AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

**Cover Sheet for a Hanford
Historical Document
Released for Public Availability**

Released 1994

**Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute**

DISCLAIMER

This is a **historical document** that is being released for public availability. This was made from the best available copy. Neither the United States Government nor any agency thereof, nor Battelle Memorial Institute, 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.** The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DECLASSIFIED

HW-63904
Page 1

1. GF Bailey - AR Kosmata
2. CE Bowers - RD Carter
3. WL Bunch
4. RA Chitwood - WR Smit
5. JE Faulkner
6. FC Franklin - AD Vaughn
7. GC Fullmer
8. FE Dearing - EC Bennett
9. RO Gumprecht
10. JE Hanson - RE Baars
11. CG Lewis - FW Van Wormer
12. BI Monnie - WL Stiede
13. WC Morgan ←
14. WS Hechodom
15. R Nilson
16. GF Owsley
17. BE Simpson
18. RE Trumble
19. 300 File
20. Record Copy

This document consists
of 9 pages. No.
copies. Serie

This document classified
by FW Van Wormer

ENERGY RELEASE PER FISSION IN THE HANFORD REACTORS

Classification Cancelled and Changed To

DECLASSIFIED

By Authority of CG-PR-2
WASnyder, 1-11-94.
J. M.oley, 1-28-94.
Verified By J. E. Savely 1-31-94

February 12, 1960

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

NOTICE

This report was prepared for use within General Electric Company in the course of work under Atomic Energy Commission Contract AT(40)-1(1352) and any views or opinions expressed in the report are those of the authors only. This report is subject to revision upon collection of additional data.

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 any 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.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

MASTER

ENERGY RELEASE PER FISSION IN THE HANFORD REACTORS

INTRODUCTION

The average energy release per fission event in a reactor is dependent on the composition and arrangement of the lattice materials. In a study of heat generation in the NPR, Nilson⁽¹⁾ developed expressions for calculating the average energy released in each material per fission event. These relationships have been used in the present calculations to obtain the energy release per fission in existing Hanford reactors.

SUMMARY

The calculated average energy release per fission, for present natural uranium fuel elements, is presented in Table I.

TABLE I

ENERGY RELEASE PER FISSION EVENT

<u>Reactor</u>	<u>Fuel Element</u>	<u>Mev/U²³⁵ fission</u>	<u>Mev/fission</u>
B, D, DR, F & H	O-II	215.5 $\pm$ 2.6	202.1 $\pm$ 3.0
C	C-II	215.1 $\pm$ 2.6	202.2 $\pm$ 3.0
KE & KW	K-III	215.4 $\pm$ 2.6	202.4 $\pm$ 3.0

The quoted uncertainties are based on experimental or calculated uncertainties in the parameters utilized in the calculations. No attempt has been made to estimate the uncertainties which might be associated with the method; the relationships are assumed to be correct and to account for all of the energy.

DISCUSSION

The method for calculating the average energy release per fission is developed in detail in reference 1. The equations are summarized below; energy sources are also discussed briefly for the convenience of the reader.

Table II is a recent compilation of the direct energy released per U²³⁵ fission event. The tabulated neutrino energy does not appear as sensible heat in the reactor; therefore, the total available energy is 194.0 $\pm$ 2.5 Mev per U²³⁵ fission event.

TABLE II

PRIMARY ENERGY RELEASE⁽²⁾

<u>Radiation</u>	<u>Energy Release per fission (Mev)</u>
Fission fragments	168.0 $\pm$ 2
Prompt neutrons	4.87 $\pm$ 0.6
Prompt gammas	7.2 $\pm$ 0.8
Delayed neutrons	0.0068 $\pm$ 0.0003
Isomeric gammas	0.24
Fission product gammas	5.7 $\pm$ 0.8
Fission product betas	8.0 $\pm$ 0.7
Fission product neutrino	10.5 $\pm$ 0.8
TOTAL	204.5 $\pm$ 2.5 Mev

DECLASSIFIED

HW-63904
Page 3

In addition to the direct energy release from fission, a potential energy source is associated with the extra neutrons emitted during fission. This secondary energy is released as the neutrons are absorbed and as the resultant radioactive nuclei decay. Nilson⁽¹⁾ has derived the relationships listed in Table III to account for these secondary sources of energy as a function of the nuclear parameters of the reactor lattice. The symbols which are used are defined in Tables IV and V. Table IV lists recent values for those parameters which are associated with natural uranium. Table V lists calculated values for the parameters which are a function of the composition and geometry of the lattice.

The values in Tables IV and V were used to evaluate the relationships of Table III. The results are presented in Table VI for each of the reactors. The energy contributed by the secondary sources is combined with the primary energy release to obtain the total energy release per fission.

The largest source of error in these calculations is due to the uncertainty associated with the direct energy release per fission event. Smaller errors are introduced because of the uncertainties in calculating ϵ and p ; all other errors are insignificant in comparison to the above.

If confidence limits of ± 0.003 and ± 0.03 are assigned to the calculated values of ϵ and p respectively, standard deviations of ± 0.21 Mev for the radiative neutron capture of U^{238} and ± 0.60 Mev for fast fissions in U^{238} are obtained. Including these values with the experimental uncertainties associated with the value of the primary energy release, leads to standard deviations of ± 2.6 Mev for the energy per U^{235} fission and ± 3.0 Mev for the energy per fission. These uncertainties assume that the relationships in Table III correctly account for all secondary energy sources.

It is shown in Appendix B that the resultant energy per fission is not strongly dependent upon the neutron temperature or equivalent graphite lattice assumptions which were made to obtain the relative thermal utilization values.

W. C. Morgan

W. L. Bunch

Irradiation Testing Operation
RESEARCH AND ENGINEERING OPERATION

WC Morgan:WL Bunch:mc

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.

TABLE III
SECONDARY ENERGY SOURCES

<u>Event</u>	<u>Events per U²³⁵ fission</u>	<u>Energy per Event</u>
<u>Radiative Neutron Capture</u>		
U ²³⁵		6.5 Mev γ
U ²³⁸ (thermal and resonance)	$(1 + \alpha_5) \frac{\sum 8/\sum 5 + \nu_5}{\sum 5} \in (1-p) \exp.(-\tau_B^2)$	4.7 Mev γ
Graphite	$(1 + \alpha_5) (1 + \sum 8/\sum 5) F_g/F_u$	5.0 Mev γ
Water	$(1 + \alpha_5) (1 + \sum 8/\sum 5) F_w/F_u$	2.24 Mev γ
Aluminum (Can and process tube)	$(1 + \alpha_5) (1 + \sum 8/\sum 5) F_{al}/F_u$	7.72 Mev γ
O ¹⁶ (n,p) N ¹⁶	negligible	~1 Mev (net)
<u>Capture Product Decay</u>		
U ²³⁹	Same as U ²³⁸ capture	0.074 Mev γ , 1.21 Mev β
Np ²³⁹	Same as U ²³⁸ capture	0.25 - 0.46 Mev γ , 0.3 - 0.47 Mev β
Al ²⁸	Same as Al capture	1.78 Mev γ , 2.87 Mev β
Fast fission in U ²³⁸	δ	194.0 Mev (see Table II)

DECLASSIFIED

TABLE IV
NUCLEAR PARAMETERS FOR NATURAL URANIUM (0.714% U²³⁵)

<u>Symbol</u>	<u>Definition</u>	<u>Value</u>	<u>Ref.</u>
α_5	Thermal capture to fission ratio for U ²³⁵	0.174 $\pm$ 0.010	3
σ_8	Microscopic thermal neutron absorption cross section for U ²³⁸	2.71 $\pm$ 0.02 b	4
σ_5	Microscopic thermal neutron absorption cross section for U ²³⁵	682. $\pm$ 3 b.	3
ν_5	Number of neutrons emitted per fission event in U ²³⁵	2.43 $\pm$ 0.03	3
Σ_8	Macroscopic thermal neutron absorption cross section for U ²³⁸	0.0741 $\pm$ 0.0005 cm ⁻¹	
Σ_5	Macroscopic thermal neutron absorption cross section for U ²³⁵	0.134 $\pm$ 0.001 cm ⁻¹	

TABLE V
CALCULATED PARAMETERS FOR CURRENT I&E FUEL LOADINGS

<u>Symbol</u>	<u>Definition</u>	<u>Value</u>			<u>Ref.</u>
		<u>B,D,DR,F&H</u>	<u>C</u>	<u>KE & KW</u>	
p	Resonance escape probability	0.8816	0.8825	0.8619	5
τ	Fermi age (cm ²)	348	337	321	5
B ²	Geometric buckling factor of pile lattice (cm ⁻²)	2.99x10 ⁻⁵	2.99x10 ⁻⁵	2.61x10 ⁻⁵	5
1 + δ	Number of fission events per U ²³⁵ fission ($\frac{\nu\epsilon - 1}{\nu - 1}$)	1.066	1.064	1.064	
ϵ	Number of fast fissions per thermal fission	1.039	1.038	1.038	5
F _g /F _u	Thermal utilization of graphite relative to uranium	0.05702	0.05693	0.04435	App. A
F _w /F _u	Thermal utilization of water relative to uranium	0.02516	0.03003	0.03736	"
F _{al} /F _u	Thermal utilization of aluminum relative to uranium	0.02587	0.02705	0.02805	"

DECLASSIFIED

DECLASSIFIED

TABLE VI

TOTAL ENERGY RELEASE

Event	Events per U ²³⁵ fission			Energy Release (Mev)		
	B,D,DR,F&H	C	KE & KW	B,D,DR,F&H	C	KE & KW
<u>Radiative Neutron Capture</u>						
U ²³⁵	0.174	0.174	0.174	1.13	1.13	1.13
U ²³⁸	0.9404	0.9381	0.9902	4.42	4.41	4.65
Graphite	0.1039	0.1038	0.0809	0.52	0.52	0.40
Water	0.0459	0.0547	0.0681	0.10	0.12	0.15
Aluminum	0.0472	0.0493	0.0511	0.36	0.38	0.39
<u>Capture Product Decay</u>						
U ²³⁹	0.9404	0.9381	0.9902	1.21	1.20	1.27
Np ²³⁹	0.9404	0.9381	0.9902	0.70	0.69	0.73
Al ²⁸	0.0472	0.0493	0.0511	0.22	0.23	0.24
<u>Fast fissions in U²³⁸</u>	0.066	0.064	0.064	12.80	12.42	12.42
<u>U²³⁵ fission</u>	1.000	1.000	1.000	194.0	194.0	194.0
Total Energy Release per U ²³⁵ fission				215.46	215.10	215.38
Average Energy Release per fission				<u>Energy/U²³⁵ fission</u>		
				1 + 8		
				202.12	202.16	202.42

APPENDIX A

Thermal Utilization

P-3 calculations were made to obtain the average neutron flux distribution in the reactor lattice using the existing Fortran program for the IBM-709(6). The assumptions made in the calculations include:

1. Infinite pile lattice and fuel element length.
2. All components of the lattice have an isotope distribution identical to the naturally occurring elements.
3. Graphite neutron temperature of 600 C.
4. Water, aluminum and fuel neutron temperatures of 300 C.
5. Cylindrical graphite lattice of the same area as the true (square) lattice.

Using the calculated relative flux distribution, the relative thermal utilizations were calculated from the following relationship:

$$F \sim \bar{\phi} A \Sigma_a$$

where F = the thermal utilization of a region.

$\bar{\phi}$ = the average flux in that region.

A = the cross sectional area of that region.

Σ_a = the macroscopic absorption cross section of the material in that region.

Absolute thermal utilizations were not obtained because only the ratio F_u/F_a is of interest in the calculations.

The cross sectional area of the fuel element was corrected for the volume of aluminum in the end caps. A correction introduced at this point in the calculations should approach more closely the true conditions than if a similar correction was made before neutron flux distributions were obtained.

During the above calculations the lattice was considered to consist of seven concentric regions:

<u>Region</u>	<u>Material</u>
1	Water in the fuel element spire
2	Aluminum spire
3	Uranium fuel
4	Aluminum fuel element can (exterior only)
5	Water in process tube (exterior of fuel element only)
6	Aluminum process tube
7	Graphite moderator

Thus, the values of Table VI are the sum of individual thermal utilization values:

$$F_g/F_u = F_7/F_3$$

$$F_w/F_u = (F_1 + F_5)/F_3$$

$$F_{al}/F_u = (F_2 + F_4 + F_6)/F_3$$

APPENDIX B

Additional P-3 calculations were made for K reactors to determine the dependence of the energy release on the assumed neutron temperature and on the assumed equivalent lattice for graphite.

TABLE VII

<u>Change Made in the Assumptions</u>	<u>Average Energy Release per Fission</u>
Original assumptions	202.42 Mev
Graphite neutron temperature of 700 C.	
Water, aluminum and uranium neutron temperature of 350 C.	202.39 Mev
Graphite neutron temperature of 500 C.	
Water, aluminum and uranium neutron temperature of 250 C.	202.48 Mev
Radius of graphite equivalent lattice increased $\frac{1}{2}$ inch.	202.45 Mev
Radius of graphite equivalent lattice decreased $\frac{1}{2}$ inch.	202.40 Mev
It is apparent that these assumptions do not significantly affect the results of the calculations.	

DECLASSIFIED

HW-63904
Page 9

REFERENCES

- (1) HW-59075, "NPR Heat Generation Rates During Operation," Nilson, R., February 13, 1959 (Secret).
- (2) CF-58-4-25, "U²³⁵ Thermal Fission," Maienschein, F. C., April 7, 1958 (Unclassified).
- (3) Nucleonics, v.17, p. 132-133, "New 'World-Average' Thermal Cross Sections," Hughes, Donald J., November 1959 (Unclassified).
- (4) BNL-325, Second Edition, "Neutron Cross Sections," Hughes, D. J. and Schwartz, R. B., July 1, 1958 (Unclassified).
- (5) Private Communication, Nechodom, W. S., February 4, 1960.
- (6) HW-60781, "Program on the IBM-709 Digital Computer of the P-3 Approximation to the Boltzmann Transport Equation in Cylindrical Geometry," Matsumoto, D. D. and Richey, C. R., June 17, 1959 (Unclassified).

DATE

FILMED

6/16/94

END