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RECOMMENDED VALUES OF DECAY HEAT POWER AND METHOD TO UTILIZE THE DATA

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Recommended Values of Decay Heat Power and Method to Utilize the Data

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Recommended values of fission-product decay heat power are presented for five fissioning systems; U-235, -238, Pu-239, -240 and Pu-241. These values are based on the summation calculations with the JNDC FP Nuclear Data Library Version 2 which was released and made open in 1990. The data are presented in two ways. One is tabular form and another is 33-term exponential fitting functions. The present report also gives the correction factors for the FP neutron-capture effect on decay heat and the energy-spectra data of gamma-ray component of decay heat. The contents of the present report are based on the work of the Decay Heat Evaluation Working Group of JNDC and the Committee on "Standardization of Decay Heat Power in Nuclear Reactors" in Atomic Energy Society of Japan.

Keywords: Decay Heat Power, Recommended Value, Exponential Fitting Function, Summation Calculation, Neutron Capture Effect, Gamma-Ray Energy Spectrum

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This report is dedicated to the late Dr. Shungo IIJIMA (1930.9.22 - 1990.11.14) who made remarkable and unforgettable contributions to the present activity.

崩壊熱の推奨値とその使用法

日本原子力研究所東海研究所シグマ研究委員会
田坂 完二[*1]・片倉 純一・吉田 正[*2]
加藤 敏郎[*1]・中嶋 龍三[*3]

（1991年2月5日受理）

核分裂生成物による崩壊熱の推奨値を五つの核分裂系（U-235, -238, Pu-239, -240 および Pu-241）に対し与えてある。これらの推奨値は1990年に公開となったJNDCによる核分裂生成物の核データライブラリー第二版を用いた総和計算に基づいたものである。推奨値は二種類の方法で表わされている。一つは表形式であり，もう一つは33項の指数関数表示である。本報告書には，また，FPによる崩壊熱への中性子吸収効果の補正因子，ガンマ線のエネルギースペクトルも与えられている。本報告書の内容はシグマ研究委員会の崩壊熱評価ワーキンググループおよび原子力学会の「原子炉崩壊熱基準」研究専門委員会の成果に基づいている。

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Foreword

Experimental and theoretical activities on the decay heat power study in Japan have been investigated, and the data have been evaluated to establish the decay heat power standard. Results were published in August 1989 as a report titled "Decay Heat Power in Nuclear Reactor and Its Recommended Values" (in Japanese). In that report, measurements and theoretical works are reviewed, the nuclear data required for calculation of the decay heat power are described, the results of the summation calculation are compared with the experimental data, and the recommended values are evaluated from the studies. The present status of the use of data of decay heat power at each nuclear power plant and the requirements for the standard data are also investigated in that report. After the publication of the main report, it was further considered useful to publish a data book as a supplement to use easily those recommended values, and the data have been compiled in a data book. It was also planned to store the data in a floppy diskette as well as programs for the calculation. The present data report "Recommended Values of Decay Heat Power and Methods to Utilize the Data" is the full English translation of the supplemental report mentioned above.

The contents of this report is based on Part I, Section IV "Recommended Values" of the main report. The report includes the description of FP decay heat power (formulae for calculation, neutron capture effect, evaluation of errors, FP gamma-ray spectra), formulae for calculation of the decay heat power of actinides, and tables of recommended values. An example of calculation for a typical case is shown in Appendix so that users can easily understand how to use this report. In the floppy diskette, contained are the fitting constants of an exponential function with 33 terms, the recommended values of decay heat power for a pulsed thermal fission, and infinite and one-year irradiations, computation programs for decay heat power, a computation program for neutron capture effect, the calculated FP gamma-ray spectra, the uncertainty of decay heat power, and the constants and default values of a typical case. The floppy diskette can be obtained upon the request to Dr. J. Katakura of JAERI.

This data report and the floppy diskette are based on the results of research work of the Decay Heat Evaluation Working Group of the Japanese Nuclear Data Committee (JNDC) and results of investigation of the Research Committee on "Standardization of Decay Heat Power in Nuclear Reactors."

People from various fields collaborated in the compilation of this report as well as members of the Research Committee. This collaboration is appreciated. On the next page, the membership of the Research Committee is placed.

We hope the present recommendation can be widely used in the design and operation of various nuclear plants.

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† Deceased, November 14th, 1990

1. Introduction

This instruction report is written for users who need to calculate the decay heat power from fission products. The decay heat power and the related quantities presented here have been evaluated and recommended by the Committee of Standardization of Decay Heat Power in Nuclear Reactors organized in the Atomic Energy Society of Japan.

The basic data on the decay heat power from fission products are based on the results of summation calculations by the Decay Heat Evaluation Working Group of the Japanese Nuclear Data Committee. The users may find the data necessary for their own purposes in Tables presented in this report or from a floppy diskette in which some computational programs are also stored.

2. Scope and Contents

The authors believe that the methods of evaluating the decay heat power described in this book are applicable to design and assessment of safety for light water reactors (both the boiling water and the pressurized water types) and fast breeder reactors. These are also useful in designing the residual-heat removal systems for handling or storing the spent fuel.

The recommended values of the decay heat power from fission products are tabulated (Tables 4.1 to 4.15) for thermal neutron fission of U-235, Pu-239 and Pu-241 and for fast neutron fission of U-238 and Pu-240, where the cooling time ranges from 0 to 10^{13} seconds (~3 x 10^5 years). Those tables include the values of beta, gamma and total decay heat power for an instantaneous burst fission, a one-year irradiation and an infinite irradiation. The decay heat power for an arbitrary reactor power history without neutron capture effect may be calculated using the parameters (Table 4.16) which best fit the values for the instantaneous burst fission by 33 terms of an exponential function.

The effect of neutron capture by fission products during the reactor operation is treated as a correction factor. The correction factors for the beta and the gamma decay heat power of U-235 and Pu-239 are given for various reactor operating conditions. Users may apply the correction factors for Pu-239 to those for U-238, Pu-240 and Pu-241.

The uncertainties in the recommended values of the decay heat power are given in tables and figures for the instantaneous burst fission and the infinite irradiation of U-235, Pu-239 and U-238.

The energy spectra of delayed gamma rays, following radioactive decays of aggregate fission products, are tabulated for a three-years irradiation of U-235, Pu-239 and U-238, where the normalized intensities per MeV of gamma rays are calculated for the cooling times up to 10^{13} seconds. The neutron capture effect on the gamma-ray spectra is also given.

The decay heat power from decays of actinide nuclides are not described here except for U-239 and Np-239, for which the expressions of decay heat power and values of necessary quantities are given. The decay heat power from activation products in structural materials and the fission power from delayed neutron fission are not considered in the present recommendation. Users are requested to evaluate these items by themselves.

3. Decay Heat Power

3.1 Definition of Terms

The following notation defines the terms to be used for the description of the recommended values of the decay heat power.

T'	: Time measured from initial reactor startup (s),
T	: Total operating period, including intermediate periods at zero power (s),
t	: Time after shutdown; cooling time (s),
∞	: Infinite time ($T = 10^{13}$ s for computational purposes),
$f_i^{\alpha}(t)$	: Decay heat power of type α (= β, γ or τ) from fissionable nuclide i at t seconds after a fission pulse (MeV/s/fission),
α	: An index specifying decay heat power type,
β	: An index specifying β decay heat power,
γ	: An index specifying γ decay heat power,
τ	: An index specifying (β+γ) decay heat power,
$\Delta f_i^{\alpha}(t)$	: One standard deviation in $f_i^{\alpha}(t)$ (MeV/s/fission),
$F_i^{\alpha}(T,t)$	: Decay heat power of type α at t seconds after an operating period of T seconds at constant fission rate of nuclide i in the absence of neutron capture in fission products (MeV/s)/(fission/s),
$\Delta F_i^{\alpha}(T,t)$	: One standard deviation of $F_i^{\alpha}(T,t)$ (MeV/s)/(fission/s),
Q_i	: Total recoverable energy associated with one fission of nuclide i (MeV/fission),
ΔQ_i	: One standard deviation in Q_i (MeV/fission),
P_i^k	: Average power from fissioning of nuclide i during operation period T_k (MeV/s),
ΔP_i^k	: One standard deviation in P_i^k (MeV/s),
k	: An index specifying an operating period at constant power,

$P_d^\alpha(T,t)$: Total fission product decay heat power of type α at t seconds after shutdown from an operating history of T seconds duration (MeV/s),

$P_{di}^\alpha(T,t)$: Fission product decay heat power contribution to $P_d^\alpha(T,t)$ by i-th fissionable nuclide (MeV/s),

d : An index specifying the decay heat power,

$R_d^\alpha(T,t)$: Total fission product decay heat power corresponding to $P_d^\alpha(T,t)$ but uncorrected for neutron capture in fission products (MeV/s),

$R_{di}^\alpha(T,t)$: Fission product decay heat power contribution to $R_d^\alpha(T,t)$ by i-th fissionable nuclide, uncorrected for neutron capture in fission products (MeV/s),

$P_i(T')$: The power generated by i-th fissionable nuclide at T'(MeV/s),

$P(T')$: The total power at $T' = \sum_i P_i(T')$ (MeV/s),

$\Delta P(T')$: One standard deviation in P(T') (MeV/s),

$G_i^\alpha(\psi,\phi,T,t)$: The factor which accounts for neutron capture in fission products for the fission of i-th fissionable nuclide, t seconds after shutdown from an operation of T seconds in the system with neutron spectrum ψ and neutron flux ϕ,

ϕ : neutron flux (n/cm^2/s).

3.2 Determining Decay Heat Power from $f_i^\alpha(t)$ or $F_i^\alpha(\infty,t)$

There are two convenient ways to express the decay heat power data of fission products. In the first method, the decay heat power data are given as $f_i^\alpha(t)$ for the instantaneous burst fission of each fissile nuclide i with the decay heat power type α (β, γ or $\beta+\gamma$) (Tables 4.1 through 4.5). The second method gives the decay heat power as $F_i^\alpha(\infty,t)$ for the infinite irradiation of each fissile nuclide i at a constant fission rate of 1 fission/s without a consideration for the neutron capture transformation effect of fission products on the decay heat power (Tables 4.6 through 4.10). Further, the data for a one-year irradiation are given in Tables 4.11 through 4.15 for user's convenience.

The decay heat power $f_i^\alpha(t)$ for an instantaneous burst fission was fitted to an exponential function with 33 terms and the fitted parameters are shown in Table 4.16.

The decay heat power $P_d^\alpha(T,t)$ after an arbitrary irradiation of period T can be expressed as follows by $R_d^\alpha(T,t)$ when the neutron capture effects can be neglected.

$$R_d^\alpha(T,t) = \sum_{i=1}^{5} R_{di}^\alpha(T,t), \qquad (3.2.1)$$

where i = 1 thermal neutron fission of ^{235}U,
2 fast neutron fission of ^{238}U,
3 thermal neutron fission of ^{239}Pu,
4 fast neutron fission of ^{240}Pu,
5 thermal neutron fission of ^{241}Pu,

With consideration of the neutron capture effect of fission products, the decay heat power $P_d^\alpha(T,t)$ can be expressed as follows by using the coefficients $G_i^\alpha(\psi,\phi,T,t)$ for the neutron capture effect, which are described in Sec. 3.3.

$$P_d^{\alpha}(T,t) = \sum_{i=1}^{5} R_{di}^{\alpha}(T,t)\cdot G_i^{\alpha}(\psi,\phi,T,t) \tag{3.2.2}$$

The decay heat power $R_{di}^{\alpha}(T,t)$ in Eq. (3.2.1) can be expressed by the instantaneous burst fission function $f_i^{\alpha}(t)$ as

$$R_{di}^{\alpha}(T,t) = \int_0^T \frac{P_i(T')}{Q_i} f_i^{\alpha}(t + T - T')\, dT', \tag{3.2.3}$$

or by the infinite irradiation function $F_i^{\alpha}(\infty,t)$ expressing the irradiation history by a histogram of N steps as

$$R_{di}^{\alpha}(T,t) = \sum_{k=1}^{N} \frac{P_i^k\, F_i^{\alpha}(T_k,t_k)}{Q_i}, \tag{3.2.4}$$

where

P_i^k = Fission power by the fissile nuclide i in the k-th time step of the operation histogram,

$$t_1 = t,\ t_2 = t+T_1,\cdots,\ t_N = t + \sum_{k=1}^{N-1} T_k, \tag{3.2.5}$$

$$T = \sum_{k=1}^{N} T_k\ , \tag{3.2.6}$$

$$F_i^{\alpha}(T_k,t_k) = F_i^{\alpha}(\infty,t_k) - F_i^{\alpha}(\infty,t_k + T_k). \tag{3.2.7}$$

The instantaneous burst fission function $f_i^{\alpha}(t)$ and the infinite irradiation function $F_i^{\alpha}(\infty,t)$ are given in Tables 4.1 through 4.10 or they can be calculated from the exponential fitting function as follows using the fitting parameters LAMBDA (λ) and ALPHA (a_i^{α}) in Table 4.16,

$$f_i^{\alpha}(t) = \sum_{i=1}^{33} a_i^{\alpha}\ e^{-\lambda_i t}\ , \tag{3.2.8}$$

$$F_i^{\alpha}(\infty,t) = \sum_{i=1}^{33} \frac{a_i^{\alpha}}{\lambda_i}\ e^{-\lambda_i t}\ . \tag{3.2.9}$$

From Eqs.(3.2.7) and (3.2.9), the finite irradiation decay heat power $F_i^{\alpha}(T,t)$ can be also expressed by the exponential fitting function as

$$F_i^{\alpha}(T,t) = \sum_{i=1}^{33} \frac{a_i^{\alpha}}{\lambda_i}\ e^{-\lambda_i t}\ (1-e^{-\lambda_i T})\ . \tag{3.2.10}$$

3.3 Effect of Neutron Capture in Fission Products

Long-life or stable fission-product nuclides produced in a reactor core may have large neutron-capture cross sections, and therefore, transform themselves to heavier nuclides and affect the decay heat power of fission products as a result. Especially the stable fission product nuclide Cs-133 is transformed by neutron capture into Cs-134, which emits strong gamma rays and increases the decay heat power considerably for the long cooling time period up to 10 years. The neutron capture effect can be expressed as the ratio of decay heat powers with and without consideration for the neutron capture effect,

$$G_i^{\alpha}(\psi,\phi,T,t) = P_{di}^{\alpha}(\psi,\phi,T,t) \,/\, P_{di}^{\alpha}(\psi,0,T,t). \qquad (3.3.1)$$

The neutron capture reaction rate λ_{eff} is necessary for the calculation of the neutron capture effect and it is given as the product of the effective neutron capture cross section σ_{eff} and the total thermal neutron flux ϕ_{th} for the thermal reactor system as

$$\lambda_{eff} = \sigma_{eff} \cdot \phi_{th} \,, \qquad (3.3.2a)$$

where,

$\sigma_{eff} = A \cdot g\sigma_0 + r \cdot RI$,

$A = (\pi \underline{T}_0 / 4\underline{T}_n)^{1/2}$,

$\underline{T}_0 = 293$ K,

$\underline{T}_n$: Neutron temperature,

$g = 1 + \frac{1}{A\sigma_0} \int_0^{0.625\,\mathrm{eV}} [\sigma(E_n) - \frac{\sigma_0 V_0}{V}]\, M(E_n;\underline{T}_n)\, dE_n$,

σ_0 : 2200 m/s cross section,

V_0 = 2200 m/s,

V = Neutron velocity,

$M(E_n;\underline{T}_n)$: Maxwell neutron spectrum for the neutron temperature T_n,

$r = \phi_{cpi}/\phi_{th}/(u_H - u_L)$ (epithermal index),

ϕ_{cpi} : Epithermal neutron flux (E_n = 0.625 eV~5.53 keV),

$(u_H - u_L) = 9.09$ (Lethargy width for the resonance region 0.625 eV~5.53 keV),

RI : Resonance integral.

For the fast reactor system, the capture reaction rate λ_{eff} is given as the product of an average neutron capture cross section σ_{av} and the total neutron flux ϕ as

$$\lambda_{eff} = \sigma_{av} \cdot \phi \qquad (3.3.2b)$$

The neutron capture reaction rate λ_{eff} can also be calculated by Eq. (3.3.2b) for other reactor system than the fast reactor if σ_{av} is a given quantity pertinent to that system.

The method is shown below to calculate the neutron capture effect G_i^α. The coefficients G_i^α ($\alpha=\beta$ or γ) are calculated as the ratio of decay heat powers calculated rigorously with ($\phi\neq0$) and without ($\phi=0$) consideration for the neutron capture effect for the varieties of neutron spectrum ψ(or reactor type), neutron flux ϕ, irradiation time T and cooling time t. The neutron capture effect G_i^α for the actual irradiation condition (ϕ, T, t) is obtained by a linear interpolation of the calculated results for ϕ, T and t after selecting the reactor type or the neutron spectrum ψ. The interpolation program CAPTCORR.BAS is given in the floppy diskette.

In the present work, the neutron capture effect G_i^α has been calculated for the following variables.

Reactor type or neutron spectrum ψ : PWR (r=0.21), BWR (r=0.14), FBR (cf. neutron spectrum in Table 3.3.1),

Fissile nuclide i : ^{235}U, ^{239}Pu (Use ^{239}Pu data for ^{238}U, ^{240}Pu and ^{241}Pu),

Neutron flux ϕ : PWR (0.00, 1.04, 5.20, 9.10, 20.8, 52.0, 104.0 [x 10^{13} th·n/cm^2/s]),
BWR (0.00, 0.52, 2.60, 5.30, 10.4, 26.0, 52.0 [x 10^{13} th·n/cm^2/s]),
FBR (0.00, 0.50, 2.50, 5.00, 10.0, 25.0, 50.0 [x 10^{15} n/cm^2/s]),

Irradiation time T : 0, 1, 3, 5 years,

Cooling time t :	0.0E+0,	1.0E+0	1.0E+1,	1.0E+2,	1.0E+3,
(sec)	1.0E+4,	2.0E+4,	5.0E+4,	1.0E+5,	2.0E+5,
	5.0E+5,	1.0E+6,	2.0E+6,	5.0E+6,	1.0E+7,
	1.5E+7,	2.0E+7,	3.0E+7,	4.0E+7,	5.0E+7,
	6.0E+7,	8.0E+7,	1.0E+8,	1.5E+8,	2.0E+8,
	3.0E+8,	4.0E+8,	5.0E+8,	6.0E+8,	8.0E+8,
	1.0E+9,	2.0E+9,	5.0E+9,	6.0E+9,	8.0E+9,
	1.0E+10,	1.5E+10,	2.0E+10,	3.0E+10,	4.0E+10,
	5.0E+10,	1.0E+11,	2.0E+11,	5.0E+11,	1.0E+12,
	2.0E+12,	5.0E+12,	1.0E+13.		

The neutron capture cross section data are supplemented in Appendix B. The neutron fluxes in typical light water reactors and fast breeder reactors are shown in Table 3.3.2. The neutron capture effects G(t) in Table 3.3.3 were calculated by using the irradiation data in Table 3.3.2. The present results in Table 3.3.2 correspond to realistic operation conditions, therefore, they are clearly smaller than the results in ANS 5.1 which correspond to the maximum capture effects.

Table 3.3.1 70-group normalized neutron spectrum in FBR demonstration reactor core

N	E(eV)	$\phi\Delta u$	N	E(eV)	$\phi\Delta u$	N	E(eV)	$\phi\Delta u$
1	+1.000E+07	+8.058E-04	26	+1.930E+04	+1.917E-02	51	+3.727E+01	+3.197E-07
2	+7.788E+06	+2.377E-03	27	+1.503E+04	+2.175E-02	52	+2.902E+01	+2.844E-07
3	+6.065E+06	+5.117E-03	28	+1.171E+04	+1.991E-02	53	+2.260E+01	+6.753E-08
4	+4.724E+06	+8.383E-03	29	+9.119E+03	+1.209E-02	54	+1.760E+01	+2.830E-08
5	+3.679E+06	+1.312E-02	30	+7.102E+03	+1.212E-02	55	+1.371E+01	+2.552E-08
6	+2.865E+06	+2.023E-02	31	+5.531E+03	+9.558E-03	56	+1.068E+01	+3.567E-08
7	+2.231E+06	+2.154E-02	32	+4.307E+03	+5.784E-03	57	+8.315E+00	+9.958E-09
8	+1.738E+06	+2.564E-02	33	+3.355E+03	+1.419E-03	58	+6.476E+00	+7.200E-09
9	+1.353E+06	+2.823E-02	34	+2.613E+03	+4.958E-03	59	+5.043E+00	+1.026E-08
10	+1.054E+06	+2.928E-02	35	+2.035E+03	+8.171E-03	60	+3.928E+00	+1.864E-08
11	+8.209E+05	+4.427E-02	36	+1.585E+03	+7.282E-03	61	+3.059E+00	+1.070E-08
12	+6.393E+05	+5.541E-02	37	+1.234E+03	+5.101E-03	62	+2.382E+00	+1.602E-08
13	+4.979E+05	+4.062E-02	38	+9.611E+02	+3.438E-03	63	+1.855E+00	+9.168E-09
14	+3.877E+05	+5.636E-02	39	+7.485E+02	+2.257E-03	64	+1.445E+00	+1.892E-09
15	+3.020E+05	+5.667E-02	40	+5.829E+02	+1.301E-03	65	+1.125E+00	+9.846E-12
16	+2.252E+05	+5.574E-02	41	+4.540E+02	+6.623E-04	66	+8.764E-01	+3.420E-10
17	+1.832E+05	+5.406E-02	42	+3.536E+02	+4.362E-04	67	+6.826E-01	+4.441E-10
18	+1.426E+05	+6.368E-02	43	+2.754E+02	+2.346E-04	68	+5.316E-01	+1.731E-10
19	+1.111E+05	+5.117E-02	44	+2.145E+02	+1.242E-04	69	+4.140E-01	+1.465E-11
20	+8.652E+04	+5.223E-02	45	+1.670E+02	+6.043E-05	70	+3.224E-01	+7.485E-10
21	+6.738E+04	+3.912E-02	46	+1.301E+02	+2.455E-05	71	+1.000E-05	
22	+5.248E+04	+4.238E-02	47	+1.013E+02	+1.213E-05			
23	+4.087E+04	+3.314E-02	48	+7.889E+01	+4.028E-06			
24	+3.183E+04	+3.377E-02	49	+6.144E+01	+1.917E-06			
25	+2.479E+04	+3.079E-02	50	+4.785E+01	+6.745E-07			

*) E corresponds to upper boundary energy of each group and the lethargy width Δu is constant 0.25 for each group. The neutron spectrum data are private information from Dr. H. Takano of JAERI.

Table 3.3.2 Neutron flux for capture effect calculation (Irradiation of 4 years)

	ϕ_{th}	ϕ_{epi}	r	T_n	Fissile
ANS-5.1	1.75×10^{14}	3.0×10^{14}	0.189	706 K	U-235
PWR	4.7×10^{13}	9.1×10^{13}	0.210	(858 K)	U-235
BWR	4.2×10^{13}	5.3×10^{13}	0.140	858 K	U-235
FBR	$\phi = 5.0 \times 10^{15}$				Pu-239

Total neutron flux : $\phi = 2.2 \times 10^{14}$ n/cm^2/s (BWR),
3.4×10^{14} n/cm^2/s (PWR).

Calculation conditions : ^{235}U enrichment:3.2%, Burnup:15 GWd/t, Power density:50.5 kW/l(BWR), 105 kW/l(PWR), Void fraction:40% (BWR), Boron concentration:500ppm(PWR).

Table 3.3.3 Calculated neutron capture effect G(t) for FP decay power

t(s)	ANS-5.1	PWR	BWR	FBR	t(s)	ANS-5.1	PWR	BWR	FBR
+1.0E+01	+1.022	+1.011	+1.009	+1.020	+2.0E+09	-	+1.000	+1.000	+0.996
+1.0E+02	+1.022	+1.010	+1.008	+1.021	+5.0E+09	-	+1.005	+1.004	+0.992
+1.0E+03	+1.033	+1.014	+1.011	+1.029	+6.0E+09	-	+1.009	+1.007	+0.990
+1.0E+04	+1.064	+1.028	+1.021	+1.050	+8.0E+09	-	+1.027	+1.021	+0.984
+1.0E+05	+1.124	+1.062	+1.052	+1.072	+1.0E+10	-	+1.072	+1.057	+0.969
+2.0E+05	+1.131	+1.067	+1.057	+1.077	+1.5E+10	-	+1.356	+1.280	+0.912
+1.0E+06	+1.124	+1.070	+1.058	+1.084	+2.0E+10	-	+1.136	+1.107	+0.939
+2.0E+06	+1.127	+1.079	+1.066	+1.084	+3.0E+10	-	+0.867	+0.896	+0.963
+5.0E+06	-	+1.104	+1.084	+1.079	+4.0E+10	-	+0.840	+0.874	+0.965
+1.0E+07	+1.181	+1.141	+1.112	+1.084	+5.0E+10	-	+0.838	+0.872	+0.965
+2.0E+07	+1.284	+1.220	+1.174	+1.104	+1.0E+11	-	+0.837	+0.872	+0.965
+4.0E+07	+1.444	+1.367	+1.290	+1.149	+2.0E+11	-	+0.836	+0.871	+0.965
+1.0E+08	+1.598	+1.442	+1.349	+1.232	+5.0E+11	-	+0.833	+0.869	+0.964
+1.5E+08	+1.498	+1.357	+1.282	+1.253	+1.0E+12	-	+0.829	+0.865	+0.963
+2.0E+08	+1.343	+1.240	+1.190	+1.214	+2.0E+12	-	+0.820	+0.857	+0.960
+3.0E+08	-	+1.093	+1.074	+1.113	+5.0E+12	-	+0.794	+0.834	+0.950
+4.0E+08	+1.065	+1.035	+1.028	+1.059	+1.0E+13	-	+0.753	+0.794	+0.934
+6.0E+08	+1.021	+1.005	+1.005	+1.024					
+1.0E+09	+1.007	+1.000	+1.000	+1.007					

3.4 Uncertainty Analysis

3.4.1 Method of Analysis

The burst function of decay heat power can be expressed as,

$$f(t) = \sum_m \lambda_m \cdot E_m \cdot N_m(t), \tag{3.4.1}$$

where N_m is the number of nuclide, λ_m a decay constant and E_m an average energy released per decay of nuclide m. The relative sensitivity coefficients $R(P_k,t)$ of the burst function f(t) to the parameters P_k, such as fission yield, decay constant and average energy, can be written as follows;

$$\begin{aligned} R(P_k,t) &= (\partial f(t)/f(t))/(\partial P_k/P_k) \\ &= \left(\frac{\partial f(t)}{\partial P_k}\right)\cdot\left(\frac{P_k}{f(t)}\right) \end{aligned} \tag{3.4.2}$$

Using the coefficients $R(P_k,t)$, the uncertainty (or the relative covariance matrix between time t and t') is expressed as follows;

$$\begin{aligned} M(t,t') &= \langle \Delta f(t) \Delta f(t') \rangle / f(t) f(t') \\ &= \sum_{k,k'} \langle R(P_k,t) R(P_k',t') \rangle \cdot \langle \Delta P_k \Delta P_k' / P_k P_k' \rangle \end{aligned} \tag{3.4.3}$$

where <--> denotes an average.
When we consider a linearized decay chain, the atom number of nuclide m at time t after a fission event (at t = 0) is written by the Batemann expression as,

$$\begin{aligned} N_m(t) &= \sum_{n=1}^{m} Y_n \cdot N_m^{(n)}(t) = \sum_{n=1}^{m} Y_n \sum_{k=n}^{m} P_{nm}(k) \cdot \exp(-\lambda_k t) \\ P_{nm}(k) &= \lambda_n \lambda_{n+1} \cdots \lambda_{m-1} / \prod_{j=n}^{m} (\lambda_j - \lambda_k) \\ &= 1, \quad \text{if } n = m \ (= k) \end{aligned} \tag{3.4.4}$$

where Y_n is the fission yield and λ_n the decay constant of nuclide n.

Thus the relative sensitivity coefficient $R(P_k,t)$ becomes

$$R(E_m,t) = \lambda_m \cdot E_m \cdot N_m(t)/f(t)$$

$$R(Y_n,t) = Y_n \sum_{m=n}^{n_{max}} \lambda_m \cdot E_m \cdot N_m^{(n)}(t)/f(t)$$

$$R(\lambda_k,t) = R(E_k,t) + \sum_{n=1}^{k} \sum_{m=k}^{n_{max}} Y_n \cdot \lambda_m \cdot E_m \cdot \lambda_k (\partial N_m^{(n)}(t)/\partial \lambda_k)/f(t) \qquad (3.4.5)$$

According to these expression, $R(E_m,t)$ gives the contribution of the average energy release E_m of nuclide m to the decay heat power f(t) at time t.

In the present uncertainty analysis, we applied the above expressions with some limitations: the mass chain was limited to the region of A = (83 - 110) and A = (126 - 153). The nuclides, which have a half-life shorter than 2 seconds, β^+ decay mode and/or delayed neutron emission mode, were ignored.

3.4.2 Uncertainty of Nuclear Data

① Uncertainty of energy release data

i) Uncertainty of estimated energy release data

The parameter Q_{00}[1] in the gross theory of beta decay[2], which has been used for estimating average decay energy release data for the nuclides with no or insufficient measured decay data, has been varied between 0.0 MeV and 2.0 MeV. The energy release values corresponding to those Q_{00} values are considered to be the lower and the upper limits of the average energy release. We assumed that the proper value exists between the limits with 99.6 % probability or with 3 times of standard deviation (σ). Thus,

$$\sigma_{\beta \text{ or } \gamma} = \frac{1}{6} \times \left| \bar{E}_{\beta \text{ or } \gamma}(Q_{00}=2.0) - \bar{E}_{\beta \text{ or } \gamma}(Q_{00}=0.0) \right| \qquad (3.4.6)$$

Calculations were performed for typical FP nuclides and the uncertainties for other nuclides were estimated by interpolation or extrapolation with Q_β.

ii) Uncertainty of measured energy release data
The evaluated uncertainty values of the first version of JNDC FP Decay Data file[3] were adopted, but those of ^{92}Y, ^{137m}Ba, ^{140}La, ^{144}Pr and ^{151}Sm were taken from ENSDF[4].

② Uncertainty of decay constant data
The evaluated uncertainty values of the first version of JNDC FP Decay Data File were used in the present work.

③ Uncertainty of fission yield data
The uncertainty values of fission yields were taken from the compilation by Rider and Meek[5] for ENDF/B-V. But the independent yield uncertainties were renormalized by those of mass chain yields because of too large unrealistic independent yield uncertainties for some nuclides.

3.4.3 Uncertainty of Decay Heat Power Calculation

The uncertainty of the decay heat power was calculated with Eq.(3.4.3) by ignoring correlations between the parameters, and between average beta and gamma energies, and between the fission yields of individual nuclides.

As mentioned in Sec. 3.4.1, the mass number and the half-life considered in the calculation were limited in order to simplify the calculation. This led to the underestimation of the calculated uncertainty values at short cooling times. In the uncertainties of energy, decay constant and fission yield, the energy one was separately calculated without any limitation, in which all FP nuclides with masses from A=66 through A=172 were taken into consideration. The comparison between the energy uncertainty calculated above way and the total one calculated by the simplified method with the limitation on the mass chain showed that at long cooling time region the total one was greater than the energy one as expected but at short cooling time region the total one became smaller than the energy one because of the limitation of the simplified calculation. Thus the excess of the total uncertainty at long cooling time region was extrapolated to compensate for the underestimation at short cooling time region.

The calculated uncertainty values for the instantaneous irradiation and for the infinite irradiation are shown in Figs. 3.4.1 and 3.4.2. The calculation method of uncertainty for a finite irradiation is described in Appendix D.

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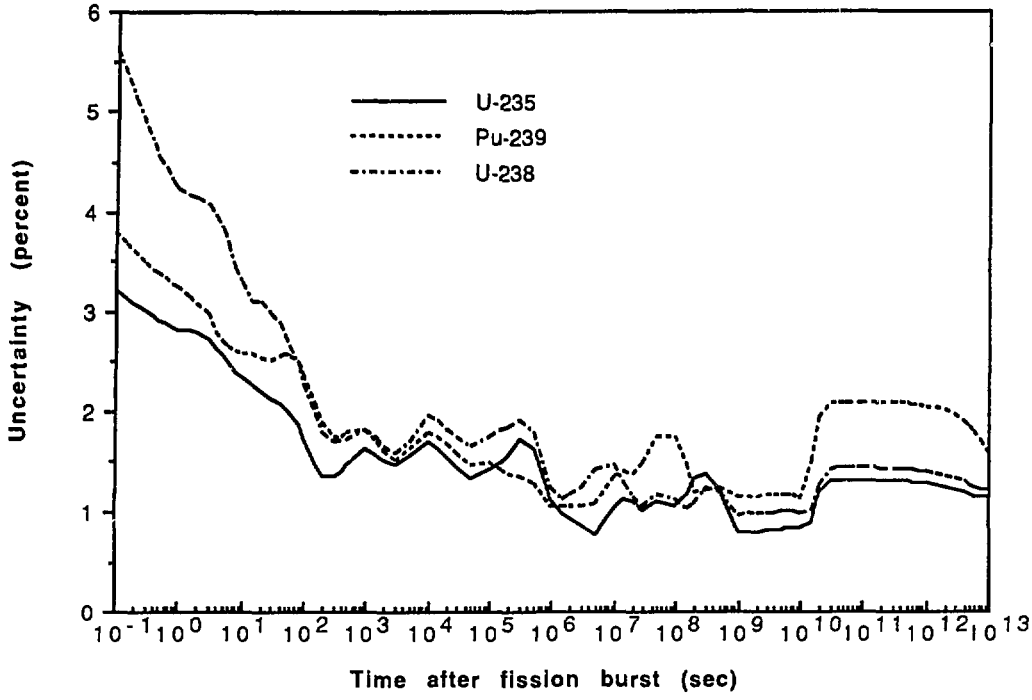

Fig. 3.4.1 Uncertainty of decay heat calculation for instantaneous irradiation

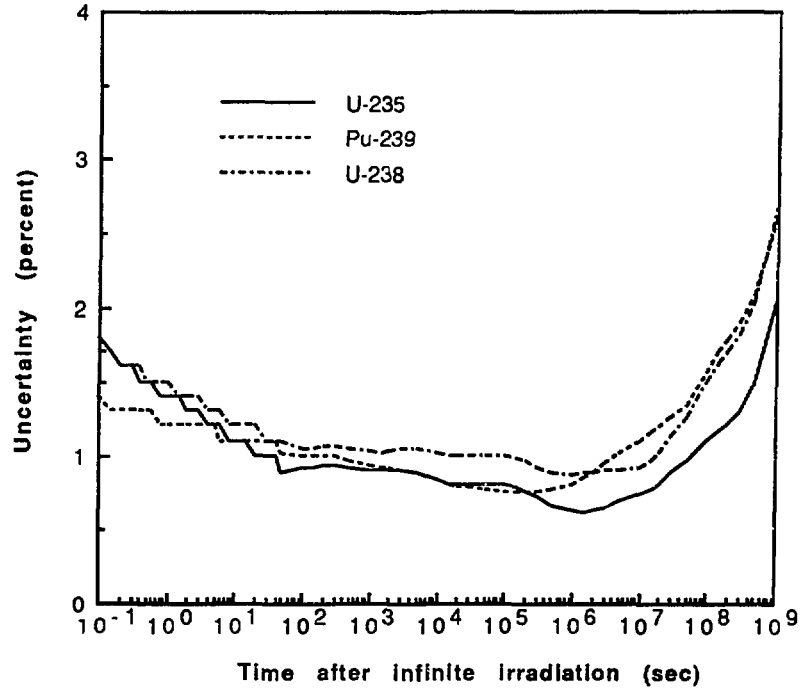

Fig. 3.4.2 Uncertainty of decay heat calculation for infinite irradiation

3.5 Delayed Gamma-Ray Energy Spectrum

The spectra of gamma rays emitted from aggregate fission products after a fission depend on a fissioning nuclide, irradiation time and cooling time. But the spectrum shape depends little on irradiation time[1] when the irradiation time is longer than one year.

If we prepare a set of normalized spectra for an irradiation time longer than one year, say three years, and for various cooling times, a spectrum at any cooling time can be accurately obtained by multiplying the normalized spectrum by the gamma-ray energy released at the cooling time.

$$S_\gamma(T,t;E) = F_\gamma(T,t) \times S_n(t;E), \qquad (3.5.1)$$

T= Irradiation time,
t = Cooling time,

$S_\gamma(T,t;E)$ = Gamma-ray energy spectrum,
$F_\gamma(T,t)$ = Released gamma ray energy,

$S_n(t;E)$ = Normalized gamma-ray energy spectrum ($\int_0^\infty S_n(t;E)dE = 1.0$).

The gamma-ray energy release $F_\gamma(T,t)$ is calculated as described in Section 3.2.

The normalized gamma-ray energy spectra of ^{235}U, ^{238}U and ^{239}Pu fissions are given in Tables 3.5.1 through 3.5.3 for each decade cooling time with an energy bin of 1 MeV from 0 to 10 MeV. A spectrum at any cooling time is obtained by an interpolation. The interpolated values should be renormalized. The tabulated spectra should be used for a fuel irradiated for more than one year. The interpolation method will be described later in this Section.

Comparison of delayed and prompt[2] gamma-ray energy spectra are shown in Figs. 3.5.1 and 3.5.2 for ^{235}U and ^{239}Pu fissions, respectively. The delayed gamma-ray spectra in the figures are those for irradiation of three years and they are normalized with the total energy release for the comparison with the prompt spectra.

Some nuclides produced through the neutron capture reaction affect the decay heat power (Capture effect). The nuclide ^{134}Cs is known as the major one whose energy release has the largest contribution to the decay heat power at cooling times between 10^4 and 10^9 s. The capture effect is described in detail in Section 3.3, where the capture effect is calculated for various irradiation times, cooling times, neutron flux levels and reactor types. The calculated results are provided for beta, gamma and total decay heat values. They are stored in a floppy diskette* in tabular forms. Using the table, the capture effect $G_\gamma(T,t)$ for the gamma component at cooling time t with irradiation time T can be obtained. Although $G_\gamma(T,t)$ includes the contribution from all of the nuclides affecting the capture reaction, ^{134}Cs is regarded as the major contributor to the gamma-ray spectrum capture effect at cooling times between 10^4 and 10^9 s and few nuclides emit gamma rays with the energy higher than 2 MeV. Therefore, the gamma-ray spectra from the capture effect in this cooling time region can be replaced with that for ^{134}Cs.

The spectrum including the capture effect, $S_c(T,t;E)$, is expressed as,

$$\begin{aligned} S_c(T,t;E) &= F_\gamma(T,t) \times S_n(t;E) + \left(G_\gamma(T,t) - 1.0\right) \times F_\gamma(T,t) \times S_{Cs}(E) \\ &= F_\gamma(T,t) \times \left[S_n(t;E) + \left(G_\gamma(T,t) - 1.0\right) \times S_{Cs}(E)\right]. \end{aligned} \tag{3.5.2}$$

where $S_{Cs}(E)$ is the normalized spectrum of ^{134}Cs, which has the value of 0.9531 for the first energy group from 0 to 1 MeV, 0.04689 for the second group from 1 to 2 MeV and 0.0 for the groups with energy above 2 MeV.

The number of ^{134}Cs can be also obtained directly by the following equation.

$$N_{Cs}(T,t) = F \cdot Y \cdot \left[\frac{1-\exp[-(\lambda_4+\sigma_4\Phi)T]}{\lambda_4+\sigma_4\Phi} + \frac{\exp(-\sigma_3\Phi T) - \exp[-(\lambda_4+\sigma_4\Phi)T]}{\sigma_3\Phi-(\lambda_4+\sigma_4\Phi)}\right] \cdot \exp(-\lambda_4 t), \tag{3.5.3}$$

where

* A copy of the floppy diskette is available upon request. See Appendix D

T = Irradiation time,
t = Cooling time,
λ_4 = Decay constant of ^{134}Cs nuclide (1.0652×10^{-8} sec^{-1}),
σ_4 = Capture cross section of ^{134}Cs nuclide,
σ_3 = Capture cross section of ^{133}Cs nuclide,
Φ = Neutron flux,
F = Fission rate (Fissions/sec),
Y = Cumulative fission yield of ^{133}Cs nuclide.

In the above expression, the nuclides preceding ^{133}Cs in the linear chain are ignored. In the case of using that expression, the gamma-ray energy spectrum taking only the ^{134}Cs effect into account is written as,

$$S_C(T,t;E) = F_\gamma(T,t) \times S_n(t,E) + \frac{\lambda_4 \bar{E}_\gamma}{F} \times N_{Cs}(T,t) \times S_{Cs}(E), \quad (3.5.4)$$

where $\bar{E}_\gamma$ is average gamma-ray energy release per one decay of ^{134}Cs.

Interpolation[3)]

When a set of data (X_j, Y_j, Y_j') $(j = 1,2,\ldots,n)$ is given, the value of Y corresponding to the value of X between X_j and X_{j+1} can be obtained by interpolation as described below. The variable Y_j' expresses the derivative at the point j. When Y_j' is not explicitly given, its calculation is necessary as described later.

Before getting the final value of Y, the following calculations are performed.

$$S_j = \frac{Y_{j+1} - Y_j}{X_{j+1} - X_j}$$
$$Y_0 = Y_j + S_j \times (X - X_j)$$
$$\Delta Y_j = Y_j + Y_j' \times (X - X_j) - Y_0$$
$$\Delta Y_{j+1} = Y_{j+1} + Y_{j+1}' \times (X - X_{j+1}) - Y_0$$

The interpolation is classified into three cases according to the sign of $\Delta Y_j \Delta Y_{j+1}$.

① $\Delta Y_j \Delta Y_{j+1} = 0$

$$Y = Y_0$$

② $\Delta Y_j \Delta Y_{j+1} > 0$

$$Y = Y_0 + \frac{\Delta Y_j \Delta Y_{j+1}}{\Delta Y_j + \Delta Y_{j+1}}$$

③ $\Delta Y_j \Delta Y_{j+1} < 0$

$$Y = Y_0 + \frac{\Delta Y_j \Delta Y_{j+1}(X - X_j + X - X_{j+1})}{(\Delta Y_j - \Delta Y_{j+1})(X_{j+1} - X_j)}$$

<u>Calculation of Y_j'</u>

① $i < j < k$

$$Y_j' = \frac{(Y_j - Y_i)\left\{(X_k - X_j)^2 + (Y_k - Y_j)^2\right\} + (Y_k - Y_j)\left\{(X_j - X_i)^2 + (Y_j - Y_i)^2\right\}}{(X_j - X_i)\left\{(X_k - X_j)^2 + (Y_k - Y_j)^2\right\} + (X_k - X_j)\left\{(X_j - X_i)^2 + (Y_j - Y_i)^2\right\}}.$$

② If i or k is an end point,

i) when $S > 0$ and $S > Y_m'$, or $S < 0$ and $S < Y_m'$,

$$Y_m' = S + (S - Y_j') = 2S - Y_j' \quad ,$$

ii) In the other cases,

$$Y_m' = S + \frac{|S|(S - Y_j')}{|S| + |S - Y_j'|} \quad .$$

Notice: The time coordinate should be a logarithm, that is, Log(t) when the interpolation of the spectra in Tables 3.5.1 through 3.5.3 is performed, because it is better to take spacing as equal as possible. When the extrapolation to the region between 0 and 0.1 s is needed, a simple linear interpolation

may be satisfactory because the change of spectral values in the region is enough small in each energy group to allow such interpolation.

References

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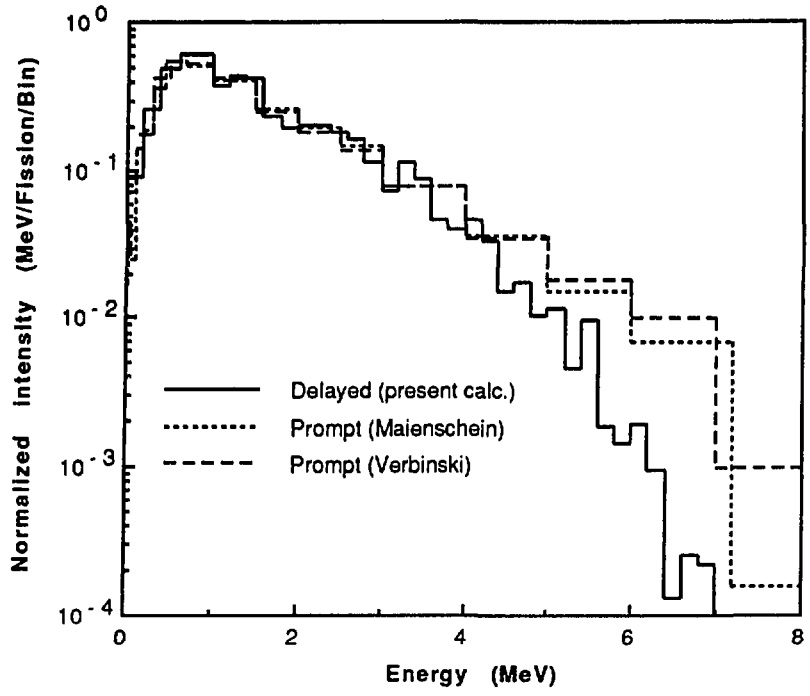


Fig. 3.5.1 Comparison of gamma-ray energy spectra of U-235 (delayed and prompt)

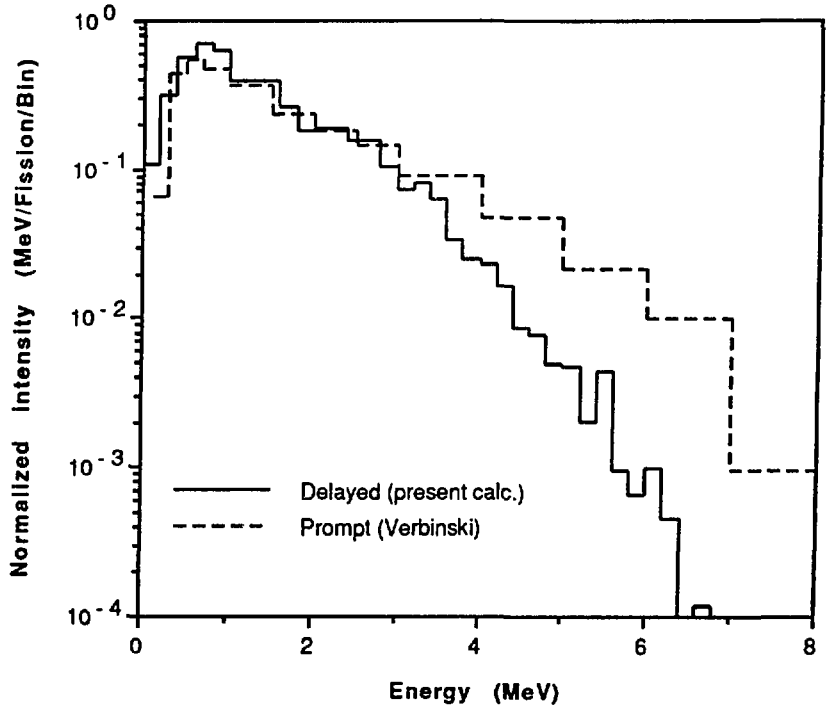


Fig. 3.5.2 Comparison of gamma-ray energy spectra of Pu-239 (delayed and prompt)

Table 3.5.1 Normalized Gamma-Ray Energy Spectrua of ^{235}U

Energy (MeV)	Time After Shutdown (sec) 0.0	1.0E-01	1.0E+00	1.0E+01	1.0E+02	1.0E+03	1.0E+04
0.0- 1.0	3.989E-01	3.996E-01	4.042E-01	4.214E-01	4.685E-01	5.253E-01	6.159E-01
1.0- 2.0	3.287E-01	3.287E-01	3.287E-01	3.304E-01	3.307E-01	3.333E-01	3.081E-01
2.0- 3.0	1.713E-01	1.710E-01	1.692E-01	1.626E-01	1.414E-01	1.228E-01	7.006E-02
3.0- 4.0	7.043E-02	7.011E-02	6.783E-02	5.690E-02	4.010E-02	1.777E-02	5.688E-03
4.0- 5.0	2.426E-02	2.417E-02	2.367E-02	2.255E-02	1.599E-02	8.299E-04	2.046E-04
5.0- 6.0	5.752E-03	5.746E-03	5.741E-03	5.479E-03	3.045E-03	4.121E-05	3.816E-16
6.0- 7.0	6.871E-04	6.816E-04	6.608E-04	6.238E-04	3.270E-04	1.193E-07	2.104E-27
7.0- 8.0	1.168E-05	1.050E-05	5.378E-06	4.380E-07	2.732E-10	3.888E-36	0.0
8.0- 9.0	9.479E-07	8.245E-07	3.351E-07	6.985E-09	4.241E-12	6.038E-38	0.0
9.0-10.0	5.761E-08	4.972E-08	1.892E-08	1.950E-11	1.956E-30	0.0	0.0

Energy (MeV)	Time After Shutdown (sec) 1.0E+05	1.0E+06	1.0E+07	1.0E+08	1.0E+09	1.0E+10	1.0E+11
0.0- 1.0	7.240E-01	7.016E-01	9.755E-01	9.694E-01	1.000E+00	9.997E-01	9.821E-01
1.0- 2.0	2.565E-01	2.737E-01	1.065E-02	1.008E-02	7.678E-07	2.914E-04	1.787E-02
2.0- 3.0	1.944E-02	2.451E-02	1.384E-02	2.047E-02	1.025E-07	1.942E-06	1.519E-11
3.0- 4.0	9.153E-05	1.454E-04	5.359E-06	8.030E-06	1.230E-08	2.331E-07	4.038E-12
4.0- 5.0	4.000E-07	1.809E-12	8.678E-12	1.594E-10	2.392E-10	4.534E-09	6.910E-14
5.0- 6.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
6.0- 7.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
7.0- 8.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
8.0- 9.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
9.0-10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Energy (MeV)	Time (sec) 1.0E+12	1.0E+13
0.0- 1.0	9.821E-01	9.821E-01
1.0- 2.0	1.787E-02	1.786E-02
2.0- 3.0	1.688E-11	4.841E-11
3.0- 4.0	4.487E-12	1.287E-11
4.0- 5.0	7.679E-14	2.203E-13
5.0- 6.0	0.0	0.0
6.0- 7.0	0.0	0.0
7.0- 8.0	0.0	0.0
8.0- 9.0	0.0	0.0
9.0-10.0	0.0	0.0

Table 3.5.2 Normalized Gamma-Ray Energy Spectra of ^{239}Pu

Energy (MeV)	Time After Shutdown (sec) 0.0	1.0E-01	1.0E+00	1.0E+01	1.0E+02	1.0E+03	1.0E+04
0.0- 1.0	4.517E-01	4.522E-01	4.564E-01	4.744E-01	5.226E-01	5.747E-01	6.684E-01
1.0- 2.0	3.210E-01	3.209E-01	3.203E-01	3.188E-01	3.129E-01	3.071E-01	2.776E-01
2.0- 3.0	1.575E-01	1.573E-01	1.564E-01	1.505E-01	1.269E-01	1.013E-01	4.915E-02
3.0- 4.0	5.480E-02	5.455E-02	5.269E-02	4.409E-02	3.023E-02	1.638E-02	4.737E-03
4.0- 5.0	1.215E-02	1.205E-02	1.143E-02	9.720E-03	6.142E-03	4.264E-04	9.986E-05
5.0- 6.0	2.561E-03	2.551E-03	2.494E-03	2.199E-03	1.136E-03	1.429E-05	2.166E-16
6.0- 7.0	3.440E-04	3.416E-04	3.292E-04	2.929E-04	1.499E-04	6.887E-08	5.467E-27
7.0- 8.0	4.571E-06	4.233E-06	2.718E-06	4.803E-07	1.052E-10	1.443E-36	0.0
8.0- 9.0	1.770E-07	1.510E-07	6.438E-08	7.393E-09	1.633E-12	2.241E-38	0.0
9.0-10.0	8.019E-09	6.456E-09	1.703E-09	1.745E-12	7.522E-31	0.0	0.0

Energy (MeV)	Time After Shutdown (sec) 1.0E+05	1.0E+06	1.0E+07	1.0E+08	1.0E+09	1.0E+10	1.0E+11
0.0- 1.0	7.439E-01	7.170E-01	9.617E-01	9.484E-01	1.000E+00	9.987E-01	9.821E-01
1.0- 2.0	2.379E-01	2.596E-01	2.638E-02	3.818E-02	8.957E-06	1.241E-03	1.787E-02
2.0- 3.0	1.815E-02	2.326E-02	1.184E-02	1.333E-02	3.524E-07	6.322E-06	3.156E-12
3.0- 4.0	8.026E-05	1.400E-04	3.502E-05	6.080E-05	4.230E-08	7.589E-07	8.390E-13
4.0- 5.0	-9.367E-08	6.906E-12	3.355E-11	4.187E-10	8.225E-10	1.476E-08	1.436E-14
5.0- 6.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
6.0- 7.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
7.0- 8.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
8.0- 9.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
9.0-10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Energy (MeV)	Time (sec) 1.0E+12	1.0E+13
0.0- 1.0	9.821E-01	9.821E-01
1.0- 2.0	1.787E-02	1.787E-02
2.0- 3.0	3.507E-12	1.006E-11
3.0- 4.0	9.324E-13	2.675E-12
4.0- 5.0	1.596E-14	4.579E-14
5.0- 6.0	0.0	0.0
6.0- 7.0	0.0	0.0
7.0- 8.0	0.0	0.0
8.0- 9.0	0.0	0.0
9.0-10.0	0.0	0.0

Table 3.5.3 Normalized Gamma-Ray Energy Spectra of ^{238}U

Energy (MeV)	Time After Shutdown (sec) 0.0	1.0E-01	1.0E+00	1.0E+01	1.0E+02	1.0E+03	1.0E+04
0.0- 1.0	3.977E-01	3.992E-01	4.081E-01	4.385E-01	4.963E-01	5.561E-01	6.455E-01
1.0- 2.0	3.157E-01	3.157E-01	3.158E-01	3.193E-01	3.218E-01	3.207E-01	2.956E-01
2.0- 3.0	1.860E-01	1.852E-01	1.808E-01	1.669E-01	1.349E-01	1.060E-01	5.416E-02
3.0- 4.0	7.530E-02	7.484E-02	7.155E-02	5.460E-02	3.367E-02	1.663E-02	4.628E-03
4.0- 5.0	2.026E-02	1.998E-02	1.892E-02	1.598E-02	1.069E-02	5.471E-04	1.331E-04
5.0- 6.0	4.361E-03	4.332E-03	4.249E-03	4.034E-03	2.251E-03	2.492E-05	1.526E-16
6.0- 7.0	6.737E-04	6.636E-04	6.306E-04	6.185E-04	3.908E-04	2.611E-07	6.007E-27
7.0- 8.0	1.965E-05	1.753E-05	9.391E-06	1.347E-06	7.982E-10	1.182E-35	0.0
8.0- 9.0	1.404E-06	1.179E-06	4.229E-07	2.098E-08	1.239E-11	1.836E-37	0.0
9.0-10.0	8.268E-08	6.828E-08	2.093E-08	2.179E-11	1.397E-30	0.0	0.0

Energy (MeV)	Time After Shutdown (sec) 1.0E+05	1.0E+06	1.0E+07	1.0E+08	1.0E+09	1.0E+10	1.0E+11
0.0- 1.0	7.325E-01	7.032E-01	9.660E-01	9.535E-01	1.000E+00	9.997E-01	9.821E-01
1.0- 2.0	2.484E-01	2.722E-01	2.033E-02	2.945E-02	8.332E-07	3.446E-04	1.787E-02
2.0- 3.0	1.898E-02	2.440E-02	1.361E-02	1.699E-02	1.733E-07	3.275E-06	1.342E-11
3.0- 4.0	8.536E-05	1.463E-04	2.264E-05	4.380E-05	2.080E-08	3.932E-07	3.568E-12
4.0- 5.0	4.846E-08	3.057E-12	1.553E-11	2.273E-10	4.046E-10	7.646E-09	6.106E-14
5.0- 6.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
6.0- 7.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
7.0- 8.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
8.0- 9.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
9.0-10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Energy (MeV)	Time (sec) 1.0E+12	1.0E+13
0.0- 1.0	9.821E-01	9.821E-01
1.0- 2.0	1.787E-02	1.785E-02
2.0- 3.0	1.491E-11	4.277E-11
3.0- 4.0	3.965E-12	1.137E-11
4.0- 5.0	6.785E-14	1.946E-13
5.0- 6.0	0.0	0.0
6.0- 7.0	0.0	0.0
7.0- 8.0	0.0	0.0
8.0- 9.0	0.0	0.0
9.0-10.0	0.0	0.0

3.6 Actinide Decay Heat Power

Primary contributors to the actinide decay heat power are beta and gamma rays emitted from nuclides U-239(23.47 m) and Np-239(2.355 d), which are produced by the neutron capture in U-238. This chain responsible for the decay heat in the cooling time shorter than 2 weeks after reactor shutdown. For the longer cooling times, alpha decay power from a variety of transuranium nuclides becomes the primary contributors to the actinide decay heat power.

The actinide decay heat power $\underline{P}(T,t)$ (MeV/fission) by U-239 (Index 1) and Np-239 (Index 2) at cooling time t after irradiation at 1 fission/s for irradiation time T can be obtained from the following expressions.

$$\underline{P}_1(T,t) = (C^0/\underline{F})E_1\cdot[1-\exp(-\lambda_1 T)]\exp(-\lambda_1 t), \qquad (3.6.1)$$

$$\underline{P}_2(T,t) = (C^0/\underline{F})E_2\cdot[1/(\lambda_1-\lambda_2)]\cdot[\lambda_1\{1-\exp(-\lambda_2 T)\}\exp(-\lambda_2 t) -\lambda_2\{1-\exp(-\lambda_1 T)\}\exp(-\lambda_1 t)]. \qquad (3.6.2)$$

where, λ_1, λ_2 : Beta decay constants,

$\lambda_1 = 4.922\times10\text{-}4\ s^{-1}$ (U-239),

$\lambda_2 = 3.406\times10\text{-}6\ s^{-1}$ (Np-239),

E_1, E_2 : Energy release by beta decay

$E_1 = 0.447$ MeV (β 88.4%, γ 11.6%) (U-239),

$E_2 = 0.426$ MeV (β 59.2%, γ 40.8%) (Np-239),

$C^0 = N^0\sigma_c\phi$: U-238 neutron capture rate (capture/s),

$\underline{F}$: Fission rate (fission/s).

The above data on decay constants and energy releases were obtained from Table of Radioactive Isotopes, E. Browne, R. B. Firestone, ed. by V. S. Shirley, Wiley-Interscience, 1986.

The actinide decay heat power is, thus, obtained as follows,

$$\underline{P}(\text{actinide}) = (P_0/Q_f)[\underline{P}_1(T,t) + \underline{P}_2(T,t)] \qquad (3.6.3)$$

for the operating power P_0 and the fission energy release Q_f (cf. Appendix A). The value of $(C_0/\underline{F})$ should be supplied and justified by the user, and P_0 should be taken as the maximum reactor power during the operating history of the last one week before shutdown.

4. Tables of Recommended Data on Decay Heat Power

Recommended data on the decay heat power of fission products are listed in Tables 4.1 through 4.15. The data for an instantaneous irradiation are given in Tables 4.1 through 4.5, those for an infinite irradiation in Tables 4.6 through 4.10, and those for one-year irradiation in Tables 4.11 through 4.15. The uncertainties of the recommended data are also given for the instantaneous irradiation and the infinite irradiation. The uncertainties for the fission of ^{235}U and ^{239}Pu are those evaluated by the method in Section 3.4, however, those for ^{240}Pu and ^{241}Pu are estimated as 1.5 times those of ^{239}Pu.

Coefficients of the fitting formula expressed by an exponential function with 33 terms are listed in Table 4.16 for the instantaneous irradiation.

Table 4.1
Tabular Data for Recommended Decay Heat Power for Pulse Thermal Fission of ^{235}U

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	7.690E-1	6.9	6.008E-1	9.9	1.370E+0	3.7
1.0E-1	7.039E-1	6.4	5.501E-1	9.4	1.254E+0	3.2
1.5E-1	6.763E-1	6.1	5.277E-1	9.1	1.204E+0	3.1
2.0E-1	6.511E-1	5.9	5.080E-1	8.8	1.159E+0	3.1
3.0E-1	6.064E-1	5.5	4.726E-1	8.5	1.079E+0	3.0
4.0E-1	5.682E-1	5.3	4.422E-1	8.4	1.010E+0	2.9
5.0E-1	5.351E-1	5.0	4.157E-1	8.3	9.509E-1	2.9
6.0E-1	5.062E-1	4.8	3.925E-1	8.2	8.987E-1	2.9
8.0E-1	4.578E-1	4.6	3.537E-1	8.1	8.115E-1	2.8
1.0E+0	4.188E-1	4.3	3.222E-1	8.1	7.410E-1	2.8
1.5E+0	3.472E-1	4.1	2.645E-1	7.9	6.116E-1	2.8
2.0E+0	2.974E-1	4.0	2.246E-1	7.8	5.220E-1	2.8
3.0E+0	2.312E-1	3.9	1.724E-1	7.3	4.036E-1	2.7
4.0E+0	1.881E-1	3.9	1.394E-1	6.8	3.276E-1	2.6
5.0E+0	1.576E-1	4.0	1.166E-1	6.3	2.742E-1	2.5
6.0E+0	1.346E-1	4.0	9.994E-2	5.8	2.346E-1	2.5
8.0E+0	1.027E-1	4.0	7.730E-2	5.2	1.800E-1	2.4
1.0E+1	8.169E-2	4.0	6.279E-2	4.9	1.445E-1	2.3
1.5E+1	5.215E-2	4.0	4.261E-2	4.4	9.476E-2	2.3
2.0E+1	3.729E-2	3.9	3.226E-2	4.1	6.955E-2	2.2
3.0E+1	2.317E-2	4.0	2.178E-2	3.5	4.496E-2	2.1
4.0E+1	1.670E-2	4.1	1.652E-2	3.1	3.323E-2	2.1
5.0E+1	1.304E-2	4.0	1.334E-2	2.8	2.637E-2	2.0
6.0E+1	1.064E-2	3.6	1.117E-2	2.7	2.182E-2	2.0
8.0E+1	7.655E-3	3.7	8.348E-3	2.5	1.600E-2	1.9
1.0E+2	5.852E-3	2.3	6.562E-3	1.8	1.241E-2	1.7
1.5E+2	3.507E-3	2.1	4.083E-3	1.9	7.591E-3	1.5
2.0E+2	2.432E-3	2.2	2.855E-3	2.0	5.286E-3	1.3
3.0E+2	1.487E-3	2.4	1.725E-3	2.0	3.212E-3	1.3
4.0E+2	1.077E-3	2.5	1.234E-3	1.9	2.312E-3	1.4
5.0E+2	8.508E-4	2.8	9.714E-4	1.7	1.822E-3	1.5
6.0E+2	7.059E-4	2.9	8.079E-4	1.6	1.514E-3	1.5
8.0E+2	5.289E-4	3.1	6.129E-4	1.4	1.142E-3	1.6
1.0E+3	4.232E-4	3.1	4.981E-4	1.4	9.213E-4	1.6
1.5E+3	2.784E-4	3.0	3.413E-4	1.4	6.197E-4	1.6
2.0E+3	2.013E-4	2.8	2.570E-4	1.5	4.583E-4	1.5
3.0E+3	1.202E-4	2.4	1.664E-4	1.8	2.865E-4	1.5
4.0E+3	8.061E-5	2.3	1.193E-4	2.0	1.999E-4	1.5
5.0E+3	5.882E-5	2.2	9.109E-5	2.1	1.499E-4	1.6
6.0E+3	4.570E-5	2.3	7.252E-5	2.2	1.182E-4	1.6
8.0E+3	3.133E-5	2.4	4.973E-5	2.3	8.106E-5	1.7

Table 4.1 (Continued)
Pulse Thermal Fission of ^{235}U

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	2.387E-5	2.5	3.638E-5	2.3	6.025E-5	1.7
1.5E+4	1.508E-5	2.5	1.952E-5	2.2	3.460E-5	1.6
2.0E+4	1.101E-5	2.4	1.223E-5	1.9	2.324E-5	1.5
3.0E+4	7.025E-6	2.3	6.566E-6	1.6	1.359E-5	1.4
4.0E+4	4.967E-6	2.3	4.443E-6	1.5	9.410E-6	1.4
5.0E+4	3.686E-6	2.2	3.336E-6	1.4	7.022E-6	1.3
6.0E+4	2.828E-6	2.3	2.648E-6	1.3	5.476E-6	1.3
8.0E+4	1.806E-6	2.5	1.836E-6	1.3	3.642E-6	1.4
1.0E+5	1.260E-6	2.6	1.376E-6	1.3	2.636E-6	1.4
1.5E+5	6.528E-7	2.9	8.051E-7	1.4	1.458E-6	1.5
2.0E+5	4.096E-7	3.0	5.518E-7	1.6	9.614E-7	1.6
3.0E+5	2.218E-7	3.4	3.387E-7	1.8	5.605E-7	1.7
4.0E+5	1.532E-7	3.4	2.504E-7	1.7	4.036E-7	1.7
5.0E+5	1.193E-7	3.4	2.018E-7	1.7	3.211E-7	1.6
6.0E+5	9.869E-8	3.2	1.698E-7	1.5	2.685E-7	1.5
8.0E+5	7.419E-8	2.7	1.280E-7	1.2	2.022E-7	1.3
1.0E+6	5.963E-8	2.5	1.012E-7	1.0	1.608E-7	1.1
1.5E+6	3.998E-8	2.2	6.390E-8	0.8	1.039E-7	1.0
2.0E+6	2.991E-8	2.1	4.530E-8	0.7	7.520E-8	0.9
3.0E+6	1.930E-8	1.7	2.671E-8	0.8	4.602E-8	0.9
4.0E+6	1.377E-8	1.4	1.758E-8	0.9	3.135E-8	0.8
5.0E+6	1.050E-8	1.0	1.256E-8	1.1	2.307E-8	0.8
6.0E+6	8.435E-9	0.8	9.634E-9	1.3	1.807E-8	0.8
8.0E+6	6.027E-9	0.7	6.591E-9	1.7	1.262E-8	0.9
1.0E+7	4.663E-9	0.7	5.015E-9	1.9	9.678E-9	1.0
1.5E+7	2.855E-9	0.8	2.827E-9	2.1	5.682E-9	1.1
2.0E+7	1.963E-9	0.9	1.604E-9	2.2	3.568E-9	1.1
3.0E+7	1.163E-9	1.1	5.181E-10	2.1	1.681E-9	1.0
4.0E+7	8.156E-10	1.2	1.867E-10	1.9	1.002E-9	1.1
5.0E+7	6.136E-10	1.2	8.607E-11	1.6	6.997E-10	1.1
6.0E+7	4.755E-10	1.2	5.364E-11	1.4	5.292E-10	1.1
8.0E+7	2.986E-10	1.2	3.605E-11	1.2	3.347E-10	1.1
1.0E+8	1.976E-10	1.2	3.053E-11	1.1	2.281E-10	1.0
1.5E+8	9.316E-11	1.4	2.494E-11	0.8	1.181E-10	1.2
2.0E+8	6.444E-11	1.8	2.279E-11	0.8	8.724E-11	1.3
3.0E+8	5.093E-11	1.9	2.068E-11	0.8	7.162E-11	1.4
4.0E+8	4.610E-11	1.8	1.915E-11	0.8	6.524E-11	1.3
5.0E+8	4.239E-11	1.7	1.778E-11	0.9	6.017E-11	1.2
6.0E+8	3.913E-11	1.5	1.652E-11	0.9	5.564E-11	1.1
8.0E+8	3.346E-11	1.2	1.427E-11	0.8	4.773E-11	0.9

Table 4.1 (Continued)
Pulse Thermal Fission of ^{235}U

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	2.867E-11	1.1	1.233E-11	0.8	4.100E-11	0.8
1.5E+9	1.953E-11	1.1	8.569E-12	0.8	2.809E-11	0.8
2.0E+9	1.332E-11	1.1	5.954E-12	0.8	1.927E-11	0.8
3.0E+9	6.211E-12	1.1	2.875E-12	0.8	9.086E-12	0.8
4.0E+9	2.901E-12	1.1	1.388E-12	0.8	4.289E-12	0.8
5.0E+9	1.357E-12	1.1	6.705E-13	0.8	2.027E-12	0.8
6.0E+9	6.358E-13	1.1	3.239E-13	0.8	9.597E-13	0.8
8.0E+9	1.412E-13	1.2	7.573E-14	0.8	2.169E-13	0.8
1.0E+10	3.255E-14	1.2	1.787E-14	0.8	5.042E-14	0.8
1.5E+10	1.908E-15	1.2	7.422E-16	0.9	2.650E-15	0.9
2.0E+10	8.795E-16	1.5	2.921E-16	1.9	1.172E-15	1.2
3.0E+10	7.228E-16	1.7	2.794E-16	2.0	1.002E-15	1.3
4.0E+10	7.098E-16	1.7	2.787E-16	2.0	9.885E-16	1.3
5.0E+10	7.079E-16	1.7	2.781E-16	2.0	9.861E-16	1.3
6.0E+10	7.071E-16	1.7	2.775E-16	2.0	9.846E-16	1.3
8.0E+10	7.055E-16	1.7	2.763E-16	2.0	9.818E-16	1.3
1.0E+11	7.039E-16	1.7	2.751E-16	2.0	9.790E-16	1.3
1.5E+11	7.000E-16	1.7	2.721E-16	2.0	9.721E-16	1.3
2.0E+11	6.962E-16	1.7	2.691E-16	2.0	9.653E-16	1.3
3.0E+11	6.886E-16	1.7	2.632E-16	2.0	9.518E-16	1.3
4.0E+11	6.811E-16	1.7	2.575E-16	2.0	9.386E-16	1.3
5.0E+11	6.736E-16	1.6	2.519E-16	2.0	9.256E-16	1.3
6.0E+11	6.663E-16	1.6	2.464E-16	2.0	9.128E-16	1.3
8.0E+11	6.520E-16	1.6	2.358E-16	2.0	8.878E-16	1.3
1.0E+12	6.381E-16	1.6	2.257E-16	2.0	8.638E-16	1.3
1.5E+12	6.048E-16	1.6	2.022E-16	2.0	8.070E-16	1.3
2.0E+12	5.737E-16	1.6	1.812E-16	2.0	7.549E-16	1.2
3.0E+12	5.174E-16	1.6	1.454E-16	2.0	6.628E-16	1.2
4.0E+12	4.678E-16	1.5	1.167E-16	2.0	5.846E-16	1.2
5.0E+12	4.242E-16	1.5	9.370E-17	2.0	5.179E-16	1.2
6.0E+12	3.856E-16	1.5	7.522E-17	2.0	4.608E-16	1.1
8.0E+12	3.210E-16	1.5	4.847E-17	2.0	3.695E-16	1.1
1.0E+13	2.698E-16	1.5	3.124E-17	2.0	3.011E-16	1.1

Table 4.2
Tabular Data for Recommended Decay Heat Power for Pulse Thermal Fission of ^{239}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	4.536E-1	5.4	3.635E-1	8.1	8.171E-1	4.3
1.0E-1	4.280E-1	4.9	3.410E-1	7.6	7.690E-1	3.8
1.5E-1	4.163E-1	4.7	3.309E-1	7.3	7.472E-1	3.7
2.0E-1	4.053E-1	4.5	3.214E-1	7.2	7.267E-1	3.6
3.0E-1	3.851E-1	4.3	3.041E-1	7.0	6.892E-1	3.5
4.0E-1	3.670E-1	4.1	2.886E-1	6.9	6.556E-1	3.4
5.0E-1	3.505E-1	4.0	2.748E-1	6.9	6.253E-1	3.4
6.0E-1	3.356E-1	3.9	2.623E-1	6.8	5.979E-1	3.3
8.0E-1	3.093E-1	3.7	2.406E-1	6.7	5.499E-1	3.3
1.0E+0	2.865E-1	3.6	2.224E-1	6.7	5.093E-1	3.2
1.5E+0	2.429E-1	3.5	1.871E-1	6.5	4.300E-1	3.1
2.0E+0	2.103E-1	3.4	1.614E-1	6.3	3.717E-1	3.1
3.0E+0	1.650E-1	3.3	1.262E-1	5.9	2.911E-1	3.0
4.0E+0	1.347E-1	3.3	1.032E-1	5.5	2.379E-1	2.8
5.0E+0	1.131E-1	3.3	8.699E-2	5.2	2.001E-1	2.7
6.0E+0	9.705E-2	3.3	7.506E-2	4.9	1.721E-1	2.6
8.0E+0	7.479E-2	3.3	5.875E-2	4.7	1.335E-1	2.6
1.0E+1	6.029E-2	3.4	4.820E-2	4.5	1.085E-1	2.6
1.5E+1	3.980E-2	3.5	3.327E-2	4.2	7.307E-2	2.6
2.0E+1	2.927E-2	3.6	2.545E-2	3.9	5.472E-2	2.5
3.0E+1	1.891E-2	3.8	1.741E-2	3.6	3.632E-2	2.5
4.0E+1	1.396E-2	3.9	1.331E-2	3.5	2.727E-2	2.6
5.0E+1	1.107E-2	3.9	1.079E-2	3.6	2.186E-2	2.6
6.0E+1	9.138E-3	3.9	9.063E-3	3.6	1.820E-2	2.5
8.0E+1	6.676E-3	3.6	6.785E-3	3.6	1.346E-2	2.5
1.0E+2	5.161E-3	2.8	5.334E-3	3.0	1.050E-2	2.4
1.5E+2	3.157E-3	2.6	3.328E-3	2.9	6.485E-3	2.1
2.0E+2	2.223E-3	2.6	2.349E-3	2.6	4.572E-3	1.9
3.0E+2	1.391E-3	2.9	1.460E-3	2.1	2.852E-3	1.7
4.0E+2	1.024E-3	3.1	1.074E-3	1.8	2.099E-3	1.7
5.0E+2	8.185E-4	3.3	8.643E-4	1.5	1.683E-3	1.8
6.0E+2	6.850E-4	3.4	7.312E-4	1.4	1.416E-3	1.8
8.0E+2	5.175E-4	3.5	5.673E-4	1.3	1.085E-3	1.8
1.0E+3	4.135E-4	3.6	4.663E-4	1.3	8.798E-4	1.8
1.5E+3	2.668E-4	3.4	3.214E-4	1.3	5.882E-4	1.7
2.0E+3	1.888E-4	3.2	2.410E-4	1.4	4.298E-4	1.6
3.0E+3	1.088E-4	2.7	1.534E-4	1.7	2.623E-4	1.5
4.0E+3	7.073E-5	2.6	1.077E-4	1.9	1.784E-4	1.6
5.0E+3	5.009E-5	2.5	8.051E-5	2.1	1.306E-4	1.6
6.0E+3	3.788E-5	2.5	6.289E-5	2.2	1.008E-4	1.7
8.0E+3	2.489E-5	2.8	4.191E-5	2.3	6.679E-5	1.8

Table 4.2 (Continued)
Pulse Thermal Fission of ^{239}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	1.845E-5	2.9	3.013E-5	2.3	4.858E-5	1.8
1.5E+4	1.130E-5	3.0	1.599E-5	2.1	2.729E-5	1.7
2.0E+4	8.202E-6	3.0	1.015E-5	1.8	1.835E-5	1.7
3.0E+4	5.29[illegible]E-6	2.8	5.710E-6	1.6	1.100E-5	1.6
4.0E+4	3.833E-6	2.7	4.022E-6	1.5	7.855E-6	1.5
5.0E+4	2.926E-6	2.6	3.108E-6	1.5	6.034E-6	1.5
6.0E+4	2.310E-6	2.7	2.517E-6	1.4	4.827E-6	1.5
8.0E+4	1.554E-6	2.8	1.792E-6	1.4	3.346E-6	1.5
1.0E+5	1.130E-6	2.9	1.369E-6	1.3	2.499E-6	1.5
1.5E+5	6.257E-7	2.9	8.366E-7	1.3	1.462E-6	1.4
2.0E+5	4.084E-7	2.8	5.939E-7	1.3	1.002E-6	1.4
3.0E+5	2.282E-7	2.9	3.760E-7	1.3	6.042E-7	1.3
4.0E+5	1.569E-7	2.9	2.777E-7	1.2	4.346E-7	1.3
5.0E+5	1.200E-7	2.9	2.211E-7	1.2	3.411E-7	1.3
6.0E+5	9.740E-8	2.8	1.833E-7	1.1	2.807E-7	1.2
8.0E+5	7.052E-8	2.6	1.344E-7	1.0	2.050E-7	1.1
1.0E+6	5.487E-8	2.5	1.039E-7	0.9	1.588E-7	1.0
1.5E+6	3.465E-8	2.4	6.327E-8	0.9	9.792E-8	1.0
2.0E+6	2.495E-8	2.4	4.407E-8	0.9	6.902E-8	1.0
3.0E+6	1.541E-8	2.1	2.576E-8	1.1	4.118E-8	1.0
4.0E+6	1.074E-8	1.7	1.694E-8	1.3	2.768E-8	1.1
5.0E+6	8.106E-9	1.4	1.206E-8	1.5	2.017E-8	1.1
6.0E+6	6.506E-9	1.3	9.177E-9	1.7	1.568E-8	1.2
8.0E+6	4.750E-9	1.2	6.130E-9	2.1	1.088E-8	1.3
1.0E+7	3.820E-9	1.2	4.553E-9	2.3	8.372E-9	1.4
1.5E+7	2.635E-9	1.4	2.476E-9	2.4	5.111E-9	1.4
2.0E+7	2.032E-9	1.6	1.404E-9	2.5	3.436E-9	1.4
3.0E+7	1.408E-9	1.8	4.976E-10	2.2	1.905E-9	1.5
4.0E+7	1.065E-9	1.9	2.223E-10	2.1	1.287E-9	1.6
5.0E+7	8.297E-10	2.0	1.316E-10	2.1	9.613E-10	1.8
6.0E+7	6.545E-10	2.1	9.592E-11	1.9	7.505E-10	1.8
8.0E+7	4.148E-10	2.1	6.634E-11	1.8	4.812E-10	1.8
1.0E+8	2.684E-10	2.1	5.147E-11	1.7	3.199E-10	1.8
1.5E+8	1.025E-10	1.9	3.379E-11	1.1	1.363E-10	1.5
2.0E+8	5.084E-11	1.8	2.723E-11	0.8	7.807E-11	1.2
3.0E+8	2.736E-11	2.1	2.277E-11	0.8	5.013E-11	1.2
4.0E+8	2.290E-11	2.2	2.074E-11	0.8	4.364E-11	1.2
5.0E+8	2.077E-11	2.2	1.916E-11	0.8	3.993E-11	1.2
6.0E+8	1.911E-11	2.2	1.777E-11	0.8	3.687E-11	1.2
8.0E+8	1.633E-11	2.1	1.533E-11	0.8	3.166E-11	1.2

Table 4.2 (Continued)
Pulse Thermal Fission of ^{239}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	1.401E-11	2.0	1.324E-11	0.8	2.725E-11	1.1
1.5E+9	9.587E-12	2.0	9.197E-12	0.8	1.878E-11	1.1
2.0E+9	6.575E-12	2.0	6.391E-12	0.8	1.297E-11	1.1
3.0E+9	3.102E-12	2.1	3.087E-12	0.8	6.188E-12	1.2
4.0E+9	1.468E-12	2.1	1.491E-12	0.8	2.959E-12	1.2
5.0E+9	6.969E-13	2.1	7.207E-13	0.8	1.418E-12	1.2
6.0E+9	3.328E-13	2.1	3.487E-13	0.8	6.815E-13	1.2
8.0E+9	7.840E-14	2.1	8.234E-14	0.8	1.607E-13	1.2
1.0E+10	2.043E-14	2.0	2.024E-14	0.8	4.067E-14	1.1
1.5E+10	2.426E-15	2.4	1.852E-15	1.5	4.277E-15	1.5
2.0E+10	1.371E-15	3.1	1.367E-15	2.0	2.738E-15	1.9
3.0E+10	1.100E-15	3.5	1.351E-15	2.0	2.451E-15	2.1
4.0E+10	1.076E-15	3.6	1.348E-15	2.0	2.424E-15	2.1
5.0E+10	1.072E-15	3.6	1.345E-15	2.0	2.417E-15	2.1
6.0E+10	1.071E-15	3.6	1.342E-15	2.0	2.413E-15	2.1
8.0E+10	1.067E-15	3.6	1.336E-15	2.0	2.404E-15	2.1
1.0E+11	1.064E-15	3.6	1.330E-15	2.0	2.395E-15	2.1
1.5E+11	1.056E-15	3.6	1.316E-15	2.0	2.372E-15	2.1
2.0E+11	1.048E-15	3.5	1.302E-15	2.0	2.350E-15	2.1
3.0E+11	1.033E-15	3.5	1.273E-15	2.0	2.306E-15	2.1
4.0E+11	1.018E-15	3.5	1.246E-15	2.0	2.263E-15	2.1
5.0E+11	1.003E-15	3.5	1.218E-15	2.0	2.221E-15	2.1
6.0E+11	9.884E-16	3.5	1.192E-15	2.0	2.180E-15	2.1
8.0E+11	9.599E-16	3.5	1.141E-15	2.0	2.101E-15	2.1
1.0E+12	9.324E-16	3.4	1.092E-15	2.0	2.024E-15	2.0
1.5E+12	8.678E-16	3.4	9.780E-16	2.0	1.846E-15	2.0
2.0E+12	8.087E-16	3.3	8.762E-16	2.0	1.685E-15	2.0
3.0E+12	7.047E-16	3.3	7.033E-16	2.0	1.408E-15	2.0
4.0E+12	6.169E-16	3.0	5.645E-16	2.0	1.181E-15	1.9
5.0E+12	5.426E-16	2.9	4.532E-16	2.0	9.957E-16	1.9
6.0E+12	4.793E-16	2.8	3.637E-16	2.0	8.431E-16	1.8
8.0E+12	3.792E-16	2.6	2.344E-16	2.0	6.136E-16	1.7
1.0E+13	3.052E-16	2.4	1.510E-16	2.0	4.562E-16	1.6

Table 4.3
Tabular Data for Recommended Decay Heat Power for Pulse Fast Fission of ^{238}U

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	1.518E+0	7.4	1.129E+0	10.1	2.647E+0	6.1
1.0E-1	1.366E+0	6.9	1.016E+0	9.6	2.382E+0	5.6
1.5E-1	1.302E+0	6.6	9.683E-1	9.1	2.271E+0	5.3
2.0E-1	1.245E+0	6.3	9.253E-1	8.8	2.170E+0	5.1
3.0E-1	1.146E+0	5.9	8.510E-1	8.4	1.997E+0	4.9
4.0E-1	1.063E+0	5.6	7.888E-1	8.2	1.852E+0	4.7
5.0E-1	9.926E-1	5.4	7.360E-1	8.1	1.729E+0	4.5
6.0E-1	9.318E-1	5.2	6.904E-1	8.0	1.622E+0	4.4
8.0E-1	8.319E-1	4.9	6.154E-1	7.9	1.447E+0	4.3
1.0E+0	7.526E-1	4.6	5.558E-1	8.0	1.308E+0	4.2
1.5E+0	6.089E-1	4.3	4.479E-1	8.1	1.057E+0	4.2
2.0E+0	5.103E-1	4.1	3.741E-1	8.1	8.844E-1	4.1
3.0E+0	3.810E-1	4.0	2.783E-1	8.1	6.593E-1	4.1
4.0E+0	2.992E-1	3.9	2.186E-1	7.9	5.178E-1	4.0
5.0E+0	2.429E-1	3.9	1.781E-1	7.5	4.209E-1	3.8
6.0E+0	2.020E-1	3.9	1.490E-1	7.0	3.510E-1	3.7
8.0E+0	1.473E-1	3.9	1.107E-1	6.2	2.579E-1	3.4
1.0E+1	1.132E-1	3.9	8.690E-2	5.7	2.001E-1	3.3
1.5E+1	6.817E-2	4.0	5.549E-2	5.2	1.237E-1	3.1
2.0E+1	4.712E-2	4.0	4.046E-2	5.0	8.758E-2	3.1
3.0E+1	2.822E-2	4.0	2.626E-2	4.5	5.449E-2	3.0
4.0E+1	1.997E-2	4.1	1.957E-2	4.2	3.954E-2	2.9
5.0E+1	1.543E-2	4.0	1.564E-2	3.9	3.107E-2	2.8
6.0E+1	1.253E-2	3.9	1.301E-2	3.7	2.553E-2	2.6
8.0E+1	8.945E-3	3.6	9.612E-3	3.4	1.856E-2	2.4
1.0E+2	6.801E-3	2.5	7.491E-3	2.4	1.429E-2	2.3
1.5E+2	4.027E-3	2.4	4.591E-3	2.5	8.619E-3	2.0
2.0E+2	2.756E-3	2.4	3.172E-3	2.4	5.928E-3	1.8
3.0E+2	1.638E-3	2.6	1.869E-3	2.2	3.507E-3	1.7
4.0E+2	1.159E-3	2.8	1.305E-3	2.1	2.464E-3	1.7
5.0E+2	9.006E-4	3.0	1.007E-3	1.9	1.908E-3	1.7
6.0E+2	7.403E-4	2.2	8.273E-4	1.8	1.568E-3	1.8
8.0E+2	5.498E-4	3.4	6.198E-4	1.6	1.170E-3	1.8
1.0E+3	4.375E-4	3.5	5.014E-4	1.6	9.389E-4	1.8
1.5E+3	2.840E-4	3.4	3.417E-4	1.6	6.257E-4	1.8
2.0E+3	2.024E-4	3.1	2.557E-4	1.7	4.581E-4	1.7
3.0E+3	1.176E-4	2.7	1.627E-4	1.9	2.804E-4	1.6
4.0E+3	7.702E-5	2.5	1.149E-4	2.1	1.919E-4	1.7
5.0E+3	5.501E-5	2.5	8.664E-5	2.3	1.417E-4	1.7
6.0E+3	4.195E-5	2.5	6.837E-5	2.5	1.103E-4	1.8
8.0E+3	2.798E-5	2.7	4.638E-5	2.6	7.437E-5	1.9

Table 4.3 (Continued)
Pulse Fast Fission of ^{238}U

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	2.095E-5	2.9	3.373E-5	2.6	5.468E-5	2.0
1.5E+4	1.296E-5	2.9	1.792E-5	2.5	3.089E-5	1.9
2.0E+4	9.417E-6	2.9	1.117E-5	2.3	2.059E-5	1.8
3.0E+4	6.044E-6	2.9	6.079E-6	2.0	1.212E-5	1.7
4.0E+4	4.331E-6	2.8	4.204E-6	1.9	8.535E-6	1.7
5.0E+4	3.259E-6	2.7	3.209E-6	1.9	6.469E-6	1.7
6.0E+4	2.533E-6	2.8	2.574E-6	1.9	5.108E-6	1.7
8.0E+4	1.655E-6	2.9	1.805E-6	1.9	3.460E-6	1.7
1.0E+5	1.175E-6	3.0	1.363E-6	2.0	2.538E-6	1.8
1.5E+5	6.291E-7	3.2	8.142E-7	2.0	1.443E-6	1.8
2.0E+5	4.041E-7	3.4	5.709E-7	2.1	9.750E-7	1.9
3.0E+5	2.244E-7	3.7	3.607E-7	2.1	5.851E-7	1.9
4.0E+5	1.555E-7	3.7	2.691E-7	2.0	4.245E-7	1.8
5.0E+5	1.201E-7	3.7	2.166E-7	1.9	3.366E-7	1.8
6.0E+5	9.829E-8	3.5	1.812E-7	1.7	2.794E-7	1.6
8.0E+5	7.218E-8	3.0	1.346E-7	1.4	2.067E-7	1.4
1.0E+6	5.682E-8	2.7	1.048E-7	1.2	1.617E-7	1.2
1.5E+6	3.674E-8	2.4	6.451E-8	1.1	1.012E-7	1.1
2.0E+6	2.691E-8	2.3	4.513E-8	1.2	7.204E-8	1.2
3.0E+6	1.697E-8	2.0	2.637E-8	1.6	4.335E-8	1.2
4.0E+6	1.196E-8	1.7	1.725E-8	1.9	2.921E-8	1.3
5.0E+6	9.064E-9	1.3	1.221E-8	2.3	2.127E-8	1.4
6.0E+6	7.269E-9	1.2	9.240E-9	2.4	1.651E-8	1.5
8.0E+6	5.244E-9	1.1	6.140E-9	2.6	1.138E-8	1.5
1.0E+7	4.135E-9	1.0	4.556E-9	2.7	8.692E-9	1.5
1.5E+7	2.698E-9	1.0	2.477E-9	2.4	5.175E-9	1.3
2.0E+7	1.980E-9	1.1	1.394E-9	2.3	3.374E-9	1.1
3.0E+7	1.290E-9	1.2	4.718E-10	2.0	1.762E-9	1.1
4.0E+7	9.480E-10	1.3	1.941E-10	1.9	1.142E-9	1.2
5.0E+7	7.286E-10	1.3	1.060E-10	1.8	8.346E-10	1.2
6.0E+7	5.701E-10	1.3	7.389E-11	1.6	6.440E-10	1.2
8.0E+7	3.584E-10	1.3	5.058E-11	1.3	4.090E-10	1.2
1.0E+8	2.322E-10	1.3	4.007E-11	1.2	2.723E-10	1.1
1.5E+8	9.372E-11	1.3	2.802E-11	0.9	1.217E-10	1.0
2.0E+8	5.238E-11	1.5	2.358E-11	0.9	7.596E-11	1.1
3.0E+8	3.356E-11	1.9	2.037E-11	0.9	5.393E-11	1.2
4.0E+8	2.920E-11	1.9	1.869E-11	0.9	4.789E-11	1.2
5.0E+8	2.664E-11	1.8	1.731E-11	0.9	4.395E-11	1.2
6.0E+8	2.454E-11	1.8	1.607E-11	0.9	4.061E-11	1.1
8.0E+8	2.097E-11	1.6	1.387E-11	0.9	3.484E-11	1.1

Table 4.3 (Continued)
Pulse Fast Fission of ^{238}U

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	1.798E-11	1.5	1.199E-11	0.9	2.997E-11	1.0
1.5E+9	1.228E-11	1.6	8.326E-12	0.9	2.060E-11	1.0
2.0E+9	8.400E-12	1.6	5.785E-12	0.9	1.418E-11	1.0
3.0E+9	3.940E-12	1.6	2.794E-12	0.9	6.734E-12	1.0
4.0E+9	1.853E-12	1.6	1.349E-12	0.9	3.202E-12	1.0
5.0E+9	8.740E-13	1.6	6.515E-13	0.9	1.526E-12	1.0
6.0E+9	4.140E-13	1.6	3.148E-13	0.9	7.288E-13	1.0
8.0E+9	9.536E-14	1.6	7.364E-14	0.9	1.690E-13	1.0
1.0E+10	2.379E-14	1.6	1.742E-14	0.9	4.120E-14	1.0
1.5E+10	2.177E-15	1.6	7.706E-16	1.0	2.948E-15	1.0
2.0E+10	1.029E-15	1.8	3.331E-16	2.0	1.362E-15	1.3
3.0E+10	7.482E-16	2.0	3.206E-16	2.1	1.069E-15	1.4
4.0E+10	7.240E-16	2.0	3.199E-16	2.1	1.044E-15	1.4
5.0E+10	7.211E-16	2.0	3.192E-16	2.1	1.040E-15	1.4
6.0E+10	7.201E-16	2.0	3.185E-16	2.1	1.039E-15	1.4
8.0E+10	7.185E-16	2.0	3.171E-16	2.1	1.036E-15	1.4
1.0E+11	7.169E-16	2.0	3.157E-16	2.1	1.033E-15	1.4
1.5E+11	7.128E-16	2.0	3.122E-16	2.1	1.025E-15	1.4
2.0E+11	7.088E-16	2.0	3.088E-16	2.1	1.018E-15	1.4
3.0E+11	7.009E-16	2.0	3.021E-16	2.1	1.003E-15	1.4
4.0E+11	6.931E-16	2.0	2.955E-16	2.1	9.886E-16	1.4
5.0E+11	6.854E-16	2.0	2.891E-16	2.1	9.745E-16	1.4
6.0E+11	6.777E-16	2.0	2.828E-16	2.1	9.606E-16	1.4
8.0E+11	6.628E-16	2.0	2.707E-16	2.1	9.335E-16	1.4
1.0E+12	6.483E-16	2.0	2.590E-16	2.1	9.074E-16	1.4
1.5E+12	6.138E-16	2.0	2.321E-16	2.1	8.459E-16	1.4
2.0E+12	5.815E-16	1.9	2.079E-16	2.1	7.894E-16	1.4
3.0E+12	5.231E-16	1.9	1.669E-16	2.1	6.900E-16	1.3
4.0E+12	4.718E-16	1.9	1.340E-16	2.1	6.058E-16	1.3
5.0E+12	4.267E-16	1.8	1.075E-16	2.1	5.343E-16	1.3
6.0E+12	3.870E-16	1.8	8.633E-17	2.1	4.733E-16	1.2
8.0E+12	3.206E-16	1.8	5.564E-17	2.1	3.762E-16	1.2
1.0E+13	2.682E-16	1.8	3.586E-17	2.1	3.040E-16	1.2

Table 4.4
Tabular Data for Recommended Decay Heat Power for Pulse Fast Fission of ^{240}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	5.842E-1	8.3	4.505E-1	12.3	1.035E+0	6.7
1.0E-1	5.468E-1	7.3	4.203E-1	11.3	9.671E-1	5.7
1.5E-1	5.302E-1	7.0	4.069E-1	11.0	9.371E-1	5.5
2.0E-1	5.148E-1	6.8	3.945E-1	10.8	9.092E-1	5.4
3.0E-1	4.867E-1	6.4	3.719E-1	10.6	8.587E-1	5.2
4.0E-1	4.619E-1	6.2	3.521E-1	10.4	8.140E-1	5.1
5.0E-1	4.398E-1	6.0	3.344E-1	10.3	7.741E-1	5.1
6.0E-1	4.198E-1	5.8	3.185E-1	10.2	7.383E-1	5.0
8.0E-1	3.853E-1	5.6	2.911E-1	10.1	6.763E-1	4.9
1.0E+0	3.563E-1	5.4	2.682E-1	10.0	6.244E-1	4.8
1.5E+0	3.001E-1	5.2	2.243E-1	9.7	5.244E-1	4.7
2.0E+0	2.591E-1	5.1	1.927E-1	9.4	4.518E-1	4.6
3.0E+0	2.025E-1	5.0	1.500E-1	8.8	3.525E-1	4.5
4.0E+0	1.650E-1	4.9	1.222E-1	8.2	2.872E-1	4.2
5.0E+0	1.382E-1	4.9	1.028E-1	7.7	2.410E-1	4.1
6.0E+0	1.182E-1	4.9	8.843E-2	7.4	2.067E-1	4.0
8.0E+0	9.048E-2	5.0	6.883E-2	7.0	1.593E-1	3.9
1.0E+1	7.238E-2	5.1	5.616E-2	6.8	1.285E-1	3.9
1.5E+1	4.692E-2	5.2	3.834E-2	6.3	8.526E-2	3.8
2.0E+1	3.403E-2	5.4	2.912E-2	5.9	6.315E-2	3.8
3.0E+1	2.160E-2	5.7	1.977E-2	5.4	4.137E-2	3.8
4.0E+1	1.578E-2	5.9	1.505E-2	5.3	3.082E-2	3.8
5.0E+1	1.241E-2	5.9	1.215E-2	5.3	2.456E-2	3.9
6.0E+1	1.019E-2	5.8	1.015E-2	5.4	2.034E-2	3.8
8.0E+1	7.380E-3	5.5	7.508E-3	5.3	1.489E-2	3.7
1.0E+2	5.664E-3	4.2	5.835E-3	4.4	1.150E-2	3.5
1.5E+2	3.416E-3	3.9	3.552E-3	4.3	6.968E-3	3.1
2.0E+2	2.380E-3	3.9	2.461E-3	3.9	4.840E-3	2.8
3.0E+2	1.470E-3	4.3	1.496E-3	3.1	2.966E-3	2.6
4.0E+2	1.075E-3	4.6	1.089E-3	2.7	2.164E-3	2.6
5.0E+2	8.551E-4	5.0	8.722E-4	2.3	1.727E-3	2.7
6.0E+2	7.134E-4	5.1	7.367E-4	2.1	1.450E-3	2.7
8.0E+2	5.367E-4	5.3	5.717E-4	1.9	1.108E-3	2.7
1.0E+3	4.277E-4	5.4	4.708E-4	1.9	8.986E-4	2.7
1.5E+3	2.752E-4	5.1	3.263E-4	2.0	6.015E-4	2.6
2.0E+3	1.947E-4	4.8	2.458E-4	2.2	4.405E-4	2.4
3.0E+3	1.124E-4	4.1	1.574E-4	2.6	2.698E-4	2.3
4.0E+3	7.313E-5	3.9	1.108E-4	2.9	1.840E-4	2.4
5.0E+3	5.172E-5	3.7	8.293E-5	3.2	1.347E-4	2.4
6.0E+3	3.898E-5	3.8	6.474E-5	3.3	1.037E-4	2.6
8.0E+3	2.538E-5	4.1	4.298E-5	3.4	6.836E-5	2.7

Table 4.4 (Continued)
Pulse Fast Fission of ^{240}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	1.862E-5	4.4	3.074E-5	3.4	4.936E-5	2.7
1.5E+4	1.119E-5	4.5	1.606E-5	3.2	2.724E-5	2.6
2.0E+4	8.037E-6	4.4	1.005E-5	2.7	1.808E-5	2.5
3.0E+4	5.155E-6	4.2	5.570E-6	2.3	1.073E-5	2.4
4.0E+4	3.740E-6	4.0	3.910E-6	2.3	7.650E-6	2.3
5.0E+4	2.864E-6	3.9	3.018E-6	2.2	5.882E-6	2.2
6.0E+4	2.269E-6	4.0	2.441E-6	2.1	4.710E-6	2.3
8.0E+4	1.533E-6	4.2	1.731E-6	2.0	3.265E-6	2.3
1.0E+5	1.118E-6	4.3	1.316E-6	2.0	2.434E-6	2.2
1.5E+5	6.202E-7	4.3	7.941E-7	1.9	1.414E-6	2.1
2.0E+5	4.042E-7	4.2	5.586E-7	1.9	9.628E-7	2.0
3.0E+5	2.246E-7	4.3	3.498E-7	1.9	5.744E-7	2.0
4.0E+5	1.535E-7	4.3	2.571E-7	1.9	4.106E-7	2.0
5.0E+5	1.169E-7	4.3	2.043E-7	1.8	3.211E-7	1.9
6.0E+5	9.452E-8	4.2	1.693E-7	1.7	2.638E-7	1.8
8.0E+5	6.820E-8	3.9	1.242E-7	1.5	1.924E-7	1.7
1.0E+6	5.303E-8	3.7	9.611E-8	1.4	1.491E-7	1.6
1.5E+6	3.353E-8	3.6	5.862E-8	1.3	9.215E-8	1.5
2.0E+6	2.419E-8	3.5	4.088E-8	1.4	6.507E-8	1.6
3.0E+6	1.500E-8	3.1	2.392E-8	1.7	3.892E-8	1.5
4.0E+6	1.051E-8	2.6	1.573E-8	2.0	2.623E-8	1.7
5.0E+6	7.989E-9	2.1	1.119E-8	2.3	1.918E-8	1.6
6.0E+6	6.465E-9	1.9	8.502E-9	2.6	1.497E-8	1.8
8.0E+6	4.802E-9	1.8	5.667E-9	3.2	1.047E-8	2.0
1.0E+7	3.923E-9	1.8	4.203E-9	3.4	8.126E-9	2.0
1.5E+7	2.795E-9	2.1	2.289E-9	3.7	5.084E-9	2.1
2.0E+7	2.204E-9	2.4	1.308E-9	3.7	3.511E-9	2.1
3.0E+7	1.562E-9	2.8	4.791E-10	3.3	2.041E-9	2.2
4.0E+7	1.190E-9	2.9	2.251E-10	3.2	1.415E-9	2.4
5.0E+7	9.300E-10	3.0	1.388E-10	3.1	1.069E-9	2.6
6.0E+7	7.338E-10	3.1	1.030E-10	2.9	8.368E-10	2.7
8.0E+7	4.640E-10	3.1	7.102E-11	2.6	5.350E-10	2.7
1.0E+8	2.988E-10	3.1	5.421E-11	2.5	3.530E-10	2.6
1.5E+8	1.110E-10	2.9	3.416E-11	1.6	1.452E-10	2.2
2.0E+8	5.243E-11	2.6	2.683E-11	1.2	7.926E-11	1.8
3.0E+8	2.603E-11	3.1	2.207E-11	1.2	4.811E-11	1.8
4.0E+8	2.133E-11	3.2	2.005E-11	1.2	4.138E-11	1.8
5.0E+8	1.926E-11	3.3	1.851E-11	1.2	3.777E-11	1.8
6.0E+8	1.769E-11	3.3	1.716E-11	1.3	3.485E-11	1.8
8.0E+8	1.510E-11	3.1	1.480E-11	1.3	2.990E-11	1.8

Table 4.4 (Continued)
Pulse Fast Fission of ^{240}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	1.295E-11	3.1	1.278E-11	1.3	2.573E-11	1.7
1.5E+9	8.864E-12	3.0	8.877E-12	1.2	1.774E-11	1.7
2.0E+9	6.082E-12	3.0	6.168E-12	1.2	1.225E-11	1.7
3.0E+9	2.873E-12	3.2	2.979E-12	1.2	5.852E-12	1.7
4.0E+9	1.361E-12	3.2	1.439E-12	1.2	2.800E-12	1.7
5.0E+9	6.478E-13	3.2	6.956E-13	1.2	1.343E-12	1.7
6.0E+9	3.102E-13	3.2	3.366E-13	1.2	6.469E-13	1.7
8.0E+9	7.384E-14	3.2	7.958E-14	1.2	1.534E-13	1.7
1.0E+10	1.965E-14	3.0	1.964E-14	1.2	3.929E-14	1.7
1.5E+10	2.500E-15	3.6	1.897E-15	2.3	4.397E-15	2.3
2.0E+10	1.403E-15	4.7	1.430E-15	3.0	2.833E-15	2.9
3.0E+10	1.108E-15	5.3	1.414E-15	3.0	2.522E-15	3.1
4.0E+10	1.082E-15	5.4	1.411E-15	3.0	2.493E-15	3.1
5.0E+10	1.078E-15	5.4	1.408E-15	3.0	2.486E-15	3.1
6.0E+10	1.077E-15	5.4	1.405E-15	3.0	2.481E-15	3.1
8.0E+10	1.073E-15	5.4	1.398E-15	3.0	2.472E-15	3.1
1.0E+11	1.070E-15	5.4	1.392E-15	3.0	2.462E-15	3.1
1.5E+11	1.062E-15	5.4	1.377E-15	3.0	2.439E-15	3.1
2.0E+11	1.054E-15	5.3	1.362E-15	3.0	2.416E-15	3.1
3.0E+11	1.038E-15	5.3	1.332E-15	3.0	2.370E-15	3.1
4.0E+11	1.023E-15	5.3	1.303E-15	3.0	2.326E-15	3.1
5.0E+11	1.008E-15	5.3	1.275E-15	3.0	2.283E-15	3.1
6.0E+11	9.926E-16	5.3	1.247E-15	3.0	2.240E-15	3.1
8.0E+11	9.636E-16	5.3	1.194E-15	3.0	2.157E-15	3.1
1.0E+12	9.356E-16	5.1	1.142E-15	3.0	2.078E-15	3.1
1.5E+12	8.698E-16	5.1	1.023E-15	3.0	1.893E-15	3.0
2.0E+12	8.096E-16	5.0	9.169E-16	3.0	1.726E-15	3.0
3.0E+12	7.039E-16	5.0	7.360E-16	3.0	1.440E-15	2.9
4.0E+12	6.148E-16	4.5	5.908E-16	3.0	1.206E-15	2.9
5.0E+12	5.396E-16	4.4	4.742E-16	3.0	1.014E-15	2.8
6.0E+12	4.758E-16	4.2	3.806E-16	3.0	8.564E-16	2.7
8.0E+12	3.750E-16	3.9	2.453E-16	3.0	6.203E-16	2.5
1.0E+13	3.008E-16	3.6	1.580E-16	3.0	4.589E-16	2.4

Table 4.5
Tabular Data for Recommended Decay Heat Power for Pulse Thermal Fission of ^{241}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	8.441E-1	8.3	6.351E-1	12.3	1.479E+0	6.7
1.0E-1	7.809E-1	7.3	5.863E-1	11.3	1.367E+0	5.7
1.5E-1	7.536E-1	7.0	5.652E-1	11.0	1.319E+0	5.5
2.0E-1	7.284E-1	6.8	5.457E-1	10.8	1.274E+0	5.4
3.0E-1	6.833E-1	6.4	5.110E-1	10.5	1.194E+0	5.2
4.0E-1	6.441E-1	6.2	4.809E-1	10.4	1.125E+0	5.1
5.0E-1	6.097E-1	6.0	4.546E-1	10.3	1.064E+0	5.1
6.0E-1	5.791E-1	5.8	4.312E-1	10.2	1.010E+0	5.0
8.0E-1	5.268E-1	5.6	3.914E-1	10.1	9.182E-1	4.9
1.0E+0	4.836E-1	5.4	3.587E-1	10.0	8.423E-1	4.8
1.5E+0	4.017E-1	5.2	2.970E-1	9.7	6.987E-1	4.7
2.0E+0	3.430E-1	5.1	2.533E-1	9.4	5.963E-1	4.6
3.0E+0	2.633E-1	5.0	1.949E-1	8.8	4.582E-1	4.5
4.0E+0	2.115E-1	4.9	1.573E-1	8.2	3.688E-1	4.2
5.0E+0	1.750E-1	4.9	1.312E-1	7.7	3.062E-1	4.1
6.0E+0	1.480E-1	4.9	1.120E-1	7.4	2.600E-1	4.0
8.0E+0	1.111E-1	5.0	8.591E-2	7.0	1.970E-1	3.9
1.0E+1	8.754E-2	5.1	6.916E-2	6.8	1.567E-1	3.9
1.5E+1	5.539E-2	5.2	4.602E-2	6.3	1.014E-1	3.8
2.0E+1	3.969E-2	5.4	3.441E-2	5.9	7.410E-2	3.8
3.0E+1	2.491E-2	5.7	2.299E-2	5.4	4.790E-2	3.8
4.0E+1	1.807E-2	5.9	1.736E-2	5.3	3.543E-2	3.8
5.0E+1	1.414E-2	5.9	1.393E-2	5.3	2.807E-2	3.9
6.0E+1	1.155E-2	5.8	1.158E-2	5.4	2.313E-2	3.8
8.0E+1	8.288E-3	5.5	8.486E-3	5.3	1.677E-2	3.7
1.0E+2	6.315E-3	4.2	6.538E-3	4.4	1.285E-2	3.5
1.5E+2	3.757E-3	3.9	3.912E-3	4.3	7.669E-3	3.1
2.0E+2	2.594E-3	3.9	2.677E-3	3.9	5.271E-3	2.8
3.0E+2	1.584E-3	4.3	1.598E-3	3.1	3.181E-3	2.6
4.0E+2	1.148E-3	4.6	1.147E-3	2.7	2.295E-3	2.6
5.0E+2	9.077E-4	5.0	9.081E-4	2.3	1.816E-3	2.7
6.0E+2	7.531E-4	5.1	7.605E-4	2.1	1.514E-3	2.7
8.0E+2	5.613E-4	5.3	5.837E-4	1.9	1.145E-3	2.7
1.0E+3	4.443E-4	5.4	4.780E-4	1.9	9.222E-4	2.7
1.5E+3	2.826E-4	5.1	3.296E-4	2.0	6.122E-4	2.6
2.0E+3	1.986E-4	4.8	2.479E-4	2.2	4.466E-4	2.4
3.0E+3	1.137E-4	4.1	1.585E-4	2.6	2.722E-4	2.3
4.0E+3	7.339E-5	3.9	1.115E-4	2.9	1.849E-4	2.4
5.0E+3	5.159E-5	3.7	8.336E-5	3.2	1.350E-4	2.4
6.0E+3	3.868E-5	3.8	6.509E-5	3.3	1.038E-4	2.6
8.0E+3	2.495E-5	4.1	4.322E-5	3.4	6.817E-5	2.7

Table 4.5 (Continued)
Pulse Thermal Fission of ^{241}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	1.817E-5	4.4	3.086E-5	3.4	4.903E-5	2.7
1.5E+4	1.074E-5	4.5	1.592E-5	3.2	2.667E-5	2.6
2.0E+4	7.626E-6	4.4	9.797E-6	2.8	1.742E-5	2.5
3.0E+4	4.839E-6	4.2	5.324E-6	2.3	1.016E-5	2.4
4.0E+4	3.509E-6	4.1	3.725E-6	2.3	7.234E-6	2.3
5.0E+4	2.696E-6	3.9	2.877E-6	2.2	5.573E-6	2.2
6.0E+4	2.145E-6	4.0	2.328E-6	2.1	4.472E-6	2.3
8.0E+4	1.461E-6	4.2	1.647E-6	2.0	3.108E-6	2.3
1.0E+5	1.072E-6	4.3	1.248E-6	2.0	2.320E-6	2.2
1.5E+5	6.013E-7	4.3	7.490E-7	1.9	1.350E-6	2.1
2.0E+5	3.951E-7	4.2	5.277E-7	1.9	9.228E-7	2.0
3.0E+5	2.224E-7	4.3	3.350E-7	1.9	5.574E-7	2.0
4.0E+5	1.535E-7	4.3	2.505E-7	1.9	4.041E-7	2.0
5.0E+5	1.179E-7	4.3	2.025E-7	1.8	3.204E-7	1.9
6.0E+5	9.606E-8	4.2	1.704E-7	1.7	2.664E-7	1.8
8.0E+5	7.021E-8	3.9	1.283E-7	1.5	1.985E-7	1.7
1.0E+6	5.517E-8	3.7	1.013E-7	1.4	1.565E-7	1.6
1.5E+6	3.550E-8	3.6	6.382E-8	1.3	9.932E-8	1.5
2.0E+6	2.578E-8	3.5	4.500E-8	1.4	7.078E-8	1.6
3.0E+6	1.595E-8	3.1	2.607E-8	1.7	4.202E-8	1.5
4.0E+6	1.108E-8	2.6	1.670E-8	2.0	2.779E-8	1.7
5.0E+6	8.370E-9	2.1	1.154E-8	2.3	1.991E-8	1.6
6.0E+6	6.752E-9	1.9	8.537E-9	2.6	1.529E-8	1.8
8.0E+6	5.035E-9	1.8	5.484E-9	3.2	1.052E-8	2.0
1.0E+7	4.161E-9	1.8	4.005E-9	3.4	8.166E-9	2.0
1.5E+7	3.062E-9	2.1	2.179E-9	3.7	5.242E-9	2.1
2.0E+7	2.473E-9	2.4	1.263E-9	3.7	3.736E-9	2.1
3.0E+7	1.798E-9	2.8	4.858E-10	3.3	2.284E-9	2.2
4.0E+7	1.383E-9	2.9	2.426E-10	3.2	1.626E-9	2.4
5.0E+7	1.085E-9	3.0	1.565E-10	3.1	1.241E-9	2.6
6.0E+7	8.574E-10	3.1	1.183E-10	2.9	9.757E-10	2.7
8.0E+7	5.423E-10	3.1	8.155E-11	2.6	6.238E-10	2.7
1.0E+8	3.482E-10	3.1	6.138E-11	2.5	4.096E-10	2.6
1.5E+8	1.261E-10	2.9	3.730E-11	1.6	1.634E-10	2.2
2.0E+8	5.629E-11	2.6	2.868E-11	1.2	8.497E-11	1.8
3.0E+8	2.507E-11	3.1	2.340E-11	1.2	4.848E-11	1.8
4.0E+8	1.995E-11	3.2	2.131E-11	1.2	4.127E-11	1.8
5.0E+8	1.793E-11	3.3	1.972E-11	1.2	3.766E-11	1.8
6.0E+8	1.647E-11	3.3	1.830E-11	1.3	3.477E-11	1.8
8.0E+8	1.407E-11	3.1	1.580E-11	1.3	2.987E-11	1.8

Table 4.5 (Continued)
Pulse Thermal Fission of ^{241}Pu

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	1.208E-11	3.1	1.365E-11	1.3	2.572E-11	1.7
1.5E+9	8.283E-12	3.0	9.480E-12	1.2	1.776E-11	1.7
2.0E+9	5.694E-12	3.0	6.587E-12	1.2	1.228E-11	1.7
3.0E+9	2.698E-12	3.2	3.181E-12	1.2	5.879E-12	1.7
4.0E+9	1.283E-12	3.2	1.536E-12	1.2	2.819E-12	1.7
5.0E+9	6.125E-13	3.2	7.418E-13	1.2	1.354E-12	1.7
6.0E+9	2.945E-13	3.2	3.584E-13	1.2	6.529E-13	1.7
8.0E+9	7.073E-14	3.2	8.386E-14	1.2	1.546E-13	1.7
1.0E+10	1.898E-14	3.0	1.985E-14	1.2	3.883E-14	1.7
1.5E+10	2.251E-15	3.6	8.966E-16	2.3	3.148E-15	2.3
2.0E+10	1.090E-15	4.7	3.985E-16	3.0	1.489E-15	2.9
3.0E+10	7.664E-16	5.3	3.842E-16	3.0	1.151E-15	3.1
4.0E+10	7.382E-16	5.4	3.833E-16	3.0	1.121E-15	3.1
5.0E+10	7.349E-16	5.4	3.825E-16	3.0	1.117E-15	3.1
6.0E+10	7.339E-16	5.4	3.816E-16	3.0	1.115E-15	3.1
8.0E+10	7.321E-16	5.4	3.799E-16	3.0	1.112E-15	3.1
1.0E+11	7.304E-16	5.4	3.783E-16	3.0	1.109E-15	3.1
1.5E+11	7.262E-16	5.4	3.741E-16	3.0	1.100E-15	3.1
2.0E+11	7.220E-16	5.3	3.701E-16	3.0	1.092E-15	3.1
3.0E+11	7.136E-16	5.3	3.620E-16	3.0	1.076E-15	3.1
4.0E+11	7.054E-16	5.3	3.541E-16	3.0	1.060E-15	3.1
5.0E+11	6.973E-16	5.3	3.464E-16	3.0	1.044E-15	3.1
6.0E+11	6.893E-16	5.3	3.389E-16	3.0	1.028E-15	3.1
8.0E+11	6.737E-16	5.3	3.243E-16	3.0	9.980E-16	3.1
1.0E+12	6.585E-16	5.1	3.104E-16	3.0	9.689E-16	3.1
1.5E+12	6.223E-16	5.1	2.781E-16	3.0	9.004E-16	3.0
2.0E+12	5.885E-16	5.0	2.491E-16	3.0	8.377E-16	3.0
3.0E+12	5.275E-16	5.0	2.000E-16	3.0	7.275E-16	2.9
4.0E+12	4.741E-16	4.5	1.605E-16	3.0	6.347E-16	2.9
5.0E+12	4.273E-16	4.4	1.289E-16	3.0	5.562E-16	2.8
6.0E+12	3.862E-16	4.2	1.034E-16	3.0	4.896E-16	2.7
8.0E+12	3.178E-16	3.9	6.666E-17	3.0	3.845E-16	2.5
1.0E+13	2.641E-16	3.6	4.296E-17	3.0	3.071E-16	2.4

Table 4.6
Tabular Data for Recommended Decay Heat Power for Thermal Fission of ^{235}U and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	6.509E+0	3.0	6.438E+0	3.1	1.295E+1	2.3
1.0E-1	6.436E+0	2.5	6.384E+0	2.6	1.282E+1	1.8
1.5E-1	6.401E+0	2.4	6.349E+0	2.4	1.275E+1	1.7
2.0E-1	6.368E+0	2.4	6.328E+0	2.4	1.270E+1	1.6
3.0E-1	6.305E+0	2.3	6.279E+0	2.3	1.258E+1	1.6
4.0E-1	6.247E+0	2.2	6.233E+0	2.2	1.248E+1	1.5
5.0E-1	6.192E+0	2.2	6.190E+0	2.1	1.238E+1	1.5
6.0E-1	6.140E+0	2.2	6.150E+0	2.1	1.229E+1	1.5
8.0E-1	6.043E+0	2.1	6.075E+0	2.0	1.212E+1	1.4
1.0E+0	5.956E+0	2.1	6.008E+0	1.9	1.196E+1	1.4
1.5E+0	5.766E+0	2.0	5.862E+0	1.8	1.163E+1	1.4
2.0E+0	5.605E+0	2.0	5.740E+0	1.7	1.135E+1	1.3
3.0E+0	5.343E+0	2.0	5.544E+0	1.6	1.089E+1	1.3
4.0E+0	5.135E+0	1.9	5.389E+0	1.5	1.052E+1	1.2
5.0E+0	4.963E+0	1.9	5.262E+0	1.5	1.022E+1	1.2
6.0E+0	4.817E+0	1.9	5.154E+0	1.4	9.971E+0	1.2
8.0E+0	4.583E+0	1.9	4.979E+0	1.4	9.561E+0	1.1
1.0E+1	4.400E+0	1.8	4.839E+0	1.3	9.239E+0	1.1
1.5E+1	4.074E+0	1.8	4.582E+0	1.2	8.657E+0	1.1
2.0E+1	3.855E+0	1.7	4.398E+0	1.2	8.252E+0	1.0
3.0E+1	3.563E+0	1.7	4.134E+0	1.1	7.697E+0	1.0
4.0E+1	3.367E+0	1.6	3.945E+0	1.1	7.312E+0	1.0
5.0E+1	3.220E+0	1.6	3.797E+0	1.1	7.017E+0	0.9
6.0E+1	3.102E+0	1.6	3.675E+0	1.1	6.777E+0	0.9
8.0E+1	2.922E+0	1.5	3.482E+0	1.1	6.404E+0	0.9
1.0E+2	2.788E+0	1.5	3.334E+0	1.1	6.123E+0	0.9
1.5E+2	2.563E+0	1.6	3.076E+0	1.1	5.639E+0	0.9
2.0E+2	2.417E+0	1.6	2.906E+0	1.1	5.323E+0	0.9
3.0E+2	2.229E+0	1.6	2.686E+0	1.1	4.915E+0	0.9
4.0E+2	2.103E+0	1.6	2.541E+0	1.1	4.644E+0	0.9
5.0E+2	2.007E+0	1.6	2.432E+0	1.1	4.439E+0	0.9
6.0E+2	1.930E+0	1.5	2.343E+0	1.1	4.273E+0	0.9
8.0E+2	1.808E+0	1.5	2.203E+0	1.1	4.012E+0	0.9
1.0E+3	1.714E+0	1.4	2.093E+0	1.1	3.807E+0	0.9
1.5E+3	1.543E+0	1.4	1.888E+0	1.2	3.431E+0	0.9
2.0E+3	1.425E+0	1.4	1.740E+0	1.2	3.165E+0	0.9
3.0E+3	1.269E+0	1.4	1.534E+0	1.2	2.804E+0	0.9
4.0E+3	1.171E+0	1.4	1.394E+0	1.1	2.565E+0	0.9
5.0E+3	1.102E+0	1.4	1.289E+0	1.1	2.392E+0	0.9
6.0E+3	1.051E+0	1.4	1.208E+0	1.1	2.259E+0	0.9
8.0E+3	9.753E-1	1.4	1.088E+0	1.0	2.064E+0	0.9

Table 4.6 (Continued)
Thermal Fission of ^{235}U and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	9.208E-1	1.4	1.003E+0	0.9	1.924E+0	0.8
1.5E+4	8.268E-1	1.4	8.699E-1	0.9	1.697E+0	0.8
2.0E+4	7.626E-1	1.4	7.928E-1	0.8	1.555E+0	0.8
3.0E+4	6.751E-1	1.4	7.042E-1	0.8	1.379E+0	0.8
4.0E+4	6.160E-1	1.4	6.505E-1	0.9	1.267E+0	0.8
5.0E+4	5.732E-1	1.4	6.121E-1	0.9	1.185E+0	0.8
6.0E+4	5.409E-1	1.4	5.825E-1	0.9	1.123E+0	0.8
8.0E+4	4.957E-1	1.4	5.385E-1	0.9	1.034E+0	0.8
1.0E+5	4.656E-1	1.4	5.067E-1	0.9	9.723E-1	0.8
1.5E+5	4.204E-1	1.3	4.544E-1	0.9	8.748E-1	0.8
2.0E+5	3.946E-1	1.3	4.212E-1	0.9	8.158E-1	0.8
3.0E+5	3.649E-1	1.2	3.785E-1	0.8	7.433E-1	0.7
4.0E+5	3.466E-1	1.1	3.495E-1	0.8	6.961E-1	0.7
5.0E+5	3.331E-1	1.1	3.271E-1	0.8	6.602E-1	0.7
6.0E+5	3.223E-1	1.0	3.087E-1	0.8	6.309E-1	0.7
8.0E+5	3.052E-1	0.9	2.792E-1	0.9	5.845E-1	0.6
1.0E+6	2.920E-1	0.8	2.565E-1	0.9	5.485E-1	0.6
1.5E+6	2.677E-1	0.7	2.164E-1	0.9	4.841E-1	0.6
2.0E+6	2.505E-1	0.7	1.896E-1	1.0	4.401E-1	0.6
3.0E+6	2.265E-1	0.6	1.548E-1	1.1	3.813E-1	0.6
4.0E+6	2.102E-1	0.6	1.332E-1	1.2	3.434E-1	0.6
5.0E+6	1.982E-1	0.6	1.183E-1	1.3	3.166E-1	0.6
6.0E+6	1.888E-1	0.7	1.073E-1	1.3	2.962E-1	0.6
8.0E+6	1.746E-1	0.7	9.152E-2	1.2	2.662E-1	0.6
1.0E+7	1.641E-1	0.7	8.005E-2	1.2	2.441E-1	0.6
1.5E+7	1.459E-1	0.7	6.104E-2	1.1	2.069E-1	0.6
2.0E+7	1.340E-1	0.7	5.024E-2	1.1	1.843E-1	0.6
3.0E+7	1.191E-1	0.8	4.066E-2	1.1	1.597E-1	0.6
4.0E+7	1.094E-1	0.8	3.746E-2	1.2	1.468E-1	0.6
5.0E+7	1.023E-1	0.8	3.620E-2	1.2	1.385E-1	0.7
6.0E+7	9.688E-2	0.9	3.553E-2	1.2	1.324E-1	0.7
8.0E+7	8.931E-2	0.9	3.467E-2	1.3	1.240E-1	0.7
1.0E+8	8.444E-2	0.9	3.401E-2	1.3	1.184E-1	0.8
1.5E+8	7.773E-2	1.0	3.265E-2	1.3	1.104E-1	0.8
2.0E+8	7.393E-2	1.0	3.147E-2	1.3	1.054E-1	0.8
3.0E+8	6.834E-2	..1	2.930E-2	1.4	9.764E-2	0.9
4.0E+8	6.351E-2	1.2	2.731E-2	1.4	9.082E-2	0.9
5.0E+8	5.909E-2	1.3	2.547E-2	1.5	8.455E-2	1.0
6.0E+8	5.501E-2	1.4	2.375E-2	1.6	7.877E-2	1.1
8.0E+8	4.777E-2	1.6	2.068E-2	1.7	6.845E-2	1.3

Table 4.6 (Continued)
Thermal Fission of ^{235}U and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	4.157E-2	1.8	1.802E-2	1.7	5.960E-2	1.4
1.5E+9	2.967E-2	1.9	1.286E-2	1.8	4.253E-2	1.5
2.0E+9	2.156E-2	2.0	9.265E-3	2.9	3.082E-2	1.7
3.0E+9	1.224E-2	2.3	5.036E-3	4.0	1.728E-2	2.0
4.0E+9	7.897E-3	2.5	2.993E-3	5.2	1.089E-2	2.3
5.0E+9	5.866E-3	2.8	2.007E-3	6.4	7.873E-3	2.6
6.0E+9	4.915E-3	3.0	1.531E-3	7.5	6.446E-3	2.8
8.0E+9	4.259E-3	3.5	1.190E-3	9.9	5.449E-3	3.4
1.0E+10	4.113E-3	4.0	1.110E-3	12.2	5.222E-3	4.0
1.5E+10	4.067E-3	4.0	1.085E-3	12.2	5.152E-3	4.0
2.0E+10	4.062E-3	4.0	1.083E-3	12.2	5.146E-3	4.0
3.0E+10	4.058E-3	4.0	1.081E-3	12.2	5.138E-3	4.0
4.0E+10	4.054E-3	4.0	1.078E-3	12.2	5.132E-3	4.0
5.0E+10	4.050E-3	4.0	1.076E-3	12.2	5.126E-3	4.0
6.0E+10	4.046E-3	4.0	1.074E-3	12.3	5.119E-3	4.0
8.0E+10	4.037E-3	4.0	1.069E-3	12.3	5.106E-3	4.0
1.0E+11	4.029E-3	4.0	1.064E-3	12.3	5.094E-3	4.0
1.5E+11	4.009E-3	4.0	1.053E-3	12.3	5.062E-3	4.0
2.0E+11	3.989E-3	4.0	1.041E-3	12.3	5.030E-3	4.0
3.0E+11	3.950E-3	3.9	1.018E-3	12.3	4.968E-3	3.9
4.0E+11	3.911E-3	3.9	9.963E-4	12.3	4.907E-3	3.9
5.0E+11	3.872E-3	3.9	9.746E-4	12.3	4.847E-3	3.9
6.0E+11	3.834E-3	3.8	9.535E-4	12.3	4.787E-3	3.8
8.0E+11	3.759E-3	3.8	9.125E-4	12.3	4.672E-3	3.8
1.0E+12	3.686E-3	3.8	8.732E-4	12.3	4.560E-3	3.8
1.5E+12	3.512E-3	3.7	7.824E-4	12.9	4.294E-3	3.7
2.0E+12	3.348E-3	3.6	7.010E-4	13.5	4.049E-3	3.7
3.0E+12	3.049E-3	3.5	5.627E-4	14.7	3.611E-3	3.5
4.0E+12	2.783E-3	3.3	4.517E-4	15.9	3.235E-3	3.4
5.0E+12	2.547E-3	3.1	3.626E-4	17.1	2.910E-3	3.3
6.0E+12	2.337E-3	3.0	2.911E-4	18.9	2.628E-3	3.1
8.0E+12	1.981E-3	2.6	1.877E-4	20.8	2.169E-3	2.9
1.0E+13	1.695E-3	2.3	1.210E-4	23.2	1.816E-3	2.6

Table 4.7
Tabular Data for Recommended Decay Heat Power for Thermal Fission of ^{239}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	5.265E+0	2.6	5.350E+0	2.5	1.062E+1	1.9
1.0E-1	5.221E+0	2.1	5.314E+0	2.0	1.054E+1	1.4
1.5E-1	5.200E+0	2.0	5.298E+0	1.9	1.050E+1	1.3
2.0E-1	5.180E+0	2.0	5.281E+0	1.9	1.046E+1	1.3
3.0E-1	5.140E+0	2.0	5.250E+0	1.8	1.039E+1	1.3
4.0E-1	5.103E+0	1.9	5.220E+0	1.8	1.032E+1	1.3
5.0E-1	5.067E+0	1.9	5.192E+0	1.8	1.026E+1	1.3
6.0E-1	5.032E+0	1.9	5.165E+0	1.8	1.020E+1	1.3
8.0E-1	4.968E+0	1.9	5.115E+0	1.7	1.008E+1	1.2
1.0E+0	4.908E+0	1.9	5.069E+0	1.7	9.977E+0	1.2
1.5E+0	4.777E+0	1.9	4.967E+0	1.6	9.744E+0	1.2
2.0E+0	4.664E+0	1.9	4.880E+0	1.6	9.544E+0	1.2
3.0E+0	4.478E+0	1.9	4.738E+0	1.5	9.216E+0	1.2
4.0E+0	4.329E+0	1.9	4.624E+0	1.5	8.953E+0	1.2
5.0E+0	4.205E+0	1.9	4.529E+0	1.5	8.735E+0	1.2
6.0E+0	4.101E+0	1.9	4.449E+0	1.5	8.549E+0	1.1
8.0E+0	3.931E+0	1.9	4.316E+0	1.4	8.247E+0	1.1
1.0E+1	3.796E+0	1.9	4.210E+0	1.4	8.006E+0	1.1
1.5E+1	3.553E+0	1.9	4.011E+0	1.4	7.563E+0	1.1
2.0E+1	3.383E+0	1.8	3.866E+0	1.3	7.248E+0	1.1
3.0E+1	3.149E+0	1.8	3.657E+0	1.3	6.806E+0	1.1
4.0E+1	2.987E+0	1.8	3.505E+0	1.2	6.492E+0	1.1
5.0E+1	2.863E+0	1.8	3.385E+0	1.2	6.248E+0	1.0
6.0E+1	2.763E+0	1.8	3.287E+0	1.2	6.049E+0	1.0
8.0E+1	2.607E+0	1.8	3.130E+0	1.1	5.737E+0	1.0
1.0E+2	2.489E+0	1.8	3.010E+0	1.1	5.499E+0	1.0
1.5E+2	2.288E+0	1.8	2.800E+0	1.0	5.088E+0	1.0
2.0E+2	2.157E+0	1.9	2.661E+0	1.0	4.817E+0	1.0
3.0E+2	1.982E+0	1.8	2.477E+0	1.0	4.460E+0	1.0
4.0E+2	1.864E+0	1.8	2.353E+0	1.0	4.216E+0	1.0
5.0E+2	1.772E+0	1.8	2.257E+0	1.0	4.029E+0	1.0
6.0E+2	1.698E+0	1.8	2.177E+0	1.0	3.875E+0	1.0
8.0E+2	1.579E+0	1.7	2.049E+0	1.1	3.628E+0	0.9
1.0E+3	1.487E+0	1.6	1.946E+0	1.1	3.433E+0	0.9
1.5E+3	1.321E+0	1.5	1.754E+0	1.1	3.074E+0	0.9
2.0E+3	1.209E+0	1.5	1.615E+0	1.1	2.823E+0	0.9
3.0E+3	1.065E+0	1.5	1.423E+0	1.1	2.488E+0	0.9
4.0E+3	9.776E-1	1.5	1.294E+0	1.1	2.272E+0	0.9
5.0E+3	9.182E-1	1.5	1.201E+0	1.0	2.119E+0	0.9
6.0E+3	8.747E-1	1.5	1.130E+0	1.0	2.005E+0	0.9
8.0E+3	8.136E-1	1.5	1.028E+0	0.9	1.841E+0	0.8

Table 4.7 (Continued)
Thermal Fission of ^{239}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	7.710E-1	1.5	9.565E-1	0.9	1.727E+0	0.8
1.5E+4	6.996E-1	1.5	8.469E-1	0.8	1.546E+0	0.8
2.0E+4	6.517E-1	1.4	7.835E-1	0.8	1.435E+0	0.8
3.0E+4	5.862E-1	1.4	7.084E-1	0.8	1.295E+0	0.8
4.0E+4	5.413E-1	1.4	6.608E-1	0.8	1.202E+0	0.8
5.0E+4	5.078E-1	1.4	6.255E-1	0.8	1.133E+0	0.8
6.0E+4	4.818E-1	1.4	5.976E-1	0.8	1.079E+0	0.8
8.0E+4	4.440E-1	1.4	5.552E-1	0.8	9.992E-1	0.8
1.0E+5	4.175E-1	1.3	5.239E-1	0.8	9.414E-1	0.8
1.5E+5	3.756E-1	1.3	4.708E-1	0.9	8.464E-1	0.8
2.0E+5	3.504E-1	1.3	4.357E-1	0.9	7.861E-1	0.8
3.0E+5	3.202E-1	1.3	3.889E-1	0.9	7.091E-1	0.8
4.0E+5	3.014E-1	1.2	3.567E-1	0.9	6.581E-1	0.8
5.0E+5	2.877E-1	1.2	3.320E-1	1.0	6.197E-1	0.8
6.0E+5	2.769E-1	1.2	3.119E-1	1.0	5.888E-1	0.8
8.0E+5	2.604E-1	1.2	2.806E-1	1.1	5.409E-1	0.8
1.0E+6	2.480E-1	1.2	2.569E-1	1.0	5.049E-1	0.8
1.5E+6	2.263E-1	1.2	2.166E-1	1.1	4.429E-1	0.8
2.0E+6	2.116E-1	1.2	1.902E-1	1.2	4.019E-1	0.8
3.0E+6	1.921E-1	1.2	1.566E-1	1.3	3.487E-1	0.9
4.0E+6	1.793E-1	1.2	1.357E-1	1.4	3.150E-1	0.9
5.0E+6	1.700E-1	1.2	1.214E-1	1.4	2.914E-1	0.9
6.0E+6	1.627E-1	1.3	1.109E-1	1.4	2.737E-1	0.9
8.0E+6	1.517E-1	1.3	9.602E-2	1.4	2.477E-1	1.0
1.0E+7	1.432E-1	1.3	8.548E-2	1.4	2.287E-1	1.0
1.5E+7	1.275E-1	1.3	6.855E-2	1.3	1.960E-1	1.0
2.0E+7	1.159E-1	1.3	5.912E-2	1.4	1.750E-1	1.0
3.0E+7	9.912E-2	1.3	5.050E-2	1.5	1.496E-1	1.0
4.0E+7	8.688E-2	1.3	4.716E-2	1.5	1.340E-1	1.0
5.0E+7	7.748E-2	1.3	4.547E-2	1.6	1.229E-1	1.0
6.0E+7	7.009E-2	1.3	4.436E-2	1.6	1.144E-1	1.0
8.0E+7	5.960E-2	1.3	4.278E-2	1.7	1.024E-1	1.0
1.0E+8	5.289E-2	1.3	4.161E-2	1.7	9.450E-2	1.0
1.5E+8	4.441E-2	1.4	3.956E-2	1.7	8.397E-2	1.1
2.0E+8	4.082E-2	1.5	3.805E-2	1.8	7.887E-2	1.2
3.0E+8	3.728E-2	1.5	3.560E-2	1.9	7.288E-2	1.2
4.0E+8	3.481E-2	1.6	3.343E-2	2.0	6.823E-2	1.3
5.0E+8	3.263E-2	1.6	3.144E-2	2.1	6.407E-2	1.3
6.0E+8	3.064E-2	1.6	2.959E-2	2.2	6.023E-2	1.4
8.0E+8	2.710E-2	1.7	2.629E-2	2.4	5.339E-2	1.5

Table 4.7 (Continued)
Thermal Fission of ^{239}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	2.408E-2	1.8	2.344E-2	2.6	4.751E-2	1.6
1.5E+9	1.825E-2	2.0	1.789E-2	2.9	3.614E-2	1.7
2.0E+9	1.426E-2	2.2	1.403E-2	3.1	2.829E-2	1.8
3.0E+9	9.635E-3	2.4	9.495E-3	3.6	1.913E-2	2.1
4.0E+9	7.452E-3	2.6	7.303E-3	4.1	1.475E-2	2.3
5.0E+9	6.418E-3	2.8	6.243E-3	4.6	1.266E-2	2.6
6.0E+9	5.926E-3	2.9	5.731E-3	5.1	1.166E-2	2.8
8.0E+9	5.576E-3	3.0	5.363E-3	6.1	1.094E-2	3.3
1.0E+10	5.492E-3	3.1	5.275E-3	7.1	1.077E-2	3.8
1.5E+10	5.458E-3	3.1	5.244E-3	7.1	1.070E-2	3.8
2.0E+10	5.451E-3	3.1	5.238E-3	7.1	1.069E-2	3.8
3.0E+10	5.442E-3	3.1	5.226E-3	7.1	1.067E-2	3.8
4.0E+10	5.435E-3	3.1	5.215E-3	7.1	1.065E-2	3.8
5.0E+10	5.428E-3	3.1	5.204E-3	7.2	1.063E-2	3.8
6.0E+10	5.421E-3	3.1	5.192E-3	7.2	1.061E-2	3.8
8.0E+10	5.406E-3	3.1	5.169E-3	7.2	1.058E-2	3.8
1.0E+11	5.392E-3	3.1	5.147E-3	7.2	1.054E-2	3.8
1.5E+11	5.356E-3	3.1	5.090E-3	7.3	1.045E-2	3.8
2.0E+11	5.321E-3	3.1	5.035E-3	7.4	1.036E-2	3.9
3.0E+11	5.251E-3	3.1	4.925E-3	7.5	1.018E-2	3.9
4.0E+11	5.183E-3	3.1	4.818E-3	7.7	1.000E-2	4.0
5.0E+11	5.115E-3	3.1	4.713E-3	7.8	9.829E-3	4.0
6.0E+11	5.049E-3	3.1	4.611E-3	8.0	9.660E-3	4.1
8.0E+11	4.920E-3	3.1	4.413E-3	8.3	9.332E-3	4.2
1.0E+12	4.794E-3	3.1	4.223E-3	8.6	9.017E-3	4.3
1.5E+12	4.499E-3	3.1	3.783E-3	9.7	8.282E-3	4.5
2.0E+12	4.227E-3	3.1	3.390E-3	10.8	7.616E-3	4.6
3.0E+12	3.743E-3	3.0	2.721E-3	13.1	6.464E-3	4.9
4.0E+12	3.330E-3	3.0	2.184E-3	15.3	5.514E-3	5.2
5.0E+12	2.976E-3	3.0	1.753E-3	17.5	4.729E-3	5.5
6.0E+12	2.671E-3	2.9	1.407E-3	19.8	4.078E-3	5.9
8.0E+12	2.178E-3	2.9	9.070E-4	24.2	3.085E-3	6.5
1.0E+13	1.805E-3	2.9	5.845E-4	28.7	2.390E-3	7.1

Table 4.8
Tabular Data for Recommended Decay Heat Power for Fast Fission of ^{238}U and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	8.307E+0	2.7	7.819E+0	3.5	1.613E+1	2.2
1.0E-1	8.162E+0	2.2	7.712E+0	3.0	1.587E+1	1.7
1.5E-1	8.096E+0	2.1	7.663E+0	2.9	1.576E+1	1.7
2.0E-1	8.032E+0	2.1	7.615E+0	2.8	1.565E+1	1.6
3.0E-1	7.912E+0	2.0	7.527E+0	2.7	1.544E+1	1.6
4.0E-1	7.802E+0	2.0	7.445E+0	2.6	1.525E+1	1.6
5.0E-1	7.699E+0	2.0	7.369E+0	2.6	1.507E+1	1.5
6.0E-1	7.603E+0	2.0	7.297E+0	2.6	1.490E+1	1.5
8.0E-1	7.427E+0	1.9	7.167E+0	2.5	1.459E+1	1.5
1.0E+0	7.269E+0	1.9	7.050E+0	2.4	1.432E+1	1.5
1.5E+0	6.931E+0	1.9	6.801E+0	2.3	1.373E+1	1.4
2.0E+0	6.653E+0	1.9	6.597E+0	2.2	1.325E+1	1.4
3.0E+0	6.213E+0	1.9	6.275E+0	2.1	1.249E+1	1.4
4.0E+0	5.875E+0	1.9	6.028E+0	2.0	1.190E+1	1.3
5.0E+0	5.606E+0	1.9	5.831E+0	1.9	1.144E+1	1.3
6.0E+0	5.385E+0	1.9	5.668E+0	1.9	1.105E+1	1.3
8.0E+0	5.040E+0	1.9	5.412E+0	1.8	1.045E+1	1.2
1.0E+1	4.782E+0	1.9	5.216E+0	1.7	9.998E+0	1.2
1.5E+1	4.345E+0	1.8	4.871E+0	1.6	9.216E+0	1.2
2.0E+1	4.062E+0	1.8	4.635E+0	1.6	8.697E+0	1.2
3.0E+1	3.701E+0	1.8	4.312E+0	1.5	8.012E+0	1.1
4.0E+1	3.464E+0	1.8	4.086E+0	1.4	7.550E+0	1.1
5.0E+1	3.289E+0	1.7	3.912E+0	1.4	7.201E+0	1.1
6.0E+1	3.150E+0	1.7	3.769E+0	1.3	6.919E+0	1.1
8.0E+1	2.939E+0	1.7	3.546E+0	1.3	6.485E+0	1.0
1.0E+2	2.783E+0	1.8	3.376E+0	1.2	6.160E+0	1.0
1.5E+2	2.522E+0	1.8	3.084E+0	1.2	5.607E+0	1.1
2.0E+2	2.356E+0	1.8	2.894E+0	1.2	5.250E+0	1.1
3.0E+2	2.146E+0	1.8	2.652E+0	1.2	4.798E+0	1.1
4.0E+2	2.009E+0	1.8	2.496E+0	1.2	4.505E+0	1.1
5.0E+2	1.907E+0	1.8	2.382E+0	1.2	4.289E+0	1.1
6.0E+2	1.825E+0	1.8	2.291E+0	1.3	4.117E+0	1.1
8.0E+2	1.698E+0	1.7	2.149E+0	1.3	3.847E+0	1.0
1.0E+3	1.600E+0	1.7	2.037E+0	1.3	3.638E+0	1.0
1.5E+3	1.425E+0	1.6	1.832E+0	1.3	3.257E+0	1.0
2.0E+3	1.305E+0	1.6	1.684E+0	1.3	2.989E+0	1.0
3.0E+3	1.151E+0	1.6	1.481E+0	1.4	2.631E+0	1.0
4.0E+3	1.056E+0	1.6	1.344E+0	1.3	2.400E+0	1.0
5.0E+3	9.906E-1	1.7	1.244E+0	1.3	2.235E+0	1.0
6.0E+3	9.427E-1	1.7	1.168E+0	1.3	2.110E+0	1.0
8.0E+3	8.745E-1	1.7	1.055E+0	1.2	1.930E+0	1.0

Table 4.8 (Continued)
Fast Fission of ^{238}U and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	8.263E-1	1.7	9.760E-1	1.2	1.802E+0	1.0
1.5E+4	7.447E-1	1.7	8.529E-1	1.2	1.598E+0	1.0
2.0E+4	6.897E-1	1.7	7.824E-1	1.2	1.472E+0	1.0
3.0E+4	6.147E-1	1.7	7.011E-1	1.2	1.316E+0	1.0
4.0E+4	5.636E-1	1.7	6.509E-1	1.2	1.215E+0	1.0
5.0E+4	5.260E-1	1.7	6.142E-1	1.2	1.140E+0	1.0
6.0E+4	4.973E-1	1.6	5.855E-1	1.2	1.083E+0	1.0
8.0E+4	4.564E-1	1.6	5.425E-1	1.2	9.989E-1	1.0
1.0E+5	4.285E-1	1.6	5.112E-1	1.2	9.397E-1	1.0
1.5E+5	3.857E-1	1.6	4.589E-1	1.2	8.446E-1	1.0
2.0E+5	3.606E-1	1.5	4.249E-1	1.2	7.855E-1	1.0
3.0E+5	3.308E-1	1.4	3.801E-1	1.2	7.109E-1	0.9
4.0E+5	3.123E-1	1.3	3.491E-1	1.2	6.614E-1	0.9
5.0E+5	2.987E-1	1.3	3.250E-1	1.2	6.237E-1	0.9
6.0E+5	2.878E-1	1.2	3.052E-1	1.3	5.930E-1	0.9
8.0E+5	2.710E-1	1.1	2.740E-1	1.3	5.451E-1	0.9
1.0E+6	2.582E-1	1.0	2.503E-1	1.3	5.085E-1	0.8
1.5E+6	2.356E-1	0.9	2.093E-1	1.4	4.449E-1	0.8
2.0E+6	2.199E-1	0.9	1.824E-1	1.5	4.023E-1	0.8
3.0E+6	1.986E-1	0.8	1.480E-1	1.6	3.466E-1	0.8
4.0E+6	1.844E-1	0.9	1.266E-1	1.6	3.110E-1	0.8
5.0E+6	1.740E-1	0.9	1.121E-1	1.6	2.861E-1	0.8
6.0E+6	1.659E-1	0.9	1.015E-1	1.5	2.674E-1	0.8
8.0E+6	1.536E-1	0.9	8.657E-2	1.4	2.402E-1	0.8
1.0E+7	1.443E-1	0.9	7.601E-2	1.3	2.203E-1	0.8
1.5E+7	1.277E-1	1.0	5.907E-2	1.2	1.868E-1	0.8
2.0E+7	1.162E-1	1.0	4.967E-2	1.1	1.659E-1	0.8
3.0E+7	1.003E-1	1.0	4.124E-2	1.2	1.416E-1	0.8
4.0E+7	8.930E-2	1.1	3.818E-2	1.3	1.275E-1	0.8
5.0E+7	8.098E-2	1.1	3.676E-2	1.3	1.177E-1	0.9
6.0E+7	7.453E-2	1.1	3.588E-2	1.4	1.104E-1	0.9
8.0E+7	6.543E-2	1.2	3.468E-2	1.4	1.001E-1	0.9
1.0E+8	5.963E-2	1.3	3.378E-2	1.4	9.341E-2	1.0
1.5E+8	5.218E-2	1.4	3.213E-2	1.5	8.431E-2	1.1
2.0E+8	4.872E-2	1.5	3.086E-2	1.5	7.958E-2	1.1
3.0E+8	4.471E-2	1.6	2.869E-2	1.6	7.340E-2	1.1
4.0E+8	4.160E-2	1.6	2.674E-2	1.6	6.834E-2	1.2
5.0E+8	3.881E-2	1.7	2.494E-2	1.7	6.375E-2	1.2
6.0E+8	3.626E-2	1.8	2.327E-2	1.8	5.953E-2	1.3
8.0E+8	3.172E-2	2.0	2.029E-2	2.0	5.200E-2	1.5

Table 4.8 (Continued)
Fast Fission of ^{238}U and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	2.783E-2	2.2	1.770E-2	2.2	4.554E-2	1.6
1.5E+9	2.036E-2	2.3	1.268E-2	2.8	3.304E-2	1.8
2.0E+9	1.525E-2	2.5	9.194E-3	3.4	2.444E-2	2.0
3.0E+9	9.360E-3	2.8	5.084E-3	4.6	1.444E-2	2.3
4.0E+9	6.594E-3	3.1	3.100E-3	5.8	9.694E-3	2.7
5.0E+9	5.292E-3	3.4	2.142E-3	7.0	7.434E-3	3.0
6.0E+9	4.677E-3	3.6	1.679E-3	8.2	6.356E-3	3.4
8.0E+9	4.245E-3	4.2	1.347E-3	10.6	5.592E-3	4.1
1.0E+10	4.144E-3	4.8	1.269E-3	13.0	5.413E-3	4.8
1.5E+10	4.106E-3	4.8	1.245E-3	13.0	5.351E-3	4.8
2.0E+10	4.101E-3	4.8	1.243E-3	13.0	5.344E-3	4.8
3.0E+10	4.095E-3	4.8	1.240E-3	13.0	5.336E-3	4.8
4.0E+10	4.091E-3	4.8	1.238E-3	13.0	5.329E-3	4.8
5.0E+10	4.087E-3	4.8	1.235E-3	13.1	5.322E-3	4.7
6.0E+10	4.083E-3	4.8	1.232E-3	13.1	5.315E-3	4.7
8.0E+10	4.074E-3	4.8	1.227E-3	13.1	5.301E-3	4.7
1.0E+11	4.066E-3	4.8	1.221E-3	13.1	5.287E-3	4.7
1.5E+11	4.045E-3	4.8	1.208E-3	13.1	5.253E-3	4.7
2.0E+11	4.024E-3	4.8	1.195E-3	13.1	5.219E-3	4.7
3.0E+11	3.984E-3	4.7	1.169E-3	13.1	5.152E-3	4.6
4.0E+11	3.943E-3	4.7	1.143E-3	13.1	5.087E-3	4.6
5.0E+11	3.903E-3	4.7	1.119E-3	13.0	5.022E-3	4.6
6.0E+11	3.864E-3	4.6	1.094E-3	13.0	4.958E-3	4.5
8.0E+11	3.787E-3	4.6	1.047E-3	13.0	4.834E-3	4.5
1.0E+12	3.711E-3	4.6	1.002E-3	13.0	4.714E-3	4.5
1.5E+12	3.531E-3	4.5	8.979E-4	13.6	4.429E-3	4.4
2.0E+12	3.362E-3	4.4	8.045E-4	14.3	4.167E-3	4.3
3.0E+12	3.054E-3	4.2	6.458E-4	15.6	3.700E-3	4.2
4.0E+12	2.781E-3	4.0	5.185E-4	16.8	3.300E-3	4.0
5.0E+12	2.539E-3	3.8	4.162E-4	18.1	2.955E-3	3.9
6.0E+12	2.324E-3	3.6	3.342E-4	19.4	2.658E-3	3.7
8.0E+12	1.960E-3	3.2	2.154E-4	21.9	2.176E-3	3.4
1.0E+13	1.670E-3	2.8	1.389E-4	24.5	1.809E-3	3.1

Table 4.9
Tabular Data for Recommended Decay Heat Power for
Pulse Fast Fission of ^{240}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	5.876E+0	4.2	5.752E+0	4.0	1.163E+1	3.1
1.0E-1	5.819E+0	3.2	5.708E+0	3.0	1.153E+1	2.1
1.5E-1	5.792E+0	3.0	5.687E+0	2.9	1.148E+1	2.0
2.0E-1	5.766E+0	3.0	5.667E+0	2.9	1.143E+1	2.0
3.0E-1	5.716E+0	3.0	5.629E+0	2.7	1.134E+1	2.0
4.0E-1	5.669E+0	2.9	5.593E+0	2.7	1.126E+1	2.0
5.0E-1	5.624E+0	2.9	5.558E+0	2.7	1.118E+1	2.0
6.0E-1	5.581E+0	2.9	5.526E+0	2.7	1.111E+1	2.0
8.0E-1	5.500E+0	2.9	5.465E+0	2.6	1.097E+1	1.8
1.0E+0	5.426E+0	2.9	5.409E+0	2.6	1.084E+1	1.8
1.5E+0	5.263E+0	2.9	5.286E+0	2.4	1.055E+1	1.8
2.0E+0	5.124E+0	2.9	5.183E+0	2.4	1.031E+1	1.8
3.0E+0	4.895E+0	2.9	5.013E+0	2.3	9.908E+0	1.8
4.0E+0	4.712E+0	2.9	4.878E+0	2.3	9.590E+0	1.8
5.0E+0	4.562E+0	2.9	4.766E+0	2.3	9.327E+0	1.8
6.0E+0	4.434E+0	2.9	4.671E+0	2.3	9.104E+0	1.7
8.0E+0	4.227E+0	2.9	4.515E+0	2.1	8.742E+0	1.7
1.0E+1	4.066E+0	2.9	4.391E+0	2.1	8.456E+0	1.7
1.5E+1	3.775E+0	2.9	4.160E+0	2.1	7.935E+0	1.7
2.0E+1	3.576E+0	2.7	3.994E+0	2.0	7.570E+0	1.7
3.0E+1	3.307E+0	2.7	3.755E+0	2.0	7.062E+0	1.7
4.0E+1	3.123E+0	2.7	3.583E+0	1.8	6.707E+0	1.7
5.0E+1	2.984E+0	2.7	3.448E+0	1.8	6.432E+0	1.5
6.0E+1	2.871E+0	2.7	3.338E+0	1.8	6.209E+0	1.5
8.0E+1	2.698E+0	2.7	3.163E+0	1.7	5.861E+0	1.5
1.0E+2	2.569E+0	2.7	3.031E+0	1.7	5.600E+0	1.5
1.5E+2	2.350E+0	2.7	2.804E+0	1.5	5.154E+0	1.5
2.0E+2	2.208E+0	2.9	2.657E+0	1.5	4.865E+0	1.5
3.0E+2	2.023E+0	2.7	2.467E+0	1.5	4.490E+0	1.5
4.0E+2	1.898E+0	2.7	2.340E+0	1.5	4.238E+0	1.5
5.0E+2	1.802E+0	2.7	2.243E+0	1.5	4.045E+0	1.5
6.0E+2	1.724E+0	2.7	2.163E+0	1.5	3.887E+0	1.5
8.0E+2	1.601E+0	2.6	2.034E+0	1.7	3.634E+0	1.4
1.0E+3	1.505E+0	2.4	1.930E+0	1.7	3.435E+0	1.4
1.5E+3	1.334E+0	2.3	1.735E+0	1.7	3.069E+0	1.4
2.0E+3	1.218E+0	2.3	1.593E+0	1.7	2.812E+0	1.4
3.0E+3	1.070E+0	2.3	1.397E+0	1.7	2.468E+0	1.4
4.0E+3	9.798E-1	2.3	1.265E+0	1.7	2.245E+0	1.4
5.0E+3	9.184E-1	2.3	1.169E+0	1.5	2.088E+0	1.3
6.0E+3	8.735E-1	2.3	1.096E+0	1.5	1.970E+0	1.4
8.0E+3	8.109E-1	2.3	9.907E-1	1.4	1.802E+0	1.2

Table 4.9 (Continued)
Pulse Fast Fission of ^{240}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	7.676E-1	2.3	9.180E-1	1.4	1.686E+0	1.2
1.5E+4	6.963E-1	2.3	8.070E-1	1.2	1.503E+0	1.2
2.0E+4	6.491E-1	2.1	7.437E-1	1.2	1.393E+0	1.2
3.0E+4	5.853E-1	2.1	6.701E-1	1.2	1.255E+0	1.2
4.0E+4	5.414E-1	2.1	6.237E-1	1.2	1.165E+0	1.2
5.0E+4	5.087E-1	2.1	5.894E-1	1.2	1.098E+0	1.2
6.0E+4	4.832E-1	2.1	5.623E-1	1.2	1.046E+0	1.2
8.0E+4	4.460E-1	2.1	5.213E-1	1.2	9.673E-1	1.2
1.0E+5	4.198E-1	2.0	4.912E-1	1.2	9.110E-1	1.1
1.5E+5	3.783E-1	2.0	4.404E-1	1.4	8.187E-1	1.1
2.0E+5	3.534E-1	2.0	4.072E-1	1.4	7.606E-1	1.1
3.0E+5	3.235E-1	2.0	3.634E-1	1.4	6.869E-1	1.1
4.0E+5	3.050E-1	1.8	3.336E-1	1.4	6.386E-1	1.2
5.0E+5	2.917E-1	1.8	3.107E-1	1.5	6.024E-1	1.2
6.0E+5	2.812E-1	1.8	2.922E-1	1.5	5.734E-1	1.2
8.0E+5	2.652E-1	1.8	2.632E-1	1.7	5.284E-1	1.2
1.0E+6	2.532E-1	1.8	2.414E-1	1.5	4.946E-1	1.2
1.5E+6	2.323E-1	1.8	2.040E-1	1.7	4.363E-1	1.2
2.0E+6	2.181E-1	1.8	1.796E-1	1.8	3.977E-1	1.2
3.0E+6	1.991E-1	1.8	1.484E-1	2.0	3.475E-1	1.4
4.0E+6	1.866E-1	1.8	1.290E-1	2.1	3.156E-1	1.4
5.0E+6	1.774E-1	1.8	1.157E-1	2.1	2.932E-1	1.4
6.0E+6	1.703E-1	2.0	1.060E-1	2.1	2.763E-1	1.4
8.0E+6	1.592E-1	2.0	9.220E-2	2.1	2.514E-1	1.5
1.0E+7	1.506E-1	2.0	8.246E-2	2.1	2.330E-1	1.5
1.5E+7	1.341E-1	2.0	6.683E-2	2.0	2.010E-1	1.5
2.0E+7	1.218E-1	2.0	5.809E-2	2.1	1.798E-1	1.5
3.0E+7	1.033E-1	2.0	4.997E-2	2.3	1.533E-1	1.5
4.0E+7	8.966E-2	2.0	4.668E-2	2.3	1.363E-1	1.5
5.0E+7	7.912E-2	2.0	4.493E-2	2.4	1.241E-1	1.5
6.0E+7	7.085E-2	2.0	4.375E-2	2.4	1.146E-1	1.5
8.0E+7	5.909E-2	2.0	4.205E-2	2.6	1.011E-1	1.5
1.0E+8	5.160E-2	2.0	4.081E-2	2.6	9.241E-2	1.7
1.5E+8	4.226E-2	2.1	3.869E-2	2.6	8.095E-2	1.8
2.0E+8	3.845E-2	2.3	3.719E-2	2.7	7.564E-2	1.8
3.0E+8	3.494E-2	2.3	3.480E-2	2.9	6.974E-2	2.0
4.0E+8	3.262E-2	2.4	3.270E-2	3.0	6.532E-2	2.0
5.0E+8	3.060E-2	2.4	3.077E-2	3.2	6.137E-2	2.1
6.0E+8	2.875E-2	2.4	2.899E-2	3.3	5.774E-2	2.3
8.0E+8	2.548E-2	2.6	2.580E-2	3.6	5.128E-2	2.4

Table 4.9 (Continued)
Pulse Fast Fission of ^{240}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	2.268E-2	2.7	2.305E-2	3.9	4.573E-2	2.6
1.5E+9	1.730E-2	3.0	1.769E-2	4.4	3.499E-2	2.7
2.0E+9	1.360E-2	3.3	1.397E-2	4.7	2.758E-2	3.2
3.0E+9	9.328E-3	3.6	9.592E-3	5.4	1.892E-2	3.5
4.0E+9	7.305E-3	3.9	7.475E-3	6.2	1.478E-2	3.9
5.0E+9	6.345E-3	4.2	6.453E-3	7.0	1.280E-2	4.2
6.0E+9	5.887E-3	4.4	5.958E-3	7.7	1.185E-2	5.0
8.0E+9	5.560E-3	4.7	5.603E-3	9.2	1.116E-2	5.7
1.0E+10	5.480E-3	4.7	5.518E-3	10.7	1.100E-2	5.7
1.5E+10	5.446E-3	4.7	5.488E-3	10.7	1.093E-2	5.7
2.0E+10	5.439E-3	4.7	5.481E-3	10.7	1.092E-2	5.7
3.0E+10	5.431E-3	4.7	5.469E-3	10.7	1.090E-2	5.7
4.0E+10	5.423E-3	4.7	5.457E-3	10.7	1.088E-2	5.7
5.0E+10	5.416E-3	4.7	5.445E-3	10.8	1.086E-2	5.7
6.0E+10	5.409E-3	4.7	5.433E-3	10.8	1.084E-2	5.7
8.0E+10	5.394E-3	4.7	5.409E-3	10.8	1.080E-2	5.7
1.0E+11	5.380E-3	4.7	5.386E-3	10.8	1.077E-2	5.7
1.5E+11	5.344E-3	4.7	5.327E-3	11.0	1.067E-2	5.7
2.0E+11	5.308E-3	4.7	5.269E-3	11.1	1.058E-2	5.9
3.0E+11	5.237E-3	4.7	5.154E-3	11.3	1.039E-2	5.9
4.0E+11	5.168E-3	4.7	5.042E-3	11.6	1.021E-2	6.0
5.0E+11	5.099E-3	4.7	4.932E-3	11.7	1.003E-2	6.0
6.0E+11	5.032E-3	4.7	4.825E-3	12.0	9.857E-3	6.2
8.0E+11	4.901E-3	4.7	4.618E-3	12.5	9.519E-3	6.3
1.0E+12	4.774E-3	4.7	4.419E-3	12.9	9.193E-3	6.5
1.5E+12	4.475E-3	4.7	3.959E-3	14.5	8.435E-3	6.8
2.0E+12	4.200E-3	4.7	3.547E-3	16.2	7.747E-3	7.0
3.0E+12	3.712E-3	4.5	2.847E-3	19.7	6.560E-3	7.4
4.0E+12	3.297E-3	4.5	2.286E-3	23.0	5.582E-3	7.8
5.0E+12	2.941E-3	4.5	1.835E-3	26.3	4.775E-3	8.3
6.0E+12	2.635E-3	4.4	1.473E-3	29.7	4.108E-3	8.9
8.0E+12	2.143E-3	4.4	9.490E-4	36.3	3.092E-3	9.8
1.0E+13	1.771E-3	4.4	6.115E-4	43.1	2.383E-3	10.7

Table 4.10
Tabular Data for Recommended Decay Heat Power for
Pulse Thermal Fission of ^{241}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
0.0	6.765E+0	4.2	6.467E+0	4.0	1.323E+1	3.1
1.0E-1	6.684E+0	3.2	6.406E+0	3.0	1.309E+1	2.1
1.5E-1	6.645E+0	3.0	6.377E+0	2.9	1.302E+1	2.0
2.0E-1	6.608E+0	3.0	6.350E+0	2.9	1.296E+1	2.0
3.0E-1	6.538E+0	3.0	6.297E+0	2.7	1.283E+1	2.0
4.0E-1	6.471E+0	2.9	6.247E+0	2.7	1.272E+1	2.0
5.0E-1	6.409E+0	2.9	6.200E+0	2.7	1.261E+1	2.0
6.0E-1	6.349E+0	2.9	6.156E+0	2.7	1.251E+1	2.0
8.0E-1	6.239E+0	2.9	6.074E+0	2.6	1.231E+1	1.8
1.0E+0	6.138E+0	2.9	5.999E+0	2.6	1.214E+1	1.8
1.5E+0	5.918E+0	2.9	5.836E+0	2.4	1.175E+1	1.8
2.0E+0	5.733E+0	2.9	5.699E+0	2.4	1.143E+1	1.8
3.0E+0	5.433E+0	2.9	5.478E+0	2.3	1.091E+1	1.8
4.0E+0	5.197E+0	2.9	5.303E+0	2.3	1.050E+1	1.8
5.0E+0	5.005E+0	2.9	5.159E+0	2.3	1.016E+1	1.8
6.0E+0	4.844E+0	2.9	5.038E+0	2.3	9.882E+0	1.7
8.0E+0	4.588E+0	2.9	4.842E+0	2.1	9.430E+0	1.7
1.0E+1	4.391E+0	2.9	4.688E+0	2.1	9.079E+0	1.7
1.5E+1	4.044E+0	2.9	4.408E+0	2.1	8.452E+0	1.7
2.0E+1	3.810E+0	2.7	4.209E+0	2.0	8.020E+0	1.7
3.0E+1	3.498E+0	2.7	3.930E+0	2.0	7.429E+0	1.7
4.0E+1	3.287E+0	2.7	3.731E+0	1.8	7.018E+0	1.7
5.0E+1	3.128E+0	2.7	3.576E+0	1.8	6.704E+0	1.5
6.0E+1	3.000E+0	2.7	3.449E+0	1.8	6.449E+0	1.5
8.0E+1	2.805E+0	2.7	3.251E+0	1.7	6.056E+0	1.5
1.0E+2	2.660E+0	2.7	3.102E+0	1.7	5.762E+0	1.5
1.5E+2	2.418E+0	2.7	2.900E+0	1.5	5.268E+0	1.5
2.0E+2	2.262E+0	2.9	2.689E+0	1.5	4.951E+0	1.5
3.0E+2	2.062E+0	2.7	2.484E+0	1.5	4.546E+0	1.5
4.0E+2	1.927E+0	2.7	2.349E+0	1.5	4.277E+0	1.5
5.0E+2	1.826E+0	2.7	2.248E+0	1.5	4.073E+0	1.5
6.0E+2	1.743E+0	2.7	2.165E+0	1.5	3.908E+0	1.5
8.0E+2	1.613E+0	2.6	2.032E+0	1.7	3.646E+0	1.4
1.0E+3	1.514E+0	2.4	1.927E+0	1.7	3.441E+0	1.4
1.5E+3	1.337E+0	2.3	1.729E+0	1.7	3.066E+0	1.4
2.0E+3	1.219E+0	2.3	1.587E+0	1.7	2.805E+0	1.4
3.0E+3	1.068E+0	2.3	1.389E+0	1.7	2.457E+0	1.4
4.0E+3	9.771E-1	2.3	1.256E+0	1.7	2.233E+0	1.4
5.0E+3	9.156E-1	2.3	1.160E+0	1.5	2.075E+0	1.3
6.0E+3	8.710E-1	2.3	1.086E+0	1.5	1.957E+0	1.4
8.0E+3	8.092E-1	2.3	9.801E-1	1.4	1.789E+0	1.2

Table 4.10 (Continued)
Pulse Thermal Fission of ^{241}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+4	7.667E-1	2.3	9.070E-1	1.4	1.674E+0	1.2
1.5E+4	6.976E-1	2.3	7.961E-1	1.2	1.494E+0	1.2
2.0E+4	6.526E-1	2.1	7.338E-1	1.2	1.386E+0	1.2
3.0E+4	5.924E-1	2.1	6.627E-1	1.2	1.255E+0	1.2
4.0E+4	5.513E-1	2.1	6.185E-1	1.2	1.170E+0	1.2
5.0E+4	5.205E-1	2.1	5.859E-1	1.2	1.106E+0	1.2
6.0E+4	4.965E-1	2.1	5.600E-1	1.2	1.057E+0	1.2
8.0E+4	4.611E-1	2.1	5.209E-1	1.2	9.821E-1	1.2
1.0E+5	4.361E-1	2.0	4.923E-1	1.2	9.284E-1	1.1
1.5E+5	3.961E-1	2.0	4.443E-1	1.4	8.404E-1	1.1
2.0E+5	3.718E-1	2.0	4.130E-1	1.4	7.848E-1	1.1
3.0E+5	3.425E-1	2.0	3.714E-1	1.4	7.139E-1	1.1
4.0E+5	3.241E-1	1.8	3.426E-1	1.4	6.667E-1	1.2
5.0E+5	3.107E-1	1.8	3.201E-1	1.5	6.308E-1	1.2
6.0E+5	3.001E-1	1.8	3.016E-1	1.5	6.017E-1	1.2
8.0E+5	2.837E-1	1.8	2.721E-1	1.7	5.558E-1	1.2
1.0E+6	2.713E-1	1.8	2.493E-1	1.5	5.206E-1	1.2
1.5E+6	2.493E-1	1.8	2.092E-1	1.7	4.585E-1	1.2
2.0E+6	2.342E-1	1.8	1.825E-1	1.8	4.167E-1	1.2
3.0E+6	2.140E-1	1.8	1.482E-1	2.0	3.622E-1	1.4
4.0E+6	2.007E-1	1.8	1.273E-1	2.1	3.280E-1	1.4
5.0E+6	1.911E-1	1.8	1.134E-1	2.1	3.045E-1	1.4
6.0E+6	1.836E-1	2.0	1.035E-1	2.1	2.871E-1	1.4
8.0E+6	1.721E-1	2.0	8.990E-2	2.1	2.620E-1	1.5
1.0E+7	1.630E-1	2.0	8.056E-2	2.1	2.435E-1	1.5
1.5E+7	1.453E-1	2.0	6.571E-2	2.0	2.110E-1	1.5
2.0E+7	1.315E-1	2.0	5.733E-2	2.1	1.889E-1	1.5
3.0E+7	1.105E-1	2.0	4.935E-2	2.3	1.599E-1	1.5
4.0E+7	9.476E-2	2.0	4.592E-2	2.3	1.407E-1	1.5
5.0E+7	8.249E-2	2.0	4.399E-2	2.4	1.265E-1	1.5
6.0E+7	7.282E-2	2.0	4.264E-2	2.4	1.155E-1	1.5
8.0E+7	5.908E-2	2.0	4.069E-2	2.6	9.977E-2	1.5
1.0E+8	5.033E-2	2.0	3.928E-2	2.6	8.961E-2	1.5
1.5E+8	3.954E-2	2.1	3.691E-2	2.6	7.645E-2	1.7
2.0E+8	3.530E-2	2.3	3.530E-2	2.7	7.060E-2	1.8
3.0E+8	3.174E-2	2.3	3.275E-2	2.9	6.449E-2	1.8
4.0E+8	2.954E-2	2.4	3.053E-2	3.0	6.007E-2	2.0
5.0E+8	2.766E-2	2.4	2.848E-2	3.2	5.613E-2	2.0
6.0E+8	2.594E-2	2.4	2.658E-2	3.3	5.252E-2	2.1
8.0E+8	2.290E-2	2.6	2.317E-2	3.6	4.607E-2	2.3

Table 4.10 (Continued)
Pulse Thermal Fission of ^{241}Pu and for Irradiation of 10^{13} Seconds

Time (s)	Beta (MeV/s/fission)	Uncertainty (percent)	Gamma (MeV/s/fission)	Uncertainty (percent)	Total (MeV/s/fission)	Uncertainty (percent)
1.0E+9	2.029E-2	2.7	2.023E-2	3.9	4.052E-2	2.4
1.5E+9	1.526E-2	3.0	1.451E-2	4.4	2.977E-2	2.6
2.0E+9	1.180E-2	3.3	1.054E-2	4.7	2.235E-2	2.7
3.0E+9	7.795E-3	3.6	5.863E-3	5.4	1.366E-2	3.2
4.0E+9	5.892E-3	3.9	3.604E-3	6.2	9.496E-3	3.5
5.0E+9	4.986E-3	4.2	2.513E-3	7.0	7.499E-3	3.9
6.0E+9	4.553E-3	4.4	1.986E-3	7.7	6.538E-3	4.2
8.0E+9	4.241E-3	4.5	1.608E-3	9.2	5.849E-3	5.0
1.0E+10	4.164E-3	4.7	1.519E-3	10.7	5.684E-3	5.7
1.5E+10	4.132E-3	4.7	1.492E-3	10.7	5.624E-3	5.7
2.0E+10	4.126E-3	4.7	1.489E-3	10.7	5.615E-3	5.7
3.0E+10	4.120E-3	4.7	1.486E-3	10.7	5.606E-3	5.7
4.0E+10	4.116E-3	4.7	1.483E-3	10.7	5.598E-3	5.7
5.0E+10	4.111E-3	4.7	1.480E-3	10.8	5.591E-3	5.7
6.0E+10	4.107E-3	4.7	1.476E-3	10.8	5.583E-3	5.7
8.0E+10	4.098E-3	4.7	1.470E-3	10.8	5.568E-3	5.7
1.0E+11	4.089E-3	4.7	1.463E-3	10.8	5.553E-3	5.7
1.5E+11	4.068E-3	4.7	1.448E-3	11.0	5.515E-3	5.7
2.0E+11	4.046E-3	4.7	1.432E-3	11.1	5.478E-3	5.9
3.0E+11	4.004E-3	4.7	1.401E-3	11.3	5.404E-3	5.9
4.0E+11	3.962E-3	4.7	1.370E-3	11.6	5.332E-3	6.0
5.0E+11	3.920E-3	4.7	1.340E-3	11.7	5.261E-3	6.0
6.0E+11	3.879E-3	4.7	1.311E-3	12.0	5.191E-3	6.2
8.0E+11	3.799E-3	4.7	1.255E-3	12.5	5.054E-3	6.3
1.0E+12	3.721E-3	4.7	1.201E-3	12.9	4.922E-3	6.5
1.5E+12	3.534E-3	4.7	1.076E-3	14.5	4.610E-3	6.8
2.0E+12	3.359E-3	4.7	9.640E-4	16.2	4.323E-3	7.0
3.0E+12	3.041E-3	4.5	7.738E-4	19.7	3.815E-3	7.4
4.0E+12	2.760E-3	4.5	6.212E-4	23.0	3.381E-3	7.8
5.0E+12	2.511E-3	4.5	4.987E-4	26.3	3.010E-3	8.3
6.0E+12	2.290E-3	4.4	4.003E-4	29.7	2.691E-3	8.9
8.0E+12	1.919E-3	4.4	2.580E-4	36.3	2.177E-3	9.8
1.0E+13	1.624E-3	4.4	1.663E-4	43.1	1.791E-3	10.7

Table 4.11
Tabular Data for Recommended Decay Heat Power for Thermal Fission of ^{235}U and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
0.0	6.392E+0	6.398E+0	1.279E+1
1.0E-1	6.319E+0	6.340E+0	1.266E+1
1.5E-1	6.284E+0	6.314E+0	1.260E+1
2.0E-1	6.251E+0	6.288E+0	1.254E+1
3.0E-1	6.188E+0	6.239E+0	1.243E+1
4.0E-1	6.129E+0	6.193E+0	1.232E+1
5.0E-1	6.074E+0	6.150E+0	1.222E+1
6.0E-1	6.022E+0	6.110E+0	1.213E+1
8.0E-1	5.926E+0	6.035E+0	1.196E+1
1.0E+0	5.839E+0	5.968E+0	1.181E+1
1.5E+0	5.648E+0	5.822E+0	1.147E+1
2.0E+0	5.488E+0	5.700E+0	1.119E+1
3.0E+0	5.226E+0	5.504E+0	1.073E+1
4.0E+0	5.018E+0	5.349E+0	1.037E+1
5.0E+0	4.846E+0	5.222E+0	1.007E+1
6.0E+0	4.700E+0	5.114E+0	9.814E+0
8.0E+0	4.465E+0	4.939E+0	9.404E+0
1.0E+1	4.282E+0	4.800E+0	9.082E+0
1.5E+1	3.957E+0	4.542E+0	8.500E+0
2.0E+1	3.737E+0	4.358E+0	8.095E+0
3.0E+1	3.445E+0	4.094E+0	7.540E+0
4.0E+1	3.250E+0	3.905E+0	7.155E+0
5.0E+1	3.102E+0	3.757E+0	6.860E+0
6.0E+1	2.985E+0	3.635E+0	6.620E+0
8.0E+1	2.805E+0	3.443E+0	6.247E+0
1.0E+2	2.671E+0	3.295E+0	5.965E+0
1.5E+2	2.445E+0	3.037E+0	5.482E+0
2.0E+2	2.300E+0	2.866E+0	5.166E+0
3.0E+2	2.112E+0	2.646E+0	4.758E+0
4.0E+2	1.986E+0	2.501E+0	4.487E+0
5.0E+2	1.890E+0	2.392E+0	4.282E+0
6.0E+2	1.813E+0	2.304E+0	4.116E+0
8.0E+2	1.691E+0	2.163E+0	3.854E+0
1.0E+3	1.597E+0	2.053E+0	3.650E+0
1.5E+3	1.426E+0	1.848E+0	3.274E+0
2.0E+3	1.307E+0	1.700E+0	3.008E+0
3.0E+3	1.152E+0	1.494E+0	2.646E+0
4.0E+3	1.054E+0	1.354E+0	2.407E+0
5.0E+3	9.850E-1	1.250E+0	2.235E+0
6.0E+3	9.333E-1	1.168E+0	2.102E+0
8.0E+3	8.580E-1	1.048E+0	1.906E+0

Table 4.11 (Continued)
Thermal Fission of ^{235}U and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+4	8.035E-1	9.633E-1	1.767E+0
1.5E+4	7.095E-1	8.300E-1	1.540E+0
2.0E+4	6.453E-1	7.529E-1	1.398E+0
3.0E+4	5.578E-1	6.643E-1	1.222E+0
4.0E+4	4.988E-1	6.106E-1	1.109E+0
5.0E+4	4.560E-1	5.722E-1	1.028E+0
6.0E+4	4.237E-1	5.426E-1	9.662E-1
8.0E+4	3.785E-1	4.986E-1	8.771E-1
1.0E+5	3.484E-1	4.669E-1	8.152E-1
1.5E+5	3.033E-1	4.145E-1	7.178E-1
2.0E+5	2.775E-1	3.814E-1	6.589E-1
3.0E+5	2.479E-1	3.387E-1	5.866E-1
4.0E+5	2.297E-1	3.098E-1	5.395E-1
5.0E+5	2.163E-1	2.874E-1	5.038E-1
6.0E+5	2.056E-1	2.690E-1	4.746E-1
8.0E+5	1.888E-1	2.396E-1	4.284E-1
1.0E+6	1.757E-1	2.170E-1	3.927E-1
1.5E+6	1.520E-1	1.771E-1	3.291E-1
2.0E+6	1.353E-1	1.505E-1	2.857E-1
3.0E+6	1.123E-1	1.160E-1	2.283E-1
4.0E+6	9.698E-2	9.466E-2	1.916E-1
5.0E+6	8.590E-2	8.010E-2	1.660E-1
6.0E+6	7.740E-2	6.937E-2	1.468E-1
8.0E+6	6.491E-2	5.398E-2	1.189E-1
1.0E+7	5.593E-2	4.286E-2	9.879E-2
1.5E+7	4.135E-2	2.451E-2	6.587E-2
2.0E+7	3.269E-2	1.418E-2	4.686E-2
3.0E+7	2.291E-2	5.219E-3	2.813E-2
4.0E+7	1.728E-2	2.469E-3	1.975E-2
5.0E+7	1.344E-2	1.579E-3	1.502E-2
6.0E+7	1.063E-2	1.247E-3	1.187E-2
8.0E+7	6.925E-3	1.000E-3	7.925E-3
1.0E+8	4.785E-3	8.890E-4	5.674E-3
1.5E+8	2.550E-3	7.608E-4	3.310E-3
2.0E+8	1.913E-3	7.058E-4	2.619E-3
3.0E+8	1.578E-3	6.445E-4	2.223E-3
4.0E+8	1.435E-3	5.971E-4	2.032E-3
5.0E+8	1.321E-3	5.544E-4	1.875E-3
6.0E+8	1.219E-3	5.151E-4	1.735E-3
8.0E+8	1.043E-3	4.451E-4	1.488E-3

Table 4.11 (Continued)
Thermal Fission of ^{235}U and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+9	8.937E-4	3.847E-4	1.278E-3
1.5E+9	6.087E-4	2.673E-4	8.760E-4
2.0E+9	4.152E-4	1.857E-4	6.010E-4
3.0E+9	1.936E-4	8.968E-5	2.833E-4
4.0E+9	9.044E-5	4.331E-5	1.338E-4
5.0E+9	4.230E-5	2.092E-5	6.322E-5
6.0E+9	1.983E-5	1.010E-5	2.993E-5
8.0E+9	4.403E-6	2.363E-6	6.765E-6
1.0E+10	1.016E-6	5.576E-7	1.573E-6
1.5E+10	5.990E-8	2.325E-8	8.315E-8
2.0E+10	2.773E-8	9.213E-9	3.694E-8
3.0E+10	2.2°1E-8	8.815E-9	3.162E-8
4.0E+10	2.240E-8	8.795E-9	3.119E-8
5.0E+10	2.234E-8	8.776E-9	3.111E-8
6.0E+10	2.231E-8	8.756E-9	3.107E-8
8.0E+10	2.226E-8	8.718E-9	3.098E-8
1.0E+11	2.221E-8	8.680E-9	3.089E-8
1.5E+11	2.209E-8	8.585E-9	3.067E-8
2.0E+11	2.197E-8	8.491E-9	3.046E-8
3.0E+11	2.173E-8	8.306E-9	3.003E-8
4.0E+11	2.149E-8	8.126E-9	2.962E-8
5.0E+11	2.126E-8	7.949E-9	2.920E-8
6.0E+11	2.103E-8	7.776E-9	2.880E-8
8.0E+11	2.057E-8	7.442E-9	2.801E-8
1.0E+12	2.013E-8	7.122E-9	2.725E-8
1.5E+12	1.908E-8	6.380E-9	2.546E-8
2.0E+12	1.810E-8	5.716E-9	2.382E-8
3.0E+12	1.632E-8	4.589E-9	2.091E-8
4.0E+12	1.476E-8	3.683E-9	1.845E-8
5.0E+12	1.338E-8	2.957E-9	1.634E-8
6.0E+12	1.217E-8	2.373E-9	1.454E-8
8.0E+12	1.013E-8	1.530E-9	1.166E-8
1.0E+13	8.514E-9	9.858E-10	9.500E-9

Table 4.12
Tabular Data for Recommended Decay Heat Power for Thermal Fission of ^{239}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
0.0	5.168E+0	5.300E+0	1.047E+1
1.0E-1	5.124E+0	5.265E+0	1.039E+1
1.5E-1	5.103E+0	5.248E+0	1.035E+1
2.0E-1	5.083E+0	5.232E+0	1.031E+1
3.0E-1	5.043E+0	5.201E+0	1.024E+1
4.0E-1	5.006E+0	5.171E+0	1.018E+1
5.0E-1	4.970E+0	5.143E+0	1.011E+1
6.0E-1	4.936E+0	5.116E+0	1.005E+1
8.0E-1	4.871E+0	5.065E+0	9.937E+0
1.0E+0	4.812E+0	5.019E+0	9.831E+0
1.5E+0	4.680E+0	4.917E+0	9.597E+0
2.0E+0	4.567E+0	4.831E+0	9.397E+0
3.0E+0	4.381E+0	4.688E+0	9.069E+0
4.0E+0	4.232E+0	4.576E+0	8.806E+0
5.0E+0	4.109E+0	4.480E+0	8.588E+0
6.0E+0	4.004E+0	4.399E+0	8.403E+0
8.0E+0	3.834E+0	4.266E+0	8.100E+0
1.0E+1	3.700E+0	4.160E+0	7.860E+0
1.5E+1	3.456E+0	3.961E+0	7.417E+0
2.0E+1	3.286E+0	3.816E+0	7.102E+0
3.0E+1	3.052E+0	3.607E+0	6.659E+0
4.0E+1	2.890E+0	3.455E+0	6.345E+0
5.0E+1	2.766E+0	3.336E+0	6.102E+0
6.0E+1	2.666E+0	3.237E+0	5.902E+0
8.0E+1	2.510E+0	3.080E+0	5.590E+0
1.0E+2	2.392E+0	2.960E+0	5.352E+0
1.5E+2	2.192E+0	2.750E+0	4.942E+0
2.0E+2	2.060E+0	2.611E+0	4.670E+0
3.0E+2	1.886E+0	2.427E+0	4.313E+0
4.0E+2	1.767E+0	2.303E+0	4.070E+0
5.0E+2	1.675E+0	2.207E+0	3.882E+0
6.0E+2	1.601E+0	2.127E+0	3.728E+0
8.0E+2	1.482E+0	1.999E+0	3.481E+0
1.0E+3	1.390E+0	1.896E+0	3.286E+0
1.5E+3	1.224E+0	1.704E+0	2.928E+0
2.0E+3	1.112E+0	1.565E+0	2.677E+0
3.0E+3	9.684E-1	1.373E+0	2.341E+0
4.0E+3	8.807E-1	1.244E+0	2.125E+0
5.0E+3	8.212E-1	1.151E+0	1.973E+0
6.0E+3	7.777E-1	1.081E+0	1.858E+0
8.0E+3	7.167E-1	9.777E-1	1.694E+0

Table 4.12 (Continued)
Thermal Fission of ^{239}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+4	6.744E-1	9.067E-1	1.581E+0
1.5E+4	6.026E-1	7.971E-1	1.400E+0
2.0E+4	5.547E-1	7.337E-1	1.288E+0
3.0E+4	4.893E-1	6.587E-1	1.148E+0
4.0E+4	4.444E-1	6.111E-1	1.055E+0
5.0E+4	4.109E-1	5.758E-1	9.867E-1
6.0E+4	3.849E-1	5.478E-1	9.327E-1
8.0E+4	3.479E-1	5.055E-1	8.526E-1
1.0E+5	3.206E-1	4.743E-1	7.949E-1
1.5E+5	2.789E-1	4.211E-1	6.999E-1
2.0E+5	2.537E-1	3.860E-1	6.397E-1
3.0E+5	2.236E-1	3.392E-1	5.628E-1
4.0E+5	2.049E-1	3.071E-1	5.120E-1
5.0E+5	1.914E-1	2.824E-1	4.738E-1
6.0E+5	1.807E-1	2.624E-1	4.431E-1
8.0E+5	1.645E-1	2.311E-1	3.956E-1
1.0E+6	1.523E-1	2.076E-1	3.599E-1
1.5E+6	1.313E-1	1.674E-1	2.987E-1
2.0E+6	1.173E-1	1.413E-1	2.585E-1
3.0E+6	9.899E-2	1.080E-1	2.070E-1
4.0E+6	8.737E-2	8.739E-2	1.748E-1
5.0E+6	7.925E-2	7.341E-2	1.527E-1
6.0E+6	7.315E-2	6.318E-2	1.363E-1
8.0E+6	6.432E-2	4.876E-2	1.131E-1
1.0E+7	5.790E-2	3.865E-2	9.658E-2
1.5E+7	4.700E-2	2.259E-2	6.959E-2
2.0E+7	3.972E-2	1.385E-2	5.357E-2
3.0E+7	3.002E-2	6.292E-3	3.631E-2
4.0E+7	2.343E-2	3.787E-3	2.721E-2
5.0E+7	1.851E-2	2.790E-3	2.130E-2
6.0E+7	1.472E-2	2.287E-3	1.701E-2
8.0E+7	9.469E-3	1.725E-3	1.119E-2
1.0E+8	6.238E-3	1.393E-3	7.631E-3
1.5E+8	2.547E-3	9.831E-4	3.530E-3
2.0E+8	1.383E-3	8.246E-4	2.205E-3
3.0E+8	8.293E-4	7.066E-4	1.536E-3
4.0E+8	7.102E-4	6.463E-4	1.356E-3
5.0E+8	6.465E-4	5.974E-4	1.244E-3
6.0E+8	5.953E-4	5.541E-4	1.149E-3
8.0E+8	5.093E-4	4.781E-4	9.871E-4

Table 4.12 (Continued)
Thermal Fission of ^{239}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+9	4.367E-4	4.133E-4	8.497E-4
1.5E+9	2.989E-4	2.869E-4	5.858E-4
2.0E+9	2.053E-4	1.994E-4	4.044E-4
3.0E+9	9.672E-5	9.628E-5	1.930E-4
4.0E+9	4.577E-5	4.651E-5	9.228E-5
5.0E+9	2.173E-5	2.248E-5	4.422E-5
6.0E+9	1.038E-5	1.088E-5	2.126E-5
8.0E+9	2.447E-6	2.569E-6	5.015E-6
1.0E+10	6.383E-7	6.318E-7	1.270E-6
1.5E+10	7.628E-8	5.825E-8	1.345E-7
2.0E+10	4.321E-8	4.314E-8	8.635E-8
3.0E+10	3.471E-8	4.263E-8	7.734E-8
4.0E+10	3.395E-8	4.254E-8	7.649E-8
5.0E+10	3.384E-8	4.245E-8	7.628E-8
6.0E+10	3.378E-8	4.235E-8	7.613E-8
8.0E+10	3.368E-8	4.217E-8	7.585E-8
1.0E+11	3.358E-8	4.198E-8	7.556E-8
1.5E+11	3.333E-8	4.152E-8	7.485E-8
2.0E+11	3.308E-8	4.107E-8	7.415E-8
3.0E+11	3.260E-8	4.017E-8	7.277E-8
4.0E+11	3.212E-8	3.930E-8	7.142E-8
5.0E+11	3.165E-8	3.845E-8	7.009E-8
6.0E+11	3.119E-8	3.761E-8	6.880E-8
8.0E+11	3.029E-8	3.599E-8	6.628E-8
1.0E+12	2.946E-8	3.440E-8	6.386E-8
1.5E+12	2.738E-8	3.086E-8	5.824E-8
2.0E+12	2.552E-8	2.765E-8	5.316E-8
3.0E+12	2.224E-8	2.219E-8	4.443E-8
4.0E+12	1.947E-8	1.781E-8	3.728E-8
5.0E+12	1.712E-8	1.430E-8	3.142E-8
6.0E+12	1.513E-8	1.148E-8	2.661E-8
8.0E+12	1.197E-8	7.395E-9	1.936E-8
1.0E+13	9.629E-9	4.765E-9	1.440E-8

Table 4.13
Tabular Data for Recommended Decay Heat Power for Fast Fission of ^{238}U and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
0.0	8.208E+0	7.779E+0	1.599E+1
1.0E-1	8.064E+0	7.672E+0	1.574E+1
1.5E-1	7.997E+0	7.620E+0	1.562E+1
2.0E-1	7.934E+0	7.575E+0	1.551E+1
3.0E-1	7.810E+0	7.486E+0	1.530E+1
4.0E-1	7.704E+0	7.400E+0	1.511E+1
5.0E-1	7.601E+0	7.328E+0	1.493E+1
6.0E-1	7.505E+0	7.257E+0	1.476E+1
8.0E-1	7.329E+0	7.126E+0	1.446E+1
1.0E+0	7.171E+0	7.010E+0	1.418E+1
1.5E+0	6.833E+0	6.761E+0	1.359E+1
2.0E+0	6.555E+0	6.556E+0	1.311E+1
3.0E+0	6.114E+0	6.234E+0	1.235E+1
4.0E+0	5.777E+0	5.988E+0	1.177E+1
5.0E+0	5.508E+0	5.791E+0	1.130E+1
6.0E+0	5.286E+0	5.628E+0	1.091E+1
8.0E+0	4.942E+0	5.371E+0	1.031E+1
1.0E+1	4.684E+0	5.176E+0	9.860E+0
1.5E+1	4.246E+0	4.831E+0	9.077E+0
2.0E+1	3.964E+0	4.595E+0	8.559E+0
3.0E+1	3.603E+0	4.271E+0	7.874E+0
4.0E+1	3.366E+0	4.046E+0	7.412E+0
5.0E+1	3.191E+0	3.871E+0	7.062E+0
6.0E+1	3.052E+0	3.729E+0	6.781E+0
8.0E+1	2.841E+0	3.505E+0	6.346E+0
1.0E+2	2.685E+0	3.336E+0	6.021E+0
1.5E+2	2.424E+0	3.044E+0	5.468E+0
2.0E+2	2.258E+0	2.853E+0	5.111E+0
3.0E+2	2.047E+0	2.611E+0	4.659E+0
4.0E+2	1.910E+0	2.456E+0	4.366E+0
5.0E+2	1.809E+0	2.342E+0	4.150E+0
6.0E+2	1.727E+0	2.251E+0	3.978E+0
8.0E+2	1.600E+0	2.108E+0	3.708E+0
1.0E+3	1.502E+0	1.997E+0	3.499E+0
1.5E+3	1.327E+0	1.791E+0	3.118E+0
2.0E+3	1.207E+0	1.644E+0	2.850E+0
3.0E+3	1.052E+0	1.440E+0	2.492E+0
4.0E+3	9.572E-1	1.304E+0	2.261E+0
5.0E+3	8.922E-1	1.204E+0	2.096E+0
6.0E+3	8.443E-1	1.127E+0	1.971E+0
8.0E+3	7.761E-1	1.015E+0	1.791E+0

Table 4.13 (Continued)
Fast Fission of ^{238}U and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+4	7.279E-1	9.354E-1	1.663E+0
1.5E+4	6.464E-1	8.124E-1	1.459E+0
2.0E+4	5.914E-1	7.418E-1	1.333E+0
3.0E+4	5.164E-1	6.606E-1	1.177E+0
4.0E+4	4.653E-1	6.103E-1	1.076E+0
5.0E+4	4.277E-1	5.737E-1	1.001E+0
6.0E+4	3.990E-1	5.450E-1	9.440E-1
8.0E+4	3.581E-1	5.020E-1	8.601E-1
1.0E+5	3.303E-1	4.707E-1	8.009E-1
1.5E+5	2.875E-1	4.184E-1	7.059E-1
2.0E+5	2.625E-1	3.845E-1	6.469E-1
3.0E+5	2.328E-1	3.396E-1	5.724E-1
4.0E+5	2.144E-1	3.087E-1	5.231E-1
5.0E+5	2.009E-1	2.846E-1	4.855E-1
6.0E+5	1.902E-1	2.649E-1	4.550E-1
8.0E+5	1.736E-1	2.338E-1	4.074E-1
1.0E+6	1.611E-1	2.101E-1	3.712E-1
1.5E+6	1.390E-1	1.690E-1	3.080E-1
2.0E+6	1.239E-1	1.426E-1	2.665E-1
3.0E+6	1.037E-1	1.085E-1	2.122E-1
4.0E+6	9.057E-2	8.743E-2	1.780E-1
5.0E+6	8.125E-2	7.319E-2	1.544E-1
6.0E+6	7.418E-2	6.284E-2	1.370E-1
8.0E+6	6.388E-2	4.832E-2	1.122E-1
1.0E+7	5.646E-2	3.812E-2	9.458E-2
1.5E+7	4.411E-2	2.192E-2	6.602E-2
2.0E+7	3.632E-2	1.307E-2	4.939E-2
3.0E+7	2.668E-2	5.470E-3	3.215E-2
4.0E+7	2.054E-2	3.046E-3	2.358E-2
5.0E+7	1.610E-2	2.157E-3	1.826E-2
6.0E+7	1.275E-2	1.746E-3	1.450E-2
8.0E+7	8.188E-3	1.335E-3	9.524E-3
1.0E+8	5.443E-3	1.106E-3	6.549E-3
1.5E+8	2.403E-3	8.280E-4	3.231E-3
2.0E+8	1.476E-3	7.200E-4	2.196E-3
3.0E+8	1.029E-3	6.331E-4	1.662E-3
4.0E+8	9.070E-4	5.825E-4	1.490E-3
5.0E+8	8.295E-4	5.398E-4	1.369E-3
6.0E+8	7.645E-4	5.011E-4	1.266E-3
8.0E+8	6.524E-4	4.324E-4	1.086E-3

Table 4.13 (Continued)
Fast Fission of ^{238}U and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+9	5.605E-4	3.739E-4	9.344E-4
1.5E+9	3.828E-4	2.597E-4	6.425E-4
2.0E+9	2.619E-4	1.805E-4	4.424E-4
3.0E+9	1.229E-4	8.714E-5	2.100E-4
4.0E+9	5.778E-5	4.208E-5	9.986E-5
5.0E+9	2.725E-5	2.032E-5	4.758E-5
6.0E+9	1.291E-5	9.819E-6	2.270E-5
8.0E+9	2.975E-6	2.297E-6	5.272E-6
1.0E+10	7.423E-7	5.434E-7	1.286E-6
1.5E+10	6.840E-8	2.415E-8	9.256E-8
2.0E+10	3.242E-8	1.051E-8	4.293E-8
3.0E+10	2.361E-8	1.012E-8	3.370E-8
4.0E+10	2.284E-8	1.009E-8	3.294E-8
5.0E+10	2.275E-8	1.007E-8	3.283E-8
6.0E+10	2.272E-8	1.005E-8	3.277E-8
8.0E+10	2.267E-8	1.000E-8	3.268E-8
1.0E+11	2.262E-8	9.960E-9	3.258E-8
1.5E+11	2.249E-8	9.852E-9	3.234E-8
2.0E+11	2.237E-8	9.744E-9	3.211E-8
3.0E+11	2.212E-8	9.532E-9	3.165E-8
4.0E+11	2.187E-8	9.325E-9	3.119E-8
5.0E+11	2.163E-8	9.120E-9	3.075E-8
6.0E+11	2.139E-8	8.924E-9	3.031E-8
8.0E+11	2.092E-8	8.540E-9	2.946E-8
1.0E+12	2.046E-8	8.173E-9	2.863E-8
1.5E+12	1.937E-8	7.322E-9	2.669E-8
2.0E+12	1.835E-8	6.561E-9	2.491E-8
3.0E+12	1.651E-8	5.266E-9	2.177E-8
4.0E+12	1.489E-8	4.227E-9	1.912E-8
5.0E+12	1.347E-8	3.393E-9	1.686E-8
6.0E+12	1.221E-8	2.720E-9	1.493E-8
8.0E+12	1.012E-8	1.756E-9	1.187E-8
1.0E+13	8.462E-9	1.132E-9	9.593E-9

Table 4.14
Tabular Data for Recommended Decay Heat Power for Fast Fission of ^{240}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
0.0	5.775E+0	5.702E+0	1.148E+1
1.0E-1	5.718E+0	5.659E+0	1.138E+1
1.5E-1	5.692E+0	5.638E+0	1.133E+1
2.0E-1	5.665E+0	5.618E+0	1.128E+1
3.0E-1	5.615E+0	5.580E+0	1.120E+1
4.0E-1	5.568E+0	5.543E+0	1.111E+1
5.0E-1	5.523E+0	5.509E+0	1.103E+1
6.0E-1	5.480E+0	5.476E+0	1.096E+1
8.0E-1	5.400E+0	5.416E+0	1.082E+1
1.0E+0	5.325E+0	5.360E+0	1.069E+1
1.5E+0	5.162E+0	5.237E+0	1.040E+1
2.0E+0	5.023E+0	5.133E+0	1.016E+1
3.0E+0	4.793E+0	4.964E+0	9.758E+0
4.0E+0	4.612E+0	4.828E+0	9.440E+0
5.0E+0	4.461E+0	4.717E+0	9.177E+0
6.0E+0	4.333E+0	4.621E+0	8.955E+0
8.0E+0	4.126E+0	4.466E+0	8.592E+0
1.0E+1	3.965E+0	4.341E+0	8.306E+0
1.5E+1	3.675E+0	4.111E+0	7.785E+0
2.0E+1	3.475E+0	3.940E+0	7.420E+0
3.0E+1	3.206E+0	3.706E+0	6.912E+0
4.0E+1	3.022E+0	3.534E+0	6.556E+0
5.0E+1	2.883E+0	3.399E+0	6.282E+0
6.0E+1	2.770E+0	3.288E+0	6.059E+0
8.0E+1	2.597E+0	3.114E+0	5.711E+0
1.0E+2	2.468E+0	2.982E+0	5.450E+0
1.5E+2	2.249E+0	2.755E+0	5.004E+0
2.0E+2	2.107E+0	2.607E+0	4.714E+0
3.0E+2	1.922E+0	2.417E+0	4.339E+0
4.0E+2	1.797E+0	2.291E+0	4.088E+0
5.0E+2	1.701E+0	2.194E+0	3.895E+0
6.0E+2	1.623E+0	2.114E+0	3.737E+0
8.0E+2	1.500E+0	1.984E+0	3.484E+0
1.0E+3	1.404E+0	1.881E+0	3.285E+0
1.5E+3	1.233E+0	1.686E+0	2.919E+0
2.0E+3	1.118E+0	1.544E+0	2.662E+0
3.0E+3	9.696E-1	1.348E+0	2.317E+0
4.0E+3	8.789E-1	1.216E+0	2.095E+0
5.0E+3	8.175E-1	1.120E+0	1.938E+0
6.0E+3	7.726E-1	1.047E+0	1.820E+0
8.0E+3	7.100E-1	9.415E-1	1.652E+0

Table 4.14 (Continued)
Fast Fission of ^{240}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+4	6.667E-1	8.688E-1	1.536E+0
1.5E+4	5.955E-1	7.578E-1	1.353E+0
2.0E+4	5.482E-1	6.945E-1	1.243E+0
3.0E+4	4.844E-1	6.208E-1	1.105E+0
4.0E+4	4.406E-1	5.745E-1	1.015E+0
5.0E+4	4.079E-1	5.402E-1	9.481E-1
6.0E+4	3.824E-1	5.131E-1	8.955E-1
8.0E+4	3.452E-1	4.721E-1	8.172E-1
1.0E+5	3.190E-1	4.419E-1	7.610E-1
1.5E+5	2.776E-1	3.912E-1	6.688E-1
2.0E+5	2.527E-1	3.580E-1	6.108E-1
3.0E+5	2.230E-1	3.143E-1	5.373E-1
4.0E+5	2.047E-1	2.845E-1	4.892E-1
5.0E+5	1.915E-1	2.617E-1	4.532E-1
6.0E+5	1.812E-1	2.431E-1	4.243E-1
8.0E+5	1.655E-1	2.143E-1	3.797E-1
1.0E+6	1.538E-1	1.925E-1	3.463E-1
1.5E+6	1.335E-1	1.553E-1	2.889E-1
2.0E+6	1.200E-1	1.311E-1	2.511E-1
3.0E+6	1.025E-1	1.002E-1	2.027E-1
4.0E+6	9.130E-2	8.114E-2	1.724E-1
5.0E+6	8.348E-2	6.819E-2	1.517E-1
6.0E+6	7.760E-2	5.872E-2	1.363E-1
8.0E+6	6.901E-2	4.542E-2	1.144E-1
1.0E+7	6.271E-2	3.611E-2	9.881E-2
1.5E+7	5.166E-2	2.139E-2	7.304E-2
2.0E+7	4.405E-2	1.337E-2	5.742E-2
3.0E+7	3.357E-2	6.377E-3	3.992E-2
4.0E+7	2.625E-2	3.990E-3	3.024E-2
5.0E+7	2.074E-2	2.992E-3	2.371E-2
6.0E+7	1.648E-2	2.452E-3	1.893E-2
8.0E+7	1.056E-2	1.825E-3	1.238E-2
1.0E+8	6.903E-3	1.448E-3	8.351E-3
1.5E+8	2.723E-3	9.845E-4	3.707E-3
2.0E+8	1.401E-3	8.088E-4	2.210E-3
3.0E+8	7.847E-4	6.844E-4	1.469E-3
4.0E+8	6.608E-4	6.245E-4	1.285E-3
5.0E+8	5.992E-4	5.771E-4	1.176E-3
6.0E+8	5.510E-4	5.351E-4	1.086E-3
8.0E+8	4.707E-4	4.615E-4	9.322E-4

Table 4.14 (Continued)
Fast Fission of ^{240}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+9	4.037E-4	3.987E-4	8.024E-4
1.5E+9	2.764E-4	2.769E-4	5.533E-4
2.0E+9	1.897E-4	1.924E-4	3.821E-4
3.0E+9	8.958E-5	9.293E-5	1.825E-4
4.0E+9	4.245E-5	4.489E-5	8.735E-5
5.0E+9	2.020E-5	2.170E-5	4.190E-5
6.0E+9	9.677E-6	1.050E-5	2.018E-5
8.0E+9	2.305E-6	2.483E-6	4.787E-6
1.0E+10	6.140E-7	6.133E-7	1.227E-6
1.5E+10	7.862E-8	5.969E-8	1.380E-7
2.0E+10	4.423E-8	4.511E-8	8.933E-8
3.0E+10	3.497E-8	4.461E-8	7.959E-8
4.0E+10	3.415E-8	4.451E-8	7.866E-8
5.0E+10	3.403E-8	4.442E-8	7.844E-8
6.0E+10	3.397E-8	4.432E-8	7.829E-8
8.0E+10	3.387E-8	4.412E-8	7.799E-8
1.0E+11	3.376E-8	4.393E-8	7.769E-8
1.5E+11	3.351E-8	4.345E-8	7.696E-8
2.0E+11	3.326E-8	4.297E-8	7.623E-8
3.0E+11	3.276E-8	4.204E-8	7.480E-8
4.0E+11	3.227E-8	4.112E-8	7.339E-8
5.0E+11	3.179E-8	4.023E-8	7.200E-8
6.0E+11	3.132E-8	3.936E-8	7.068E-8
8.0E+11	3.041E-8	3.766E-8	6.807E-8
1.0E+12	2.953E-8	3.603E-8	6.556E-8
1.5E+12	2.745E-8	3.229E-8	5.974E-8
2.0E+12	2.555E-8	2.893E-8	5.448E-8
3.0E+12	2.221E-8	2.322E-8	4.543E-8
4.0E+12	1.940E-8	1.864E-8	3.804E-8
5.0E+12	1.703E-8	1.496E-8	3.199E-8
6.0E+12	1.501E-8	1.201E-8	2.702E-8
8.0E+12	1.183E-8	7.739E-9	1.957E-8
1.0E+13	9.492E-9	4.986E-9	1.448E-8

Table 4.15
Tabular Data for Recommended Decay Heat Power for Thermal Fission of ^{241}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
0.0	6.657E+0	6.419E+0	1.308E+1
1.0E-1	6.576E+0	6.357E+0	1.293E+1
1.5E-1	6.538E+0	6.329E+0	1.287E+1
2.0E-1	6.501E+0	6.301E+0	1.280E+1
3.0E-1	6.430E+0	6.248E+0	1.268E+1
4.0E-1	6.364E+0	6.199E+0	1.256E+1
5.0E-1	6.301E+0	6.152E+0	1.245E+1
6.0E-1	6.242E+0	6.108E+0	1.235E+1
8.0E-1	6.131E+0	6.025E+0	1.216E+1
1.0E+0	6.030E+0	5.951E+0	1.198E+1
1.5E+0	5.810E+0	5.788E+0	1.160E+1
2.0E+0	5.625E+0	5.651E+0	1.128E+1
3.0E+0	5.325E+0	5.429E+0	1.075E+1
4.0E+0	5.089E+0	5.254E+0	1.034E+1
5.0E+0	4.897E+0	5.110E+0	1.001E+1
6.0E+0	4.736E+0	4.989E+0	9.725E+0
8.0E+0	4.480E+0	4.793E+0	9.273E+0
1.0E+1	4.283E+0	4.640E+0	8.922E+0
1.5E+1	3.936E+0	4.359E+0	8.295E+0
2.0E+1	3.703E+0	4.161E+0	7.864E+0
3.0E+1	3.391E+0	3.882E+0	7.272E+0
4.0E+1	3.179E+0	3.683E+0	6.862E+0
5.0E+1	3.020E+0	3.527E+0	6.547E+0
6.0E+1	2.892E+0	3.401E+0	6.293E+0
8.0E+1	2.697E+0	3.203E+0	5.899E+0
1.0E+2	2.552E+0	3.054E+0	5.606E+0
1.5E+2	2.310E+0	2.801E+0	5.111E+0
2.0E+2	2.154E+0	2.640E+0	4.795E+0
3.0E+2	1.954E+0	2.435E+0	4.389E+0
4.0E+2	1.820E+0	2.301E+0	4.121E+0
5.0E+2	1.718E+0	2.199E+0	3.917E+0
6.0E+2	1.635E+0	2.116E+0	3.752E+0
8.0E+2	1.506E+0	1.984E+0	3.489E+0
1.0E+3	1.406E+0	1.878E+0	3.284E+0
1.5E+3	1.229E+0	1.681E+0	2.910E+0
2.0E+3	1.111E+0	1.538E+0	2.649E+0
3.0E+3	9.606E-1	1.340E+0	2.301E+0
4.0E+3	8.693E-1	1.207E+0	2.077E+0
5.0E+3	8.078E-1	1.111E+0	1.919E+0
6.0E+3	7.632E-1	1.037E+0	1.801E+0
8.0E+3	7.014E-1	9.314E-1	1.633E+0

Table 4.15 (Continued)
Thermal Fission of ^{241}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+4	6.590E-1	8.584E-1	1.517E+0
1.5E+4	5.899E-1	7.474E-1	1.337E+0
2.0E+4	5.449E-1	6.852E-1	1.230E+0
3.0E+4	4.847E-1	6.141E-1	1.099E+0
4.0E+4	4.436E-1	5.699E-1	1.013E+0
5.0E+4	4.128E-1	5.372E-1	9.501E-1
6.0E+4	3.888E-1	5.114E-1	9.002E-1
8.0E+4	3.535E-1	4.723E-1	8.258E-1
1.0E+5	3.285E-1	4.437E-1	7.722E-1
1.5E+5	2.886E-1	3.957E-1	6.843E-1
2.0E+5	2.644E-1	3.645E-1	6.288E-1
3.0E+5	2.352E-1	3.229E-1	5.581E-1
4.0E+5	2.170E-1	2.941E-1	5.111E-1
5.0E+5	2.038E-1	2.717E-1	4.755E-1
6.0E+5	1.933E-1	2.532E-1	4.465E-1
8.0E+5	1.773E-1	2.238E-1	4.011E-1
1.0E+6	1.652E-1	2.011E-1	3.663E-1
1.5E+6	1.441E-1	1.612E-1	3.053E-1
2.0E+6	1.298E-1	1.346E-1	2.644E-1
3.0E+6	1.112E-1	1.007E-1	2.119E-1
4.0E+6	9.948E-2	8.013E-2	1.796E-1
5.0E+6	9.140E-2	6.655E-2	1.580E-1
6.0E+6	8.540E-2	5.693E-2	1.423E-1
8.0E+6	7.670E-2	4.387E-2	1.206E-1
1.0E+7	7.032E-2	3.499E-2	1.053E-1
1.5E+7	5.888E-2	2.114E-2	8.002E-2
2.0E+7	5.071E-2	1.358E-2	6.429E-2
3.0E+7	3.902E-2	6.884E-3	4.590E-2
4.0E+7	3.063E-2	4.497E-3	3.512E-2
5.0E+7	2.423E-2	3.431E-3	2.766E-2
6.0E+7	1.926E-2	2.817E-3	2.208E-2
8.0E+7	1.231E-2	2.074E-3	1.438E-2
1.0E+8	8.006E-3	1.621E-3	9.627E-3
1.5E+8	3.050E-3	1.066E-3	4.116E-3
2.0E+8	1.474E-3	8.619E-4	2.336E-3
3.0E+8	7.496E-4	7.258E-4	1.475E-3
4.0E+8	6.172E-4	6.641E-4	1.281E-3
5.0E+8	5.580E-4	6.150E-4	1.173E-3
6.0E+8	5.130E-4	5.708E-4	1.084E-3
8.0E+8	4.386E-4	4.927E-4	9.313E-4

Table 4.15 (Continued)
Thermal Fission of ^{241}Pu and for Irradiation of One Year

Time (s)	Beta (MeV/s/fission)	Gamma (MeV/s/fission)	Total (MeV/s/fission)
1.0E+9	3.765E-4	4.257E-4	8.022E-4
1.5E+9	2.583E-4	2.957E-4	5.540E-4
2.0E+9	1.776E-4	2.055E-4	3.830E-4
3.0E+9	8.414E-5	9.921E-5	1.834E-4
4.0E+9	4.001E-5	4.791E-5	8.792E-5
5.0E+9	1.911E-5	2.314E-5	4.225E-5
6.0E+9	9.186E-6	1.118E-5	2.037E-5
8.0E+9	2.208E-6	2.616E-6	4.824E-6
1.0E+10	5.932E-7	6.193E-7	1.213E-6
1.5E+10	7.076E-8	2.811E-8	9.887E-8
2.0E+10	3.436E-8	1.257E-8	4.693E-8
3.0E+10	2.418E-8	1.212E-8	3.630E-8
4.0E+10	2.329E-8	1.209E-8	3.539E-8
5.0E+10	2.319E-8	1.207E-8	3.526E-8
6.0E+10	2.316E-8	1.204E-8	3.520E-8
8.0E+10	2.310E-8	1.199E-8	3.509E-8
1.0E+11	2.305E-8	1.194E-8	3.498E-8
1.5E+11	2.291E-8	1.181E-8	3.472E-8
2.0E+11	2.278E-8	1.168E-8	3.446E-8
3.0E+11	2.252E-8	1.142E-8	3.394E-8
4.0E+11	2.226E-8	1.117E-8	3.343E-8
5.0E+11	2.200E-8	1.093E-8	3.293E-8
6.0E+11	2.175E-8	1.069E-8	3.244E-8
8.0E+11	2.126E-8	1.023E-8	3.149E-8
1.0E+12	2.078E-8	9.794E-9	3.057E-8
1.5E+12	1.964E-8	8.775E-9	2.841E-8
2.0E+12	1.857E-8	7.861E-9	2.643E-8
3.0E+12	1.664E-8	6.310E-9	2.296E-8
4.0E+12	1.496E-8	5.065E-9	2.003E-8
5.0E+12	1.348E-8	4.066E-9	1.755E-8
6.0E+12	1.219E-8	3.264E-9	1.545E-8
8.0E+12	1.003E-8	2.103E-9	1.213E-8
1.0E+13	8.334E-9	1.356E-9	9.689E-9

Table 4.16 Coefficients of The Fitting Formula to The Recommended Decay Heat Power Expressed by An Exponential Function with 33 Terms

NO.	FITTING PARAMETER(U235(T)) LAMBDA	ALPHA BETA	ALPHA GAMMA	ALPHA (B+G)	FITTING PARAMETER(U238(F)) LAMBDA	ALPHA BETA	ALPHA GAMMA	ALPHA (B+G)	FITTING PARAMETER(PU239(T)) LAMBDA	ALPHA BETA	ALPHA GAMMA	ALPHA (B+G)
1	3.290E+00	1.150E-01	8.444E-02	1.994E-01	3.290E+00	3.123E-01	2.291E-01	5.414E-01	3.290E+00	2.662E-02	2.583E-02	5.245E-02
2	1.001E+00	2.508E-01	2.155E-01	4.664E-01	1.001E+00	4.471E-01	3.489E-01	7.961E-01	1.001E+00	1.114E-01	1.030E-01	2.144E-01
3	5.157E-01	4.057E-02	1.714E-02	5.771E-02	5.157E-01	1.784E-01	1.285E-01	3.069E-01	5.157E-01	7.752E-02	4.459E-02	1.221E-01
4	2.951E-01	1.611E-01	1.853E-01	3.465E-01	2.951E-01	2.916E-01	2.422E-01	5.338E-01	2.951E-01	9.289E-02	9.995E-02	1.928E-01
5	1.959E-01	6.855E-02	-1.216E-02	5.639E-02	1.959E-01	1.145E-01	4.232E-02	1.569E-01	1.959E-01	4.694E-02	8.433E-03	5.537E-02
6	1.037E-01	9.400E-02	7.517E-02	1.692E-01	1.037E-01	1.275E-01	9.535E-02	2.228E-01	1.037E-01	6.619E-02	5.279E-02	1.190E-01
7	3.488E-02	2.527E-02	1.917E-02	4.445E-02	3.488E-02	3.019E-02	2.458E-02	5.477E-02	3.488E-02	1.989E-02	1.571E-02	3.561E-02
8	1.330E-02	1.054E-02	1.229E-02	2.283E-02	1.330E-02	1.236E-02	1.369E-02	2.605E-02	1.330E-02	9.396E-03	1.016E-02	1.955E-02
9	5.004E-03	1.548E-03	2.407E-03	3.954E-03	5.004E-03	2.330E-03	3.547E-03	5.877E-03	5.004E-03	1.117E-03	1.420E-03	2.537E-03
10	3.591E-03	6.661E-04	3.944E-04	1.061E-03	3.591E-03	3.496E-04	-9.722E-05	2.524E-04	3.591E-03	7.202E-04	5.707E-04	1.291E-03
11	1.357E-03	5.515E-04	6.150E-04	1.166E-03	1.357E-03	6.272E-04	6.641E-04	1.291E-03	1.357E-03	5.854E-04	5.116E-04	1.097E-03
12	5.645E-04	3.301E-04	3.098E-04	6.399E-04	5.645E-04	3.373E-04	3.188E-04	6.560E-04	5.645E-04	3.212E-04	3.433E-04	6.645E-04
13	1.850E-04	5.432E-05	1.397E-04	1.940E-04	1.850E-04	5.122E-05	1.333E-04	1.846E-04	1.850E-04	4.628E-05	1.184E-04	1.647E-04
14	5.435E-05	-1.095E-06	3.115E-05	3.005E-05	5.435E-05	-4.952E-06	1.834E-05	1.338E-05	5.435E-05	-1.940E-06	8.409E-06	6.469E-06
15	4.918E-05	1.712E-05	-1.563E-05	1.487E-06	4.918E-05	1.751E-05	-5.795E-06	1.172E-05	4.918E-05	1.231E-05	2.127E-06	1.444E-05
16	1.922E-05	3.978E-06	4.368E-06	8.346E-06	1.922E-05	3.409E-06	3.972E-06	7.381E-06	1.922E-05	3.081E-06	3.367E-06	6.447E-06
17	8.422E-06	8.348E-07	8.133E-07	1.648E-06	8.422E-06	8.375E-07	7.797E-07	1.617E-06	8.422E-06	8.943E-07	9.373E-07	1.832E-06
18	2.443E-06	1.622E-07	3.057E-07	4.679E-07	2.443E-06	1.836E-07	3.661E-07	5.497E-07	2.443E-06	1.903E-07	3.843E-07	5.746E-07
19	6.925E-07	2.042E-08	7.855E-08	9.897E-08	6.925E-07	1.857E-08	6.085E-08	7.942E-08	6.925E-07	2.648E-08	6.020E-08	8.668E-08
20	6.202E-07	4.074E-08	4.294E-08	8.368E-08	6.202E-07	3.839E-08	5.611E-08	9.450E-08	6.202E-07	2.905E-08	5.173E-08	8.078E-08
21	1.503E-07	2.611E-08	-3.612E-08	-1.001E-08	1.503E-07	2.611E-08	-1.504E-08	1.107E-08	1.503E-07	2.835E-08	-1.092E-08	1.744E-08
22	1.277E-07	-1.816E-07	2.691E-07	8.752E-08	1.277E-07	-2.058E-07	1.184E-07	-8.740E-08	1.277E-07	-2.518E-07	8.901E-08	-1.628E-07
23	1.262E-07	1.686E-07	-2.200E-07	-5.134E-08	1.262E-07	1.897E-07	-8.989E-08	9.985E-08	1.262E-07	2.315E-07	-6.438E-08	1.672E-07
24	2.714E-08	2.183E-09	1.546E-10	2.337E-09	2.714E-08	1.602E-09	8.249E-11	1.685E-09	2.714E-08	1.084E-09	4.515E-11	1.129E-09
25	2.251E-08	-9.660E-11	-4.788E-11	-1.445E-10	2.251E-08	7.845E-10	9.137E-11	8.759E-10	2.251E-08	1.536E-09	1.899E-10	1.726E-09
26	8.985E-09	9.695E-12	4.467E-12	1.416E-11	8.985E-09	1.639E-11	4.566E-12	2.096E-11	8.985E-09	1.549E-11	6.951E-12	2.244E-11
27	4.366E-09	3.932E-12	-3.961E-13	3.536E-12	4.366E-09	2.338E-12	6.572E-15	2.345E-12	4.366E-09	2.146E-12	1.951E-13	2.341E-12
28	7.707E-10	5.085E-11	1.105E-13	5.096E-11	7.707E-10	2.786E-11	6.335E-14	2.792E-11	7.707E-10	1.812E-11	7.235E-14	1.819E-11
29	7.280E-10	1.041E-11	2.544E-11	3.586E-11	7.280E-10	1.036E-11	2.476E-11	3.512E-11	7.280E-10	1.148E-11	2.734E-11	3.882E-11
30	2.430E-10	2.003E-14	1.667E-17	2.005E-14	2.430E-10	3.826E-14	1.336E-17	3.827E-14	2.430E-10	3.704E-14	4.726E-17	3.709E-14
31	2.198E-13	1.077E-16	2.813E-16	3.889E-16	2.198E-13	1.197E-16	3.228E-16	4.424E-16	2.198E-13	4.830E-16	1.359E-15	1.843E-15
32	1.026E-13	5.286E-16	-1.609E-19	5.285E-16	1.026E-13	5.362E-16	-1.543E-19	5.361E-16	1.026E-13	5.285E-16	4.372E-19	5.289E-16
33	9.550E-15	7.527E-17	7.903E-20	7.535E-17	9.550E-15	6.903E-17	9.018E-20	6.912E-17	9.550E-15	6.839E-17	-9.745E-20	6.829E-17

$$f(t) = \sum_{i=1}^{33} \alpha_i \exp(-\lambda_i t) \qquad \text{MeV/s/fission}$$

$$F(T,t) = \sum_{i=1}^{33} \frac{\alpha_i}{\lambda_i} \exp(-\lambda_i t)\cdot(1 - \exp(-\lambda_i T)) \qquad \text{MeV/fission}$$

Table 4.16 Coefficients of The Fitting Formula to The Recommended Decay Heat Power Expressed by An Exponential Function with 33 Terms (Continued)

	FITTING PARAMETER(PU240(F))				FITTING PARAMETER(PU241(T))			
	LAMBDA	ALPHA			LAMBDA	ALPHA		
NO.		BETA	GAMMA	(B+G)		BETA	GAMMA	(B+G)
1	3.290E+00	4.756E-02	3.858E-02	8.614E-02	3.290E+00	9.660E-02	7.512E-02	1.717E-01
2	1.001E+00	1.561E-01	1.324E-01	2.884E-01	1.001E+00	2.408E-01	1.925E-01	4.332E-01
3	5.157E-01	7.998E-02	5.854E-02	1.385E-01	5.157E-01	1.072E-01	8.029E-02	1.875E-01
4	2.951E-01	1.214E-01	1.026E-01	2.240E-01	2.951E-01	1.821E-01	1.258E-01	3.079E-01
5	1.959E-01	6.224E-02	2.831E-02	9.055E-02	1.959E-01	8.150E-02	5.665E-02	1.382E-01
6	1.037E-01	8.037E-02	5.647E-02	1.368E-01	1.037E-01	9.318E-02	6.498E-02	1.582E-01
7	3.488E-02	2.308E-02	1.903E-02	4.211E-02	3.488E-02	2.766E-02	2.349E-02	5.116E-02
8	1.330E-02	1.056E-02	1.160E-02	2.217E-02	1.330E-02	1.183E-02	1.299E-02	2.483E-02
9	5.004E-03	1.056E-03	1.395E-03	2.452E-03	5.004E-03	1.124E-03	1.708E-03	2.832E-03
10	3.591E-03	8.841E-04	6.189E-04	1.503E-03	3.591E-03	1.016E-03	5.894E-04	1.605E-03
11	1.357E-03	5.887E-04	4.905E-04	1.079E-03	1.357E-03	6.283E-04	5.195E-04	1.148E-03
12	5.645E-04	3.329E-04	3.506E-04	6.835E-04	5.645E-04	3.373E-04	3.443E-04	6.816E-04
13	1.850E-04	4.998E-05	1.248E-04	1.748E-04	1.850E-04	5.070E-05	1.287E-04	1.794E-04
14	5.435E-05	-3.139E-06	6.714E-06	3.575E-06	5.4[illegible]E-05	-1.248E-06	6.907E-06	5.659E-06
15	4.918E-05	1.283E-05	3.206E-06	1.604E-05	4.918E-05	1.032E-05	2.364E-06	1.268E-05
16	1.922E-05	3.000E-06	3.347E-06	6.347E-06	1.922E-05	2.848E-06	3.377E-06	6.225E-06
17	8.422E-06	9.128E-07	8.970E-07	1.810E-06	8.422E-06	8.852E-07	7.819E-07	1.667E-06
18	2.443E-06	1.845E-07	3.527E-07	5.372E-07	2.443E-06	1.761E-07	3.119E-07	4.880E-07
19	6.925E-07	2.440E-08	5.305E-08	7.746E-08	6.925E-07	2.070E-08	5.130E-08	7.199E-08
20	6.202E-07	2.922E-08	5.059E-08	7.981E-08	6.202E-07	3.836E-08	6.984E-08	1.082E-07
21	1.503E-07	2.631E-08	-8.566E-09	1.774E-08	1.503E-07	2.580E-08	-7.571E-09	1.823E-08
22	1.277E-07	-2.390E-07	7.275E-08	-1.662E-07	1.277E-07	-2.420E-07	6.274E-08	-1.793E-07
23	1.262E-07	2.200E-07	-5.150E-08	1.685E-07	1.262E-07	2.228E-07	-4.340E-08	1.794E-07
24	2.714E-08	1.144E-09	3.962E-11	1.183E-09	2.714E-08	1.132E-09	2.088E-11	1.153E-09
25	2.251E-08	1.801E-09	2.277E-10	2.029E-09	2.251E-08	2.291E-09	3.016E-10	2.592E-09
26	8.985E-09	1.678E-11	6.813E-12	2.360E-11	8.985E-09	1.944E-11	4.521E-12	2.396E-11
27	4.366E-09	2.416E-12	3.192E-13	2.735E-12	4.366E-09	1.582E-12	1.572E-13	1.739E-12
28	7.707E-10	1.613E-11	8.809E-14	1.622E-11	7.707E-10	1.353E-11	8.343E-14	1.361E-11
29	7.280E-10	1.117E-11	2.637E-11	3.755E-11	7.280E-10	1.190E-11	2.818E-11	4.007E-11
30	2.430E-10	4.057E-14	3.648E-17	4.060E-14	2.430E-10	4.470E-14	2.103E-17	4.473E-14
31	2.198E-13	5.054E-16	1.423E-15	1.928E-15	2.198E-13	1.371E-16	3.867E-16	5.238E-16
32	1.026E-13	5.143E-16	4.736E-20	5.143E-16	1.026E-13	5.416E-16	-1.390E-19	5.414E-16
33	9.550E-15	6.643E-17	1.366E-20	6.644E-17	9.550E-15	6.027E-17	8.109E-20	6.035E-17

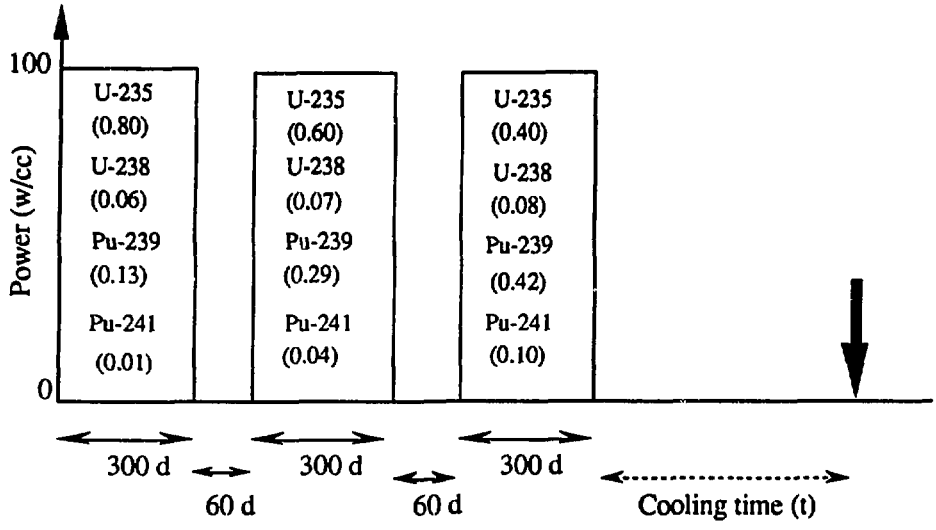
Concluding Remarks

The data for the radio-activity and the decay heat from the spent fuels are basic data for the radiation shielding, the instrumentation and the hazard evaluation of nuclear facilities as well as their design and operation. Therefore, the basic study on this subject has continuously been done in the US and Europe and the efforts have been directed at standardization of these data. In Japan, noticeable progress has been made, too, both in theoretical and experimental aspects of the decay heat study for the last decade. Starting from the fruits of those efforts, the Committee of "Standardization of Decay Heat Power in Nuclear Reactors", organized in the Atomic Energy Society of Japan, critically reviewed the status of the decay heat study, and prepared and published two reports; "Decay Heat Power in Nuclear Reactor and Its Recommended Values" (Aug., 1989) and "Recommended Values of Decay Heat Power and Methods to Utilize the Data" (July, 1990). Both of these are written, originally in Japanese, in order to present the recommended value in an easy-to-handle way. The present report is an English translation of the latter.

The present recommendation is more inclusive and has wider scope than the existing recommendations or standards. Or more specifically, the present recommendation has the following new features in addition to an intention of improving the reliability when being applied to the problems related to planned or accidental shutdown of LWRs; ① refined treatment for the FP neutron-capture effect, which guarantees the reasonable corrections even in very long cooling-time ranges, ② explicit inclusion of Pu-240 and -241 leading to more proper application to FBRs, ③ presentation of the delayed (or FP) gamma-ray energy spectrum data, and ④ preparation of the floppy diskettes for personal computers, which assists the users in applying the present recommendation to their own problems. Though the inclusion of the decay heat uncertainties is one of the new and useful features of the present work, much effort is left to be done before these uncertainties become quite reliable and widely acceptable. In fact, uncertainties estimated by many authors in the world differ fairly largely from each other especially in the short and the long cooling time ranges. This may be caused by the differences in the error estimations in important physical quantities such as the decay energies, the decay constants, and the fission yields, as well as the difference in the treatment of the error propagation.

Appendix A Example

Here we adopt the sample problem given in the ANS5.1 Draft (1979) as an example. The reactor operating history is shown in the following figure.



The number in parentheses indicates the power fraction of each nuclide.

C28/F = 0.80 (U-238 capture/total fission)

The calculated decay heat power is shown in Table A.1. The column indicated as "ACT.HEAT" gives the non-FP decay heat due to the decays of U-239 and Np-239 generated from U-238 by its neutron-capture. Table A.2 shows the fissile-wise contributions to the total FP decay heat power with their uncertainties on the 1-σ confidence level.

Table A.1 Calculated Decay Heat (W/cc)

T(S)	BETA	GAMMA	TOTAL	ACT.HEAT
+1.000E+00	+2.678E+00	+2.725E+00	+5.404E+00	+3.439E-01
+1.000E+01	+1.972E+00	+2.192E+00	+4.164E+00	+3.431E-01
+1.000E+02	+1.237E+00	+1.511E+00	+2.748E+00	+3.357E-01
+1.000E+03	+7.262E-01	+9.480E-01	+1.674E+00	+2.769E-01
+1.000E+04	+3.629E-01	+4.476E-01	+8.105E-01	+1.684E-01
+1.000E+05	+1.720E-01	+2.265E-01	+3.985E-01	+1.231E-01
+1.000E+06	+8.999E-02	+1.028E-01	+1.928E-01	+5.739E-03
+1.000E+07	+3.714E-02	+2.031E-02	+5.744E-02	+2.796E-16
+1.000E+08	+4.500E-03	+1.164E-03	+5.663E-03	+0.000E+00
+1.000E+09	+8.406E-04	+4.615E-04	+1.302E-03	+0.000E+00

Table A.2 Fissile-wise Contributions to the Decay Heat (W/cc) and their Uncertainties (1 σ, %)

T(S)	U-235	STD	U-238	STD	PU-239	STD
+1.00E+00	+2.335E+00	+0.9	+5.489E-01	+1.0	+1.960E+00	+1.0
+1.00E+01	+1.798E+00	+0.9	+3.818E-01	+1.1	+1.567E+00	+1.0
+1.00E+02	+1.184E+00	+0.9	+2.335E-01	+1.0	+1.068E+00	+1.0
+1.00E+03	+7.278E-01	+0.9	+1.360E-01	+1.0	+6.567E-01	+0.9
+1.00E+04	+3.568E-01	+0.8	+6.511E-02	+1.0	+3.175E-01	+0.8
+1.00E+05	+1.694E-01	+0.8	+3.180E-02	+1.0	+1.610E-01	+0.7
+1.00E+06	+8.596E-02	+0.5	+1.517E-02	+0.8	+7.438E-02	+0.6
+1.00E+07	+2.638E-02	+0.3	+4.346E-03	+0.4	+2.159E-02	+0.7
+1.00E+08	+2.887E-03	+0.7	+3.907E-04	+1.0	+1.920E-03	+1.0
+1.00E+09	+9.058E-04	+2.1	+7.635E-05	+2.8	+2.741E-04	+2.6

T(S)	PU-240	STD	PU-241	STD	TOTAL	STD
+1.00E+00	+0.000E+00	+0.0	+5.601E-01	+1.5	+5.404E+00	+1.0
+1.00E+01	+0.000E+00	+0.0	+4.171E-01	+1.7	+4.164E+00	+1.0
+1.00E+02	+0.000E+00	+0.0	+2.622E-01	+1.5	+2.748E+00	+1.0
+1.00E+03	+0.000E+00	+0.0	+1.537E-01	+1.5	+1.674E+00	+1.0
+1.00E+04	+0.000E+00	+0.0	+7.109E-02	+1.5	+8.105E-01	+0.8
+1.00E+05	+0.000E+00	+0.0	+3.628E-02	+1.5	+3.985E-01	+0.8
+1.00E+06	+0.000E+00	+0.0	+1.730E-02	+1.3	+1.928E-01	+0.6
+1.00E+07	+0.000E+00	+0.0	+5.120E-03	+0.8	+5.744E-02	+0.5
+1.00E+08	+0.000E+00	+0.0	+4.656E-04	+1.8	+5.663E-03	+0.9
+1.00E+09	+0.000E+00	+0.0	+4.584E-05	+4.2	+1.302E-03	+2.3

Appendix B Physical Constants and Conversion Factors

Here are summarized several kinds of physical constants and conversion factors which are pertinent to decay heat problems. Some of these are the constants which the present recommended values are based on in some respects, and others may help the users in applying the recommended values to a particular problem.

B.1 Energy Conversion: 1 Watt = 6.2415×10^{12} MeV/sec

B.2 Total Energy Release per Fission (MeV)

		Q_{eff}	Q_c	Q_T
U-235	thermal	193.58	8.60	202.2
	fast	191.61	10.04	201.7
U-236	fast	191.67	10.66	202.3
U-238	fast	195.16	10.71	205.9
Pu-239	thermal	199.62	11.29	210.9
	fast	197.05	13.10	210.2
Pu-240	fast	198.00	12.96	211.0
Pu-241	thermal	201.65	11.57	213.2
Pu-242	fast	201.36	12.88	214.2
U-233	thermal	190.85	8.94	199.8
Th-232	fast	184.92	8.53	193.5

Q_{eff} = Effective energy release per fission (exclusive of the unrecoverable anti-neutrino energy)

Q_c = Neutron capture gamma-ray energy in typical thermal and fast reactors. Q_c is expressed as $Q_c = (\nu-1)Q_{nc}$, where Q_{nc} is taken to be 5.99 MeV due to Tasaka et al.(JAERI-1320 (1990)).

Q_T = Total recoverable energy ($Q_{eff}+Q_c$)

B.3 Neutron Capture Cross Sections

The neutron capture cross sections for 27 FP nuclides, which are predominantly responsible for the neutron capture effect on decay heat, are listed in Table B.3.1. These are derived from JENDL-3, the Japanese Evaluated Nuclear Data Library, version 3. They are the 2200 m/s cross sections, the Westcott's g-factors and the resonance integrals defined for two varieties of the energy intervals, namely, 0.625 eV - 5.53 keV (RI-1) and 0.5 eV - 20 MeV (RI-2). In calculating the g-factors the Maxwellian temperature, which governs the neutron-spectrum shape, is assumed to be 858 K, and the Doppler effect on the cross section is taken into account with the same temperature. The Westcott's g-factor here is defined below 0.625 eV. Lowering this upper energy limit from 0.625 eV to 0.5 eV affects the g-factor only by 0.1 % at most except the case for Eu-155, where the g-factor is reduced by 10 % with the same change of the upper energy limit. Table B.3.2 lists the one-group cross sections collapsed with three varieties of the neutron spectra typical for BWR(BWRU), PWR(PWRU) and for FBR(EMOPU/U/U/C). The symbols given in parentheses are the spectrum identifications given in the ORIGEN-2 code library from which the collapsing spectra are taken. The data in Tables B.3.1 and B.3.2 were supplied by the courtesy of T. Nakagawa and F. Masukawa of JAERI.

B.4 Typical Reactor Conditions Pertinent to Decay Heat Calculations

B.4.1 Fuel Compositions

PWR : 4 w/o enriched U-235
BWR : 3 w/o enriched U-235
FBR Demonstration Plant : 24 w/o Pu/(Pu+U) (IC:20, OC:28)
FBR Commercial Plant : 17 w/o Pu/(Pu+U) (IC:15, OC:19)
Pu239/Pu240/Pu241/Pu242=58/24/14/4

B.4.2 Neutron Flux

PWR $\phi_{th}=4.7\times10^{13}$, r=0.210, T_n=858K, $\phi_{tot}=3.4\times10^{14}$
BWR $\phi_{th}=4.2\times10^{13}$, r=0.140, T_n=858K, $\phi_{tot}=2.2\times10^{14}$
FBR $\phi_{tot}=5.0\times10^{15}$,

r= epi-thermal index $=\phi_{epi}/\phi_{th}/\Delta u$, $\Delta u=\ln(5530\text{eV}/0.625\text{eV})$

B.4.3 Power Density: PWR:105kW/l, BWR:50.5kW/l, FBR:250kW/l

B.4.4 C28/F: PWR: 0.5 - 0.7,
BWR: 0.5 - 0.8,
FBR(demonstration plant): 0.5-0.7(core), 4-20(blanket)
FBR(commercial plant) : 0.7-0.9(core), 4-10(blanket)

B.4.5 Power Contribution Ratio

PWR	BWR	U-235	U-238	Pu-239	Pu-240	Pu-241
10MWD/T	5MWD/T	0.75	0.05	0.20	0	0
20	10	0.60	0.06	0.31	0	0.03
30	20	0.45	0.06	0.40	0	0.09
40	30	0.29	0.06	0.50	0	0.15

FBR(demo-plant)	U-235	U-238	Pu-239	Pu-240	Pu-241
25MWD/T	0.01	0.13	0.63	0.06	0.17
50	0.01	0.13	0.65	0.06	0.15
70	0.01	0.13	0.67	0.06	0.13
90	0.01	0.13	0.68	0.06	0.12

Table B.3.1 2200m/s Cross Sections, Westcotts' g-Factors and Resonance Integrals for Important FP Nuclides

ISOTOPE	ZA	SIG-2200	g-FACTOR	RI - 1	RI - 2
MO- 98	42098	+1.304E-01	+1.004E+00	+6.064E+00	+6.557E+00
TC- 99	43099	+1.986E+01	+1.031E+00	+3.138E+02	+3.183E+02
RU- 102	44102	+1.236E+00	+9.982E-01	+3.324E+00	+4.306E+00
RU- 103	44103	+8.004E+00	+1.000E+00	+8.822E+01	+9.113E+01
RH- 105	45105	+1.601E+04	+1.000E+00	+4.696E+03	+5.459E+03
PD- 107	46107	+2.016E+00	+1.006E+00	+1.034E+02	+1.089E+02
PD- 108	46108	+8.534E+00	+1.007E+00	+2.510E+02	+2.526E+02
AG- 109	47109	+9.086E+01	+1.035E+00	+1.463E+03	+1.471E+03
AG- 110M	47110	+8.221E+01	+9.272E-01	+7.899E+01	+9.341E+01
I- 129	53129	+2.712E+01	+9.833E-01	+2.617E+01	+2.932E+01
XE- 135	54135	+2.710E+06	+1.088E+00	+4.171E+03	+7.669E+03
CS- 133	55133	+2.911E+01	+1.016E+00	+3.916E+02	+3.953E+02
CS- 134	55134	+1.402E+02	+9.885E-01	+9.361E+01	+1.048E+02
CS- 135	55135	+8.730E+00	+9.989E-01	+6.097E+01	+6.230E+01
CS- 136	55136	+1.301E+01	+1.001E+00	+5.558E+01	+5.719E+01
CS- 137	55137	+1.104E-01	+1.000E+00	+5.610E-01	+6.771E-01
LA- 139	57139	+8.957E+00	+9.995E-01	+1.102E+01	+1.161E+01
PR- 141	59141	+1.339E+01	+9.959E-01	+1.778E+01	+1.887E+01
PM- 147	61147	+1.682E+02	+9.811E-01	+2.195E+03	+2.207E+03
PM- 148	61148	+2.001E+03	+1.000E+00	+2.401E+03	+2.503E+03
PM- 148M	61148	+1.078E+04	+3.835E+00	+3.211E+03	+3.364E+03
SM- 149	62149	+4.246E+04	+2.774E+00	+3.386E+03	+3.514E+03
SM- 150	62150	+1.089E+02	+9.589E-01	+3.194E+02	+3.257E+02
SM- 152	62152	+2.072E+02	+1.018E+00	+2.754E+03	+2.768E+03
EU- 153	63153	+3.931E+02	+8.970E-01	+1.387E+03	+1.420E+03
EU- 154	63154	+1.365E+03	+2.367E+00	+1.269E+03	+1.315E+03
EU- 155	63155	+4.071E+03	+1.532E+00	+6.751E+03	+1.848E+04
EU- 156	63156	+4.002E+00	+1.000E+00	+2.564E+02	+2.591E+02

*)Maxwell temperature = 858K. Cross sections for T=858K.
g-factor (<0.625 eV), RI-1(0.625 eV - 5.53 keV), RI-2(0.5eV-20MeV).

Table B.3.2 One-group Neutron Cross Sections for Important FPs (barns)

ISOTOPE	ZA	SIG(PWR)	SIG(BWR)	SIG(FBR)
MO- 98	42098	+2.325E-01	+2.430E-01	+9.813E-02
TC- 99	43099	+9.144E+00	+9.588E+00	+5.537E-01
RU- 102	44102	+2.620E-01	+2.710E-01	+1.493E-01
RU- 103	44103	+3.287E+00	+3.429E+00	+4.315E-01
RH- 105	45105	+1.497E+03	+1.532E+03	+5.785E-01
PD- 107	46107	+3.486E+00	+3.650E+00	+8.576E-01
PD- 108	46108	+7.790E+00	+8.188E+00	+1.975E-01
AG- 109	47109	+4.152E+01	+4.388E+01	+5.912E-01
AG- 110m	47110	+9.756E+00	+1.005E+01	+2.062E+00
I- 129	53129	+3.198E+00	+3.292E+00	+3.378E-01
XE- 135	54135	+2.624E+05	+2.673E+05	+6.126E-02
CS- 133	55133	+1.172E+01	+1,228E+01	+4.089E-01
CS- 134	55134	+1.492E+01	+1.532E+01	+9.648E-01
CS- 135	55135	+2.548E+00	+2.653E+00	+1.851E-01
CS- 136	55136	+2.810E+00	+2.917E+00	+2.034E-01
CS- 137	55137	+3.449E-02	+3.600E-02	+2.069E-02
LA- 139	57139	+1.090E+00	+1,123E+00	+2.841E-01
PR- 141	59141	+1.730E+00	+1.780E+00	+1.149E-01
PM- 147	61147	+6.380E+01	+6.687E+01	+1.022E+00
PM- 148	61148	+2.337E+02	+2.400E+02	+1.600E+00
PM- 148M	61148	+3.077E+03	+3.124E+03	+2.651E+00
SM- 149	62149	+1.005E+04	+1.010E+04	+2.498E+00
SM- 150	62150	+1.710E+01	+1.781E+01	+4.264E-02
SM- 152	62152	+7.793E+01	+8.253E+01	+3.980E-02
EU- 153	63153	+6.872E+01	+7.092E+01	+2.152E+00
EU- 154	63154	+2.744E+02	+2.800E+02	+2.664E+00
EU- 155	63155	+9.810E+02	+1.008E+03	+1.063E+02
EU- 156	63156	+7.764E+00	+8.136E+00	+5.635E-01

*)ORIGEN SPECTRA : PWR(PWRU), BWR(BWRU),FBR(EMOPU/U/U/C)

Appendix C Comparison of Decay Heat Power between JNDC and ORIGEN-2/82

Table C.1 compares the decay heat power in U-235, U-238, Pu-239 and Pu-241 between the present JNDC recommendation and the ORIGEN-2/82 results for the case of four-year irradiation.

Table C.1 Relative Decay Heat P/P0 after A Four-Year Irradiation in Comparison with ORIGEN-2/82 Results

t(SEC)	U-235		PU-239	
	JNDC	ORIGEN/JNDC	JNDC	ORIGEN/JNDC
+1.0E+01	+4.495E-02	+1.011E+00	+3.753E-02	+1.006E+00
+1.0E+02	+2.960E-02	+1.016E+00	+2.564E-02	+1.014E+00
+1.0E+03	+1.819E-02	+1.002E+00	+1.585E-02	+9.942E-01
+1.0E+04	+8.918E-03	+1.000E+00	+7.771E-03	+9.886E-01
+1.0E+05	+4.231E-03	+9.922E-01	+4.045E-03	+9.883E-01
+2.0E+05	+3.459E-03	+9.954E-01	+3.309E-03	+9.937E-01
+4.0E+05	+2.870E-03	+9.990E-01	+2.703E-03	+1.001E+00
+1.0E+06	+2.143E-03	+1.024E+00	+1.977E-03	+1.008E+00
+2.0E+06	+1.609E-03	+1.004E+00	+1.490E-03	+1.012E+00
+4.0E+06	+1.135E-03	+1.005E+00	+1.080E-03	+1.014E+00
+1.0E+07	+6.498E-04	+1.003E+00	+6.758E-04	+1.013E+00
+2.0E+07	+3.619E-04	+1.000E+00	+4.292E-04	+1.012E+00
+4.0E+07	+1.883E-04	+9.996E-01	+2.467E-04	+1.010E+00
+1.0E+08	+7.503E-05	+9.985E-01	+8.285E-05	+1.004E+00
+2.0E+08	+4.732E-05	+9.963E-01	+3.448E-05	+1.005E+00
+4.0E+08	+3.852E-05	+9.973E-01	+2.465E-05	+1.008E+00
+1.0E+09	+2.427E-05	+1.002E+00	+1.553E-05	+1.010E+00

T(SEC)	U-238		PU-241	
	JNDC	ORIGEN/JNDC	JNDC	ORIGEN/JNDC
+1.0E+01	+4.787E-02	+1.008E+00	+4.224E-02	+9.261E-01
+1.0E+02	+2.933E-02	+1.011E+00	+2.667E-02	+9.301E-01
+1.0E+03	+1.715E-02	+9.937E-01	+1.577E-02	+9.043E-01
+1.0E+04	+8.281E-03	+9.874E-01	+7.476E-03	+9.087E-01
+1.0E+05	+4.117E-03	+9.722E-01	+3.978E-03	+9.070E-01
+2.0E+05	+3.371E-03	+9.692E-01	+3.303E-03	+9.088E-01
+4.0E+05	+2.771E-03	+9.659E-01	+2.749E-03	+9.098E-01
+1.0E+06	+2.034E-03	+9.662E-01	+2.064E-03	+9.110E-01
+2.0E+06	+1.521E-03	+9.750E-01	+1.577E-03	+9.177E-01
+4.0E+06	+1.082E-03	+9.919E-01	+1.163E-03	+9.286E-01
+1.0E+07	+6.481E-04	+1.003E+00	+7.724E-04	+9.384E-01
+2.0E+07	+3.917E-04	+1.000E+00	+5.248E-04	+9.428E-01
+4.0E+07	+2.172E-04	+1.003E+00	+3.125E-04	+9.456E-01
+1.0E+08	+7.572E-05	+1.026E+00	+9.856E-05	+9.479E-01
+2.0E+08	+3.655E-05	+1.021E+00	+3.443E-05	+9.636E-01
+4.0E+08	+2.764E-05	+9.842E-01	+2.304E-05	+9.729E-01
+1.0E+09	+1.740E-05	+9.728E-01	+1.453E-05	+9.738E-01

note) Total energy release per fission given in Appendix B.2 is used both for the JNDC and the ORIGEN cases in common.

Appendix D Basic Programs DECANT, CAPCORR and Related Data Sets

As an attachment to the present report, two BASIC computer programs have been developed. One is DECANT which calculates the decay heat curves for practical applications on the basis of the fitting-formulae proposed in Sec.3.2, and another is CAPCORR which supplements the former by generating the correction factors for the neutron capture effect according to Sec.3.3. A floppy diskette is available from Dr. J. Katakura, JAERI, which contains the source program of the codes DECANT and CAPCORR as well as the following data sets pertinent to the present report.

D.1 Data Set Names and their Contents

(0) DH-ANNEX.JXW ---	Original documents of Appendix A - D in Japanese
(1) J33COEFF.DAT ---	Parameters in the 33-term exponential fitting formulae for the β- and the γ-ray components of the decay heat power after a burst fission in U-235, U-238, Pu-239, Pu-240 and Pu-241 (see Table 4.16)
(2) ANSCOEFF.DAT ---	Parameters in the ANS5.1 fitting formulae (1979) for the total (β+γ) decay heat power after a burst fission in U-235, U-238 and Pu-239
(3) JHEAT***.DAT --- (for ***, see a - d below)	Numerical tables of the β- and the γ-ray components of the decay heat power in U-235, U-238, Pu-239, Pu-240 and Pu-241 calculated at 59 time-points in the cooling time range of $1.0\text{-}1.0\times10^{13}$ seconds. Inclusive of four different irradiation periods in four separate files; instantaneous, 1-year, infinite and 1.0×10^{13} seconds. Note): The decay heat power after "true" infinite irradiation is greater than that after 1.0×10^{13} sec irradiation usually used as "infinite irradiation" at long cooling time

region. The differences between them are 2, 4, 8, 15 and 170% at 10^6, 10^7, 10^8, 10^9 and 10^{10} sec cooling-times, respectively.

a. JHEATBST.DAT --- f(t): Total decay heat power after a burst irradiation

b. JHEAT1Y.DAT --- F(T=1-year,t): Total decay heat power after a 1-year irradiation

c. JHEATINF.DAT --- F(T=∞,t): Total decay heat power after an infinite irradiation

d. JHEAT13.DAT --- F(T=1.0×10^{13}s,t): Total decay heat power after a 1.0×10^{13} second irradiation

(4) DHINFERR.DAT --- Uncertainty (1-σ,%) of the total decay heat power after an infinite irradiation

(5) FPGAMMA.DAT --- Normalized FP gamma-ray spectra in U-235, U-238, and Pu-239 calculated in the cooling-time range of 0 - 1.0×10^{13} second after an infinite irradiation

(6) GT***.DAT --- $G_i^{\alpha}(\phi,T,t)$: Correction factors for the neutron capture effect in FPs to be applied to both β- and γ-ray components of the decay heat power (Here the superscript α stands for β or γ).

(for ***, see a - f below)

The data are given for the following parameter values of the neutron flux ϕ, the irradiation time T and the cooling time t in separate neutron spectra ψ and fissioning nuclide i, a through f.

Flux ϕ: 0 - 1×10^{15} for PWR thermal flux, r=0.21
0 - 5.2×10^{14} for BWR thermal flux, r=0.14
0 - 5×10^{15} for FBR total flux
(all in n/sec/cm^2)
Irradiation time T: burst, 1, 3, 5 years

Cooling time t: 0 - 10^{13} s

a. GTPWRU5.DAT --- PWR, U-235
b. GTPWRP9.DAT --- PWR, Pu-239
c. GTBWRU5.DAT --- BWR, U-235
d. GTBWRP9.DAT --- BWR, Pu-239
e. GTFBRU5.DAT --- FBR, U-235
f. GTFBRP9.DAT --- FBR, Pu-239

(7) ANSWER.DAT --- Output data file generated by the "DECANT.BAS" BASIC program given below. The problem treated here is the sample problem described in Appendix A.

(8) PWRCAPT.DAT --- Output data file generated by the "CAPTCORR.BAS" BASIC program given below. The problem treated here is the decay-heat in a PWR after a 4-years operation.

(9) DECANT.BAS --- A BASIC program for calculating the reactor decay heat power after a series of constant power operations. The operation history should be input on the key board as well as the fissile-wise contributions to the reactor operating power, and a ratio of the U-238 capture to the total fission. The ratio is used to calculate the decay heat contribution from the actinide chain U-239→Np-239→Pu-239. The code finally outputs the decay heat power and its uncertainty at the cooling times from 1 to 1.0×10^9 seconds on the screen, and on the data file "ANSWER.DAT" (see (7) above) as well.

(10)CAPTCORR.BAS --- A BASIC program for calculating the correction factors for the neutron capture effect in FPs. The code interpolates the data "GT***.DAT" given in (6) above depending on the reactor type (PWR/

BWR/FBR), the fissile (U-235/Pu-239), the neutron flux ϕ and the irradiation time T. The results will be written on the data file like an example "PWRCAPT.DAT" given in (8).

D.2 How to Use the BASIC Programs

The BASIC programs DECANT and CAPTCORR described above are written on N88BASIC. The user is advised to make some necessary modifications, which are assumed to be minimal, in order that the programs are workable on his own version of the BASIC system.

The program DECANT will read the data files of J33COEF.DAT, JHEATINF.DAT and DHINFERR.DAT, which are prepared in the diskette introduced in Sect. D.1. along with the programs themselves. Another program CAPTCORR will make use of the files of J33COEF, JHEATINF and GT***.DAT.

D.2.1 DECANT.BAS

This BASIC program calculates the dec ay heat power of FP and contribution from the actinide chain U-239→Np-239→Pu-239. The input data should be entered from a keyboard, which includes the operation history, the fissile-wise contributions to the reactor operating power, and the U-238 capture ratio to the total fission number. The actinide decay heat is calculated only for the final cycle of the reactor burnup. This can be justified because the U-239→Np-239→Pu-239 part of the actinide decay heat dies out after a few days after the last neutron-capture event in U-238. Evaluation of the neutron capture effect in FPs is left to another program CAPTCORR. As for the FP decay heat the program outputs the fissile-wise contributions and the uncertainties as well.

The calculation will be carried out as follows. The step-wise operating power history illustrated in Fig.D.1 is assumed.
Time t after the reactor shutdown at $T_{k_{max}}$, the decay heat is written, for a single fissile for simplicity, as

$$R(t)=\sum_k P_k/Q_t[\sum_i^{33}(\alpha i/\lambda i)\exp(-\lambda_i t)\{\exp(-\lambda_i(T-T_k))-\exp(-\lambda_i(T-T_k))\}] \quad \text{(D.1a)}$$

or

$$R(t)=\sum_k P_k/Q_t[F(t+T-T_k)-F(t+T-T_{k-1})], \quad \text{(D.1b)}$$

where Q_t stands for the recoverable energy released at one fission event, and α_i and λ_i are the parameters in the 33-term exponential fitting formula,

$$f(t) = \sum_{i}^{33} \alpha_i \exp(-\lambda_i t),$$

for the decay heat power after an instantaneous irradiation as defined and tabulated in Table 4.16. In the latter expression (D.1b), F(t) stands for the decay heat after an infinite irradiation. Although the computation based on the expression (D.1b) saves a lot of time, it introduces numerical error when the first and the second terms in the right-hand side have almost the same values as each other. In such case, the program uses the expression (D.1a) to avoid the error.

The program calculates the uncertainty from the expression (D.1b) as,

$$|\Delta R(t)| = \sum_{k} (P_k/Q_t)[\,|\Delta F(t+T-T_k)| - |\Delta F(t+T-T_{k-1})|\,] \qquad (D.2)$$

assuming a full correlation for the infinite-irradiation functions:

$$<\Delta F(t_1)\Delta F(t_2)> = |\Delta F(t_1)|\cdot|\Delta F(t_2)|.$$

When the relative error is a constant, $|\Delta F(t)/F(t)| = \sigma$, then we will get $|\Delta R(t)/R(t)| = \sigma$. When we are concerned in a long cooling-time range, the assumption of the full correlation can be applied well, because the decay heat is dominated by only a few number of decaying nuclides. When the cooling-time is shorter than the irradiation-time, the expression (D.2) is still valid, because the second term in the parentheses [ ] in (D.1b) can be neglected in comparison with the first term.

When a plural number of fissile nuclides contribute to the decay heat, it is assumed that their errors correlate strongly each other. This leads to summing up the expression (D.2) over fissile nuclides to get the total error,

$$|\Delta R(t)/R(t)| = \sum_{i} |\Delta R_i(t)/R_i(t)|\cdot R_i(t)/R(t), \qquad (D.3)$$

as is adopted in the DECANT program. When no correlation is assumed among the fissile nuclides, the resultant uncertainty becomes fairly smaller than that given from the expression (D.3).

D.2.2 CAPTCORR.BAS

This program reads the $G^{\alpha}(\phi_i,T_j,t)$ table ($\alpha=\beta$ or γ) stored in the data files, such as GTPWRU5.DAT, depending on the reactor type and the fissile nuclide, and then interpolates the data to get the correction factor for the neutron-capture effect $G^{\alpha}(\phi,T,t)$ for a given flux ϕ and a given irradiation time T. The interpolation is carried out in a two-dimensional plane by transforming a rectangular, which has four vertexes (ϕ_n,T_n, n=1,4) and holds the point (ϕ,T) in it, into a square $\xi_n=\pm1$, $\eta_n=\pm1$ as shown in Fig. D.2. In practice, the interpolation is done with the aid of "shape functions" F_n as

$$G(\eta,\xi;t)=\sum_{n}^{4}G(\eta_n,\xi_n;t)F_n, \qquad (D.4)$$

with

$$F_1=(1-\eta)(1-\xi)/4, \quad F_2=(1-\eta)(1+\xi)/4$$
$$F_3=(1+\eta)(1+\xi)/4, \quad F_4=(1+\eta)(1-\xi)/4,$$

and

$$\eta=[2/\log(\phi_2/\phi_1)][\log\phi-\log(\phi_2\phi_1)/2]$$
$$\xi=[2/\log(T_2/T_1)][\log T-\log(T_2T_1)/2].$$

The corrected decay heat power for a constant fission rate F and a neutron flux ϕ is, then,

$$P(F,T,\phi;t)=P(F,T,0;t)G(\phi,T;t). \qquad (D.5)$$

For a more complicated operation history, such as Fig. D.1 for example, we approximate the corrected decay heat, by introducing an effective constant values F_{eff} for the fission rate, ϕ_{eff} for the flux, and T_{eff} for the irradiation time, as

$$\Delta P=P(F_{eff},T_{eff},0;t)[G(\phi_{eff},T_{eff};t)-1],$$

and then, further,

$$P(F^*,T^*,f^*;t)=P(F^*,T^*,0;t)+P(F_{eff},T_{eff},0;t)[G(\phi_{eff},T_{eff};t)-1], \qquad (D.6)$$

where the superscript * indicate that the parameter represents a complex operation history. The first term in the right-hand side and $P(F_{eff},T_{eff},0;t)$ can be calculated by the program DECANT.BAS. The effective values for F, T and ϕ are given as

$$T_{eff} = \Sigma\ \Delta T_k, \quad \text{(sum over k for } \phi_k \neq 0) \tag{D.7a}$$

$$\phi_{eff} = \Sigma\ \phi_k \Delta T_k / T_{eff}, \tag{D.7b}$$

$$F_{eff} = \Sigma (P_k / Q_f) \Delta T_k / T_{eff}. \tag{D.7c}$$

When several fissiles contribute to the decay heat, F_{eff} should be calculated for each fissile nuclide. For U-238, Pu-240 and Pu-241, we use the $G(\phi,T;t)$ data for Pu-239.

The program CAPTCORR.BAS outputs $P(F,T,0;t)$ and $G(\phi,T;t)$ for β, γ and $\beta+\gamma$ onto a printer and/or an output-file disk. The corrected decay heat can be calculated based on this output-file combined with the output-file from the DECANT code.

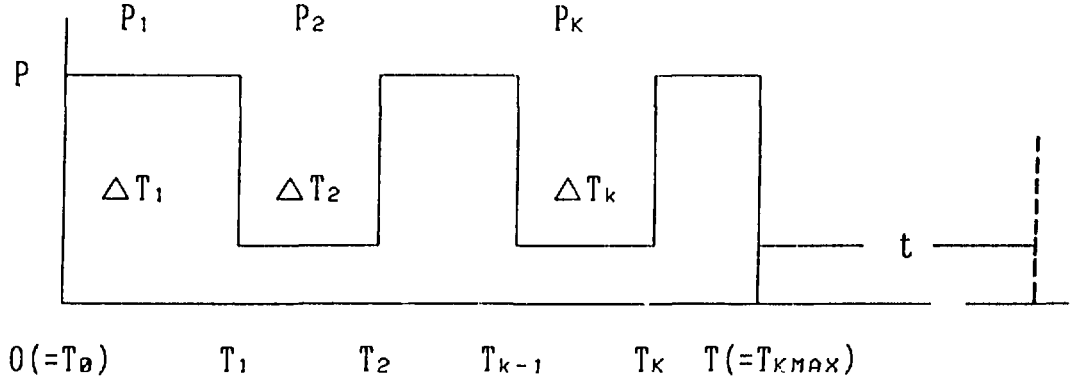


Fig. D.1 Step-wise Irradiation Power History; An Example

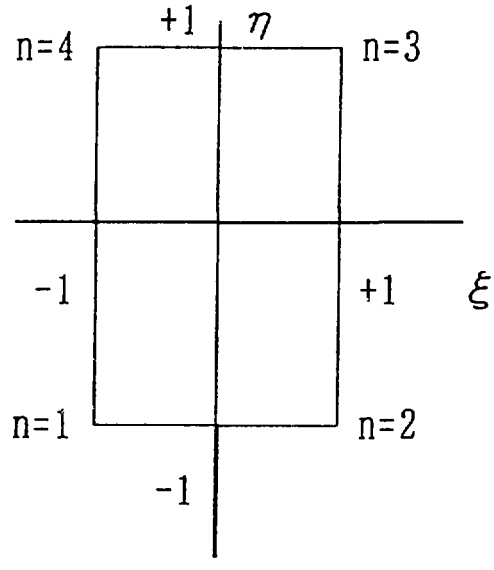


Fig. D.2 Two-Dimensional Plane for Data Interpolation

国際単位系 (SI) と換算表

表 1　SI 基本単位および補助単位

量	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd
平面角	ラジアン	rad
立体角	ステラジアン	sr

表 2　SI と併用される単位

名称	記号
分、時、日	min, h, d
度、分、秒	°, ′, ″
リットル	l, L
トン	t
電子ボルト	eV
原子質量単位	u

1 eV = 1.60218 × 10^{-19} J

1 u = 1.66054 × 10^{-27} kg

表 3　固有の名称をもつ SI 組立単位

量	名称	記号	他の SI 単位による表現
周波数	ヘルツ	Hz	s^{-1}
力	ニュートン	N	m·kg/s^2
圧力、応力	パスカル	Pa	N/m^2
エネルギー、仕事、熱量	ジュール	J	N·m
仕事率、放射束	ワット	W	J/s
電気量、電荷	クーロン	C	A·s
電位、電圧、起電力	ボルト	V	W/A
静電容量	ファラド	F	C/V
電気抵抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁束	ウェーバ	Wb	V·s
磁束密度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	°C	
光束	ルーメン	lm	cd·sr
照度	ルクス	lx	lm/m^2
放射能	ベクレル	Bq	s^{-1}
吸収線量	グレイ	Gy	J/kg
線量当量	シーベルト	Sv	J/kg

表 4　SI と組み合わせて用いられる非 SI 単位

名称	記号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

1 Å = 0.1 nm = 10^{-10} m

1 b = 100 fm² = 10^{-28} m²

1 bar = 0.1 MPa = 10^5 Pa

1 Gal = 1 cm/s² = 10^{-2} m/s²

1 Ci = 3.7 × 10^{10} Bq

1 R = 2.58 × 10^{-4} C/kg

1 rad = 1 cGy = 10^{-2} Gy

1 rem = 1 cSv = 10^{-2} Sv

表 5　SI 接頭語

倍数	接頭語	記号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

注

1　表 1～5 は「国際単位系」第 5 版（国際度量衡局 1985 年刊）による。ただし、1 eV および 1 u の値は CODATA の 1986 年推奨値による。

2　表 4 には慣用されているものを含めたが、その他にも多くの単位があるので、単位名の一覧表としては網羅的ではない。

3　bar は、JIS では圧力の単位として用いる表示は認められていないが、ISO では使用してもよいとしている。

4　EC 関係の単位では、bar、barn および mmHg 等の使用は、原則として認められない。

換算表

力	N	10^5 dyn	kgf	lbf
	1	10^5	0.101972	0.224809
	9.80665	9.80665×10^5	1	2.20462
	4.44822	4.44822×10^5	0.453592	1

粘度　1 Pa·s = 1 N·s/m² = 10^5 P（ポアズ）　1 g/(cm·s)

動粘度　1 m²/s = 10^4 St（ストークス）　1 cm²/s

圧力	MPa	10^{-5} bar	kgf/cm²	atm	mmHg（Torr）	lbf/in²（psi）
	1	10^{-5}	10.1972	9.86923	7.50062×10^{3}	145.038
力	0.0980665	1	1	0.967841	735.559	14.2233
	0.101325	1.03323	1.03323	1	760	14.6959
	1.33322×10^{-4}	1.33322×10^{-3}	1.35951×10^{-3}	1.31579×10^{-3}	1	1.93368×10^{-2}
	6.89476×10^{-3}	6.89476×10^{-2}	7.03070×10^{-2}	6.80460×10^{-2}	51.7149	1

エネルギー・仕事・熱量	J	10^7 erg	kgf·m	kW·h	cal（計量法）	Btu	ft·lbf	eV
	1	10^7	0.101972	2.77778×10^{-7}	0.238889	9.47813×10^{-4}	0.737562	6.24150×10^{18}
	9.80665	9.80665×10^7	1	2.72407×10^{-6}	2.34270	9.29487×10^{-3}	7.23301	6.12082×10^{19}
	3.6×10^6	3.67098×10^{13}	1	8.59999×10^{5}	3412.13	2.65522×10^{6}	2.21694×10^{25}	
	4.18605	4.18605×10^7	0.426858	1.16279×10^{-6}	1	3.96759×10^{-3}	3.08747	2.61272×10^{19}
	1055.06	10^{10}	107.586	2.93072×10^{-4}	251.996	1	778.172	6.58515×10^{21}
	1.35582	1.35582×10^7	0.138255	3.76616×10^{-7}	0.323890	1.28506×10^{-3}	1	8.46233×10^{18}
	1.60218×10^{-19}	1.60218×10^{-12}	1.63377×10^{-20}	4.45050×10^{-26}	3.82743×10^{-20}	1.51857×10^{-22}	1.18171×10^{-19}	1

1 cal = 4.18605 J（計量法）
= 4.187 J（熱化学）
= 4.1855 J（15 °C）
= 4.1868 J（国際蒸気表）

仕事率　1 PS（仏馬力）
= 75 kgf·m/s
= 735.499 W

放射能	Bq	Ci
	1	2.70270×10^{-11}
	3.7×10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58×10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

（86 年 12 月 26 日現在）