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DESCRIPTION:
Ltr pursuant to Para 50.59 of 10 CFR 50.... notarized 01-31-73...request change to Tech Specs for Nine Mile Point, Unit 1 w/attached Figures 1 thru 5 and Tables 1 thru 3.......

ENCLOSURES:

ACKNOWLEDGED

DO NOT REMOVE

PLANT NAMES: Nine Mile Point, Unit 1

FOR ACTION/INFORMATION 02-02-73 rht

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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD, WEST
SYRACUSE, N. Y. 13202

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United States Atomic Energy
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RE: Docket 50-220; License DPR-17

Dear

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

The first major refueling outage, previously described(1), is tentatively scheduled for April 22, 1973. However, circumstances may dictate changing the date to as early as March 1, 1973. At this outage, it is planned to replace approximately one fourth of the initial core fuel bundles. These replacements will provide reactivity augmentation for continued high load factor operation through to the next refueling now anticipated for the Spring, 1974.

This letter describes the insertion of up to 32 Type 2 fuel bundles, previously described(2), and up to 108 new Type 4 fuel bundles reported herein. 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 initial core fuel bundles may be reinserted. These latter bundles were discharged during the April 1972 outage.

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

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

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Mechanical Design

The mechanical design has been previously described for the initial core[3], Type 2[4] and Type 3[5] fuel. Type 4 fuel bundles have the same envelope dimensions as the initial core and previous reload fuel. They can be inserted, without restriction, in all locations within the reactor core. Figure 1 is a schematic diagram of the Type 4 fuel assembly; Table 1 summarizes the significant characteristics of all four fuel types.

Basic lattice arrangement of Type 4 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 pheriphery. Figure 2 shows the location of four enrichments within the Type 4 fuel bundle as well as the location of gadolinia-urania rods, tie rods and the spacer capture rod.

Design analyses on Type 4 fuel have been performed which show that the stress integrity limits[6] applied to previous Nine Mile Point reactor fuel are not exceeded at continuous operation with linear heat generation rates up to the operating limit of 17.5 kw/ft.

A value of 1% of plastic strain of the Zircaloy fuel rod cladding has been defined as the limit below which fuel damage due to overstraining the fuel cladding is not expected to occur. Available data indicate that there is a small, but finite, probability that some cladding segments may have plastic elongation less than 1% at failure, but the distribution also indicates that 1% strain value to be the 95% point in the total population. For fresh Type 4 fuel, the calculated linear heat generation rate (LHGR) corresponding to 1% diametral plastic strain of the cladding is 28 kw/ft. Later in life the high exposure fuel has less nuclear capability, due to depletion of fissionable material, and will operate at correspondingly lower powers so that a wide margin is maintained between the operational LHGR and that calculated to cause 1% cladding diametral strain.

The principal changes in the Type 4 fuel, as compared to the previous fuel, are in the areas of pellet geometry, and cladding heat treatment and dimensions. Specifically, there has been a change in pellet diameter from 0.487 to 0.477 inch, a change in pellet length from ~3/4 to ~1/2 inch, a change in the pellet shape to slightly chamfered ends and there will be no fuel rods employing dished fuel pellets. In addition to these changes there

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(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

(5) Docket No. 50-220; Letter: Niagara Mohawk Power Corporation (T. J. Brosnan) to United States Atomic Energy Commission (P. A. Morris), dated February 24, 1972

(6) Docket No. 50-220; Final Safety Analysis Report, Volume I, Section IV p. 60

has also been a change in the Zircaloy cladding heat treatment intended to result in lower statistical variability in cladding mechanical properties. The change in cladding heat treatment involves an increase in stress relief annealing temperature which also results in an increase in ductility and an associated reduction of strength of the material. The reduction in cladding strength has been compensated by an increase in cladding wall thickness from 0.032 inch to 0.037 inch in order to assure that cladding design stress limits are satisfied. These design changes represent improvements which have been incorporated to reduce the probability of fuel rod failure from clad strain localizations caused by pellet-to-cladding interaction. Additional information on the design changes implemented in the Type 4 fuel, including the bases for these changes, can be found in the licensing topical report, NEDO 10505, "Experience with BWR Fuel Through September 1971", by H. E. Williamson and D. C. Ditmore, dated May 1972.

There will also be a hydrogen getter introduced into the plenum of each fuel rod. This getter is introduced as an added precaution against the inadvertent introduction of extraneous hydrogenous impurity during fabrication.

A thermal wafer is being introduced between the bottom pellet and the lower end plug to reduce the operating temperature of the lower end plug. This change is being made to assure that cladding design stress limits in the weld region are met.

The average enrichment of the Type 4 fuel is 2.5 w/o. Fuel Type 4 has 4 gadolinia-urania rods for supplementary reactivity control. The peak power gadolinia-urania rod in Type 4 fuel will operate at a power no greater than 0.91 of the peak UO_2 rod. This, then, limits the gadolinia rods to a linear heat generation rate (LHGR) of 15.9 kw/ft, as compared with a LHGR of 17.5 kw/ft for the limiting UO_2 rods. The gadolinia-urania rods are less limiting in thermal-mechanical performance than the UO_2 rods throughout fuel life.

Thermal-Hydraulic Characteristics

The refueled core is designed to operate with a maximum linear heat generation rate $\leqslant$ 17.5 kw/ft and a minimum critical heat flux ratio $\geqslant$ 1.90. The thermal-hydraulic characteristics of Type 4 fuel are essentially the same as for the initial core fuel and identical to the Type 2 and Type 3 fuel.

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 temperature and void dependent characteristics of the reloaded core will not differ significantly from the previous loading condition and will satisfy all design and operating criteria.

B. Reactor Shutdown Margin

The reloaded core can 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 0.022 Δk was calculated for the reloaded core as shown in Figure 3. This core loading assumes insertion of the 32 remaining unexposed Type 2 reload bundles, and insertion of 88 of the 108 available Type 4 reload bundles. This minimum shutdown margin of 0.022 Δk was calculated for an assumed refueling at a pre-outage core average exposure of 9800 MWD/T (an exposure increase of 3800 MWD/T from the spring 1972 outage). The shutdown margin will increase throughout the balance of this cycle as shown in Figure 4.

A 0.022 Δk shutdown margin has also been calculated for a refueling utilizing all available 140 bundles (32 Type 2 and 108 Type 4 bundles) in a scatter pattern similar to Figure 3 at a pre-outage core average exposure of 9800 MWD/T.

Shutdown margins were also considered for a refueling at a pre-outage core average exposure of only 9200 MWD/T (an exposure increase of 3200 MWD/T from the spring 1972 outage). For loading up to 140 fresh bundles in a scatter pattern, the shutdown margins are significantly greater than the 1% Δk design limit.

For all cases considered, 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.

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 4 fuel since the k_∞ of the replacement fuel in its most reactive condition is less than k_∞ of the initial fuel assemblies without curtains.

Safety Analysis

The safety analyses described in the Petition to Increase Power Level[7] and its Addendums and more recent analyses[8] have been reviewed to see which of the transients and/or accidents, if any, may be affected by the introduction of Type 4 reload fuel. For these reviews, the Hench-Levy correlation has continued to be used to determine the safety limit and to establish margins from the normal operating points to the safety limit. Similarly, the same considerations, margins, damage limits, and operating limits described in previous licensing submittals were employed.

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(7) Docket No. 50-220; Technical Supplement to Petition to Increase Power Level, Section II

(8) Docket No. 50-220; Letter: Niagara Mohawk Power Corporation (T. J. Brosnan) to United States Atomic Energy Commission (P. A. Morris), dated February 28, 1972.

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 analyses for the control rod drop, refueling and main steam line break accidents show that the principal parameters are not significantly affected. Thus, these accident results remain unchanged.

In the case of the postulated control rod drop accident, utilization of the rod worth minimizer supplemented by procedural controls, will continue to ensure that the peak enthalpy for this accident will not exceed the 280 cal/gm design basis limit; and the calculated fission product releases would be no more than previously reported(9). The rods containing the Gd_2O_3 will have a lower failure threshold. However, as also shown in the section on mechanical design, the operating power levels will be lower than the maximum powered UO_2 rods. The net result to the gadolinia-containing rods during the postulated accident is that they remain further away from the design basis limit than the maximum powered UO_2 rods and, therefore, do not affect the results of the accident.

With regard to postulated loss of coolant accidents, previously reported(10),(11) calculations have demonstrated that the initial core and Type 2 and Type 3 fuel comply with AEC Interim Acceptance Criteria for the performance of Emergency Core Cooling Systems.

The enrichment levels of Type 4 fuel are slightly different than those of initial core or reload fuel previously reviewed. In addition, the gadolinia content is different than previously reviewed reload fuel. The distribution of local peaking factors and changes in peaking factors with exposure are, therefore, different for Type 4 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 4 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 4 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.

(9) Docket No. 50-220; Final Safety Analysis Report, Appendix E

(10) 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, 1971

(11) Docket No. 50-220; Letter: Niagara Mohawk Power Corporation (T. J. Brosnan) to United States Atomic Energy Commission (P. A. Morris), dated February 24, 1972

The analyses are based on models previously used for Nine Mile Point Unit 1, the results of which are summarized below.

	Max. Peak Clad Temperature (F)	Max. Total Metal Water Reaction (%)
AEC Criteria	2300	1
Initial Core Fuel	2237	0.25
Type 2 Reload Fuel	2265	0.25
Type 3 Reload Fuel	2280	0.25
Type 4 Reload Fuel	2270	0.25

These results show that the refueled core will continue to meet the AEC Interim Acceptance Criteria.

It should be recognized that there is additional conservatism in these calculations as discussed in a recent paper, "Realistic Evaluation of Core Thermal Response during a Loss-of-Coolant Accident for BWR", by Allan Rodgers presented at the 1972 Winter ASME meeting (No. 72-WA/HT-48).

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 worst case loading error for the reference core configuration occurs when 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. The Type 2 loading error is worse than the similar Type 3 or Type 4 loading error because the Type 2 bundle contains a higher enrichment and/or lower gadolinia concentration than the other bundles.

Conclusions

Type 2 and Type 4 reload fuel have been designed to be compatible with and closely match the mechanical, nuclear and thermal-hydraulic characteristics of the initial core, 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 Station's Operation Review Committee and the Safety Review and Audit Board which concur with these findings.

Very truly yours,

T J Brosnan

T. J. Brosnan
Vice President and Chief Engineer

State of New York, County of Onondaga,
Sworn to before me this 31st day of January, 1973

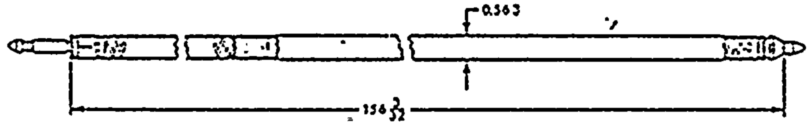
Carolyn F. Robertson

Notary Public

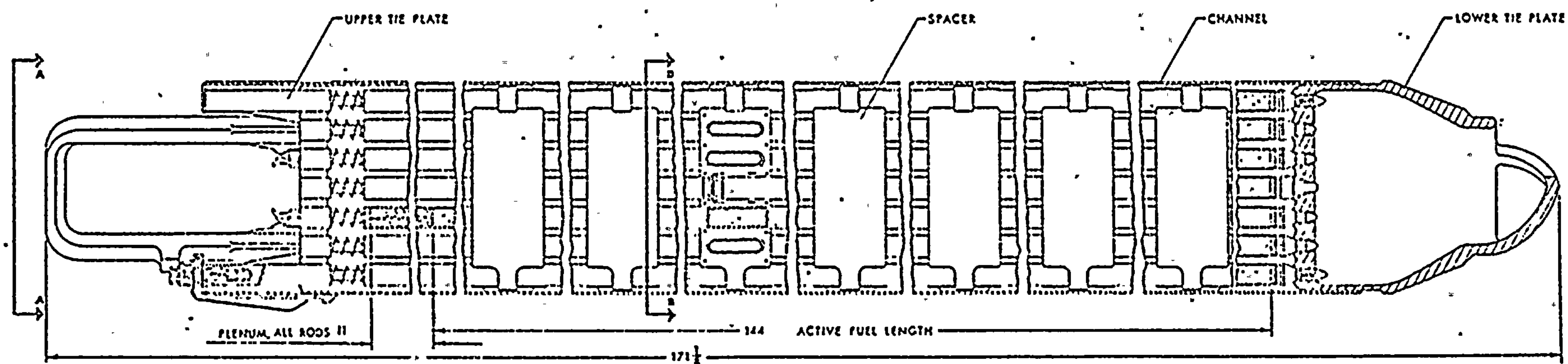
CAROLYN F. ROBERTSON
Notary Public in the State of New York
Qualified in Onon. Co. No. 34-8599125
My Commission Expires March 30, 1974

TJB/sq

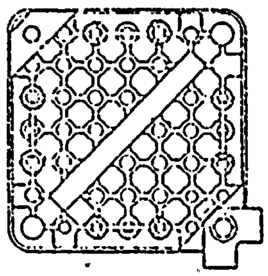
Attachments



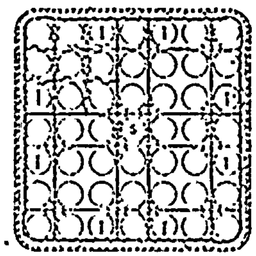
TYPICAL FUEL ROD



TYPICAL FUEL BUNDLE-TYPE 4 (DISUNITED VIEW SHOWN)



VIEW A-A



SECTION B-B

S-SEGMENTED SPACER CAPTURE ROD

T-TIE RODS

NOTE: ALL DIMENSIONS ARE IN INCHES.

TYPE 4 FUEL DESCRIPTION

FIGURE 1

Figure 2

TYPE 4 FUEL ROD LOCATIONS

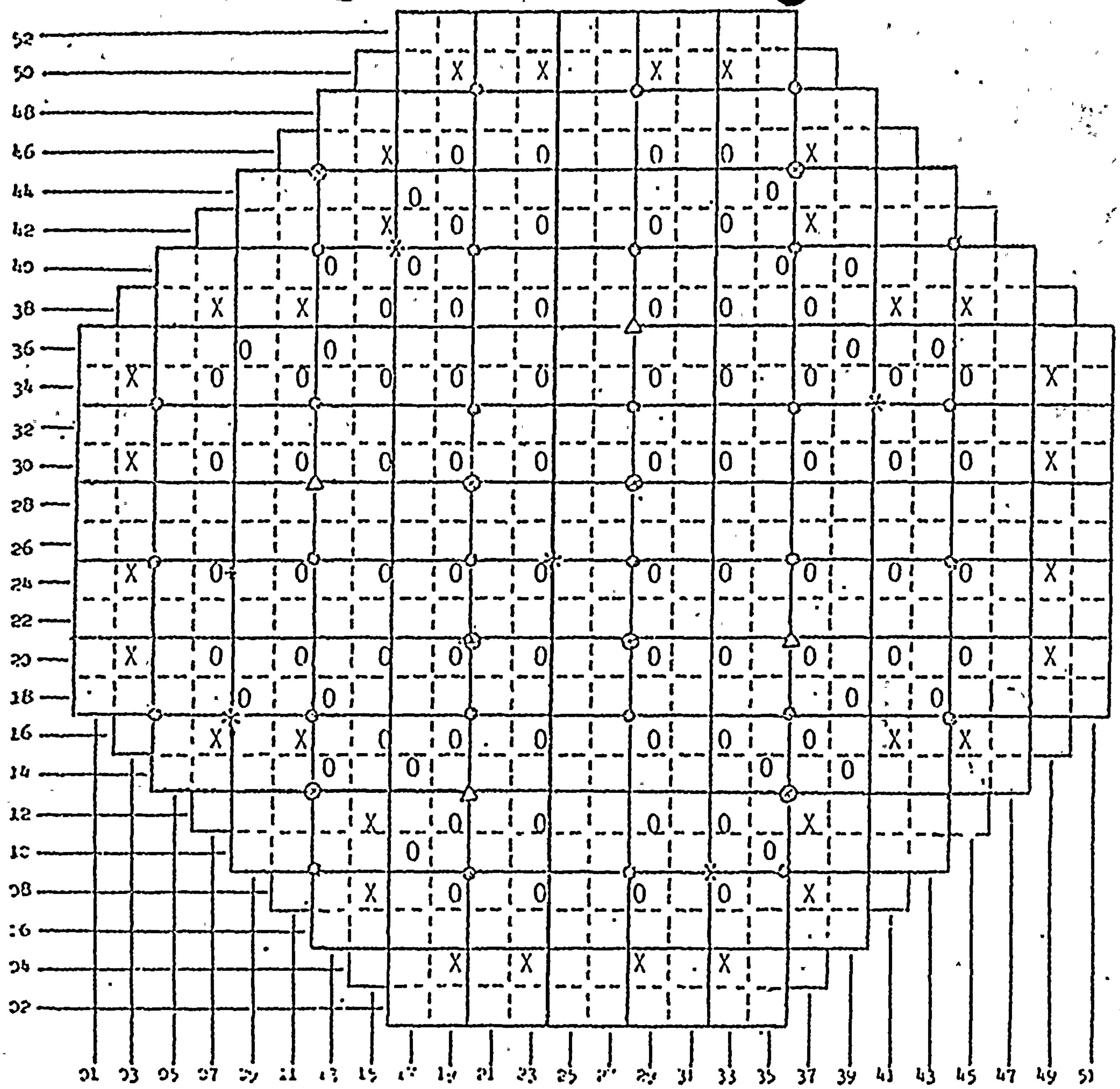
wide-wide corner

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

ROD TYPE	ENRICHMENT w/o U-235	NUMBER OF RODS
1	2.79	32
2	2.09	10
3	1.80	6
4	1.40	1

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

Figure 3



X = Unexposed Type 2

0 = Unexposed Type 4

⊙ LPRM LOCATION

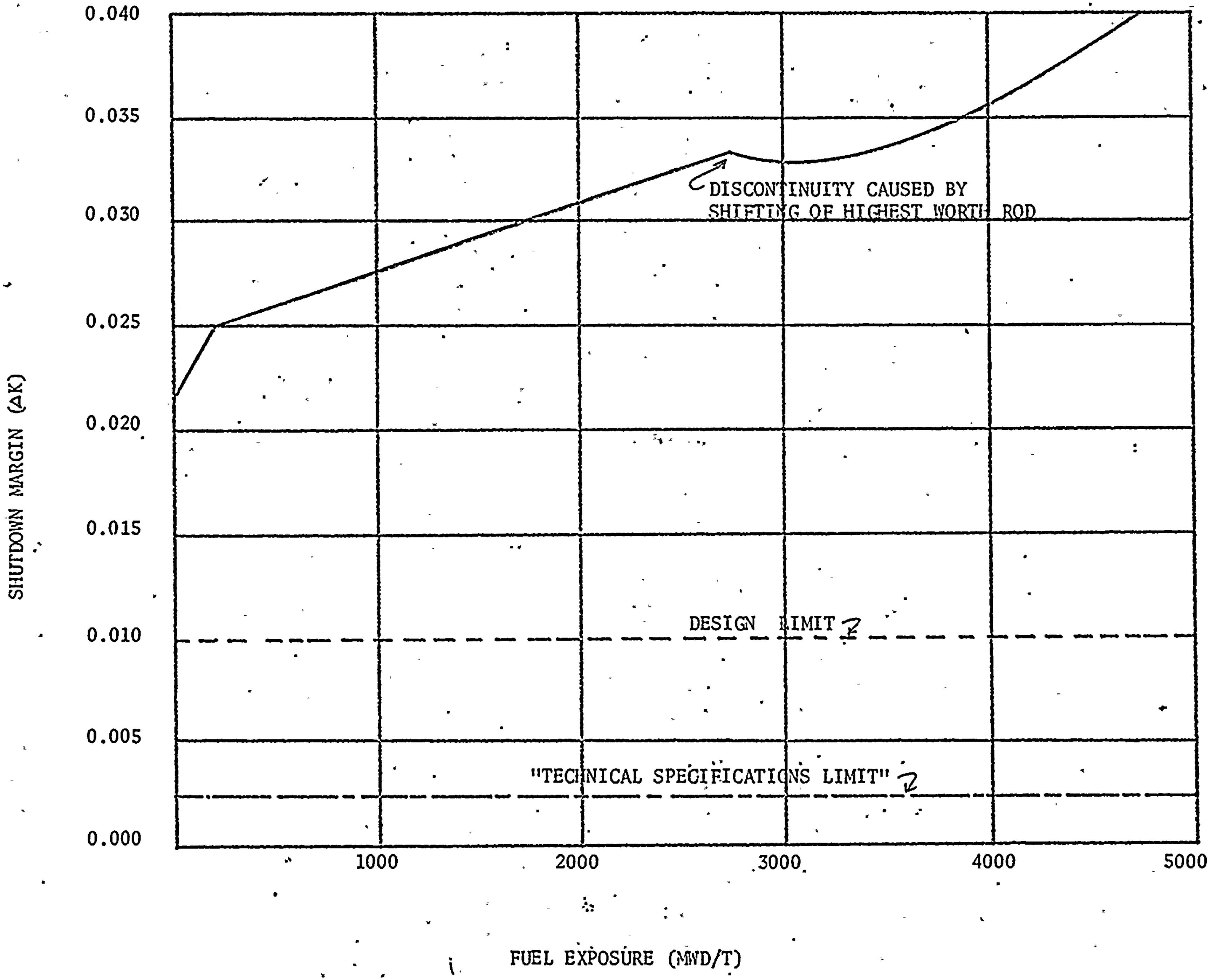
⊗ IRM LOCATION

△ SRM LOCATION

✳ SOURCE LOCATION

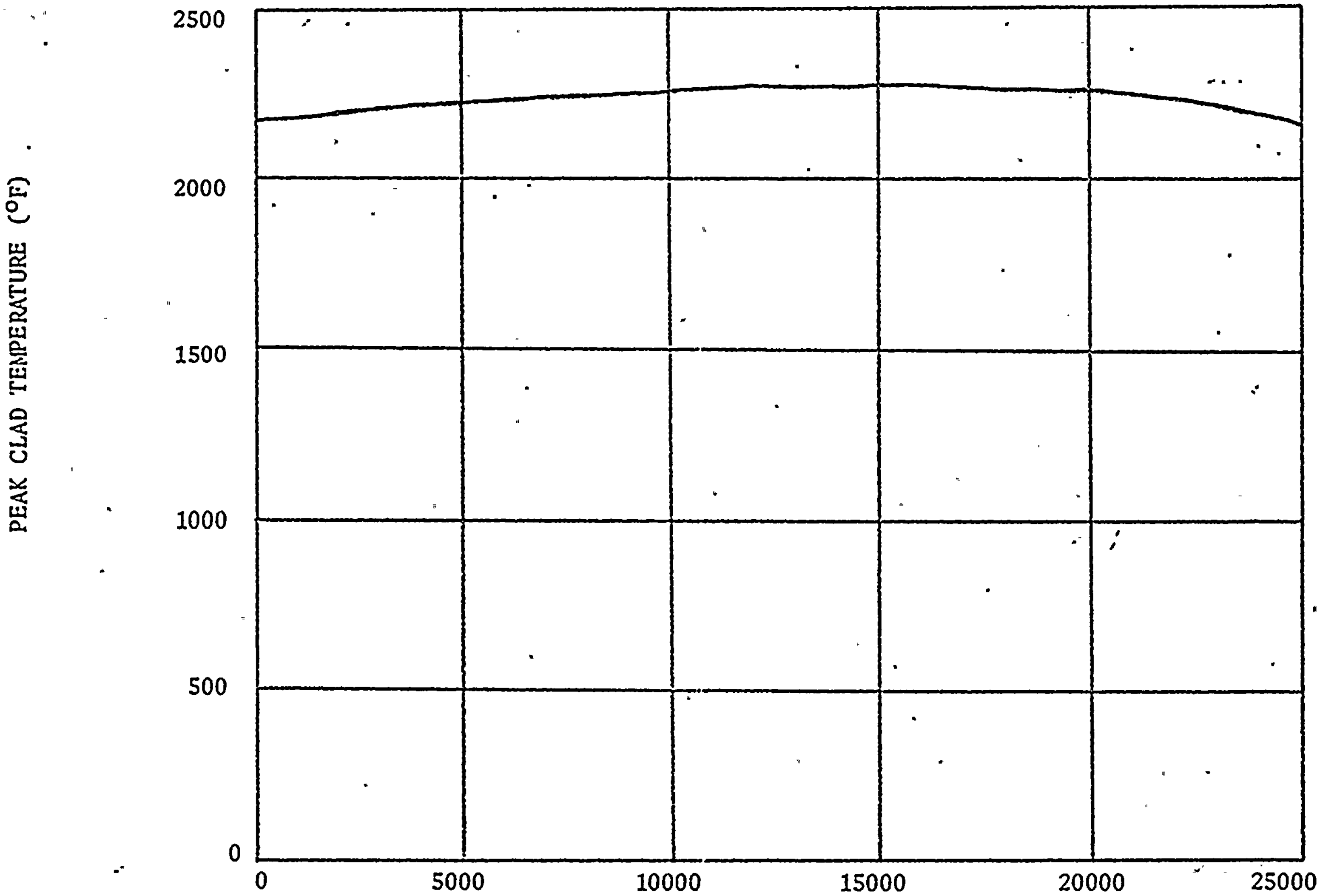
CORE PLAN
NINE MILE POINT
UNIT NO. 1

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



SHUTDOWN MARGIN VERSUS EXPOSURE

FIGURE 4



EXPOSURE (MWD/T)
PEAK CLAD TEMPERATURES FOR TYPE 4 FUEL
DESIGN BASIS ACCIDENT

FIGURE 5

TABLE 1
INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS

FUEL	INITIAL CORE FUEL	TYPE 2 FUEL	TYPE 3 FUEL	TYPE 4 FUEL
Material	SINTERED UO_2 PELLETS	SINTERED UO_2 PELLETS	SINTERED UO_2 PELLETS	SINTERED UO_2 PELLETS
Initial Enrichment, w/o U^{235}				
High	2.42	2.79	2.56	2.79
Medium High	1.67	2.10	1.94	2.09
Medium Low	1.19	1.80	1.69	1.80
Low		1.40	1.33	1.40
Average for Bundle	2.11	2.50	2.30	2.50
Pellet Geometry				
Long Dished	23 rods	21 rods	17 rods	0 rods
Long Undished	26 rods	28 rods	32 rods	0 rods
Short Chamfered (undished)	0 rods	0 rods	0 rods	49 rods
Pellet Diameter, inches	0.488	0.487	0.487	0.477
Density, % of Theoretical				
Dished Rods	91.5	91.5	92.4	N. A.
Undished Rods	94.3	94.3	94.3	94.1
Melting Point, °F	5080	5080	5080	5080
CLADDING				
Material	Zr-2	Zr-2	Zr-2	Zr-2
Thickness, inches	0.0355	0.032	0.032	0.037
Fuel Rod O. D., inches	0.570	0.563	0.563	0.563
FUEL RODS				
Active length, inches	144.0	144.0	144.0	144.0
Gas Plenum Length	11-1/4	11-1/4	11-1/4	11
Fill Gas	Helium	Helium	Helium	Helium
Getter	No	No	No	Yes
SPACERS				
Number per bundle	7	7	7	7
Material	Stainless Steel with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs	Zr-4 with Inconel Springs

TABLE 1
INITIAL CORE AND RELOAD FUEL ASSEMBLY DESIGN SPECIFICATIONS (Cont.)

FUEL BUNDLE	INITIAL CORE FUEL	TYPE 2 FUEL	TYPE 3 FUEL	TYPE 4 FUEL
Geometry	7X7	7X7	7X7	7X7
High Enrichment Rods	30	32	32	32
Medium High Enrichment Rods	16	10	10	10
Medium Low Enrichment Rods	3	6	6	6
Low Enrichment Rods		1	1	1
Poison Rods per Bundle	None	4	3	4
Rod Pitch, inches	0.738	0.738	0.738	0.738
Water to Fuel Volume Ratio	2.38	2.43	2.43	2.53
Heat Transfer Area, ft^2	87.6	86.5	86.5	86.5

CHANNELS				
Material	Zr-4	Zr-4	Zr-4	Zr-4
Outside Dimension, inches	5.438	5.438	5.438	5.438
Wall Thickness, inches	0.080	0.080	0.080	0.080
Channel Length, inches	162-1/8	162-1/8	162-1/8	162-1/8

TABLE 2

THERMAL-HYDRAULIC INFORMATION BASED ON DESIGN POWER

DISTRIBUTION FOR REFERENCE DESIGN LOADING

POWER PEAKING FACTORS	INITIAL CORE FUEL	TYPE 2 and TYPE 3 FUEL	TYPE 4 FUEL	DESIGN LIMITS
RADIAL	1.50	1.50	1.50	
AXIAL	1.57	1.57	1.57	
LOCAL	1.30	1.30	1.30	
TOTAL	3.06	3.06	3.06	
BUNDLE POWER (Mw)	5.22	5.22	5.22	
MAXIMUM LINEAR HEAT GENERATION RATE (kw/ft)	17.5	17.5	17.5	17.5
MINIMUM CRITICAL HEAT FLUX RATIO	1.92	1.95	1.94	1.9

TABLE 3

NINE MILE POINT UNIT NO. 1 NUCLEAR CHARACTERISTICS OF CORE

Core Effective Multiplication and Control System Worth	Pre-Outage Core*	Reloaded Core*
(0% Voids, 68°F)		
k_{eff} Uncontrolled	1.08	1.11
Δk Control Rods	-0.15	-0.16
K_{eff} Fully Controlled	0.93	0.95
k_{eff} Strongest Single Rod Withdrawn	0.96	0.98
Reactivity Coefficients		
Steam Void Coefficient at 1850 MWt, 526 BTU/lb Inlet Enthalpy, 30.5% Voids $\Delta k/k/\%$ Void, Range of Values	-8.9×10^{-4}	-9.42×10^{-4} to -8.83×10^{-4}
Power Coefficient at 1850 MWt, 526 BTU/lb Inlet Enthalpy, $\Delta k/k/\Delta P/P$, Range of Values	-0.050	-0.050 to -0.043
Fuel Temperature Coefficient, $\Delta k/k/°F$ Fuel, Range of Values	-1.13×10^{-5}	-1.08×10^{-5} to -1.14×10^{-5}

* Analysis for refueling at a core exposure of 9800 MWD/T.