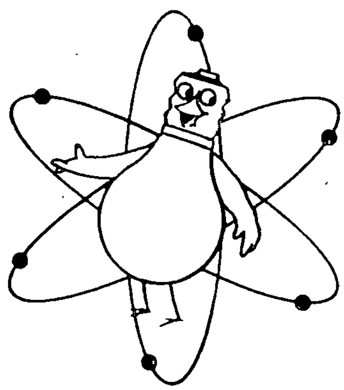
DRESDEN NUCLEAR POWER STATION

UNIT 2

TOPICAL REPORT TO DPR-19

MODIFICATION 71-1

REFUELING



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D2

DRESDEN NUCLEAR POWER STATION UNIT 2

TOPICAL REPORT DPR-19 MODIFICATION 71-1 -- REFUELING

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1.0 INTRODUCTION

Proposed Modification 71-1 to the Dresden Unit 2 Final Safety Analysis Report provides criteria for the use of gadolinia replacement fuel, and an evaluation of the performance of the core that results. In view of the fact that the average core exposure for Dresden 2 at the time of shutdown will be approximately 1200 MWD/STU, this fuel will retain basically the same enrichment as it did initially. Therefore, in reloading the core we will use fuel of the same enrichment as the original loading so that the reactivity differences between the fuels are minimized.

The replacement fuel will use gadolinium for control augmentation in lieu of boron-steel curtains. This is consistent with all General Electric Company refueling of boiling water power reactors since 1965 (Humboldt, Dresden I, Big Rock Point). The first core reactor design for Quad-Cities (to be licensed in the near future) will use gadolinium for control augmentation; boron-steel curtains will not be used.

The specific number of fuel bundles to be replaced will not be known until after refueling outage fuel inspection is completed. This Proposed Modification makes provision for replacement batch sizes up to and including full core replacement (724 assemblies).

2.0 SUMMARY

The replacement fuels for the Dresden 2 reactor are identified as type 1 and type 2 assemblies. They use gadolinium for control augmentation instead of curtains. The type 1 fuel has more control augmentation than the type 2 fuel. The flexibility of using two types of fuel assemblies as replacements provides adequate reactor shutdown margin while maintaining sufficient hot operating reactivity to enable use of control rods to shape the power.

After the failed fuel has been replaced, the reconstituted Dresden-2 core will differ from the original core in several respects. There will be three types of fuel assemblies, two with zero exposure and one with exposure approximating 1200 MWD/ST. The replacement fuel has control augmentation consisting of low concentration Gd_2O_3 in solid solution in UO_2 in a few selected rods per assembly. The exact location of each fuel assembly and type will be determined during the refueling outage. All fuel assemblies are the standard D-2-Quad-Cities design. In the above respects, the planned Dresden 2 refueling is no different from refueling circumstances in other boiling water reactors. Gadolinia augmented refueling has been the subject of past AEC dockets, i.e., Dresden I, Big Rock Point, and Humboldt Bay Unit 3.

This modification differs from normal refueling in these respects:

(1) The exposure of the original fuel is low enough that the associated poison curtains will be retained in the core for the next operating period, and

(2) For a partial loading of gadolinia fuel, replacements may be loaded in the periphery and not scattered.

Accordingly, for purposes of safety review, the information presented shows the compatibility of the replacement fuel with the curtained original fuel in any of several loading patterns.

The replacement fuels have the same assembly average design enrichment (2.12 w/o U-235)* as the first core fuel and can be utilized in batch sizes up to complete core replacement. Type 1 and 2 replacement fuels are described in detail in Amendment 10 to Quad-Cities FSAR AEC Dockets 50-254 and 50-265. The geometry and mechanical description of the type 1 and type 2 replacement fuels are identical to the original core fuel with the exception of the use of gadolinia (Gd_2O_3) for control augmentation in lieu of curtains.

The nuclear design is almost identical to the original core fuel. Three enrichments of similar value with a minor difference in placement are used to control local peaking to within design limits. Since the nuclear characteristics of the replacement fuels are so similar to the core 1 fuel, a great deal of flexibility is available for varying replacement batch sizes and core configurations.

The thermal and reactivity margins which will exist in the replacement core have been evaluated. Due to the nearly equal performance characteristics of the control augmented fuels involved, the thermal and reactivity margins are essentially unchanged. Control augmentation of the replacement fuel is an integral part of the assembly, leading to an improvement in the reactivity margins in fuel storage areas. The reactivity coefficients for gadolinium fuel are essentially the same as those of the original Dresden 2 core.

During the startup phase of Dresden Unit 2, critical control rod patterns were very close to those anticipated. Operating control rod patterns and power distribution have also been in close agreement with prior analysis. Thus, there is considerable verification of the adequacy of analytical methods used on Dresden 2.

* W/O indicates weight percent

3.0 MECHANICAL DESIGN

The geometry and mechanical design of replacement fuels are identical to the original fuel with the exception that gadolinia has been added to the UO_2 in selected locations in each fuel assembly. The gadolinia is uniformly distributed in the UO_2 and forms a solid solution.

The addition of small amounts of gadolinia to UO_2 reduces the melting temperature and also reduces thermal conductivity over a portion of the temperature range of interest. Consequences of these are changes in the linear heat generation rates corresponding to center melting and damage (1% plastic strain) for the gadolinia-bearing rods. These changes for low exposure fuel are:

	UO_2	Gd_2O_3-UO_2	
Onset of Center Melting	22	17.3	kw/ft
Damage (1% Plastic Strain)	28.0	25.2	kw/ft

These values are based on a conservative interpretation of known technological data. Development of the values has been described in Quad-Cities FSAR, Amendment 12, Question 3 of Pages 3 and 4, (AEC Dockets 50-254 and 50-265).

The locations of the gadolinia-bearing rods in the fuel cell assembly are chosen to limit the peak heat flux in these rods to a maximum of 80 percent of the peak heat flux in the pure UO_2 rods of the same assembly. As a result, the linear heat generation in the gadolinia UO_2 rods will not exceed 14 kw/ft. during normal operation. The effects of reduced conductivity and melting point are appropriately offset by the peak heat flux reduction. Any abnormal operating condition which causes a stress or strain limit to be exceeded will produce no more severe conditions in the gadolinia-bearing rods than in the highest powered UO_2 rod in the same assembly.

4.0 THERMAL AND HYDRAULIC CHARACTERISTICS

4.1 Local Limits

Section 3.2.2 of the Dresden 2 FSAR defines the local thermal-hydraulic limits to be applied during reactor operation. These are:

MCHFR $\geq$ 1.90 (APED 5286)

Linear Heat Generation $\leq$ 17.5 kw/ft.

4.2 Thermal Margins

The local power density within the core will be controlled to levels such that the fuel is maintained within these limits. If the gadolinia-bearing rods were to experience local powers greater than 14 kw/ft., it would be necessary to monitor and limit the power of those rods individually. However, by establishing a 17.5 kw/ft. limit for the rods of maximum power, the gadolinia-bearing rods are restricted to levels of lesser concern. This is shown by reviewing Figures 4 and 5 of Quad-Cities FSAR, Amendment 10. For all assembly power levels at which the design basis stress limits (FSAR Section 3.4) are satisfied in the UO_2 rods, the same limits are satisfied in the gadolinia-bearing rods. Also,as concluded from the above date, the assembly power margin to 1% plastic strain is larger in the gadolinia-bearing rods than in the pure UO_2 rods.

In the gadolinia-controlled fuel assembly, the gadolinia-bearing rods operate at only one-third of the peak UO_2 rod power at the start of the first cycle of residence. As conversion of the high cross-section gadolinium isotopes nears completion, the relative power of these rods increases, approaching 80 percent of the peak rod value at the end of the cycle. In subsequent core cycles, the relative power of the gadolinia rods is roughly constant. However, because of its depletion, the assembly cannot operate at limiting power levels; this causes the maximum absolute power values to decrease. Peak UO_2 rods operate below 17.5 kw/ft. and the gadolinium-bearing rods operate below 14.0 kw/ft.

Table 3.2.2 of the FSAR evaluates key thermal-hydraulic parameters at the design basis peaking factors and at other assumed reactor power distributions at rated power (2527 MWth) and rated flow. One comparison of sets of peaking factors is given below:

	FSAR (Table 3.2.2)			Following Fuel Replacement	
	Design Basis	Typical	D2 Operation	Typical	Design Basis
*Local	1.30	1.28	1.28	1.30	1.28
*Axial	1.57	1.53	1.54	1.57	1.54
*Relative Assembly	1.47	1.15	1.21	1.47	1.25

Expected replacement configurations have been evaluated and they have all demonstrated more thermal margin relative to local limits than that provided by the design basis peaking factors; this is indicated by the values in the fifth column in the table above.

The exposure dependent local peaking factors are provided in Table 1 showing the fuels used in the Dresden 2 replacement core. As the table indicates, the maximum local peaking is 1.28, occurs in original D2 fuel, and is less than the design basis peaking of 1.30.

* These have been defined in the FSAR, para. 3.2.8.1.

TABLE 1

EXPOSURE DEPENDENT LOCAL POWER PEAKING

REPLACEMENT CORE FUELS--LATTICE WITHOUT CONTROL RODS

Exposure MWD/STU	Original Core Fuel	Gadolinia Type-1	Gadolinia Type-2	1 Curtain Against Type-2
0	1.22	1.26	1.24	1.27
500	1.26	1.26	1.25	1.25
1000	1.27	1.27	1.26	1.24
1500	1.28	1.27	1.27	1.25
2000	1.28	1.26	1.26	1.25
3000	1.28	1.25	1.25	1.24
5000	1.27	1.20	1.20	1.23
7000	1.26	1.20	1.20	1.20
10,000	1.24	1.18	1.18	1.19
Max. Value First Cycle	1.28	1.27	1.27	1.27

NOTES:
1. Gadolinia TYPE-1 is the reload fuel with greatest burnable poison concentration.
2. At some (interface) reactor locations, the reload TYPE-2 will have an adjacent curtain.
3. The highest local peaking is expected to occur in the original fuel assemblies at an exposure of 2000 MWD/STU
4. Data presented for 10,000 MWD/ST are representative of the end of cycle condition.

4.3 Transient Core Performance

Operational transients at rated power are analyzed in Section 3.2.8.4 of the D-2 FSAR. These transients are evaluated using the design power peaking factors as a basis. This provides the same initial conditions after fuel replacement as before. These transients are analyzed using coefficients which are more conservative then those expected in either the original or the reload core. Thus, the calculated transient performance is the same for both cores.

Dresden 2 operating experience has demonstrated that the actual transients were as predicted. The replacement core and the original core have almost identical fuel temperature and moderator void reactivity coefficients; thus, the replacement core will have the same predicted transient response.

5.0 NUCLEAR CHARACTERISTICS

5.1 Design Bases

The nuclear design bases from Section 3.0 of the D2 FSAR are used for the replacement Dresden 2 core. These are:

- Local peaking ≤ 1.30
- Shutdown K-EFF ≤ 0.99
- Negative void coefficient all reactor states
- Reactivity coefficients consistent with safe and efficient operation

5.2 Nuclear Design Description

The nuclear design of the reload Type 1 and Type 2 fuels is almost identical to the original core fuel. Table 2 provides a comparison of the nuclear properties of the original core fuel and the replacement fuel. Three fuel rod enrichments similar to the original core fuel are used in the replacement fuel to control local power peaking. The enrichment distribution is given in Figure 1. Note that the distribution is almost identical to the original distribution. Due to this distribution similarity, replacement assembly performance will be satisfactory when placed adjacent to a curtain: peak power location is changed and its exposure dependence is affected, but the magnitude is within limits at all exposures.

An interface of curtain and gadolinia fuel will occur in the replacement core; its lattice property (Type-2 reload adjacent to one boron-SS curtain) is included in the table of nuclear properties (Table 2).

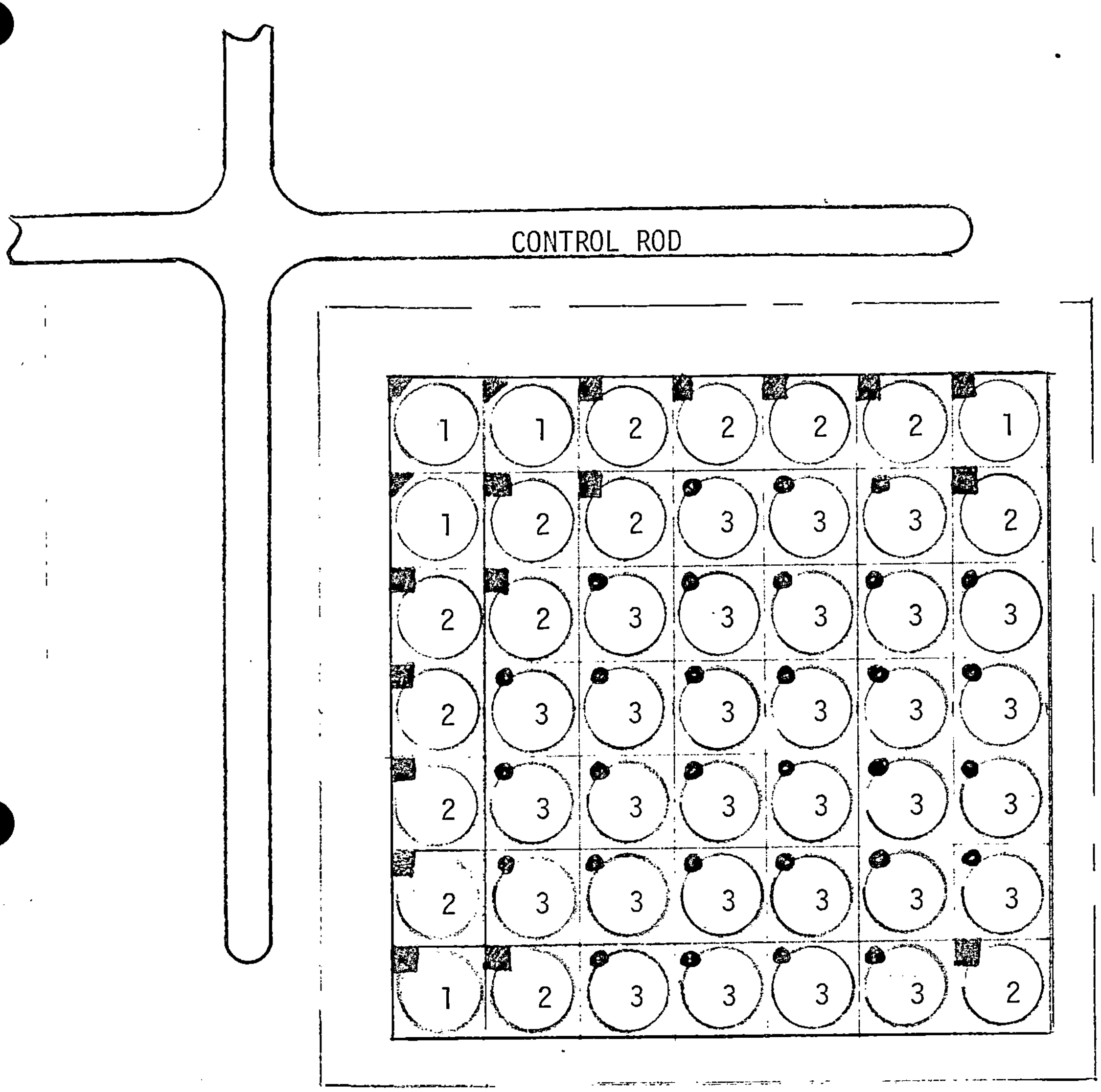
TABLE 2

NUCLEAR PROPERTIES OF FUELS USED IN REPLACEMENT CORE

Property***	Original Core (Curtains)	REPLACEMENT FUELS TYPE-1	TYPE-2	TYPE-2 With 1 Curtain**
Peak local power during cycle	1.28	1.27	1.27	1.27
Assembly average enrichment (w/o)	2.12	2.12	2.12	2.12
Enrichment distribution w/o (number rods)	2.44 (30) 1.69 (16) 1.20 (3)	2.47 (30) 1.70 (14) 1.20 (5)	2.47(30) 1.70(14) 1.20(5)	2.47 (30) 1.70 (14) 1.20 (5)
K_{eff}, single control rod of maximum worth withdrawn Cold, zero exposure	0.975	0.961	0.986	0.942
Operating coefficients				
Operating Moderator void (ΔK/K)/% void	-1.4×10^{-3}	-1.2×10^{-3}	-1.4×10^{-3}	
Operating Fuel temperature (ΔK/K)/°F	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	

**With curtain characteristic of ~~1000~~ 1200 MWD/STU depletion

***Fuel assembly reactivities are at time of reactor replacement. Hot operating reactivities include Xe and Sm.



Replacement Core

1 = 1.20 w/o

2 = 1.70 w/o

3 = 2.47 w/o

ORIGINAL CORE

◤ = 1.20 W/O

■ = 1.69 W/O

● = 2.44 W/O (Remainder)

w/o = weight percent U-235

NOTE: By comparing the two assemblies, note that the two fuel rods in the corners adjacent to the wide-narrow water gap intersections have been shifted from the intermediate enrichment group into the low enrichment category. This was necessary to compensate for changes in the location of the maximum local peaking factor caused by the different location of the supplementary absorber.

FIGURE 1 ENRICHMENT DISTRIBUTION BOTH TYPE-1 AND TYPE-2 RELOAD FUELS 7 X 7 FUEL ASSEMBLY

Analytical methods discussed in the D2 FSAR have been improved to accommodate the performance analysis of highly self-shielding absorbers like the $Gd_2O_3UO_2$ rods. A description of these improvements follows: The nuclear analysis of the first core fuel was performed using the standard design procedures that have been used in all recent BWR designs(1). The gadolinia-bearing fuel rods, however, due to the very large thermal absorption cross section, required special treatment. Few group cross sections for these regions were obtained using a method similar to the THERMOS technique as discussed in reference 1, combined with a standard fast and intermediate treatment. This is the same basic method that was employed in the analysis of the TYPE V reload fuel in Dresden Unit 1. This fuel was subjected to critical assembly tests prior to insertion in the reactor. The number of assemblies in the observed minimum critical array was equal to the number predicted. The calculation of the removal rate of the high cross-section isotopes has been compared to the irradiation tests results. Observation of the operating control rod pattern during full power operation showed the removal of the gadolinia control to be well matched to the fissionable isotope removal.

Data from operating reactors and minimum criticals using gadolinia confirms that nuclear analysis of gadolinia control augmentation is as accurate as the analysis of curtained fuel. Following fuel replacement, the effective rate of depletion can be monitored by observing the operating reactivity status. Thus, any trend toward an unacceptably small shutdown margin due to faster than anticipated absorber removal could be detected and remedial action applied before any unsafe condition could be created. Any tendency toward slower removal rates would affect only cycle length and would be an economic problem unrelated to safety.

5.3 Reactor Shutdown Margin

Acceptable core design requires the reactor to be at least one percent subcritical in its most reactive state and with a single highest worth control blade fully withdrawn. This shutdown capability is evaluated in the cold xenon-free core.

Figure 2 presents the evaluation of shutdown K-effective for any batch size up to full core replacement. The replacement core uses reload configurations of section 6.2 (Figure 3). The envelope in Figure 2 presents the range of shutdown margin resulting from the use of the modules in Figure 3. Dependence on batch size and core location is also included. The shutdown K_{eff} has maximum value of approximately 0.98 and occurs at the beginning

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(1) Aline, P.G., et al, "The Physics of Non-Uniform BWR Lattice," BNES International Conference on the Physics Problems in Thermal Reactor Design, June 1967.

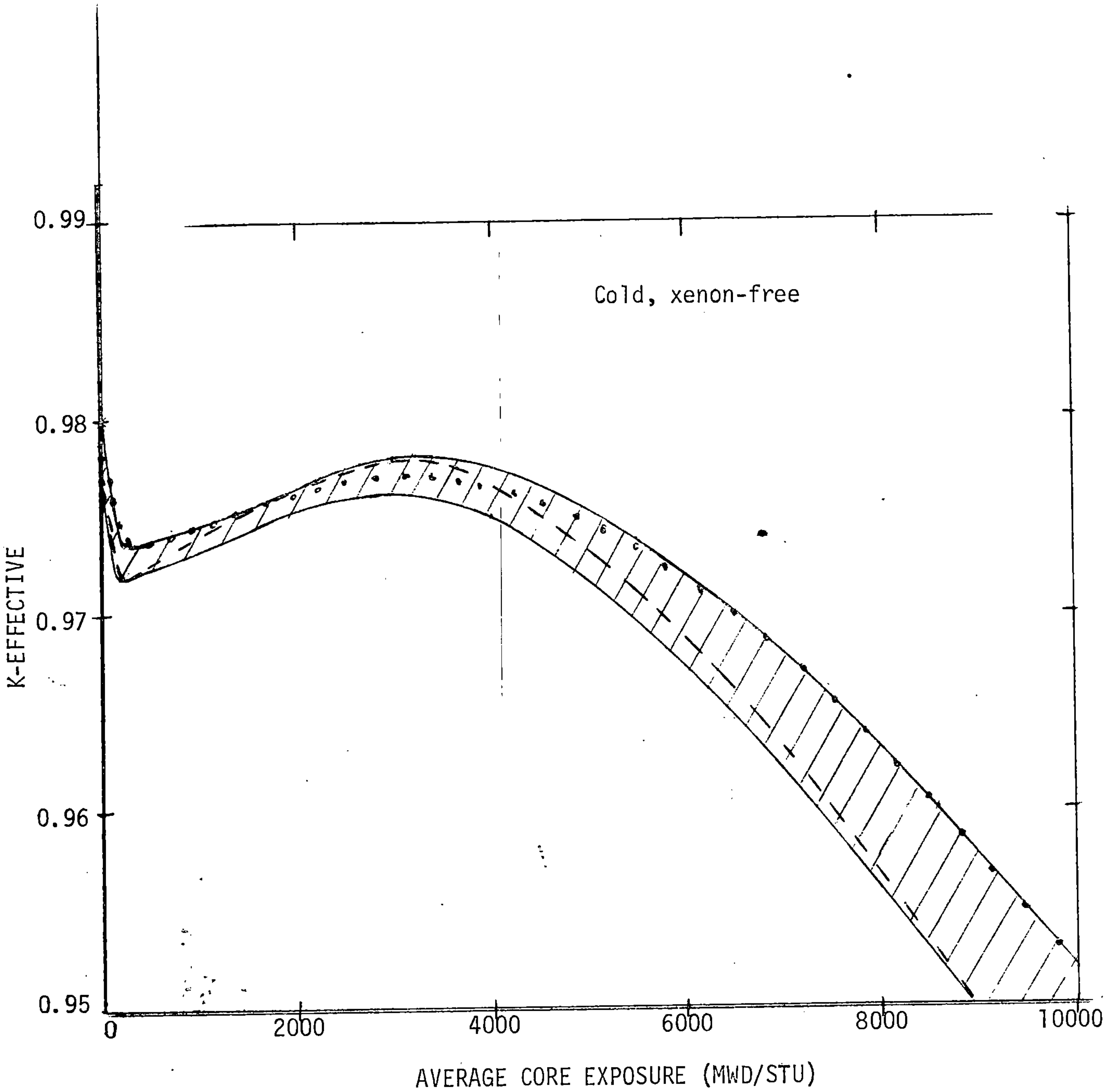
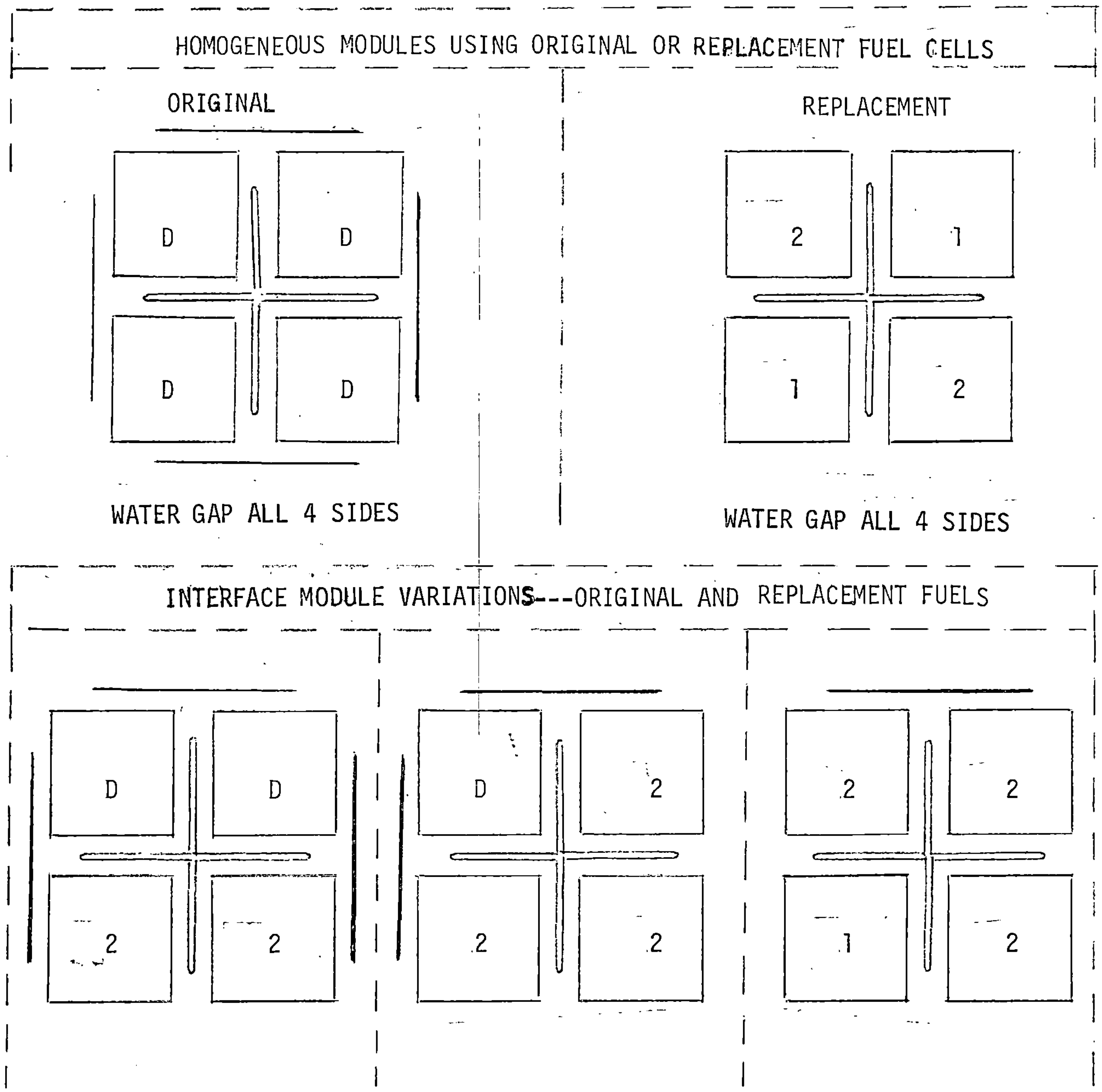


FIGURE 2 SHUTDOWN K-EFFECTIVE. MAXIMUM WORTH ROD FULLY WITHDRAWN. RANGE OF VALUES WHICH RESULTS FROM PARTIAL TO FULL CORE REPLACEMENT.

---- ORIGINAL CORE FUEL. FULL CORE LOADING

.... RELOAD FUEL. FULL CORE LOADING

D = ORIGINAL CORE FUEL
1 = TYPE-1 GADOLINIA REPLACEMENT
2 = TYPE-2 GADOLINIA REPLACEMENT



NOTES: 1. CURTAINS AND D FUEL HAVE 1000 MWD/STU DEPLETION

FIGURE 3 REFUELING MODULES USED FOR ALL CORE CONFIGURATIONS AND REPLACEMENT BATCH SIZES.

of life. This is one percent Δk more margin than the design basis. Thus, it is concluded that if replacement control modules as described in section 6.2 are used, the shutdown margin will be more than adequate, twice as large as is required. Although at some exposures the core may have slightly smaller shutdown margins than the original core, these differences are so small compared to the margin available that they are insignificant.

5.4 Reactivity Coefficients

Values of the void reactivity coefficient for both the replacement and original core fuel are given in Table 2 showing nuclear properties. Since the replacement fuel has the same geometric characteristics as the original core fuel, the coefficients for the two fuels are approximately the same. The Doppler coefficient, dependent primarily on U-238 content and on geometry, remains the same.

As discussed in the D2 FSAR (Section 14.0) and from Dresden 2 reactor operating experience, it is concluded that the coefficients are consistent with safe and efficient reactor operation. Since the plant abnormal operational transients (such as load rejection) depend on the void and Doppler coefficients, an absence of significant changes in these coefficients indicates that there will be no change in core response. The transient results in the FSAR have been determined using conservative values of the void coefficient, values more negative than expected.

6.0 REFUELING

6.1 Design Bases

To ensure that refueling patterns described in this section are consistent with the evaluations and requirements mentioned in earlier sections, the following two design bases for the replacement core have been established:

- Replacement configurations will be consistent with the bases and design limits for thermal-hydraulic and nuclear design (Table 2)

- Replacement configurations will be consistent with safe and efficient reactor operation throughout the fuel cycle.

The remainder of this section will describe the refueling logic which satisfies the above bases for any replacement batch size using TYPE-1 and TYPE-2 gadolinia fuels.

6.2 Control Modules

Interactions between fuel types can be studied by considering the five typical control modules which may be used. A control module consists of the four fuel assemblies surrounding a control blade together with the associated water gaps and curtains. Figure 3 presents arrangement diagrams of the five modules considered. In terms of fuel assembly cells (one assembly with water gap and curtains if used) there are four constituent types as follows:

-- Original core fuel with two curtains (as in D-2 core loading)

-- TYPE-1 gadolinia replacement

-- TYPE-2 gadolinia replacement

-- TYPE-2 gadolinia replacement with one curtain

The properties of the assembly cell types have been discussed either in previous sections of this report or in the FSAR (2-curtain fuel only).

The control modules are of two general classes. They are: (1) homogeneous modules which consist entirely of original core fuel or entirely of replacement fuel; and (2) interface modules which use a combination of original and replacement fuel.

The control modules satisfy the design basis by being consistent with thermal-hydraulic and nuclear design criteria presented in this report. Specifically, they provide satisfactory local power peaking and shutdown margin.

6.3 Replacement Patterns Using Defined Control Modules

The purpose of this section is to illustrate reactor configuration which utilize the control modules. The specific patterns shown illustrate the configurations that can be developed to accommodate replacement fuel batches over the entire range from zero to 724 assemblies employing only the five control modules that have been analyzed. Therefore, these loading patterns, which are quadrant symmetric, can be regarded as a demonstration of the capability to employ any replacement batch size needed. The replacement patterns are:

Figure 4 -- 164 assemblies	Figure 7 -- 336 assemblies
Figure 5 -- 280 assemblies	Figure 8 -- 724 assemblies
Figure 6 -- 364 assemblies	

These refueling patterns satisfy the design bases of 6.1. The planned replacement scheme is a peripheral loading of fresh fuel (Figure 4). However, if larger batch sizes are required, loading which scatters the fresh fuel in the central zone of the reactor may be utilized (Figure 7). In the unlikely event a full core is required, loading will be in accordance with the Quad-Cities loading pattern (Figure 8).

6.4 Loading Errors

The manufacturing quality control applied to the replacement assemblies has been described in Amendment 9 of Quad-Cities FSAR, Page 2.

Four procedural errors might occur: (1) placing fuel assemblies at incorrect reactor locations, (2) curtain omission, (3) curtain misplacement, (4) rotational misorientation of a fuel assembly. The probability of procedural errors is minimized by use of detailed written refueling procedures and by fuel design features. Any significant effect upon reactor shutdown resulting from these loading errors would be identified during cord loading shutdown margin tests. At rated power, fuel assembly and curtain loading errors could result in an additional five percent in power peaking for errors (2) and (3). Consideration of the worst case, a combined fuel assembly rotation error with assembly and curtain positioning errors results in MCHFR$>$1.0 and peak linear heat generation less than 28 kw/ft.

Accordingly, loading errors due both to manufacture and to reactor configuration will not cause fuel to exceed the damage limits defined in Section 3.4.3.4 of the FSAR. Therefore, the design basis, Para. 3.2.1.1 of the D2 FSAR has been satisfied.

7.0 CORE LOADING AND TESTING

The detailed plan for replacement and repositioning of fuel and curtains will be prepared during the shutdown. This plan will be available for inspection by AEC compliance personnel. In addition to listing the sequence of fuel moves, it will contain requirements for inspections to assure that:

1. Curtains are correctly placed adjacent to original Dresden 2 fuel assemblies.

2. Rotational orientation of all fuel assemblies is correct.

3. Each type of fuel assembly is in its proper location.

The required tests following fuel replacement are specified in Para. 7.1. Shutdown margin demonstrations or checks will be performed during and following completion of reloading. Control rods to be checked will be chosen to exemplify use of the five typical control modules of Figure 3 expected to exist in the final loading pattern.

7.1 STARTUP AND POWER TEST PROGRAM

As discussed in the previous sections, the reactor core is expected to demonstrate the same general steady-state and transient performance after replacement as did the original core. The tests specified in f, g, h, and k, below will utilize local instrument response to verify predicted characteristics of the core configuration dominated by either the original core fuel or the replacement fuel.

Two test bases have been established:

1. Run performance tests to verify anticipated integral core steady-state and transient behavior.

2. Run tests to verify in-core instrument performance and calibration.

The following tests will be conducted:

a. Control Rod System tests which monitor scram times and freedom of movement.

b. Intermediate Range Monitor instrument calibration will be performed during appropriate reactor startup states.

c. Shutdown Margin will be demonstrated periodically during fuel loading to assure that the reactor is subcritical with the strongest single control rod withdrawn. The shutdown margin requirement is a limitation on the amount of reactivity which can be loaded into the core. The test has three parts: (a) The analytical determination of the strongest control rod, (b) The calibration of an adjacent control rod, experimentally or analytically, and (c) The demonstration of subcriticality with the strongest rod fully withdrawn and the second at a position equal to the margin. This demonstration will be made for the fully loaded core, and with selected smaller core loadings.

d. Flow Control performance of the replacement core will be observed along the 100 percent power flow control line and compared with expected performance. This test will serve as a major verification of expected integral core behavior in the power range.

e. Pressure Regulator tests will be made to verify anticipated transient integral core behavior. Pressure steps will be introduced to cause a moderator variation. The transient response of the reloaded core to the moderator change will be analyzed.

f. Flux Response to Rods will be determined in both equilibrium and transient conditions. Steady-state noise will be measured as will the flux response to control rod motion. Power-void loop stability will be verified from this data.

g. LPRM Calibrations which include use of the TIP system, will be made at 50, 75 and 100% of rated power. Each local power range monitor will be calibrated to read in terms of local fuel rod surface heat flux.

h. APRM Calibrations will be performed after making significant power level changes. Reactor heat balances will form the bases of these calibrations of the average power range monitors.

i. Core Performance Evaluations will be made periodically to assure that the core is operating within allowable limits on maximum local surface heat flux and Minimum Critical Heat Flux Ratio. This test includes reactor heat balance determinations.

j. Calibration of Rods will be performed to obtain reference relationships between control rod motion and reactor power and steam flow in the specified control rod sequence.

k. Axial Power Distribution measurements will be made with the traversing in-core probe system after planned significant changes in power, control rod pattern, or flow rate. The TIP system will supply data for core performance evaluations and LPRM calibrations.

l. Process Computer functions will be verified as sensed variables come into range during the ascension to and at rated power. The process computer library will be updated to reflect the replacement reactor configuration.

m. Chemical and Radiochemical Tests are conducted to establish water conditions prior to initial operation and to maintain these throughout the test program. Chemical and radiochemical checks are made at primary coolant, offgas exhaust, waste and auxiliary system sample locations. Base or background radioactivity levels are determined at this time for use in fuel assembly failure detection and long range activity buildup studies.

8.0 CONCLUSIONS

The descriptions of the proposed replacement fuel, the replacement procedure, and other analyses that have been presented have demonstrated that safety margins will be preserved and that future plant operation can satisfy the limits of the license. Specifically:

1. Reactivity

Shutdown	-	satisfies limits
Coefficients	-	essentially unchanged

2. Thermal Limits

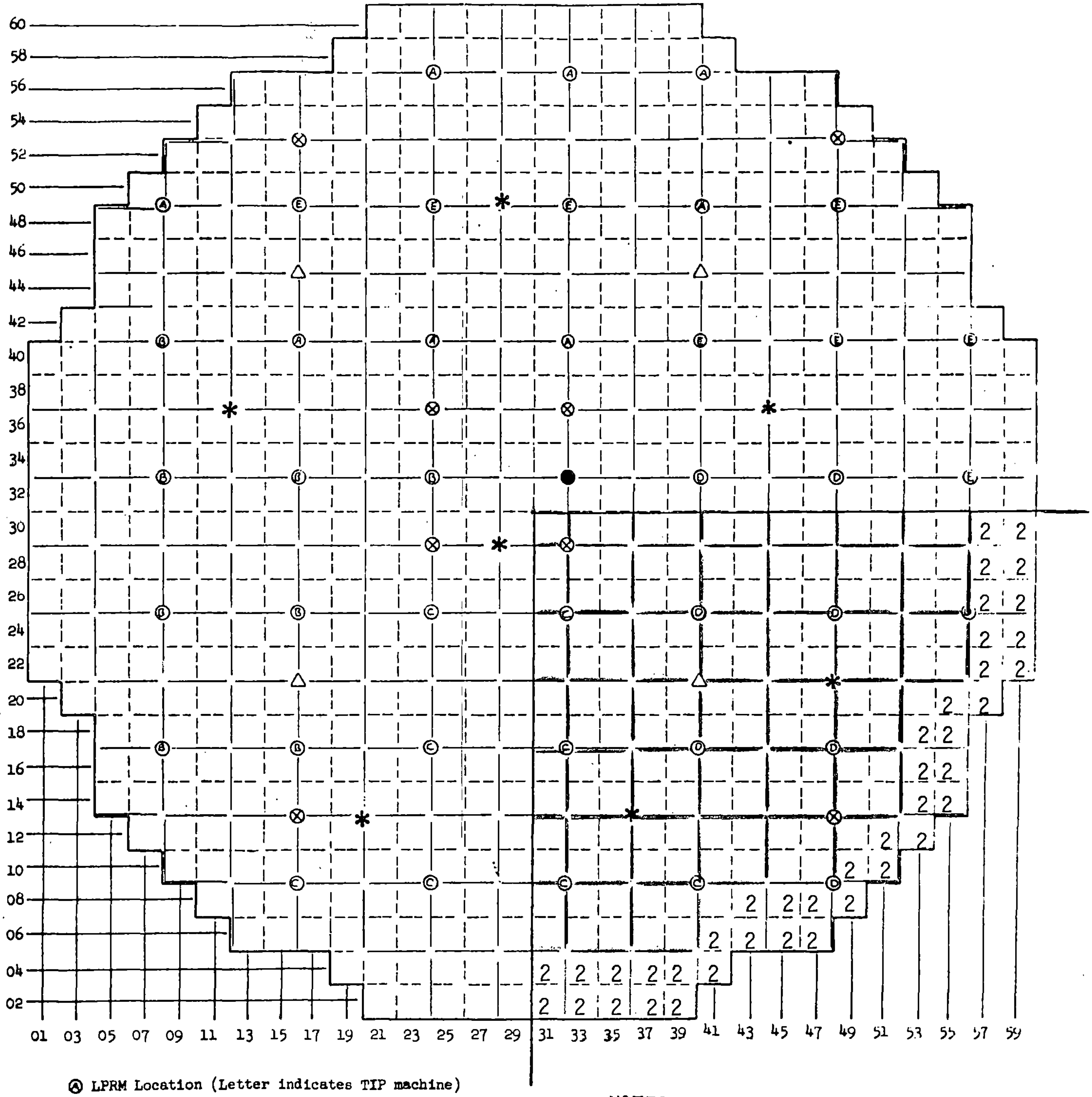
Values	-	identical
Margins	-	greater than design

3. Transient Performance- unaffected

4. Refueling Plan

Control modules and loading configurations	-	presented and analyzed
Test Program	-	described

The planned replacement of Dresden 2 fuel will not alter the safety of the core during the next operating period; moreover, the refueling during subsequent outages will not differ in principle from that employed generally in other General Electric Boiling Water Reactors.



Ⓐ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machines)

⊗ IRM Locations

△ SRM Locations

—— Curtain

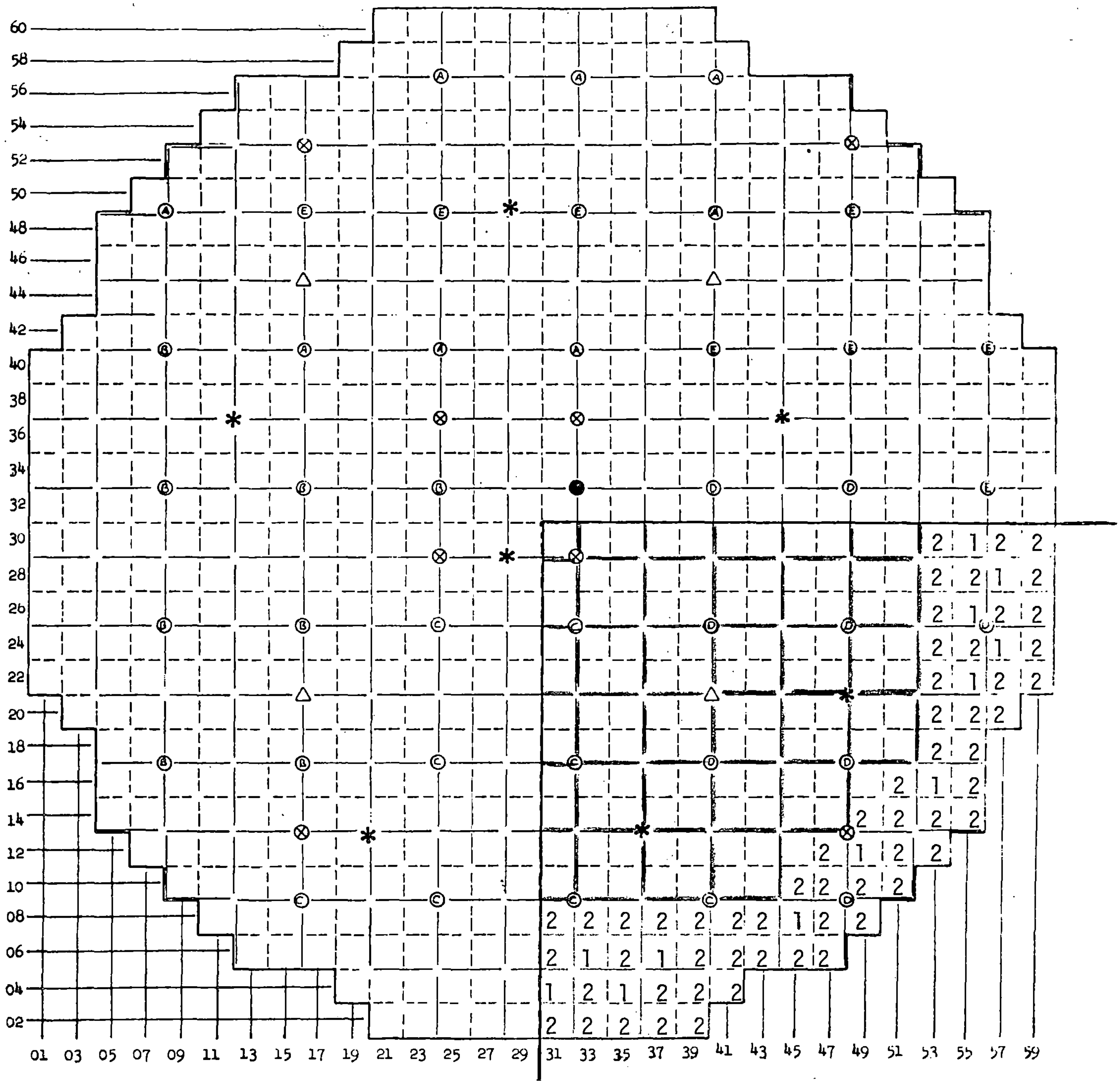
* Source Locations

Control Rod

NOTES:

1. Rotational and mirror symmetry.
2. 2=TYPE-2 Reload
3. 40 Curtains removed

FIGURE 4 REPLACEMENT 164 ASSEMBLIES



Ⓐ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machine

⊗ IRM Locations

△ SRM Locations

—— Curtain

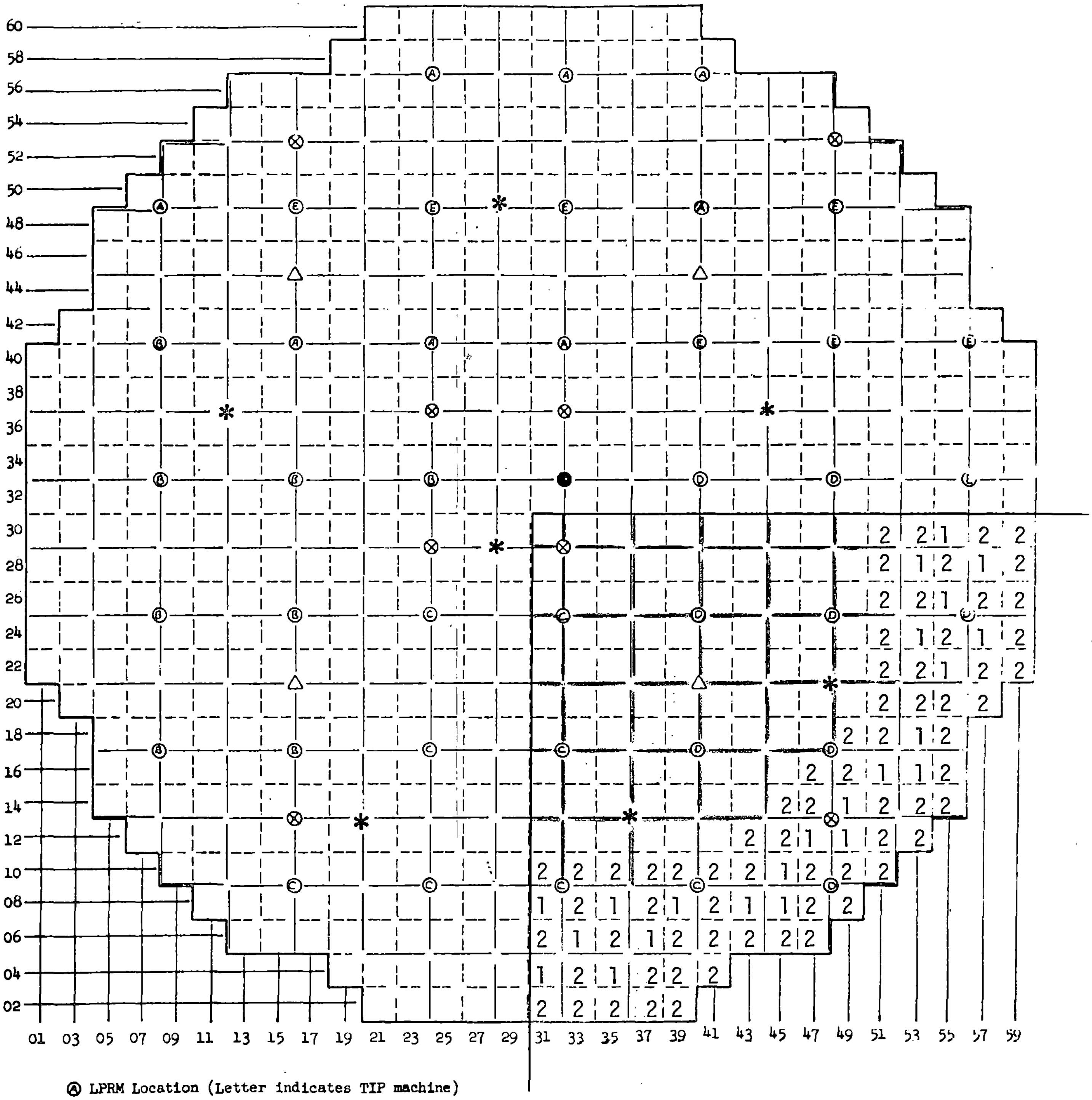
* Source Locations

Control Rod

NOTES:

1. Rotational symmetry
2. 1=TYPE-1 Reload
   2=TYPE-2 Reload
3. 100 Curtains Removed

FIGURE 5 REPLACEMENT 280 ASSEMBLIES



Ⓐ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machines)

⊗ IRM Locations

△ SRM Locations

——— Curtain

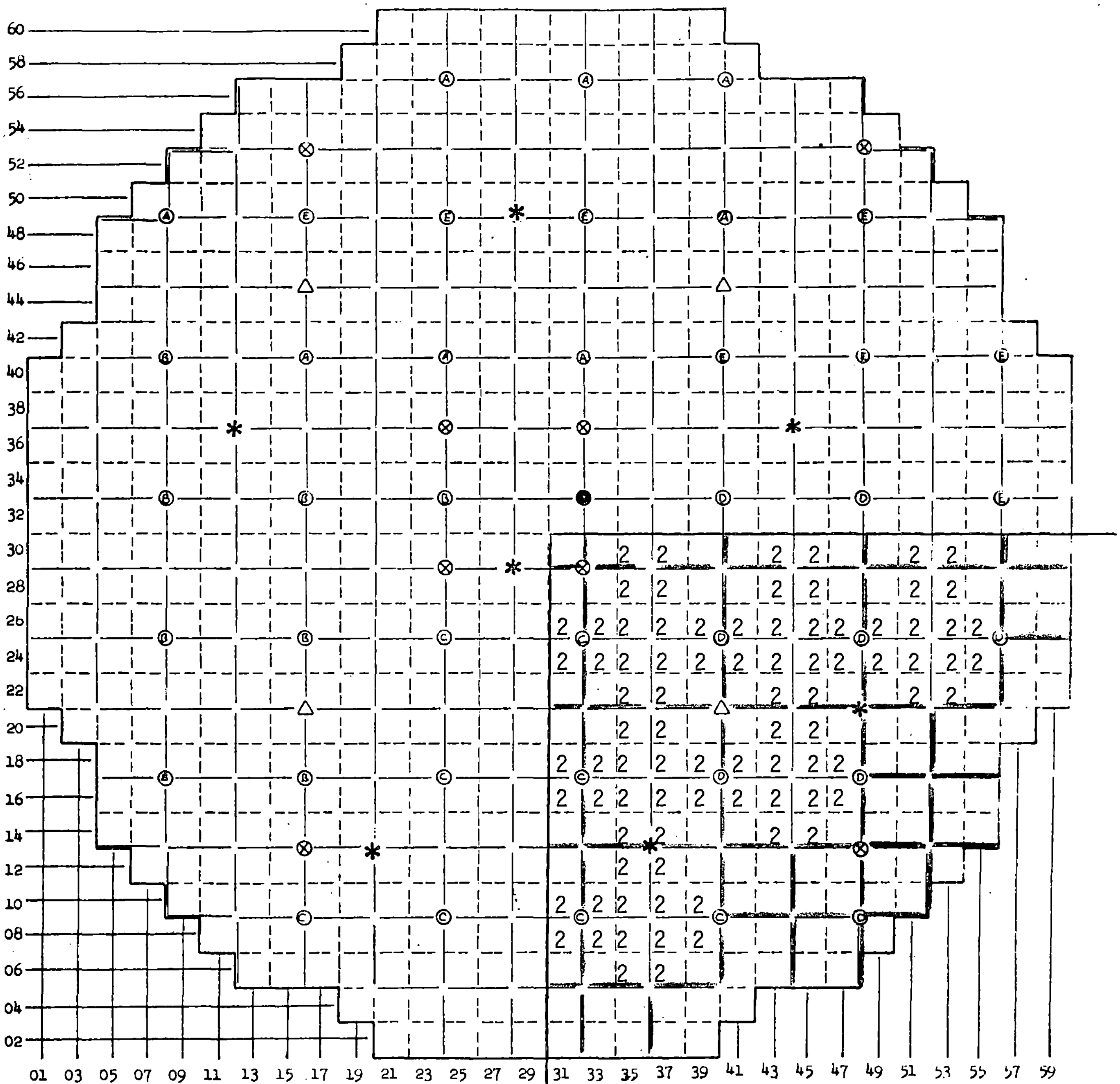
✱ Source Locations

--+-- Control Rod

NOTES:

1. 90° Rotational Symmetry.
2. 1=TYPE-1 Reload
   2=TYPE-2 Reload
3. 144 Curtains Removed

FIGURE 6 REPLACEMENT 364 ASSEMBLIES



Ⓐ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machines)

⊗ IRM Locations

△ SRM Locations

——— Curtain

* Source Locations

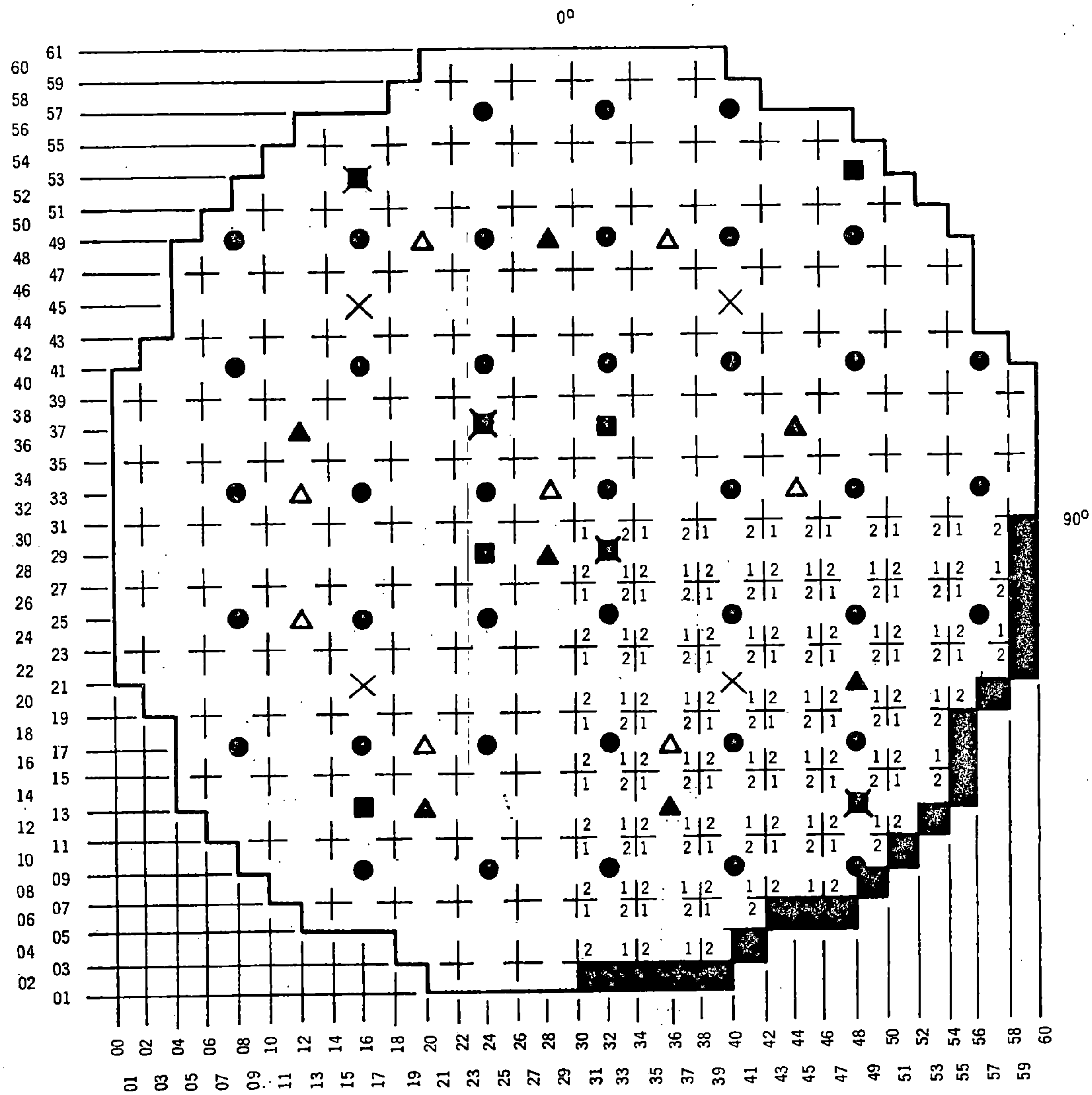
Control Rod

NOTES:

1. 90° Rotational and Mirror Symmetry.
2. 2=TYPE-2 Reload
3. 108 Curtains Removed.

FIGURE 7 REPLACEMENT 336 ASSEMBLIES

TOP PLAN VIEW



- NUMBER OF FUEL ASSEMBLIES – 724
- + NUMBER OF CONTROL RODS – 177
- ● NUMBER OF IN–CORE INSTRUMENT ASSEMBLIES – 41
- ▲ NUMBER OF NEUTRON SOURCES – 7
- △ ALTERNATE NEUTRON SOURCE LOCATIONS – 8
- × STARTUP RANGE INSTRUMENTS – 4
- INTERMEDIATE RANGE – BUS A – 4
- ■ INTERMEDIATE RANGE – BUS B – 4

1 FUEL BUNDLES WITH 3 Gd_2O_3 RODS – 312

2 FUEL BUNDLES WITH 2 Gd_2O_3 RODS – 328

FUEL BUNDLES WITH 2 Gd_2O_3 RODS – 84 AND MORE RESTRICTIVE ORIFICING

NOTE: QUARTER CORE ROTATIONAL AND MIRROR SYMMETRY

FIGURE 8 REPLACEMENT 724 ASSEMBLIES