GULF GENERAL ATOMIC

Gulf-GA-A12499
(GA-LTR-4)

AFTERHEAT CALCULATIONS FOR THE HTGR

Work done by:

R. E. Sund
D. E. Hurd
M. J. Haire
C. J. Hamilton

Report written by:

R. E. Sund

November 1, 1973

GULF GENERAL ATOMIC COMPANY
P.O. BOX 81608, SAN DIEGO, CALIFORNIA 92138

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

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

DISCLAIMER

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

CONTENTS

FOREWARD.	iii
ABSTRACT.	iv
1. INTRODUCTION	1
2. AFTERHEAT CALCULATIONS	3
2.1 Inventory of Radioactive Isotopes.	3
2.2 Decay Energies of Radioactive Isotopes	5
2.3 Delayed Neutrons	6
2.4 Afterheat Results for HTGR	7
3. UNCERTAINTY ANALYSIS FOR AFTERHEAT CALCULATIONS.	9
3.1 Uncertainties from Neglect of Fission Fragments with Low Yields.	9
3.2 Uncertainties from Neglect of Fission Fragments with Short Half-Lives.	10
3.3 Uncertainties from Fission-Fragment Yields	10
3.4 Uncertainties from Decay Energies and Half-Lives	11
3.5 Uncertainties from Neglect of Some Neutron-Capture Nuclei.	13
3.6 Uncertainties in Actinide-Nuclide Inventories.	13
3.7 Overall Uncertainties.	13
4. COMPARISON OF NEW AFTERHEAT RESULTS WITH PREVIOUS HTGR RESULTS AND WITH RESULTS FROM OTHER LABORATORIES.	14
4.1 Comparison with Previous HTGR Results.	14
4.2 Comparison with Results from Other Laboratories.	15
5. AFTERHEAT FROM SPENT FUEL ELEMENTS	17
6. GAMMA-RAY DECAY SPECTRA AND DOSE RATES	18
7. CONCLUSION	19
ACKNOWLEDGEMENTS.	37
REFERENCES.	38

TABLES

1.	Half-Lives and Energies of Radioactive Nuclides in HTGR Fuel.	20
2.	Calculated Afterheat Rates (P/P_0 in %) for Four-Year Fuel Operation	30
3.	Calculated Afterheat Rates (P/P_0 in %) for Various Contributing Radiations and for Four-Year Fuel Operation	31
4.	Percentage Uncertainties in Afterheat Results	32
5.	Gamma Decay Spectra for Four-Year Operation of Fuel Element	33
6.	Gamma-Ray Dose Rate from a Fuel Element Versus Time after Shutdown. . .	34

FIGURES

1.	New and Previous Afterheat Results.	35
2.	Gamma-Ray Decay Spectra for Four-Year Operation	36

APPENDIX

1.	Number of Curies Per Element at 100 Sec after Shutdown	
A.	Values for ^{235}U Fission Fragments and Neutron-Capture Products. .	40
B.	Values for Actinide Nuclide	42

FOREWARD

This Licensing Topical Report (LTR) has been prepared by Gulf General Atomic to update the afterheat rates for the HTGR and to document the technical bases used in the calculations. The report provides the detailed information necessary for the U. S. Atomic Energy Commission to confirm the adequacy of the new HTGR afterheat rates including the uncertainties of these rates in the 100 second to ten day time region.

ABSTRACT

Afterheat rates were calculated for the HTGR utilizing new input information. The new afterheat results are a few percent higher than the HTGR afterheat results previously obtained in the 100-to 10,000-sec time range and are 11 to 15% higher than the previous results in the one-day to ten-days time range. The uncertainties were studied in the time region from 100 sec to ten days. The new results were also compared with calculations at other laboratories.

The gamma-ray spectra were computed at several shutdown times for future applications in shielding studies.

This report completely replaces and updates previous afterheat calculations for the HTGR.

1. INTRODUCTION

Accurate calculation of afterheat generation in power reactors is currently receiving much emphasis as a result of exacting requirements for the design of shutdown core cooling systems. Better knowledge of afterheat and gamma-ray decay also benefits the areas of fuel handling, storage, and shipping. Therefore, new afterheat rates for the HTGR fuel have been calculated using detailed nuclide inventories and the latest available nuclear data. This report completely updates and replaces the previous afterheat calculations (Ref. 1).

The new afterheat calculations were done separately for the ^{233}U and ^{235}U fission products, so the maximum afterheat contribution for any HTGR fuel composition could be determined. The effect of the fuel composition on the actinide-nuclide contributions was also investigated. Approximately 25 additional isotopes were included in this afterheat calculation, as compared with the previous calculation (Ref. 1). In addition, new input information was used for fission yields, for decay schemes of the radioactive nuclides, and for the method of calculating the average beta-ray energy from the end-point energy.

The uncertainties on the afterheat results were determined from uncertainties in the fission yields, the decay schemes of the radioactive nuclides, and the calculation of the average beta-ray energies. In addition, the effect of the omission of fission fragments with low yields and the effects of other approximations in the calculations were considered.

The HTGR discussed in this document is a semi-homogeneous graphite moderated core based on the thorium-uranium cycle. Partial refueling is utilized, with approximately 25% of the core replaced at each reload on an approximately annual basis. Uranium enriched to 93% in ^{235}U is used as the feed fissile material for the initial core and reload segments. In addition, bred ^{233}U can be recycled and used as fissile material in the reload segments, with enriched uranium added as make-up fuel as necessary. Thus, there are possible equilibrium cycles with and without ^{233}U recycled fuel.

The afterheat contributions for the various nuclei considered in this calculation differ from those for other types of reactors (Refs. 2-5), including those in the proposed ANS Standard (Ref. 6). One difference is the afterheat contribution from the radioactive isotopes in the actinide series. Because of the large amount of thorium in the HTGR, a significant amount of ^{233}Th and ^{233}Pa are produced, and these nuclei are significant contributors to the afterheat values. In addition, the ^{235}U enrichment in the HTGR is significantly larger than that in light water reactors so there is a different buildup of the nuclei which are produced following neutron capture, including the transuranic nuclei. Another important aspect of the HTGR calculations is that ^{233}U and ^{235}U are present in varying fuel mixes, and the fission of both isotopes must be considered, as compared with the fission of ^{235}U and ^{239}Pu in light water reactors.

In Section 4 the afterheat results are compared with a number of results reported in the literature, and the differences in the results are discussed.

2. AFTERHEAT CALCULATIONS

2.1 INVENTORY OF RADIOACTIVE ISOTOPES

The number of curies for each radioactive fission fragment in a fuel element was calculated for a four-year steady power of 0.802 MW(t) per fuel element and at 80% load factor. This load factor means that an equivalent full-power year is 292.2 days for the reactor. The calculations were done with the use of the RADC code (Ref. 7). The fission product yields and branching fractions for the decay were obtained from the report by Meek and Rider (Ref. 8). The half-lives were typically obtained from this report (Ref. 8); almost all of the half-lives in the report (Ref. 8) were based on the Chart of the Nuclides (Ref. 9). Fission products were included in the calculation if the yield from either ^{235}U or ^{233}U was greater than 0.1% and the half-life was greater than 1 minute. In addition, any fission fragment with a known, shorter half-life was included if the yield of the isotope after correcting for a 100 sec decay was greater than 0.1%. The time of 100 sec was chosen because the HTGR core has a large heat capacity so the uncertainties in the afterheat during the first 100 sec will have a negligible effect on peak-core temperature. The calculation of the curie values was done for all of the fission events from ^{235}U and then repeated for the ^{233}U case.

For neutron-activation products resulting from fission fragments and for the radioactive nuclides in the actinide series, the curie values in a fuel element were calculated for operation of the element in a steady-state equilibrium reactor at an 80% load factor and at an average power of 0.802 MW(t) per fuel element. These calculations were performed with GARGOYLE code II (Ref. 10), which is a nine-energy group burnup code. The calculation included

the following nuclei which result from the neutron activation of fission fragments: ^{110}mAg , ^{134}Cs , ^{136}Cs , ^{148}mPm , ^{154}Eu , and ^{155}Eu . For these nuclei, the calculation was performed for four-year operation of the fuel element.

The calculation of the afterheat rates from the actinide nuclides was performed with the use of radioactive decay chains and neutron-activation reactions which are significant in the HTGR. The nuclei which result from double neutron capture produce only a small amount of the afterheat; however, to be conservative in regard to the increase in double-neutron capture events with an increase in reactor operating time, the calculation for the actinide nuclei was done for six-year operation of the fuel element. No burnup of the ^{232}Th and ^{238}U was assumed, so the nuclei produced by neutron capture of these nuclei would be conservative. The calculation was done for a carbon-to-thorium atom ratio of 190 to allow for possible future changes in the ratio. This atom ratio could be as high as roughly 270. The lower ratio (i.e., the ratio of 190) means a higher Th atom density, so the ^{233}Th and ^{233}Pa contributions to the afterheat are higher.

The decay of the fission fragments, neutron-activation products, and the nuclides in the actinide series were calculated with the RADC code. To indicate the magnitude of these results, the number of curies per fuel element at 100 sec after shutdown is given in Appendix 1 for each isotope.

The actual afterheat rates for the HTGR increase with the time of operation after the initial startup and then decrease each time the reactor fuel is reloaded, because of the replacement of some activated fuel elements and because of the decay of the activities in the remaining fuel elements during the shutdown time. If the reactor is shut down for a longer time or more frequently than the normal yearly shutdown of 14 days, the actual afterheat results will be lower than those calculated. Also, since the calculations were done for a minimum operation time of four years, the results for an equilibrium reactor will be lower than those calculated.

2.2 DECAY ENERGIES OF RADIOACTIVE ISOTOPES

The list of beta-ray and gamma-ray decay energies used in the present afterheat calculations is given in Table 1. Much of the information for the fission fragments was taken from the report by Tobias (Ref. 11); this report includes experimental data through August, 1972. The report gives a tabulation for many isotopes of the total gamma-ray energy, the beta-ray energy plus neutrino energy, the average beta-ray energy, the conversion-electron energy plus x-ray energy, and the total disintegration energy. In this report (Ref. 11), the average energy of each beta-ray was determined from calculations (Refs. 12 and 13) which included screening corrections and the effect of the forbiddenness of the transition; the results of these calculations are given as a function of the atomic number of the daughter nuclide as well as the beta-ray end-point energy. If the forbiddenness of a transition was not known, it was assumed to be allowed. In a few cases, the tabulation of Tobias indicated a significant difference between the disintegration energy and the sum of the beta-ray energy, neutrino energy, gamma-ray energy, conversion electron energy, and x-ray energy. In these cases, when the difference could not be explained (such as by the energy of an isomeric state), the data from Tobias were readjusted in a conservative manner for the afterheat calculation. For instance, in some cases the gamma-ray energy was adjusted upward to make the above sum agree with the disintegration energy. In Table 1, the conversion electron energy and x-ray energy are included with the gamma-ray energy.

Some of the information given in Table 1 was obtained from the report by Blachot and Tourreil (Ref. 14)*, from Nuclear Data Sheets (Ref. 15) for the actinide nuclei and for a few of the fission fragments, and from particular decay schemes in the literature. No definitive information

*Unfortunately, the best copy of this report available at the time of this calculation had many illegible parts.

could be found for the decay schemes of some nuclei with short half-lives; in these cases, half of the disintegration energy (Ref. 11) for each of these nuclei was arbitrarily assigned to the beta rays plus neutrinos and half to the gamma rays. The energies of the nuclei with unknown decay schemes are indicated by a question mark in Table 1.

To obtain the average beta-ray energies given in Table 1 from the references given in the previous paragraph, the method of England (Ref. 3) was used to convert the end-point energy to the average energy for each beta-ray. This result was then multiplied by the corresponding transition intensity before summing to obtain the average beta-ray energy for each isotope.

A check was made to be sure that the disintegration energy for each isotope was consistent with the sum of all the component energies. The values for the decay constants, energies and half-lives in Table 1 are taken from the computer output and were not rounded off to reflect the number of significant figures in the original data.

2.3 DELAYED NEUTRONS

The following is given as a simplified treatment to illustrate the effect of delayed neutrons. The contribution to afterheat by delayed-neutron fissions is approximately given by (Ref. 1):

$$\frac{P}{P_0} \approx e^{\rho t / \ell} - \sum_{i=1}^6 \frac{\beta_i}{\rho} e^{-\lambda_i t}$$

where P = power from afterheat at time t

P_0 = four-year power of 0.802 MW(t) per fuel element for this calculation

ρ = shutdown reactivity

t = time after shutdown

ℓ = neutron lifetime in HTGR

β_i = fraction of the total neutrons belonging to the i^{th} delayed-neutron group

λ_i = decay constant of precursors of i^{th} delayed-neutron group

and it is assumed that $|\rho| \gg \sum_{i=1}^6 \beta_i$.

Because rapid control-rod insertion is not required in an HTGR, the control-rod insertion time is approximately 22 seconds; hence, shutdown reactivity is really a function of time. This effect was not taken into account in the present illustrative treatment, since the delayed neutrons do not contribute significantly to the afterheat for times after 100 sec; it was simply assumed that the entire shutdown reactivity was applied at $t = 0$. The parameters used in the calculation were $\lambda = 3.4 \times 10^{-4}$ sec and $\rho = -0.2$; the β_i and λ_i were obtained from Reference 16.

2.4 AFTERHEAT RESULTS FOR HTGR

The afterheat results for the fission fragments and delayed neutrons are listed separately for ^{233}U and ^{235}U in Table 2. The afterheat yields from the fission fragments were increased by 0.5% to account for the neglect of fragments with low yields. The rough value of 0.5% was obtained by comparing the yields of the neglected isotopes with the yields of the isotopes included in the study. The afterheat values given in the table at 10 and 100 seconds are low because of the neglect of some short-lived fission fragments (see Section 3.2). To cover all possible fuel compositions of the HTGR, including the fuel at the beginning of reactor operation and also recycled fuel, the higher of the two afterheat values for ^{233}U and ^{235}U was used, as shown in column 4 of Table 2. The results for the fission fragments are for four-year operation at constant power.

The afterheat for the actinide nuclei was calculated for various reactor fuels, including ^{235}U fuel, recycled ^{233}U fuel, and residual recycled fuel. A reactor with the latter fuel consists of 92% normal fuel (with all ^{235}U and no ^{233}U when fresh) and a maximum of 8% fuel (with a composition of ~30% ^{235}U , ~50% ^{236}U and ~20% ^{238}U) from previously irradiated fissile particles. Calculations indicated that the actinide nuclei in the residual recycle fuel produce a slightly larger afterheat level than those from the other types of fuel, and the results for this fuel are given in

Table 2, column 6. Of all the actinide nuclei, ^{233}Th and ^{233}Pa are by far the most important contributors at times less than 1,000 sec, and ^{233}Pa is the most important contributor from 10,000 sec to 100 days.

The overall afterheat result is given in column 7 of Table 2 and is also plotted in Figure 1. Although the contributions in both columns 4 and 6 were determined for the cases giving the largest value, the data indicate that over much of the time range of interest, the maximum afterheat values for various fuels would not be significantly lower than those given in column 7 of Table 2.

Table 3 gives the calculated afterheat results for delayed neutrons, beta rays, gamma rays, and alpha rays. The contributions from fission fragments, delayed neutrons, and neutron-capture products were obtained from ^{235}U data for all times except 10,000 sec and 1,000 days, when the ^{233}U results were higher and were therefore used.

The values of afterheat given in Tables 2 and 3 are higher than those that would exist in an equilibrium HTGR, since the average age of fuel in an equilibrium reactor varies between $\sim 1\text{-}1/2$ and $\sim 2\text{-}1/2$ years. Consequently, the short-lived fission-fragment activities for the four-year old fuel are the same as those for an equilibrium core, and the long-lived fission-fragment activities are considerably higher than those for an equilibrium core. Also, the afterheat contributions in Tables 2 and 3 for the actinide nuclides are higher than those for an equilibrium core.

3. UNCERTAINTY ANALYSIS FOR AFTERHEAT CALCULATIONS

An uncertainty analysis was performed on the calculated afterheat rates in order to evaluate a suitably conservative margin of uncertainty. This analysis was done for the uncertainties that might increase the afterheat results for shutdown times of 100 sec, 1,000 sec, 10,000 sec, 1 day, and 10 days. Because of the high heat capacity of the HTGR, the uncertainties in the heat input rate during the first 100 sec after shutdown will have a negligible effect on peak core temperatures.

Because of the extremely large number of independent variables in the analysis and because of the multiple and diverse data sources, a strict statistical approach to the uncertainties for many of the parameters is extremely difficult. Furthermore, in some cases no statistical uncertainties are given for the input data. However, by utilizing the reported information on some data deviations and by basing some of the uncertainties on a knowledge of pertinent experimental data, a suitably conservative uncertainty may be estimated for each parameter in the calculation. Within the limitations just described, the specified uncertainty in the afterheat from each of the parameters is typically the result of an estimated 2σ variation in the input parameter. In several cases, the uncertainty given below is essentially an upper limit; however, for conservatism, the uncertainties are propagated as if all parametric deviations are on the order of 2σ . The results of the uncertainty analysis are summarized in Table 4 and discussed in the following sections.

3.1 UNCERTAINTIES FROM NEGLECT OF FISSION FRAGMENTS WITH LOW YIELDS

As discussed in Section 2.4, the afterheat results were increased by 0.5% to account for the neglect of fission fragments with low yields. The afterheat

uncertainty from this correction is estimated to be 2%, considering that the decay energies of the fragments with low yields were not treated in the calculation.

3.2 UNCERTAINTIES FROM NEGLECT OF FISSION FRAGMENTS WITH SHORT HALF-LIVES

The neglect of fission fragments with short half-lives has a negligible effect on the calculated results at 1,000 seconds and later times. The effect at 100 seconds was studied by listing all of the fission yields of the fragments which were not included in the calculation. Calculations based on this information and the half-lives indicate that the fission-fragment contribution to the afterheat results at 100 seconds are low by very roughly 2%. An afterheat uncertainty of 6% is used, because of the unknown half-lives for some short-lived nuclei and because of the approximations in the analysis.

3.3 UNCERTAINTIES FROM FISSION-FRAGMENT YIELDS

The effect of fission-fragment uncertainties was calculated for each shutdown time of interest as follows. First, the isotopes were listed in order of importance for afterheat. The uncertainties in the fission yields were then obtained from Reference 8 for the important fission-fragment contributors to the afterheat. This was done for the fragments on the ordered list until the fission yield of the A-chains for these fragments equaled 50% or more for the light-mass peak of the mass-yield curve. (An "A-chain" is a decay chain with unchanging mass number.) A similar procedure was followed for the heavy-mass peak of the mass-yield curve. The yields of the nuclei with the uncertainties obtained in this manner were then changed to yields at the upper limits of the uncertainties. Yields of successive nuclei in the same A-chain were also increased correspondingly. The nuclei with increased yields and the actinide nuclei, which are treated separately in the uncertainty analysis, account for about 65 to 95% of the total afterheat contribution, depending upon the shutdown time of interest. The afterheat results for ^{235}U were calculated with the changed fission yields, and compared with those given in Table 2.

The uncertainties for ^{233}U were also studied. These were typically somewhat larger than those for ^{235}U . However, in the time region from 100 sec to 1 day, the ^{233}U afterheat result plus the uncertainty from the ^{233}U fission fragment yields is smaller than the ^{235}U afterheat result except at 10,000 sec, when the ^{233}U afterheat result is larger than the ^{235}U afterheat result. Consequently, the uncertainty listed at 10,000 sec in Table 4 was determined from ^{233}U data. The other uncertainties given in the table were determined from the ^{235}U data.

The calculation for the fission yield uncertainties can be done in a number of ways, and the simplified procedure described in the previous paragraph should give essentially an upper limit for the afterheat uncertainties. For instance, in a more exact calculation, the fission yields of those nuclei in A-chains which contribute the least to the afterheat in the above analysis should be decreased to maintain two fragments per fission. Alternatively, the yields of all A-chains could be decreased from the values used in the above analysis to obtain two fragments per fission. In addition, the calculation should give a conservative result for the uncertainty, since the upper uncertainties were used for all of the important afterheat contributors, and it would be very unlikely that the actual yields would all deviate in this way from the stated yields. The uncertainty calculation is also conservative since the errors at the upper end of the uncertainty range in Reference 8 were used.

3.4 UNCERTAINTIES FROM DECAY ENERGIES AND HALF-LIVES

To determine the maximum afterheat uncertainty for the energies of nuclei with no known decay schemes, all of the disintegration energy for these nuclei was assigned to gamma rays, so no energy would be available for neutrinos; the afterheat rates were then recomputed.

The uncertainties on the afterheat caused by uncertainties in the energies of nuclei with known decay schemes were estimated for the input data used in the

calculation. Also, use was made of previous studies (Ref. 4) which showed the afterheat results for different data libraries. Since much new experimental information has become available over the last several years, the data libraries now available should be more accurate than those libraries that are several years old.

The uncertainties for the determination of the average beta-ray energies from the end-point energies should typically be small for the tabulations taken from Tobias (Ref. 11), since the calculations (Refs. 12 and 13) used by Tobias were performed more rigorously than previous ones. The main uncertainties occur for beta rays between levels with unknown spins and parities, so the forbiddenness of the transitions is not known.

As stated in Section 2.2, the method of England was used to compute the average energy of the beta rays for information taken from References 14 and 15. This method is easy to use, since the conversion to average energy is expressed in analytic form. For allowed and most non-unique, first-forbidden transitions this method gives an accurate result for fission fragments with a low atomic number and gives a result which is high by about 2-3% for nuclei with atomic numbers in the middle of the heavy-mass peak of the mass-yield curve, i.e., with atomic numbers of ~ 56 (this is in a direction that is conservative for the afterheat calculation). The forbiddenness of the transition is not treated in the method of England.

For atomic numbers of interest in this study and for a one-MeV beta ray, the average energy divided by the end-point energy for a first-forbidden, unique transition is ~ 4 to $\sim 6\%$ higher than that for an allowed transition (Ref. 12). This percentage increases significantly with a decrease in energy and with an increase in the degree of forbiddenness (Ref. 12). Considering the probability of beta rays with the higher degrees of forbiddenness, typical beta-ray energies that occur, and the percentage contribution of the beta rays to afterheat, an uncertainty in the afterheat of 4% was estimated for the determination of the average beta-ray energy from the end-point energy.

Values for estimated uncertainties from half-lives given in Table 4 were obtained from a list of the most important afterheat contributors.

3.5 UNCERTAINTIES FROM NEGLECT OF SOME NEUTRON-CAPTURE NUCLEI

The uncertainties from the neglect of some neutron-capture nuclei were estimated on the basis of a previous study (Ref. 4). Use of that study has limitations because it was made with a different (and more complete) set of nuclei than that used in the present calculation.

3.6 UNCERTAINTIES IN ACTINIDE-NUCLEI INVENTORIES

The uncertainties in the inventories of the actinide nuclei were determined from uncertainties in the input data for the calculations. The following uncertainty values were used for the main afterheat contributors: 5% for ^{233}Th , ^{233}Pa , and ^{234}Pa ; 10% for ^{237}U , and ^{239}Np ; 20% for ^{238}Pu ; and 50% for ^{238}Np . An additional 100% variation was used for the uncertainties in the production of extra ^{237}U , ^{238}Np , and ^{238}Pu in the residual recycled fuel.

3.7 OVERALL UNCERTAINTIES

The overall uncertainties were obtained from the square root of the sums of the squares of the components, and the results are given in Table 4. With a much more systematic study of the uncertainties from some of the neutron-capture nuclei and from some of the other effects, the overall uncertainties could possibly be reduced.

4. COMPARISON OF NEW AFTERHEAT RESULTS WITH PREVIOUS HTGR RESULTS AND WITH RESULTS FROM OTHER LABORATORIES

The new afterheat results, the results of previous HTGR calculations and the results obtained at other laboratories for three- to four-year operating times are shown for comparison in Figure 1. Following is a discussion of the results.

4.1 COMPARISON WITH PREVIOUS HTGR RESULTS

The new afterheat results are a few percent higher than the previous HTGR afterheat results (Ref. 1) in the 100- to 10,000-sec range, which is the important time period for the auxiliary core cooling system, and are 11 to 15% higher in the 1-day to 10-days time range. The main contributions to these differences in the 100-sec to 10,000-sec range are the following: the new, separate afterheat calculations for the ^{233}U fission fragments and the ^{235}U fission fragments decreased the yield by roughly 8% compared with the previous calculations (Ref. 1) in which each fission-fragment yield was assigned to the higher value from ^{233}U and ^{235}U ; the additional isotopes included in the new afterheat calculation increased the results by approximately 9%, 3% and 3% at 100, 1,000 and 10,000 sec, respectively; the new afterheat calculation with the residual-recycle fuel rather than all non-recycle fuel increased the afterheat from $\sim 0.5\%$ to $\sim 2\%$; and the use of a C-to-Th atom ratio of 190 in the new calculation rather than the previous ratio of 250 increased the afterheat by approximately 3%, 4%, and 4% at 100, 1,000 and 10,000 sec, respectively. The use of new input data also accounted for some of the differences.

If the uncertainties for the present results (i.e., the new results) are added to the present afterheat values, the results for P/P_0 are 4.24, 2.43, and 1.24% at 100, 1,000 and 10,000 sec. These results compare with P/P_0 values of 4.22, 2.63, and 1.22% obtained by adding a 20% uncertainty to the previous HTGR afterheat results (Ref. 1) at 100 and 1,000 sec and adding a 10% uncertainty to the previous result at 10,000 sec. (The 20 and 10% uncertainties correspond to those given in Reference 6 for uranium-fueled reactors.)

The increase in the new afterheat values at 1 and 10 days is primarily from the use of the C-to-Th ratio of 190, with the resultant increase in the amount of ^{233}Pa . Also, the new decay energy for ^{233}Pa is higher than that used previously. In the 1 to 10 day region, ^{233}Pa is relatively more important to afterheat than at earlier times.

4.2 COMPARISON WITH RESULTS FROM OTHER LABORATORIES

The afterheat values for Shure shown in Figure 1 were obtained by converting his ^{235}U fission-fragment results for infinite reactor operation to four-year operation with the method described in Reference 6. In doing this, Shure's recent data (Ref. 4) for times greater than 1,000 sec after shutdown and his older data (Refs. 2 and 4) for earlier times were used. A value of 200 MeV/fission was used to obtain the curve shown in Figure 1; data are shown for times between 100 sec and 100 days. This curve does not include a contribution from delayed neutrons or from the actinide nuclei.

The present ^{235}U fission-fragment results are ~1% lower than the value shown for Shure in Figure 1 at 100 sec after shutdown, are ~1% higher at 1,000 sec after shutdown than the value obtained from Shure's older data, are ~4% lower at 1,000 sec than the value obtained from his recent results (shown in Fig. 1), and are ~4% lower than the value shown for Shure in Figure 1 at 10,000 sec.

In comparison with the results of M. J. Bell shown in Figure 1, the present ^{235}U fission-fragment and neutron-capture results are very roughly 10%, 2% and 4% higher at 100, 1,000 and 10,000 sec, respectively. (It was difficult to read the values of Bell from his figure, resulting in a large uncertainty in these comparisons.)

Recent calculations (Ref. 4) with the CINDER code developed by England give ^{235}U afterheat results of 3.32, 1.97, and 0.927% at 100, 1,000, and 10,000 sec, respectively. These results were done for ~1.15-year operation and assume 200 MeV per fission. The CINDER results are higher than the present ^{235}U fission-fragment and neutron-capture results by approximately 3, 7, and 5% at 100, 1,000, and 10,000 sec, respectively. These values would be increased slightly for a comparison with CINDER results for four-year operation.

The differences between the present ^{235}U fission-fragment and neutron-capture results and the previous calculations are within the uncertainties of the present results. Some of the differences are caused by the differences between the input data for the previous calculations and the data (Refs. 8, 11, 14, and 15) used in the present calculations. Also, some of the difference between the present results and the CINDER results may be caused by the inclusion of more neutron-capture nuclei in the CINDER calculation. The additional fission-fragment nuclei in the CINDER calculation may contribute to some of the difference at 100 sec, but these nuclei are not expected to contribute significantly at later times.

The main difference between the previous ^{235}U results and the present total results is the contribution of the actinide nuclei. In particular, the ^{233}Th and ^{233}Pa contribution to the present afterheat results is significant for all times shown except 1,000 days.

5. AFTERHEAT FROM SPENT FUEL ELEMENTS

The actual operating power of a typical HTGR fuel element at any particular time during its four-year life is a function of element age, its position in the core, the control-rod configuration, and the age and power of adjacent elements. Hence, a large variety of operating histories and end-of-life power levels will be encountered among the spent fuel elements being unloaded.

The average fuel element generates 0.802 MW(t) when the reactor is operating. While this figure may represent the power level of the average element halfway through its lifetime, this same element operates at considerably higher power when fresh, and at significantly reduced power when nearly spent. Typically, this average element operates at 0.955 MW(t) during its first year, 0.851 MW(t) during its second year, 0.754 MW(t) in the third year, and 0.650 MW(t) in its last year.

It is obvious that the afterheat level from this spent fuel element shortly after shutdown will depend primarily upon its most recent operating history, i.e., 0.650 MW(t). After long shutdown times, the average power of the element over its lifetime will become determining; i.e., 0.802 MW(t). Therefore, the following approximations can be made:

- a. For calculations of afterheat from HTGR spent fuel during the refueling period, use the power level which existed in the spent fuel during its last year of operation.
- b. For calculations of afterheat from HTGR spent fuel during shipping, use the average power level in the spent fuel over its entire four-year life.

6. GAMMA-RAY DECAY SPECTRA AND DOSE RATES

For shielding purposes the PATH code (Ref. 17) was used to convert the nuclide inventories in a four-year-old fuel element into MeV/cm³-sec for the gamma rays. This calculation was done for the ²³⁵U fission-fragments, for actinide nuclei produced by only ²³⁵U fissions, and for a Th to C ratio of 250, rather than the conservative ratio of 190 used in the afterheat calculation. The fuel-element power was 0.802 MW(t) or ~9 W/cm³. The latter value was combined with the number of MeV per cm³-sec to obtain the number of MeV per W-sec. The results are shown in Table 5 and Figure 2 for various photon energy intervals and different times.

The gamma-ray dose rate from one fuel element at a distance of one meter from the side surface was calculated at various times after shutdown. The results are given in Table 6.

7. CONCLUSION

New afterheat rates have been calculated for the HTGR. In the time region of importance for the shutdown cooling system, i.e., in the 100-to 10,000-sec range, the new afterheat rates are a few percent higher than those previously calculated (Ref. 1), and in the 1-day to 10-days range, the new values are 11 to 15% higher than the previous values. These differences result from the use of new input data, the inclusion of additional isotopes in the present calculation, and calculational changes which make the present afterheat rates applicable to both recycle and non-recycle fuel and also applicable to a range of C to Th ratios. Also, the method of inputting the ^{233}U and ^{235}U fission yields was changed to eliminate the approximation previously made (Ref. 1).

The uncertainties that might increase the afterheat rates were studied in the time region from 100 sec to 10 days. The uncertainty results were 14% at 100 sec and 10% at later times.

The present ^{235}U fission fragment and neutron-capture results were compared to those from other laboratories. The differences in the results are within the uncertainties of the present afterheat rates. The main difference between the results from other laboratories and the present total results is the contribution from the actinide nuclei, particularly from ^{233}Th and ^{233}Pa .

For shielding purposes, the nuclide inventories in a four-year-old fuel element were converted into the number of MeV per W-sec for various photon energy intervals and times.

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL

Nuclide	Half-Life*	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
⁷⁸ As	1.52 h	1.266-04**	1.425+00	1.025+00	-
⁷⁹ Ge	1.00 m	1.155-02	1.783+00	2.514-01	-
^{79m} Se	3.90 m	2.962-03	0.000-00	9.600-02	-
⁸¹ Se	18.00 m	6.417-04	6.050-01	7.600-03	-
^{83m} Se	69.00 s	1.004-02	1.200+00	1.105+00	-
⁸³ Se	23.00 m	5.022-04	4.650-01	2.345+00	-
⁸³ Br	2.40 h	8.021-05	3.240-01	7.300-03	-
^{83m} Kr	1.86 h	1.035-04	-	4.140-02	-
⁸⁴ Se	3.20 m	3.609-03	5.340-01	4.077-01	-
⁸⁴ Br	31.80 m	3.632-04	1.243+00	1.795+00	-
⁸⁵ Se	39.00 s	1.777-02	1.425+00?	3.300+00?	-
⁸⁵ Br	3.00 m	3.850-03	1.003+00	6.460-02	-
^{85m} Kr	4.40 h	4.375-05	2.290-01	1.859-01	-
⁸⁵ Kr	10.76 y	2.041-09	2.290-01	2.200-03	-
⁸⁶ Br	54.00 s	1.283-02	1.765+00	3.296+00	-
⁸⁷ Br	55.60 s	1.246-02	2.136+00	1.865+00	-
⁸⁷ Kr	76.00 m	1.520-04	1.375+00	7.638-01	-
⁸⁸ Kr	2.79 h	6.900-05	3.070-01	2.031+00	-
⁸⁸ Rb	17.70 m	6.525-04	2.000+00	6.765-01	-
⁸⁹ Kr	3.18 m	3.632-03	1.279+00	2.106+00	-

*The notation s stands for seconds, m for minutes, h for hours, d for days and y for years.

**The notation 1.266-04 means 1.266×10^{-4} . The same notation applies to the energies and half-lives.

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
⁸⁹ Rb	15.20 m	7.599-04	8.960-01	2.245+00	-
⁸⁹ Sr	50.80 d	1.579-07	5.530-01	0.000-00	-
⁹⁰ Kr	32.30 s	2.146-02	1.116+00	1.749+00	-
^{90m} Rb	4.28 m	2.699-03	1.300+00	3.530+00	-
⁹⁰ Rb	2.70 m	4.278-03	1.789+00	2.560+00	-
⁹⁰ Sr	28.90 y	7.599-10	1.966-01	-	-
⁹⁰ Y	64.00 h	3.008-06	9.380-01	-	-
⁹¹ Rb	57.90 s	1.197-02	1.261+00	2.840+00	-
⁹¹ Sr	9.67 h	1.991-05	6.250-01	7.479-01	-
^{91m} Y	50.50 m	2.287-04	-	5.555-01	-
⁹¹ Y	58.80 d	1.364-07	6.090-01	2.600-03	-
⁹² Sr	2.69 h	7.156-05	1.980-01	1.327+00	-
⁹² Y	3.53 h	5.453-05	1.454+00	2.475-01	-
⁹³ Sr	7.50 m	1.540-03	1.060+00	2.129+00	-
⁹³ Y	10.20 h	1.887-05	1.166+00	1.092-01	-
⁹⁴ Sr	1.29 m	8.954-03	8.420-01	1.420+00	-
⁹⁴ Y	20.30 m	5.690-04	1.193+00	1.043+00	-
⁹⁵ Sr	26.00 s	2.665-02	1.306+00?	3.055+00?	-
⁹⁵ Y	10.50 m	1.100-03	1.713+00	4.829-01	-
⁹⁵ Zr	65.50 d	1.225-07	1.180-01	7.303-01	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
^{95m} Nb	87.00 h	2.213-06	-	2.355-01	-
⁹⁵ Nb	35.10 d	2.285-07	4.350-02	7.657-01	-
⁹⁶ Y	2.30 m	5.022-03	1.474+00?	3.400+00?	-
⁹⁷ Zr	16.80 h	1.146-05	7.070-01	1.708-01	-
^{97m} Nb	58.00 s	1.195-02	-	7.432-01	-
⁹⁷ Nb	73.60 m	1.569-04	4.620-01	6.653-01	-
⁹⁸ Zr	31.00 s	2.236-02	9.020-01	-	-
⁹⁸ Nb	51.00 m	2.265-04	7.670-01	2.418+00	-
^{99m} Nb	2.60 m	4.442-03	8.930-01	1.424+00	-
⁹⁹ Mo	66.60 h	2.890-06	3.900-01	1.711-01	-
^{99m} Tc	6.01 h	3.205-05	-	1.425-01	-
^{100m} Nb	12.00 m	9.625-04	1.279+00?	3.000+00?	-
¹⁰⁰ Nb	2.90 m	3.983-03	1.469+00	2.710+00	-
¹⁰¹ Mo	14.60 m	7.911-04	4.300-01	1.852+00	-
¹⁰¹ Tc	14.20 m	8.134-04	4.740-01	3.439-01	-
¹⁰² Mo	11.10 m	1.041-03	3.527-01	-	-
^{103m} Mo	5.30 h	3.632-05	8.955-01?	2.200+00?	-
¹⁰³ Mo	66.00 s	1.050-02	8.955-01?	2.200+00?	-
¹⁰³ Tc	50.00 s	1.386-02	8.080-01	3.400-01	-
¹⁰³ Ru	39.80 d	2.015-07	6.730-02	5.260-01	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
^{103m}Rh	55.00 m	2.100-04	-	3.960-02	-
^{104}Mo	1.30 m	8.885-03	4.820-01	1.300+00	-
^{104}Tc	18.00 m	6.417-04	1.197+00	1.437+00	-
^{105}Mo	41.00 s	1.690-02	1.158+00?	2.750+00?	-
^{105}Tc	7.80 m	1.481-03	6.623-01	1.700+00	-
^{105}Ru	4.44 h	4.336-05	4.090-01	7.869-01	-
^{105}Rh	35.50 h	5.422-06	1.520-01	7.870-02	-
^{106}Tc	37.00 s	1.873-02	2.848+00	6.570-01	-
^{106}Ru	368.00 d	2.180-08	1.000-02	-	-
^{106}Rh	30.00 s	2.310-02	1.414+00	2.045-01	-
^{107}Ru	4.20 m	2.750-03	1.254+00	2.405-01	-
^{107}Rh	22.40 m	5.156-04	4.400-01	3.126-01	-
^{110m}Ag	253.00 d	3.170-08	6.800-02	2.790+00	-
^{119m}Sn	245.00 d	3.274-08	-	8.890-02	-
^{123}Sn	129.00 d	6.218-08	5.160-01	6.800-03	-
^{125}Sn	9.65 d	8.312-07	8.020-01	3.117-01	-
^{125}Sb	2.73 y	8.044-09	8.120-02	4.548-01	-
^{125m}Te	58.00 d	1.383-07	-	1.448-01	-
^{127m}Sn	4.20 m	2.750-03	1.091+00	4.900-01	-
^{127}Sn	2.12 h	9.080-05	7.830-01	1.300+00	-
^{127}Sb	3.80 d	2.111-06	3.150-01	6.621-01	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
^{127m}Te	109.00 d	7.359-08	6.100-03	8.800-02	-
^{127}Te	9.30 h	2.070-05	2.390-01	4.800-03	-
^{128}Sn	59.00 m	1.958-04	2.705-01	5.000-01	-
^{128m}Sb	10.40 m	1.111-03	1.000+00	1.986+00	-
^{128}Sb	9.00 h	2.139-05	4.200-01	3.160+00	-
^{129m}Sn	2.00 m	5.775-03	8.014-01?	2.000+00?	-
^{129}Sn	7.50 m	1.540-03	8.014-01?	2.000+00?	-
^{129}Sb	4.34 h	4.436-05	3.510-01	1.237+00	-
^{129m}Te	34.10 d	2.352-07	2.060-01	9.630-02	-
^{129}Te	69.00 m	1.674-04	5.170-01	8.590-02	-
^{130}Sn	2.60 m	4.442-03	3.527-01	1.000+00	-
^{130m}Sb	8.00 m	1.444-03	1.261+00	2.242+00	-
^{130}Sb	37.00 m	3.122-04	1.112+00	2.490+00	-
^{131}Sn	1.30 m	8.885-03	1.038+00	2.500+00	-
^{131}Sb	23.00 m	5.022-04	6.852-01	1.665+00	-
^{131m}Te	30.00 h	6.417-06	1.860-01	1.340+00	-
^{131}Te	25.00 m	4.620-04	6.730-01	4.489-01	-
^{131}I	8.07 d	9.945-07	1.838-01	3.847-01	-
^{131m}Xe	11.96 d	6.706-07	-	1.639-01	-
^{132}Sn	1.00 m	1.155-02	1.038+00	2.500+00	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
^{132m} Sb	45.00 s	1.540-02	1.182+00?	2.800+00?	-
¹³² Sb	3.10 m	3.726-03	1.182+00	2.800+00	-
¹³² Te	78.00 h	2.468-06	6.270-02	2.780-01	-
¹³² I	2.28 h	8.428-05	4.780-01	2.280+00	-
¹³³ Sb	2.70 m	4.278-03	5.360-01	1.947+00	-
^{133m} Te	53.00 m	2.179-04	4.600-01	1.592+00	-
¹³³ Te	12.50 m	9.240-04	7.980-01	9.674-01	-
¹³³ I	20.80 h	9.255-06	4.060-01	6.240-01	-
^{133m} Xe	2.26 d	3.549-06	-	2.328-01	-
¹³³ Xe	5.27 d	1.522-06	1.005-01	8.200-02	-
¹³⁴ Te	43.00 m	2.686-04	3.070-01	4.955-01	-
¹³⁴ I	52.30 m	2.208-04	6.370-01	2.575+00	-
¹³⁴ Cs	2.06 y	1.066-08	1.630-01	1.570+00	-
¹³⁵ I	6.70 h	2.873-05	3.740-01	1.593+00	-
^{135m} Xe	15.70 m	7.357-04	-	5.263-01	-
¹³⁵ Xe	9.16 h	2.102-05	3.030-01	2.616-01	-
¹³⁶ Te	33.00 s	2.100-02	9.121-01?	2.235+00?	-
¹³⁶ I	85.00 s	8.153-03	1.808+00	2.070+00	-
¹³⁶ Cs	13.00 d	6.169-07	1.066-01	2.185+00	-
¹³⁷ I	22.30 s	3.108-02	1.228+00?	2.895+00?	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
¹³⁷ Xe	3.82 m	3.024-03	1.813+00	1.876-01	-
¹³⁷ Cs	30.20 y	7.272-10	1.895-01	-	-
^{137m} Ba	2.55 m	4.528-03	-	6.615-01	-
¹³⁸ Xe	14.20 m	8.134-04	4.570-01	2.184+00	-
¹³⁸ Cs	32.20 m	3.587-04	1.143+00	2.310+00	-
¹³⁹ Xe	40.00 s	1.733-02	1.863+00	5.051-01	-
¹³⁹ Cs	9.30 m	1.242-03	1.600+00	4.150-01	-
¹³⁹ Ba	83.20 m	1.388-04	8.950-01	5.080-02	-
¹⁴⁰ Cs	63.80 s	1.086-02	2.207+00	1.964+00	-
¹⁴⁰ Ba	12.80 d	6.266-07	2.730-01	2.190-01	-
¹⁴⁰ La	40.23 h	4.785-06	5.360-01	2.311+00	-
¹⁴¹ Cs	24.70 s	2.806-02	1.031+00?	2.485+00?	-
¹⁴¹ Ba	18.30 m	6.312-04	1.038+00	5.487-01	-
¹⁴¹ La	3.87 h	4.974-05	9.530-01	2.750-02	-
¹⁴¹ Ce	32.53 d	2.466-07	1.458-01	1.010-01	-
¹⁴² Ba	10.70 m	1.079-03	2.880-01	1.039+00	-
¹⁴² La	92.40 m	1.250-04	8.772-01	2.407+00	-
¹⁴³ La	14.00 m	8.250-04	6.577-01	1.690+00	-
¹⁴³ Ce	33.00 h	5.833-06	4.060-01	2.930-01	-
¹⁴³ Pr	13.58 d	5.906-07	3.130-01	-	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
¹⁴⁴ La	41.00 s	1.690-02	1.161+00 ?	2.750+00 ?	-
¹⁴⁴ Ce	284.40 d	2.820-08	8.180-02	2.820-02	-
¹⁴⁴ Pr	17.30 m	6.676-04	1.204+00	3.130-02	-
¹⁴⁵ Ce	2.90 m	3.983-03	6.210-01	8.290-01	-
¹⁴⁵ Pr	5.98 h	3.219-05	6.770-01	1.360-02	-
¹⁴⁶ Ce	14.20 m	8.134-04	2.180-01	4.059-01	-
¹⁴⁶ Pr	24.20 m	4.773-04	1.199+00	1.149+00	-
¹⁴⁷ Ce	70.00 s	9.900-03	6.394-01	1.650+00	-
¹⁴⁷ Pr	12.00 m	9.625-04	6.700-01	9.300-01	-
¹⁴⁷ Nd	11.06 d	7.252-07	2.350-01	1.717-01	-
¹⁴⁷ Pm	2.62 y	8.371-09	6.220-02	-	-
¹⁴⁸ Ce	43.00 s	1.612-02	3.111-01 ?	9.000-01 ?	-
¹⁴⁸ Pr	2.00 m	5.775-03	1.866+00	3.900-01	-
^{148m} Pm	41.50 d	1.933-07	1.408-01	2.031+00	-
¹⁴⁸ Pm	5.40 d	1.485-06	7.030-01	6.286-01	-
¹⁴⁹ Pr	2.30 m	5.022-03	1.158+00	2.303-01	-
¹⁴⁹ Nd	1.73 h	1.113-04	4.500-01	4.110-01	-
¹⁴⁹ Pm	53.10 h	3.625-06	3.630-01	1.390-02	-
¹⁵¹ Nd	12.40 m	9.315-04	7.060-01	7.662-01	-
¹⁵¹ Pm	28.40 h	6.778-06	2.650-01	3.909-01	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
¹⁵¹ Sm	93.00 y	2.361-10	1.950-02	4.000-04	-
¹⁵² Nd	11.70 m	9.872-04	1.964-01?	6.100-01?	-
¹⁵² Pm	4.20 y	2.750-03	1.364+00	2.839-01	-
¹⁵² Eu	14.00 y	1.569-09	8.900-02	1.588+00	-
¹⁵³ Pm	5.50 m	2.100-03	6.420-01	8.090-02	-
¹⁵³ Sm	46.80 h	4.113-06	2.230-01	1.040-01	-
¹⁵⁴ Eu	7.80 y	2.815-09	2.470-01	1.245+00	-
¹⁵⁵ Eu	5.00 y	4.392-09	4.470-02	8.150-02	-
²³¹ Th	25.52 h	7.543-06	8.030+00	1.072-01	-
²³² Pa	1.31 d	6.123-06	9.766-02	1.020+00	-
²³² U	72.00 y	3.050-10	-	2.000-02	5.390+00
²³³ Th	22.30 m	5.179-04	4.473-01	3.139-02	-
²³³ Pa	27.00 d	2.971-07	6.336-02	3.499-01	-
²³³ U	1.59+05 y	1.390-13	-	8.938-03	4.900+00
²³⁴ Th	24.10 d	3.328-07	4.453-02	3.043-02	-
²³⁴ Pa	6.70 h	2.873-05	2.042-01	1.591+00	-
²³⁷ U	6.75 d	1.188-06	6.779-02	2.780-01	-
²³⁸ Np	2.12 d	3.789-06	2.298-01	6.370-01	-
²³⁸ Pu	87.75 y	2.503-10	-	1.253-02	5.580+00
²³⁹ U	23.54 m	4.907-04	4.500-01	6.000-02	-

TABLE 1. HALF-LIVES AND ENERGIES OF RADIOACTIVE NUCLIDES IN HTGR FUEL (Cont'd)

Nuclide	Half-Life	Decay Const. (1/sec)	Beta-Ray Average Energy (MeV)	Gamma-Ray Energy (MeV)	Alpha-Ray Energy (MeV)
^{239}Np	2.35 d	3.406-06	1.150-01	3.278-01	-
^{240}Np	65.00 m	1.777-04	2.948-01	1.310+00	-
^{241}Pu	14.80 y	1.484-09	5.244-03	-	-
^{241}Am	433.00 y	5.072-11	-	6.343-02	5.588+00
^{242}Am	16.02 h	1.202-05	1.737-01	4.552-02	-
^{242}Cm	163.00 d	4.921-08	-	1.076-02	6.206+00
^{243}Pu	4.95 h	3.885-05	1.781-01	3.283-02	-
^{244}Am	10.10 h	1.906-05	1.159-01	1.043+00	-
^{244}Cm	18.00 y	1.220-09	-	8.562-03	5.891+00

TABLE 2. CALCULATED AFTERHEAT RATES (P/P_0 in %) FOR FOUR-YEAR FUEL OPERATION⁺

Time After Shutdown	Afterheat Rates for Fission Fragments and Delayed Neutrons			Afterheat Rates for Neutron-Capture Products*	Afterheat Rates for Actinide Nuclei	Total Afterheat Rates for HTGR
	²³³ U	²³⁵ U	Higher Value for ²³³ U, ²³⁵ U			
10 sec	4.26%**	5.10%**	5.10%**	0.014%	0.45%	5.56%**
100 sec	2.99%**	3.20%**	3.20%**	0.014	0.45	3.66%**
1,000 sec	1.81	1.83	1.83	0.014	0.37	2.21
10,000 sec	.879	.871	0.879	0.013	0.242	1.13
1 day	.420	.430	0.430	0.013	0.211	0.654
10 days	.225	.228	0.228	0.010	0.141	0.379
100 days	.067	.071	0.071	0.006	0.015	0.092
1,000 days	.0076	.0071	0.0076	0.0027	0.0016	0.0119

⁺See Section 2.1 for operating time for actinide nuclei.

*Only nuclei produced from neutron activation of fission fragments are included in this column.

**Because of the neglect of short-lived fission fragments, very roughly a value of 0.06% should be added to those given in the table at 100 sec. The correction at 10 sec should be significantly larger.

TABLE 3. CALCULATED AFTERHEAT RATES (P/P_0 in %) FOR VARIOUS CONTRIBUTING RADIATIONS AND FOR FOUR-YEAR FUEL OPERATION⁺

Time After Shutdown	Delayed Neutrons	Beta Rays	Gamma Rays	Alpha Rays	Total
10 sec	0.89%	1.96%	2.71%	nil	5.56%*
100 sec	0.06	1.55	2.05	nil	3.66*
1,000 sec	0.000	0.93	1.28	nil	2.21
10,000 sec	0.000	0.443	0.690	nil	1.13
1 day	0.000	0.218	0.434	0.002	0.654
10 days	0.000	0.124	0.253	0.002	0.379
100 days	0.000	0.043	0.047	0.002	0.092
1,000 days	0.000	0.0060	0.0043	0.0016	0.0119

⁺See Section 2.1 for operating time of actinide nuclei.

*Because of the neglect of short-lived fission fragments, very roughly a value of 0.06% should be added to that given in the table at 100 sec. The correction at 10 sec should be significantly larger.

TABLE 4. PERCENTAGE UNCERTAINTIES IN AFTERHEAT RATES

Time After Shutdown	Neglect of Fission Fragments with Low Yields	Neglect of Fission Fragments with Short Half-Lives	Uncertainties on Fission-Fragment Yields	Uncertainties in Energies of Nuclei with Unknown Decay Schemes	Uncertainties in Energies of Nuclei with Known Decay Schemes	Uncertainties in Calculation of Average Energy of Beta Rays	Uncertainties in Half-Lives	Neglect of Some Neutron-Capture Nuclei	Uncertainties in Actinide-Nuclide Inventories	Overall Uncertainty
100 sec	2	6	5	3.5	7	4	6	3	1.4	14
1,000 sec	2	0	3.8	0.8	6	4	4	3	2	10
10,000 sec	2	0	3.5	0.3	6	4	3	4	3	10
1 day	2	0	1.5	0.3	4	4	3	6	4.5	10
10 days	2	0	1.5	0.0	4	4	3	7	3	10

TABLE 5. GAMMA DECAY SPECTRA FOR FOUR-YEAR OPERATION OF FUEL ELEMENT

Photon Energy Range (MeV)	10 Seconds After Shutdown		1 Day After Shutdown		10 Days After Shutdown		100 Days After Shutdown	
	MeV/W-sec for Gamma Rays	Major Contributors*	MeV/W-sec for Gamma Rays	Major Contributors	MeV/W-sec for Gamma Rays	Major Contributors	MeV/W-sec for Gamma Rays	Major Contributors
0-0.2	$4.1 + 9^{**}$	^{99m}Tc , ^{103}Tc , ^{141}Ba , ^{233}Th , ^{233}Pa , ^{234}Pa	$1.55 + 9$	^{233}Pa	$9.4 + 8$	^{233}Pa	$1.10 + 8$	^{233}Pa
0.2-0.45	$1.47 + 10$	^{135}I , ^{139}Xe , ^{141}Ba , ^{144}La , ^{233}Pa	$6.8 + 9$	^{233}Pa	$4.7 + 9$	^{233}Pa	$4.3 + 8$	^{233}Pa
0.45-0.7	$1.69 + 10$	Not Determined	$3.4 + 9$	^{103}Ru , ^{132}I , ^{133}I , ^{140}La	$1.35 + 9$	^{103}Ru , ^{140}La	$3.4 + 8$	^{134}Cs , ^{137m}Ba
0.7-0.9	$2.23 + 10$	^{93}Sr , ^{95}Zr , ^{95}Nb , ^{97m}Nb , ^{98}Nb , ^{134}I	$5.7 + 9$	^{95}Zr , ^{95}Nb	$3.7 + 9$	^{95}Zr , ^{95}Nb	$1.62 + 9$	^{95}Zr , ^{95}Nb
0.9-1.125	$1.45 + 10$	Not Determined	$1.02 + 9$	^{132}I , ^{140}La , ^{238}Np	$2.48 + 8$	^{140}La	$5.9 + 6$	^{134}Cs , ^{149m}Pm
1.125-1.425	$1.67 + 10$	^{92}Sr , ^{94}Sr , ^{135}I , ^{136}I , ^{143}La	$5.8 + 8$	^{132}I , ^{135}I	$8.4 + 7$	^{132}I , ^{134}Cs	$1.79 + 7$	^{134}Cs
1.425-1.8	$1.56 + 10$	^{135}I , ^{138}Cs , ^{140}La , ^{143}La	$3.2 + 9$	^{140}La	$2.05 + 9$	^{140}La	$2.09 + 7$	^{140}La
1.8-2.2	$8.4 + 9$	^{88}Kr , ^{89}Pb , ^{91}Rb , ^{138}Xe , ^{142}La , ^{143}La	$1.26 + 8$	^{132}I	$3.4 + 7$	^{144}Pr	$1.96 + 7$	^{144}Pr
2.2-2.65	$1.03 + 10$	^{88}Kr , ^{138}Cs , ^{142}La , ^{143}La	$2.34 + 8$	^{140}La	$1.45 + 8$	^{140}La	$1.10 + 6$	^{140}La
2.65-3.25	$4.5 + 9$	^{133}Sb , ^{140}Cs , ^{142}La , ^{143}La	$6.0 + 6$	^{140}La	$3.8 + 6$	^{140}La	$2.9 + 4$	^{140}La
73.25	$7.5 + 9$	^{90}Rb , ^{91}Rb	$2.91 + 5$	^{140}La	$1.33 + 5$	^{140}La	$1.02 + 3$	^{140}La

* $\geq 50\%$ of intensity, ≤ 6 different isotopes** $4.1 + 9$ means 4.1×10^9

TABLE 6. GAMMA-RAY DOSE RATE FROM A FUEL ELEMENT VERSUS
TIME AFTER SHUTDOWN

Time After Shutdown	Dose Rate (rad/hr) at 1 Meter From Surface of Fuel Element
10 sec	2.7×10^5
100 sec	2.3×10^5
1,000 sec	1.5×10^5
10,000 sec	7.2×10^4
1 day	4.0×10^4
10 days	2.3×10^4
100 days	4.9×10^3
1,000 days	4.7×10^2

FIGURE 1. NEW AND PREVIOUS AFTERHEAT RESULTS

FIGURE 2. GAMMA-RAY DECAY SPECTRA FOR FOUR-YEAR OPERATION

ACKNOWLEDGEMENTS

The author gratefully acknowledges D. E. Hurd for programming some of the computations and for performing many of the calculations for this report, B. A. Engholm for helpful advice, C. J. Hamilton for calculations and helpful discussions concerning the activities of the actinide nuclei, and M. J. Haire for calculations and advise concerning the activities of the fission fragments. Also, other people at GGA as well as some other laboratories were very helpful in discussions.

REFERENCES

1. Engholm, B. A., Gulf General Atomic, "Shutdown Heat Generation in the HTGR," unpublished data.
2. Shure, K., "Fission Product Decay Energy," Bettis Technical Review, WAPD-BT-24, Westinghouse Electrical Corporation, December, 1961.
3. England, T. R., "An Investigation of Fission Product Behavior and Decay Heating in Nuclear Reactors," Ph. D. Thesis, University of Wisconsin, August, 1969.
4. Shure, K., "²³⁵U Fission Product Decay Energy, 1972 Re-Evaluation," Bettis Atomic Power Laboratory, WAPD-TM-1119, Westinghouse Electric Corporation, October, 1972.
5. Bell, M. J., "Calculated Radiation Properties of Spent Plutonium Fuels," Nuclear Technology 18, 5 (1973).
6. ANS Standard ANS-5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors (Draft)," American Nuclear Society, Hinsdale, Illinois, October, 1971.
7. Vanslager, F. E., "RAD 2, A Computer Program for Calculating Fission Product Radioactivities," GAMD-6519, July 23, 1965.
8. Meek, M. E., and B. F. Rider, "Compilation of Fission Product Yields," Vallecitos Nuclear Center, NEDO-12154, 71 NED 41, January, 1972.
9. Chart of the Nuclides, 10th Edition, 1969, U.S.A.E.C.
10. Todt, F. W., "GARGOYLE, An Infinite Medium Fuel Cycle Analysis Code with Fuel and Poison Searches," U.S.A.E.C. Report, GA-6622, General Atomic Division, General Dynamics Corporation, 1966.
11. Tobias, A., "Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products (Revision 2)," Central Electricity Generating Board, Berkeley Nuclear Laboratories, RD/B/M2453, FREDSWG/P (72)12, October, 1972.
12. Widman, J. C., J. Mantel, N. H. Horwitz, and E. R. Powsner, Int. J. Appl. Rad. and Isot. 19, 1 (1968).
13. Dillman, L. T., J. Nuc. Medicine 10, Supplement 2, pamphlet 4, p. 7, March, 1969.

14. Blachot, Jean, and Roland de Tourriel, "Bibliotheque de Donnees Nuclearies Relatives Aux Produits de Fission," (3rd edition) CEA-N-1526, March, 1972.
15. Nuclear Data Sheets from 1969-1972, published by Academic Press, Inc., New York.
16. Keepin, G. R., Physics of Nuclear Kinetics, Addison-Wesley Publishing Co., Inc., Reading, Massachusetts (1965).
17. Clark, S. S., and B. A. Engholm, "PATH - A Highly Flexible, General Purpose Shielding Program," GA-9908, December 10, 1969.

APPENDIX 1. NUMBER OF CURIES PER ELEMENT AT 100 SEC AFTER SHUTDOWN

A. VALUES FOR ^{235}U FISSION FRAGMENTS AND NEUTRON-CAPTURE PRODUCTS

^{78}As	1.38+02*	$^{90\text{m}}\text{Rb}$	1.94+03	^{98}Zr	4.09+03
^{79}Ge	1.18+02	^{90}Rb	3.15+04	^{98}Nb	4.49+03
$^{79\text{m}}\text{Se}$	4.16+02	^{90}Sr	2.99+03	$^{99\text{m}}\text{Nb}$	8.22+03
^{81}Se	1.29+03	^{90}Y	2.99+03	^{99}Mo	4.17+04
$^{83\text{m}}\text{Se}$	9.48+02	^{91}Rb	1.26+04	$^{99\text{m}}\text{Tc}$	3.68+04
^{83}Se	1.35+03	^{91}Sr	3.90+04	$^{100\text{m}}\text{Nb}$	1.93+04
^{83}Br	3.63+03	$^{91\text{m}}\text{Y}$	2.39+04	^{100}Nb	1.42+04
$^{83\text{m}}\text{Kr}$	3.72+03	^{91}Y	3.94+04	^{101}Mo	3.21+04
^{84}Se	4.43+03	^{92}Sr	4.04+04	^{101}Tc	3.47+04
^{84}Br	6.33+03	^{92}Y	4.07+04	^{102}Mo	2.59+04
^{85}Se	1.29+03	^{93}Sr	3.69+04	$^{103\text{m}}\text{Mo}$	2.98+03
^{85}Br	6.55+03	^{93}Y	4.37+04	^{103}Mo	7.77+03
$^{85\text{m}}\text{Kr}$	9.02+03	^{94}Sr	1.67+04	^{103}Tc	1.84+04
^{85}Kr	3.42+02	^{94}Y	4.27+04	^{103}Ru	2.12+04
^{86}Br	4.59+03	^{95}Sr	2.67+03	$^{103\text{m}}\text{Rh}$	2.09+04
^{87}Br	4.35+03	^{95}Y	4.09+04	^{104}Mo	4.94+03
^{87}Kr	1.60+04	^{95}Zr	4.43+04	^{104}Tc	1.22+04
^{88}Kr	2.46+04	$^{95\text{m}}\text{Nb}$	5.78+02	^{105}Mo	9.49+02
^{88}Rb	2.49+04	^{95}Nb	4.43+04	^{105}Tc	5.23+03
^{89}Kr	2.20+04	^{96}Y	2.45+04	^{105}Ru	6.44+03
^{89}Rb	3.23+04	^{97}Zr	4.04+04	^{105}Rh	5.47+03
^{89}Sr	3.28+04	$^{97\text{m}}\text{Nb}$	3.76+04	^{106}Tc	4.05+02
^{90}Kr	3.85+03	^{97}Nb	4.00+04	^{106}Ru	2.37+03

*1.38+02 means 1.38×10^2

^{106}Rh 2.37+03	^{132}Sb 1.43+04	^{141}Ce 4.00+04
^{107}Ru 9.89+02	^{132}Te 2.80+04	^{142}Ba 3.26+04
^{107}Rh 1.29+03	^{132}I 2.81+04	^{142}La 3.69+04
$^{110\text{m}}\text{Ag}$ 5.26-01	^{133}Sb 1.44+04	^{143}La 3.65+04
$^{119\text{m}}\text{Sn}$ 1.28+00	$^{133\text{m}}\text{Te}$ 2.82+04	^{143}Ce 4.02+04
^{123}Sn 9.53+01	^{133}Te 1.78+04	^{143}Pr 4.04+04
^{125}Sn 4.41+01	^{133}I 4.60+04	^{144}La 6.29+03
^{125}Sb 9.59+01	$^{133\text{m}}\text{Xe}$ 1.28+03	^{144}Ce 3.33+04
$^{125\text{m}}\text{Te}$ 2.00+01	^{133}Xe 4.60+04	^{144}Pr 3.33+04
$^{127\text{m}}\text{Sn}$ 4.25+02	^{134}Te 4.23+04	^{145}Ce 1.81+04
^{127}Sn 1.73+02	^{134}I 4.88+04	^{145}Pr 2.69+04
^{127}Sb 7.84+02	^{134}Cs 5.39+03	^{146}Ce 1.84+04
$^{127\text{m}}\text{Te}$ 2.96+02	^{135}I 4.34+04	^{146}Pr 2.04+04
^{127}Te 9.44+02	$^{135\text{m}}\text{Xe}$ 7.10+03	^{147}Ce 5.09+03
^{128}Sn 2.13+03	^{135}Xe 5.38+03	^{147}Pr 1.47+04
$^{128\text{m}}\text{Sb}$ 2.34+03	^{136}Te 6.03+02	^{147}Nd 1.54+04
^{128}Sb 3.40+02	^{136}I 1.15+04	^{147}Pm 6.28+03
$^{129\text{m}}\text{Sn}$ 1.46+03	^{136}Cs 2.73+03	^{148}Ce 1.50+03
^{129}Sn 1.05+03	^{137}I 1.17+03	^{148}Pr 7.91+03
^{129}Sb 4.40+03	^{137}Xe 3.19+04	$^{148\text{m}}\text{Pm}$ 4.84+02
$^{129\text{m}}\text{Te}$ 6.98+02	^{137}Cs 3.00+03	^{148}Pm 2.05+03
^{129}Te 4.16+03	$^{137\text{m}}\text{Ba}$ 2.79+03	^{149}Pr 4.08+03
^{130}Sn 4.72+03	^{138}Xe 3.92+04	^{149}Nd 7.27+03
$^{130\text{m}}\text{Sb}$ 2.50+03	^{138}Cs 4.58+04	^{149}Pm 7.17+03
^{130}Sb 1.02+04	^{139}Xe 5.98+03	^{151}Nd 2.51+03
^{131}Sn 2.75+03	^{139}Cs 4.10+04	^{151}Pm 2.88+03
^{131}Sb 1.48+04	^{139}Ba 4.46+04	^{151}Sm 1.00+00
$^{131\text{m}}\text{Te}$ 2.21+03	^{140}Cs 1.58+04	^{152}Nd 1.33+03
^{131}Te 1.46+04	^{140}Ba 4.30+04	^{152}Pm 1.70+03
^{131}I 1.88+04	^{140}La 4.31+04	^{152}Eu 4.00-02
$^{131\text{m}}\text{Xe}$ 1.13+02	^{141}Cs 1.80+03	^{153}Pm 8.07+02
^{132}Sn 2.03+03	^{141}Ba 3.79+04	^{153}Sm 6.31+02
$^{132\text{m}}\text{Sb}$ 3.72+03	^{141}La 4.00+04	^{154}Eu 1.86+01
		^{155}Eu 6.35+00

APPENDIX 1. NUMBER OF CURIES PER ELEMENT AT 100 SEC AFTER SHUTDOWN

B. VALUES FOR ACTINIDE NUCLIDE

^{231}Th 1.41+02
 ^{232}Pa 1.52+02
 ^{232}U 2.56+00
 ^{233}Th 5.705+05
 ^{233}Pa 5.575+05
 ^{233}U 1.78+00
 ^{234}Th 4.81+01
 ^{234}Pa 1.846+04
 ^{237}U 6.014+04
 ^{238}Np 5.18+04
 ^{238}Pu 3.85+02
 ^{239}U 9.012+03
 ^{239}Np 1.035+04
 ^{240}Np 1.28+01
 ^{241}Pu 6.78+01
 ^{241}Am 2.93-02
 ^{242}Am 4.61+01
 ^{242}Cm 3.77+01
 ^{243}Pu 4.74+02
 ^{244}Am 2.03+02
 ^{244}Cm 6.78+00