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Mr. J. F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Dresden Unit 3 Supplement A to Proposed Technical Specification Changes Associated with Second Reload - AEC Dkt 50-249

Dear Mr. O'Leary:

The attached proposed changes to Appendix A to DPR-25 supplement the changes requested by letter to you dated September 14, 1973. These supplementary changes are associated with extending the second reload for Dresden Unit 3 from a total of 60 to 68 new fuel assemblies. Of the 68 assemblies up to 44 will be 8 x 8 assemblies of the type described in the original "Second Reload License Submittal" and up to 24 will be 7 x 7 assemblies of the type described in the attached "Supplement A to Second Reload License Submittal."

The attached supplementary report provides an evaluation of the revised reload program. The attached proposed changes to the Technical Specifications are based on the results of this evaluation. The proposed changes involve reducing the upscale trip level setpoint of the Rod Block Monitor (RBM). This change ensures that MCHFR will remain above 1.0 for the postulated limiting case control rod withdrawal error. This change is necessary as a result of the evaluation of the twenty-four 7 x 7 fuel assemblies.

Pursuant to Section 50.59 of 10 CFR 50 and Paragraph 3.B of the Facility Operating License DPR-25, Commonwealth Edison submits proposed changes to Appendix A of DPR-25. The proposed changes are indicated on the attached revised Pages 24 and 48 from Appendix A of DPR-25. The purpose of these changes and the safety evaluation are included in the attached report, "Dresden Station, Unit 3 Supplement A to Second Reload License Submittal."

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Commonwealth Edison Company

This proposed Technical Specification change has received Onsite and Offsite Review and approval.

Three signed originals and 37 copies of this letter are provided for your use.

Very truly yours,

J. S. Abel
Nuclear Licensing Administrator
Boiling Water Reactors

SUBSCRIBED and SWORN to before me this 27th day of November, 1973.

Brenda Pammer
Notary Public

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias)(7)	$\leq$ 0.650W + 43(2)
* 1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$ 12/125 full scale
2	APRM downscale (7)	$\geq$ 3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	$\leq$ 0.650W + 42 (2)
1	Rod block monitor downscale (7)	$\geq$ 5/125 full scale
3	IRM downscale (3)	$\geq$ 5/125 full scale
3	IRM upscale	$\leq$ 108/125 full scale
* 3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32" of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves; i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCHFR does not decrease to 1.0. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCHFR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCHFR is maintained greater than 1.0

*The APRM rod block which is set at 12% of rated power is functional in the refuel and Startup/Hot Standby mode. This control rod block provides the same type of protection in the refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the Run mode; i.e., it prevents MCHFR from decreasing below 1.0 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked when MCHFR is ~1.16, thus allowing adequate margin. Ref. Section 7.4.5.3 SAR. Below ~70% power the worst case withdrawal of a single control rod results in a MCHFR >1.0 without rod block action, thus below this level it is not required.

SUPPLEMENTAL INFORMATION TO THE DRESDEN 3 SECOND RELOAD LICENSE SUBMITTAL

The Dresden 3 reference loading plan for Cycle 3 specified the insertion of up to 60 8x8 2.50 $\bar{e}$ assemblies into the core.[1] The reference loading plan in this supplement provides for the loading of up to 68 assemblies consisting of 44 8x8 2.50 $\bar{e}$ assemblies (R2 Type A), and 24 7x7 2.30 $\bar{e}$ assemblies (R2 Type B). The 8x8 fuel is as previously specified[1], and the 7x7 fuel is identical in design to the previously loaded Reload-1 fuel bundle.[2] The new design reference core configuration consists of the bundles defined in Table 1 . The relative locations of the 68 assemblies are shown in Figure 1 .

FUEL ASSEMBLY THERMAL-HYDRAULIC EVALUATION

The results of the thermal-hydraulic analyses previously reported[1] applies to the reference core loading specified in this supplement. Table 4-1 of Reference 1 provides the thermal-hydraulic characteristics of both the Reload-2 Type B fuel assemblies (as represented by the 7x7 fuel), as well as the full range of possible loadings of 8x8 fuel.

NUCLEAR CHARACTERISTICS OF THE CORE

The nuclear characteristics of the core reconstituted with 68 Reload 2 assemblies have been recalculated, and have been found to be as previously reported.[1] This is due to the relatively small change in the original loading plan; sixteen 8x8 assemblies replaced with less reactive 7x7 assemblies, and 8 initial core bundles replaced with 7x7 reload assemblies out of a total of 724 bundles in the core.

TABLE 1

FUEL TYPE AND NUMBER

FUEL TYPE	NUMBER
Initial	604
Reload 1	52
Reload 2 Type A	44
Reload 2 Type B	24
TOTAL	724

TABLE 2

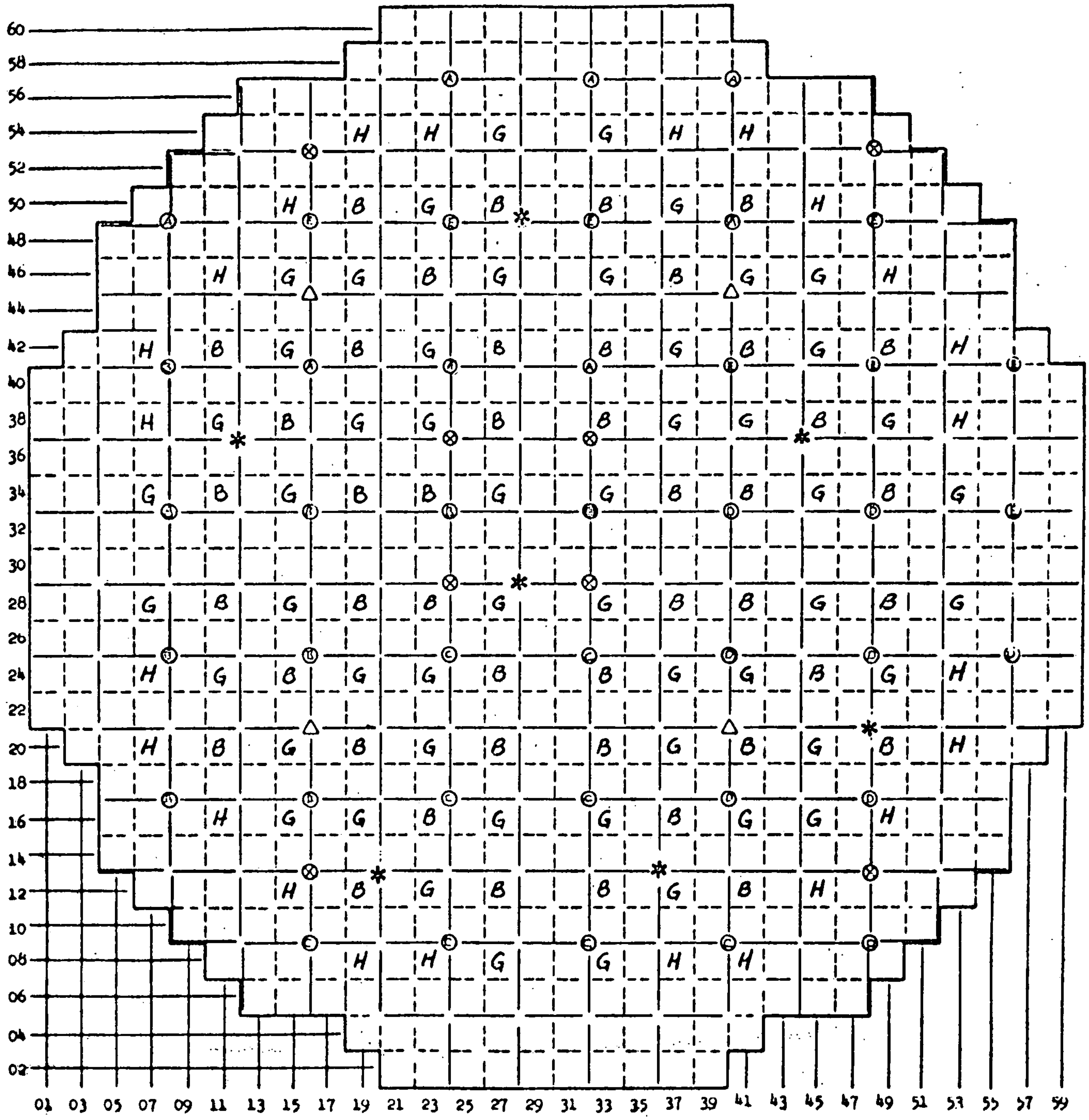
LOCA PCT

	MAX PCT °F
AEC Criteria	2300
Initial Core	2015
(R1 and R2 Type B)	1960
(R Type A)	1920

Since the previous analyses already considered a mixed core, and since the nuclear inputs to the analyses are unchanged, the results of transient analyses and core dynamics, as previously reported, apply.[1]

Results of the Rod Withdrawal Error (RWE) analyses have changed from the original Reload 2 license submittal due to the insertion of Reload 2 Type B fuel. The results indicate that the Rod Block Monitor (RBM) should be re-set to a lower value to ensure that MCHFR will be above 1.0 with adequate margin in the postulated event of a RWE.

FIGURE 1 DRESDEN 3 CYCLE 3 DESIGN REFERENCE CORE LOADING
(68 New Bundles)



Ⓐ LPRM Location (Letter indicates TIP machine)

Ⓔ LPRM Location (Common location for all TIP machines)

⊗ IRM Locations

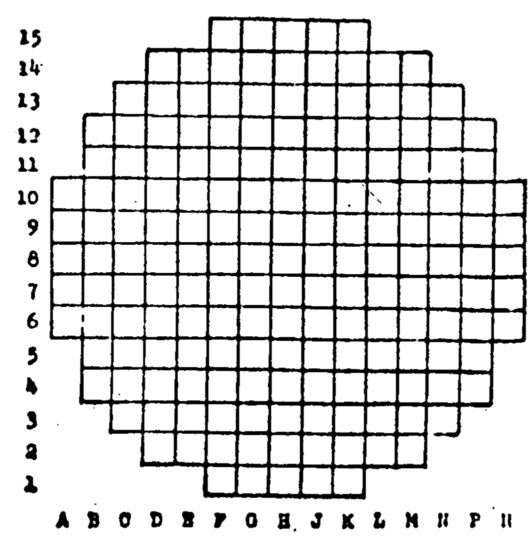
△ SRM Locations

* Source Locations

G - Reload 1 Bundle (7x7, 2.3$\overline{e}$)-52

B - Reload 2 Type A Bundle (8x8, 2.5$\overline{e}$) -44

H - Reload 2 Type B Bundle (7x7,2.3$\overline{e}$) -24



TECHNICAL SPECIFICATION CHANGES

The following changes are required in the Technical Specifications as a result of the change in reference core loaidng.

ITEM	LOCATION	CHANGE	REASON
Table 3.2.3	Page 42 Rod Block Monitor Upscale -	Change trip level setting from 0.650W + 45[2] to 0.650W + 44[2]	This change is the result of the new re-analysis
Basis statement for 3.2	Next to last sentence on pg 48	Change to read "..... MCHFR is ~1.16...."	This is a result of the new evaluation

REFERENCES

(1) Dresden 3 Nuclear Power Station Second Reload License Submittal, September 1973.

(2) Dresden 3 Nuclear Power Station First Reload License Submittal, April 1973.

SAFETY ANALYSIS

The change in the reference core loading to include 7x7 fuel does not change the results of the previously presented safety analyses for Reload 2[1] except for the rod withdrawal error. Since the Reload 2 Type B fuel is the same as previously inserted Reload 1, the safety analyses for that reload are applicable to the Type B fuel for this reload with the same exception as above.

The fuel damage limits, operating limits, and operating margins are the same as previously discussed for the individual fuels.

The accident analyses, including the main steam-line break, refueling, control rod drop, and loss of coolant accidents have not changed from that previously presented in the respective Reload 1 and 2 license submittals.

As previously stated, the change in reference core loading pattern does not change the nuclear inputs to accident analyses; thus, there is no change in the rod drop accident analysis, and the previously reported results apply.[1]

The principle parameters affecting the results of the loss of coolant accident analyses are the distribution at local peaking factors, and the changes in peaking factors with exposure. The Reload 2 Type A fuel remains unchanged[1], and the Reload 2 Type B fuel is the same fuel that was previously reviewed for Reload 1. Thus, the analysis presented in the Reload 1 license submittal is applicable to the Reload 2 Type B fuel. For comparative purposes, the results of these analyses are shown in Table 2.