OCT 30 1964

UNCLASSIFIED

CLASSIFICATION LEVEL
(S, C, OR U)

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

MASTER

NAA-SR-MEMO 5397

Do not remove
this sheet

This document contains 17 pages

This is copy _____ of _____ series _____

CLASSIFICATION TYPE
(RD OR DI)

*NAA-SR-MEMOs are working papers and may be expanded,
modified, or withdrawn at any time,
and are intended for internal use only.*

This report may not be published without the approval of the Patent Branch, AEC.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission to the extent that such employee or contractor prepares, handles or distributes, or provides access to, any information pursuant to his employment or contract with the Commission.

UNCLASSIFIED

CLASSIFICATION LEVEL
(S, C, OR U)

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

WRITTEN BY J. R. Hume		ATOMICS INTERNATIONAL A Division of North American Aviation, Inc. TECHNICAL DATA RECORD	TDR NO. 5397	
SECTION Product Engineering			GO 9304	
GROUP Analytical Engineering			LEDGER ACCT 3621	
UNIT Nuclear & Rad. Analysis			SUB-ACCT. 4575	
APPROVED BY: (SUPERVISOR) <i>R. L. Ashley</i>		PROGRAM General Research Program	TWR 6352	
OTHER		PROJECT Power Generation	DATE June 21, 1960	
			PAGE 1 OF 17	

TO: J. W. Flora *

COPIES TO: G. Wegmann*
M. Dunenfeld *
P. Spiegler *
M. Silberberg *

R. L. Ashley *
M. P. Heisler *
C. J. Brous
R. Davidson *

SUBJECT:

Transient Energy Release Analysis of An Experimental Fuel Pin in TREAT

CONTENTS:

I	STATEMENT OF PROBLEM	PAGE 1
II	SUMMARY OF RESULTS AND RECOMMENDATIONS	PAGE 1
III	METHOD USED, DESCRIPTION OF EQUIPMENT, SAMPLE CALCULATIONS . . .	PAGE 2
IV	REFERENCES AND APPENDICES	PAGE 3
V	REFERENCES	Page 5

I STATEMENT OF PROBLEM

Determine the time and space dependent energy release rate in an experimental fuel pin placed in the TREAT.

II. SUMMARY OF RESULTS AND RECOMMENDATIONS

The time and space dependent energy release in an experimental fuel pin placed in the Transient Reactor Test Facility (TREAT) has been analysed. The problem is essentially the determination of the time and space neutron flux which is, of course, proportional to the heat generation rate. A simplification results from the assumption of a "stationary" spatial flux which is established during steady state and persists while the reactor is experiencing a transient. Calculation of the stationary flux determines the relationship between energy released in the fuel pin and energy released in the TREAT core. Then the time dependent fuel pin energy release problem is completed if the time dependent and integral energy release of the TREAT core can be computed or determined experimentally. The numerical results reported here are normalized to an integral energy release of 1000 Mw-sec in the whole TREAT core. Users of this data can make suitable adjustments to match the time dependent and integral energy release of a particular transient.

The analysis considers the case of a 0.3" diameter, 4' long, fuel pin with a 0.010" stainless steel cladding. An annulus of organic coolant surrounds the pin and is contained by a 0.75" OD, 0.050" wall stainless steel tube. The entire assembly is presumed to be located in an experimental opening on the vertical center line of TREAT.

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. 5397
DATE June 21, 1960
PAGE 2 OF 17

In addition to the energy release averaged over the whole fuel pin, the average in each of two annular regions within the pin was determined. The two regions were divided at a radius equal to half the pin radius. The ratio of surface-to-pin average release rate was also determined. The results for uranium-10 percent molybdenum alloy fuel and uranium dioxide fuel were as follows:

	Average Energy Release (Mw-sec/cm ³)			Surface- to-Average Ratio
	Interior Region	Exterior Region	Whole Pin	
U-10 Mo				
Natural Uranium	0.0092	0.0096	0.0095	1.04
2 w/o Enriched	0.0223	0.0241	0.0236	1.075
5 w/o Enriched	0.0380	0.0455	0.0435	1.11
8 w/o Enriched	0.0430	0.0585	0.0546	1.18
UO ₂				
Natural Uranium	0.0055	0.0057	0.0056	1.04

Details of the radial distributions in the fuel pins are shown in Figures 2 through 7.

The vertical distribution of heat generation was approximated by a chopped-sine function with a peak-to-average of 1.2. This function is plotted in Figure 9.

III. DISCUSSION OF THE PROBLEM

Suppose an experimental fuel pin typical of current design is placed in a test hole in the center of the TREAT (Transient Reactor Test). A calculation of the time and space dependent heat generation rate during a power transient is desired.

The experimental apparatus is presumed to have the following specification: The fuel pin is 0.300" in diameter and has a 0.010" cladding of stainless steel. The fuel is 4' long, to correspond with the length of the TREAT core. A concentric stainless steel tube with 0.75" OD and 0.050" wall serves as a primary containment vessel. An annulus of organic coolant fills the intervening space. Additional secondary containment is left unspecified except for the requirement that the entire assembly fit into an 8 inch square test hole. Fuel materials to be considered are uranium dioxide and uranium-10 percent molybdenum alloy. A sketch of the experimental apparatus is presented in Figure 1.

IV. ANALYSIS

Energy release in the fuel pin will be proportional to $\sum_f \phi(r, z, t)$ where $\sum_f$ is the macroscopic fission cross-section and $\phi(r, z, t)$ is the time and space dependent neutron flux. Hence, determination of the flux term is the major part of this problem. Simplification follows from the usual practice of separating space and time effects. It will be assumed that the "stationary" flux distribution in the reactor and fuel pin which is established at steady state persists even when the reactor is on a transient. Then the ratio of flux in the fuel pin to flux in TREAT is constant during transients. Independent calculations or measurements of the TREAT flux and energy release then imply the time dependence of flux in the fuel pin. Examples of TREAT transient flux are given in references 1, 2, and 3.

We now concentrate on the spatial distribution. With the objective of creating a large heat sink, the designers of TREAT diluted the uranium with as much graphite as possible. This results in a very well thermalized reactor so that fissions from fast neutrons may be neglected. The determination of the thermal flux may be divided into several parts. First we define a fuel pin disadvantage factor as the ratio of average thermal flux in the fuel pin to thermal flux in adjacent TREAT fuel. Next we define a term which relates thermal flux in fuel adjacent to the fuel pin to the reactor average. Considering a horizontal plane with the fuel pin in the center, the latter term amounts to the horizontal peak-to-average since the maximum flux will be located essentially adjacent to the fuel pin. Following the usual practice of assuming flux separability in terms of the spatial coordinates, the horizontal plane is considered to be at the vertical average power level and the vertical distribution is determined separately. Now the energy release rate in the fuel pin can be determined as follows:

$$\left(\begin{array}{l} \text{energy release} \\ \text{rate in fuel pin} \\ \text{per cm}^3 \end{array} \right) = \left(\frac{\sum_f \text{fuel pin}}{\sum_f \text{TREAT}} \right) \left(\begin{array}{l} \text{Pin disadvantage factor} \\ \\ \end{array} \right) \times \left(\begin{array}{l} \text{horizontal} \\ \text{peak-to-average} \end{array} \right) \left(\begin{array}{l} \text{average energy release} \\ \text{rate in TREAT per cm}^3 \end{array} \right)$$

The pin disadvantage factor can be determined by considering a circular cell of the following composition from inside to outside: 1) fuel material, 2) cladding, 3) organic coolant, 4) primary containment, 5) void--secondary containment is neglected, 6) TREAT fuel. If the outer boundary is located sufficiently far from the fuel pin, a zero current boundary condition is established "naturally" so that the particular location of the boundary has no bearing on the flux distribution. Use of transport theory is necessitated here because diffusion theory cannot be expected to give good results in the fuel pin where the absorption rate is high. The results of S_n (approximation to transport theory) calculations are shown in Figures 2 through 6. Data has been normalized to the asymptotic value in TREAT fuel so that the flux disadvantage can be read directly. Details of the flux distributions in the experimental assembly are shown in Figure 7.

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. 5397
DATE June 21, 1960
PAGE 4 OF 17

The horizontal peak-to-average can be determined with the AIM V diffusion theory code using the multi-group cross-sections developed in the Appendix. The result of such a calculation is shown in Figure 8. The fuel pin was omitted in the ~~calculation~~ and the experimental channel was treated as a void. Neutron streaming through the channel was ignored but this is partially offset by fission neutrons produced in the fuel pin.

Although the vertical flux distribution was not calculated, a reasonable estimate can be obtained by ignoring control rod disturbances and using a chopped sine distribution. A vertical peak-to-average of 1.2 was reported in reference 1. Then the vertical flux can be described analytically as $\phi = 1.2 \cos(\pi/2L)X$ where L is the sum of core half-height and reflector savings and X is the vertical distance measured from the core midpoint. Using the expression for average flux,

$$\bar{\phi} = \frac{1.2 \int_0^{H/2} \cos\left(\frac{\pi}{2L} X\right) dX}{\int_0^{H/2} dX} = 1.0,$$

a reflector savings of 26.8 cm results. The analytic expression for flux is thus specified. This vertical distribution is plotted in Figure 9.

Sample Calculation

For illustrative purposes, the calculations for the U-10 Mo, natural uranium fuel will be shown. The following factors were obtained from an S_n calculation:

$$\frac{\text{Average flux in fuel pin}}{\text{Flux in TREAT fuel}} = 0.566$$

$$\frac{\text{Average flux in interior region}}{\text{Flux in TREAT fuel}} = 0.548$$

$$\frac{\text{Average flux in outer region}}{\text{Flux in TREAT fuel}} = 0.572$$

$$\frac{\text{Pin surface flux}}{\text{Pin average flux}} = 1.04$$

The microscopic fission cross-section for fuel with this enrichment is 0.1419. The calculated fission cross-section for TREAT is 0.00382. An AIM V calculation of the horizontal flux distribution in TREAT yields a peak-to-average of 1.28.

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. 5397
DATE June 21, 1960
PAGE 5 OF 17

$$\begin{aligned} \left(\begin{array}{l} \text{energy release} \\ \text{rate in fuel pin} \\ \text{per cm}^3 \end{array} \right) &= \left(\frac{\sum \text{fuel pin}}{\sum \text{TREAT}} \right) \left(\begin{array}{l} \text{pin disadvantage factor} \\ \\ \end{array} \right) \\ &\times \left(\begin{array}{l} \text{horizontal} \\ \text{peak-to-average} \end{array} \right) \left(\begin{array}{l} \text{average energy release} \\ \text{rate in TREAT per cm}^3 \end{array} \right) \\ &= \left(\frac{.1419}{.00382} \right) (.566) (1.28) \left(\frac{1,000 \text{ Mw-sec}}{2.83 \times 10^6 \text{ cm}^3} \right) \\ &= 0.0095 \text{ Mw-sec/cm}^3 \end{aligned}$$

V. REFERENCES

1. ANL-5923, "Hazards Summary Report on the Transient Reactor Test Facility (TREAT)", D. R. MacFarlane, G. A. Freund, and J. F. Boland.
2. ANL-6025, "Physics Analyses of the TREAT Reactor Design," H. P. Iskenderian, August, 1959.
3. A/CONF. 15/P/1848, "TREAT, A Pulsed Graphite-Moderated Reactor for Kinetic Experiments," G. A. Freund, H. P. Iskenderian, and D. Okrent, Second U. N. Int. Conf. on the Peaceful Uses of Atomic Energy, June, 1958.

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. 5397
DATE June 21, 1960
PAGE 6 OF 17

APPENDIX

Calculation of the TREAT Core

The physics analysis of the TREAT reactor described by Haig Iskenderian (reference 2) was directed primarily toward the calculation of reactivity and was not readily applicable to the problem here. Therefore new nuclear constants which are consistent with the normal practice and available tools at AI were developed. Use was made of references 1, 2, and 3 to obtain a physical description of the TREAT core and to substantiate calculations wherever possible. The analysis is reported in detail here since it may be useful in subsequent studies of TREAT reactor experiments.

TREAT is a nominal 5' x 5' x 4' high core surrounded by an effectively infinite graphite reflector. The Zircaloy-3 clad fuel elements are 4" x 4" in cross-section. The fuel itself is a homogeneous mixture of uranium oxide and graphite fabricated by die-pressing and baking. The reactor is cooled by air flowing in channels between fuel elements. Slots and test holes are provided for experimental equipment.

Typical core composition (no slots or test holes) is characterized by a density of 1.71 and the following volume fractions:

Graphite + U	.867
Zircaloy	.033
Void	.100
	<u>1.000</u>

The properties of Zircaloy-2 were substituted for Zircaloy-3 since they were readily available:

$$\sigma_a (2200 \text{ m}) = 0.212 \text{ b} \quad \rho = 6.57$$

$$\sigma_s (\text{thermal}) = 8.0 \text{ b}$$

The density of the graphite-uranium was eventually located in the Argonne Laboratory literature but may be deduced as follows:

$$V_{\text{Zr}} (\rho_{\text{Zr}}) + V_{\text{G+U}} (\rho_{\text{G+U}}) = 1.71$$

$$\rho_{\text{G+U}} = \frac{1.71 - (.033) (6.57)}{0.867} = 1.72$$

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. 5397
DATE June 21, 1960
PAGE 7 OF 17

Reference 3 states that 93.1 percent enriched uranium was added to the uranium-graphite mixture as 0.248 w/o U_3O_8 . Then

$$\rho_U = 1.72 \times 0.00248 \times \left(\frac{705}{833} \right) = 0.00362 \text{ gm/cm}^3$$

The volume of a 5' x 5' x 4' core is $2.83 \times 10^6 \text{ cm}^3$.

The loading of U^{235} is

$$0.00362 \times 2.83 \times 0.867 \times 0.931 = 8.35 \text{ kg.}$$

This can be compared to the figure 8.44 kg quoted in reference 1. Using the calculated value

$$\rho_{U^{235}} = \frac{8.35 \times 10^3 \text{ gm}}{2.83 \times 10^6 \text{ cm}^3} = 0.00295 \text{ gm/cm}^3.$$

The number density of U^{235} can now be computed

$$N_{U^{235}} = \frac{0.00295 \text{ gm/cm}^3}{235.1 \text{ amu/atom} \times 1.66 \times 10^{-24} \text{ gm/amu}} = 0.00000756 \times 10^{24} \text{ atoms/cm}^3$$

Using a carbon to U^{235} atom ratio of 10,000 to 1, from reference 1,

$$N_C = 0.0756 \times 10^{24}.$$

Also,

$$N_{U^{238}} \approx \frac{6.9}{93.1} = 0.00000056 \times 10^{24} \text{ atoms/cm}^3.$$

The Muft IV code was used to compute core diffusion constants for the energy range 0.625 ev to 10 Mev. Zirconium and all impurities were neglected. The constants were determined as follows:

$$D_1 = 1.274$$

$$\Sigma_{a_1} = 0.000162$$

$$\Sigma_R = 0.002946$$

$$\nu \Sigma_{f1} = 0.000260$$

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. 5397
DATE June 21, 1960
PAGE 8 OF 17

The following microscopic thermal cross-sections were used:

Material	σ_a	σ_f	σ_s	$\bar{\mu}$	ν
Carbon	0.00728	---	4.8	0.05556	---
Zr	0.212	---	8.0	0.00732	---
U ²³⁵	603.0	505.0	10.0	0.00284	2.47
U ²³⁸	2.4	---	8.3	0.0028	---

The resulting microscopic cross-sections are:

$$D_2 = 0.925 \quad \Sigma_{a2} = 0.005425$$

$$\Sigma_{s2} = 0.3748 \quad \bar{\mu} \Sigma_s = 0.02027$$

Diffusion constants for the reflector were developed from some of the Hallam data by eliminating steel and impurities:

$$D_1 = 1.1612 \quad \Sigma_{a1} = 0.0000531$$

$$\Sigma_R = 0.00345$$

$$D_2 = 0.8425 \quad \Sigma_{a2} = 0.00061$$

The horizontal neutron flux distribution was computed with the aid of the AIM V diffusion theory code. It was considered better to cylindricize the outer boundary with cross-sectional area preserved than to solve the slab problem with its strong sensitivity to transverse buckling. Accordingly, the core was divided into three circular regions: a 10.15 cm radius void, an 86 cm radius core region, and a reflector region.

A partial measure of the success of this analysis is the comparison of the $k_{eff} = 1.257$ resulting here and $k_{eff} = 1.194$ which Iskenderian calculated. The difference is quite reasonable considering the uncertainties involved. Some of the factors causing the discrepancy are: failure to account for neutron streaming through the experimental port; uncertainty regarding the actual U²³⁵ density; failure to include the boron impurity; and inadequate treatment of neutron leakage resulting from cylindricizing the core geometry. The results should, however, be valid to within 20 percent, which is sufficiently reliable for the present purpose.

PREPARED BY: <i>J. Hume</i>	ATOMICS INTERNATIONAL A DIVISION OF NORTH AMERICAN AVIATION, INC.	PAGE NO. 9 of 17
CHECKED BY:		REPORT NO. TDR 5397
DATE:	EQUIP. SKETCH	MODEL NO.

FIGURE 1

FIGURE 2

FIGURE 3

FIGURE 4

FIGURE 6

FIGURE 4

FIGURE 7

Estimated Power Distribution
for Fuel Pin in TREAT

FIGURE 3