1.0 Introduction and Summary

This Reload Safety Evaluation (RSE) report presents a description and safety evaluation for the Cycle 2 reload core design of Unit 1 of the North Anna Power Station. Unit 1 completed its first cycle of operation in September, 1979 and achieved a core average burnup of 15,892 MWD/MTU. During the subsequent refueling, 52 Region 1 fuel assemblies will be replaced with 52 fresh Region 4 fuel assemblies. Unit 1 is projected to begin Cycle 2 operation in December, 1979 and will extend to the Fall of 1980, producing approximately 9400 MWD/MTU (246 EFPD) of energy. Operation of up to approximately 10,400 MWD/MTU (272 EFPD) is allowed in a power coastdown mode.

All design and safety analyses performed for the Cycle 2 reload core were based on the following assumptions:

1) Cycle 1 operation is terminated between 14,300 and 15,900 MWD/MTU.
2) Cycle 2 burnup will not exceed 10,400 MWD/MTU.
3) Adherence to the plant operating limitations delineated in the approved Technical Specifications(1) along with the proposed Technical Specifications changes given in Attachment 2 is maintained. The Technical Specifications changes are discussed in Section 4.0 of this report.

The safety evaluation methodology applied to the Cycle 2 reload core is described in detail in Reference 2. All of the postulated accidents which were analyzed and reported in the FSAR(3) have been reviewed as described in Reference 2 to determine the potential impact of the Cycle 2 reload core design on the transient results. The conclusions in the FSAR for the following accidents were found to be potentially affected: Rod Cluster Control Assembly Ejection from hot zero power at both beginning and end of life, the Main Stream Line Break with a loss of offsite power, Excessive Heat Removal due to Feedwater System Malfunction and Excessive Load Increase. These accidents were reanalyzed,

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and the results are discussed in Section 3.0 of this report.

Based on the evaluation contained in this report, it has been determined that the proposed Cycle 2 reload core will not adversely affect the safety of the station and, at the same time, will accomplish its energy generation requirement.

2.0 Reload Core Design

2.1 Basic Design Parameters

The basic design parameters for Cycle 2 are a rated core power of 2775 MWt, a system pressure of 2250 psia, a reactor average temperature (T_{avg}) of 580.3^{o}F, a core linear power density of 5.44 kw/ft (based on 143.7 inches average active fuel length) and a thermal design flow rate of 278,400 gpm.(3)

2.2 Design Loading Pattern

The fuel assembly loading pattern is shown in Figure 1 and significant core design parameters are denoted in Table 1. The core loading will contain 304 depleted burnable poison rods. The burnable poison rod locations and distributions are shown in Figure 2. Also shown in Figure 2 are the locations of full length control rod and source assemblies. As shown, two unirradiated secondary sources will be activated during Cycle 2 operation.

2.3 Mechanical Design

The mechanical design of the fresh Region 4 fuel assemblies is identical to that of Region 3. Clad flattening will not occur during Cycle 2, as predicted by the currently approved model.(4)

For all fuel regions, the fuel rod internal pressure design basis limits the internal pressure of the lead fuel rod to a value below that which would cause 1) the diametral gap to increase due to outward cladding creep during steady state operation and 2) the occurrence of extensive DNB

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propagation. These criteria have been shown to be acceptable during normal operation and accident conditions.[5] The above criteria are satisfied for the reactor for Cycle 2 operation as determined by the NRC approved application of the fuel performance model in Reference 6. Considerable operating experience has been obtained with Westinghouse fuel and is extensively described in WCAP 8183,[7] which is updated periodically.

2.4 Thermal and Hydraulic Design

The present DNB core limits[3] have been found to be conservative for Cycle 2.

The minimum fuel temperatures at power for Cycle 2 were found to be lower than those previously used in the FSAR. This resulted from the use of more conservative methodology in the calculation of the fuel temperatures. The transients affected by this change are the cooldown accidents. Of these accidents, the Excessive Load Increase and Excessive Heat Removal due to Feedwater System Malfunction transients (which are assumed to be initiated at full power) were potentially adversely affected. These transients were reanalyzed and are discussed further in Section 3.2.

The maximum linear power density limit of 21.1 kw/ft for overpower accidents is established by the burnup-dependent fuel centerline temperature limit of 4700°F, and the time dependent fuel densification model of Reference 8. The maximum local linear power density calculated for any overpower transient resulting from allowable Cycle 2 operating conditions does not exceed this limit.

Recent data on the effects of fuel rod bow on DNB and the magnitude of rod bow occurring during irradiation indicate that the current Technical Specifications reductions on the $F^N_{\Delta H}$ limit are very conservative. Thus a change to the Technical Specifications which still conservatively accommodates the effects of fuel rod bow is proposed in Attachment 2 and is discussed in detail in Section 4.0.

2.5 Nuclear Design

Representative radial power distributions have been calculated with the model documented in Reference 9 and are provided in Figures 3-6. Assemblywise beginning-of-cycle burnup distributions are provided in Figure 1 and estimated region average end of cycle burnups are provided in Table 1. Burnup calculations were performed with the model documented in Reference 10. The radial peaking factors associated with routine steady state operation are within appropriate design limits.

The moderator temperature coefficient will be zero or negative above Hot Zero Power, as required by the current limits. The Cycle 2 reload core shutdown margin is adequate, as shown in Table 2. The control rod insertion limits have been reviewed and found to meet all the criteria specified in Reference 11. Consequently, the insertion limits have not changed and are the current limits as shown in Figure 7.

2.6 Startup Physics Test Program

A startup physics testing program will be conducted to ensure that Cycle 2 physics performance characteristics will be bounded by the appropriate design calculations. This program will include measurement of temperature coefficients, boron end points, rod bank worths, boron worths, and power distribution, and is identical to that documented in References 12 and 13, except for the deletion of measurements of the isothermal temperature coefficient with D and C rod banks inserted and of the power coefficient. A report on the North Anna 1, Cycle 2 startup program will be completed in a timely manner and made available to the NRC. Design calculations to support the testing program and core operation will be performed with the models documented in References 9, 10 and 14.

3.0 Safety Evaluation

3.1 General

This section provides an evaluation of the impact of the Cycle 2 reload core on the design basis and postulated incidents previously analyzed in the FSAR. This impact, which is dicussed below, does not adversely affect the ability to safely operate the reactor at up to 100% of rated thermal power during Cycle 2.

A reload core can typically affect accident analysis input parameters in three major areas: Kinetics characteristics, control rod worths, and core peaking factors. The Cycle 2 reload core parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

Kinetics Parmeters

A comparison of the values of the Cycle 2 kinetics parameters with current limits is given in Table 3. The delayed neutron fraction, Doppler only power coefficient, Doppler temperature coefficient, and the prompt neutron lifetime are all within the bounds of the current limits for these parameters.

Control Rod Worth

Changes in control rod worths may affect the shutdown margin, the maximum positive reactivity insertion rate, the trip reactivity and magnitude of the rod worths assumed in the safety analysis (i.e., rod ejection). As discussed in Section 2.5, the Cycle 2 reload core shutdown margin is adequate. As shown in Table 3, the maximum positive reactivity insertion rate from a subcritical condition associated with the withdrawal of two RCCA control banks moving together in their highest worth region for Cycle 2 is less than the current limit value of 75 pcm/sec. Ejected rod worths for Cycle 2 are within the bounds of the current limits, except for the

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BOL hot zero power case. This case was reanalyzed, and the results of the analysis are discussed in Section 3.2. A review of all dropped RCCA incidents shows that protection is provided by the NIS High Negative Rate Protection System as modified by Reference 15.

Cycle 2 has a trip reactivity insertion characteristic which differs slightly from that used in the FSAR. The difference consists of a very small reduction in trip reactivity insertion rate after the rods are inserted at least 45 percent. The effects of this deviation have been evaluated for those accidents which are potentially affected. For fast transients, the minimum DNBR is reach prior to 45% insertion of the rods. 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 following a trip. This potential has been investigated, and it has been shown that these effects will not change the safety conclusions of the FSAR. Therefore, no reanalysis is required.

Core Peaking Factors

Core peaking factors during postulated abnormal conditions influence the maximum fuel rod centerline temperature, the maximum heat flux and the initial stored energy in the fuel.

Frequent axial power distribution monitoring will be maintained in accordance with the Technical Specifications[1] to ensure that the limiting total peaking factors ($F_Q(Z)$) obtained during Cycle 2 routine steady state and load follow (Condition I) operation do not exceed the total peaking factor limits delineated in the Technical Specifications.* The radial peaking factor, $F_{xy}(Z)$, associated with Condition I operation is predicted to exceed the current $F_{xy}(Z)$ Technical Specifications limit. (See Section 4.2.2.2 of Reference 1). Consequently, a change to the Technical Specifications is

*The bases for the current Technical Specifications limit on F_Q is found in Reference 16. However, a new LOCA-ECCS analysis is now being performed to accommodate small percentages (e.g., up to 5%) of steam generator tube plugging.

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being proposed and is discussed in detail in Section 4.0.

For most postulated accident conditions, Cycle 2 peaking factors were within previously analyzed limits. However, the maximum $F^N_{\Delta H}$ obtained during the return to power portion of the hypothetical Main Steam Line Break Accident is greater than that used in the FSAR. This accident was reanalyzed, and the results of the reanalysis are discussed in Section 3.2. The peaking factors associated with the Rod Ejection Accident are within current limits for the BOL and EOL hot full power cases, but the BOL and EOL hot zero power ejected rod peaking factors exceed the previously analyzed values. The cases have been reanalyzed and the reanalysis results are discussed in Section 3.2.

3.2 Incidents Reanalyzed

3.2.1 Control Rod Ejection Accident

The Control Rod Ejection Accident analysis is affected by an increased ejected rod worth for the BOL hot zero power case and by increased power peaking factors for the BOL and EOL hot zero power cases. These cases were reanalyzed. The hot spot fuel rod and system parameters do not exceed the limiting criteria of the FSAR and Reference 17 for these cases. Therefore, the conclusions of the FSAR remain valid. Renaalysis assumptions for the Control Rod Ejection Accident are given in Table 4, and the results are presented in Table 5.

3.2.2 Main Steam Line Break Accident

The hypothetical Main Steam Line Break accident (loss of offsite power case) was reanalyzed due to an increase in the maximum $F^N_{\Delta H}$ obtained during the return to power portion of the transient. A limiting statepoint analysis was performed using a detailed reactivity feedback calculation which was conservative for Cycle 2 but more realistic than that used in the FSAR analysis. The results of the analysis (See Table 6) show that the calculated minimum DNBR remains greater than 1.30. Thus, all safety criteria

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are met and the conclusions presented in the FSAR remain valid.

3.2.3 Cooldown Transients

The Excessive Heat Removal due to Feedwater System Malfunction and Excessive Load Increase Transients were reanalyzed as a result of the calculated minimum fuel temperatures for Cycle 2 being lower than those used in the FSAR analyses. Initial conditons of power, temperature, pressure and flow were consistent with the analyses presented in the FSAR. The results are essentially the same as shown in the FSAR; the minimum DNBRs remain greater than 1.3 for both accidents.

4.0 Technical Specifications

4.1 Rod Bow Reduction in $F^N_{\Delta H}$

The current reductions being applied to $F^N_{\Delta H}$ to accommodate the impact of fuel rod bow on core thermal margins are defined in Section 3.2.3 of the Technical Specifications.(1) The reductions are based on the DNBR reduction as given in Reference 18, and are based on rod bow DNB tests in which selected fuel rods (forming a thimble cell) were bowed into contact. However, recent data obtained and evaluated by Westinghouse indicate that the appropriate reductions in DNBR (or $F^N_{\Delta H}$) resulting from fuel rod bow during irradiation are significantly less than those currently being accommodated in the Technical Specifications.

Westinghouse has now determined, as documented in References 19, 20 and 21, that irradiation of a region of fuel up to 33,000 MWD/MTU (the nominal region average burnup) would result in channel closure of less than 85% on a 95 x 95 basis. The limiting DNBR reductions associated with 85% channel closure tests are found to be 11.4% for North Anna fuel for full flow operation, and 14% for N-1 loop operation and the Loss of Flow Transient(22).

(N-1 loop operation is not a consideration, since this mode of operation is precluded by the North Anna Unit 1 Operating License).

This DNBR reduction can be partially offset by existing generic thermal margin in the core design. The margins are delineated on page 14 of Reference 18, and for North Anna Units No. 1 and 2, provide thermal margins of 9.1%. The remaining DNBR reductions for full flow operation and for the Loss of Flow transient are shown in Table 7. The DNBR reduction associated with full flow operation will be accommodated in the Technical Specifications through the region averaged burnup dependent reduction in the $F^N_{\Delta H}$ limit shown in Figure 8. As displayed in Figure 8, the generic thermal margins discussed above offset the DNBR (or $F^N_{\Delta H}$) reduction until region averaged burnups of 26,500 MWD/MTU are attained. Above 26,500 MWD/MTU, the $F^N_{\Delta H}$ reduction increases linearly from zero to 1.3% at 33,000 MWD/MTU.

Elimination of the additional DNBR penalty (2.6%) associated with the Loss of Flow transient is accomplished by taking partial credit for the margin available (5.8%) in the limiting analysis described in Section 15.3.4 of Reference 3. Therefore, further reduction of the $F^N_{\Delta H}$ limit is not required.

By applying the $F^N_{\Delta H}$ reduction shown in Figure 8, the effects of fuel rod bow on DNBR will continue to be conservatively accommodated.

The proposed Technical Specifications change is provided in Attachment 2.

4.2 $F_{xy}(Z)$ Limits

The Technical Specifications currently define an upper limit on the radial peaking factor, $F_{xy}(Z)$, of 1.55 for all unrodded core planes and 1.71 for core planes containing Bank D control rods. (See Section 4.2.2.2 of the Technical Specifications[1]). Calculations for Cycle 2 have shown that limiting values of $F_{xy}(Z)$ associated with Condition 1 operation will exceed this limit at some axial elevations. Therefore, the following revised limits are proposed:

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1) $F_{xy} \leq 1.71$ for all core planes containing Bank D control rods,

2) $F_{xy} \leq 1.57$ for all unrodded core planes above 8 ft. elevation, and,

3) $F_{xy} \leq 1.65$ for unrodded core planes 0 to 8 ft. elevation.

These limits conservatively bound the Cycle 2 calculated values, and are consistent with the development of the heat flux hot channel factor, $F_Q^T(Z)$, which meets the limit as discussed previously in Section 3.1.

The proposed Technical Specifications change is provided in Attachment 2.

5.0 Conclusions

The Cycle 2 reload core will not adversely affect the safety of North Anna Unit No. 1 while achieving the requirement for nominal energy generation of approximately 9400 MWD/MTU and maximum energy generation of 10,400 MWD/MTU in a power coastdown mode of operation. This conclusion is based on the development of a core loading pattern which meets the reactivity requirements while maintaining most reactor safety and design parameters within their current limits. For those parameters which are predicted to fall outside their current limits, the impact on safety has been evaluated. This evaluation was supported by specific accident reanalyses. The results of these reanalyses were within appropriate limiting criteria and the conclusions presented in the FSAR are thus demonstrated to be valid for Cycle 2 operation, subject to incorporation of the proposed Technical Specifications changes discussed in Section 4.0.

6.0 References

1. Technical Specifications, North Anna Power Station, Unit 1, dated November 26, 1977, as amended.

2. F. M. Bordelon, et.al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272 , March 1978.

3. North Anna Power Station, Units 1 and 2, Final Safety Analysis Report, Docket Nos. 50-338 and 50-339, dated January 3, 1973, as amended.

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

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

6. Miller, J. V. (ed) "Improved Analytical Models Used in Westinghouse fuel Rod Design Computations," WCAP 8720 (Proprietary) and WCAP - 8795 (Non-Proprietary), October 1976.

7. O'Hara, T. L. And Lorii, J. A., "Operational Experience with Westinghouse Cores," WCAP - 8183, Revision 8, April 1979.

8. Hellman, J. M. (ed), "Fuel Densification Experimental Results and Model for Reactor Operation," WCAP 8219-A, March 1975.

9. Letter from W. N. Thomas (Vepco) to B. C. Rusche (NRC), Serial No. 166, July 26, 1976, (VEP-FRD-19, "The PDQ07 Discrete Model, Virginia Electric and Power Company')

10. Letter from W. N. Thomas (Vepco) to B. C. Rusche (NRC), Serial No. 011 January 25, 1977, (VEP-FRD-20, "The PDQ07 One Zone Model, Virginia Electric and Power Company").

11. Letter from C. M. Stallings (Vepco) to R. W. Reid (NRC), Serial No. 746, October 22, 1975, transmitting Supplemental Information in Support of Technical Specification Change No. 33 (Amendment to Operating License DPR-32 and DPR-37), Surry Power Station - Units Nos. 1 and 2.

12. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC), Serial No. 272, May 11, 1978, transmitting Supplemental Information to Amendment to the Operating License, Technical Specifications Change No. 65, Surry Power Station - Units No. 1 and 2.

13. Letter from W. N. Thomas (Vepco) to H. R. Denton (NRC), Serial No. 581, October 13, 1978, (VEP-FRD-30, "Surry Unit 1, Cycle 5 Startup Physics Test Report, Virginia Electric and Power Company").

14. Letter from W. N. Thomas (Vepco) to H. R. Denton (NRC), Serial No. 017, January 9, 1979, (VEP-FRD-24, "The Vepco FLAME Model, Virginia Electric and Power Company").

15. Letter from C. M. Stallings (Vepco) to H. R. Denton (NRC), Serial No. 504, July 6, 1979, transmitting request for Amendment to Operating License, Proposed Technical Specification Change No. 21, North Anna Power Station Unit No. 1.

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16. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC), Serial No. 258, May 5, 1978, transmitting request for Amendment to Operating License, North Anna Power Station Unit No. 1, Proposed Technical Specification Change No. 11.

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

18. "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactor (Revision 1)", U. S. Nuclear Regulatory Commission, February 16, 1977.

19. J. R. Reavis, et. al., "Fuel Rod Bowing", WCAP 8692, December, 1975, Westinghouse Electric Corporation .

20. "Rod Bow Effects in Westinghouse Fuel", Westinghouse (E. Eicheldinger) to NRC (D. F. Ross) letter, NS-CE-1580, dated October 24, 1977.

21. "Fuel Rod Bowing", Westinghouse (T. M. Anderson) to NRC (D. G. Ross, Jr.) Letter, NS-TMA-1760, dated May 25, 1978.

22. "Staff Review of WCAP-8691", NRC (F. F. Stolz) to Westinghouse (T. M. Anderson) letter dated April 5, 1979.

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TABLE 1

CORE DESIGN PARAMETERS FOR
NORTH ANNA UNIT 1, CYCLE 2

Region	1	2	3	4
Enrichment (w/o U235)	2.11	2.60	3.10	3.2
Density (% Theoretical)	94.43	94.53	94.50	95*
Number of Assemblies	1	52	52	52
Approximate** Burnup at Beginning of Cycle 2 (MWD/MTU)	13800	18300	12000	0
Estimate of Max Burnup at End of Cycle 2** (MWD/MTU)	23600	28300	23600	9500
MTU per Region	0.46	23.91	23.92	24.25

*Nominal

**Based on an end of Cycle 1 core average burnup of 15,900 MWD/MTU and/or a Cycle 2 core lifetime of 10400 MWD/MTU.

TABLE 2

SHUTDOWN REQUIREMENTS AND MARGINS
NORTH ANNA UNIT 1 - CYCLE 2

Control Rod Worth (%$\Delta\rho$)	BOL	EOL
All Rods Inserted	7.83	8.46
All Rods Inserted Less Worst Stuck Rod	6.28	6.67
(1) Less 10%	5.65	6.01
Control Rod Requirements (%$\Delta\rho$)		
Reactivity Defects (Combined Doppler, Tavg, Void and Redistribution Effects)	1.87	2.97
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	2.37	3.47
Shutdown Margin (1)-(2) (%$\Delta\rho$)	3.28	2.54
Required Shutdown Margin (%$\Delta\rho$)	1.77	1.77

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TABLE 3

KINETICS CHARACTERISTICS

	Current Limit	Cycle 2
Moderator Temperature Coefficient (pcm/^{o}F)*	0.0 to -43	-0.7 to -34
Most Negative Doppler Temperature Coefficient (pcm/^{o}F)	-2.2	-2.2
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)	-10.3 (HZP) to -6.67 (HFP)	-11.97 (HZP) to -8.18 (HFP)
Most Negative Doppler - Only Power Coefficient Zero to Full Power (pcm/% power)	-19.4 (HZP) to -12.8 (HFP)	-12.24 (HZP) to -8.48 (HFP)
Delayed Neutron Fraction βeff (%)	0.44 (EOL) to 0.75 (BOL)	0.52 (EOL) to 0.60 (BOL)
Maximum Prompt Neutron Lifetime (μ sec)	26	17
Maximum Positive Reactivity Insertion Rate From Subcritical (pcm/sec.)	75	61

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

TABLE 4

REANALYSIS ASSUMPTION FOR
THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT
HOT ZERO POWER CASES

Time in Life	BOL		EOL	
	Current Analysis	Previous Analysis	Current Analysis	Previous Analysis
Power Level, %	0	0	0	0
Ejected Rod Worth, %Δk	0.878	0.785	0.98	0.98
Delayed Neutron Fraction %	0.52	0.52	0.44	0.44
Feedback Reactivity Weighting	2.725	2.40	4.50	3.55
Trip Reactivity, %Δk	2.0	2.0	2.0	2.0
F_Q after rod ejection	16.07	13.0	19.2	18.7

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TABLE 5

REANALYSIS RESULTS FOR THE ROD
CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT
HOT ZERO POWER CASES

	Analysis Results		Design Limit[17]
Time In Life	BOL	EOL	
Maximum Fuel Pellet Average Temperature	3654	3772	-
Maximum Fuel Centerline Temperature, ^{o}F	4271	4381	-
Maximum Clad Average Temperature, ^{o}F	2672	2677	2700
Maximum Fuel Enthalpy (Cal/gm)	157	145	200
Fuel Pellet Melting, %	<10	<10	10

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TABLE 6

MAIN STEAM LINE BREAK RESULTS
INSIDE BREAK LOSS OF OFFSITE POWER CASE(D)

Parameter	Previous Analysis[2]	Cycle 2
Peak Core Average Power, %	13.70	6.92
Reactor Inlet Temp., Failed Loop, °F	310.8	310.8
Reactor Inlet Temp., Intact Loops, °F	521.2	521.2
Reactor Coolant Pressure, psia	913.8	913.8
Reactor Coolant Flow, % of Nominal	22.1	22.1
Minimum DNBR	>1.3	>1.3

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TABLE 7

REDUCTION IN DNBR RESULTING FROM FUEL ROD BOW*

	Full Flow Operation	N-1 Loop and Loss of Flow Transient
Maximum DNBR Penalty for 85% Closure	11.4%	14.0%
Generic Thermal Margin	9.1%	9.1%
Net DNBR Penalty for 85% Closure	2.3%	4.9%
DNBR Differential For Loss of Flow Transient		2.6%

*North Anna 1 and 2 are currently using Westinghouse 17 x 17 fuel assemblies

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FIGURE 1

NORTH ANNA UNIT 1, CYCLE 2 LOADING PAT[illegible] C2

	08	09	10	11	12	13	14	15
H	1 M12 13405	3 R09 8643	2 N10 16045	3 H15 11038	2 M11 16418	3 12 (H13) J15 8643	2 L12 16418	4 FRESH 0
J	3 R09 8643	2 N08 17885	3 M13 9867	2 J12 18290	3 K14 12979	2 J10 19006	4 16 (J14) FRESH 0	4 FRESH 0
K	2 N10 16045	3 N12 9867	2 K13 16045	3 L14 8956	3 L13 14859	2 K11 18517	4 FRESH 0	
L	3 R08 11038	2 M09 18290	3 P11 8956	2 J08 18653	3 J14 15221	4 16 (K13) FRESH 0	4 FRESH 0	
M	2 M11 16418	3 P10 12979	3 N11 14859	3 P09 15221	2 L08 18474	4 FRESH 0		
N	3 12 (N08) J15 8643	2 K09 19006	2 L10 18517	4 16 (N10) FRESH 0	4 FRESH 0			
P	2 L12 16418	4 16 (P09) FRESH 0	4 FRESH 0	4 FRESH 0				
R	4 FRESH 0	4 FRESH 0						

Batch	#F/A	Initial w/o U235
1	1	2.1
2	52	2.6
3	52	3.1
4	52	3.2

304 Depleted BP Rods

BATCH ---No. Depleted BP Rods
---BPRA Previous Cycle Location
---F/A Previous Cycle Location
---BOC2 Burnup (MWD/MTU)

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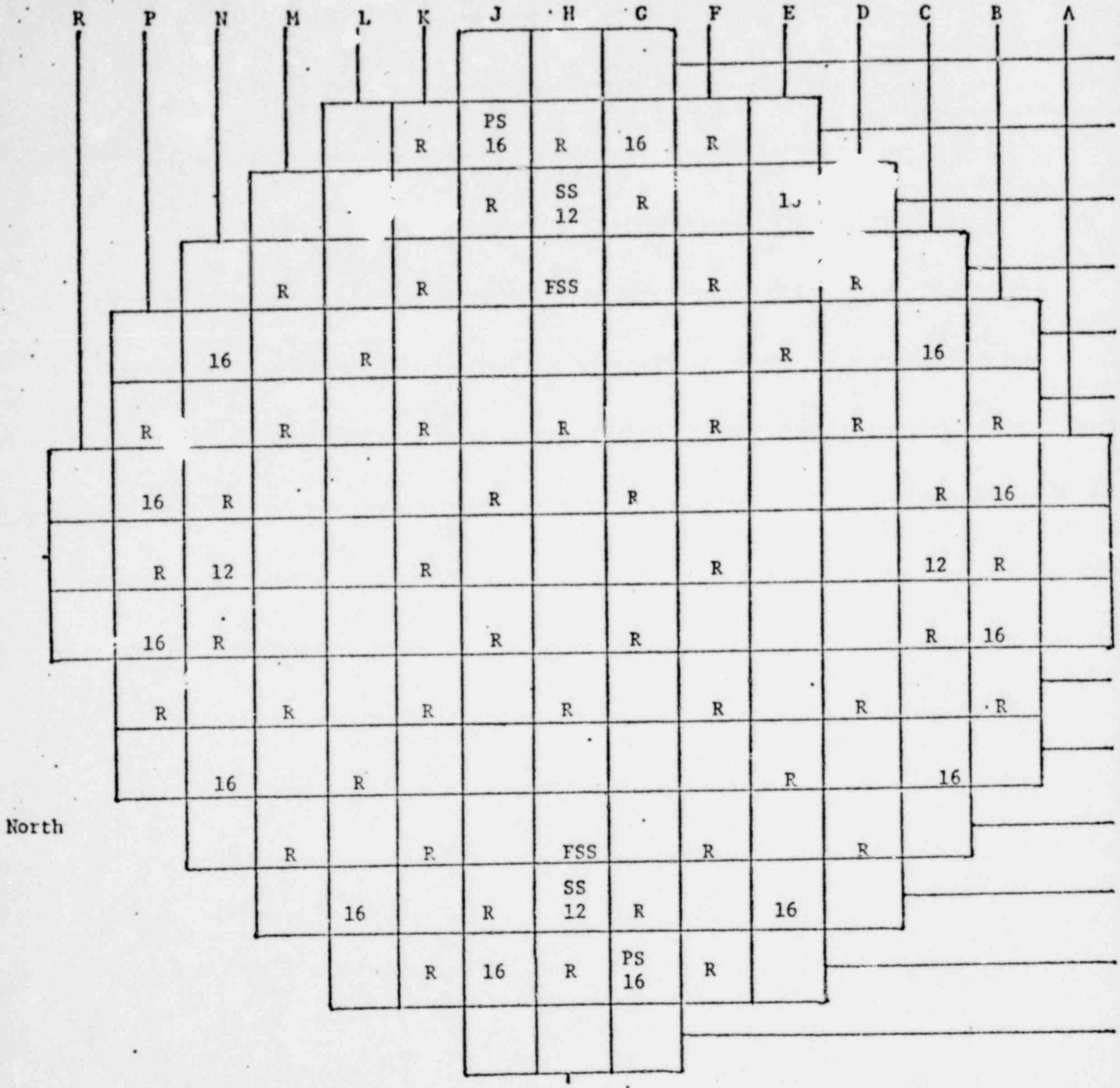
CONTROL ROD, BURNABLE POISON, AND SOURCE ASSEMBLY
LOCATIONS FOR NORTH ANNA UNIT 1, CYCLE 2



x - Number of Depleted Burnable Poison Rods
PS - Primary Source Location
SS - Secondary Source Location
FSS - Fresh Secondary Source
R - Rod Cluster Control Assembly

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FIGURE 3

HFP, ARO, BOC RADIAL POWER DISTRIBUTION FOR
NORTH ANNA UNIT 1, CYCLE 2 *

0.889 0.906							
1.160 1.240	0.968 1.006						
0.996 1.046	1.158 1.262	1.027 1.093					
1.105 1.174	0.964 1.031	1.189 1.286	0.959 1.037				
0.963 1.022	1.078 1.149	1.098 1.153	1.049 1.162	0.784 0.881			
1.141 1.231	0.955 1.034	0.949 1.016	1.171 1.310	0.713 1.092			
1.003 1.068	1.220 1.342	1.031 1.305	0.720 1.141				
0.892 1.171	0.720 1.133						

--------Assembly Average Relative Power

--------Peak Pin

* Based on a cycle 1 length of 15,500 MWD/MTU.

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FIGURE 4

HFP, D-BANK IN, BOC RADIAL POWER DISTRIBUTION FOR
NORTH ANNA UNIT 1, CYCLE 2 *

0.932 0.951							
1.189 1.263	0.948 1.005						
1.001 1.055	1.055 1.152	0.567 0.848					
1.140 1.228	0.960 1.036	1.136 1.323	1.010 1.076				
0.997 1.055	1.132 1.219	1.196 1.300	1.203 1.341	0.940 1.041			
1.028 1.075	0.945 1.022	1.050 1.157	1.374 1.529	0.869 1.317			
0.503 0.775	1.064 1.238	1.097 1.367	0.839 1.314				
0.657 0.833	0.597 0.925						

--------Assembly Average Relative Power
--------Peak Pin

* Based on a cycle 1 length of 15,500 MWD/MTU.

FIGURE 5

HFP, ARO RADIAL POWER DISTRIBUTION FOR
NORTH ANNA UNIT 1, CYCLE 2 AT 9600 MWD/MTU*

0.987 1.002							
1.194 1.250	1.031 1.060						
1.050 1.088	1.178 1.250	1.058 1.103					
1.143 1.201	1.016 1.063	1.182 1.246	0.994 1.053				
1.013 1.052	1.104 1.158	1.098 1.134	1.051 1.122	0.831 0.929			
1.123 1.177	0.969 1.016	0.950 1.005	1.101 1.209	0.733 1.061			
0.964 1.027	1.129 1.222	0.962 1.186	0.703 1.063				
0.849 1.092	0.697 1.047						

--------Assembly Average Relative Power
--------Peak Pin

* Based on a cycle 1 length of 15,500 MWD/MTU.

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FIGURE 6

HFP, D-BANK IN RADIAL POWER DISTRIBUTION FOR
NORTH ANNA UNIT 1, CYCLE 2 AT 9600 MWD/MTU

1.037 1.055							
1.231 1.278	1.014 1.075						
1.061 1.103	1.076 1.177	0.561 0.853					
1.186 1.265	1.016 1.083	1.124 1.283	1.045 1.105				
1.057 1.108	1.166 1.229	1.197 1.272	1.204 1.293	0.992 1.093			
1.022 1.099	0.966 1.052	1.050 1.138	1.289 1.406	0.888 1.267			
0.463 0.740	0.982 1.164	1.023 1.258	0.816 1.213				
0.617 0.756	0.575 0.845						

----------Assembly Average Relative Power
----------Peak Pin

* Based on a cycle 1 length of 15,500 MWD/MTU.

FIGURE 7

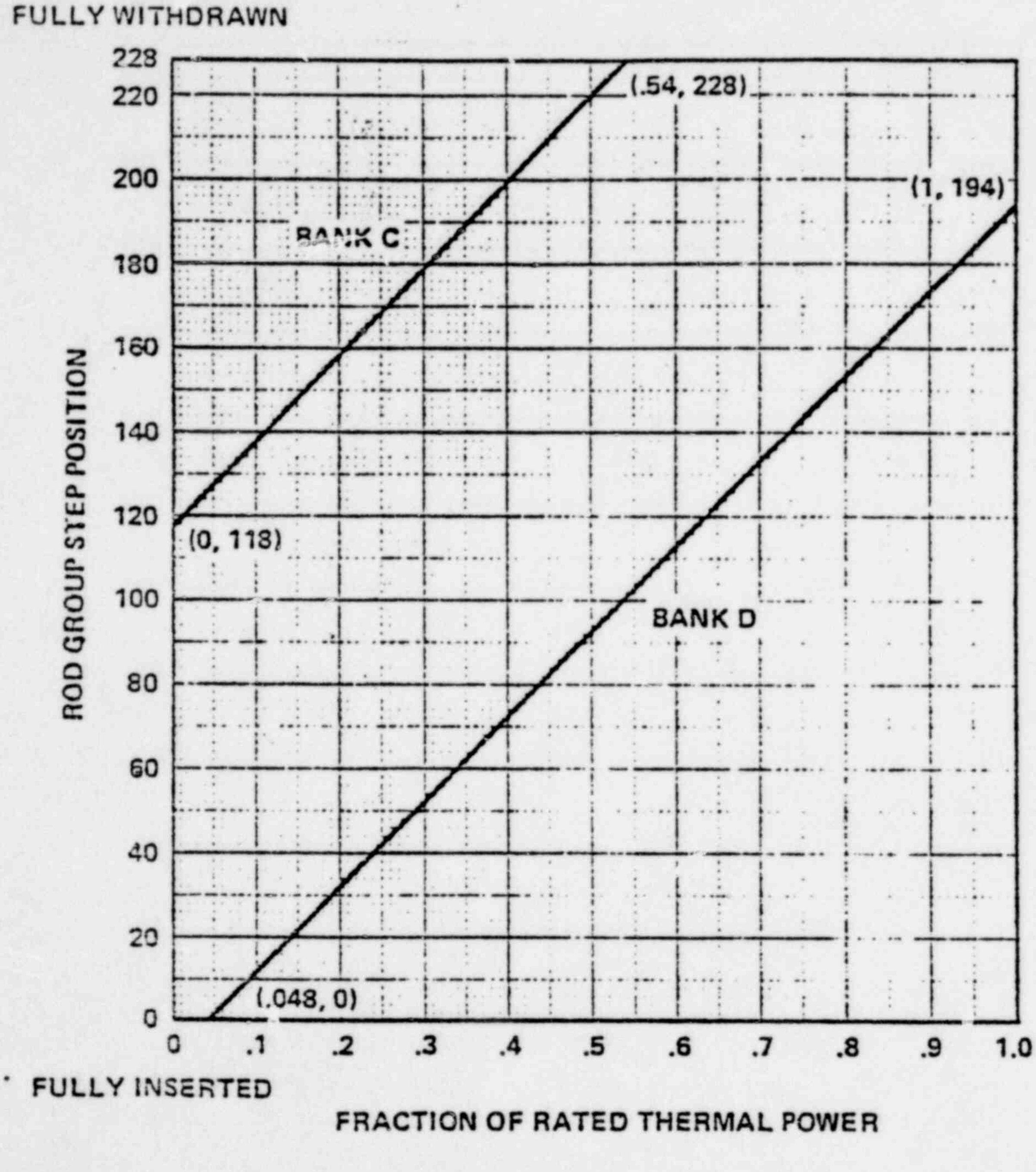

Figure 3.1-1 Rod Group Insertion Limits Versus Thermal Power

POOR ORIGINAL

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FIGURE 8

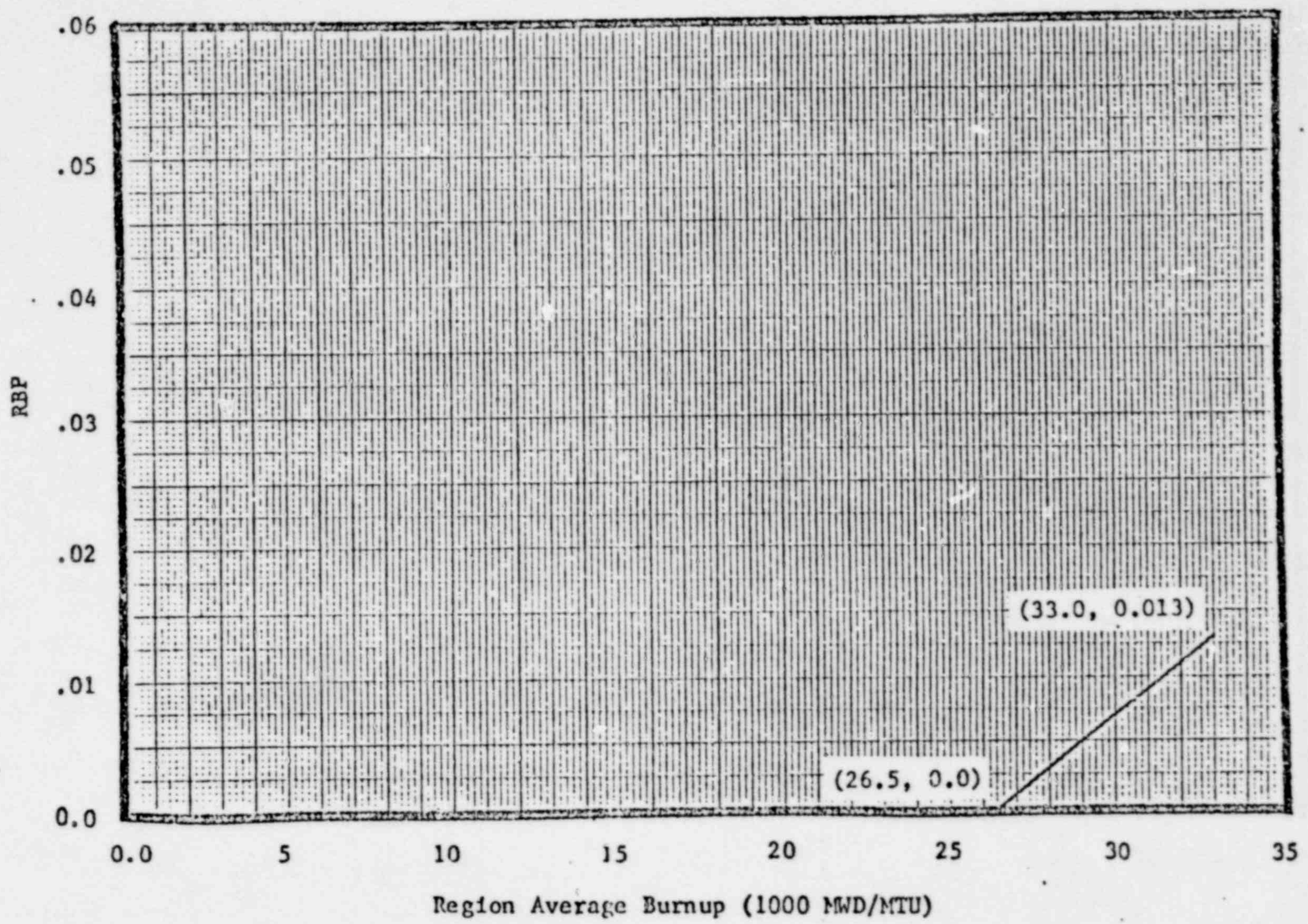

Figure 8 - Rod Bow Penalty Fraction Versus Region Average Burnup

CHANGE NO. 24 TO THE
TECHNICAL SPECIFICATIONS
NORTH ANNA POWER STATION
UNIT 1, CYCLE 2

VIRGINIA ELECTRIC AND POWER COMPANY

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ATTACHMENT 2

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{x}^{L}. at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods,

2. $F_{xy}^{RTP} \leq 1.65$ for all unrodded core planes from 0 to 65% of core height, and

3. $F_{xy}^{RTP} \leq 1.57$ for unrodded core planes above 65% of core height.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D".

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

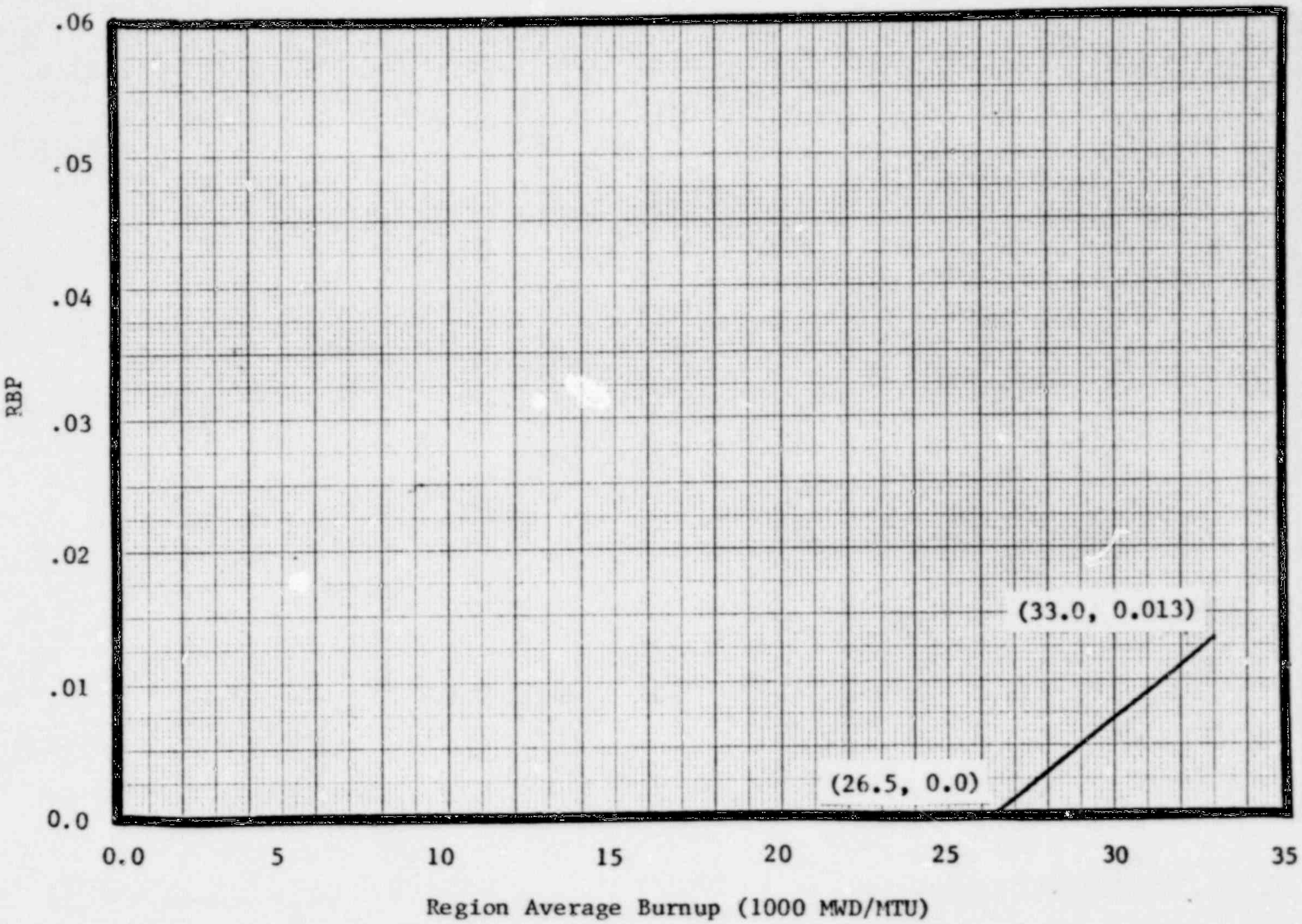

Figure 3.2-3 Rod Bow Penalty Fraction Versus Region Average Burnup

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