Proposed Change No. 125
Supplement No. 9

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WYR 76-82

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE
20 Turnpike Road Westborough, Massachusetts 01581

August 17, 1976

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation

Subject: Yankee Rowe Core XII Rod Ejection Analysis

Reference:
1) License No. DPR-3 (Docket No. 50-29)
2) Proposed Change No. 125 (July 14, 1975)
3) Proposed Change No. 125, Supplement No. 1 (October 10, 1975).

Dear Sir:

Pursuant to Section 50.59 of the Commission's Rules and Regulations, Yankee Atomic Electric Company hereby proposes the following modifications to Appendix A of the current operating Technical Specifications and to Section 3.1.3.1 of the Standard Technical Specifications.

PROPOSED CHANGE: It is proposed that the attached revised pages (Attachment A) replace those submitted in Reference 2) and subsequently amended in Reference 3). In addition it is proposed that the revised pages (Attachment B) replace those currently contained in Yankee Rowe Core XII Standard Technical Specifications.

REASON FOR CHANGE: In preparation for the safety analysis required to license Core XIII a review of the analysis performed for Core XI and Core XII was performed. This review revealed an error in the methodology used in the Core XI rod ejection analysis. Since the Core XI analysis was referenced in the Core XII licensing submittal via References 2) and 3), an analysis was performed to determine the effect of the error on Core XII operating limits specified in both the current Plant Technical Specifications and in the proposed Standard Technical Specifications.

The results of the reanalysis indicate that the allowable full power ejected rod worth is unchanged (i.e., 0.5 percent $\Delta\rho$) but that the allowable zero power ejected rod worth must be limited to 0.75 percent $\Delta\rho$ (a reduction below the 1.0 percent $\Delta\rho$ currently contained in the Technical Specifications). Proposed amendments to Appendix A of the current operating Technical Specifications and to the proposed Standard Technical Specifications are provided in Attachments A and B, respectively.

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SAFETY CONSIDERATIONS: The proposed change results in a more restrictive limit on the allowable worth of a single control rod in the core at zero power conditions. Since operation has always been maintained within the proposed specifications (0.75% $\Delta\rho$), the health and safety of the public have not (and will not) be endangered.

This proposed change has been reviewed by the Nuclear Safety Audit and Review Committee.

SCHEDULE OF CHANGE: Core XII operation is currently being maintained within the more restrictive limits of the proposed specifications. We respectfully request review of this Proposed Change by the Commission as soon as practical.

Any questions regarding this Proposed Change should be directed to Mr. Alan E. Ladieu at our Engineering Office, 20 Turnpike Road, Westboro, Massachusetts, 01581, (617) 366-9011, Extension 2867.

Respectfully submitted,

YANKEE ATOMIC ELECTRIC COMPANY

W P Johnson

W. P. Johnson
Vice President

COMMONWEALTH OF MASSACHUSETTS)
)ss.
COUNTY OF WORCESTER)

Then personally appeared before me, W. P. Johnson, who, being duly sworn did state that he is Vice President of Yankee Atomic Electric Company, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Yankee Atomic Electric Company, and that the statements therein are true to the best of his knowledge and belief.

Frederic E Greenman

Frederic E. Greenman Notary Public
My Commission Expires October 6, 1978

ATTACHMENT A TO SUPPLEMENT NO. 9

OF PROPOSED CHANGE NO. 125

Instructions for Implementation:

1. Replace the current Table of Contents (con't.) of the Performance Analysis by that contained within.

2. Replace page 50 of the current document by pages 50.a through 50.d.

3. Replace page 53 by pages 53.a through 53.h.

TABLE OF CONTENTS (Con't.)

7.8 CONTROL ROD EJECTION ACCIDENT

GENERAL

The rod ejection accident is the most rapid reactivity insertion that can be reasonably postulated. A rapid reactivity insertion causes the core power to rise rapidly for a brief period until the increasing reactivity loss due to the increased resonant absorption in U238 terminates the power rise and then causes it to decline. This power increase results in excessive fuel heat generation that could lead to clad failure and fuel melting unless suitable rod restrictions and protection are provided.

The rapid ejection of a control rod from the core would require a complete circumferential rupture of the control rod drive mechanism (CRDM) housing or of the CRDM nozzle on the reactor vessel head. The CRDM housing and CRDM nozzle are an extension of the reactor coolant system boundary and are designed and manufactured to Section VIII of the ASME Boiler and Pressure Vessel Code (1956 Edition). Hence, the occurrence of such a failure is considered highly unlikely.

The amount of reactivity which can be introduced into the core as a result of a rod ejection is minimized by operation with chemical shim. Most of the control rods are withdrawn before reaching full power, thus minimizing the severity of any rod ejection accident. Furthermore, should a rod ejection occur, the resulting high power level initiates a high neutron flux trip signal which causes the shutdown rods to insert, thus reducing the neutron generated power to negligible levels.

The analytical results presented in this section pertain to the nuclear portion of the transients. The loss of coolant resulting from the primary system rupture and its consequences are similar to those for small breaks which are discussed in the section describing the Loss of Coolant Accident.

METHOD OF ANALYSIS

The analysis of the rod ejection accident was performed using the CHIC-KIN digital computer program to simulate the core hot channel response. This model incorporates the standard six delay group representation along with explicit reactivity contributions from rod motion, fuel temperature effect and moderator temperature variations.

In the rod ejection transient, the principal reactivity feedback mechanism affecting the course of the nuclear transient is the fuel temperature. Although the axial shape does not change significantly during the course of a rod ejection accident, the radial shapes at various axial slices undergo marked changes. However, the use of the static, non-Doppler flattened radial pin peaking factor (obtained from two-dimensional diffusion theory results) in conjunction with the average core energy release (obtained from the point kinetics results) yields hot spot energy releases that are conservative since the fuel temperature effect during a transient is strongest at the radial peak location, limiting the radial peak to a value below that obtained from the static ejected rod configuration for the major portion of the transient.

By combining the point kinetics results with the radial pin peaking factor as discussed above, the peak fuel centerline temperatures are determined from the CHIC-KIN computer program. These peak fuel centerline temperatures are converted to the hot spot total centerline enthalpies and compared to threshold enthalpy values to determine the type and degree of fuel damage. The threshold enthalpy values are: (1, 2, 3).

Clad Damage Threshold:	Total Average Enthalpy = 200 cal/gm
Incipient Centerline Melting Threshold:	Total Centerline Enthalpy = 250 cal/gm
Fully Molten Centerline Threshold:	Total Centerline Enthalpy = 310 cal/gm

The analyses of the rod ejection accidents were performed using the following assumptions:

(a) The fully inserted rod ejects instantaneously;

(b) The fuel temperature coefficient is the least negative value expected throughout core life during power operation;

(c) The maximum ejected rod worth throughout core life is assumed to be 0.75% $\Delta\rho$ at zero power and 0.5% $\Delta\rho$ at full power conditions;

(d) The calculated maximum ejected rod peaking throughout core life is increased by 10 percent in order to account for calculational uncertainties.

RESULTS

The cases which have been analyzed were selected on the basis of a calculational survey which determined the worst cases from the standpoint of power peaking and reactivity worth of the ejected rod. In all cases analyzed, the highest three-dimensional power peak resulted when the maximum reactivity worth rod was ejected.

Tables 7.8-1 through 7.8-4 and Figures 7.8-1 through 7.8-4 present the initial conditions and results for the full power and zero power cases.

CONCLUSIONS

The rod insertion limits which have been established, in conjunction with the reactor protective system, assure that no clad damage or fuel melting occurs for the rod ejection accident.

REFERENCES

1. Brassfield, H.C., et. al., "Recommended Property and Reaction Kinetics Data for use in Evaluating a Light Water-Cooled Reactor Loss-of-Coolant Incident Involving Zircaloy-4, or 304-SS Clad UO_2, GEMP-482, April 1968.

2. Idaho Nuclear Corporation, Monthly Report, Ny-123-68, October, 1969.

3. Idaho Nuclear Corporation, Monthly Report, Hai-127-70, March, 1970.

7.9 STEAM LINE RUPTURE ACCIDENT

The analysis of this event for the reference cycle formed the basis for selecting the rod insertion limits. Thus, any degradation in performance for the reload cycle would have to be reflected in more restrictive rod insertion limits. The shutdown margin has increased for the reload cycle relative to the reference cycle. Thus, no modification to the rod insertion limit curves is required to maintain the same margin of safety.

7.10 STEAM GENERATOR TUBE RUPTURE ACCIDENT

The analysis of this event for the reference cycle demonstrated that the results are not sensitive to the core design. Thus, the results of the analysis presented in Proposed Change No. 115 will also apply to the reload cycle.

7.11 LOSS OF COOLANT ACCIDENT

Analysis of this accident is presented in the Appendix to this report.

7.12 OTHER ACCIDENTS AND TRANSIENTS

Previous analyses (Reference 13) have demonstrated that the Waste Gas Incident, Fuel Handling Incident, Reactor Containment Pressure Analysis, Hypothetical Accident, and Post Accident Hydrogen Considerations are not sensitive to core design changes and therefore the results presented in Reference 12 still apply to the reload cycle.

Table 7.8-1

Rod Ejection Initial Conditions At Full Power

Parameter	Value
Core Power Level (Mw)	618
Hot Spot Fuel Centerline Temperature (^{o}F)	3381
Ejected Rod Worth (%$\Delta\rho$)	0.5
Post Ejection 3-D Peak	3.17
Moderator Temperature Coefficient ($10^{-4}\Delta\rho/F$)	-0.5
Fuel Temperature Coefficient ($10^{-5}\Delta\rho/F$)	-.766
Delayed Neutron Fraction	.005743
Neutron Lifetime (μsec)	17.0

Table 7.8-2

Rod Ejection Initial Conditions at Zero Power

Parameter	Value
Core Power Level (Mw)	1
Hot Spot Fuel Centerline Temperature (^oF)	548
Ejected Rod Worth (%$\Delta\rho$)	0.75
Post Ejection 3-D Peak	5.52
Moderator Temperature Coefficient ($10^{-4}\Delta\rho/F$)	0.0
Fuel Temperature Coefficient ($10^{-5}\Delta\rho/F$)	-1.0
Delayed Neutron Fraction	.005743
Neutron Lifetime (μsec)	17.0

Table 7.8.3

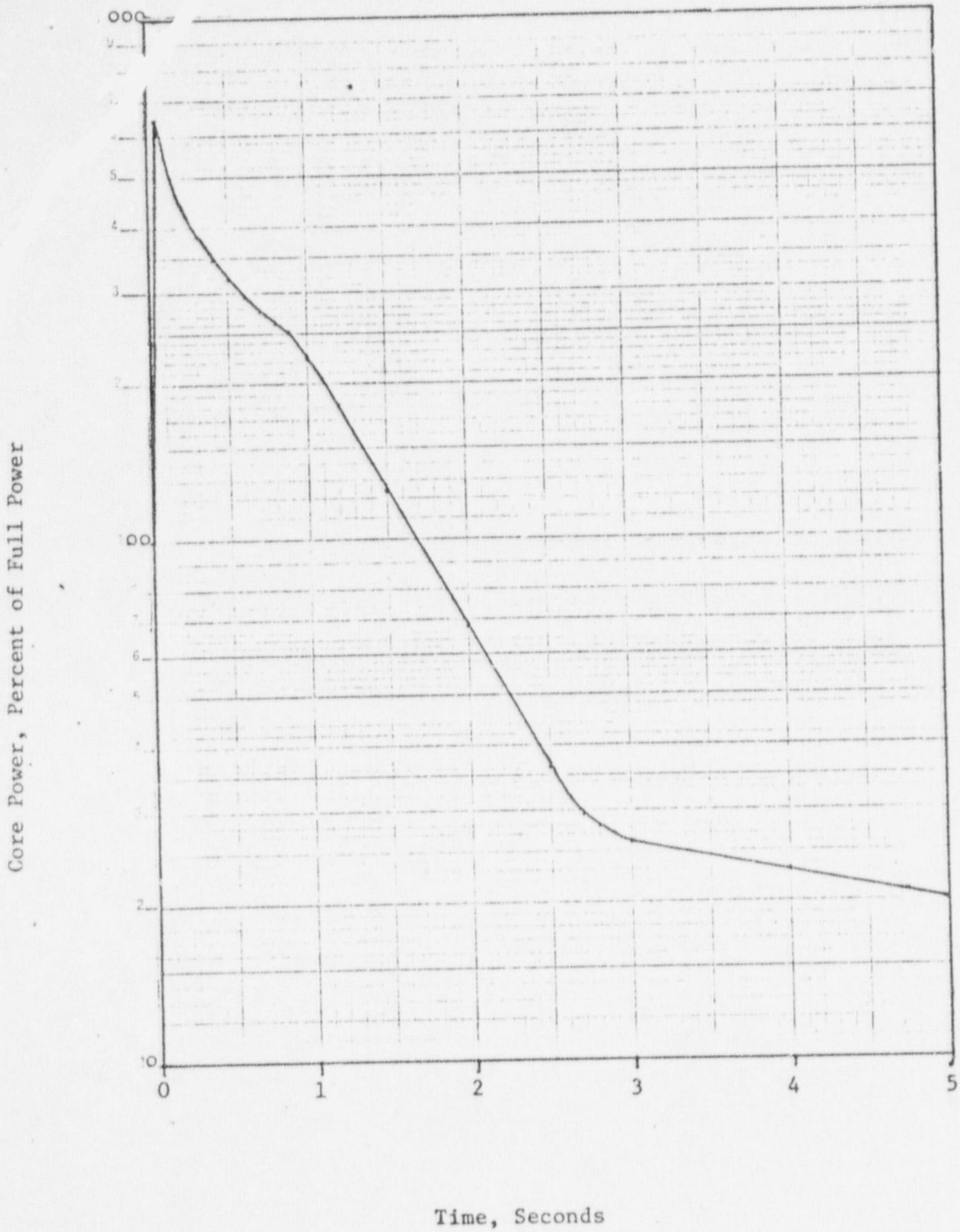
Rod Ejection Results at Full Power

Parameter	Value
Maximum Average Enthalpy at Hot Spot (cal/gm)	134
Maximum Fuel Centerline Enthalpy at Hot Spot (cal/gm)	209
Fraction of Rods that Suffer Clad Damage (Average Enthalpy $\geq$ 200 cal/gm)	0.0
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy $\geq$ 250 cal/gm)	0.0

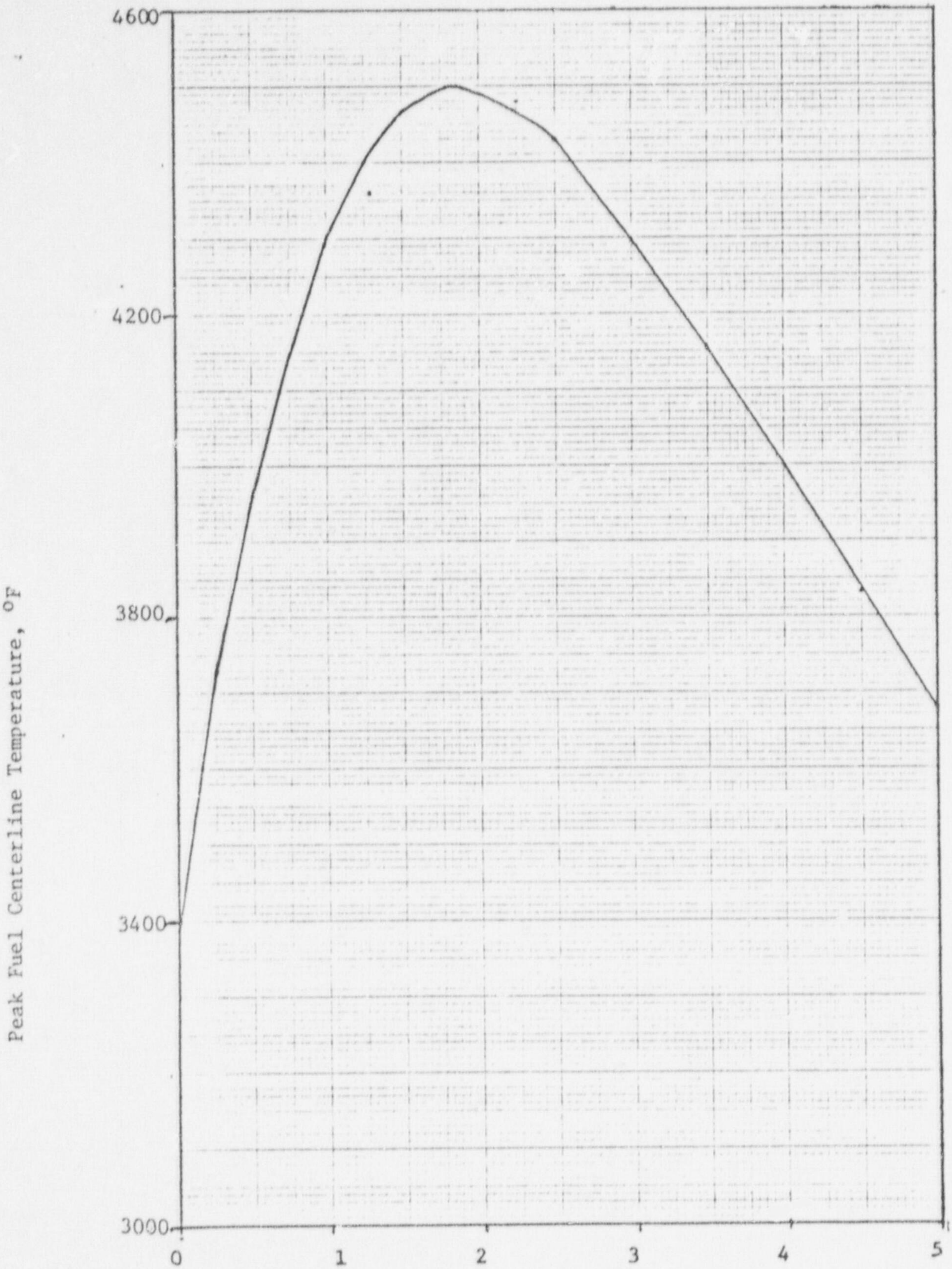
Table 7.8-4

Rod Ejection Results at Zero Power

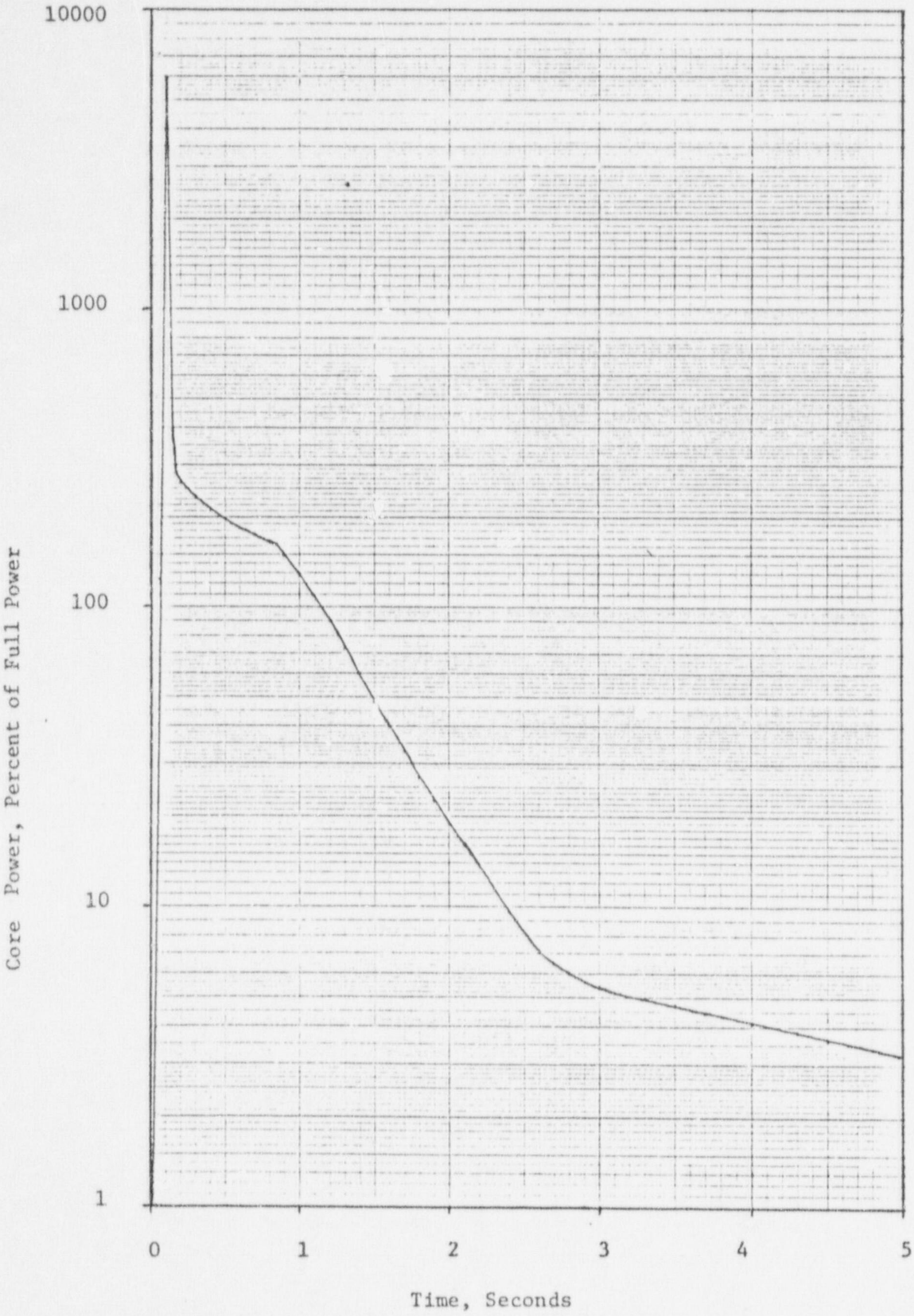
Parameter	Value
Maximum Average Enthalpy at Hot Spot (cal/gm)	183
Maximum Fuel Centerline Enthalpy at Hot Spot (cal/gm)	213
Fraction of Rods that Suffer Clad Damage (Average Enthalpy $\geq$ 200 cal/gm)	0.0
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy $\geq$ 250 cal/gm)	0.0



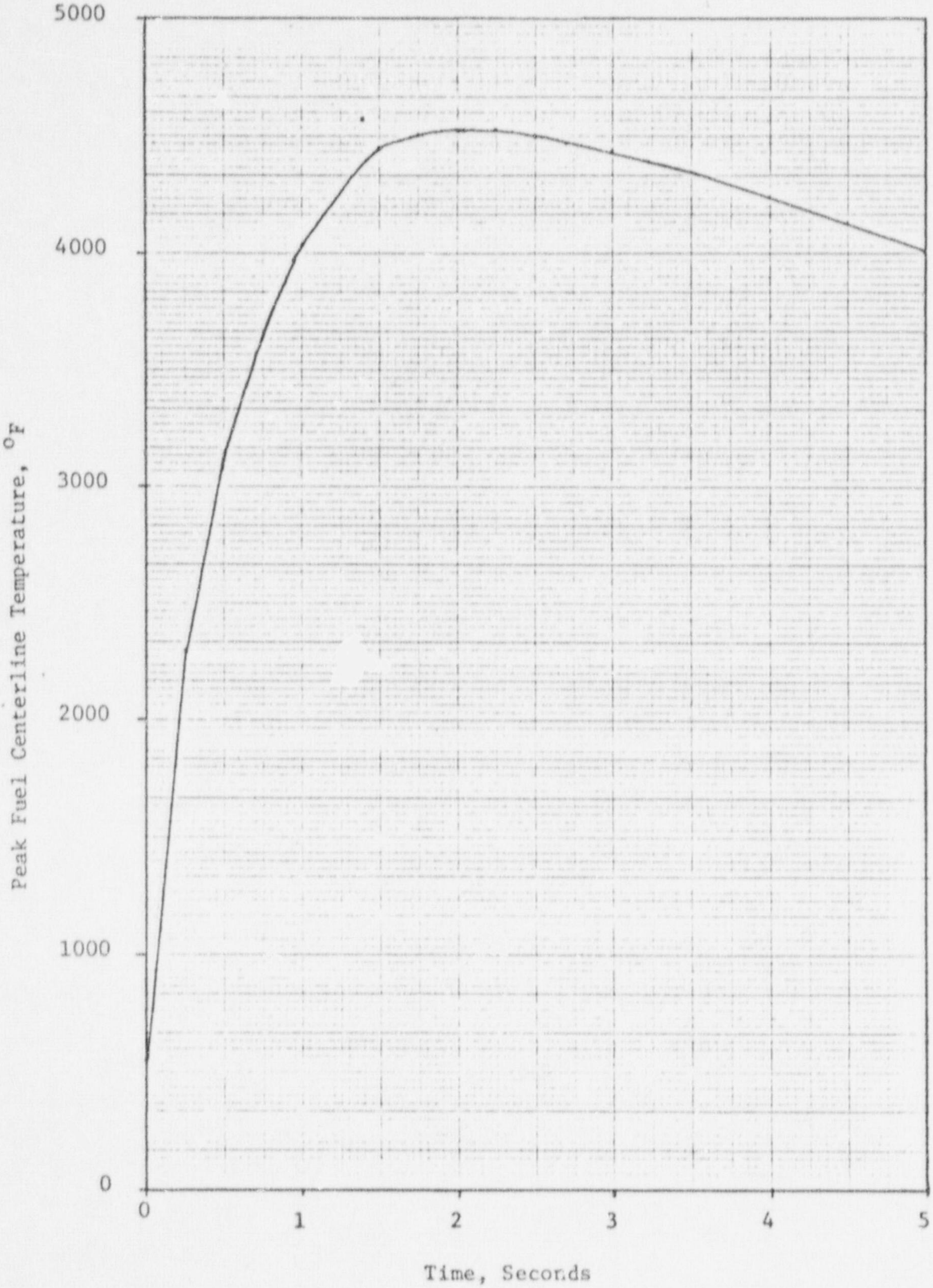
Yankee Nuclear Power Station	Rod Ejection Accident at Full Power Core Average Power vs. Time	Figure 7.8-1



Yankee Nuclear Power Station	Rod Ejection Accident at Full Power Peak Fuel Centerline Temperature vs. Time	Figure 7.8-2



Yankee Nuclear Power Station	Rod Ejection Accident at Zero Power Core Average Power vs. Time	Figure 7.8-3



Yankee Nuclear Power Station	Rod Ejection Accident at Zero Power Peak Fuel Centerline Temperature vs. Time	Figure 7.8-4

ATTACHMENT B TO SUPPLEMENT NO. 9

OF PROPOSED CHANGE NO. 125

Instructions for Implementation:

Replace page 3/4 1-23 of the Proposed Yankee Rowe Standard Technical Specifications by the attached page.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods which are inserted in the core shall be OPERABLE and positioned within $\pm$ 8 inches (indicated position) of every other rod in their group.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more control rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one control rod inoperable or misaligned from any other rod in its group by more than $\pm$ 8 inches (indicated position), be in HOT STANDBY within 6 hours.

c. With one control rod inoperable or misaligned from any other rod in its group by more than $\pm$ 8 inches (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) An analysis of the potential ejected rod worth is performed within 3 days and the rod worth is determined to be $\leq$ 0.75% Δk at zero power and $\leq$ 0.5% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

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*See Special Test Exceptions 3.10.2 and 3.10.4.