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DESCRIPTION:

Ltr re our 6-3-75 ltr furn responses to GETAB Questions .... trans the following:

ACKNOWLEDGED
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ENCLOSURES:

Appendix A- Justification For APRM Rod Block/ SCRAM Setting Change
Appendix B- modified pages for the Appendix A Tech- Specs
Appendix C- Typical BWR GETAB application Plant Info Request

PLANT NAME: Monticello

FOR ACTION/INFORMATION 7-16-75 JGB

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

July 10, 1975

DOCKETED USNRC JUL 12, 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

RECEIVED JUL 12 1975 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. D. L. Ziemann, Chief
Operating Reactors Branch # 2
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ziemann,

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Response to June 3, 1975 GETAB Questions

This letter is in response to your June 3, 1975 letter transmitting two questions concerning our GETAB Technical Specification change request. The questions with their respective answers are included below. The analysis of Cycle 5, which will commence this fall, will be based on the GETAB analysis discussed in our License Amendment Request Dated March 12, 1975. We therefore request that you schedule completion of your review and issuance of these changes accordingly.

Question No. 1. The APRM flux scram trip setting and rod block equations in Technical Specification 2.3 are the same as those used with the Hench-Levy CHF correlation. Either modify the equations or provide justification for retaining the equations based on critical heat flux rather than using units consistent with the GEXL/GETAB Analysis. The trip setting and rod block equation constants for the 8 x 8 fuel are different from those used by 7 x 7 fuel. Explain the difference.

Response. The present Technical Specification settings referenced are in terms of heat flux; the constants are the heat flux levels corresponding to the maximum peaking factor for 7 x 7 and 8 x 8 fuel respectively. As your question infers, it is appropriate that the settings be expressed in terms of peaking factor directly rather than extending the peaking factor to a heat flux value. We understand that you are also reviewing, on a generic basis, a change in the slope of the flow biased APRM scram and rod block settings. This subject is addressed in Appendix A. Since these changes effect four pages of our Technical Specifications we are including as Appendix B, the appropriate Technical Specification pages modified to reflect these changes, should you wish to effect them at this time. These pages would then serve as replacements and additions to Exhibit B of our License Amendment Request Dated March 12, 1975.

Question No. 2. Does the relative bundle to bundle power distribution used in the GETAB statistical analysis for Monticello remain fixed throughout the series of Monte Carlo simulations? If not, how does it vary, and why? Show that the peak radial power factor used is the worst case that could occur during the cycle.

Response. A series of questions on GETAB were answered in a January 20, 1975 letter from L. H. Frauenholz of General Electric to A. J. Ignatonis of the Nuclear Regulatory Commission. These answers were specific to Monticello, which was the lead plant GETAB analysis. The bundle to bundle power distribution used in the Monte Carlo simulation to derive the safety limit and operating limit on MCPR is identified and justified in the detail in that document. For your convenience, Pages 1-4 and Figures 3-1 and 4-1, which answer the above question, are attached.

Yours very truly,

M H Voth / for

L. O. Mayer, PE
Manager, Nuclear Support Services

LOM/MHV/deb

cc: J. G. Keppler
G. Charnoff
MPCA
Attn: J. W. Ferman

APPENDIX A

MONTICELLO

JUSTIFICATION FOR APRM ROD BLOCK/SCRAM SETTING CHANGE

To improve control rod operability in the core flow range above 20% of drive pump speed and below 50% of rated flow and to provide greater flexibility to accommodate situtations such as Xenon compensation during power ascensions with fixed control rod patterns, the APRM flow biased rod block (RB) and scram lines (S) should be redefined as shown below:

$$RB = 0.58\ W + 50$$

$$S = 0.58\ W + 62$$

Where W is the recirculation loop flow in percent of rated flow.

These revised lines were selected on the following bases:

1. The rod block lines intercept the rated power flow control line at natural circulation, thus raising the rod block set point by about 10% at low flows. This APRM rod block setting will provide a reasonable range for maneuvering during plant startup.

2. As part of the original design requirements, stability calculations were performed on the natural circulation line at 50% and 62% power (the new intercepts of the rod block and scram lines).

3. At the rated flow conditions, the rod block and scram lines remain at 108% and 120% of rated power.

Discussion

Stability calculations for Monticello have been performed and are presented in Section 7.2 of the FSAR (1) and were recalculated in conjunction with the Third Reload Submittal (2). Figure 7-2-10 of the FSAR and Figure 6-12 of the Third Reload Submittal show that the system is stable at the intersection of natural circulation flow and the nominal expected flow control line which is the new limit of the rod block line. The least stable analyzed point is the 62% power-natural circulation flow point which is still extremely stable and bounds the system stability as the least stable point. Thus, while raising the APRM rod block and flow biased scram allows operation in regions of reduced stability margin there is clearly no threat to the ultimate stability limit of 1.0 decay ratio.

Analyses reported in the Third Reload Submittal and the FSAR, Section 7 demonstrate that with the scram trip set at 120% of rated power none of the abnormal operational transients analysed violate the fuel safety limit and there is substantial margin from fuel damage. Credit is not taken in these analyses for the APRM flow-biased scram. Thus, any setting for the APRM flow-biased scram provides additional margin from the Fuel Cladding integrity Safety Limit.

Because GETAB analyses have not taken credit for the APRM flow biased scram, the derived operating limit MCPR is also valid for any APRM scram less than 120 percent of rated power. As with MCHFR, the worst case MCPR occurs at 100% of rated power so that maintaining the 108% rated power rod block setting assures no decrease in operating margin.

(1) Monticello Nuclear Generating Plant, Final Safety Analysis Report, DAR-22, Docket 50-263.

(2) Northern States Power Company, Monticello Nuclear Generating Plant. Third Reload Submittal, December 11, 1974.

APPENDIX B

Appendix B, attached, consists of modified pages for the Appendix A Technical Specifications as listed below. These pages incorporate the changes discussed in the letter.

Page 6
7
11
12

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Core Thermal Power Limit (Reactor Pressure > 800 Psia and Core Flow is > 10% of Rated)

When the reactor pressure is > 800 Psia and core flow is > 10% of rated, the existence of a minimum critical power ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

2.3 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM -- The APRM flux scram trip setting shall be as shown in Figure 2.3.1 and shall be:

$$S \leq 0.58\ W + 62\ \%$$

where:

S = Scram setting in percent of rated thermal power

W = Percent of design recirculation driving flow

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 Psia or Core Flow $\leq$ 10% of Rated)

When the reactor pressure is $\leq$ 800 Psia or Core Flow is $\leq$ 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transients

To insure that the safety limit established in Specification 2.1.A is not exceeded, each required scram shall be initiated by its primary source signal as indicated by the plant process computer.

In the event of operation with a total peaking factor (PF) greater than the design peaking factor (DPF), the setting shall be modified as follows:

$$S \leq (0.58\ W + 62\ \%)\ \frac{DPF}{PF}$$

where:

DPF = 3.08 for 7 x 7 fuel
= 3.04 for 8 x 8 fuel

2. IRM--Flux Scram setting shall be $\leq$ 20% of rated neutron flux

B. APRM Rod Block--The APRM rod block setting shall be shown in Figure 2.3.1 and shall be:

$$RB \leq 0.58\ W + 50\%$$

where:

RB = Rod Block setting in percent of rated thermal power
W = Percent of design recirculation driving flow

In the event of operation with a total peaking factor (PF) greater than the design peaking factor (DPF), the setting shall be modified as follows:

$$RB \leq (0.58\ W + 50\ \%)\ \frac{DPF}{PF}$$

where:

DPF = 3.08 for 7 x 7 fuel
= 3.04 for 8 x 8 fuel

C. Reactor Low Water Level Scram setting shall be $\geq$ 10'6 above the top of the active fuel.

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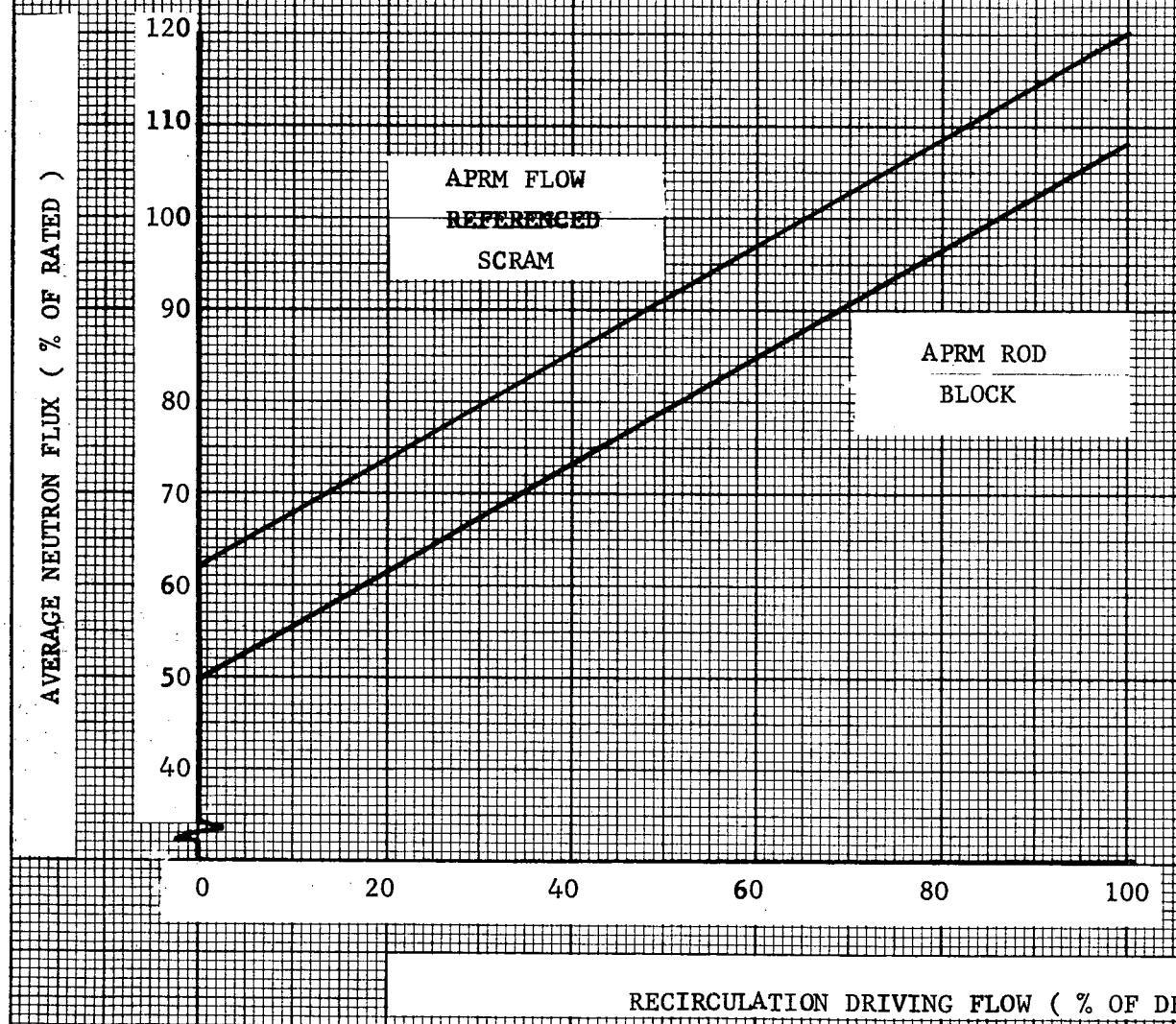
FIGURE 2.3.1
APRM FLOW REFERENCED
SCRAM AND ROD BLOCK
TRIP SETTINGS



NOTES:

1. APRM SCRAM SETTING SHALL BE:
   $S \leq 0.58\ W + 62\ \%$
2. APRM ROD BLOCK TRIP SETTING SHALL BE:
   $RB \leq 0.58\ W + 50\ \%$
3. 100 % NEUTRON FLUX IS EQUIVALENT TO 1670 MWT
4. W = PERCENT OF DESIGN RECIRCULATION DRIVING FLOW.

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TYPICAL BWR GETAB APPLICATION

Plant Information Request

1. QUESTION:

Identify the scram reactivity curve used to calculate the system response following a turbine trip at the end of the equilibrium cycle. Is it the design basis curve ("D" curve)?

RESPONSE:

The scram reactivity curve applied is the Monticello End-of-Cycle 4 "C" curve (EOC-C), multiplied by the Design Conservative Factor (0.8). The EOC4-C curve reflects the expected Cycle 4 End of Cycle scram reactivity function.

2. QUESTION:

Provide the nominal values of the following parameters used in the derivation of MCPR=1.06 and 1.36: R factor, axial power shape, axial and local peaking factors, radial peaking factor, nonfuel power fraction, average power density, bundle power, bundle flow, and inlet enthalpy. Justify the parameters selected.

RESPONSE:

The following list summarizes the requested information.

1. R-factor = 1.102. This value is the highest expected R-factor during the operating cycle.

2. Axial Power Shape:

APF	NODE		APF	NODE	
0.38	1	(Bottom)	1.44	13	
0.45	2		1.51	14	
0.50	3		1.55	15	
0.53	4		1.57	16	
0.60	5		1.55	17	
0.65	6		1.50	18	
0.70	7		1.42	19	
0.79	8		1.28	20	
0.88	9		1.10	21	
1.00	10		0.90	22	
1.16	11		0.69	23	
1.34	12		0.51	24	(Top)

2. RESPONSE:

2. Axial Power Shape: (Continued)

The above design axial power shape yields a conservative representation of the conditions which might exist during an operating cycle as discussed further in NEDO-10958 Appendix 1.

3. Axial Peaking Factor = 1.57 at node 16 as shown above.

4. Local Peaking Factor = 1.22 (design value, but not used directly in GETAB analyses.)

5. Radial Peaking Factor = 1.47. This design value is larger than the highest expected bundle radial peaking factor; therefore, it establishes a conservative upper limit.

6. Nonfuel Power Fraction = 0.035 (Design value, but not critical to GETAB evaluations.

7. Average Power Density = 40.6 KW/L (Design value, but not used directly in GETAB analyses.

8. Bundle Power = 4.962 (Based on the design core power and the maximum radial peaking factor discussed in item 5.)

9. Bundle Flow = 1.065×10^5 Lbs/hr (This value is the steady-state flow in the fuel bundle operating at the highest bundle power.

10. Inlet Enthalpy = 523.0 Btu/Lb (Design value determined from a reactor heat balance operating with rated core flow, 100% power and maximum feedwater temperature.)

3. QUESTION:

Identify the relative bundle to bundle power distribution (by histogram method) that is employed in the application of the GETAB statistical analysis to Monticello.

RESPONSE:

The histogram of relative bundle power used in GETAB statistical analysis for Monticello is shown in Figure 3-1. This histogram was generated by arranging the control rod pattern such that the maximum number of fuel assemblies were placed at positions of minimum MCPR as described further in NEDO-10958 Appendix IV.

4.a QUESTION:

What is the relative bundle power distribution of the Monticello plant at the worst time of the fuel cycle? Is it the same as the one used in the statistical rod boiling transition analysis. If not, justify why it is not used in the derivation of MCPR values.

RESPONSE:

Because the power distribution described in the answer to Question 3 is selected and forced to be conservative (NEDO-10958 Appendix IV-4), the worst distribution for Monticello during its fuel cycle is not expected to be as severe as the distribution used in the analysis.

To illustrate the conservatism inherent in the selected power distribution, comparative analyses were performed using an actual operating power distribution from a typical BWR/4. The analysis showed that the 99.9% limit is met with an MCPR 0.05 lower than the value derived with the conservative power distribution. Hence, the selected power distribution is indeed very conservative.

4.b QUESTION:

Provide a Monticello bundle power histogram (calculated result) in order to see that the predicted Monticello case is not as severe as that used in GETAB analyses and reported in the response to 4.a, above.

RESPONSE:

The requested histogram is shown in Figure 4-1. The case chosen is an end-of-cycle condition which typically exhibits highest radial peaking. Comparison with that used for GETAB analysis (question 3) will show that the Histogram used in GETAB analyses is skewed more to the right than the calculated histogram. Therefore, in the response to question 4a above, it can be concluded that the Histogram used for GETAB analyses is indeed conservative for Monticello.

5. QUESTION:

The TIP uncertainty of 8.7% is presumably based on a symmetric, reload core, and LPRM extrapolated TIP data (based on the information shown in NEDO-20340). Justify the assumption of power symmetry for the Monticello reload core.

RESPONSE:

The 8.7% uncertainty accounts for no identified physical assymmetry in the Monticello core. If some assymmetry exists, it has not been detected and is therefore not taken into account. The 8.7% relates only to TIP assymmetry.

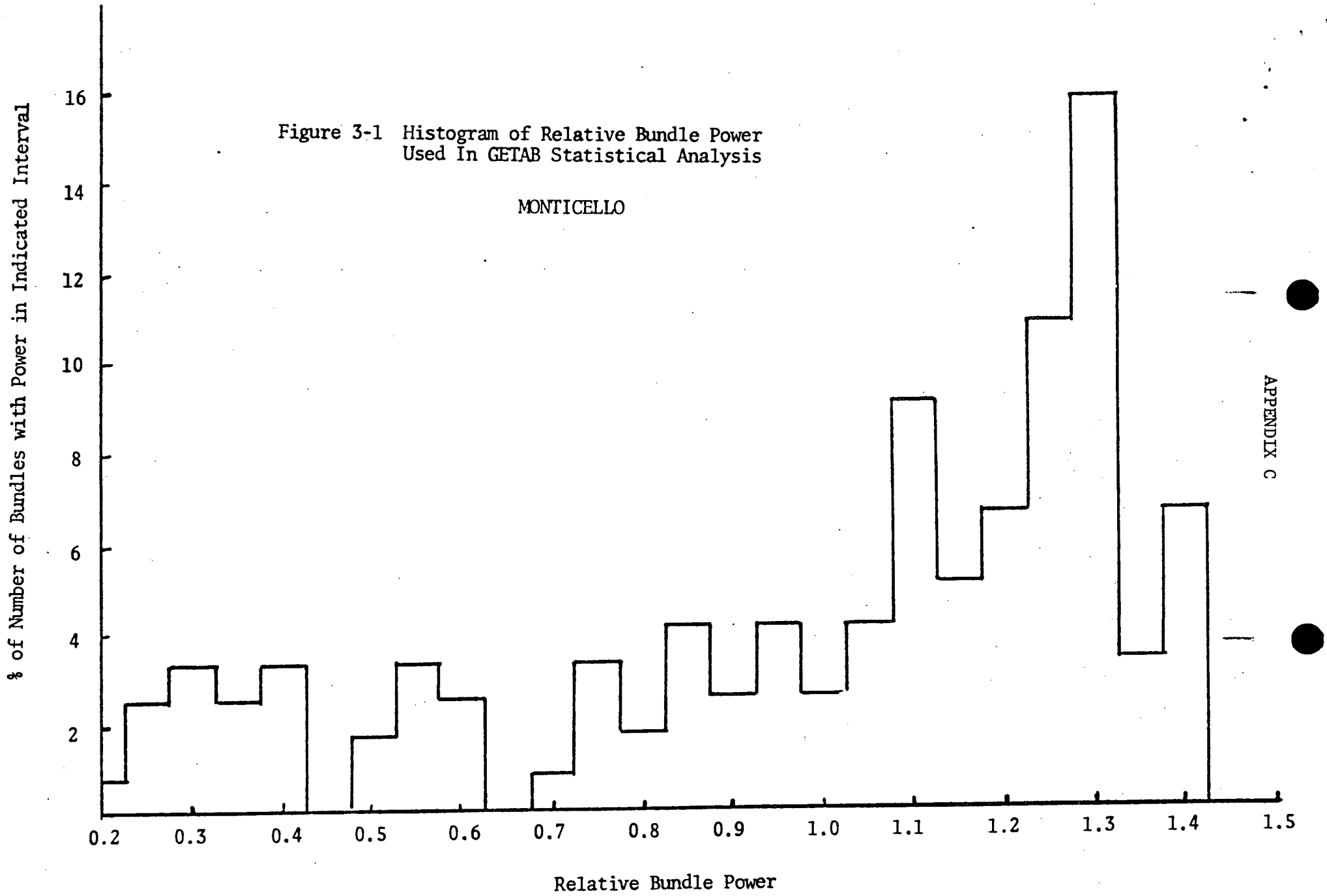
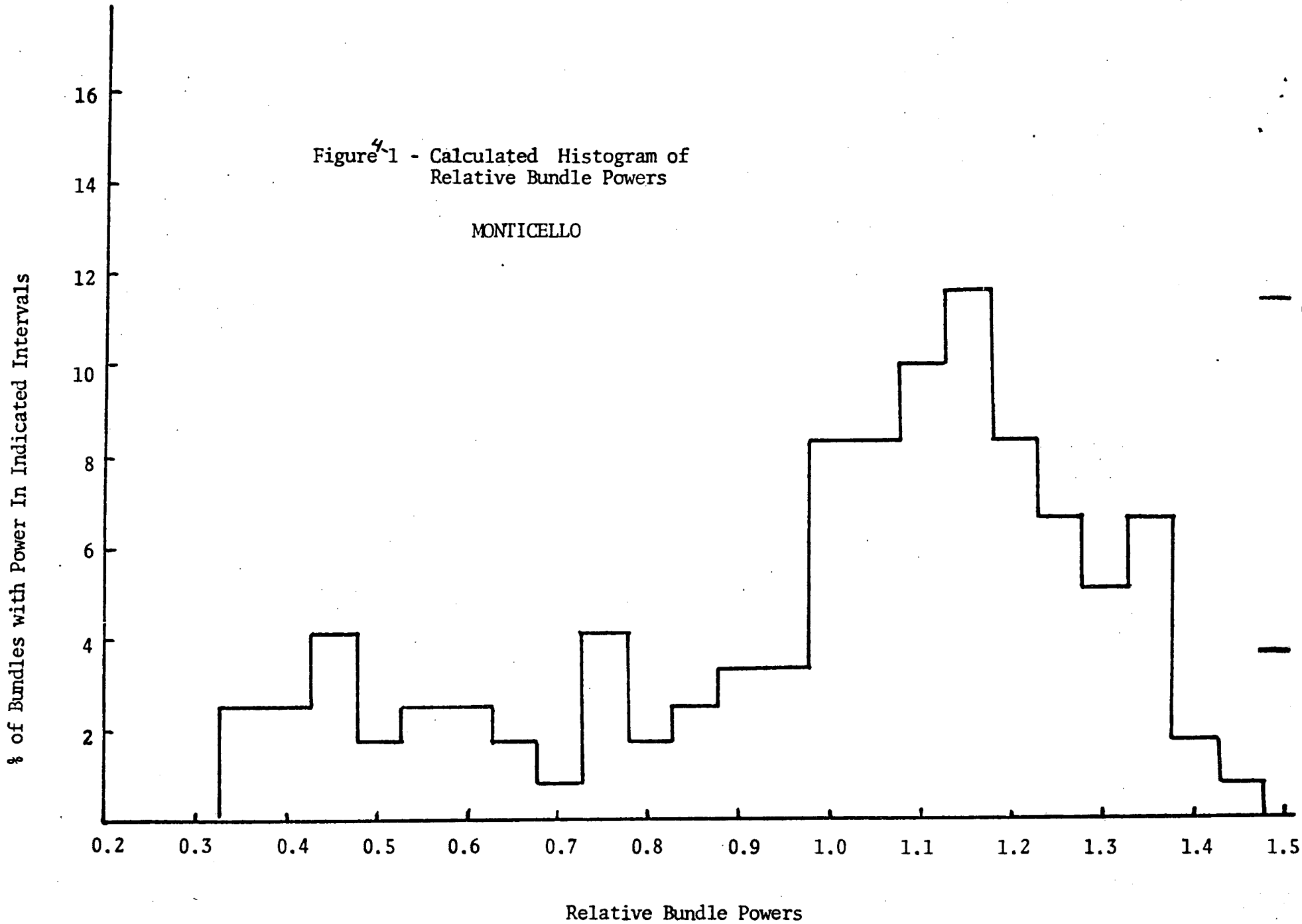


Figure 3-1 Histogram of Relative Bundle Power Used In GETAB Statistical Analysis

MONTICELLO

Figure 4-1 - Calculated Histogram of Relative Bundle Powers

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