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Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. R. A. Purple, Chief
Operating Reactors Branch #1

Gentlemen:

REF: Docket 50-305
Operating License DPR-43
Letter from Mr. K. R. Goller, NRC to
Wisconsin Public Service dated July 1, 1975

Attached are ten copies of a safety evaluation for the proposed cycle 2 core of the Kewaunee Power Plant. This safety evaluation, performed by Westinghouse Electric Corporation and reviewed by our engineering staff, verifies that the proposed cycle 2 core will not adversely affect the safety features and margin designed into the Kewaunee Plant. This evaluation demonstrates that the proposed cycle 2 core characteristics are either enveloped by previous analyses or the acceptance criteria of previous analyses are not exceeded by assuming the characteristics of the proposed cycle 2 core as analysis assumptions.

Since the operation of the proposed cycle 2 core is either demonstrated to be within the envelope of acceptance criteria for analyses previously submitted, or does not increase the probability of accident, or does not reduce the margin of safety as defined in the Technical Specifications basis; the change in core loading of the Kewaunee Plant to the proposed cycle 2 core does not result in an unreviewed safety question as specified in Title 10 CFR Part 50, Section 50.59(a). The attached evaluations are being transmitted to document this position.

Very truly yours,

E. W. James, Senior Vice President
Power Supply & Engineering

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Attach.

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Repro 2/1

RELOAD SAFETY EVALUATION

KEWAUNEE NUCLEAR PLANT

CYCLE 2

Rec'd w/ltr dtd
2-3-76

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LIST OF FIGURES

FIGURE	TITLE	PAGE
1	Loading Pattern	18
2	Hot Channel Factor Operating Envelope	19

LIST OF TABLES

TABLE	TITLE	PAGE
1	Fuel Assembly Design Parameters	13
2	Kinetics Characteristics	14
3	Shutdown Requirements and Margins	15
4	Rod Ejection Parameters	16
5	Results of Rod Ejection Analysis Hot Spot Fuel and Clad Temperatures	17

TABLE OF CONTENTS

	TITLE	PAGE
1.0	INTRODUCTION AND SUMMARY	1
2.0	REACTOR DESIGN	2
	2.1 Mechanical Design	2
	2.2 Nuclear Design	2
	2.3 Thermal and Hydraulic Design	3
3.0	ACCIDENT EVALUATION	4
	3.1 Power Capability	4
	3.2 Accident Evaluation	5
	3.3 Incidents Reanalyzed	7
4.0	TECHNICAL SPECIFICATIONS	9
5.0	TESTING OF CYCLE 2 CORE	10
6.0	REFERENCES	12

1.0 INTRODUCTION AND SUMMARY

The Kewaunee Nuclear Power Plant is in its first cycle of operation. The unit will be refueled and ready for Cycle 2 startup in Mid-April 1976.

The proposed core loading pattern for Cycle 2 is shown as Figure 1. Forty of the Region 1 assemblies will be discharged and forty Region 4 fuel assemblies (see Table 1) added to form the proposed Cycle 2 core.

This report presents an evaluation for Cycle 2 which demonstrates that the core reload will not adversely affect the safety of the plant and that no unreviewed safety questions exist in regards to the refueling. It is not the purpose of this report to present a reanalysis of all potential incidents. Those incidents analyzed and reported in the FSAR which could potentially be affected by fuel reload have been reviewed for the Cycle 2 design are described herein. The results of new analyses have been included and the justification for the applicability of previous results for other analyses are presented. It has been concluded that the Cycle 2 design does not cause the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the assumptions that:

1. Cycle 1 operation is terminated after 16,800 $\pm$ 1000 MWD/MTU, and
2. There is adherence to plant operating limitations presently in the Technical Specifications.

The present evaluation did not explicitly take into account the effects of generic issues currently under consideration by the NRC. The rod bow issue, in particular, is under generic discussion with the NRC and the effects of row bow on DNB or LOCA related accidents has been generically addressed by Westinghouse in WCAP 8691.

Nominal design parameters for Cycle 2 are 1650 MWt core power, 2250 psia system pressure, 535.5°F core inlet temperature, and 6.2 kw/ft average linear fuel power density (based on 144" active fuel length)

2.0 REACTOR DESIGN

2.1 Mechanical Design

The mechanical design of Region 4 fuel is dimensionally similar to Regions 1, 2, and 3 fuel. Region 4 fuel has a different enrichment and pellet density as noted in Table 1.

Clad flattening time is predicted to be 20,400 EFPH for the limiting region (Region 1) using the current Westinghouse evaluation model[4]. Therefore, the single Region 1 assembly has a nominal Cycle 2 allowed residence time of 8765 EFPH (assumes a Cycle 1 lifetime of 11,635 EFPH). The predicted lifetime of Cycle 2 is 7840 EFPH; therefore, clad flatting should not occur during Cycle 2.

The fuel vendor, Westinghouse, has had considerable experience with Zircaloy-clad fuel. This experience is described in Reference 6.

2.2 Nuclear Design

The ECCS analysis per the March 15, 1975 version of the Westinghouse ECCS Evaluation Model results in a maximum $F_Q \times P_{Rel} \leq 2.15$. The specific axial maximum $F_Q \times P_{Rel}$ is presented in Figure 2 which was included in proposed amendment no. 3 to the Operating License dated September 4, 1974. An analysis was performed for the proposed cycle 2 core to verify that operation will be within this $F_Q \times P_{Rel}$ envelope

This analysis, transmitted to the NRC on January 22, 1975, indicated that the predicted axial maximum values of $F_Q \times P_{Rel}$ do not exceed the ECCS assumed values.

Table 2 provides a comparison of proposed cycle 2 core kinetics characteristics with the cycle 1 core kinetic characteristics or assumed values for accident calculations presented in the FSAR accident analyses. All cycle 2 core kinetic parameters, with the exception of a 5% variation in prompt neutron lifetime, are within the range of the kinetic parameters assumed in the FSAR. Section 3.0 addresses the results of specific analysis performed assuming the cycle 2 core kinetic characteristics.

Table 3 provides a comparison of total rod worth and shutdown requirements for cores of cycles 1 and 2. The beginning of cycle and end of cycle values are provided to assist in the comparison. The shutdown margin specified on Table 3 are the margins assumed in the accident analyses provided in the FSAR and required by the Technical Specifications.[2] The shutdown margin available in the proposed cycle 2 core exceeds the minimum requirements of the analyses.

2.3 Thermal and Hydraulic Design

The mechanical design of the region 4 fuel is, as stated above, similar to the regions comprising the cycle 1 core. The region 4 fuel assemblies are of the same general two loop 14 X 14 fuel element design described in Section 3.2.3 of the Kewaunee FSAR. The Reactor Coolant System hydraulic design will not be altered from the design employed in cycle 1.

The power distribution limits of cycle 2 are the same as cycle 1.

The existing DNB limits are conservative for the cycle 2 core. The loading of the proposed cycle 2 core will not result in a significant variation in the thermal-hydraulic design margins included in the first cycle.

3.0 ACCIDENT EVALUATION

3.1 Power Capability

The following addresses the capability of the Kewaunee Plant to operate at 100% power during cycle 2 considering the consequences and associated design basis of accidents addressed in the FSAR.

The ECCS performance was evaluated for the Kewaunee Plant per the March 15, 1975 version of the Westinghouse ECCS evaluation model. This ECCS performance evaluation is included in the FSAR as revision number 28 submitted December 29, 1975. The ECCS analysis assumed an $F_Q \times P_{Rel}$ operating limit as presented on Figure 2. The LOCTA analysis was performed for an assumed power level of 102% of licensed rating (1650 Mwth). The SATAN analysis was performed for an assumed power level of 102% of the engineered safeguards design rating (1721.7 Mwth).

An F_Q envelope analysis was performed for the proposed cycle 2 core. This F_Q envelope analysis verified the operability of cycle 2 within the ECCS evaluation assumed envelope. The F_Q envelope analysis was transmitted to the NRC on January 22, 1975.

The operation of the cycle 2 core within the conditions specified in the proposed amendment number 13 to the Technical Specifications will result in adherence to the assumptions of the ECCS analysis.

The overpower transient maximum local rod power of 21.0 kw/ft was assumed as the limit in the facility safety evaluation.[1] The ECCS analysis limits the maximum local rod power to 13.53 kw/ft. Since cycle 2 satisfies the ECCS requirements of 13.53 kw/ft, the overpower transient limit is also satisfied with significant margin.

The DNB safety analysis assumed, an F^N_H of 1.55 including an 8% uncertainty, a referenced cosine shape with an axial peak to average ratio of 1.55, penalties for pellet eccentricity, and a local power spike penalty. The $F^N_{\Delta H}$ assumption is assured by the existing Technical Specifications and their associated limiting conditions for operation. The basic assumptions of the DNB analysis are satisfied by the cycle 2 core.

3.2 Accident Evaluation

Each of the postulated core related incidents analyzed in the FSAR have been evaluated to determine the effect of the proposed reload core upon the analysis assumptions and results.[2] In the majority of the incident analyses examined, the reload core results in a more conservative result, due to additional conservatism in the analysis assumptions. For certain incidents, the conservatism of the initial assumptions of the analysis accommodated the effects of the proposed

core. Reanalysis was required for incidents where the analysis assumptions were invalid and/or nonconservative for the proposed cycle 2 core.

The cases reanalyzed were compared to the design bases to determine if the FSAR conclusions were valid. If the FSAR conclusions remained valid, then no unreviewed safety questions exist. Prior to discussion of specific incident analyses performed by cycle 2, a comparison of the proposed cycle 2 core physics characteristics is necessary. Core loading patterns, point-wise isotopic concentrations, fuel enrichments, inclusion of burnable poison assemblies, etc., are altered during a reload core assembly. The alteration of these parameters affect the core kinetic characteristics, control rod worths, power distribution, etc. Table 1 provides general fuel assembly design information for the proposed cycle 2 core.

Table 2 includes the range of kinetic characteristics employed in analyses previously submitted for the Kewaunee facility and the range of characteristics for the proposed cycle 2 core. The kinetic characteristic range of previously submitted analyses envelopes the range of kinetic characteristics of the proposed core.

The reactivity worth of control rods and thereby shutdown margin may be affected by the redistribution of fuel in the core assembly. Table 3 presents the predicted control rod reactivity worth for proposed cycle 2 versus the predicted worth of the cycle 1 rods. Also included are comparisons of cycle related reactivity

requirements and incident analysis assumed shutdown margins. It can be seen that adequate shutdown margin is provided in the proposed Cycle 2 core calculated rod worths. (The beginning of life Cycle 1 rod worths as measured were within +5% of the predicted worth.)

3.3 Incidents Reanalyzed

The rod ejection accident analysis previously submitted assumed a larger B_{eff} value for the beginning of life case. The end of life full power case incorporated a maximum steady state hot spot fuel temperature lower than the appropriate assumed values for the proposed Cycle 2 core. Rather than employ values only appropriate for the proposed Cycle 2 core, a conservative set of assumptions were selected to envelope the proposed Cycle 2 core and subsequent cycle cores. The parameters considered in the reanalysis along with the proposed Cycle 2 parameters and previously employed parameters are presented on Table 4. The "re-analysis assumed values" noted on Table 4 are significantly conservative in comparison to either of the other sets.

The reanalysis was performed in accordance with the Method of Reference 3. Additional description of the analysis is presented in Section 14.2.6 of the FSAR.

Table 5 includes the results of the reanalysis in accordance with the method described in WCAP 7588. These results are within the Limiting Criteria of the WCAP 7588 and the Kewaunee Plant FSAR specified criteria. The conclusions presented in the FSAR are not altered by the results of this analysis since:

1. The maximum fuel pellet center temperature is below 5080°F for unirradiated fuel and 4890°F for high burn-up fuel.
2. The maximum average clad temperature is below the 2700°F clad embrittlement value.
3. The maximum fuel enthalpy is below the 225 cal/gm for non-irradiated fuel and 200 cal/gm for irradiated fuel.

Fission product release would be within the specifications of 10 CFR 100 as in the case of the previous analysis.

4.0 Technical Specifications

The evaluation of the Cycle 2 core assumed limits established for Cycle 1 of the Kewaunee facility as specified by the Technical Specifications or as included in the ECCS related request for changes to the Technical Specifications. The ECCS related changes to the Technical Specifications were transmitted to the Commission as proposed Amendments No. 3 and 13, dated September 4, 1974 and December 28, 1975 respectively.

No additional changes to the Technical Specifications are required as a result of the reload.

5.0 TESTING OF THE CYCLE 2 CORE

The calculated physics parameters of the reload core were employed in the safety evaluation and provide the basis for justifying the applicability of the analysis. To verify the validity of these bases a physic verification test program will be performed on the cycle 2 core. The following measurements are intended:

1. The critical conditions for the cycle 2 core will be determined - All rods out boron endpoint measurement.
2. An all rods out (D bank > 210 steps out) flux map will be performed.
3. The isothermal temperature coefficient will be determined for the all rods out configuration.
4. The Bank worths of control banks D and C will be measured.
5. An at power flux map will be performed prior to exceeding 75% power as required by the Technical Specifications.
6. The power range detectors, overpower ΔT, and overtemperature ΔT channels shall be calibrated for individual detector sensitivity and correlation to incore power distribution prior to escalation above 75% power. These final protection system calibrations are performed at 75% to minimize entrapulation errors.
7. A flux map will be compared to predicted values prior to exceeding 90% and a comparison will be made at 100%.
8. Other tests required by the Technical Specifications, i.e., rod drops, rod position calibrations, etc.

Acceptance Criteria

The following lists the specific acceptance criteria which if exceeded will require an internal safety review and/or re-analysis of the parameter

prior to proceeding with power escalation:

1. Boron endpoint all rods out $\pm$ 50 ppm.
2. Rod worth all control banks $\pm$ 10%.
3. Isothermal temperature coefficient $\pm$ 3 pcm/^{o}F.
4. Assembly to average power distribution $\pm$7%.

6.0 REFERENCES

1. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation", WCAP 8218-P-A, March 1975 (Proprietary) and WCAP 8219-A, March 1975 (Non-Proprietary).
2. Final Safety Analysis Report - Kewaunee Nuclear Power Plant.
3. Risher, D. H. Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP 7588, Revision 1, December 1971.
4. George, R. A., et. al., "Revised Clad Flattening Model", WCAP 8377 (Proprietary) and SCAP 8381 (Non-Proprietary), July 1974.
5. "Fuel Densification, Kewaunee Nuclear Power Plant", WCAP 8377 (Proprietary) and WCAP 8093 (Non-Proprietary), March 1973.
6. Plocido, V. J., and Schreiber, R. E., "Operational Experience with Westinghouse Cores", WCAP 8183, Revision 3, March 1975.

Table 1

Kewaunee Nuclear Power Plant - Cycle 2

Fuel Assembly Design Parameters

Region	1	2	3	4
Enrichment (w/o U-235)	2.26	3.04	3.41	3.30
Density (Percent Theoretical)*	93.6	92.7	93.1	95.0
Number of Assemblies	1	40	40	40
Approximate Burnup at Beginning of Cycle 2 (MWD/MTU)	17,200	19,200	12,400	0

* All region's values except Region 4 are region average as-built values; Region 4 values are nominal.

Table 2

Kewaunee Cycle 2

Kinetics Characteristics

	Previously Analyzed Values	Cycle 2
Moderator Temperature Coefficient ($\Delta\rho/^{o}F$) x 10^4	-4.0 to 0[2]	-3.5 to 0
Doppler Coefficient ($\Delta\rho/^{o}F$) x 10^5	-1.63 to -1.0[2]	-2.5 to -1.0
Delayed Neutron Fraction β_{eff}(%)	0.50 to 0.70[2]	0.50 to 0.70
Maximum Prompt Neutron Lifetime (μsec)	20[2]	19
Maximum Reactivity Withdrawal Rate (pcm/sec)**	82[2]	82

**pcm = 10^{-5} $\Delta\rho$

Table 3

Kewaunee Nuclear Power Plant - Cycle 1 and 2

Shutdown Requirements and Margins

	Cycle 1		Cycle 2	
Control Rod Worth (%Δρ)	BOC	EOC	BOC	EOC
All Rods Inserted Less Worst Stuck Rod	6.36	6.27	6.00	6.20
(1) Less 10%	5.72	5.64	5.40	5.58
Control Rod Requirements (%Δρ)				
Reactivity Defects (Doppler, T_{avg}, Void Redistribution)	1.71	2.76	1.75	2.64
Rod Insertion Allowance	0.50	0.50	0.50	0.50
(2) Total Requirements	2.21	3.26	2.25	3.14
Shutdown Margin [(1)-(2)] (%Δρ)	3.51	2.38	3.15	2.44
Required Shutdown Margin (%Δρ)	1.00	2.00	1.00	2.00

Table 4

Rod Ejection Parameters
(Reference 5)

HZP - BOL	Previously Analyzed Value	Cycle 2 Value	Used in Re-Analysis
Max. Ejected Rod Worth, %Δρ	0.91	0.43	0.91
Max. F_Q	11.2	7.32	11.2
β_{eff}	0.0070	0.0060	0.0055
HFP - BOL			
Max. Ejected Rod Worth, %Δρ	0.23	0.14	0.27
Max. F_Q	4.83	4.37	5.00
β_{eff}	0.0070	0.0060	0.0055
Initial Fuel Avg. Temperature, ^oF	3185	2730	2780
HZP - EOL			
Max. Ejected Rod Worth, %Δρ	0.89	0.66	0.92
Max. F_Q	12.5	11.66	13.0
β_{eff}	0.0050	0.005	0.0050
HFP - EOL			
Max. Ejected Rod Worth, %Δρ	0.42	0.16	0.42
Max. F_Q	4.00	4.64	4.64
β_{eff}	0.0050	0.005	0.0050
Initial Fuel Avg. Temperature, ^oF	2245	2730	2780

Table 5

Results of Rod Ejection Analysis
Hot Spot Fuel and Clad Temperatures

	BOL	BOL	EOL	EOL
Initial Power, %	0%	102%	0%	102%
Maximum Fuel Pellet Center Temperature (^{o}F)	3552	4900	3904	4800
Maximum Fuel Average Temperature (^{o}F)	3081	3867	3431	3805
Maximum Clad Average Temperature (^{o}F)	2327	2242	2629	2199
Maximum Fuel Enthalpy (Cal/gm)	128	167	145	164

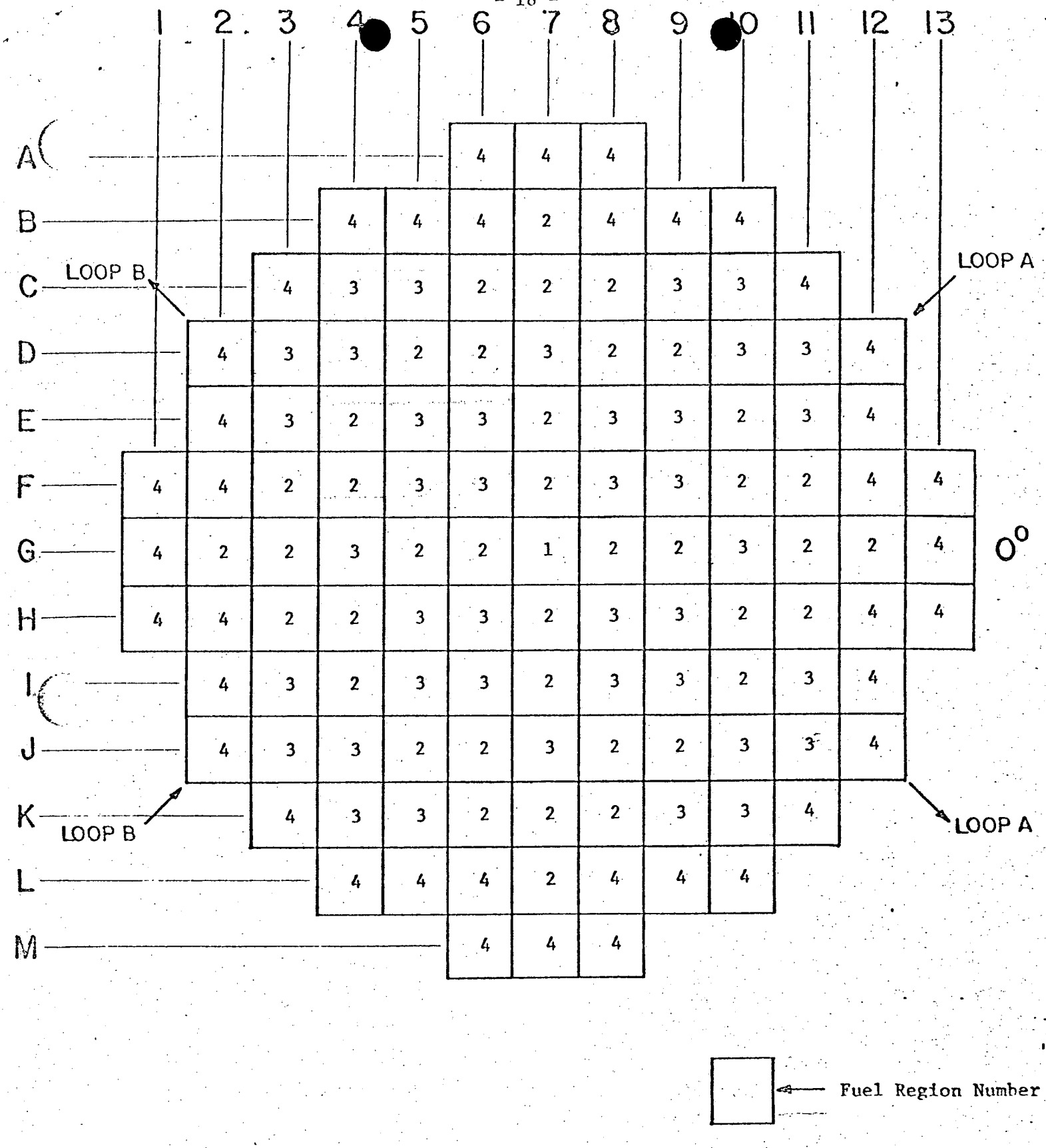

Figure 1
Proposed
Core Loading Pattern
Kewaunee Nuclear Plant
Unit 1 - Cycle 2

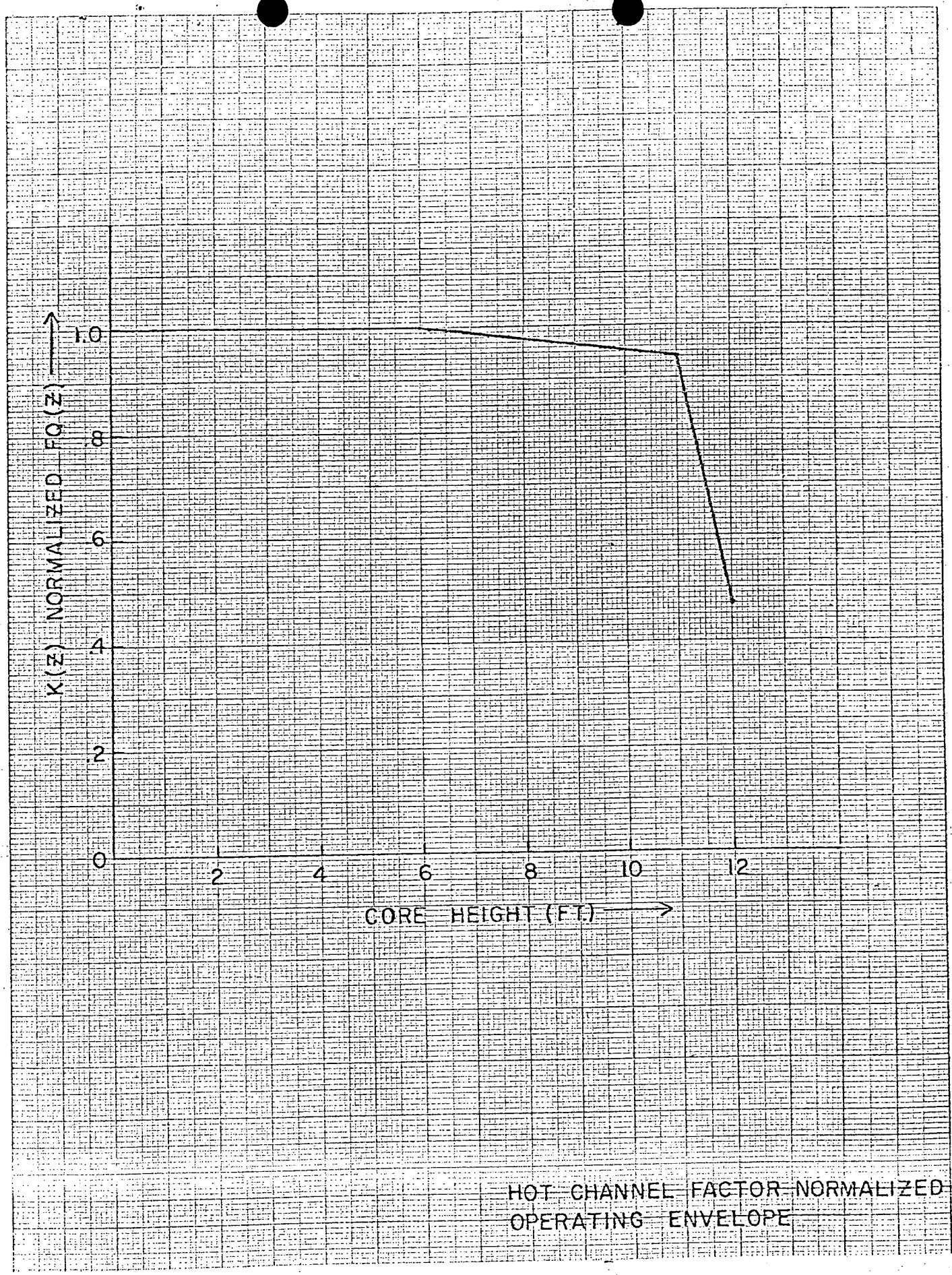

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Figure 2