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LICENSE NO DPR-43 APPL FOR AMEND: TECH SPECS PROPOSED CHANGE NO 33 CONCERNING REVISION TO THE ROD INSERTION LIMITS FOR CYCLE 4... W/ATT REACTOR DESIGN AND ACCIDENT EVALUATION.

PLANT NAME: KEWAUNEE

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March 17, 1978

Office of Nuclear Reactor Regulation
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Washington, D. C. 20555

Attention Mr. A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

RECEIVED MAR 1978 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Gentlemen:

Docket 50-305
Operating License DPR-43
Proposed Technical Specification Amendment No. 33

Please find enclosed forty (40) copies of our proposed Amendment No. 33 to the Kewaunee Nuclear Power Plant Technical Specifications.

The proposed change incorporates a change in the rod insertion limits for Cycle 4. The design of the Cycle 4 core is such that the rod worth requires adjustment of the insertion limit requirements to assure sufficient shutdown margin for the cycle. This change is proposed as an interim requirement for Cycle 4 and subsequent cycles would revert back to the original insertion limit requirements.

The Reload Safety Evaluation for the Kewaunee Nuclear Power Plant Cycle 4 core load has been performed by Westinghouse and 40 copies of this evaluation are being provided as supplemental information to support this proposed amendment.

Very truly yours,

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

sa

Enc.

Subscribed and Sworn to
Before Me This 17th Day
of March 1978

Notary Public, State of Wisconsin

My Commission Expires
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RELOAD SAFETY EVALUATION
KEWAUNEE NUCLEAR PLANT
CYCLE 4

FEBRUARY, 1978

Edited by:

J. A. Iorii
M. D. Beaumont

Work Performed Under Shop Order WPDF-460

Approved: M. G. Arlotti
M. G. Arlotti, Manager
Fuel Licensing and Coordination

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1.0 INTRODUCTION AND SUMMARY

The Kewaunee Nuclear Plant is in its third cycle of operation. The unit is expected to be refueled and be ready for Cycle 4 startup in June, 1978.

This report presents an evaluation for Cycle 4 operation which demonstrates that the core reload will not adversely affect the safety of the plant. It is not the purpose of this report to present a reanalysis of all potential incidents. Those incidents analyzed and reported in the FSAR[1] which could potentially be affected by fuel reload have been reviewed for the Cycle 4 design described herein. The results of new analyses have been included, and the justification for the applicability of previous results from the remaining analyses is presented. These analyses assume that: (1) Cycle 3 operation is terminated at 12,700 ± 500 MWD/MTU, (2) Cycle 4 burnup is limited to the end-of-full power capability*, and (3) there is adherence to plant operating limitations given in the Technical Specifications with modifications as discussed in Section 4.0.

During the Cycle 3/4 refueling, 1 Region 1 fuel assembly and 40 Region 3 fuel assemblies will be discharged and replaced by 1 Region 1 assembly (from Cycle 1) and 40 Region 6 assemblies. See Table 1 for the number of fuel assemblies in each region and Figure 1 for the core loading pattern. Figure 2 shows the locations of the 112 depleted and 224 fresh burnable posion rods used in Cycle 4.

Nominal design parameters for Cycle 4 are 1650 MWt core power, 2250 psia system pressure, nominal core inlet temperature of 532.5, and core average linear power of 6.2 kw/ft, average, based on a 144.0" active fuel length.

*Definition: Full rated power and core inlet temperature of 532.5°F, control rods fully withdrawn, and zero ppm of residual boron.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of Region 6 fuel is the same as Region 5 fuel except as discussed below. Double leaf fuel assembly hold-down springs and their mounting hardware have been incorporated in lieu of single leaf hold-down springs and their mounting hardware used on all previous regions. The use of two leaf springs provides additional hold-down force margin but does not result in overall dimensional change. The fuel assembly shipping, storage, and handling requirements are satisfied using the two leaf spring. The region enrichments are shown in Table 1. Other physical aspects of Region 6 fuel are the same as Region 5. The Region 6 fuel has been designed according to the fuel performance model in Reference 2. The internal pressure of the lead rod in the reactor is limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur. Reference 3 shows that the DNB propagation criteria is satisfied.

Clad flattening time is predicted to be greater than 21,500 EFPH for all fuel regions operating in Cycle 4 using the current Westinghouse evaluation model(4). Since the maximum cumulative irradiation time through Cycle 4 for the limiting region (Region 1) is expected to be approximately 20,500 EFPH, clad flattening does not occur.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is described in WCAP-8183, "Operational Experience with Westinghouse Cores(5)," which is updated periodically.

2.2 NUCLEAR DESIGN

Cycle 4 core loading is designed to meet an $F_Q^T \times P$ limit of ≤ 2.25 (see Section 3.1). A subset of the 18 case ($F_Q^T \times P$) vs. core elevation analysis was performed. Since a subset of cases was used an appropriate multiplier was applied to the calculated $F_Q^T \times P$ points. Table 2 provides a comparison of the Cycle 4 kinetics characteristics with the current limit based on previously submitted accident analysis. It can be seen from Table 2 that most of the Cycle 4 values fall within the current limits. Table 3 provides the minimum end of life control rod worths and the maximum requirements at the most limiting condition during the cycle. The required shutdown margin is based on previously submitted accident analysis. The available shutdown margin exceeds the minimum required provided that the full power core inlet temperature is maintained at $\leq$536.5°F.

The trip reactivity insertion rate for Cycle 4 is slower than the one used in previous cycles (see Section 3.3). The reactivity insertion rate is different because the combined bank worth as a function of time (axial location) has changed. The reactivity insertion rate for Cycle 4 was calculated by a conservative method that produces a flux distribution skewed towards the bottom of the core. This reduces the reactivity worth of the banks at the top of the core relative to the total worth. Such a calculation provides a conservative trip reactivity shape for accident analysis since the axial flux distribution is normally distributed evenly with constant axial offset control.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variation in thermal margins will result from the Cycle 4 reload. The present DNB core limits have been found to be conservative.

3.0 ACCIDENT EVALUATION

3.1 POWER CAPABILITY

This section evaluates the plant power capability considering the consequences of those incidents examined in the FSAR using the previously accepted design bases. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% of rated power during Cycle 4. For the overpower transient, the burnup dependent fuel centerline temperature limit of 4700°F can be accomodated with margin in the Cycle 4 core. The time dependent densification model[6] was used for fuel temperature evaluations. The LOCA limit is met by maintaining $F_Q \times P$ at or below 2.25[7].

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[1] and fuel densification report[8] have been examined. In most cases, it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable Westinghouse safety analysis. For those incidents which were reanalyzed, it was determined that the applicable design basis limits are not exceeded, and therefore, the conclusions presented in the FSAR are still valid.

Reloads can affect kinetics characteristics, control rod worths, and core peaking factors. Cycle 4 parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

3.2.1 KINETICS PARAMETERS

A comparison of the Cycle 4 kinetics parameters with current limits is given in Table 2. With the exception of the Doppler coefficient, all the kinetics values fall within the bounds of the current limits. The

Doppler temperature coefficient is slightly more negative than the current limit resulting in about a 2% increase in the total reactivity feedback during cooldown transients. This has an essentially negligable effect on accident analyses.

3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margins, ejected rod worths and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 4 is less than or equal to the current limits. Table 3 shows that the Cycle 4 shutdown margin requirements are satisfied. Ejected rod worths for Cycle 4 are within the bounds of the current limits, with the exception of the BOL hot full power case as shown in Table 4. This case is reanalyzed in Section 3.3.

Cycle 4 has a slower trip reactivity insertion rate than that used for previous cycles. The effects of this reduced reactivity trip rate have been evaluated for those accidents affected, and compared with previous cycle analyses. Slow transients are relatively insensitive to trip reactivity insertion rate and need be investigated only for increases in total energy release from the fuel to the coolant after the trip. For Cycle 4, the increase in energy generated, compared to the previous trip reactivity curve used, is more than offset by having less stored energy in the fuel than in previous cycles prior to the trip.

Fast transients such as rod ejection and rod withdrawal from subcritical in which negative reactivity insertion is due primarily to Doppler feedback, will be insensitive to the change in trip reactivity rate since the transient is essentially turned around before rod insertion starts. Both of these transients were reanalyzed for other reasons, as described in Section 3.3.

An evaluation of the loss of flow accident with the slower trip reactivity insertion shows that the minimum DNBR remains greater than 1.3 and the FSAR safety criteria are satisfied.

The slower trip reactivity insertion also affects the rod withdrawal at power and the locked rotor accidents which are reanalyzed in Section 3.3.

3.2.3 CORE PEAKING FACTORS

Evaluation of peaking factors for the rod out of position and dropped RCCA incidents show that DNBR is maintained above 1.3. A peaking factor evaluation for the hypothetical steamline break transient showed that the DNBR is maintained above 1.3. The peaking factors following control rod ejection are outside the bounds of previous analyses for the hot full power cases. These rod ejection cases have been reanalyzed in Section 3.3.

Cycle 4 loading pattern gives a slightly more limiting fuel rod census than the current limit. Therefore, the locked rotor accident was reanalyzed in Section 3.3.

3.3 INCIDENTS REANALYZED

The control rod ejection analysis is affected adversely by increased power peaking factors for the hot full power cases. For the BOL hot full power case, additional adverse affects are seen due to increased ejected rod worth. All rod ejection cases were reanalyzed and as shown in Table 5, the hot spot fuel rod does not exceed the limiting criteria[9].

The locked rotor accident was reanalyzed due to a slower trip reactivity insertion rate. As a result of this slower trip reactivity insertion, the number of rods expected to experience DNB has increased to 40%. This analysis used the same methods as were used in the Cycle 3 analysis, including the old densification model(8). Also, as a result of the slower trip reactivity insertion, the peak clad temperature rose to 1800°F for previous cycles. The peak clad temperature is well below the 2700°F limiting value, and the reactor coolant pressure remains below 2750 psia. Therefore, the conclusions of Reference 1 remain valid for Cycle 4.

Uncontrolled rod withdrawal at power was reanalyzed for the critical range of rod withdrawal rates shown in the FSAR (2 to 3 pcm/sec at full power and 15 pcm/sec at 60% power), using the revised slower trip reactivity insertion rate. It was found that the effect was small, and the minimum DNBR remains above the 1.3 minimum limit for all cases.

Uncontrolled RCCA bank withdrawal from a subcritical condition was reanalyzed using the revised slow trip reactivity insertion rate. For the worst case, insertion of +82 pcm per second starting from 10^{-13} full power, the peak heat flux increased from 75.5% full power for Cycle 1 to 84% for Cycle 4. The core water temperature remained below design value. Therefore, the DNB ratio remained well above the limiting value of 1.3, and the conclusions of Reference 4 are still valid for Cycle 4.

4.0 TECHNICAL SPECIFICATIONS

New control rod insertion limits as a function of power are provided in Figure 3. The control rod insertion limits have been changed since the EOC rod insertion allowance has been decreased to 0.45% in order to preserve excess shutdown margin.

5.0 REFERENCES

1. Final Safety Analysis Report - Kewaunee Nuclear Power Plant.

2. Miller, J. V. (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

3. Risher, D. H. et. al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977.

4. George, R. A., et. al., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

5. Schreiber, R. E. and Iorii, J. A., "Operational Experience With Westinghouse Cores," WCAP-8183 Revision 6, June 1977.

6. 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).

7. Letter, WPS-76-84, Lordi to Geisler, December 2, 1976, Transmittal of ECCS Analysis.

8. "Fuel Densification, Kewaunee Nuclear Power Plant," WCAP-8377 (Proprietary) and WCAP-8093 (Non-Proprietary), March, 1973.

9. Risher, D. H. Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

Table 1

Kewaunee Cycle 4
Fuel Assembly Design Parameters

Region	1	4A	4B	5	6
Enrichment (w/o U-235)*	2.26	3.28	3.28	3.30	3.1
Density (Percent Theoretical)*	93.6	94.5	94.5	94.4	95.0
Number of Assemblies	1	32	8	40	40
Approximate Burnup at Beginning of Cycle 4 (MWD/MTU)	18000	19000	12100	14000	0

*All regions except Region 6 are as-built values; Region 6 is the nominal value. However, an average density of 94.5% theoretical was used in thermal and nuclear evaluations.

Table 2

Kewaunee Cycle 4
Kinetics Characteristics

	Previously Analyzed Values(1)	Cycle 4
Moderator Temperature Coefficient, (pcm/°F)*	-35 to 0	-35 to 0
Doppler Temperature Coefficient (pcm/°F)*	-1.65 to -1.0	-2.32 to -1.0
Delayed Neutron Fraction β_{eff} (%)	0.51 to 0.71	0.51 to 0.71
Maximum Prompt Neutron Lifetime (μ sec)	20	20
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (pcm/sec)*	82	82

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

Table 3

Kewaunee - Cycle 3 and 4
Shutdown Requirements and Margins

	Cycle 3		Cycle 4	
Control Rod Worth (%$\Delta\rho$)	BOC	EOC	BOC	EOC
All Rods Inserted Less Worst Stuck Rod	5.69	5.80	5.69	5.77
(1) Less 10%	5.12	5.22	5.12	5.19
Control Rod Requirements (%$\Delta\rho$)				
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	2.09	2.67	1.86	2.69
Rod Insertion Allowance	0.50	0.50	0.93	0.45
(2) Total Requirements	2.59	3.17	2.79	3.14
Shutdown Margin [(1)-(2)] (%$\Delta\rho$)	2.53	2.05	2.33	2.05
Required Shutdown Margin (%$\Delta\rho$)	1.00	2.00	1.00	2.00

TABLE 4
ROD EJECTION PARAMETERS
KEWAUNEE UNIT 1, CYCLE 4

	PREVIOUS ANALYSIS VALUES	VALUES CYCLE 4	VALUE USED IN REANALYSIS
HZP - BOL			
Max. Ejected Rod Worth, $\%\Delta\rho$	0.91	.85	N/A
Max. F_Q	11.2	11.2	
β_{eff}	0.0055	0.0055	
HFP - BOL			
Max. Ejected Rod Worth, $\%\Delta\rho$	0.23	0.30	0.30
Max. F_Q	4.88	5.03	5.03
β_{eff}	0.0055	0.0055	0.0055
HZP - EOL			
Max. Ejected Rod Worth, $\%\Delta\rho$	0.89	$\leq$ 0.89	N/A
Max. F_Q	12.5	$\leq$ 12.5	
β_{eff}	0.005	0.005	
HFP - EOL			
Max. Ejected Rod Worth, $\%\Delta\rho$	0.42	$\leq$ 0.42	0.42
Max. F_Q	4.64	5.10	5.10
β_{eff}	0.005	0.005	0.005

HZP - Hot Zero Power
HFP - Hot Full Power
N/A - No analyses needed for Cycle 4
BOL - Beginning of Life (Cycle 4)
EOL - End of Life (Cycle 4)

Table 5

Results of Rod Ejection Analysis - Hot Spot Fuels
And Clad Temperatures
Kewaunee Cycle 4

	BOL	EOL
Initial Power %	102%	102%
Maximum Fuel Pellet Center Temperature (°F)	4951	4872
Maximum Fuel Average Temperature (°F)	3833	3810
Maximum Clad Average Temperature (°F)	2259	2240
Maximum Fuel Enthalpy (BTU/LB)	298.3	296.1

FIGURE 1

CORE LOADING PATTERN
KEWAUNEE NUCLEAR PLANT
UNIT 1 - CYCLE 4

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						4A	6	4A					
B				5	6	6	4A	6	6	5			
C			5	6	4A	5	4A	5	4A	6	5		
D		5	6	5	6	5	6	5	6	5	6	5	
E		6	4A	6	4A	4B	4A	4B	4A	6	4A	6	
F	4A	6	5	5	4B	5	5	5	4B	5	5	6	4A
G	6	4A	4A	6	4A	5	1	5	4A	5	4A	4A	6
H	4A	6	5	5	4B	5	5	5	4B	5	5	6	4A
I		6	4A	6	4A	4B	4A	4B	4A	6	4A	6	
J		5	6	5	6	5	6	5	6	5	6	5	
K			5	6	4A	5	4A	5	4A	6	5		
L				5	6	6	4A	6	6	5			
M						4A	6	4A					

X	Region Number

FIGURE 2

SOURCE AND BURNABLE POISON LOCATIONS
KEWAUNEE NUCLEAR PLANT
UNIT 1 - CYCLE 4

	1	2	3	4	5	6	7	8	9	10	11	12	13
A													
B							12 D						
C				8 D						8 D			
D			8 D		16 F		12 F		16 F		8 D		
E				16 F		4 F		4 F		16 F			
F					4 F		4 F		4 F				
G		12* D		12 F		4 F		4 F		12 F		12* D	
H					4 F		4 F		4 F				
I				16 F		4 F		4 F		16 F			
J			8 D		16 F		12 F		16 F		8 D		
K				8 D						8 D			
L							12 D						
M													

X* - Source Location

F - Fresh Burnable Poisons
D - Depleted Burnable Poisons

Figure 3

Control Rod Insertion Limits

as a Function of Power

Kewaunee Nuclear Plant

Unit 1 - Cycle 4