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PLANT NAME: H. B. Robinson Unit #2

ENCLOSURES:
Addl Info for Operation at 2300 MWt Core Power
(1) Rev & addl pgs & figs to the FSAR
(2) Cert of Sv.

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Carolina Power & Light Company

April 12, 1974

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Mr. John F. O'Leary
Directorate of Licensing
Office of Regulation
United States Atomic Energy Commission
Washington, D. C. 20545

H. B. ROBINSON UNIT 2 - DOCKET NO. 50-261
FACILITY OPERATING LICENSE DPR-23
ADDITIONAL INFORMATION FOR OPERATION AT 2300 MWt CORE POWER

Dear Mr. O'Leary:

Transmitted herewith are three signed originals and forty (40) copies of additional FSAR and Technical Specification page changes in support of our application for amendment of the H. B. Robinson Unit 2 license to allow operation at a core power level of 2300 MWt. The FSAR page changes incorporate information contained in WCAP-8243, "H. B. Robinson Unit 2, Justification for Operation at 2300 MWt," December, 1973, concerning overpower protection system limits based on excore surveillance including the effects of fuel densification. The page change to the Technical Specifications updates the Control Group insertion limits for three loop or two loop operation based on final design calculations for Cycle 3 operation.

As required by Commission Regulations, this submittal is signed under oath by a duly authorized officer of the Company.

Yours very truly,

J. A. Jones
Executive Vice President

JAJ/kr
Enclosure

3253

Sworn to and subscribed before me this 12th day of April, 1974.

Margaret M. Cox

My commission expires: July 4, 1975

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON UNIT 2

DOCKET No. 50-261

INSTRUCTION SHEET

This amendment contains information in support of CP&L's request for an Amendment to the Operating License to allow operation at 2300 MWt core power. The revised pages contain the results of the analyses done in support of 2300 MWt operation as well as other information important to the review.

Each revised page bears the date April, 1974, in the upper right hand corner. Vertical bars have been used in the margins of the revised pages to indicate the location of the revision on the page.

The following page removals and insertions should be made to incorporate these page changes into the FSAR.

REMOVE (Existing Pages)	INSERT (Amendment Pages)
3.2.1-5/3.2.1-6	3.2.1-5/3.2.1-6
3.2.1-7/3.2.1-8	3.2.1-7/3.2.1-8
---	3.2.1-8a
3.2.1-23	3.2.1-23
Figure 3.2.1-5/3.2.1-6	Figure 3.2.1-5/3.2.1-6

The following page removal and insertion should be made to incorporate this page change into the proposed Technical Specifications for 2300 MWt operation.

REMOVE	INSERT
---	Figure 3.10-1

Part Length Rods

The eight RCC assemblies with part length rods can be inserted into the core to control the axial power distribution. These assemblies do not drop when the reactor is tripped. The part length rods do not contribute to shutdown by themselves; they can, however, increase shutdown by flattening the power distribution at low power levels. No credit has been taken for this additional shutdown(3).

Xenon Stability Control

Part length control rods are provided to suppress xenon induced power oscillations in the axial direction, should they occur. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced power oscillations. Extensive analysis, with confirmation of methods by spatial transient experiments at Haddam Neck, has shown that any induced radial or diametral xenon transients would die away naturally. A full discussion of xenon stability control can be found in Reference 3. 4

Excess Reactivity Insertion Upon Reactor Trip

The control rod requirements have been based on providing one per cent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position or to prevent return to criticality following a credible steam-line break, whichever is the more limiting. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam break accident is considered.

Calculated Rod Worths

The complement of full length rods arranged in the pattern shown in Figure 3.2.1-1 meets the shutdown requirements. Table 3.2.1-3 lists the calculated worths of this rod configuration for beginning and end of the first cycle. In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed are decreased in the design by 10 per cent to account for any errors or uncertainties in the calculation. This worth is established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in operating reactor shows the calculation to be well within the allowed uncertainty of 10%.

Reactor Core Power Distribution

In order to meet the design objectives the peak to average power density must be within the limits set by the nuclear hot channel factors. For the peak power point in the core, as defined by the LOCA and overpower limit criteria, the nuclear heat flux hot channel factor, at rated power, F_q^N, was established as specified in Table 3.2.1-1, Line 18. For the hottest channel the nuclear enthalpy rise hot channel factors, $F_{\Delta H}^N$, was established as specified in Table 3.2.1-1, Line 19. The value of F_q^N includes the local power peaking due to fuel densification as shown in Figure 3.2.1-36. Variation in hot channel factors are illustrated for typical rod configurations in Figures 3.2.1-2 through 3.2.1-5. The calculations shown in these figures do not include the power flattening effect of equilibrium xenon and non-uniform Doppler broadening. Reactivity feedback effects associated with non-uniform xenon, water boron density and Doppler broadening are discussed in detail in Reference 3.

In core instrumentation will be employed to check the power distributions throughout core lifetime.

Eight part length rods, which are similar to the standard control rods but which have absorber material in only the bottom three feet, are located in the core as shown in Figure 3.2.1-1. The function of these rods is to shape the axial power distribution and to control axial xenon oscillations.

The control system for axial power distribution control is based on manual operation of the part length rods. Administrative procedures, alarm functions, and automatic rod stops, guide and monitor the operator in performing these tasks. The out of core nuclear instrumentation system supplies the necessary information for the operator to control the core power distribution within the limits established for the protection system design. This information consists of a two pen recorder for each long ion chamber which displays the upper and lower ion chamber currents and an indicator which gives the difference in these two currents for each long ion chamber. These ion chamber currents to the recorders and indicators are calibrated against in-core power distribution obtained from the movable detector system so that the eight individual signals are directly related to the power generated in the adjacent section of the core. This essentially divides the core into eight sections, four in the upper half and four in the lower half, and the operator manually positions the part length rods to maintain a prescribed relationship between the power generated in the upper and lower sections of the core.

The relationship between core power distribution and out of core nuclear instrumentation readings will be established during the startup testing program. In core flux measurements will be made over the range of relative positions between part length rods and the full length rod control banks for reactor power in the range of 25% to 100%. These measurements, together with long ion chamber currents, will be processed to yield the relationships between core average axial power generation, the axial peaking factor and axial offset as indicated by the out of core nuclear instrumentation. These relationships can be checked during operation to assess the effect of core burnup on the sensitivity between in-core power distribution and out of core readings.

A more detailed discussion of the background, analytical and experimental, data which forms the basis for this approach, is given in reference 3.

Local Power Density Limit Protection

The basis for protection against exceeding local power density limits which is not dependent upon incore surveillance of the core power density is given in topical reports[3,23] which describe the use of excore detector signals (axial offset) in the overpower protection logic. Since the magnitude of the local power peaking spike varies as a function of height in the core, it must be applied to the basic power shape data which determines the design value of the total power peaking factor F_q^T. F_q^T includes an engineering subfactor $F_q^E = 1.03$, which is defined in Section 3.2.2 and relates F_q^T to F_q^N.

An upper bound on F_q^T is set as a function of axial offset by consideration of the allowable operating situations. When F_q^T is increased locally by the height dependent local power spike, the individual points are increased by different amounts. The result is a revised plot of F_q^T vs. axial offset which requires a revised upper bound different in shape and magnitude from the previous upper bound.

For Cycle 3 of the H. B. Robinson 2, the limiting F_q^T, including the effects of the local power peaking of Figure 3.2.1-36, are given in Figure 3.2.1-6. The results presented also include a 5% factor for uncertainty and 3% for manufacturing tolerances. At reduced power, high local rod power could occur as a result of a large axial offset and its resultant high F_q^T. For this reason, the plot contains points at large axial offset evaluated for less than full power, with an F_{xy} appropriate to rodded planes as determined by the control rod insertion limits in the Technical Specifications. The values of F_{xy} for the plane of the peak power were assumed to be no lower than 1.435 in the unrodded and rodded planes independent of burnup. The data supports an F_q^T boundary as low as 2.50 as shown in Figure 3.2.1-6, which provides additional margin to the limiting F_q^T of 2.65 determined from the LOCA analysis in Section 14.3.2.

Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection System.

Moderator Temperature Coefficient

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to the coefficient and in a chemical shim core, the rods are only partially inserted. Also, the chemical poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component

18. J. M. Hellman (ed.) "Fuel Densification Experimental Results and Model for Reactor Application." WCAP 8219, October, 1973.

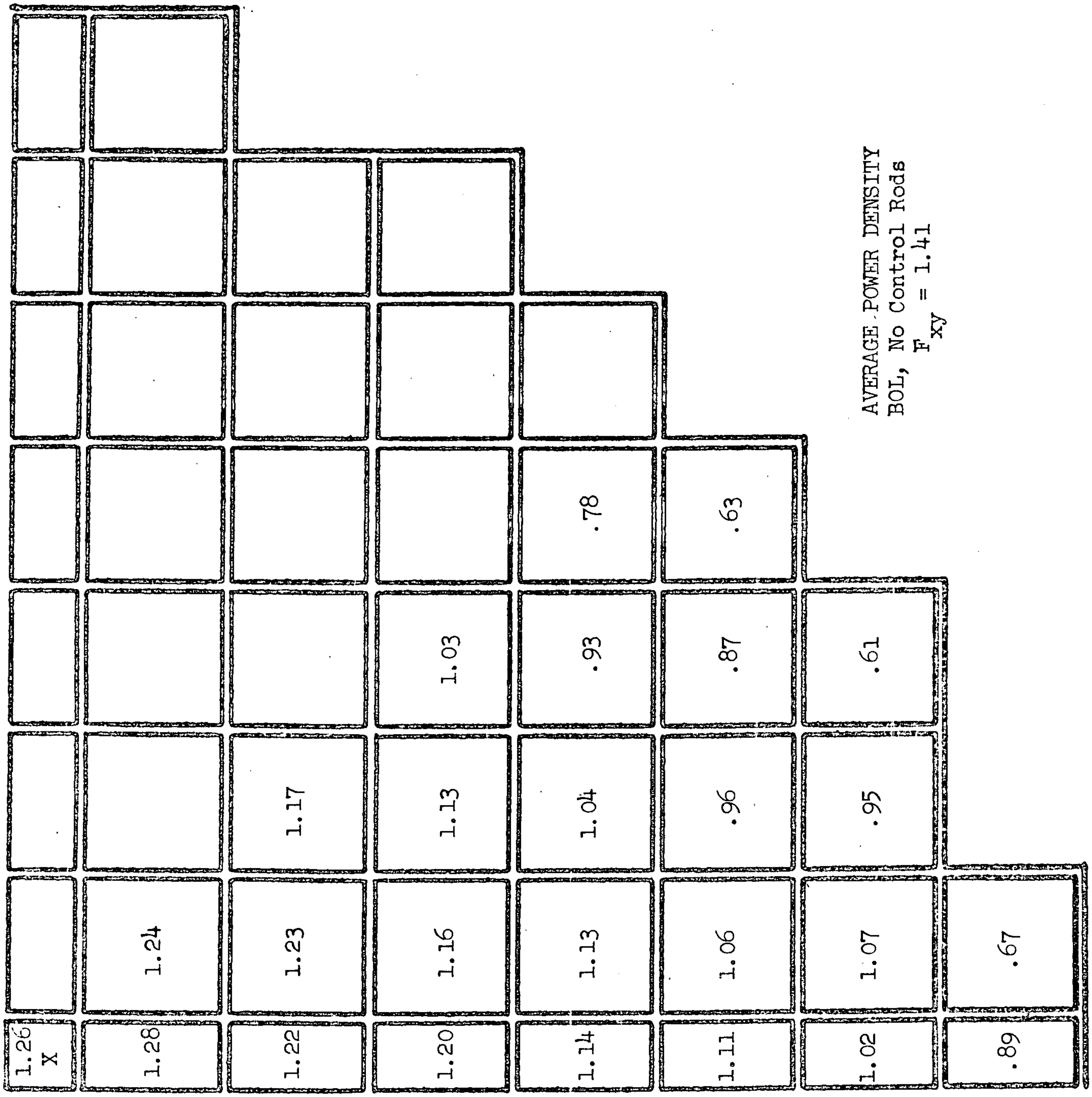
19. R. Salvatori, "Fuel Densification H. B. Robinson Steam Electric Plant, Unit No. 2, Cycle 2," WCAP 8114, April, 1973 (Proprietary).

20. R. Salvatori, "Fuel Densification H. B. Robinson Steam Electric Plant, Unit No. 2, Cycle 2," WCAP 8115, April, 1973 (Non-Proprietary).

21. D. W. Peacock, "H. B. Robinson Unit 2 Justification for Operation at 2300 MWt," WCAP 8243, December, 1973 (Proprietary).

22. D. W. Peacock, "H. B. Robinson Unit 2 Justification for Operation at 2300 MWt," WCAP 8243, December, 1973 (Non-Proprietary).

23. R. F. Barry, R. W. Brandon, J. P. Katz and R. J. Nath, "Power Distribution Monitoring in the R. E. Ginna PWR," WCAP 7542-L, September, 1970 (Westinghouse Proprietary Class 2).



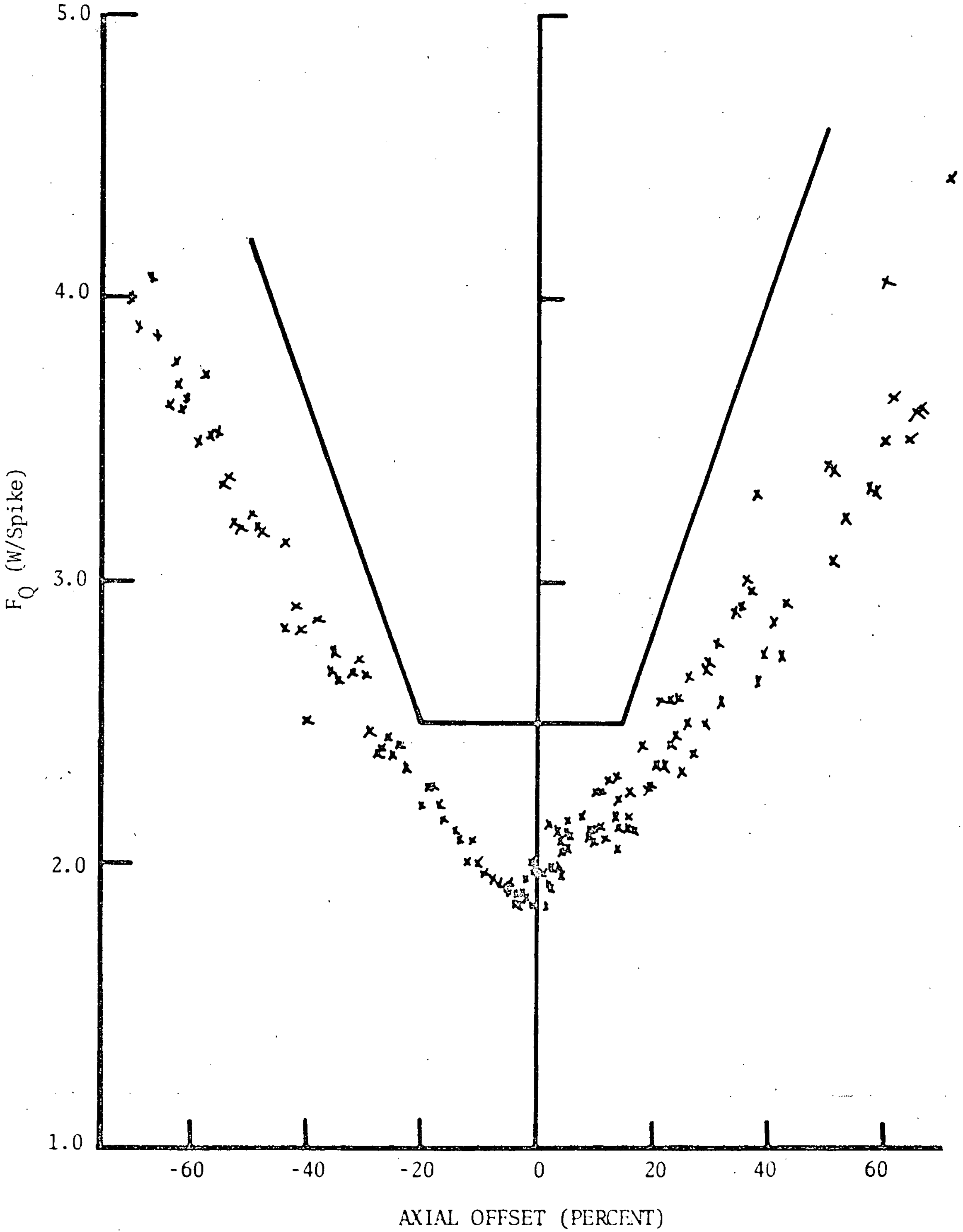
AVERAGE POWER DENSITY
BOL, No Control Rods
$F_{xy} = 1.41$

F_{XY} = $\frac{\text{Enthalpy Rise-Hot Channel Power Density}}{\text{Enthalpy Rise-Average Channel Power Density}}$

X = Location of Hottest Rod

AVERAGE POWER DENSITY (BOL), NO CONTROL RODS

FIGURE 3.2.1-5



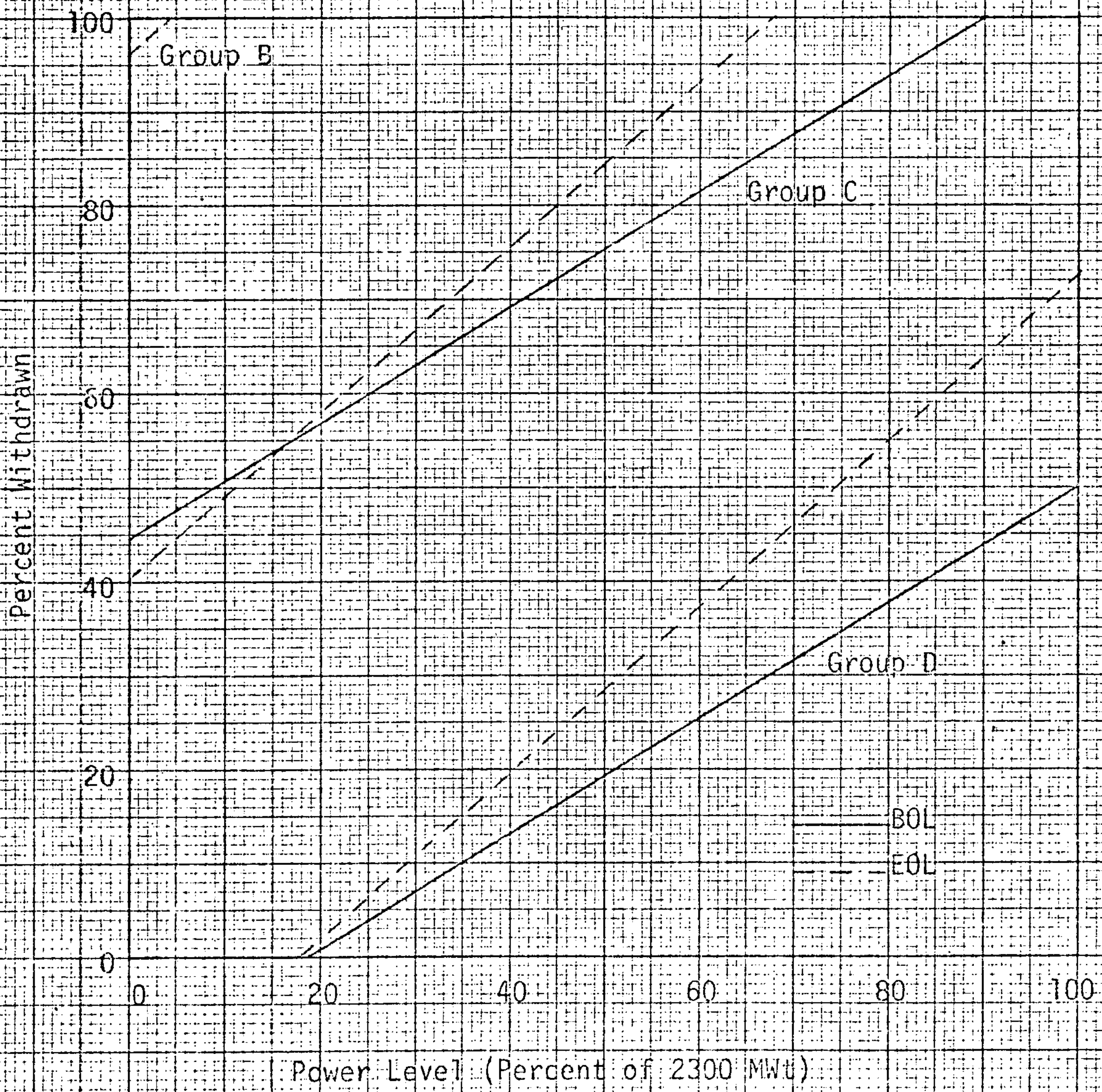
F_Q VERSUS AXIAL OFFSET

Figure 3.2.1-6

Figure 3.10-1

Control Group Insertion Limits for

Three Loop or Two Loop Operation



UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-261
	)	
H. B. Robinson, Unit 2	)	

CERTIFICATE OF SERVICE

This is to certify that a copy of additional proposed page changes to the Final Safety Analysis Report and the Technical Specifications in support of the Petition Requesting Amendment to Facility Operating License DPR-23 has this 12th day of April, 1974, been served upon the following by deposit of same in the United States mail, addressed as follows:

Mr. Harrell L. Gardner
Chairman - Darlington County Board of Commissioners
Route 2
Darlington, South Carolina 29532

W. Brian Howell

W. Brian Howell
Associate General Counsel
Carolina Power & Light Company

Business Address: 336 Fayetteville Street
Raleigh, N. C. 27602
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828-8211

Dated: April 12, 1974

3258