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| Niagara Mohawk Power Corporation<br>Syracuse, N.Y. 13202<br>T. J. Brosnan | July 27, 1971 | July 29, 1971 | 3430 |

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| Dr. Peter A. Morris | 1 signed & 22 conf'd | | |

| ACTION NECESSARY ☐ | CONCURRENCE ☐ | DATE ANSWERED: |
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DESCRIPTION: (Must Be Unclassified)
Ltr re meeting on 5-20-71...providing info re status of off-gas & tentative plans for partial refueling in the Fall of 1971 at Nine Mile Point Plant w/attchmt's Table 1-3 & Fig's 1-4....

ENCLOSURES:

REMARKS:

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U.S. ATOMIC ENERGY COMMISSION MAIL CONTROL FORM FORM AEC-326S (6-60)

Regulatory File Cy. 50-220

# NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

SYRACUSE, NEW YORK 13202

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July 27, 1971

RECEIVED JUL 29 1971 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

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

Dear Dr. Morris:

During our meeting of May 20, 1971, we described the status of off-gas at Nine Mile Point and our tentative plans for a partial refueling in the Fall of 1971 in order to maintain releases as low as practicable. Immediately following this meeting, we proceeded to prepare detailed, descriptive materials and analyses covering this heretofore unanticipated refueling.

The outage for this refueling is now planned to begin on September 19, 1971. We have, in the short time since our meeting with the Commission on May 20, 1971, taken steps, as outlined below, to assure that this schedule can be maintained.

(a) Design for this reload fuel was finalized and the preparation of the data reported herein was completed.

(b) Special arrangements were made with the Commission to provide the necessary enrichment services on an abbreviated schedule and with General Electric for immediate fabrication of this reload fuel.

(c) Schedules for planned maintenance within Niagara Mohawk's system and within the New York Power Pool were modified to accomodate this refueling.

Described and reported herein are analyses supporting (1) refueling of up to 56 fuel assemblies (Type 2) on the periphery of the core, (2) removal from the periphery of the core of up to 48 temporary control curtains, and (3) reconstitution of initial core fuel bundles.

3430

MECHANICAL DESIGN

The Type 2 reload fuel bundle for the Nine Mile Point reactor is designed to be compatible with the initial core fuel. The Type 2 reload fuel bundle is illustrated in Figure 1, and Table 1 compares the characteristics of the initial and reload fuel. Basic lattice arrangement for reload fuel remains the same as the initial core fuel, with 49 rods in a 7 x 7 array, a centrally located spacer capture rod, and eight tie rods located on the periphery. Figure 2 shows the location of the four enrichments within the fuel assembly as well as the location of dished rods, tie rods and spacer capture rod.

Type 2 fuel rods incorporate slight dimensional changes from initial core fuel in pellet diameter, clad thickness and rod diameter as indicated in Table 1. These changes reflect the mechanical design common to current General Electric product line fuel. Type 2 reload core fuel, which also uses zirconium spacers instead of stainless steel, has been analyzed for the design operating and damage limits listed in the section on thermal hydraulics.

The average design enrichment of Type 2 reload fuel is 2.50 weight percent as compared with an as-built enrichment of 2.11 percent for the initial core. The reload enrichment is representative of the average enrichment required for future equilibrium reload cores. The reload bundle contains four enrichments, as compared with three in the initial core.

The reactivity control for the proposed core loading is supplemented by the use of burnable poison in each reload bundle. Four fuel rods of each reload fuel bundle contain a small fraction of gadolinium oxide ($Gd_2O_3$) homogeneously mixed with the $UO_2$. The dimensions of the gadolinium bearing rods are the identical to the $UO_2$ rods. The presence of gadolinium in the $UO_2$ pellets reduces the lineal heat generation rates at which incipient center melting and 1% cladding strain will occur relative to the lineal heat generation rate (LHGR) for $UO_2$ rods. The LHGR to reach incipient center melting for the particular $Gd_2O_3$ concentration in Type 2 fuel rods is approximately 19.5 - 20.5 kw/ft as compared with 21 - 22 kw/ft for $UO_2$ rods, and the LHGR for 1% cladding strain is greater than 26.5 kw/ft compared with 28 kw/ft for $UO_2$ rods. However, the design for reload fuel assures that the margin between the operating power and the damage limit is greater for rods containing $Gd_2O_3$ than for $UO_2$ rods because $Gd_2O_3$ rods will never exceed 0.80 of the peak design power in the fuel bundle. The lower power for $Gd_2O_3$ rods in each bundle is due to their placement away from the higher flux region which is adjacent to the water surrounding each bundle. Their performance has been successfully demonstrated in over 300 assemblies in the Dresden 1, Big Rock Point and Humboldt reactors. Similar gadolinium-containing fuel bundles have been reviewed by the Commission on several other dockets; e.g., Dresden 1, (50 - 10), Big Rock Point (50 - 155), Humboldt Bay Unit 3 (50 - 153), Dresden 2 (50 - 237), and Quad Cities 1 (50 - 254) and Quad Cities 2 (50 - 265).

THERMAL HYDRAULIC CHARACTERISTICS

As shown in Figure 3, the reloaded core will contain up to 56 Type 2 reload bundles loaded on the periphery with the balance of the fuel being initial core fuel. Type 2 reload fuel will be subject to operating conditions substantially below those for initial core fuel because of their placement in the relatively low power region near the periphery of the core. The operating parameters, which are monitored periodically during power operation, are shown below for the design power distribution of the refueled core.

| | Type 1 Initial Fuel Center | Type 1 Initial Fuel Periphery | Type 2 Reload Fuel Periphery |
|---|---|---|---|
| Bundle Power (MW) | 5.22 | 3.30 | 3.30 |
| Maximum Lineal Heat Generation Rate (kw/ft) | 17.5 | 12.8 | 12.8 |
| Minimum Critical Heat Flux Ratio | 1.9 | 2.23 | 2.25 |

Therefore, the initial core fuel continues to be the most limiting fuel in the core and previous* core operating limits and the margin between operating and damage limit remain applicable to the refueled core.

The thermal hydraulic characteristics of Type 2 reload fuel are very nearly the same as for initial core fuel. Type 2 fuel has a smaller fuel rod diameter and a different spacer loss coefficient and will, therefore, receive approximately 2 percent more coolant flow than a comparable initial core fuel bundle in the peripheral location. The smaller fuel rod diameter also reduces the heat transfer area by approximately 1.3 percent, thereby increasing the heat flux relative to an initial core fuel rod at the same power. The higher flow in Type 2 reload fuel will reduce the coolant flow in the initial core fuel bundles, but such reduction will be less than 0.2 percent. These slight perturbations in flow and heat flux have no substantial effect on the fuel; but are, nevertheless, accounted for in maintaining a minimum critical heat flux ratio $\geqslant$1.9 during normal operation. Thus, Type 2 fuel has no major effect on the thermal hydraulic characteristics of the refueled core. Table 2, which has been taken from the Petition to Increase Power Level**, is applicable to the refueled core with the addition of appropriate footnotes.

*Technical Supplement to Petition to Increase Power Level, pp. II-4 -II-8

**Technical Supplement to Petition to Increase Power Level, p. II-5

## NUCLEAR CHARACTERISTICS OF THE FUEL AND CORE

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

A tabulation of the typical nuclear characteristics of the initial core and of the reloaded core is provided in Table 3. Since the geometry and materials used in the construction of the replacement fuel are nearly identical to those of the initial fuel, the nuclear characteristics of the reload fuel are sufficiently close to those of the initial fuel that temperature and void dependent characteristics of the reloaded core will not differ significantly from the values previously reported.*

### B. Reactor Shutdown Margin

The refueled core fully meets criteria established for the initial core in that it may be maintained subcritical in the most reactive condition throughout the subsequent operating cycle with the most reactive rod in its full-out position and all other rods fully inserted. A shutdown margin of greater than $0.02\Delta k$ was calculated for a reload core with the 48 peripherally located temporary control curtains removed and 56 Type 2 reload fuel bundles loaded on the periphery as shown in Figure 3. This shutdown margin increases with fuel exposure. Figure 4 assumes the refueling occurs at an initial core exposure of 4,750 MWD/T. Alternate calculations have been made for a refueling at 4,000 MWD/T which show that shutdown margin will be greater than $0.02\Delta k$ at all times throughout the subsequent cycle. It is anticipated that the refueling will occur at an initial core exposure greater than 4,000 MWD/T.

Removal of the remaining temporary control curtains may be scheduled prior to the next refueling but in no event will curtains be removed so as to reduce the calculated shutdown margin to less than $0.01\Delta k$. Testing will be performed at the beginning of the second cycle and at the time curtains are removed to assure that the shutdown margin, as specified in the Technical Specifications**, is maintained. The reloaded core satisfies the shutdown criteria throughout the cycle as shown in Figure 4. The value of R referred to in the Technical Specifications remains at zero for the second cycle because the shutdown margin does not decrease from its initial value at the beginning of the cycle.

### C, Liquid Poison System

The liquid poison system is designed to provide the capability to bring the reactor from full design rating (1850 MWt) to a cold xenon-

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* Final Safety Analysis Report, Volume 1, Section IV and Volume II, Appendix E and Petition to Increase Power level, Section II, pp. II-4-II8.

** Technical Specifications and Bases for Nine Mile Point Nuclear Station Operation at 1850 Thermal Megawatts, p. 21.

free shutdown condition (keff <0.97) assuming none of the control rods can be inserted. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full-power and cold shutdown. The liquid poison system capability has been examined and is found to be adequate as the reference power level of 1850 MWt has not changed and the core reactivity effects of voids and temperature (Table 3) have not been significantly altered by introduction of reload fuel.

D. Reactivity of Fuel in Storage

There is no new safety implication with the spent fuel storage pool and new fuel storage rack configuration because the $k_\infty$ of the replacement fuel in its most reactive condition is not more than $k_\infty$ of the initial fuel assemblies without temporary control curtain.

REFUELING PROCEDURE

During the refueling shutdown the initial core fuel will be inspected so as to locate fuel bundles and fuel rods which are contributing to radioactivity in off-gas. Such fuel rods having been identified will be removed from the fuel bundle and replaced with sound rods from other initial core fuel bundles. This reconstitution of initial core fuel bundles is planned during the refueling outage. Criteria governing this refueling operation are outlined below which assure that the reconstituted fuel will not differ significantly from the original fuel bundle.

(a) The mechanical design of the fuel bundle precludes the possibility of putting a rod of high initial enrichment into a bundle location for a lower enrichment rod because the and plug shank diameter is smaller for a lower enrichment location. Exposure differences between defective rods and replacement rods will be limited to values such that the peaking factor and power distribution will be essentially the same as those for initial fuel. This assures that operating conditions of reconstituted fuel bundles will not be more limiting than those for initial core fuel bundles. All the same limits and safety analyses discussed herein and in the Petition to Increase Power Level and its Addendums apply.

(b) The maximum reactivity change of any reconstituted bundle will be less than $\pm$ .003 $\Delta k$ when compared with the original bundle. The net core reactivity change due to the reconstitution of fuel bundles will be significantly less than 0.003 $\Delta k$.

Type 2 reload fuel will be thoroughly inspected at Nine Mile Point prior to its use in the reactor. In addition, Niagara Mohawk will, with the assistance of a consultant, review the quality control procedures and records appropriate to the fabrication of Type 2 fuel.

The refueling and subsequent startup will be conducted in accordance with the Technical Specifications and the established Operating Procedures. Periodic tests and surveillance requirements will be performed during the refueling outage as required by the Technical Specification. Where appropriate, detailed instructions will be prepared to supplement the existing procedures with respect to fuel handling and fuel movements during the refueling. In particular, detailed records will be kept of all rod movements by the personnel performing the reconstitution procedure. In addition, all rod movements will be observed and recorded by personnel other than those actually performing the reconstitution.

## SAFETY ANALYSIS

The safety analyses described in the Petition to Increase Power Level and its Addendums have been reviewed to see which of the transients and/or accidents may be affected by the introduction of Type 2 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.*

The possible effect of Type 2 reload fuel on transients, as described in the Petition to Increase Power Level, caused by a single operator error or equipment malfunction which are abnormal but could be reasonably anticipated is directly related to the void coefficient, power level, and scram reactivity. As shown in Table 3, there is no change in the power or temperature coefficients for the reload core and only a slight change in the void coefficient. However, this latter value is well within the range of void coefficients assumed in previous analyses.* It is concluded that the results of the transient analyses as reported in the Petition to Increase Power Level are applicable to the reloaded core because (1) transient effects on the fuel and primary system are not significantly affected, and (2) the operating and damage limits assumed in the transient analyses remain applicable and, therefore, the results of the analyses do not change. The results of analyses for postulated accidents which might result in a release of fission products as reported in the Petition to Increase Power Level, have been reviewed to see if they could be affected by use of Type 2 reload fuel. The control rod worths, the thermal-hydraulic parameters and the fission product inventories for the refueled core are substantially the same as for the original core. Therefore, the analyses for those accidents dependent on these parameters (Control Rod Drop, Refueling, and Main Steam Line Break Accidents) remain unchanged.

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*Technical Supplement to Petition to Increase Power Level, Section II-xv

The Loss of Coolant Accident is dependent on the expected bundle power. Operating parameters for the Type 2 reload fuel on the periphery are substantially below those for fuel operating in the center of the reactor. With the lineal heat generation rate and total bundle power for the reload fuel less than 75 percent of that for limiting initial core fuel, it is assured that previous analyses for this accident also remain applicable.

CONCLUSIONS

Type 2 reload fuel has 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 accomodate this fuel; (2) this refueling 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 (3) there is reasonable assurance that the health and safety of the public will not be endangered by this refueling. This matter has been reviewed by the Safety Review and Audit Board which concurs with these findings. Therefore, Niagara Mohawk believes this is the type of facility change permitted to be made under Section 50.59 of the Commission's Regulations.

Very truly yours,

T. J. Brosnan

T. J. Brosnan
Vice President and Chief Engineer

Attachments

Regulatory File Cy.

Received w/Ltr Dated 7-27-71

TABLE 1

NINE MILE POINT UNIT NO. 1

COMPARISON: INITIAL CORE AND TYPE 2 RELOAD FUEL ASSEMBLY DESIGN

| Fuel | Initial Core Fuel | Type 2 Reload Fuel |
|---|---|---|
| Material | Sintered $UO_2$ Pellets | Sintered $UO_2$ Pellets |
| Initial Enrichment, w/o $U^{235}$ | | |
| High | 2.42 | 2.79 |
| Medium | 1.67 | 2.10 |
| Medium Low | 1.19 | 1.80 |
| Low | | 1.40 |
| Average for Bundle | 2.11* | 2.50 |
| Pellets, Undished | 26 rods | 28 rods |
| Pellets, Dished | 23 rods | 21 rods |
| Pellet diameter, inches | 0.488 | 0.487 |
| Density (% of Theoretical) | 94.3 | 94.3 |
| Melting Point, $^{o}F$ | 5080 | 5080 |
| **Cladding** | | |
| Material | Zr-2 | Zr-2 |
| Thickness, inches | 0.0355 | 0.032 |
| Fuel Rod O.D., inches | 0.570 | 0.563 |
| **Fuel Rods** | | |
| Active length, inches | 144.0 | 144.0 |
| Gas Plenum Length, inches | 11-½ | 11-½ |
| Fill Gas | Helium | Helium |

*As-built enrichment

Regulatory File Cy.

Received W/Ltr Dated 7-27-76

TABLE 1 (Cont.)

NINE MILE POINT UNIT NO. 1

COMPARISON: INITIAL CORE AND TYPE 2 RELOAD FUEL ASSEMBLY DESIGN (CONT.)

| | | |
|---|---|---|
| **Fuel Bundle** | | |
| Geometry | 7 x 7 | 7 x 7 |
| High Enrichment Rods | 30 | 32 |
| Medium Enrichment Rods | 16 | 10 |
| Medium Low Enrichment Rods | 3 | 6 |
| Low Enrichment Rods | | 1 |
| $Gd_2O_3$ Rods per Bundle | NONE | 4 |
| Rod Pitch, inches | 0.738 | 0.738 |
| Water to Fuel Volume Ratio | 2.38 | 2.43 |
| Heat Transfer Area, $ft^2$ | 87.6 | 86.5 |
| **Spacers** | | |
| Number per bundle | 7 | 7 |
| Material | Stainless Steel with inconel springs | Zircaloy with inconel springs |
| **Channels** | | |
| Material | Zr-4 | Zr-4 |
| Outside Dimension, inches | 5.438 | 5.438 |
| Wall Thickness, inches | 0.080 | 0.080 |
| Channel Length, inches | 162-1/8 | 162-1/8 |

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TABLE 2

NINE MILE POINT UNIT NO. 1

CORE THERMAL HYDRAULIC CHARACTERISTICS

| | |
|---|---|
| Power level, MW(t) | 1850 |
| No. of fuel bundles | 532 |
| Power per bundle, MW(t) | 3.2 |
| Minimum critical heat flux ratio | 1.9 at the above power (Hench-Levy) |
| Average power density, kw/liter | 41 |
| Maximum lineal heat generation rate, kw/ft | 17.5 |
| Peak heat flux, Btu/hr-ft$^2$ | $4.0 \times 10^5$ |
| Average heat flux, Btu/hr-ft$^2$ | $1.31 \times 10^5$ |
| Peak centerline fuel temperature, F | 4250 |
| Peaking factors | |
| Local | 1.30 |
| Axial | 1.57 |
| Radial | 1.50 |
| Gross Product | 3.06 |
| Steam Dome pressure, psig | 1030 |
| Core flow rate, $10^6$ lbs/hr | 67.5 |
| Steam flow rate, $10^6$ lbs/hr | 7.29 |
| Core inlet enthalpy, Btu/lb | 526 |
| Core average void fraction, % | 30.5 |
| Water/$UO_2$ volume ratio[1] | 2.38 |
| Hot channel coolant flow, $10^6$ lbs/hr | 0.112 |
| Hot channel exit void fraction[2] | 0.73 |

[1] Type 2 reload fuel water/$UO_2$ volume ratio is 2.43.

[2] Reload core hot channel exit void fraction will be slightly less.

TABLE 3

NINE MILE POINT UNIT NO. 1

NUCLEAR CHARACTERISTICS OF INITIAL CORE AND RELOADED CORE*

| Core Effective Multiplication and Control System Worth | Initial Core | Reloaded Core |
|---|---|---|
| (0% Voids, 68° F) | | |
| $K_{eff}$ Uncontrolled | 1.13 | 1.15** |
| Δk Poison Curtains | -0.05 | -0.05 |
| Δk Control Rods | -0.16 | -0.16 |
| $K_{eff}$ Fully Controlled | 0.92 | 0.94 |
| $K_{eff}$ Strongest Single Rod Withdrawn | 0.96 | 0.976*** |
| **Reactivity Coefficients** | | |
| Steam Void Coefficient at 1850 MWt, 526 BTU/lb inlet enthalpy, 30.5% in Δk/k% Void | $-11.53 \times 10^{-4}$ | $-11.95 \times 10^{-4}$ |
| Power Coefficient at 1850 MWt, 526 BTU/lb inlet enthalpy Δk/k P/P | -0.06 | -0.06 |
| Fuel Temperature Coefficient Δk/k/°F Fuel | $- 1.2 \times 10^{-5}$ | $- 1.2 \times 10^{-5}$ |

*Analysis for refueling at an initial core exposure of 4750 MWD/t with 56 Type 2 reload bundles loaded on periphery and 48 peripheral curtains removed.

**Includes 0.005 Δk worth of burnable poison in reload bundles.

***At 4000 MWD/t the calculated $K_{eff}$ is 0.980.

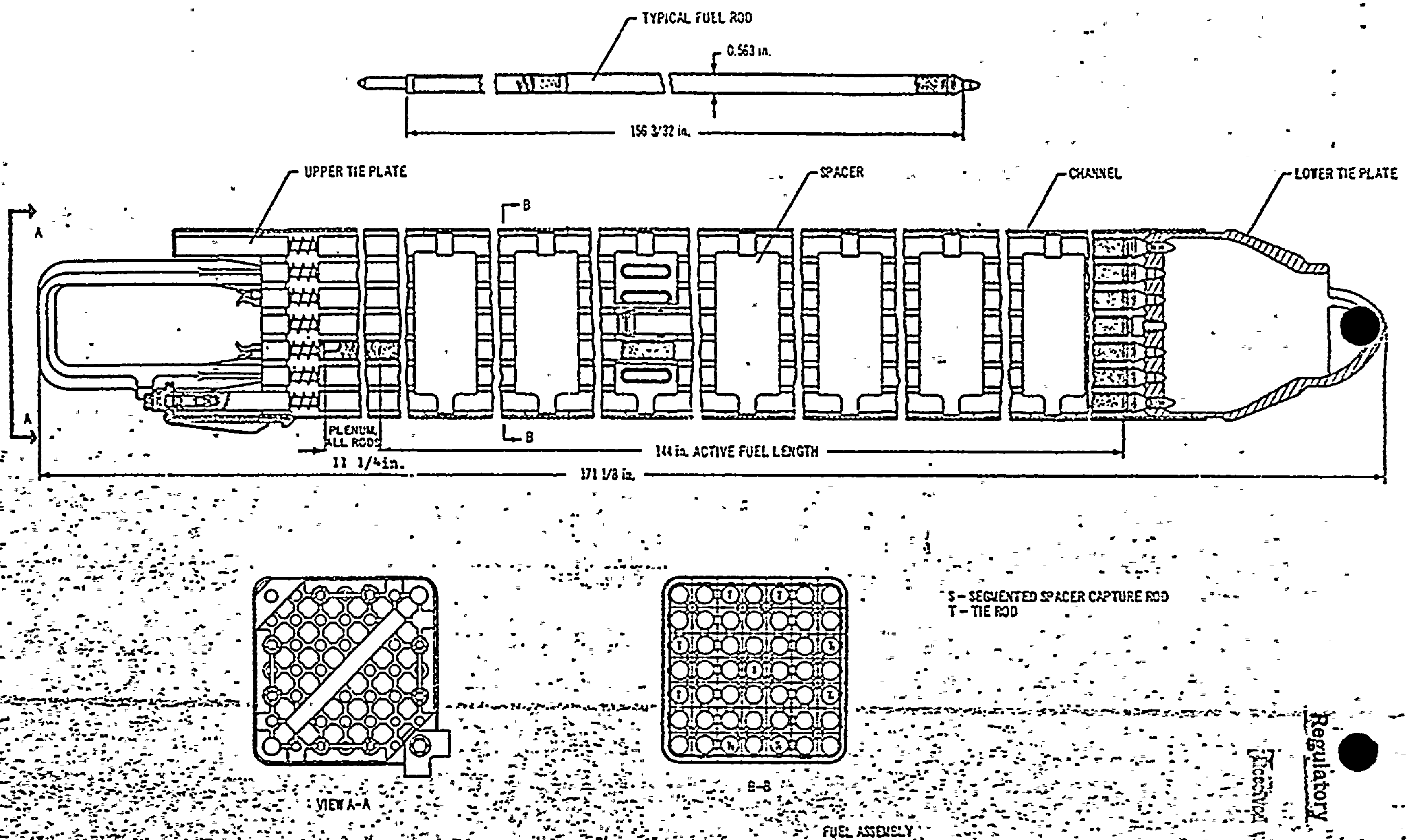


Figure 1
Nine Mile Point Unit No. 1
Type 2 - Reload Fuel Schematic

Wide-Wide Water Gap

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| | | | | | | |
|---|---|---|---|---|---|---|
| D 4 | D 3 | D T 3 | 2 | D T 2 | D 2 | D 3 |
| D 3 | 2 | D 1 | Gd 1 | 1 | D 1 | 2 |
| D 3 | D 1 | 1 | 1 | 1 | 1 | D T 1 |
| 2 | Gd 1 | 1 | 1 S | 1 | Gd 1 | 1 |
| D T 2 | 1 | 1 | 1 | 1 | 1 | D T 1 |
| D 2 | D 1 | 1 | Gd 1 | 1 | 1 | D 1 |
| D 3 | 2 | D T 1 | 1 | D T 1 | D 1 | 2 |

| | |
|---|---|
| 1 | High Enrichment |
| 2 | Medium High Enrichment |
| 3 | Medium Low Enrichment |
| 4 | Low Enrichment |
| D | Dished Rods |
| T | Tie Rods |
| S | Spacer Capture Rod |
| Gd | Gadolinium Containing Rods |

Figure 2
Nine Mile Point Unit No. 1
Reload Fuel Type 2

Figure 3
Nine Mile Point Unit #1
Core Plane



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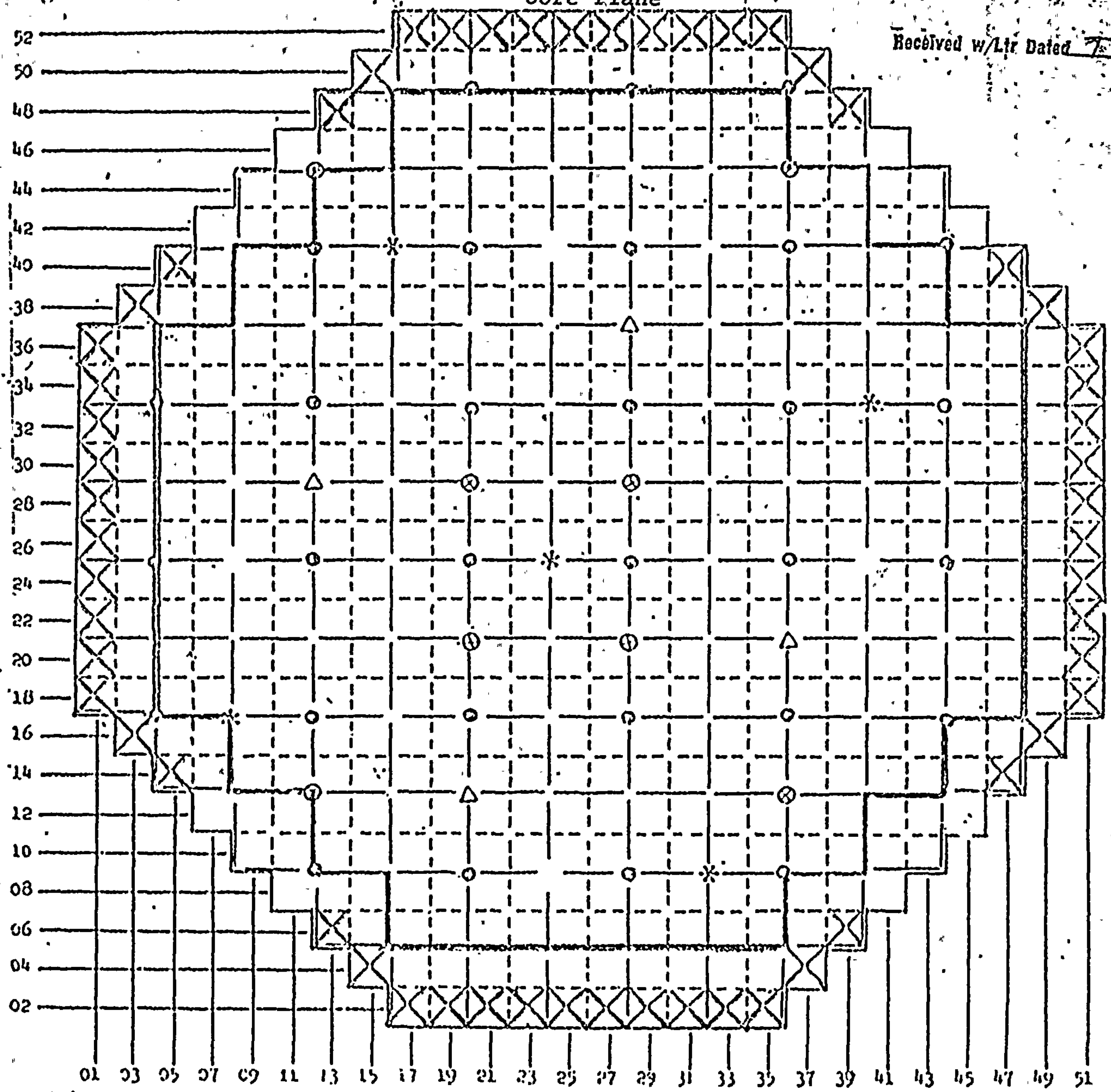
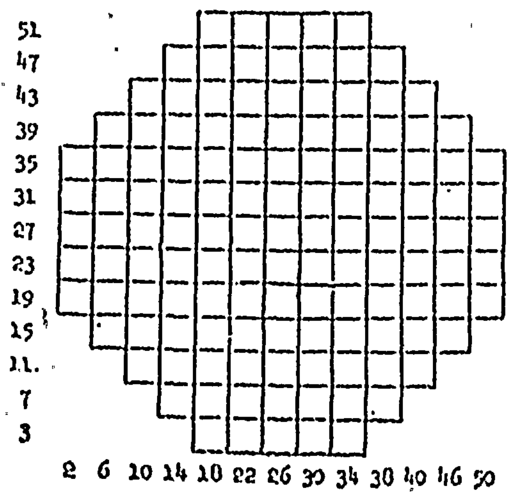



X TYPE 12 - RELOAD FUEL

OUTER ROW OF REMAINING CURTAINS EMPHASIZED BY HEAVY LINE

- ◎ LPRM LOCATION
- ⊗ IRM LOCATION
- △ SRM LOCATION
- ✻ SOURCE LOCATION




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

Figure 4

Nine Mile Point, Unit No. 1

SHUTDOWN MARGIN vs. FUEL EXPOSURE IN SECOND CYCLE