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**DECAY HEAT
FOR THE
FAST TEST REACTOR (FTR)**

Hanford Engineering Development Laboratory

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Hanford Engineering Development Laboratory

F. Schmittroth
May 1978

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DECAY HEAT FOR THE FAST TEST REACTOR (FTR)

F. Schmittroth

ABSTRACT

The decay heat for the Fast Test Reactor (FTR) is reevaluated for cooling times up to 30 years. Decay heat from both fission-products and actinides is included. Fission-product decay data used in this evaluation are taken from ENDF/B version IV. Consideration is given to the effects of fuel cycles and loadings and to the importance of fissions in the minor fuels such as ^{235}U , ^{240}Pu , and ^{241}Pu , as well as in the major fuels, ^{238}U and ^{239}Pu . Decay-heat uncertainties are evaluated and are typically less than 5% for cooling times beyond 50 seconds. The effective energy release per fission for FTR is also reevaluated.

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DECAY HEAT FOR THE FAST TEST REACTOR (FTR)

F. Schmittroth

I. INTRODUCTION

This report presents an updated evaluation of the decay heat for the Fast Test Reactor (FTR) central to the Fast Flux Test Facility (FFTF) at Hanford. Because of the need to maintain adequate cooling to dissipate decay heat in both normal and emergency^(1,2) shutdowns, decay-heat studies have always been an integral part of reactor design for fast reactors as well as thermal reactors. The present review is motivated by several reasons. First of all, decay heat is often a constraining parameter,^(1,2) not only in reactor design and operation, but in other phases of the fuel cycle as well, including waste management and spent fuel shipping and storage. Secondly, significant improvements in decay heat values are now available due to a broad spectrum of decay-heat research for both thermal and fast reactors. Summation calculations^(3,4) used for the previous FTR decay-heat evaluation⁽⁵⁾ are better as a consequence of an improved nuclear data base. Specifically, the Evaluated Nuclear Data File, ENDF/B, Version IV,⁽⁶⁾ recently released, represents a nationwide effort to provide the latest available data. Also, sensitivity studies^(7,8,9,10) have provided more rigorous assessments of the validity of these calculations. And on the experimental side, new measurements both in the U.S.^(11,12,13) and abroad,⁽¹⁴⁾ combine with calculations to provide added assurance in the final results, especially for the very short cooling times (less than 1000 sec) important in emergency shutdowns. Finally, due to the prototypic aspect of the FFTF, this work is applicable to the LMFBR program, in general.

The improved precision now attainable requires closer attention to details that were previously ignored or given rather cursory treatment. Along with fission-product heating, the decay of ^{239}U and ^{239}Np is reexamined along with a study of other actinides such as ^{242}Cm . Other considerations encompass a more detailed study of fuel cycles and fuel inventories including ^{235}U , ^{240}Pu , and ^{241}Pu . And finally, the energy released per fission is

reviewed. This parameter is important, not only as a conversion factor needed to express decay heat relative to the operating power, but for all applications where a given operating power must be converted to the corresponding fission rate.

One major limitation of this report needs to be emphasized. For long cooling times (beyond several days), decay heat from structural activation may need to be estimated for some applications. However, this component of decay heat is beyond the scope of the present study where we have restricted our attention to those components directly associated with the fuel.

II. FTR PARAMETERS

FTR parameters such as fission rates, fluxes, and fuel inventories, are required in decay-heat analyses for two related reasons. First of all, decay heat values are more easily evaluated in terms of indirect parameters like the fission rate rather than more direct parameters like operating power. Secondly, a detailed understanding of these underlying parameters aids a broader application of our results. Some studies require values averaged over the entire core, but other applications demand a more detailed picture. Before turning to the specific problem of decay heat, we review FTR parameters in this section.

A. Reaction Rates

Although detailed space- and energy-dependent fluxes are available for the FTR, a simpler description is appropriate here. It is undesirable to redo large reactor calculations to study the consequences of variations in fuel loading, for example.

In general, reaction rates can be computed for any material, i , and cross section, σ , by averaging all space and energy values:

$$\langle \phi \sigma \rho_i \rangle = \int dE \int d\vec{r} \left\{ \phi_E(\vec{r}, E) \sigma(E) \rho_i(\vec{r}, t) \right\}, \quad (1)$$

where, $\rho_i(\vec{r}, t)$ denotes the concentration of the i -th material at time, t . The remaining notation should be clear. Also, the flux may be slightly time-dependent.

There are only two enrichments^(15,16) for the drivers (inner and outer) in FTR. Within these two zones, material densities are constant except for experimental assemblies) so that it is appropriate to compute average fluxes for each zone, α :

$$\phi_{\alpha i}(E) = \frac{1}{n_{\alpha i}} \int_{V_{\alpha}} \left(\phi_E \rho_i \right) d\vec{r} \quad (2)$$

where the integration is over the volume, V_{α} , in zone, α . The total inventory of material, i , in zone, α , is

$$n_{\alpha i} = \int_{V_{\alpha}} \rho_i d\vec{r}. \quad (3)$$

Because ρ_i is constant over the zone, the flux $\phi_{\alpha i}$, is independent of the material and is given by

$$\phi_{\alpha}(E) = \frac{1}{V_{\alpha}} \int \phi_E(\vec{r}, E) d\vec{r}. \quad (4)$$

Similarly, one can define one-group cross sections for each zone:

$$\sigma_{\alpha} = \frac{1}{\phi_{\alpha}} \int \phi_{\alpha}(E) \sigma(E) dE \quad (5)$$

where, ϕ_{α} is the total flux,

$$\phi_{\alpha} = \int \phi_{\alpha}(E) dE. \quad (6)$$

The core-averaged reaction rate can now be expressed in terms of zone fluxes and one-group cross sections:

$$\langle \phi \sigma_{fi} \rangle = \sum_{\alpha=1}^2 \left(\phi_{\alpha} \sigma_{\alpha} n_{\alpha i} \right) . \quad (7)$$

As an application of this result, we consider the total FTR fission rate given by

$$R_f = \sum_i \langle \phi \sigma_{fi} \rangle , \quad (8)$$

where the sum is over all fissionable nuclides. Equation (8) can be reexpressed as

$$R_f = \sum_{\alpha} \phi_{\alpha} \sum_i \left(\sigma_{f,\alpha i} n_{\alpha i} \right) \quad (9)$$

where, $\sigma_{f,\alpha i}$ is the microscopic fission cross section for nuclide, i , in zone, α , and $n_{\alpha i}$ is the corresponding inventory. To obtain one-group cross sections, Equation (5) was used to collapse effective 42-group resonance shelf-shielded cross sections generated by the one-dimensional diffusion-theory code, 1DX.⁽¹⁷⁾ The required 42-group fluxes were obtained along with the cross sections for both the inner and outer driver zones and are depicted in Figure 1. Input to 1DX consisted of the multigroup cross section library, FTR Set 300-S,⁽¹⁸⁾ generated with the ETOX⁽¹⁹⁾ processing code. Values for the fission cross sections are given in Table I along with the ^{238}U capture cross section. Note that spectral changes in the neutron flux are relatively unimportant, as indicated by the small variations in zone cross sections. Variations in the magnitude of the flux between the inner and outer drivers are much larger, of course. The ratio of the outer driver flux to the inner driver flux obtained from 1DX is 0.702, in reasonable agreement with a value of 0.670 obtained from a complete three-dimensional calculation⁽²⁰⁾ with the diffusion code, 3DB.⁽²¹⁾ Along with the cross sections discussed previously, Table I presents the fissionable inventories for the FTR as obtained from the basic design⁽¹⁵⁾ for Cores 1 and 2. Except for an overall flux normalization, Equation (9) can now be used to compute the total fission

FIGURE 1. FTR Fluxes for Inner and Outer Drivers.

TABLE I

ONE-GROUP CROSS SECTIONS AND FISSIONABLE INVENTORIES FOR THE FTR
(CORES 1 AND 2) FOR BOTH INNER (I.D.) AND OUTER (O.D.) DRIVER ZONES

Nuclide	Inventories, kg		σ_f , barns		σ_γ , barns		$n\sigma_f$, atoms-cm ²	
	I.D. Zone	O.D. Zone	I.D. Zone	O.D. Zone	I.D. Zone	O.D. Zone	I.D. Zone	O.D. Zone
²³⁵ U	4.78	7.64	1.840*	-	-	-	22.5*	36.0
²³⁸ U	687	1098	0.0515	0.0541	0.272	0.270	89.56	150.2
²³⁹ Pu	173	360	1.816	1.815	-	-	791	1645
²⁴⁰ Pu	23.5	48.9	0.427*	-	-	-	25.2*	52.4*
²⁴¹ Pu	3.37	7.00	2.39*	-	-	-	20.1*	41.8*
$\Sigma(n\sigma_f)$							949	1926

*For ²³⁵U, ²⁴⁰Pu, and ²⁴¹Pu, only infinitely dilute values of σ_f computed at the FTR core center are given.

rate. In order to obtain the flux normalization for a given reactor power level, one must determine the effective energy release per fission, a problem we consider in the next section.

B. Energy Released per Fission, E_f

The energy released per fission, E_f , is the conversion parameter needed to relate operating power, P_o , to fission rate, R_f :

$$P_o = E_f R_f. \quad (10)$$

Since there are several fissionable nuclides, and since some of the energy associated with fission in a reactor actually comes from neutron capture, E_f is an effective reactor-dependent number. Two recent comprehensive studies of E_f have been made, one by James⁽²²⁾ and another by Unik and Gindler.⁽²³⁾ An earlier evaluation was also made by Bunch⁽²⁴⁾ for the FTR. In each case, the total energy, E_f , was found by adding the various components such as fragment kinetic energy and beta and gamma energy. Unik and Gindler concluded, along with James, that a more accurate value of the total energy is obtained by considering the mass difference of the initial fissioning nucleus and the final stable decay products. In addition, this method bypasses a troublesome discrepancy in the experimental measurements^(22,23) of the fragment kinetic energy (nevertheless, new measurements would be desirable to resolve the problem).

With some modifications, we use the mass-difference technique to recompute the effective fission energy, Q_{eff} , for several important nuclides. In Table II, we first list the mass differences, $\Delta M_t - \Delta M_{f.p.}$, between the target nucleus and the stable fission products, as summarized in Refs. 22 and 23 (for the moment, we ignore the mass carried away by neutrons). Since the mass difference includes the neutrino energy, $\bar{E}_\nu$, that arises from fission-product decay, one must subtract $\bar{E}_\nu$ to obtain the true deposited energy. The same summation calculations used to obtain the fission-product decay heat can be used to evaluate the neutrino energies. Based on ENDF/B-IV, we

TABLE II
EFFECTIVE ENERGY RELEASED PER FISSION

∞	Nuclide	$\Delta M_t - \Delta M_{f.p.}$	$\bar{E}_v$	$\bar{\nu}$	$Q_c = (\bar{\nu} - 1) q_c$	$(\bar{\nu} - 1) \Delta M_n$	Q_T (1 yr)
	$^{235}\text{U(T)}$	214.23 ± 0.20	9.1 ± 1.4	2.42 ± 0.01	-	11.46 ± 0.08	-
	$^{235}\text{U(F)}$	214.19 ± 0.20	9.1 ± 1.4	2.46 ± 0.03	10.42 ± 0.76	11.79 ± 0.24	203.5 ± 1.6
	$^{238}\text{U(F)}$	220.50 ± 0.20	11.4 ± 1.7	2.66 ± 0.05	11.85 ± 0.90	13.34 ± 0.40	207.3 ± 1.9
	$^{239}\text{Pu(F)}$	222.11 ± 0.20	7.6 ± 1.1	2.92 ± 0.05	13.71 ± 1.02	15.50 ± 0.40	212.5 ± 1.5

calculate the values shown in Table II. Although slightly below those in James' evaluation and substantially below (1-3 MeV) those found by Unik and Gindler, the values obtained from ENDF/B should represent the best evaluation and are used here.

Part of the sensible fission energy, namely that due to neutron capture, is not included in the mass difference and must be added. Because this energy is material dependent, it must be reevaluated for each reactor. For the FTR, roughly 0.5 neutrons/fission are captured in ^{238}U . Another 0.2 are captured in plutonium, mainly ^{239}Pu , while about 1 neutron/fission is captured in structural materials. Using values⁽²²⁾ of 5.69 and 6.53 MeV for the capture Q-values of ^{238}U and ^{239}Pu , and a value of 7.99 MeV obtained from Ref. 23 for stainless steel, we find a value of 7.14 ± 0.5 MeV for the energy released per capture, q_c . Mainly because of the relatively large number of neutrons lost to the radial and axial reflectors in the FTR, this value is larger than for a typical thermal reactor⁽²²⁾ (6.1 MeV) or for a reactor such as EBR-II⁽²³⁾ (6.5 MeV). The lack of a blanket in the FTR allows a relatively large number of neutrons to be captured in stainless steel with its high energy release.

The total capture energy, Q_c , is now easily computed by multiplying the energy released per capture, q_c , by the number, $(\bar{\nu}-1)$, of captures per fission. Values for the number of neutrons released per fission, $\bar{\nu}$, are given in Table II and are used to find values for Q_c , also shown in Table II.

The mass excess carried away by the fission neutrons must also be subtracted from the previous results and is equal to $(\bar{\nu}-1)\Delta M_n$ where ΔM_n is the neutron mass excess. Not only is this term and the neutron capture term proportional to $(\bar{\nu}-1)$, the two terms are nearly equal and opposite in sign (see Table II). This high degree of cancellation, as shown by the last term of

$$Q_T(\infty) = (\Delta M_t - \Delta M_{f.p.}) - \bar{E}_v + (\bar{\nu}-1)(q_c - \Delta M_n) \quad (11)$$

for the total fission energy, makes Q_T very insensitive to $\bar{\nu}$.

A final small correction is needed to obtain the total energy per fission, Q_T , for a finite operating period. Because the decay power does not reach equilibrium during a normal reactor operating history, we subtract an amount ΔE_d to account for a slight overestimate of the decay energy as given by the mass differences. For a one-year irradiation, ΔE_d is 0.23 MeV⁽²³⁾, and the corresponding values of Q_T (1 yr.) are given in the last column of Table II.

To obtain an overall value for the fission energy, E_f , in FTR, we first note that the value of Q_T for ^{240}Pu is slightly larger than for ^{239}Pu while the value for ^{241}Pu is about 2-3 MeV larger.^(22,23) These larger values compensate somewhat the smaller value for ^{235}U . Together with the relatively few number of fissions in ^{235}U , ^{240}Pu , and ^{241}Pu , we are thus justified in weighting the ^{238}U and ^{239}Pu values with their respective fission rates (see Table I) to obtain a final average of 212.0 ± 1.5 MeV for the energy released per fission, E_f , in the FTR. As shown in Table I, we note that the uncertainty in E_f arises mainly from the uncertainties in the neutrino energy, $\bar{E}_\nu$, and in the neutron capture energy, Q_c . Variations in the fuel inventories during reactor operations give some additional uncertainty in E_f . Nevertheless, because of the small differences in Q_T for the various fissionable nuclides, this error is small.

C. Absolute Fission Rate and Neutron Flux

The total fission rate in the FTR is now trivially obtained by dividing the reactor power by the energy per fission, E_f , [Equation (10)]. At a power of 400 MW, the corresponding fission rate is given by

$$R_f = 1.178 \times 10^{19} \text{ fissions/sec.} \quad (12)$$

From Equation (9), and with values taken from Table I, we can also express R_f in terms of the inner driver (i.d.) and outer driver (o.d.) fluxes:

$$R_f = (949)\phi_{i.d.} + (1926)\phi_{o.d.} \quad (13)$$

Then, with a value of 0.670 (see the previous section) for the ratio of the outer driver flux to the inner driver flux, one can easily compute absolute values for the fluxes:

$$\phi_{i.d.} = 5.26 \times 10^{15} \text{ n/cm}^2\text{-S}, \quad (14a)$$

$$\phi_{o.d.} = 3.52 \times 10^{15} \text{ n/cm}^2\text{-S}. \quad (14b)$$

These flux values will vary somewhat, of course, with different fuel inventories.

Because values for the cross sections differ so little between the inner and outer driver zones, it is of interest to compute fluxes averaged over the entire core. Unlike the zone fluxes, fluxes averaged over the entire core are material dependent. For example, using values from Table I, we find the following core-averaged fluxes for ^{238}U and ^{239}Pu :

$$\phi(^{238}\text{U}) = 4.19 \times 10^{15} \text{ n/cm}^2\text{-S}, \quad (15a)$$

$$\phi(^{239}\text{Pu}) = 4.08 \times 10^{15} \text{ n/cm}^2\text{-S}. \quad (15b)$$

Even though the outer drivers are substantially enriched in ^{239}Pu , the average flux seen by ^{239}Pu differs little from the ^{238}U value. These average values may be used to obtain the core-wide reaction rate for any material whose concentrations are proportional to those for ^{238}U and ^{239}Pu .

III. FISSION-PRODUCT DECAY HEAT

A. ^{239}Pu and ^{238}U Values

The major component of FTR decay heat is due to fission products. Since most of the fissions (~85%) are in ^{239}Pu , this nuclide is also the most

important for decay heat. The next major contributor is ^{238}U which accounts for about 8% of the fissions with the remainder accounted for by ^{235}U , ^{240}Pu , and ^{241}Pu .

Because of their importance, typical decay heat curves are shown in Figure 2 for the fast fission of ^{238}U and ^{239}Pu . The values shown are for a 10^7 sec (116 days) exposure and are expressed relative to a unit fission rate for each nuclide. Except for a known error in the ^{98}Zr branching ratio which was remedied, these curves are based strictly on summation calculations with ENDF/B version IV⁶ as the data base. Corresponding uncertainties at the one-sigma (one standard deviation) confidence level are given in Figure 3. The uncertainties stem from detailed sensitivity analyses^(7,8) of the summation method and are further supported by recent decay heat experiments on ^{235}U . (11-14)

B. Fission-Product Decay-Heat Calculations

In order to obtain a final value for the fission-product decay heat, one must account for neutron capture, burnup, fuel cycling, and fuel inventories. The summation code, RIBD,^(3,4) is well suited to compute these effects, which are discussed in the context of RIBD input parameters.

Two fissionable species are allowed in RIBD which requires the specification of the ratio of the fission rates in the two fuels. As with earlier calculations, ^{239}Pu is taken as fuel one while ^{238}U is taken as fuel two. However, although they individually contribute in a minor way to the overall fission rate, ^{235}U , ^{240}Pu , and ^{241}Pu are nearly as important in total as ^{238}U whose fission rate is close to one-tenth of the fission rate for ^{239}Pu . Rather than obtaining detailed decay-heat functions for each of the minor nuclides, we show in the Appendix how to estimate in a semi-quantitative fashion the way they differ from ^{238}U and ^{239}Pu . Specifically, results in the Appendix show that an average value of the decay heat for the fast fission of ^{235}U , ^{240}Pu , and ^{241}Pu is close to the thermal fission of ^{235}U for short cooling times. Because this latter value is roughly 4 MeV/fission below the

FIGURE 2. Fission-Product Decay Heat for ^{239}Pu and ^{238}U (Exposure 1×10^7 Seconds).

FIGURE 3. Fission-Product Decay Heat Uncertainties (Exposure 1×10^7 Seconds; 1-Sigma Confidence Level).

value for ^{238}U and 2 MeV/fission above the value for ^{239}Pu , we allocate two-thirds of the fissions in the three minor nuclides to ^{239}Pu and one-third to ^{238}U for the purpose of calculating the fission-product decay heat. Since the relative fission rates in ^{238}U and ^{239}Pu are in the ratio of 1 to 10, we choose an effective fission rate ratio for ^{238}U to ^{239}Pu given by

$$\frac{R_f^{\text{eff}}(^{238}\text{U})}{R_f^{\text{eff}}(^{239}\text{Pu})} = \frac{1 + 1/3(0.82)}{10 + 2/3(0.82)} = 0.121 ,$$

where 0.82 is the number of fissions in the minor nuclides relative to ^{238}U . At longer cooling times, the differences in decay heat for different nuclides are typically smaller (see Figure 2). Also, the total decay heat is rather insensitive to this ratio, since the decay-heat functions for the different nuclides are very similar.

Another RIBD parameter, burnup, also relates to the fission rate ratio between ^{238}U and ^{239}Pu . During a typical operating cycle in FTR, this ratio increases by about 3%. This effect leads to a burnup parameter, as defined by RIBD, of

$$T = \ln \left[\frac{1 + (1.03)(0.121)}{1 + 0.121} \right] = 0.0032 .$$

In practice, the ^{238}U to ^{239}Pu fission rate ratio can change by a larger fraction due to fuel reloading. Fortunately, as noted previously, the fission-product decay heat is rather insensitive to this ratio.

For a particular irradiation history, we can now compute the FTR fission-product decay heat. A typical exposure history for the FTR fuel at the end-of-equilibrium cycle is three cycles which consist of three 100-day irradiation periods separated by 30-day shutdown periods. Nevertheless, to accommodate the variety of exposures expected in actual operations, calculations were made for a variety of exposure histories and a nominal neutron flux of $5 \times 10^{15} \text{ n/cm}^2\text{-sec}$.

It is more useful to express the decay power, P_d , as a fraction of the operating power, P_o , rather than relative to the fission rate, R_f , as is given in Figure 2. P_o can represent the overall reactor power or some fraction, as for a single subassembly. As noted earlier [Equation (10)], the appropriate conversion factor is the energy released per fission, E_f :

$$\frac{P_d}{P_o} = \frac{P_d/R_f}{P_o/R_f} = \left(\frac{1}{E_f} \frac{P_d}{R_f} \right) \quad (16)$$

Fission-product results are shown in Figures 4-A and 4-B as a function of the cooling time after reactor shutdown.

To apply these results with confidence to individual subassemblies where the flux may deviate substantially from the nominal value given, we have made a parametric investigation of the effect of neutron capture.

For fission-product decay-heat, the neutron flux is important only as a small correction caused by the transmutation of fission products. It is therefore reasonable to express the dependence of decay heat on the neutron flux by the following parametric form:

$$F(t,T) \approx F^*(t,T) \left[1 + \left(\frac{\phi - \phi^*}{\phi^*} \right) C(t,T) \right] \quad (17)$$

where $F(t,T)$ is the decay power for an irradiation period, T , and cooling time, t . $F^*(t,T)$ is the decay power for a reference flux, ϕ^* , taken to be 5×10^{15} n/cm²-sec in this study. $C(t,T)$ is a flux-independent coefficient that allows one to obtain $F(t,T)$ from the reference function, F^* , according to Equation (17). The function, $C(t,T)$, was obtained and the accuracy of Equation (17) was checked by parametric RIBD runs for neutron fluxes from 0 to 8×10^{15} n/cm²-sec and for irradiation histories up to six cycles of operation. For this range, the accuracy of Equation (17) was always better than 1%. The results for $C(t,T)$ are shown in Figure 5, and enable one to correct

FIGURE 4-A. Fission-Product Decay Heat for FTR Fuel (Flux = 5×10^{15} n/[cm²-sec]).

FIGURE 4-B. Fission-Product Decay Heat for FTR Fuel (Flux = 5×10^{15} n/[cm²-sec]).

FIGURE 5. Neutron Capture Corrections for Fission-Product Decay Heat (Flux = 5×10^{15} n/[cm²-sec]).

the results shown in Figure 4 for an arbitrary flux. Notice from the values in Figure 5 that these corrections are at most a few percent for cooling times less than 10^7 -seconds (116 days).

C. Semi-Analytic Representation of Decay Heat

The RIBD computer code can be used to obtain the fission-product decay heat for an arbitrary exposure history. And although RIBD does not require excessive amounts of computer time to run, it is desirable to have a simpler computational scheme for the variety of applications that arise.

We note, first of all, that the capture correction, $C(t,T)$, depicted in Figure 5 is remarkably linear in the irradiation time, T . Thuse we define a new function, $\Theta(t)$, that depends only on the cooling time, t , by

$$C(t,T) \approx \left(\frac{T}{T^*} \right) \Theta(t) \quad (18)$$

where T^* is a reference irradiation time.

Next, we use Equation (17) to obtain the function $F^*(t,T)$ in terms of $F_0(t,T)$, the decay-heat function for zero flux ($\phi=0$). A little algebra easily yields

$$F(t,T) \approx F_0(t,T) \left[1 + \left(\frac{C}{1-C} \right) \left(\frac{\phi}{\phi^*} \right) \right]. \quad (19)$$

Then, with the use of Equation (18) and recognizing that $C(t,T)$ is much less than one, we obtain

$$F(t,T) \approx F_0(t,T) \left[1 + \left(\frac{\phi}{\phi^*} \right) \left(\frac{T}{T^*} \right) \Theta(t) \right]. \quad (20)$$

Finally, we note that for any irradiation history, the decay heat function, F_0 , can be obtained by superposition of $f(t)$, the decay heat function for a single fission. In particular, for an irradiation period, T , we have

$$F_0(t,T) = \int_t^{t+T} f(t) dt \quad . \quad (21)$$

Therefore, if one is given $f(t)$ and $\phi(t)$, one can obtain $F(t,T)$ for an arbitrary flux and irradiation history.

In practice, a sum of exponential terms provides a convenient representation for $f(t)$:

$$f(t) = \sum_{\alpha} a_{\alpha} e^{-\lambda_{\alpha} t} \quad , \quad (22)$$

since the functions $e^{-\lambda_{\alpha} t}$ are easily integrated to obtain $F_0(t,T)$. Table III gives a set of values for $\{a_{\alpha}\}$ and $\{\lambda_{\alpha}\}$ adequate to describe $f(t)$ from 0 to 10^9 seconds.

The expansion coefficients, $\{a_{\alpha}\}$, were obtained from a least-squares fit to a RIBD calculation made for a burst type irradiation and FTR fuel.

The function, $\phi(t)$, was obtained from $C(t,T)$ for a 3-cycle irradiation via Equation (18) for a reference irradiation time of $T^*=300$ days. Tabular values $\phi(t)$ are given in Table IV and may be used with a linear interpolation scheme. To summarize, the coefficients in Table III are used with Equations (21) and (22) to obtain $F_0(t,T)$ for an arbitrary operating history with zero flux. Then Equation (20) is used in conjunction with the tabular values of $\phi(t)$ in Table IV to obtain the neutron capture corrections to $F_0(t,T)$. Numerical tests indicate that this scheme is accurate to within 1% for irradiation periods less than 400 days. For longer irradiation times and long cooling times, some additional error is expected.

IV. DECAY HEAT FROM ^{239}U , ^{239}Np , AND OTHER ACTINIDES

At short cooling times, one must account for the decay heat released by ^{239}U and ^{239}Np while for longer times, one must consider the buildup of heavier actinides, most notably ^{242}Cm .

TABLE III

COEFFICIENTS FOR THE EXPONENTIAL REPRESENTATION
OF THE DECAY-HEAT FUNCTION, $f(t)$

α	λ_{α}	a_{α}
1	1	2.3482×10^{-3}
2	3×10^{-1}	1.0759×10^{-3}
3	1×10^{-1}	5.6486×10^{-4}
4	3×10^{-2}	1.9598×10^{-4}
5	1×10^{-2}	8.9814×10^{-5}
6	3×10^{-3}	8.5333×10^{-6}
7	1×10^{-3}	6.1089×10^{-6}
8	3×10^{-4}	1.6523×10^{-6}
9	1×10^{-4}	2.3204×10^{-7}
10	3×10^{-5}	6.3981×10^{-8}
11	1×10^{-5}	1.3890×10^{-8}
12	3×10^{-6}	2.4085×10^{-9}
13	1×10^{-6}	1.1487×10^{-9}
14	3×10^{-7}	1.9397×10^{-10}
15	1×10^{-7}	4.9865×10^{-11}
16	5×10^{-8}	5.9745×10^{-12}
17	2×10^{-8}	1.1311×10^{-11}
18	1×10^{-8}	-1.4108×10^{-12}
19	5×10^{-9}	6.0479×10^{-13}
20	2×10^{-9}	-3.8288×10^{-13}
21	1×10^{-9}	4.7912×10^{-13}

TABLE IV

CAPTURE CORRECTION FUNCTION, $\theta(t)$

<u>t, sec</u>	<u>$\theta(t)$</u>
0	3.6146×10^{-3}
1	3.8430×10^{-3}
1×10^1	4.340×10^{-3}
1×10^2	4.9390×10^{-3}
1×10^3	6.5927×10^{-3}
1×10^4	1.0923×10^{-2}
1×10^5	1.5599×10^{-2}
1×10^6	1.7410×10^{-2}
4×10^6	1.2789×10^{-2}
1×10^7	8.9656×10^{-3}
1.5×10^7	8.9983×10^{-3}
2.5×10^7	1.1414×10^{-2}
4×10^7	1.5742×10^{-2}
6.7×10^7	2.3313×10^{-2}
1×10^8	3.3965×10^{-2}
1.5×10^8	4.8069×10^{-2}
2.5×10^8	4.2239×10^{-2}
4×10^8	1.7687×10^{-2}
6.7×10^8	6.0914×10^{-3}
1×10^9	2.8212×10^{-3}

The decay heat from ^{239}U and ^{239}Np , which can contribute as much as 20% of the total decay power for cooling times near a day, is well understood and we review it briefly here. The major source of both nuclides is from neutron capture in ^{238}U as given by

so that their decay power is sensitive to the ^{238}U inventory. Because of their relatively short half-lives (23.5 min for ^{239}U and 2.35 days for ^{239}Np), the inventories of these two nuclides quickly reach equilibrium with the amount of ^{238}U during reactor operation. Their inventories are thus given by

$$\lambda_{29}n_{29}(0) = \lambda_{39}n_{39}(0) = \phi\sigma_c^{28}n_{28} \quad (23)$$

where the λ 's denote decay constants, and the last term gives the production rate due to neutron capture in ^{238}U . As an example of the subscript convention, 39 denotes ^{239}Np whose charge and atomic number are 93 and 239, respectively. After shutdown, the inventories follow the usual decay equations. Decay power is then given by

$$P_i = \bar{\epsilon}_i \lambda_i n_i(t) \quad (24)$$

for both ^{239}U and ^{239}Np (as indexed by i) and where $\bar{\epsilon}_i$ is the corresponding average decay energy. Table V gives both the beta and gamma decay energies as recently evaluated by Reich.⁽²⁵⁾ Internal conversion energy is included as both gamma and beta energy as appropriate, e.g. half of the ^{239}Np beta energy arises from conversion electrons.

The evaluation of Equation (24) is straightforward and depends [see Equation (23)] on the conversion rate of ^{238}U as given by $\phi\sigma_c^{28}n_{28}$. Equations (7), (9) and (13) can be used to obtain a core-wide average relative to the total fission rate, R_f , as follows:

TABLE V

AVERAGE ^{239}U AND ^{239}Np DECAY ENERGIES, MeV

	$\overline{\epsilon}_{\gamma}$	$\overline{\epsilon}_{\beta}$	$\overline{\epsilon}_{\text{Total}}$
^{239}U	0.06325	0.4103	0.4735
^{239}Np	0.1777	0.2414	0.4191

$$\begin{aligned}
\frac{\langle \phi \sigma_c^{28} n_{28} \rangle}{R_f} &= \frac{\sum_{\alpha} \phi_{\alpha} \sigma_{c\alpha}^{28} n_{\alpha}}{\sum_{\alpha} \phi_{\alpha} \sum_i \sigma_{f,\alpha i} n_{\alpha i}} \\
&= 0.4980 \left[\frac{1 + 1.586 r_{\phi}}{1 + 2.029 r_{\phi}} \right]
\end{aligned} \tag{25}$$

where r_{ϕ} is the ratio of the outer driver flux to the inner driver flux. From this result we see that the conversion rate of ^{238}U varies from 0.425 atoms/fission for the outer drivers ($r_{\phi}=\infty$) to 0.498 atoms/fission for the inner drivers ($r_{\phi}=0$) reflecting the relatively greater fraction of ^{238}U in the inner drivers. For a ratio, $r_{\phi} = 0.67$ as discussed earlier, we obtain

$$\frac{\langle \phi \sigma_c^{238} n_{28} \rangle}{R_f} = 0.4353 \text{ atoms/fission} \tag{26}$$

for a core-wide average of the ^{238}U conversion rate. For the FTR, one source of uncertainty in this rate stems from variations in the ^{238}U inventories. Fuel cycles and various test loadings can increase the relative amount of ^{238}U by about 7% with a comparable increase in the ^{239}U and ^{239}Np heating rates. The ^{238}U capture cross section also contributes an uncertainty of a few percent.

Although they are relatively unimportant at short times, the buildup of other transuranium nuclides can contribute significantly for cooling times greater than a few days. This buildup has previously been studied by Boroughs et al⁽²⁶⁾ in order to deduce neutron source strengths for the FTR. From their inventories, we have computed the decay heat for these nuclides and find that the major contributor for cooling times less than several hundred days is ^{242}Cm which alpha decays with a decay energy^(27,28) of 6.11 MeV and a half-life of 163 days. Given the initial inventory, $n_c(0)$, of ^{242}Cm , the decay power, $P_c(t)$, as a function of the cooling time, t , is readily obtained from

$$P_c(t) = \bar{\epsilon}_c \lambda_c n_c(o) e^{-\lambda_c t} \quad (27)$$

where $\bar{\epsilon}_c$ and λ_c are the ^{242}Cm decay constants. The initial inventory, $n_c(o)$, is a consequence of multiple buildup paths during reactor operation. Nevertheless, the main source of ^{242}Cm is from the following pathway:⁽²⁶⁾ After reprocessing, the amount, n_A , of ^{241}Am builds up from the decay of ^{241}Pu as described by

$$n_A(t_s) = n_p(o) \left[1 - e^{-\lambda_p t_s} \right], \quad (28)$$

where $n_p(o)$ is the initial ^{241}Pu inventory after reprocessing, and t_s is the shelf life of the reprocessed fuel prior to the start of irradiation. The decay constant, λ_p , corresponds to a 15y half-life⁽²⁷⁾ for ^{241}Pu . During irradiation, neutron capture in ^{241}Am leads to the ground state of ^{242}Am which quickly beta decays to ^{242}Cm . For typical operating conditions, the slow depletion of ^{241}Am is partially compensated by the continuing decay of ^{241}Pu . The relatively constant inventory of ^{241}Am during irradiation then leads to the following simple result for the amount of ^{242}Cm at the end of an irradiation period, T :

$$n_c(o) = \phi \sigma_c^g n_A(t_s) \left[1 - e^{-\lambda_c T} \right] / \lambda_c, \quad (29)$$

($n_c(o)$ is an initial inventory [see Equation (27)] in the sense that the decay power is measured from the end of the irradiation period). The capture cross section, σ_c^g , is for capture in ^{241}Am to the ground state of ^{242}Am . Comparisons of Equation (29) with detailed calculations indicate an accuracy of about 10%. If the ^{242}Cm decay constant, λ_c , is increased to account for absorption, this accuracy is substantially improved.

To obtain numerical values, we first express the ^{242}Cm decay heat relative to the fission rate, R_f , as

$$\frac{P_c(t)}{R_f} = \frac{1}{\epsilon_c} \frac{\langle \phi \sigma_c^g n_A(t_s) \rangle}{R_f} e^{-\lambda_c t} \left[1 - e^{-\lambda_c T} \right] . \quad (30)$$

With the aid of Equation (28), the ^{241}Am inventories can be obtained from the ^{241}Pu inventories in Table I. Because the inventories in Table I correspond to a shelf-life of 3.75 years after reprocessing, the initial ^{241}Pu are bigger by a factor of $e^{\lambda_p(3.75 \text{ yrs})}$. If we then assume a shelf-life⁽²⁶⁾ of 7.75 years between reprocessing and reactor startup, the ratio of the amount of ^{241}Am at reactor startup to the amount of ^{241}Pu given in Table I is 0.358. For the FTR, the ^{241}Am capture cross section, σ_c^g , to the ground state of ^{242}Am is about 1.1 barns. The conversion rate of ^{241}Am to ^{242}Cm is now readily found as before for the conversion rate of ^{238}U :

$$\frac{\langle \phi \sigma_c^g n_A \rangle}{R_f} = 0.0035 \text{ atoms/fission} . \quad (31)$$

Unlike the ^{238}U conversion rate, this result is independent of the fluxes and may be applied to any part of the reactor. The reason is that the total fission rate is closely correlated to the Pu inventories and hence the ^{241}Am inventory as well.

In addition to the small errors in Equation (30) that can be eliminated by more sophisticated burnup codes, other uncertainties in the curium heating arise from the capture cross section and the fuel shelf-life. At one-sigma, the uncertainties in σ_c^g are about 20%. For a shelf-life small compared to the 15-year half-life of ^{241}Pu , the ^{242}Cm inventory is roughly proportional to the shelf-life. Thus a variation of 2 years out of 6 in the shelf-life, for example, would introduce a further variation of about 30% in the curium heating rate. We should also emphasize that any variations in the initial ^{241}Pu concentrations are important. If fuels with high ^{241}Pu content are used, the curium heating would increase accordingly. Other isotopes, such as ^{244}Cm , may also become important. In practice, these variations can be difficult to properly assess. The fuel can come from multiple sources with a variety of reprocessing times.

V. TOTAL DECAY HEAT FROM FTR FUEL

In the preceding sections, we have separately considered the decay heat from fission-products, ^{239}U , ^{239}Np , and other actinides. Although the fission products are the dominate source of decay heat in most circumstances, the contribution of the remaining components depends on the particular situation. For instance, the decay heat from ^{239}U and ^{239}Np would be relatively larger for a blanket test assembly with its higher ^{238}U content. Or as we saw earlier, fuels with a higher initial ^{241}Pu inventory have more decay heat from the decay of ^{242}Cm . In this section, we combine the individual components to obtain the total decay heat for the expected operation of the FTR for cores 1 and 2. Besides providing an overall value needed for the design of decay-heat removal systems, this exercise allows us to easily compare the relative importance of the separate components. As mentioned before, structural activation is beyond the scope of this study and is not included.

A. Calculations

The results given up to this point have been quite general. We now address the specific details needed to obtain a final answer, particularly the fuel cycles and power histories. The general pattern of FTR operation is expected to be a series of 100-day irradiation periods separated by 30-day intervals for reloading. Goal exposure for most of the driver assemblies is 3 or 4 cycles depending on whether or not they are in high or low power positions. However, there are plans to irradiate some drivers for as long as 5 or 6 cycles, and blanket experiments may go for 8.

The actual irradiation history will not be known with certainty, of course, until the reactor has been operated. Fortunately, as shown in previous sections, the decay heat is rather insensitive to the details of the fuel cycle. The most important parameters are the total exposure time and the operating power. A possible reactor irradiation history is given in Table VI where, in order to be conservative, maximum expected exposures are

TABLE VI
EXAMPLE IRRADIATION HISTORY FOR THE FTR

<u>Assembly Type</u>	<u>Quantity</u>	<u>Typical Power Level When Fresh (MW)</u>	<u>Typical In-Reactor Lifetime (Full Power Days)</u>
<u>Driver Assembly</u>			
High-Power Positions	20	6.05	300
Low-Power Positions	24	4.65	400
<u>High-Burnup Driver</u>			
High-Power Positions	5	6.05	500
Low-Power Positions	3	4.65	600
<u>Advanced Fuels Experiments</u>	14	5.32	500
<u>Oxide Fuels Experiments</u>	11	4.26	500 to 600
<u>Closed Loops</u>	2	2.3 max.	500 to 600
<u>Blanket Experiments</u>	4	0.5	800

given. After each 100 days, about one-third of the core is reloaded so that in an equilibrium cycle only one-third of the fuel has attained goal exposure. Therefore, in Table VI, the interpretation of 20 driver assemblies with 300-day lifetimes is that, in equilibrium, there are 7 drivers that have been irradiated for 3 cycles, 7 drivers for 2 cycles, and 6 drivers for 1 cycle. A more concise summary is given in Table VII, which was used as a basis for computing the decay heat for FTR at the end-of-equilibrium cycle.

The decay heat for any particular irradiation history may be found by the superposition of intervals of constant power. We first represent the power history of the reactor by a set of contiguous time intervals, $\{\alpha\}$. Denote the irradiation time for each interval, α , by T_α ; and denote the associated operating power (which may be zero) by P_α . Then by superposition, the decay power, P_d , is given by

$$P_d(t) = \frac{1}{E_f} \sum_{\alpha} P_{\alpha} F_T(\tau_{\alpha}, T_{\alpha}) \quad (32)$$

where $F_T(\tau_{\alpha}, T_{\alpha})$ is the total decay power for interval, α , expressed relative to the fission rate and where the energy released per fission, E_f , is the necessary conversion factor [see Equation (10)]. The cooling time, τ_{α} , relative to interval, α , is related to the cooling time, t , measured from the end of the final irradiation period by

$$\tau_{\alpha} = t + \sum_{\alpha' > \alpha} T_{\alpha'} \quad (33)$$

The total decay heat function, F_T , is the sum of terms due to fission products, ^{239}U + ^{239}Np , and ^{242}Cm :

$$F_T(\tau_{\alpha}, T_{\alpha}) = F_{f.p.}(\tau_{\alpha}, T_{\alpha}) + F_{239}(\tau_{\alpha}, T_{\alpha}) + F_{242}(\tau_{\alpha}, T_{\alpha}) \quad (34)$$

The fission-product term, $F_{f.p.}$, is readily computed from Equations (20)-(22). Due to transmutation by neutron capture, the superposition principle implied by Equation (32) is not strictly true. For this reason, the neutron capture correction in Equation (20) is evaluated by using the total irradiation

TABLE VII

IRRADIATION HISTORY SUMMARY AT THE END-OF-EQUILIBRIUM CYCLE

<u>Core Power, MW</u>	<u>Fuel Cycles</u>
80.88	1
92.23	2
92.23	3
54.53	4
50.73	5
33.00	6
0.50	7
0.50	8
<hr/>	
TOTAL 404.60	

(4.6 MW is from 2 closed loops.)

interval, $\sum_{\alpha' > \alpha} T_{\alpha'}$, rather than T_{α} for the irradiation time, T . This problem is completely eliminated if one uses a general summation code such as RIBD which treats the problem in detail. However, so long as one does not apply Equation (22) to situations where the operating history is divided into many very short irradiation periods, little error will be incurred.

It is easy to show that each of the remaining terms in Equation (34) has the form,

$$F_i(\tau_{\alpha}, T_{\alpha}) = \frac{C_i}{\epsilon_i} e^{-\lambda_i \tau_{\alpha}} \left(1 - e^{-\lambda_i T_{\alpha}} \right) \quad (35)$$

where C_i is the appropriate conversion rate [see Equations (26) and (31)]. The F_{239} term consists of two such terms, one for ^{239}U and one for ^{239}Np . Actually, for ^{239}Np there is a negligible correction to Equation (35) due to the parent, ^{239}U , decay.

Based on the irradiation history summarized in Table VII, these equations, Equations (32)-(35), were used to obtain the decay heat from FTR fuel. The results are given in Table VIII for a total core power of 404.6 MW (400 + 4.6 maximum for two closed loops). The FTR allows one to store irradiated fuel in the reactor vessel (in-vessel storage, IVS). Based on a plausible, yet conservative with respect to decay-heat removal, set of 23 fuel assemblies in IVS, Table VIII also shows the additional heat from IVS. The relative importance of the individual components is shown in Figure 6. The contribution of ^{239}U and ^{239}Np is most important near 2×10^5 sec which corresponds to the 2.35-day half-life of ^{239}Np . Likewise, ^{242}Cm is a significant contributor only for times near its 163-day (1.41×10^7 sec) half-life. To allow for variations in the fuel inventories, the conversion rates in Equation (35) were increased by a few percent. The effect on the total decay heat is small.

It is worthwhile to note that, for a fixed operating power, the dependence of these results on the flux is small. Not only are the conversion

TABLE VIII
DECAY HEAT FROM FTR FUEL

Time After Shutdown, Sec.	From Core	From Core + IVS	% Uncertainty At 1-Sigma
1.000×10^0	2.003×10^1	2.008×10^1	9
2.000×10^0	1.921×10^1	1.926×10^1	8
5.000×10^0	1.779×10^1	1.784×10^1	7
1.000×10^1	1.647×10^1	1.652×10^1	6
2.000×10^1	1.498×10^1	1.503×10^1	6
5.000×10^1	1.285×10^1	1.290×10^1	5
1.000×10^2	1.121×10^1	1.126×10^1	4
2.000×10^2	9.706×10^0	9.755×10^0	4
5.000×10^2	8.063×10^0	8.112×10^0	4
1.000×10^3	6.852×10^0	6.901×10^0	4
2.000×10^3	5.582×10^0	5.631×10^0	4
5.000×10^3	4.123×10^0	4.172×10^0	3
1.000×10^4	3.321×10^0	3.369×10^0	3
2.000×10^4	2.750×10^0	2.799×10^0	3
5.000×10^4	2.142×10^0	2.191×10^0	3
1.000×10^5	1.733×10^0	1.782×10^0	3
2.000×10^5	1.366×10^0	1.414×10^0	3
5.000×10^5	9.282×10^{-1}	9.756×10^{-1}	3
1.000×10^6	6.570×10^{-1}	7.030×10^{-1}	3
2.000×10^6	4.541×10^{-1}	4.975×10^{-1}	4
5.000×10^6	2.667×10^{-1}	3.038×10^{-1}	4
1.000×10^7	1.669×10^{-1}	1.964×10^{-1}	5
2.000×10^7	9.439×10^{-2}	1.148×10^{-1}	6
5.000×10^7	3.498×10^{-2}	4.428×10^{-2}	5
1.000×10^8	1.232×10^{-2}	1.594×10^{-2}	4
2.000×10^8	3.577×10^{-3}	4.811×10^{-3}	4
5.000×10^8	1.993×10^{-3}	2.723×10^{-3}	4

FIGURE 6. FTR Decay Heat From The Core Plus In-Vessel Storage.

rates for the production of ^{239}U , ^{239}Np , and ^{242}Cm rather insensitive to the flux, but in addition these components only contribute a small fraction of the decay heat. Furthermore, the only direct dependence of fission-product decay heat on the neutron flux is through the small neutron capture correction. As a consequence, the decay-heat results obtained for a core-wide average can be applied to individual fuel assemblies in many cases without incurring large additional errors. One simply assumes that the decay heat for a particular assembly is the same fraction of the total decay heat as for the corresponding operating powers. Some caution is advised. As shown in earlier sections, fuel inventories and irradiation history are moderately important parameters. The irradiation history is especially important for cooling times that are comparable to or longer than the irradiation time.

B. Uncertainties

An expression for the fractional uncertainty in the decay power, P_d , may be obtained from Equation (32):

$$\frac{V(P_d)}{P_d^2} = \frac{V(P_o)}{P_o^2} + \frac{V(E_f)}{E_f^2} + \frac{V(F_T)}{F_T^2} \quad (36)$$

where $V(X)$ denotes the variance of X . In this expression, the uncertainty in the operating power, $V(P_o)/P_o^2$, is assumed to represent an error that is common to each of the operating powers, P_α . Because this uncertainty is not related to the factors discussed in this study, the corresponding term in Equation (36) is not included here. Nevertheless, it should be added as deemed necessary in any final use. The uncertainty in the energy released-per-fission, E_f , is relatively small, about 1%. Based on Equation (34), the remaining uncertainty is simply

$$V(F_T) = V(F_{f.p.}) + V(F_{239}) + V(F_{242}) \quad (37)$$

Because decay heat is dominated by the fission-product contribution, the total uncertainty, $V(F_T)$, is largely determined by uncertainties in this

component for most cooling times. The fission-product uncertainties are shown in Figure 3 for ^{239}Pu and ^{238}U . The uncertainties due to other fissionable nuclides are very similar so they need not be separately considered. As discussed earlier, variations in the ^{242}Cm component may be difficult to establish. To allow for possible large variations in the ^{241}Pu content of the fuel as well as other uncertainties described before, a 50% uncertainty was assigned to the ^{242}Cm term. Since this term is always relatively small, the total decay heat error is not drastically changed by such a large uncertainty. Finally, a 5% uncertainty applied to the F_{239} term has an insignificant effect on the total uncertainty. Uncertainties based on Equations (36) and (37), but neglecting any uncertainty in the operating power, are included in Table VIII. Except for cooling times near 2×10^7 sec where ^{242}Cm causes a small increase and below 100 sec where the fission product heat becomes increasingly uncertain, the total decay-heat uncertainty is less than or equal to 4% at the one-sigma confidence limit. These uncertainties represent a major reduction from previous values.

VI. SUMMARY

The decay heat associated with the FTR fuel was reevaluated. The new evaluation is in agreement with an earlier evaluation⁽⁵⁾ within the uncertainties given in the prior study; however, for cooling times less than 1000 seconds, values were reduced by as much as 18%. In the range of cooling times between 100 and 10^6 seconds (~ 12 days), a detailed uncertainty assessment has reduced earlier uncertainties⁽⁵⁾ of 15-25% to 4% or less at the one-sigma confidence level.

Major contributors to decay heat that were reexamined include the fission-product heat from ^{235}U , ^{240}Pu , and ^{241}Pu as well as ^{238}U and ^{239}Pu . The decay heat from ^{239}U and ^{239}Np was reviewed, and the decay heat from the buildup of other actinides was studied. ^{242}Cm was found to be important for cooling times near its 163-day half-life. A by-product of this work was a reevaluation of the effective energy released per fission in FTR.

A reduction to 212.0 ± 1.5 MeV was obtained compared to an earlier estimate⁽²⁴⁾ of 215.5 MeV.

Parametric studies were done to describe the effect of irradiation history and transmutation by neutron capture. And finally, semi-analytic expressions were developed that allow one to quickly ascertain decay-heat values for specific applications such as a high-burnup driver assembly. Decay-heat not associated with the fuel, in particular structural activation, was excluded from this study.

VII. ACKNOWLEDGMENT

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APPENDIX A

GROSS ESTIMATES OF DECAY HEAT FOR NUCLIDES WITH UNKNOWN YIELDS

At shutdown, after an infinite irradiation period, the difference in decay heat between two fissionable nuclides can be estimated⁽⁷⁾ from

$$\Delta F(0, \infty) = -2 \bar{\epsilon}_p \langle \Delta Z_p \rangle \quad (A.1)$$

where the average decay energy, $\bar{\epsilon}_p$, for fission-product nuclides near the peak of the fission charge distribution is about 3.5 MeV. The value, ΔZ_p , is an average, taken over all mass chains, of the corresponding differences in the most probable charges, $\{Z_{pc}\}$, created in fission. The small difference in decay heat between a long but finite exposure and the infinite exposure considered here allows us to apply the above result to typical reactor exposures. At longer cooling times, the difference between two decay heat functions decreases compared to the shutdown time estimate given here. And for much longer cooling times where decay heat is more sensitive to chain yields, the aforementioned estimate becomes invalid. Within the content of these limitations, we proceed to evaluate ΔZ_p .

By definition, $\langle \Delta Z_p \rangle$ is given by

$$\langle \Delta Z_p \rangle = \frac{1}{2} \sum_c Y_c \Delta Z_{pc} \quad (A.2)$$

where the sum is over all mass chains, c , and where the set of chain yields, $\{Y_c\}$, are normalized to 2. We see from this equation that Eq. (A.1) depends on the assumption (valid only for the shorter cooling times) that any differences in the decay heat, F , are due solely to changes in the most probable charges and that any differences, $\{\Delta Y_c\}$, in the chain yields are unimportant. Nevertheless, charge conservation links changes in the chain yields with changes in the most probable charges so that the chain yields are indirectly involved.

Both neutron and charge conservation are given by

$$\sum_c A_c Y_c = A_f - \bar{\nu}_p \quad (A.3)$$

and

$$\sum_c Z_{pc} Y_c = Z_f \quad (A.4)$$

where $\bar{\nu}_p$ is the number of prompt neutrons. The charge and atomic number of the fissioning nucleus are denoted by Z_f and A_f , while A_c gives the atomic number of mass chain, c . Eq. (A.4) can now be applied to relate ΔZ_p to changes in $\{Y_c\}$ and Z_f :

$$2\langle\Delta Z_p\rangle = \Delta Z_f - \sum_c Z_{pc} \Delta Y_c \quad (A.5)$$

This last term can be evaluated from Eq. (A.3), if we employ the following model description for the most probable changes:

$$Z_{pc} = \frac{A_c Z_f}{A_f - \bar{\nu}_p} \pm \delta \quad , \quad (A.6)$$

where δ denotes the deviation in the light and heavy mass peaks from a uniform distribution of change in fission. The result is

$$2\langle\Delta Z_p\rangle = \Delta Z_f - \left(\frac{Z_f}{A_f - \bar{\nu}_p} \right) (\Delta A_f - \Delta \bar{\nu}_p) \quad (A.7)$$

To test this formula, we calculated $F(0, \infty)$ from ENDF/B-IV yields for several nuclides listed in Table A.I. The difference, ΔF , between the decay heat of each nuclide and the decay heat released in the thermal fission of ^{235}U was then obtained and is compared to the results found from the use of Eqs. (A.1) and (A.7). As also shown, in Table A.I, the agreement is quite good for such a simple model. Even the apparent discrepancy for ^{241}Pu is not serious if one notes the large amount of cancellation required by the

TABLE A.I
ESTIMATES OF DECAY HEAT AT SHUTDOWN

<u>Nuclide</u>	<u>Z_f</u>	<u>A_f</u>	<u>$\bar{\nu}_p$</u>	<u>$F(0, \infty), \frac{\text{MeV}}{\text{fiss}}$</u> <u>ENDF/B-IV</u>	<u>$\Delta F = F(\text{nuclide}) - F[{}^{235}\text{U(T)}]$</u> <u>ENDF/B-IV</u>	<u>Eq. (A.1)</u>
${}^{235}\text{U(T)}$	92	236	2.41	12.38	0	0
${}^{235}\text{U(F)}$	92	236	2.38	-		0.04
${}^{238}\text{U(F)}$	92	239	2.70	16.06	3.68	3.69
${}^{239}\text{Pu(F)}$	94	240	2.77	10.07	-2.31	-1.95
${}^{240}\text{Pu(F)}$	94	241	2.90†	-		-0.78
${}^{241}\text{Pu(F)}$	94	242	3.00*	12.36*	-0.02	0.45

*Calculated from thermal yields.

†Evaluated value.

terms in Eq. (A.7). In fact, if one extrapolates from ^{239}Pu to ^{241}Pu so that cancellation is not needed, the agreement is very good. The value of $\bar{\nu}_p$ for ^{240}Pu whose yields are not in ENDF/B-IV is evaluated. For the other nuclides, $\bar{\nu}_p$ is calculated from the ENDF/B-IV yields.