JUN 29 1973

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Edward J. Nelson, President
89 East Avenue
Rochester, New York 14604

Change No. 9
License No. DPR-18

Gentlemen:

Your letter dated June 12, 1973, submitted proposed changes to the Technical Specifications for the R. E. Ginna Nuclear Power Plant. These changes are based on the fuel densification evaluation submitted by you on February 16, 1973, and entitled "Fuel Densification - R. E. Ginna Nuclear Power Plant Unit 1 - Cycle 3", (Westinghouse Report WCAP-8059, non-proprietary). The analysis uses the same methods previously presented for Point Beach Unit 1. This includes the effects of expected clad flattening in Region 4A of the Cycle 3 core. You have concluded from the analysis that full power operation (1520 MWt) to a fuel exposure of 7×10^5 MWD is justified with appropriate provisions made in the Technical Specifications for taking into account the effects of fuel densification and clad collapse.

In our letter to you dated November 8, 1972, we concluded that operation of the R. E. Ginna Plant for Cycle 3 with the restrictions of our June 23, 1972 letter and Table 1 of your June 22, 1972 submittal, as modified by our November 8, 1972 letter, provided an acceptable margin of safety until the fuel densification evaluation was completed.

We have now completed our review of the report on fuel densification referred to above. In order to assure safe operation with collapsed fuel rods, you have:

1. limited the clad temperature in collapsed sections of a fuel rod to less than 1800°F during a postulated loss-of-coolant accident;

2. adequately allowed for the power spike that will result in rods adjacent to collapsed rods;

3. included an additional 1.9% decrease in DNBR to account for increased pellet clad eccentricity and reduced fuel rod circumference and heat transfer area;

4. included a 10% penalty applied at the point of minimum DNBR to conservatively account for possible contact of rods due to flattening and bowing;

OFFICE ▶

5. changed the reference axial power distribution for DNB analysis from a chopped cosine with a 1.64 peak to average power to one with a 1.55 peak to average power; and

6. reduced the total heat flux hot channel factor, F_Q, in Region 4A, the only region predicted to collapse during Cycle 3, from 2.80 to 2.54.

On the basis of the above and our review of your report, we have determined that the effects of fuel densifications have been adequately analyzed and that the plant can be operated at 100% of rated power (1520 MWt) with appropriate changes to the Technical Specifications.

We have made some modifications and some additions to the Technical Specifications changes that you proposed.

Changes to the Technical Specifications have been made to Section 1 to add the definitions of hot channel factors; to Section 2.1 to limit fuel residence time to 7×10^5 MWD; to Section 2.3 to revise the overtemperature constants; to Section 3.1 to limit primary to secondary steam generator leakage so that in the event of an overpower transient and failure of all flattened fuel rods the radiation exposure at the site boundary will not exceed the limits of 10 CFR Part 20; to Section 3.10 to incorporate revised control rod insertion limits, power distribution limits, and additional power distribution surveillance requirements; and to Section 4.1 to clarify the requirement for control rod drive surveillance.

In addition, we are taking this opportunity to up-date your Technical Specifications in another area.

Recent tests by the AEC Respiratory Studies Group at LASL of full facepiece respirators operating in the demand mode as currently approved by the U. S. Bureau of Mines have indicated that the protection factor for particulates and vapors and gases is considerably less than the expected factor of 500. As a result of these tests, technical specifications containing allowances for the use of respiratory protective equipment shall be changed to reflect a protection factor of no more than 100 for full facepiece respirators operating in the demand mode for atmosphere-supplying respirators, both airline and self-contained breathing apparatus.

Also, your procedures for determining protection factors afforded by each of the respirator types currently in use at your facility should be reviewed to determine that the protection afforded the individual

wearing such respiratory equipment is at least the protection factor assigned the particular respiratory equipment by your technical specifications. If the actual protection factor is less than that given in your technical specification, you should take corrective action to either repair your respiratory equipment, replace the defective equipment, or reduce the allowable protection factor assigned to such respiratory equipment and reevaluate the exposures to any individual that used such equipment in plant activities with an assigned protection factor greater than the actual value.

We conclude that all the changes to the Technical Specifications described above do not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Provisional Operating License No. DPR-18 are hereby changed as indicated in Attachment A.

The Interim Conditions for Operation of the R. E. Ginna Power Plant Unit No. 1, Attachment A to our letter dated June 23, 1972, and the modification of those conditions contained in our letter dated November 8, 1972, are superseded by this change to the Technical Specifications.

Sincerely,

Original signed by
Robert J. Schemel for

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Attachment A

cc w/enclosure:
see next page

JUN 29 1973

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Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43—16—81465-1 445-678

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

June 29, 1973

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ATTN: Mr. Edward J. Nelson, President
89 East Avenue
Rochester, New York 14604

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Sincerely,

Robert J. Schemel, for

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Attachment A

cc w/enclosure:
see next page

cc w/enclosure:
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ATTACHMENT A

CHANGE NO. 9 TO THE TECHNICAL SPECIFICATIONS

R. E. GINNA NUCLEAR POWER PLANT UNIT NO. 1

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

SECTION 1.0

Add the following new paragraph:

"1.11 Hot Channel Factors

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density assuming nominal fuel pellet and rod dimensions.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the ratio between F_Q and F_Q^N and is the allowance on heat flux required for manufacturing tolerances.

$F_\Delta^N H$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power."

SECTION 2.1

Specification

Number the paragraph "1."

Add the following new paragraph:

"2. The fuel residence time for Cycle 3 shall be presently limited to 7×10^5 MWD under design operating conditions. The licensee may propose to operate the core in excess of 7×10^5 MWD by providing an analysis which includes the effect of further clad flattening or a change in operating conditions. Any such analysis, if proposed, shall be approved by the Regulatory staff prior to operation in excess of 7×10^5 MWD."

Basis

Replace the third paragraph (starting on page 2.1-3) and the first sentence of the fourth paragraph (page 2.1-4) with the following:

"The safety limit curves allow for heat flux peaking effects due to fuel densification and flattened fuel cladding sections.

"Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length, as well as a penalty to account for rod bowing, have been included in the calculation of the curves shown in Figure 2.1-1.

"These curves are based on an $F^N_{\Delta H}$ of 1.66, a 1.55 cosine axial flux shape, and a DNB analysis as described in Section 4.3 of WCAP-8058, "Fuel Densification, R. E. Ginna Nuclear Plant Unit 1, Cycle 3" (including the effects of fuel densification and flattened cladding).

"Since it is possible to have somewhat greater enthalpy rise hot channel factors at part power than at full power due to the deeper control bank insertion which is permitted at part power, a conservative allowance has been made in obtaining the curves in Figure 2.1-1 for an increase in $F^N_{\Delta H}$ with decreasing power levels."

Add the following new paragraph to the Basis.

"The fuel residence time for Cycle 3 is limited to 7×10^5 MWD to assure no additional fuel clad flattening without prior review by the Regulatory staff. Prior to 7×10^5 MWD, the licensee may provide the additional analyses required for operation beyond 7×10^5 MWD."

Replace Figure 2.1-1 with the attached revised figure.

SECTION 2.3

Paragraph 2.3.1.2

Replace subparagraphs d. and e. with the following:

"d. Overtemperature ΔT

$$\leq {}_{\Delta}T_o \left[K_1 + K_2(P-P^1) - K_3(T-T^1) \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right) - f(\Delta I)\right]$$

where: ${}_{\Delta}T_o$ = indicated ΔT at rated power, °F
T = average temperature, °F
T^1 = 573.5°F
P = pressurizer pressure, psig
P^1 = 2235 psig
K_1 = 1.10

K_2 = 0.000789
K_3 = 0.0123
τ_1 = 25 sec
τ_2 = 5 sec

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power, such that:

(i) for $q_t - q_b$ within -15, +5 percent, f (ΔI) = 0

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip set point shall be automatically reduced by an equivalent of two percent of rated power.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -15 percent, the ΔT set point shall be automatically reduced by an equivalent of 2.5 percent of rated power.

e. Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5 (T-T^1) - K_6 \frac{\tau_3 S T}{\tau_3 S + 1} - f(\Delta I)\right]$$

where: ΔT_o = indicated ΔT at rated power, °F
T = average temperature, °F
T^1 = indicated T_{avg} at nominal conditions at rated power, °F
K_4 = 1.11
K_5 = 0.0 for $T<T^1$
= 0.002 for $T \geq T^1$
K_6 = 0.0262 for increasing T
= 0.0 for decreasing T
τ_3 = 10 sec
f(ΔI) = as defined in 2.3.1.2.d."

Basis

Replace the fourth sentence of the first paragraph with the following:

"The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips."

Second paragraph, fourth sentence - replace the words "a power density of 18.5 kW/ft as described in Sections 7.2.3 and 14.1.2 of the FSAR" with the following words:

> "a value at which fuel pellet centerline melting would occur as described in Section 7.2.3 of the FSAR and in WCAP-8058, "Fuel Densification, R. E. Ginna Nuclear Power Plant Unit 1, Cycle 3."

SECTION 3.0

Paragraph 3.1.5.2 - replace with the following:

> "3.1.5.2 Steam generator tube leakage in any one steam generator shall not exceed the limit derived from Figure 3.1-4 when averaged over a period of 24 hours. If this limit is exceeded, the reactor shall be shut down within 8 hours."

Basis of paragraph 3.1.5.2.

Replace next to last paragraph with the following:

> "Steam generator tube leakage limits are based upon offsite dose considerations as limited by 10 CFR Part 20 in the event of an overpower transient with the presence of collapsed rods and 10 CFR Part 100 limits in the event of a steam line break or rod ejection accident."

Add the attached new Figure 3.1-4 entitled "Maximum Steam Generator Leak Rate versus Time."

Paragraph 3.10.1.3 - replace the last sentence with the following:

> "After the boron concentration has been reduced to 600 ppm, the limits may be adjusted to the end of core life values as shown by the dotted lines in Figure 3.10.1. Furthermore, the control rod banks are moved sequentially with a 100 (±5)-step overlap between successive banks."

Add the following new paragraph:

> "3.10.1.5 The part length control rods will not be inserted except for physics tests or for axial offset calibration performed at 75% power or less."

Replace paragraph 3.10.2.2 with the following:

"3.10.2.2 Power distribution limits are expressed as hot channel factors. Limiting values at rated power are:

F_Q^N (Core) = 2.74

F_Q^N (Region 4A) = 2.47

$F_\Delta^N H$ = 1.66

To obtain F_Q^N from the measurement, the measured value of F_Q^N shall be multiplied by the peaking penalty as shown in Figure 3.10-3 (at the appropriate core location) to account for fuel densification and potential clad flattening. If the measured F_Q^N and $F_\Delta^N H$ exceed the limiting values with due allowance for measurement error, the maximum allowable reactor power level and the Nuclear Overpower Trip set point shall be reduced by one percent for each percent which $F_\Delta^N H$ or F_Q^N (for the core or Region 4A) exceed the limiting values, whichever is most restrictive. If the hot channel factors cannot be reduced below the limiting values within one day, the Overpower ΔT trip set point and the Overtemperature ΔT trip set point shall be similarly reduced."

Add the following new paragraphs:

"3.10.2.6 Except for physics tests or axial offset calibration if the indicated axial flux difference where flux difference is $q_t - q_b$ as defined in Section 2.3.1.2 is outside the range +5 percent to -15 percent, then within two hours:

(a) Correct the situation, or

(b) Determine by measurement the hot channel factors and apply Specification 3.10.2.2, or

(c) Limit power to 2 percent below rated power for each 1.0 percent the indicated axial flux difference is greater than +5 percent or to 2.5 percent below rated power for each 1.0 percent the indicated axial flux difference is more negative than -15 percent."

"3.10.2.7 When operating above 75% of rated power, a power distribution map using the movable In-core Detector System shall be made at least once a month to confirm that power distribution limits are being met.

Paragraph 3.10.4.4, second sentence - replace the words "The end-of-life" with the words "The beginning of life."

Basis

Second paragraph, after the second sentence insert the following two sentences:

> "At the time in the cycle when the boron concentration reaches 600 ppm, the insertion limits are lowered to accommodate the larger power reactivity defect existing at lower boron concentration (due to the increase in the magnitude of the negative moderator temperature coefficient). The less restrictive EOL insertion limits still provide for adequate shutdown margin."

Replace the third paragraph beginning "Positioning of the part-length" with the following:

> "Part length rod insertion has been eliminated for this cycle to eliminate potential adverse power shapes and to preclude rapid local power changes caused by part length rod travel through the core."

Replace Figure 3.10-1, "Control Rod Insertion Limits" with the attached revised Figure 3.10-1.

Insert new Figure 3.10-3, "Power Spike Factor versus Elevation."

Table 4.1-2, Item 6, change the words "Control Rod" to "Full Length Control Rod."

SECTION 6.0

In paragraph 6.4.2 reduce the protection factors given in Table 6.4.1, Protection Factors for Respirators; Section II, "Atmosphere-supplying respirators;" Subsection 1, "Air-line respirators: Facepiece, full - Demand mode;" and Subsection 2, "Self-contained breathing apparatus (SCBA): Facepiece, full - Demand mode" for Column 3, "Particulates and vapors and gases except tritium oxide," from 500 to 100.

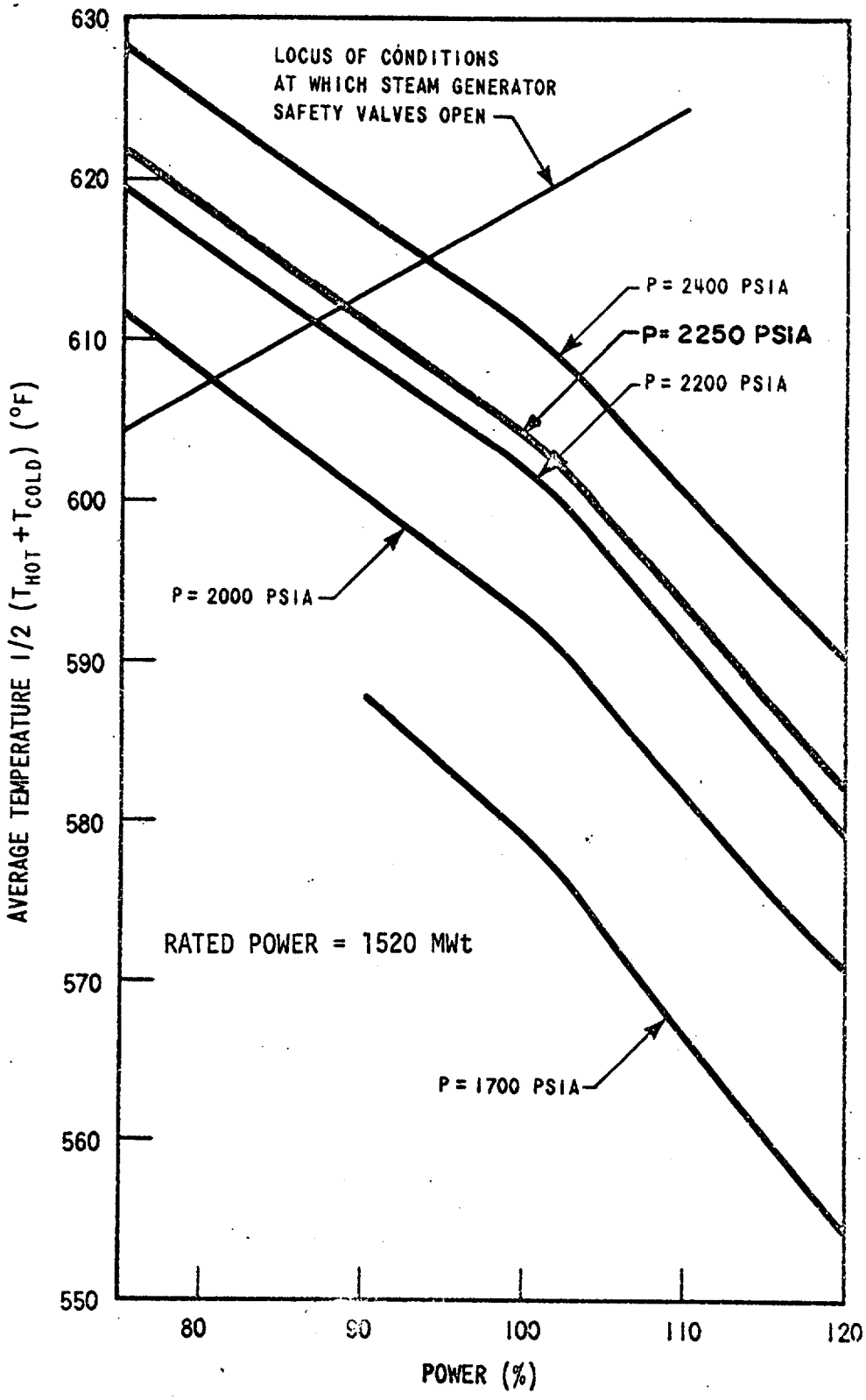
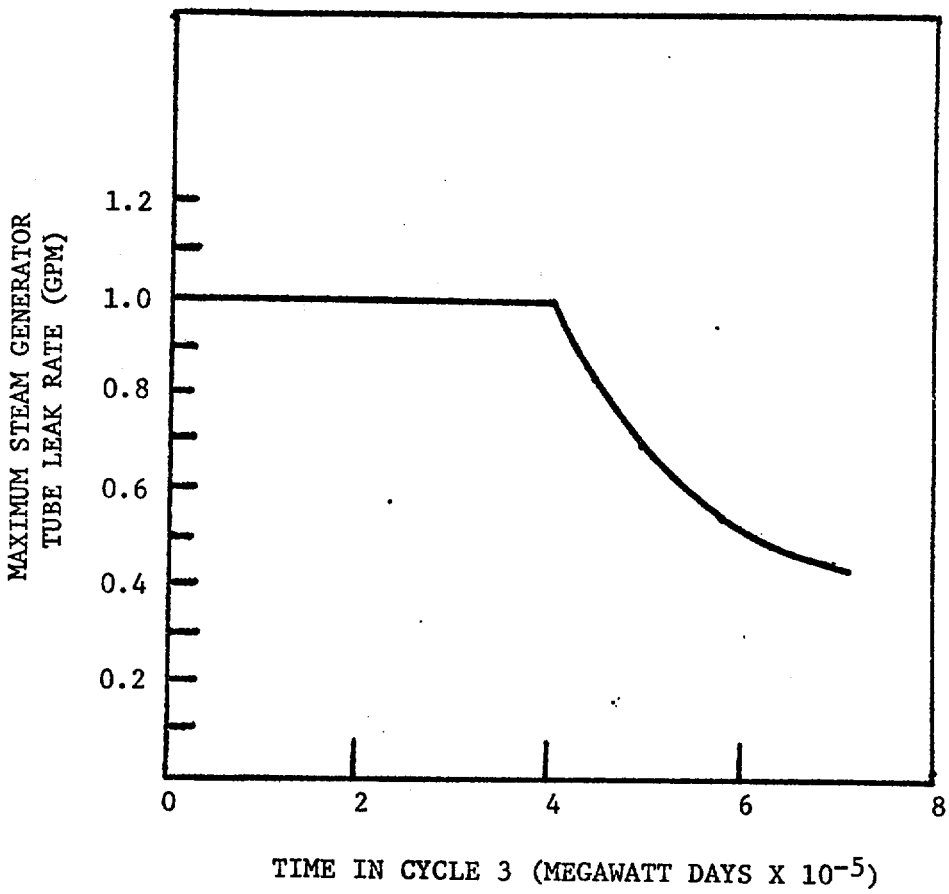


Figure 2.1-1. Core DNB Safety Limits. Two Loop Operation, 100% Flow Ginna Cycle 3



Based on: Overpower Transient
10 CFR Part 20 Dose Limits
1.5 Rem Thyroid Dose

Figure 3.1-4 Maximum Steam Generator Leak Rate versus Time

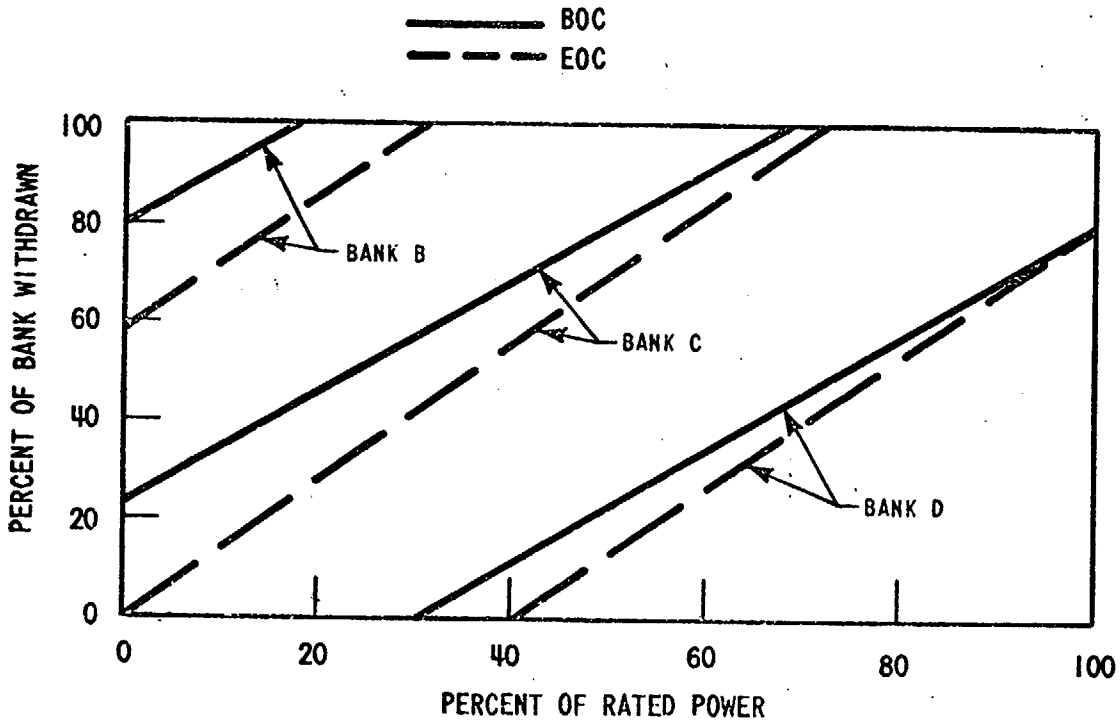


Figure 3.10-1. Control Bank Insertion Limits
Ginna Cycle 3

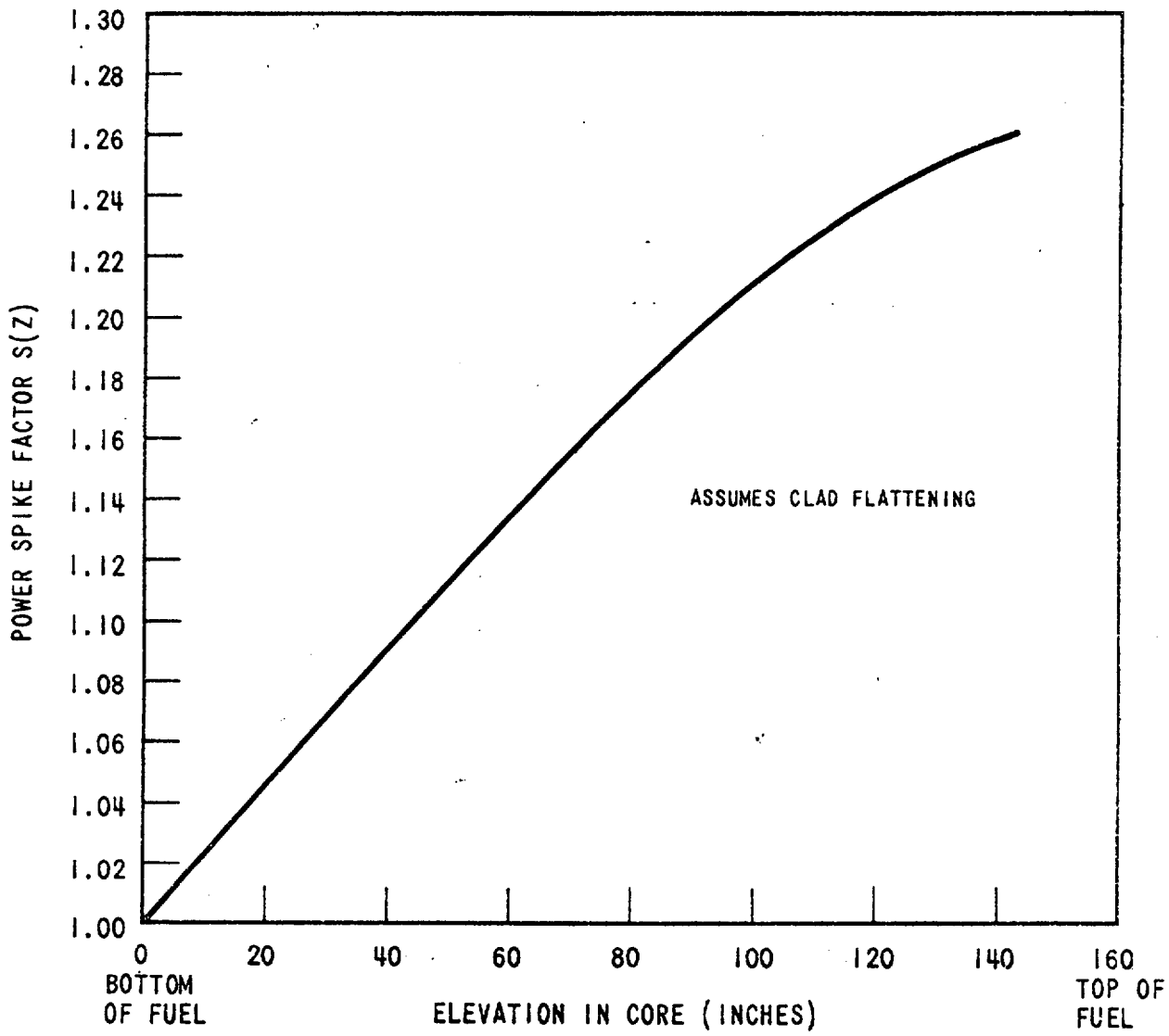


Figure 3.10-3. Power Spike Factor versus Elevation