods per Bundle	None	4	3
Rod Pitch, inches	0.738	0.738	0.738
Water to Fuel Volume Ratio	2.38	2.43	2.43
Heat Transfer Area, ft^2	87.6	86.5	86.5
CHANNELS			
Material	Zr-4	Zr-4	Zr-4
Outside Dimension, inches	5.438	5.438	5.438
Wall Thickness, inches	0.080	0.080	0.080
Channel Length, inches	162-1/8	162-1/8	162-1/8

TABLE 2

THERMAL-HYDRAULIC INFORMATION BASED ON DESIGN POWER DISTRIBUTION FOR REFERENCE DESIGN LOADING

	INITIAL CORE FUEL	TYPE 2 and TYPE 3 FUEL	DESIGN LIMITS
POWER PEAKING FACTORS			
RADIAL	1.50	1.50	
AXIAL	1.57	1.57	
LOCAL	1.30	1.30	
TOTAL	3.06	3.06	
BUNDLE POWER (Mw)	5.22	5.22	
MAXIMUM LINEAR HEAT GENERATION RATE (kw/ft)	17.5	17.5	≤17.5
MINIMUM CRITICAL HEAT FLUX RATIO	1.92	1.95	≥1.9

TABLE 3

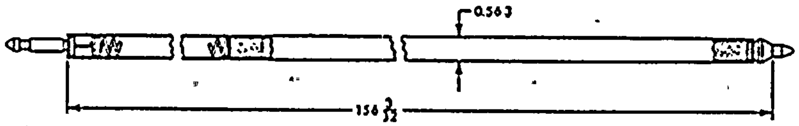
NINE MILE POINT UNIT NO. 1

NUCLEAR CHARACTERISTICS OF CORE

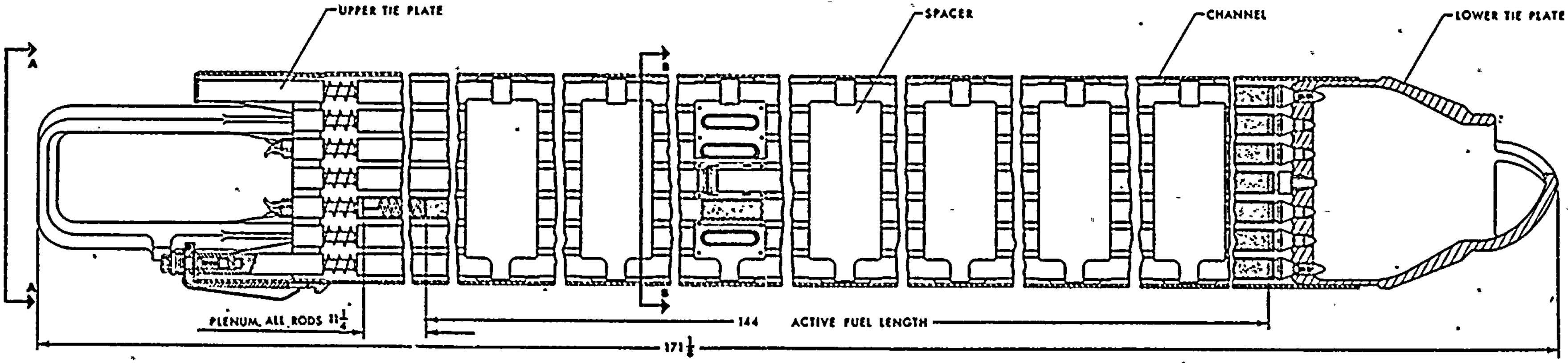
Core Effective Multiplication and Control System Worth	Pre-Outage Core	Refueled Core
(0% Voids, 68° F)		
k_{eff} Uncontrolled	1.13	1.13
Δk Poison Curtains	-0.04	0.00
Δk Control Rods	-0.16	-0.17
k_{eff} Fully Controlled	0.93	0.96
k_{eff} Strongest Single Rod Withdrawn	0.96	0.99

Reactivity Coefficients	Pre-Outage Core	Refueled Core
Steam Void Coefficient at 1850 MWt, 526 BTU/lb inlet enthalpy, 30.5% Voids Δk/k/% Void	-10.2×10^{-4}	-11.25×10^{-4}
Power Coefficient at 1850 MWt, 526 BTU/lb inlet enthalpy Δk/k/Δp/p	-0.060	-0.067
Fuel Temperature Coefficient Δk/k/°F Fuel	-1.2×10^{-5}	-1.2×10^{-5}

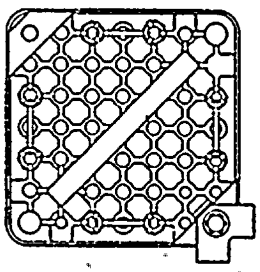
NOTE: Analysis for reloading at a core exposure of 6500 MWD/T.



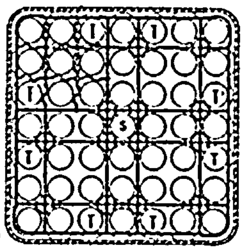
TYPICAL FUEL ROD



TYPICAL FUEL BUNDLE-TYP. 3 (DISUNITED VIEW SHOWN)



VIEW A-A



SECTION B-B

S-SEGMENTED SPACER CAPTURE ROD

T-TIE RODS

NOTE: ALL DIMENSIONS ARE IN INCHES.

TYPE 3 FUEL DESCRIPTION

FIGURE 1

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Received w/Ltr Dated 2-24-72

 FIGURE 2

Wide-Wide Corner

D 4	D 3	D T 3	2	D T 2	D 2	D 3
D 3	2	1	1	1	1	2
D T 3	1	G 1	1	1	1	D T 1
2	1	1	S 1	1	G 1	1
D T 2	1	1	1	1	1	D T 1
D 2	1	1	G 1	1	1	D 1
D 3	2	D T 1	1	D T 1	D 1	2

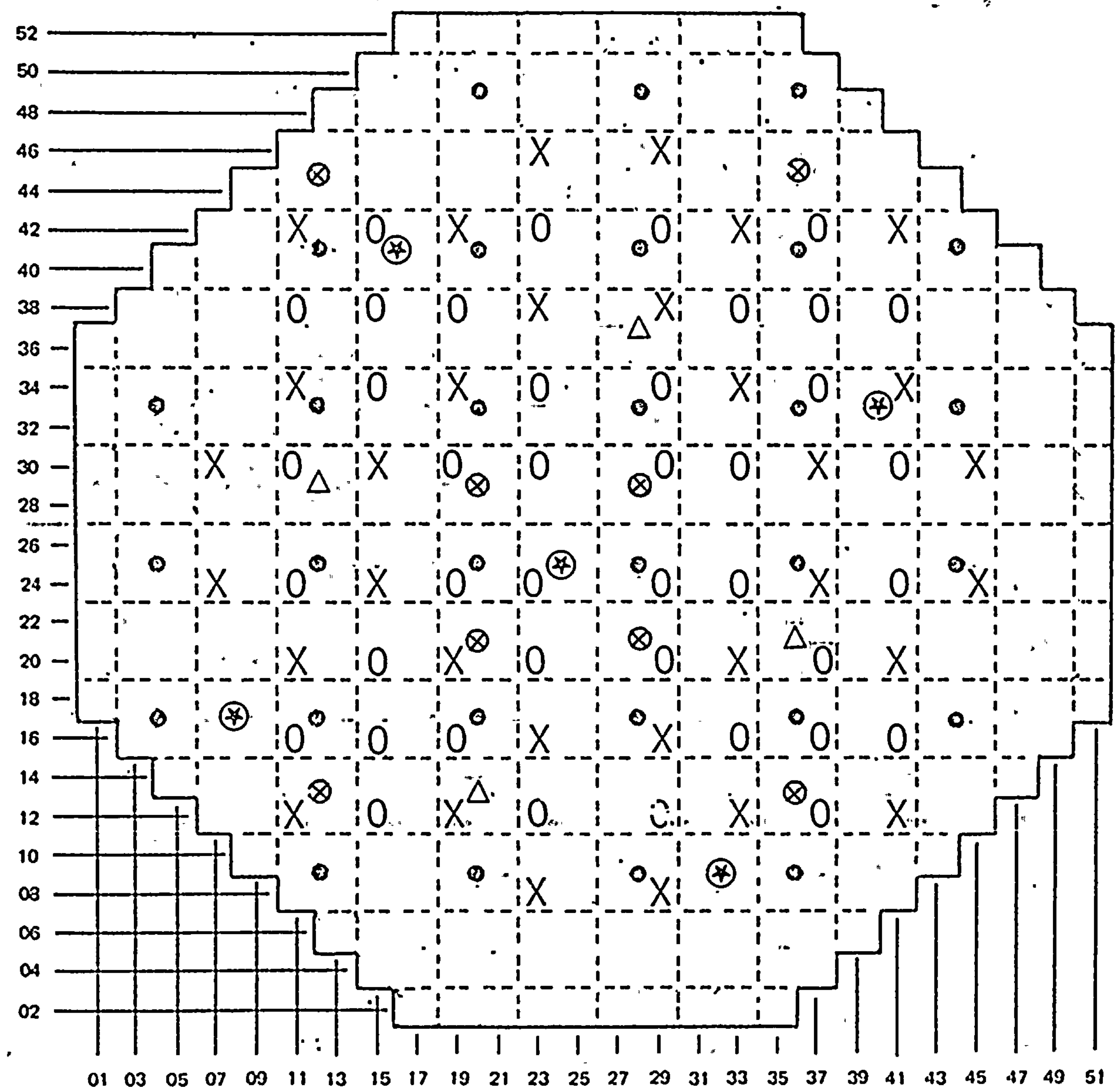
ROD TYPE	ENRICHMENT w/o U-235	NUMBER OF RODS
1	2.56	32
2	1.94	10
3	1.69	6
4	1.33	1

D - Dished Rods
S - Spacer Capture Rod
T - Tie Rods
G - Gadolinia Rods

TYPE 3 FUEL ROD LOCATIONS

Regulatory File Cy. FIGURE 3

Received w/Ltr Dated 2-24-72



CORE PLAN

X UNEXPOSED TYPE 2

O UNEXPOSED TYPE 3

⊙ LPRM LOCATION

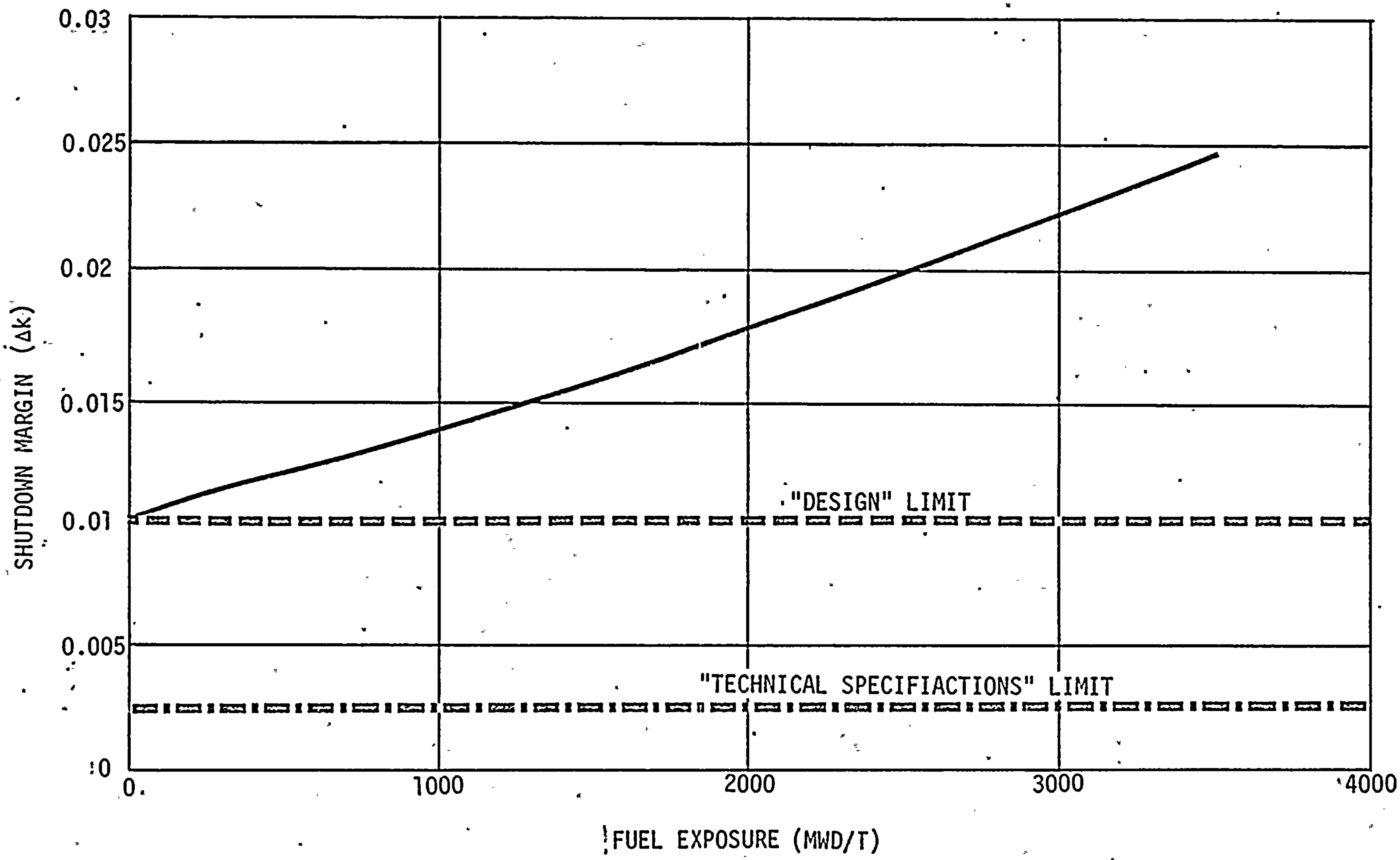
⊗ IRM LOCATION

△ SRM LOCATION

⊛ SOURCE LOCATION

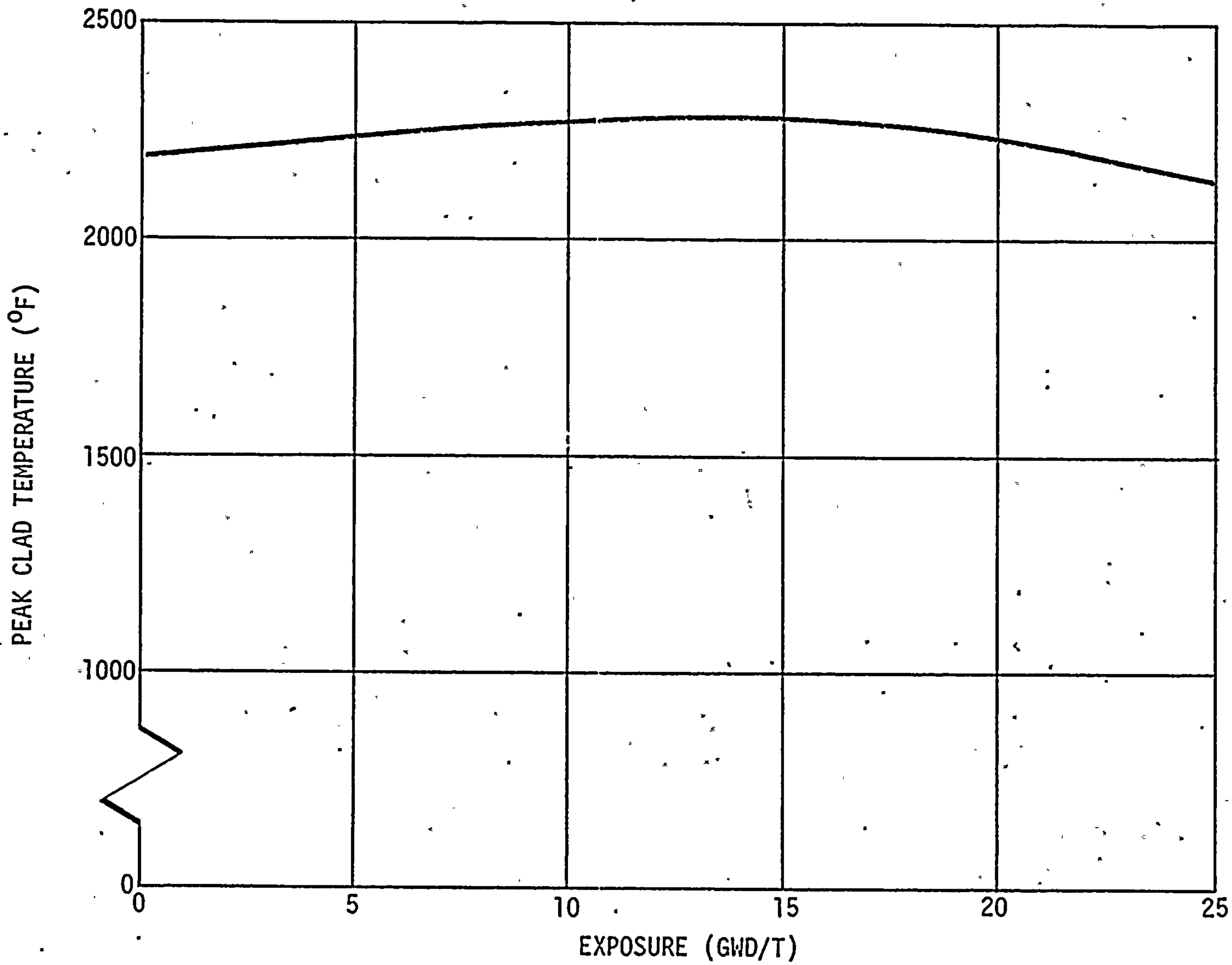
NOTE: ASSEMBLIES WITH A FACE ON THE CORE PERIPHERY HAVE FLOW RESTRICTIVE ORIFICING.

CORE PLAN
NINE MILE POINT
UNIT NO. 1



SHUTDOWN MARGIN VERSUS EXPOSURE

FIGURE 4



PEAK CLAD TEMPERATURES FOR TYPE 3 FUEL
DESIGN BASIS ACCIDENT

FIGURE 5