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ENCLOSURE
RELOAD SAFETY EVALUATION TURKEY POINT PLANT UNIT 3, CYCLE 4

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DOCKETED USNRC 10-8-76 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

October 4, 1976
L-76-347

RECEIVED OCT 8 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Office of Nuclear Reactor Regulation
Attn: Karl R. Goller, Assistant Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Goller:

Re: Turkey Point Unit No. 3 (Docket No. 50-250)
Unit 3 - Cycle 4 Reload Information

Attached herewith is the Safety Evaluation performed for the reload of Turkey Point Unit 3 and the subsequent return to operation. This report is being forwarded to you for your information. No changes in the facility operating license or the technical specifications are required to conduct the reload or the return to full power following the reload.

As discussed in our letters L-76-300 of August 19, 1976, and L-76-307 of August 25, 1976, the effect of plugged steam generator tubes was considered in the development of the revised F_Q limit of 2.11 which is currently applicable to both Unit 3 and Unit 4. The revised limit includes a tube plugging factor which is based on approximately 4% of the total number of Unit 4 steam generator tubes having been plugged. Since only approximately 2.6% of the Unit 3 steam generator tubes have been plugged, the revised F_Q limit for Unit 3 conservatively accommodates the effect of plugged steam generator tubes on ECCS performance.

In our letter L-76-300 of August 19, 1976, we also stated how rod bow affected the DNB limits of the Turkey Point units. Rod bow for Cycle 4 will be treated in the same manner during Cycle 3.

The Cycle 4 reload and the attached safety analysis have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board. They have concluded that the reload and return to operation following the reload do not involve an unreviewed safety question.

10191

The attached safety evaluation considered operation at a system pressure of 2100 psia and a core inlet temperature of 539°F, and at the FSAR design conditions of 2250 psia system pressure and 546.2°F core inlet temperature. However, at this time, consideration is not being given to operating at other than 2100 psia system pressure and 539°F core inlet temperature, which are the current Unit 3 operating parameters. Westinghouse has indicated that after 2000 MWD/MT of operation in Cycle 4, system operating pressure may have to be increased. Once the necessity for this change is determined, a separate review will be conducted in accordance with 10 CFR 50.59, and any required license amendments will be submitted to your staff.

Very truly yours,

J. A. De Mastry
for

Robert E. Uhrig
Vice President

REU/GDW/hlc
Attachments

cc: Jack R. Newman, Esq.

RELOAD SAFETY EVALUATION

TURKEY POINT PLANT

UNIT 3, CYCLE 4

FLORIDA POWER & LIGHT COMPANY

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1.0 INTRODUCTION AND SUMMARY

Turkey Point Unit 3 is in its third cycle of operation. The unit will refuel and be ready for Cycle 4 startup in mid December, 1976.

The Turkey Point Unit 3 Cycle 4 core loading pattern is shown as Figure 1. Fifty-two Region 3 assemblies will be removed and replaced by forty Region 6 assemblies; and four Region 1 and eight Region 2 assemblies stored in the spent fuel pit during Cycle 3 (see Table 1). Depleted borosilicate burnable poison rods will be used in Cycle 4. The location of these rods is shown in Figure 2.

This report presents an evaluation for Cycle 4 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 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 for the remaining analyses is presented. It has been concluded that the Cycle 4 design does not cause the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the assumption that: (1) Cycle 3 operation is terminated after 8200 $\pm$ 1000 MWD/MTU and (2) there is adherence to plant operating limitations in the Technical Specifications.

Nominal design parameters for the beginning of Cycle 4 are 2192 Mwt core power, 2100 psia system pressure, 539°F core inlet temperature, and 5.56 kw/ft average linear fuel power density (based on 144" active fuel length). However, the safety evaluation of Cycle 4 also considered operation at the FSAR design conditions of 2250 psia system pressure and 546.2°F core inlet temperature. The conclusions presented herein are applicable to both sets of initial conditions.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The only fuel that has not been in the core previously is Region 6 fuel. The mechanical design of Region 6 fuel is dimensionally the same as Region 5 fuel. Region 6 fuel has a different enrichment, as noted in Table 1. Other physical design aspects of Region 6 are the same as Region 5, except that the initial pressurization level of the fuel rods has been decreased by 65 psi. Region 6 is comprised of fuel made using UO_2 powder which meets Westinghouse specifications, but was fabricated by a process which differs from the standard Westinghouse powder process. The Draft Regulatory Guide on Densification dated July 29, 1975, will be used to determine the amount of densification in Region 6 fuel.

Clad flattening time is predicted to be greater than 23,500 EFPH for the limiting region (Region 2) using the current Westinghouse evaluation model[1]. Therefore, Region 2 has a nominal Cycle 4 allowed residence time of 6400 EFPH (Cycle 2 lifetime was 6800 EFPH, and Cycle 1 lifetime was 10,300 EFPH).

Westinghouse has had considerable experience with Zircaloy-clad fuel. This experience is extensively described in WCAP-8183, "Operational Experience with Westinghouse Cores"[3]. This report is updated about every six months.

2.2 NUCLEAR DESIGN

Table 2 provides a comparison of the range of values encompassing the Cycle 4 core kinetics parameters with the current limits based on previously submitted accident analyses. It can be seen from the table that most of the Cycle 4 range of values fall within the current limits. These parameters are evaluted in Section 3.0. Table 3 provides the control rod worths and requirements. The required shutdown margin is based on previously submitted accident analysis[4]. The available shutdown margin exceeds the minimum required.

The reactivity insertion rate for Cycle 4 is slower than the one used in previous cycles (see Section 3.3 and Figure 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 very 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 variations in thermal margins will result from the Cycle 4 reload.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

This section reviews 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% rated power during Cycle 4. A maximum local rod power of 23 kw/ft corresponds to the fuel centerline temperature limit of 4700°F for Region 5 or 6 fuel. This can be accommodated with margin in the Cycle 4 core. The time dependent densification model[2] was used for this evaluation. No significant variation in the LOCA limit will result from the Cycle 4 reload.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[4] 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 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.

A reload can typically affect accident analysis input parameters in three major areas: 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 are required.

Kinetics Parameters

A comparison of the range of values encompassing the Cycle 4 kinetics parameters with the current limits is given in Table 2. Most of the

range of values remain within the bounds of current limits. The small changes in core physics parameters have a negligible effect on transient analysis. Therefore, no additional accident analysis is required due to changes in these parameters.

Control Rod Worths

Changes in control rod worths may affect shutdown margin, differential rod worths, ejected rod worths, and trip reactivity. Table 3 shows that the Cycle 4 shutdown margin requirements are satisfied. As shown in Table 2, 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 limit.

Ejected rod worths for Cycle 4 are within the bounds of the current limits.

Cycle 4 has a slower trip reactivity insertion rate than Cycle 1; however, the total trip reactivity is significantly greater than the value assumed in Cycle 1. The effects of this reduced reactivity trip rate has been evaluted for those accidents affected and compared with the Cycle 1 analyses. Slow transients are relatively insensitive to changes in trip reactivity insertion rate, and therefore need not be reanalyzed due to the change in trip reactivity versus rod position. Fast transients such as rod ejection and rod withdrawal from subcritical, in which negative reactivity insertion is due primarily to Doppler feedback, will be unaffected by the change in trip reactivity since the transient is essentially turned around before rod insertion starts. The effect of variations in trip reactivity insertion rates for the rod withdrawal at power incident has been investigated. The results of this analysis show that the minimum DNBR is unaffected, since the minimum DNBR for the transient occurs at relatively low reactivity insertion rates.

For the loss of flow and locked rotor transients, the change in trip reactivity versus rod insertion will result in a slightly higher transient heat flux. Since the minimum DNB ratios for these transients are sensitive to heat flux relative to flow, these incidents were reanalyzed. The results of these calculations are discussed in Section 3.3.

Core Peaking Factors

Peaking factors following control rod ejection are within the bounds of the current limits. Evaluation of peaking factors for the rod out of position and dropped RCCA incidents shows that DNBR is maintained above 1.3. For the dropped bank incident, the minimum DNBR criteria of 1.3 is satisfied without taking credit for a turbine runback. A peaking factor evaluation for the hypothetical steamline break transient showed that the DNBR is maintained above 1.30.

3.3 INCIDENTS REANALYZED

Certain incidents were reanalyzed at a system pressure of 2250 psia. The results described below are also conservative for a system pressure of 2100 psia.

The complete loss of flow and locked rotor transients were reanalyzed due to a slower trip reactivity insertion rate. A comparison of the FSAR and Cycle 4 trip reactivity versus rod position values are shown in Figure 3 along with the conservative values assumed in the reanalysis. As noted in Section 3.2, the assumed total trip worth has been increased from a value of 2.8% to 4.0% $\Delta k/k$ which is conservative for Cycle 4. The calculations were performed using the same methods and assumptions used for Cycle 1[5].

For the complete 3/3 pump loss of flow incident, the minimum DNB ratio does not fall below the limiting value of 1.30. The other cases described in the FSAR incident show larger DNB ratios than the 3/3 pump incident. Thus it is concluded that all cases would remain above the 1.30 limit and the conclusions as presented in the FSAR for this incident are still valid.

For the two and three loop locked rotor cases, no rods are expected to experience DNB; however, the hot channel quality at the location of the minimum DNB ratio exceeds the range of quality over which the DNB correlation was derived. For this reason, DNB was conservatively assumed to occur. The three loop case is most limiting from a peak clad temperature standpoint. The results show a peak clad temperature of 1500°F which is well below the limiting value of 2700°F. The amount of $Zr-H_2O$ reaction is small (less than 1% by weight) and contributes less than 10°F to the peak cladding temperature. The two loop case is most limiting from a peak pressure standpoint. The results show a peak pressure of less than 2720 psia which is below the reactor vessel stress limit. Thus, ~~the conclusions as presented~~ in the FSAR are still valid for Cycle 4.

4.0 CONCLUSIONS

No changes in the Technical Specifications are required for the upcoming Unit 3 refueling or for the return to power operation following the refueling.

The refueling and the subsequent return to full power operation have been evaluated to verify that they do not involve an unreviewed safety question. Title 10 of the Code of Federal Regulations, Part 50.59(a), specifies that an unreviewed safety question exists if:

(1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or

(2) A possibility of an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or

(3) The margin of safety as defined in the basis for any technical specification is reduced.

The next refueling of Unit 3, which is scheduled for the Fall of 1976, is the third refueling of this Unit. There are no significant differences between this refueling operation and previous refueling operations. The refueling operation has been previously analyzed as reported in the Turkey Point Units 3 and 4 Final Safety Analysis Report.

The NRC Staff Safety Evaluation Report dated March 15, 1972 found this refueling operation to be acceptable. Therefore, there is no unreviewed safety question associated with the refueling operation.

Reactor design, power capability, and postulated incidents have also been evaluated to determine if they involve an unreviewed safety question. The conclusion presented above in Section 2.0 and 3.0 of this safety evaluation show that the Unit 3 Cycle 4 reload satisfies the criteria of 10 CFR Part 50.59(a), and that all limits previously found to be acceptable by the NRC Staff will still apply.

In summary, we have concluded that:

(1) Since the core refueling does not increase the probability or consequences of a previously analyzed accident, decrease safety margin, or create the possibility of an accident not previously analyzed, the refueling does not involve an unreviewed safety question;

(2) The existing Limiting Conditions for Operation incorporated in the Technical Specifications are appropriate for use during Unit 3 Cycle 4 operation;

(3) There is reasonable assurance that the health and safety of the public will not be endangered by the operation as described herein; and

(4) Such activities will be conducted in compliance with the Commission's regulations and will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

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

2. 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).

3. Schreiber, R. E., Iorii, J. A., Plocido, V. J., "Operational Experience With Westinghouse Cores", WCAP 8183, Revision 4, March 1976.

4. Final Safety Analysis Report, Turkey Point Units No. 3 and 4.

5. "Fuel Densification, Turkey Point Plant Unit No. 3", WCAP 8074 (Proprietary) and WCAP 8075 (Non Proprietary), February 1973.

TABLE 1

TURKEY POINT 3 - CYCLE 4

FUEL ASSEMBLY DESIGN PARAMETERS

Region	1	2	4	5A	5B	6
Enrichment (w/o U 235)	1.86	2.56	2.56	2.60	2.90	3.10
Density (% Theoretical)*	93.8	92.8	94.6	94.8	94.5	94.5
Number of Assemblies	5	12	52	24	24	40
Approximate Burnup at Beginning of Cycle 4 (MWD/MTU)	12000	23700	16700	8100	6600	0
Change in Internal Rod Pressure compared to Region 1 (psi)	--	0	+130	+100	+100	+35

* All regions except Region 6 are as-built values.

TABLE 2

TURKEY POINT UNIT 3

KINETICS CHARACTERISTICS

	Current Limit (2250 psia)	Cycle 3 (2100 psia)	Cycle 4 † (2250 psia)
Moderator Temperature Coefficient, $(\Delta\rho/°F)x10^4$	-3.5 to +0.3*	-3.5 to 0	-3.5 to 0
Doppler Coefficient, $(\Delta\rho/°F)x10^5$	-1.6 to -1.0	-2.6 to -1.0	-2.6 to -1.0
Delayed Neutron Fraction, β_{eff}, (%)	.50 to .72	.50 to .72	.50 to .72
Prompt Neutron Lifetime (μsec)	14 to 18	20	20
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (pcm/in)**	80	80	80

* The positive coefficient does not occur at operating conditions

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

† The Cycle 4 range of values is valid for both the nominal design conditions of the plant (2250 psia pressure, 546.2°F inlet temperature) and for the planned Cycle 4 operating conditions (2100 psia pressure, 539°F inlet temperature).

TABLE 3

TURKEY POINT 3 - CYCLE 3 AND 4

SHUTDOWN REQUIREMENTS AND MARGINS

	CYCLE 3		CYCLE 4 *	
Control Rod Worth (% Δρ)	BOC	EOC	BOC	EOC
All Rods Inserted Less Worst Stuck Rod	5.87	6.19	6.34	6.39
(1) Less 10%	5.28	5.57	5.70	5.75
Control Rod Requirements (% Δρ)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution)	1.64	2.52	1.62	2.57
Rod Insertion Allowance	.50	.50	.60	.60
(2) Total Requirements	2.14	3.02	2.22	3.17
Shutdown Margin [(1)-(2)] (% Δρ)	3.14	2.55	3.48	2.58
Required Shutdown Margin (% Δρ)	1.00	1.77	1.00	1.77

* The Cycle 4 range of values is valid for both the nominal design conditions of the plant (2250 psia pressure, 546.2°F inlet temperature) and for the planned Cycle 4 operating conditions (2100 psia pressure, 539°F inlet temperature).

FIGURE 1

CORE LOADING PATTERN
TURKEY POINT UNIT 3 CYCLE 4

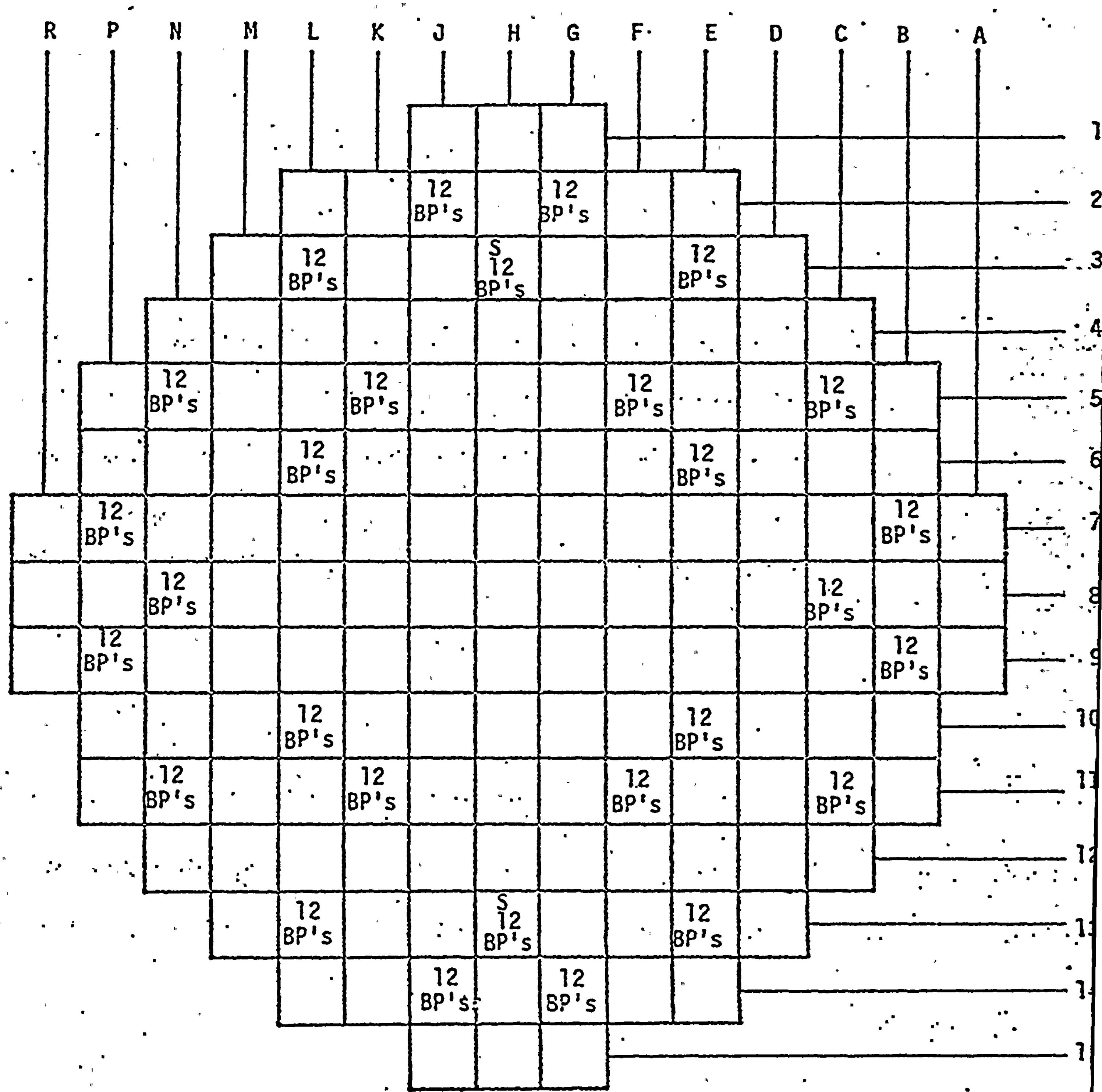
R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
						6	4*	6						
				2	6	6	4	6	6	2				
			6	6	5A	4	4	4	5A	6	6			
		6	4	5A	4	5B	4	5B	4	5A	4	6		
	2	6	5A	4	5B	4	5A	4	5B	4	5A	6	2	
	6	5A	4	5B	1	5B	2	5B	1	5B	4	5A	6	
6	6	4	5B	4	5B	4	5A	4	5B	4	5B	4	6	6
4*	4	4	4	5A	2	5A	1	5A	2	5A	4	4	4	4
6	6	4	5B	4	5B	4	5A	4	5B	4	5B	4	6	6
	6	5A	4	5B	1	5B	2	5B	1	5B	4	5A	6	
	2	6	5A	4	5B	4	5A	4	5B	4	5A	6	2	
		6	4	5A	4	5B	4	5B	4	5A	4	6		
			6	6	5A	4	4	4	5A	6	6			
				2	6	6	4	6	6	2				
						6	4	6						

X Fuel Region

* Removable Rod Assembly

FIGURE 2

SOURCE AND TURKEY POINT UNIT 3 CYCLE 4
BURNABLE POISON LOCATIONS



S BP's Source Location
Depleted Burnable Poisons

Figure 3

TURKEY POINT UNIT 3 CYCLE 4
TRIP REACTIVITY VERSUS ROD POSITION