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PARAMETRIC SCRAM TRANSIENT ANALYSES
FOR THE FFTF FAST TEST REACTOR

J. W. Hagan, A. Padilla, and G. A. Worth

November 1967

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PARAMETRIC SCRAM TRANSIENT ANALYSES
FOR THE FFTF FAST TEST REACTOR

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ABSTRACT

Parametric analyses of the transient characteristics of the FFTF fast test reactor for various assumed ramp reactivity inputs which are terminated by trip of the reactor safety system have been performed and the results of these studies are described. In each case the reactor was assumed at operating conditions at initiation of the ramp. These results will be used in evaluation of the transient sensitivities to the various design parameters for establishing the design trade-offs and system limits.

System variables included in the study were: the magnitude of the ramp reactivity insertion, the system response time, the safety rod acceleration rate, and the magnitude of the Doppler coefficient. In each case, the maximum fuel overheating factor was determined for the transient response by comparing the peak fuel energy density to the steady-state value at 100% power.

Values of the maximum overheating factors have been analyzed to estimate the values for various fuel damage thresholds. These thresholds include (a) incipient fuel melting of the hottest pin, (b) centerline fuel completely molten, and (c) 50% of the pin cross-sectional area at or above the fuel melting point. Each fuel damage threshold corresponds to some degree of accident severity and would require some type of corrective action before resumption of operation, e.g., fuel inspection, replacement of fuel, AEC approval for resumption of operation.

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PARAMETRIC SCRAM TRANSIENT ANALYSES FOR THE FFTF FAST TEST REACTOR

1. INTRODUCTION

Scram transient analyses have been performed to evaluate the sensitivity of the maximum fuel overheating factor to various system parameters to establish design bases for the core, safety system, and instrumentation. The primary variables investigated included (a) reactivity ramp insertion rates, (b) safety system response time, (c) scram acceleration rate, and (d) the magnitude of the Doppler reactivity coefficient. Conservative estimates were chosen for system characteristics which were not varied throughout the analysis, e.g., the safety system rod worth and the coolant temperature coefficient.

The fuel overheating factor is an effective measure of the change in the relative stored energy in the fuel during the transient and is equivalent to the ratio of the peak energy during the transient or overpower condition, to the operating steady-state value where these energies are evaluated relative to the sink temperature condition at that location. Neglecting phase transitions, the overheating factor for a given transient may be computed by:

$$F_{OH} = \frac{(\hat{T}_f - \hat{T}_c)_z}{(\bar{T}_f - \bar{T}_c)_z}$$

where: $\hat{T}_f$ = peak fuel center temperature during the transient
at some axial location z .

$\hat{T}_c$ = coolant temperature when $T_f = \hat{T}_f$ at axial location z .

$\bar{T}_f$ = fuel center temperature at steady-state, full power
at axial location z .

$\bar{T}_c$ = coolant temperature when $T_f = \bar{T}_f$ at axial location z .

When fuel reaches incipient melting, the fuel temperature remains constant until all the fuel at that point has passed through the phase transition. Using the above equation under these conditions would be in error unless the amount of energy absorbed in melting were converted into an equivalent tem-

perature increase. However, it was found that the overheating factors could be calculated within the accuracy of the results by computing the values using fuel modes which did not reach incipient melting due to their location away from the hottest mode.

Fuel damage thresholds are assumed to be dependent upon the peak temperatures and the degree of melting of the fuel and hence, the overheating factor. With oxide fuel, no damage is expected until some amount of fuel becomes molten to allow some relocation of fuel within the clad. The fuel damage thresholds for the various severity levels have been tentatively defined⁽¹⁾ and are related to the AEC parametric limits⁽²⁾ as shown in Table 1 below.

Table 1
Fuel Damage Severity Limits

AEC Limits on Safety Parameters		Fuel Damage Severity	
		R @ T _{melt}	Fuel T _{max}
Danger Threshold	<div><div>Uncertainty</div><div>Allowance to Danger</div></div>	Disruptive Fuel Damage	(Violent fuel release)
		$\sqrt{0.5} R_0$	~5400°F
Safety Limit	<div><div>Safety Margin</div><div>Operating Margin</div></div>	Gross Fuel Damage	(Release of fission products and nonviolent fuel release)
		$\sqrt{0.1} R_0$	4850°F
Maximum Safety System Setting		0	4850°F
Nominal Operation	<div><div>Normal Operating Zone</div></div>	0	~4350°F

In the analyses, the relationships between fuel damage thresholds, or severity limits, and the maximum fuel overheating factor were investigated. However, due to the difficulty in running transient cases which just peak out at a defined threshold, these relationships were estimated and the approximate values statistically weighted to indicate the approximate range of the estimate.

2. SUMMARY

The results of this study are shown in Figures 1 through 4. In each case the results are presented with the overheating factor as the dependent variable.

The design ramp rate for the reactor should be as high as practical to avoid undue limitations on the facility in either the design or flexibility of operation. The allowable ramp rate influences the design of components whose function, or whose failure, may result in a change in the reactivity status of the reactor. Components which fall in this category include: the control and safety system drives, fuel subassemblies, test elements, and hold-down techniques to prevent or limit component movement. Higher allowable ramp rates require more sophisticated designs of the control and safety systems to limit the system delays and to accelerate the control elements at greater rates upon release.

Figure 1 shows the overheating factors as a function of the input ramp rate with five different combinations of system delay times and rod accelerations. These plots show that if a design requirement of a 4\$/second ramp and negligible fuel damage were chosen, the requirement could be met with a 1 g (32.2 ft/sec^2) system and a 50 msec delay or a 3 g system with a 100 msec delay.

Figure 1 also shows the levels of the various damage thresholds. No damage is expected up to incipient melting of the hottest pin which was not observed until the overheating factor exceeded about 1.13. However, little additional overheating is required to advance incipient melting for the inner radius to 10% of the cross-sectional fuel area; this was estimated to be 1.14. An overheating factor of 1.39 was evaluated when the central fuel was completely molten (observed that the fuel temperature of the innermost node began rising above the melting point). Finally, an overheating factor of about 1.6 resulted in 50% of the cross-sectional area at or above the melting point.

Figure 2 shows the dependency of the overheating factor upon the Doppler coefficient. The plots show that the overheating factor is fairly independent of Doppler over the ranges investigated for cases in which the peak overheating factors are less than 1.2. However, for cases which result in much higher overheating factors, the Doppler coefficient becomes

a quite important parameter.

Figures 3 and 4 show the effects of rod acceleration and time delay using the same data plotted in Figure 1. These figures show that with a design requirement of 4\$/second and negligible fuel damage, the requirement can be met with a 2 g rod and 100 msec delay.

3. CALCULATIONAL TECHNIQUES

3.1 Fuel Model

The physical characteristics of the fuel pin assumed in the analysis are summarized in Table 2 below:

Table 2
Fuel Pin Description

Cladding material	Stainless steel -- Type 304
Clad O.D.	0.250 inches
Clad thickness	0.016 inches
Fuel pellet density	93% T.D.*
Fuel pellet smeared density	88% T.D.
Fuel pellet sintered density	98% T.D.
Fuel pin length	36 inches
Pitch to diameter ratio	1.12

*T.D. - theoretical density

Figure 5 is a diagram of the fuel pin showing its physical configuration following sintering at design operating power at which the temperatures of the fuel, cladding, and coolant are as shown in the figure. This figure represents the hottest fuel pin in the reactor and the linear power of 15.1 kW/ft was determined from the condition that incipient fuel melting (assumed to be 4850°F) is reached at 25% overpower. The following core parameters assumed to establish the design power thermal conditions on this pin were:

Radial peak-to-average power factor	1.15
Axial peak-to-average power factor	1.24
Engineering hot channel factor	1.20
Average coolant temperature rise across core, °F	300
Flow orificing	None

Since no orificing is assumed, the coolant temperature rise of 414°F across the hottest pin is obtained by multiplying the core temperature rise by the radial power peaking factor and the engineering hot

channel factor.

It is assumed that during operation the initial mixed-oxide pellet (93% of theoretical density) expands uniformly out to the cladding resulting in a fuel density of 88% of theoretical. The fuel above 1800°C (3272°F) under normal operating conditions is then assumed to be sintered to 98% of the theoretical density, thus giving the variable axial void shown in Figure 5; the maximum void is shown at axial nodal position 10 in which the void radius is 0.0247 inches. The outer radius of the sintered fuel at this same position is 0.0774 inches. It is further assumed that no additional sintering occurs during the transient.

Figure 6 shows the thermal conductivities of the sintered and unsintered fuel⁽³⁾ used in the analyses.

3.2 Heat Transfer Models

The SINTR code⁽⁴⁾ was used to predict the internal fuel relocation under normal steady-state operating conditions. The fuel regions determined by SINTR were then divided into nodes for transient analysis with the NUTIGER code⁽⁵⁾. NUTIGER is an expansion of the three-dimensional heat transfer code TIGER V to include nuclear feedbacks due to Doppler, coolant density change, fuel expansion, and fuel bowing.

For the transient analysis, the fuel pin was divided into 18 two-inch-long axial segments. Each segment consisted of seven internal (volumetric) fuel nodes, an outer fuel surface node, an internal cladding node, inner and outer cladding surface nodes, and a coolant node. Axial heat conduction within the fuel, cladding, and coolant was neglected.

NUTIGER solves the zero-dimensional, multi-group (up to six delayed neutron groups) nuclear kinetics equations after each TIGER time step and supplies TIGER with the current value of power. Changes in temperature are fed back to NUTIGER as changes in reactivity, causing subsequent changes in power. Transients can

be initiated by time-dependent external reactivity insertion, coolant flow rate, or coolant inlet temperature. For these analyses, the external reactivity insertion consisted of the input ramp and, after a given time delay, the reactivity worth due to scram of the safety system. The feedbacks used were those due to the Doppler effect and the sodium density change.

3.3 System Parameters

3.3.1 External Reactivity Insertions

External reactivity effects result from the input ramp rate and the safety system worth. These effects are entered into the analysis by absolute time values with sufficient number of points to allow program interpolation for each iteration.

The positive reactivity worths of the ramp rates are shown in Figure 7 where it is assumed that the ramp is imposed upon the steady-state operating condition at time, $t = 0$.

The safety system reactivity worths for various conditions are shown in Figure 8. The insertion rates of the rods depend upon the rod acceleration after release as shown in Figure 9. The initial position of the poison tips of the rods was assumed to be four inches (10 cm) above the fueled boundary and a total system worth of 10\$ was chosen for the analyses to represent the minimum expected scram-mable worth for a case where no backup scram system is provided and the safety system worth is reduced by a single failed rod. The differential worths of the rod system, shown in Figure 9, were computed using predicted worth curves⁽⁶⁾.

Two time delays determine the time after initiation of the ramp at which the rods are released. These are:

- (a) The time required for the flux to rise to the trip setting of the flux instruments (τ_s). Neglecting delayed neutron

changes, this time, τ_s , equals:

$$\tau_s = \frac{F - 1}{\alpha F}$$

where F = the ratio of the flux at the
trip setting to the initial
flux level,

α = reactivity ramp rate, \$/sec

The trip setting in these runs was assumed to be 125% of design operating power and the initial power at the beginning of the transient was assumed to be at 110% (to conservatively account for any operational fluctuations before initiation of the transient condition). Therefore, for these cases, $F = \frac{1.25}{1.10} = 1.136$ and the values of τ_s were as follows:

<u>Table 3</u>	
<u>Trip Time Delay</u>	
<u>α, (\$/sec)</u>	<u>τ_s, (sec)</u>
1	0.120
2	0.060
3	0.040
4	0.030

- (b) The time required for the instrumentation to sense the trip condition and for the latch mechanisms to open (τ_d). This delay time was varied in the analyses from 0.050 to 0.200 sec.

Comparison of Figures 7 and 8 shows that the excursion is essentially terminated in less than 0.5 seconds and the initial 1-2\$ of rod insertion is all that is required to reverse the power transient. For this reason, no attempt

was made in these transients to terminate the ramp insertions at some maximum value, although in reality, the ramps would be limited at some maximum level of insertion.

3.3.2 Reactivity Coefficients

Two temperature dependent reactivity effects were included in the analysis. These were:

- (a) Sodium temperature effect.
- (b) Doppler effect.

The sodium temperature reactivity coefficient was $-2.0 \times 10^{-6} \delta k/^{\circ}\text{C}$. The reactivity effect was computed by multiplying this coefficient by the average coolant temperature over the length of the fuel pin. Since the fuel model in the analyses simulates the hottest pin, the sodium temperature reactivity effect was slightly overestimated, however, scoping runs showed this to be a negligible effect.

The Doppler coefficient ($T \frac{dk}{dT}$) was varied between -0.002 to -0.004 throughout the studies and the reactivity effect was computed using the average temperature of the sintered fuel in the pin being analyzed. In the actual case, the Doppler effect would depend upon some importance-weighted average of the fuel temperature throughout the core. Using only the sintered fuel temperature of the hottest pin tends to underestimate the Doppler effect and provide additional conservatism in the results for the overheating factors. This is explained by:

- (a) Using a higher temperature base reduces the differential negative reactivity per unit temperature increase and thus provides a smaller negative feedback during the transient.

- (b) Once a fuel node reaches the melting point its contribution to the Doppler magnitude is held constant throughout the melting process within that node. Using the fuel which melts first, i.e., sintered fuel, and neglecting other fuel nodes which continue to rise in temperature throughout the transient results in a smaller negative feedback for that transient.

The effect of the Doppler coefficient upon the overheating factor is shown in Figure 2. This shows that the effect of varying the Doppler coefficient from $-.002$ to $-.004$ has little effect for transients in which the overheating factor is less than 1.2 to 1.3.

3.4 Severity Thresholds

In the analyses, the results are presented in terms of the overheating factors. This parameter is meaningless without relating it to fuel damage severity thresholds such that the effects of various transients can be evaluated as to cost and potential damage to the facility.

The severity thresholds shown in Table 1 have been tentatively chosen. Negligible damage to the fuel is expected if the fuel does not reach incipient melting. However, if fuel does become molten, it may relocate within the clad. Gross movement is not expected if the area of the fuel cross section which is at the melting point is less than 10% of the total cross sectional area. If fuel does relocate, the effective lifetime of the subassembly may be reduced. Gross clad damage is not expected until molten fuel flows against the clad or until the internal gas pressure from fission product accumulation causes localized stresses to rise above the strength of the clad.

Runs in which the temperatures peaked near these established thresholds were plotted to determine the estimated overheating factor at that threshold. These values were then averaged and the variance

of the calculated values determined to get an estimate of the probable error of the calculation. The ranges are shown as brackets about the average values in Figure 1 and have not been computed by rigorous statistical techniques but are only estimates of the probable error (50% probability limits).

As shown in Figure 1, the overheating factors for incipient melting and 10% areal melting are very close and probably indistinguishable insofar as allowable ramp reactivity insertion rates are concerned for the different severity thresholds. As experimental data are received from transient in-reactor tests, the threshold for significant fuel movement will be re-evaluated as a function of the amount of fuel melted in a given region or the amount of fuel at the melting temperature.

4. COMPARISON TO TYPICAL PARAMETERS

The cases analyzed in these studies used conservative estimates of the scrammable reactivity, trip settings, and Doppler coefficients to define the minimum design requirements. Figure 10 compares these results with those of another study using expected values of the parameters⁽⁷⁾. As shown, the allowable ramp rate for a given overheating factor is about 1-1/2 \$/sec greater for the runs using the typical parameters.

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FIGURE 1. Maximum Transient Fuel Overheating Factor

FIGURE 2. Effect of Doppler Coefficient

FIGURE 3. Effect of Safety Rod Acceleration

FIGURE 4. Effect of Time Delay to Rod Release

FIGURE 5. Fuel Geometry and Temperature Distribution for Hottest Fuel Pin

FIGURE 6. Fuel Thermal Conductivity

FIGURE 7. External Reactivity Insertion

FIGURE 8. Rod Insertion Scram Worth

FIGURE 9. Safety System Worth After Scram

FIGURE 10. Comparison to Typical Parameters

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