Data Compilation and Analysis of Fission Product Decay Heat Experiments

Summary report for special agreement

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Report prepared for the Project on Compilation of Nuclear Data Experiments for Radiation Characterisation (CoNDERC)

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Abstract

A project entitled Compilation of Nuclear Data Experiments for Radiation Characterisation (CoNDERC) has been formed with the purpose of developing a database of integral experimental measurement data that can be used in the Validation and Verification processes of nuclear data files and code systems used for radiation characterization. The project includes compiling experimental data, developing code models and input data used in the analysis of these experimental data to facilitate transfer and application of this information to existing and next-generation nuclear technologies and nuclear data validation. An important integral measure of the radiation emission process is the energy release rate (decay heat power) following nuclear fission that results from the formation and decay of fission products. The selected experiments compiled and analysed in this report include energy release measurements following ^{235}U, ^{233}U, and ^{239}Pu fission. In several experiments the contributions of gamma and beta energy release are reported separately, providing a more rigorous benchmark for nuclear decay data testing.

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1 Introduction

As part of the project on Compilation of Nuclear Data Experiments for Radiation Characterisation (CoNDERC), experimental data on the energy release rate from the decay of fission products following fission have been compiled for use in the Validation and Verification (V&V) processes for nuclear data and to provide standardized schema to perform the computer code V&V at the engineered level. Under this project, measurement data are being captured in a database to facilitate transfer of the information for nuclear technology applications. In addition, benchmark descriptions have been developed and computational modelling and simulation of the experiments performed comparing predictions and measurements using modern nuclear databases.

Experimental data on energy release rates following fission (decay heat power) represent and important integral benchmark quantity for computational systems and nuclear data. These experiments involve measurements of the radiation energy released by nuclear decay processes of fission products formed by fission. The measurements therefore exercise both the fission product yields and the nuclear decay data. The decay heat experiments considered under this project to date include fission of ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu, ^{237}Np, and ^{232}Th. Measurements include both thermal and fast neutron induced fission. Measurements use several different detector designs including a standard thermal calorimeter, a total absorption (beta and gamma radiation) detector (so-called nuclear calorimeter), and beta and gamma spectroscopy. In addition to the total decay heat power, several experiments include measurements of the separate beta and gamma contributions of decay heat providing more rigorous benchmark of nuclear decay data. Several spectroscopic experiments have also reported the measured energy distributions (spectra) of the emitted beta and gamma radiations.

2 Experimental data

Measurements of decay energy release rates following fission derive primary from requirements and applications to reactor safety and loss of coolant accident analysis. Consequently, most of the measured data is obtained from relatively short irradiation of small samples and are limited to cooling times less than about 48 hours after fission. Notably, several experiments involving longer reactor irradiations with larger fuel samples provide measurements for much longer cooling times out to several months. The wide range of irradiation times and cooling times of the measurements provides an extensive database for nuclear data validation.

The experiments compiled and analysed in this report have been widely used in the development of national [1] and international [2] decay heat standards for light water reactors. A complete list of experiments (not all included in the present report) being considered in the CoNDERC project are listed below.

Summary of decay heat experiments widely used in V&V studies

Institute	Year	First Auth	Publication	Measured nuclide(s)
Bettis	1975	Gunst	NSE 56 241	^{233}U, ^{235}U, ^{239}Pu, ^{232}Th
Bettis	1979	Shure	NSE 71 327	^{235}U, ^{239}Pu
CEA	1976	Fiche	NEACRP-L-212	^{239}Pu, ^{233}U
CEN	1973	Lott	Nucl. En. 27	^{235}U
IRT (EPRI)	1976	Friesenhahn	EPRI NP-180	^{235}U
UC Berkeley (EPRI)	1978	Schrock	EPRI NP-616	^{235}U
IRT (EPRI)	1979	Friesenhahn	EPRI NP-998	^{235}U, ^{239}Pu
Karlsruhe	1981	Baumung	KfK 3262	^{235}U
Los Alamos	1978	Yarnell	LA-7452-MS	^{239}Pu, ^{233}U, ^{235}U
Los Alamos	1977	Yarnell	LA-NUREG-6713	^{235}U
Oak Ridge	1978	Dickens	ORNL/NUREG-39	^{235}U
Oak Ridge	1978	Dickens	ORNL/NUREG-34	^{239}Pu
Oak Ridge	1978	Dickens	ORNL/NUREG-47	^{241}Pu
Uppsala	1987	Johansson	NEACRP-L-302	^{235}U, ^{239}Pu, ^{238}U
UK AWRE	1965	Johnston	Nucl. En. 19	^{239}Pu
Massachusetts Lowell	1993	Schier	DOE/ER/40723-1	^{235}U, ^{239}Pu, ^{238}U
Massachusetts Lowell	1993	Schier	DOE/ER/40723-2	^{235}U, ^{239}Pu, ^{238}U
Massachusetts Lowell	1994	Schier	DOE/ER/40723-3	^{235}U, ^{239}Pu, ^{238}U
Massachusetts Lowell	1996	Schier	DOE/ER/40723-4	^{235}U, ^{239}Pu, ^{238}U
YAYOI	1982	Akiyama	AESJ 24(9) (10)	^{235}U, ^{239}Pu, ^{233}U
YAYOI	1988	Akiyama	JAERI-M 88-252	^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu
YAYOI	2001	Ohkawachi	ND 2001	^{235}U, ^{237}Np
LANL	1964	Fisher	Phys Rev B, 1959	^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu
UK-AWRE	1969	McNair	J Nucl En 23	^{235}U, ^{239}Pu
SRRC	1970	MacMahon		^{235}U
LANL	1979	Jurney	LA-7620-MS	^{239}Pu, ^{233}U, ^{235}U
UKAEA	1979	Murphy	AEEW-R 1212	^{235}U, ^{239}Pu
SRRC	1971	Scobie	J Nucl En 25	^{235}U

Analysis of a subset of these experiments is presented in this report, including the the following measurements:

1. Friesenhahn et al. (1976), that include total, beta, and gamma components of the energy release rates following ^{235}U thermal fission [3],
2. Friesenhahn et al. (1979), that include total, beta, and gamma components of the energy release rates following ^{235}U and ^{239}Pu thermal fission [4],
3. Schrock et al. (1978), that include total energy release rates following ^{235}U thermal fission [5],
4. Fiche et al. (1976), that include total energy release rates following ^{233}U and ^{239}Pu thermal fission [6],
5. Johnson (1965), that include total, beta, and gamma components of the energy release rates following ^{239}Pu fast fission [7] during irradiation in the Dounreay fast reactor, and

6. Gunst et al. (1975), that include extended irradiations for fuel rods in test reactors at Idaho National Laboratory and thermal fission of ^{233}U, ^{235}U, ^{239}Pu, and ^{232}Th [8].

3 Computer code and nuclear data

Computational models of the experiments were developed and calculations of energy release rates performed using modern nuclear data evaluations. Comparisons of calculations to the measurements are provided. The models are also made available to provide end users with examples to facilitate use of the experiments to nuclear technology applications.

In this report, simulations were performed using the ORIGEN code, developed and maintained at Oak Ridge National Laboratory as the depletion and decay module of the SCALE code system https://www.ornl.gov/scale. The version of the code used in the present study is from the SCALE release 6.1.3 (2011). However, newest version of the code (6.2.3) has the same capabilities and uses the same nuclear data. Therefore, results presented in this report will be the same as those obtained using the most recent release. ORIGEN uses the following nuclear data:

a) Nuclear decay data from the Evaluated Nuclear Data File ENDF/B-VII.1 [9].
b) Fission product yields from ENDF/B-VII.0 [10] (Note that these data derive from the work of England and Rider in 1994 [11]). The yields for ^{239}Pu fission were revised [12] in the release of ENDF/B-VII.1 but were not adopted in this study.
c) Neutron cross sections are derived primarily from ENDF/B-VII.1. For the present work however, the neutron cross sections were not important to the analysis results since neutron transmutation is insignificant for the relatively short irradiations in most of these experiments.
d) Gamma ray emission spectra are calculated from data derived from the ENDF/B-VII.1 nuclear decay data file.
e) Neutron sources and spectra are from data compiled and distributed in the SOURCES-4C code (https://rsicc.ornl.gov/codes/ccc/ccc6/ccc-661.html) developed at Los Alamos National Laboratory.

4 Description and analysis of experiments

This section provides a summary description of the experiments compiled and analysed in this report. The measurement data associated with each experiment have been captured and stored electronically in Excel files and as standard text format files. Each file is identified by the nuclide, author and year of measurement. The file identifiers are listed in Appendix A.

Each experimental dataset was modelled and simulated, and the predicted radiation energy release rates are compared to measurements in this section. Due to the large amount of data involved in these measurements, most comparisons are limited to the total energy release rate (total decay heat power) although separate beta and gamma contributions are measured in some experiments.

4.1 Friesenhahn et al. (1976)

4.1.1 Measurements

Under an experimental program supported by the Electric Power Research Institute (EPRI), fission product decay heat measurements were performed for thermal neutron fission of ^{235}U in the time range of about 1 to 10^5 seconds by the IRT Corporation [3].

Measurements were made with a large volume total absorption liquid scintillator to absorb the majority of emitted beta and gamma radiation. The light output of the scintillator was proportional to the energy deposited by the emitted radiation. Irradiations were performed using a water-moderated ^{252}Cf source with a rapid pneumatic system to transfer the irradiated sample to the detector. The ^{235}U sample irradiation time was 24 hours. Irradiated samples consisted of uranium-aluminum alloy foils with highly enriched ^{235}U (93.26%). The reported systematic uncertainty of the measured decay heat power is 2.4%, with statistical uncertainties of 2% at 1 second increasing to 4% at 10^5 seconds.

4.1.2 Analysis of experiment

The ^{235}U sample (U-Al alloy foil) irradiation was simulated using the following reported initial uranium isotopic composition: ^{234}U (1.113%), ^{235}U (93.26%), ^{236}U (0.259%), ^{238}U (5.37%).

The computational model used an initial uranium sample mass of 1 gram. Since results scale linearly with sample mass the actual sample mass was not required. The decay heat results are reported in units of MeV/fission (MeV/s per fission/s). Therefore, a constant fission rate was required in the simulations, with results being normalize to 1 fission/s. Since the ORIGEN code prints decay heat in units of Watts (1 J = 6.242 x 10^{12} MeV), the neutron flux in the calculations was adjusted to induce a fission rate of 6.242 x 10^{12} fission/s such that the printed decay heat results have units of MeV/s. Calculations were performed for each of 86 measured cooling times that extended to 151,276 seconds (42 hours) after fission. The total and gamma contribution to decay heat were calculated. The beta contribution is derived from the difference between the total and gamma contribution since the alpha decay contribution from fission products is very small. The results are listed in Table 1 and the ratio of calculated (C) to measured (M) values is presented. The results for the total, gamma, and beta contributions to decay heat are illustrated in Fig. 1, 2, and 3.

4.2 Friesenhahn et al. (1979)

4.2.1 Measurements

Following the measurements by Friesenhahn et al. in 1976, under support by EPRI, additional decay heat measurements were made following thermal neutron fission of ^{235}U and ^{239}Pu in the time range of about 1 to 10^5 seconds [4]. Irradiation time periods ranged from 1000 seconds to 35 days. Measurements were made using the same total absorption detector as used in the earlier 1976 experiments [3]. Separate contributions of beta and gamma rays to the total decay heat are reported. Irradiations were performed in a water-moderated ^{252}Cf source with a rapid pneumatic system to transfer the irradiated sample to the detector.

Irradiated samples were aluminum alloy foils. The uranium samples had the same isotopic composition as those used in the 1976 measurements (see Sect. 4.1). The reported plutonium isotopic distribution was: ^{239}Pu (98.76%), ^{240}Pu (1.2%), ^{241}Pu (0.04%).

The reported systematic uncertainties are typically 2-4%, with statistical uncertainties generally less than 2%.

4.2.2 Analysis of experiment

The ^{235}U measurements included sample irradiation times of 1000 seconds, 20,000 seconds, 24 hours, and 35 days. The ^{239}Pu measurements used irradiation times of 1000 seconds 20,000 seconds and 24 hours. A 24-hour irradiation of ^{235}U reported in 1976 was also repeated in the 1979 measurements. While these two measurements generally agree within the estimated uncertainty bands, the 1979 results are systematically lower than the 1976 measurements and do not agree as well with the present calculations. The experimental report states that the later measurements are more accurate and represent improvements in the previously reported data for ^{235}U fission.

Simulations were performed using the reported irradiation times of each sample, and a constant fission rate was maintained. Measurement results are presented as MeV/fission. The simulations used a constant fission rate of 6.242×10^{12} fissions/s (1 Joule = 6.242×10^{12} MeV) so that the printed decay heat values are in units of MeV/s to avoid having to perform the conversion from watts to MeV after the calculation. The total, gamma, and beta components of decay heat power were measured. Only the total decay heat power results are compared to calculations in this report. The results are listed in Tables 2-7, and comparison of the measured and calculated total decay heat values are compared in Figs. 4-9. The total measurement uncertainty listed in these tables is the sum of the reported systematic and random uncertainty components.

4.3 Schrock et al. (1978)

4.3.1 Measurements

Under an experimental program supported by EPRI, measurements were reported for ^{235}U thermal fission using a fast-response calorimeter by researchers at the University of California, Berkeley [5]. Irradiations were made in a thermal column of the General Electric Test Reactor at the Vallecitos Laboratory (California). The samples were 0.2-mm-diameter uranium wires enriched to 40% ^{235}U. Measurements started at cooling times of 11 seconds and extended to 10^5 seconds. Three different irradiation times of 1, 4, and 22.3 hours were used. A total of 12 irradiations (runs) were performed, of which 7 were reported (other runs were excluded due to experimental problems). After irradiation and calorimeter measurements, the samples were transferred to Berkeley were the number of fissions in each sample was determined experimentally by gamma counting.

Although the calorimeter was developed for highly accurate measurements, this potential accuracy at short cooling times was not realized due to design. The estimated relative uncertainty of the measurements was 3.4% for cooling times from about 400 to 10,000 seconds, increasing to about 23% at 11 seconds (shortest time) with measurements being systematically higher than other data, and increasing to 21% at longer times owing to the low decay heat power of the samples.

4.3.2 Analysis of experiment

Simulations of the measurements were performed using the three irradiation times of 1, 4, and 22.35 hours. Measurements of the total decay heat power are reported in units of MeV/fission (MeV/s per fission/s). Therefore, a constant fission rate of 1 fission/s over the irradiation time is required in the simulations. The neutron flux used in the calculations was adjusted to give a constant fission rate of 6.242 x 10^{12} fissions/s (1 Joule = 6.242 x 10^{12} MeV) so that the printed decay heat values (Watts) are in units of MeV/s to avoid having to perform the unit conversion to MeV/fission after the calculation. Several measurements (runs) were performed for each irradiation time. The different measurements for each irradiation are combined in the plots comparing measurements and calculations. The results for irradiation times of 1, 4, and 22.3 hours are listed in Table 8, 9 and 10, respectively, and are plotted in Figs. 10, 11, and 12.

Experimental issues were identified with the calorimeter for measurements at cooling times less than about 600 seconds that required corrections that introduced large uncertainties. The results in this range are systematically larger than other similar measurements but are in good agreement between 500 and 10,000 seconds. For cooling times larger than 10^4 seconds the accuracy of the measurements deteriorated rapidly due to the small heat output and the results were not considered significant.

4.4 Fiche et al. (1976)

4.4.1 Measurements

Decay heat measurements are reported for ^{239}Pu and ^{233}U thermal fission by the Atomic Energy Commission (CEA) in Cadarache, France [6]. Measurements include the total decay heat for cooling times between 60 and 10^5 seconds after fission. Sample irradiations were made in the graphite reflector of the Zoe (EL-1) heavy water zero-power reactor. The Pu samples were Pu-Al containing 89% ^{239}Pu. The ^{233}U samples were uranium oxide. Uncertainties are relatively large, ranging from roughly 8-15% for the ^{239}Pu measurements, and roughly 10-25% for ^{233}U. The experimental data is reported as the energy release rate (MeV/s) per fission.

4.4.2 Analysis of experiment

Measurements are presented for the decay heat power in MeV/s per fission for ^{239}Pu and ^{233}U thermal fission. Simulations were performed assuming a 1 second irradiation followed by decay to the measured cooling times. For the purpose of these calculations, the flux was adjusted in the calculations to yield 6.242 x 10^{12} fissions/s (1 Joule = 6.242 x 10^{12} MeV) so that the printed heat values (Watts) are in units of MeV/s to avoid having to perform the conversion after the calculation. The results are presented in Table 11 and illustrated in Figs. 13 and 14.

4.5 Johnson (1965)

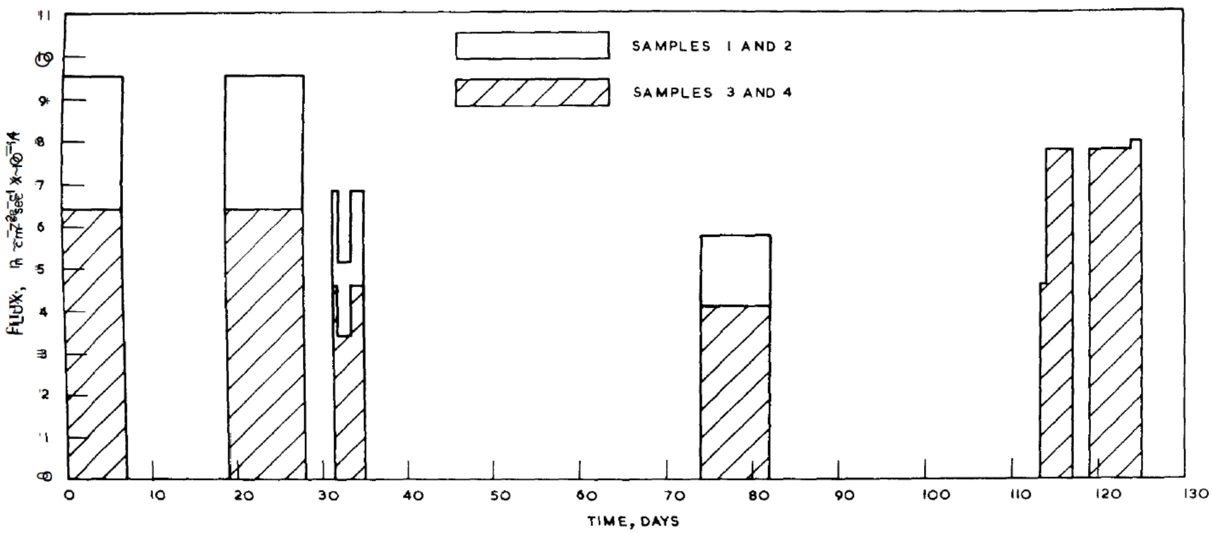
4.5.1 Measurements

Microcalorimeter measurements are reported by the United Kingdom Atom Weapons Research Establishment (UK AWRE) for Pu fuel samples irradiated in the Dounreay fast reactor for decay times from 40 to 150 days after irradiation [7]. This extended decay times make these measurements particularly valuable as there are very few measurements reported in literature in this time range. The samples consisted of 40 mg PuO_2 with a 96.38% ^{239}Pu and small (unreported) concentrations of ^{240}Pu and ^{241}Pu. Measured contributions of both gamma and beta decay heat power are determined with a multicell calorimeter using a uranium shell around the sample that absorbed a substantial fraction of the emitted γ rays, and bare measurements. The integral number of fissions is determined by radiochemical analysis of the irradiated samples with an estimated accuracy of 3%. The maximum error of the calorimeter measurements is given as less than 1%. Given the measurement error and other sources of error in the derivation of the partial contributions to decay heat, the 95% (2 σ) error in the reported beta, gamma, and total decay values are 4%, 7%, and 3% respectively. Results are reported for three samples that did not have measured fission rates. The sample that was measured for fissions however did not report decay heat results. Therefore, the number of fissions in the samples was inferred based on flux monitoring detectors at the sample positions. Consequently, some additional level of uncertainty is inherent in these data based on the core monitoring (flux level) data for each sample.

4.5.2 Analysis of experiment

Simulations were performed for the 40 mg PuO_2 samples irradiated in a fast neutron flux spectrum. The reference article states that samples were analysed chemically to determine the proportions of the minor constituent isotopes ^{240}Pu and ^{241}Pu, and the small proportion of ^{241}Am present. However, the minor constituents are not reported, and it was assumed for the purposes of the simulations that ^{239}Pu was the major contributor to fission product heat generation. The fast neutron spectrum was defined in the COUPLE code in SCALE, which is the library management code for ORIGEN, to generate the cross sections and energy-dependent fission product yields in the simulations.

Four samples were irradiated in the Dounreay reactor. Samples 1 and 2 were co-located near the center of the core and were irradiated in four irradiation stages. Samples 3 and 4 were located closer to the edge of the core in a lower neutron flux and were irradiated during the same periods as samples 1 and 2 plus two extra stages (total of six irradiation stages). The irradiation history of all samples is illustrated below.



Sample irradiation histories in the Dounreay reactor

The irradiation history was digitized to estimate the irradiation times and flux levels of the samples at each stage. The digitized data are listed in table below.

Simulation of the irradiations were performed using the flux levels and irradiation history listed in the table. For the irradiation stages 3, 5, and 6 which experienced flux changes during the stage, an average flux over the stage was used in the simulations. The neutron flux was then scaled to obtain the total number of fissions as determined by radiochemical analysis. The number of fissions was only measured for sample 1, and calorimeter measurements were reported for samples 2, 3 and 4 only. Therefore, the number of fissions in the calorimeter samples had to be inferred from the sample 1 results and the flux level in the sample positions that were determined from the reactor power rating and the computed flux distribution in the core. The computed flux values have some additional uncertainty which is not quantified. The number of fissions measured in sample 1 was 2.67 x 10^{17} fissions +/- 3%. The neutron fluence was given as 1.88 x 10^{21} n/cm^2 for samples 1 and 2 and 1.89 x 10^{21} n/cm^2 for samples 3 and 4. For the simulations, the integral number of fissions in each sample was determined from the number of fission product atoms generated in the sample, divided by two.

Irradiation history of the Dounreay reactor samples

Irradiation stage	Cumulative time (days)	Incremental time (days)	Neutron flux (× 10^{14} n/cm^2/s)	
			Samples (1,2)	Samples (3,4)
1	7.01	7.01	9.46	6.34
–	18.97	11.96	0	0
2	27.85	8.87	9.46	6.34
–	31.22	3.38	0	0
3	31.51	0.29	6.8	4.53
3	32.96	1.45	5.1	3.35
3	34.79	1.83	6.78	4.53
–	73.76	38.97	0	0
4	81.67	7.91	5.75	4.08
–	113.31	31.64		0
5	113.98	0.67		4.58
5	117.07	3.09		7.76
–	119.00	1.93		0
6	123.92	4.92		7.78
6	125.17	1.25		8.00
Total neutron fluence (n/cm^2)			1.88 x 10^{21}	1.89x10^{21}

The measurement and simulation results are compared in Table 12 for the total decay heat generated by the fission products in the samples and plotted in Figs. 15, 16, and 17. The uncertainty in the fission product decay heat values derived from calorimeter measurements is given as 3%. With the added uncertainty associated with the number of fissions in the sample of 3%, the total experimental uncertainty is conservatively estimated to be about 5%. It is emphasized that the number of fissions in the calorimeter samples were not measured directly but have been inferred from radiochemical analysis of sample 1 and the uncertainty may be underestimated. Note that for samples 3 and 4, which were irradiated in the same location and experienced a similar flux history, the decay heat values in sample 3 were systematically 4 to 7% greater than in sample 4, suggesting either larger errors in the calorimeter measurements than have been estimated, or that the samples experienced different neutron flux exposures.

A supplemental article on these measurements [13] was published which will be analysed for additional information that may improve the quality of the measurements as a benchmark.

4.6 Gunst et al. (1975)

4.6.1 Measurements

Decay heat measurements are reported for irradiated samples containing ^{235}U, ^{233}U, ^{239}Pu, and ^{232}Th over cooling times from 14 to 4500 hours after irradiation in the Materials Test Reactor at Idaho National Laboratory (INL) [8]. Measurements were made using a standard calorimeter. Irradiations were made using four-element assemblies containing Zircaloy clad metal-alloy (samples) or oxide pellets (samples) containing the measured nuclide. The calorimeter measurements were made on the entire assembly and therefore included small contributions from actinides and structural activation. The irradiation times were relatively long compared to many experiments, from about 3 week to 15 weeks. Because of the extended irradiation times, the measurements also include contributions from actinide transmutation and decay and neutron capture on fission products. Corrections to the calorimeter measurements to obtain only the fission product contribution to decay heat are made using calculations performed by the authors and are presented in the paper. A detailed uncertainty analysis is not presented, although agreement with calculations is shown to be generally better than 3% of the range of measurement data. These experiments are particularly important as they cover extended cooling times, up to 187 days, as compared to many experiments involving smaller samples and shorter irradiation times that do not extend beyond cooling times of about 2 days.

4.6.2 Analysis of experiment

There is currently insufficient documented information on the irradiation histories of the sample that were performed at the MTR and ATR reactors at Idaho National Laboratory to allow simulation of the experiments. The irradiation information is currently given only as the number of three-week cycles of exposure in the reactor, with no information on the power during each cycle or the downtime between cycles. Further, it is stated later in the paper that there was a gap of roughly a year between irradiations in the MTR and ATR reactors, which is not adequately documented.

Further information on the measurements is being sought from other Bettis Laboratory reports. Detailed design information on the irradiated fuel rods would also be valuable for modelling to derive accurate cross sections for the purposes of transmutation simulations of the heavy element and structural activation components of decay heat (which are provided in the paper based on calculations performed at the time). Additional information may be available in report WAPD-TM-1183 [14].

5 Conclusions and recommendations

Six fission product decay heat experiments have been analysed and the measurement data compiled in electronic format for use in developing a benchmark database as part of the IAEA project on Compilation of Nuclear Data Experiments for Radiation Characterisation (CoNDERC). Input models have been developed and results obtained with the ORIGEN computer code are compared to the measurements.

Comparisons of calculations and the total decay heat measurements by Freisenhahn et al. [3,4], Schrock et al. [5], and Fiche [6], are in good agreement, whereas comparisons with individual gamma and beta decay heat components [3] show larger deviations. The results are generally consistent with earlier calculations.

The experiments by Johnson [7] and Gunst et al. [8] involve more complex reactor irradiations and include measurements at much longer cooling times compared to other experiments, and therefore exercise a different range of nuclear data. The current calculated results for the Johnson experiments (Dounraey fast reactor) are systematically lower than measurement, but additional information is being analysed that may improve this experiment as a benchmark [13]. There is currently insufficient information on the irradiation history of the Gunst experiments [8], and additional information is being pursued [14]. Because these experiments cover extended cooling times not available in other experiments, it is recommended that both the Johnson and Gunst experiments be included in the V&V decay heat database provided the current deficiencies in the experimental descriptions can be resolved.

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Table 1. Measured and calculated ^{235}U decay heat power following a 24-hour irradiation (Friesenhahn et al. 1976)

Cooling	Measured decay heat power						Calculated decay heat power			C/M		
time	Total		Gamma		Beta		Total	Gamma	Beta	Total	Gamma	Beta
(s)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	(MeV/fiss)	(MeV/fiss)			
0.75	11.0762	2.1	5.5485	2.5	5.5277	4.8	10.980	5.482	5.498	0.991	0.988	0.995
1.00	10.9971	2.1	5.4051	2.5	5.5920	4.7	10.780	5.393	5.387	0.980	0.998	0.963
1.25	10.8334	2.1	5.3354	2.5	5.4980	4.7	10.600	5.315	5.285	0.978	0.996	0.961
1.65	10.6185	2.1	5.2670	2.5	5.3515	4.8	10.350	5.206	5.144	0.975	0.988	0.961
2.00	10.3734	2.1	5.1638	2.5	5.2095	4.8	10.160	5.123	5.037	0.979	0.992	0.967
2.45	10.2440	2.1	5.0332	1.4	5.2109	4.3	9.936	5.029	4.907	0.970	0.999	0.942
2.95	9.9426	2.1	4.9197	1.5	5.0229	4.3	9.721	4.937	4.784	0.978	1.004	0.952
3.60	9.6949	2.1	4.8618	1.5	4.8331	4.4	9.477	4.835	4.642	0.978	0.994	0.960
4.25	9.4732	2.1	4.7577	1.5	4.7155	4.4	9.265	4.746	4.519	0.978	0.998	0.958
5.00	9.2479	2.1	4.6521	1.5	4.5958	4.4	9.051	4.656	4.395	0.979	1.001	0.956
5.95	9.0097	2.1	4.5141	1.5	4.4956	4.4	8.815	4.558	4.257	0.978	1.010	0.947
7.00	8.7726	2.1	4.4588	1.5	4.3138	4.5	8.590	4.465	4.125	0.979	1.001	0.956
8.25	8.5536	2.1	4.3467	1.5	4.2069	4.5	8.361	4.369	3.992	0.977	1.005	0.949
9.65	8.3101	2.1	4.2427	1.5	4.0673	4.5	8.140	4.276	3.864	0.980	1.008	0.950
11.25	8.0536	2.1	4.1400	1.5	3.9136	4.5	7.924	4.184	3.740	0.984	1.011	0.956
13.10	7.8674	2.1	4.0674	1.5	3.7999	4.6	7.710	4.091	3.619	0.980	1.006	0.952
15.25	7.6355	2.1	3.9598	1.5	3.6757	4.6	7.498	3.998	3.500	0.982	1.010	0.952
17.80	7.4010	2.1	3.8661	1.5	3.5349	4.6	7.284	3.902	3.382	0.984	1.009	0.957
20.65	7.1882	2.1	3.7719	1.5	3.4163	4.6	7.081	3.809	3.272	0.985	1.010	0.958
24.10	6.9722	2.1	3.6594	1.5	3.3128	4.6	6.872	3.710	3.162	0.986	1.014	0.954
27.95	6.7623	2.1	3.5595	1.5	3.2028	4.7	6.673	3.615	3.058	0.987	1.016	0.955
32.50	6.5620	2.1	3.4524	1.5	3.1096	4.7	6.473	3.517	2.956	0.986	1.019	0.951
37.70	6.3674	2.1	3.3467	1.5	3.0207	4.7	6.278	3.419	2.859	0.986	1.022	0.946
43.75	6.1491	2.1	3.2447	1.5	2.9045	4.7	6.083	3.320	2.763	0.989	1.023	0.951
60.85	5.6910	2.1	3.0043	1.5	2.6867	4.7	5.654	3.098	2.556	0.994	1.031	0.951
70.25	5.5116	2.1	2.9216	1.6	2.5900	4.7	5.469	3.001	2.468	0.992	1.027	0.953
81.10	5.3184	2.1	2.8211	1.6	2.4973	4.7	5.286	2.905	2.381	0.994	1.030	0.953
93.70	5.1371	2.1	2.7212	1.6	2.4159	4.7	5.106	2.809	2.297	0.994	1.032	0.951
108.30	4.9596	2.1	2.6223	1.6	2.3372	4.7	4.929	2.714	2.215	0.994	1.035	0.948
125.15	4.7807	2.1	2.5331	1.6	2.2476	4.7	4.757	2.622	2.135	0.995	1.035	0.950
144.70	4.5997	2.1	2.4458	1.6	2.1539	4.8	4.590	2.532	2.058	0.998	1.035	0.955
167.35	4.4340	2.1	2.3521	1.6	2.0820	4.8	4.428	2.444	1.984	0.999	1.039	0.953
193.55	4.2732	2.1	2.2626	1.6	2.0106	4.8	4.272	2.360	1.912	1.000	1.043	0.951
223.90	4.1218	2.1	2.1875	1.6	1.9343	4.8	4.122	2.279	1.843	1.000	1.042	0.953
259.20	3.9651	2.1	2.1023	1.7	1.8628	4.8	3.976	2.201	1.775	1.003	1.047	0.953
300.25	3.8218	2.1	2.0229	1.7	1.7989	4.8	3.834	2.124	1.710	1.003	1.050	0.951
347.90	3.6806	2.1	1.9465	1.7	1.7340	4.8	3.695	2.050	1.645	1.004	1.053	0.949
403.15	3.5420	2.1	1.8759	1.7	1.6662	4.8	3.559	1.977	1.582	1.005	1.054	0.949
542.70	3.2669	2.1	1.7308	1.7	1.5361	4.8	3.289	1.833	1.456	1.007	1.059	0.948
621.50	3.1279	2.1	1.6658	1.7	1.4621	4.9	3.167	1.767	1.400	1.012	1.061	0.958
700.30	3.0135	2.1	1.6101	1.7	1.4034	4.9	3.059	1.710	1.349	1.015	1.062	0.961
779.10	2.9150	2.1	1.5585	1.7	1.3565	4.9	2.963	1.658	1.305	1.016	1.064	0.962
857.95	2.8257	2.1	1.5070	1.8	1.3187	4.9	2.875	1.610	1.265	1.017	1.068	0.959
936.75	2.7413	2.1	1.4751	1.8	1.2662	4.9	2.796	1.567	1.229	1.020	1.062	0.971
1015.55	2.6553	2.1	1.4307	1.8	1.2246	4.9	2.722	1.527	1.195	1.025	1.067	0.976
1094.30	2.5859	2.1	1.3953	1.8	1.1906	5.0	2.654	1.490	1.164	1.026	1.068	0.978
1253.75	2.4611	2.1	1.3351	1.3	1.1260	4.8	2.531	1.422	1.109	1.028	1.065	0.985
1604.15	2.2173	1.6	1.1868	1.5	1.0304	3.8	2.307	1.298	1.009	1.040	1.094	0.979
1845.95	2.0819	1.5	1.1134	0.6	0.9685	3.3	2.180	1.228	0.9520	1.047	1.103	0.983
2169.15	1.9375	1.3	1.0588	2.2	0.8787	3.9	2.037	1.147	0.8900	1.051	1.083	1.013
2395.75	1.8835	0.8	1.0068	2.3	0.8767	3.1	1.950	1.098	0.8520	1.035	1.091	0.972
2721.30	1.7378	1.7	0.9496	1.4	0.7881	4.0	1.841	1.036	0.8050	1.059	1.091	1.021
2952.90	1.6836	2.2	0.9091	1.6	0.7745	5.2	1.773	0.9961	0.7769	1.053	1.096	1.003
3494.55	1.5754	1.5	0.8261	2.1	0.7493	3.9	1.636	0.9161	0.7199	1.038	1.109	0.961
3808.25	1.5039	0.6	0.7882	2.2	0.7157	2.7	1.568	0.8762	0.6918	1.043	1.112	0.967
4033.65	1.4696	1.0	0.7634	2.3	0.7062	3.2	1.524	0.8499	0.6741	1.037	1.113	0.955
4353.65	1.4044	1.1	0.7314	2.1	0.6730	3.3	1.466	0.8154	0.6506	1.044	1.115	0.967

4585.45	1.3602	1.6	0.7094	2.2	0.6508	4.0	1.428	0.7922	0.6358	1.050	1.117	0.977
4898.10	1.3135	1.0	0.6817	1.5	0.6317	2.5	1.380	0.7632	0.6168	1.051	1.119	0.976
5128.30	1.2947	1.2	0.6626	1.5	0.6321	2.9	1.347	0.7433	0.6037	1.040	1.122	0.955
5570.25	1.2326	1.9	0.6344	2.2	0.5982	4.5	1.289	0.7080	0.5810	1.046	1.116	0.971
6057.25	1.1778	1.7	0.6030	2.2	0.5748	4.2	1.232	0.6730	0.5590	1.046	1.116	0.973
7033.45	1.0803	1.6	0.5526	3.1	0.5277	4.7	1.135	0.6128	0.5222	1.051	1.109	0.990
7547.80	1.0382	2.3	0.5313	3.3	0.5069	5.8	1.090	0.5853	0.5047	1.050	1.102	0.996
8054.20	0.9706	10.0	0.5130	2.8	0.4576	21.4	1.050	0.5606	0.4894	1.082	1.093	1.069
8562.00	0.9679	1.7	0.4930	2.9	0.4748	4.7	1.013	0.5380	0.4750	1.047	1.091	1.000
9390.05	0.9057	1.5	0.4656	3.1	0.4402	4.5	0.9593	0.5049	0.4544	1.059	1.084	1.032
10414.40	0.8672	1.7	0.4248	2.5	0.4424	4.2	0.9010	0.4694	0.4316	1.039	1.105	0.976
12906.90	0.7649	2.0	0.3622	1.6	0.4027	4.0	0.7877	0.4018	0.3859	1.030	1.109	0.958
15410.25	0.6652	4.7	0.3196	2.4	0.3456	9.3	0.7021	0.3525	0.3496	1.056	1.103	1.012
17891.80	0.6065	3.0	0.2824	2.6	0.3241	6.0	0.6354	0.3157	0.3197	1.048	1.118	0.986
20390.50	0.5599	1.7	0.2556	3.6	0.3042	4.4	0.5811	0.2867	0.2944	1.038	1.122	0.968
22914.50	0.5177	2.4	0.2344	1.5	0.2834	4.5	0.5354	0.2632	0.2722	1.034	1.123	0.961
25406.50	0.4721	2.6	0.2168	2.9	0.2554	5.3	0.4970	0.2441	0.2529	1.053	1.126	0.990
30409.95	0.4052	2.8	0.1907	3.0	0.2144	5.9	0.4345	0.2139	0.2206	1.072	1.121	1.029
35411.60	0.3678	3.3	0.1675	3.4	0.2003	6.8	0.3852	0.1909	0.1943	1.047	1.140	0.970
40408.70	0.3249	3.1	0.1542	3.6	0.1707	6.8	0.3452	0.1726	0.1726	1.063	1.119	1.011
45416.80	0.2909	3.3	0.1393	2.2	0.1515	6.7	0.3119	0.1574	0.1545	1.072	1.130	1.020
50413.55	0.2662	3.7	0.1269	3.6	0.1393	7.9	0.2839	0.1447	0.1392	1.067	1.140	1.000
60398.85	0.2191	3.7	0.1096	4.6	0.1095	8.7	0.2393	0.1243	0.1150	1.092	1.134	1.051
70421.05	0.1979	4.7	0.0924	5.8	0.1055	10.2	0.2055	0.1086	0.0969	1.038	1.175	0.918
89320.22	0.1459	1.5	0.0742	5.3	0.0717	6.3	0.1604	0.0871	0.0734	1.100	1.173	1.024
97189.97	0.1343	7.8	0.0690	6.3	0.0653	17.3	0.1465	0.0802	0.0663	1.091	1.163	1.015
100383.75	0.1275	5.0	0.0672	6.5	0.0603	12.8	0.1414	0.0777	0.0637	1.109	1.157	1.057
102018.66	0.1241	2.8	0.0662	5.6	0.0579	8.8	0.1389	0.0765	0.0624	1.119	1.155	1.078
151276.25	0.0755	4.7	0.0429	12.7	0.0326	20.0	0.0885	0.0509	0.0376	1.173	1.186	1.156

Table 2. Measured and calculated ^{239}Pu decay heat power following a 1000-second irradiation (Friesenhahn et al. 1979)

Cooling time (s)	Measured decay heat power						Calculated decay heat power	
	Total		Gamma		Beta		Total	Total
	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	C/M
1.06	6.9178	4.8	3.0289	4.9	3.8890	7.3	6.547	0.946
1.31	6.8651	4.6	3.0641	4.6	3.8010	6.9	6.427	0.936
1.56	6.7250	4.5	3.0116	5.4	3.7134	6.8	6.317	0.939
1.91	6.6464	4.4	2.9412	5.3	3.7052	6.6	6.176	0.929
2.31	6.4391	4.4	2.8722	5.3	3.5669	6.7	6.032	0.937
2.75	6.2648	4.4	2.8186	5.3	3.4461	6.6	5.890	0.940
3.25	6.0960	4.4	2.7051	5.2	3.3909	6.6	5.746	0.943
3.81	5.9269	4.3	2.6775	5.2	3.2493	6.5	5.602	0.945
4.56	5.7660	4.3	2.5954	5.2	3.1706	7.4	5.433	0.942
5.31	5.5323	4.3	2.5393	4.4	2.9929	6.6	5.284	0.955
6.20	5.3854	4.3	2.4635	4.3	2.9219	7.4	5.129	0.952
7.25	5.2397	4.3	2.3867	4.3	2.8529	6.5	4.968	0.948
8.45	5.0710	4.3	2.3212	4.3	2.7499	6.5	4.809	0.948
9.86	4.8489	4.3	2.2546	4.3	2.5943	6.5	4.647	0.958
11.56	4.6724	4.2	2.1848	4.3	2.4877	6.5	4.478	0.958
13.41	4.5004	4.1	2.1225	4.2	2.3779	6.5	4.319	0.960
15.56	4.3255	4.1	2.0471	4.2	2.2784	6.5	4.159	0.962
18.11	4.1537	4.0	1.9630	4.2	2.1908	6.5	3.997	0.962
21.06	3.9767	4.1	1.8866	4.2	2.0902	6.5	3.835	0.964
24.41	3.8058	4.1	1.8229	4.2	1.9829	6.5	3.677	0.966
28.31	3.6545	4.1	1.7415	4.2	1.9129	6.5	3.519	0.963
32.81	3.4693	4.1	1.6703	4.2	1.7989	6.5	3.361	0.969
38.00	3.3194	4.1	1.5904	4.2	1.7210	6.5	3.205	0.966
44.05	3.1541	4.1	1.5428	4.2	1.6112	6.5	3.048	0.966
51.05	2.9882	4.1	1.4613	4.2	1.5269	6.5	2.891	0.967
59.16	2.8401	4.1	1.3758	4.2	1.4643	6.5	2.735	0.963
68.56	2.6617	4.1	1.3063	4.2	1.3554	6.5	2.580	0.969
79.41	2.5108	4.1	1.2236	4.2	1.2872	6.5	2.428	0.967
92.05	2.3472	4.1	1.1563	4.1	1.1909	6.5	2.278	0.971
106.61	2.2086	4.1	1.0785	4.2	1.1301	6.5	2.134	0.966
123.45	2.0628	4.1	1.0127	4.1	1.0501	6.6	1.995	0.967
143.00	1.9207	4.1	0.9395	4.2	0.9812	6.6	1.863	0.970
165.66	1.7960	4.1	0.8856	4.2	0.9103	6.7	1.736	0.967
191.95	1.6744	4.1	0.8331	4.2	0.8412	6.7	1.617	0.966
260.00	1.4456	4.1	0.7252	4.2	0.7193	6.7	1.391	0.962
301.16	1.3440	4.1	0.6835	4.1	0.6605	6.7	1.291	0.961
348.75	1.2472	4.1	0.6346	4.2	0.6127	6.7	1.196	0.959
403.91	1.1576	4.2	0.5909	4.1	0.5667	6.7	1.106	0.955
543.27	0.9747	4.2	0.5091	4.2	0.4656	6.9	0.935	0.959
621.95	0.9009	4.2	0.4708	4.2	0.4302	6.9	0.862	0.956
700.66	0.8297	4.2	0.4407	4.2	0.3890	7.0	0.799	0.963
779.36	0.7746	4.2	0.4183	4.2	0.3563	7.2	0.745	0.962
858.06	0.7185	4.2	0.3950	4.2	0.3236	7.3	0.697	0.970
936.86	0.6761	4.2	0.3685	4.2	0.3076	7.3	0.655	0.969
1015.56	0.6398	4.2	0.3541	4.2	0.2857	7.4	0.617	0.965
1094.27	0.6004	4.2	0.3302	4.2	0.2702	7.4	0.583	0.971
1253.45	0.5302	4.3	0.3055	3.6	0.2247	7.2	0.523	0.986
1814.20	0.3747	5.0	0.2228	4.0	0.1520	9.0	0.376	1.003
2049.25	0.3252	4.5	0.1991	3.9	0.1261	8.3	0.333	1.024
2540.15	0.2554	4.6	0.1601	4.0	0.0953	8.7	0.265	1.038
2777.60	0.2352	4.6	0.1477	3.6	0.0874	8.4	0.240	1.020
3283.60	0.1890	5.3	0.1238	3.8	0.0652	11.0	0.197	1.042
4008.05	0.1479	4.2	0.0957	4.0	0.0522	8.2	0.154	1.041

4225.15	0.1382	5.1	0.0895	4.2	0.0488	10.4	0.144	1.042
4695.60	0.1218	4.4	0.0776	4.1	0.0442	8.5	0.126	1.032
4947.95	0.1116	4.6	0.0726	3.9	0.0390	8.9	0.117	1.052
5503.80	0.0973	4.6	0.0645	3.8	0.0328	9.0	0.102	1.048

Table 3. Measured and calculated ^{239}Pu decay heat power following a 24-hour irradiation (Friesenhahn et al. 1979)

Cooling time (s)	Measured decay heat power						Calculated decay heat power	
	Total		Gamma		Beta		Total	Total
	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	C/M
0.85	9.4349	3.5	4.5000	2.9	4.9349	6.2	9.026	0.957
1.05	9.4267	3.3	4.5201	2.5	4.9065	5.7	8.920	0.946
1.35	9.2972	3.3	4.4519	2.5	4.8453	5.7	8.776	0.944
1.7	9.1260	3.3	4.4019	2.4	4.7241	5.6	8.626	0.945
2.1	8.9752	3.3	4.2870	2.4	4.6882	5.6	8.473	0.944
2.55	8.7358	3.2	4.2578	2.4	4.4780	5.6	8.320	0.952
3.05	8.6173	3.2	4.2041	2.4	4.4132	5.6	8.169	0.948
3.6	8.4366	3.2	4.1040	2.4	4.3326	5.6	8.022	0.951
4.35	8.2263	3.2	4.0330	2.3	4.1933	5.6	7.845	0.954
5.1	8.0509	3.2	3.9681	2.3	4.0828	5.6	7.690	0.955
6.0	7.9049	3.2	3.8963	2.3	4.0086	5.6	7.528	0.952
7.05	7.7180	3.2	3.8133	2.3	3.9047	5.6	7.363	0.954
8.25	7.5309	3.2	3.7552	2.3	3.7756	5.6	7.199	0.956
9.65	7.3510	3.2	3.6731	2.3	3.6780	5.5	7.034	0.957
11.25	7.1854	3.2	3.6086	2.3	3.5768	5.5	6.870	0.956
13.15	6.9598	3.2	3.5307	2.3	3.4291	5.6	6.702	0.963
15.35	6.7883	3.2	3.4599	2.3	3.3284	5.6	6.535	0.963
17.85	6.6281	3.2	3.4019	2.2	3.2262	5.6	6.371	0.961
20.75	6.4402	3.2	3.3234	2.2	3.1168	5.6	6.208	0.964
24.1	6.2673	3.2	3.2503	2.2	3.0170	5.7	6.046	0.965
27.95	6.0871	3.2	3.1700	2.2	2.9172	5.7	5.885	0.967
32.4	5.9352	3.2	3.0898	2.3	2.8454	5.7	5.724	0.964
37.65	5.7709	3.2	3.0120	2.2	2.7589	5.7	5.560	0.963
43.65	5.5933	3.2	2.9382	2.3	2.6551	5.7	5.399	0.965
50.65	5.4282	3.2	2.8531	2.3	2.5750	5.7	5.236	0.965
58.75	5.2527	3.2	2.7725	2.3	2.4802	5.7	5.073	0.966
68.15	5.0849	3.1	2.7025	2.3	2.3824	5.7	4.910	0.966
79.05	4.9220	3.2	2.6063	2.3	2.3157	5.7	4.748	0.965
91.65	4.7506	3.2	2.5283	2.3	2.2223	5.8	4.589	0.966
106.3	4.6034	3.2	2.4565	2.3	2.1469	5.8	4.433	0.963
123.28	4.4403	3.1	2.3821	2.3	2.0583	5.8	4.280	0.964
142.85	4.2959	3.1	2.3024	2.3	1.9935	5.8	4.133	0.962
165.55	4.1467	3.1	2.2364	2.3	1.9103	5.8	3.990	0.962
191.85	4.0032	3.1	2.1698	2.3	1.8335	5.9	3.852	0.962
259.58	3.7293	3.1	2.0287	2.3	1.7005	6.0	3.580	0.960
300.4	3.5969	3.1	1.9669	2.3	1.6300	6.0	3.454	0.960
347.8	3.4656	3.1	1.9078	2.3	1.5578	6.0	3.328	0.960
402.65	3.3390	3.1	1.8370	2.4	1.5021	6.1	3.205	0.960
542.0	3.0734	3.2	1.7057	2.4	1.3677	6.1	2.953	0.961
620.35	2.9521	3.2	1.6470	2.4	1.3051	6.2	2.838	0.961
698.75	2.8398	3.2	1.5891	2.4	1.2507	6.3	2.735	0.963
777.05	2.7405	3.2	1.5387	2.4	1.2018	6.4	2.643	0.964
855.35	2.6553	3.2	1.4876	2.5	1.1676	6.4	2.560	0.964
933.65	2.5762	3.2	1.4441	2.5	1.1321	6.4	2.483	0.964
1011.95	2.4998	3.2	1.4064	2.5	1.0934	6.4	2.413	0.965
1090.25	2.4241	3.2	1.3684	2.5	1.0558	6.5	2.347	0.968
1249.3	2.2976	3.2	1.2971	2.9	1.0005	6.9	2.227	0.969
2542.15	1.6274	4.0	0.9365	2.2	0.6909	8.0	1.622	0.997
2751.5	1.5844	3.7	0.8972	2.2	0.6872	7.2	1.559	0.984
3234.4	1.4372	3.9	0.8169	2.9	0.6203	8.0	1.435	0.998
4169.2	1.2576	3.9	0.7064	2.6	0.5512	7.9	1.255	0.998
4637.0	1.1909	3.9	0.6706	2.5	0.5202	7.8	1.184	0.994
4859.8	1.1592	3.9	0.6489	2.5	0.5103	7.9	1.154	0.996

5326.6	1.0960	3.9	0.6086	2.5	0.4874	7.7	1.097	1.001
5951.0	1.0023	3.8	0.5640	2.4	0.4383	7.7	1.032	1.030
7988.7	0.8685	4.0	0.4637	2.5	0.4048	7.7	0.873	1.005
8988.4	0.8153	4.1	0.4331	2.4	0.3822	7.7	0.815	1.000
10014.4	0.7593	4.1	0.3971	2.6	0.3622	7.8	0.765	1.008
11028.3	0.7216	4.1	0.3737	2.4	0.3479	7.7	0.723	1.002
14502.5	0.6123	4.2	0.3066	2.3	0.3057	7.7	0.613	1.001
17077.6	0.5541	4.2	0.2749	2.3	0.2792	7.7	0.555	1.001
19653.8	0.5053	4.2	0.2511	2.1	0.2543	7.7	0.509	1.006
22251.3	0.4669	4.3	0.2280	2.1	0.2389	7.6	0.470	1.007
24832.0	0.4335	4.3	0.2113	2.1	0.2223	7.7	0.438	1.011
27409.2	0.4060	4.3	0.1981	2.1	0.2079	7.7	0.411	1.011
32006.6	0.3654	4.3	0.1779	2.1	0.1875	7.7	0.369	1.011
37024.7	0.3265	4.3	0.1609	2.1	0.1656	7.8	0.333	1.019
42029.7	0.2956	4.4	0.1459	2.1	0.1497	7.9	0.302	1.023
47025.9	0.2708	4.4	0.1338	2.1	0.1371	7.9	0.277	1.022
64958.5	0.2040	4.4	0.1039	2.2	0.1001	8.2	0.210	1.031
74953.5	0.1778	4.4	0.0911	2.1	0.0867	8.3	0.185	1.038
94948.2	0.1400	4.5	0.0733	2.2	0.0667	8.6	0.147	1.051
104968	0.1265	4.5	0.0669	2.1	0.0596	8.7	0.133	1.054
129967	0.1008	4.4	0.0543	2.2	0.0466	8.7	0.107	1.063

Table 4. Measured and calculated ^{235}U decay heat power following a 1000-second irradiation (Friesenhahn et al. 1979)

Cooling time (s)	Measured decay heat power						Calculated decay heat power	
	Total		Gamma		Beta		Total	Total
	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	C/M
1.05	8.4428	3.4	4.2337	3.6	4.2092	4.6	8.076	0.957
1.35	8.2162	3.4	3.9275	3.7	4.2007	4.6	7.868	0.958
1.7	8.0232	3.4	3.0151	3.6	4.2082	4.6	7.654	0.954
2.17	7.7760	3.4	3.7127	3.6	4.0632	4.6	7.403	0.952
2.5	7.6129	3.4	3.6436	3.7	3.9693	4.6	7.247	0.952
3.05	7.3776	3.4	3.5567	3.6	3.8209	4.6	7.015	0.951
3.6	7.1322	3.4	3.4815	3.7	3.6507	4.6	6.811	0.955
4.25	6.9178	3.4	3.4054	3.7	3.5124	4.6	6.600	0.954
5.1	6.6650	3.4	3.2959	3.6	3.3691	4.6	6.359	0.954
6	6.4539	3.4	3.1931	3.6	3.2608	4.6	6.139	0.951
7.05	6.2096	3.4	3.0863	3.7	3.1233	4.7	5.916	0.953
8.25	5.9557	3.4	3.0062	3.7	2.9496	4.7	5.697	0.957
9.65	5.7563	3.4	2.9057	3.7	2.8506	4.7	5.477	0.951
11.25	5.5248	3.4	2.8104	3.7	2.7144	4.7	5.262	0.952
13.1	5.2804	3.4	2.6950	3.7	2.5854	4.7	5.049	0.956
15.25	5.0489	3.4	2.5993	3.7	2.4497	4.8	4.839	0.958
17.75	4.8332	3.4	2.5028	3.7	2.3304	4.8	4.630	0.958
20.65	4.6243	3.4	2.3958	3.7	2.2286	4.8	4.425	0.957
24	4.4107	3.4	2.2946	3.7	2.1161	4.8	4.223	0.957
27.9	4.1869	3.4	2.2016	3.7	1.9853	4.9	4.024	0.961
32.4	3.9981	3.5	2.0969	3.7	1.9012	4.9	3.829	0.958
37.6	3.7871	3.5	2.0084	3.7	1.7787	4.9	3.636	0.960
43.75	3.5795	3.5	1.8894	3.7	1.6901	4.9	3.442	0.962
50.75	3.3904	3.5	1.7928	3.7	1.5976	5.0	3.253	0.959
58.85	3.1809	3.5	1.6849	3.8	1.4960	5.1	3.067	0.964
68.25	2.9889	3.5	1.5893	3.8	1.3996	5.1	2.883	0.965
79.15	2.8025	3.5	1.4889	3.8	1.3136	5.1	2.702	0.964
91.8	2.6135	3.5	1.3948	3.8	1.2187	5.2	2.525	0.966
106.45	2.4384	3.5	1.2977	3.8	1.1407	5.2	2.354	0.965
123.35	2.2640	3.6	1.1965	3.8	1.0675	5.2	2.191	0.968
143	2.0974	3.6	1.1084	3.8	0.9890	5.3	2.034	0.970
165.65	1.9369	3.6	1.0285	3.9	0.9084	5.3	1.887	0.974
191.95	1.7973	3.6	0.9455	3.9	0.8518	5.3	1.747	0.972
259.8	1.5231	3.7	0.8010	3.9	0.7221	5.4	1.487	0.976
300.6	1.4067	3.7	0.7411	4.0	0.6656	5.5	1.373	0.976
347.85	1.2928	3.7	0.6835	4.0	0.6093	5.6	1.267	0.980
402.6	1.1853	3.7	0.6258	4.0	0.5595	5.7	1.167	0.985
541.9	0.9900	3.8	0.5274	4.0	0.4627	5.9	0.980	0.990
620.7	0.9103	3.9	0.4829	4.1	0.4273	6.0	0.901	0.990
699.5	0.8407	3.9	0.4508	4.1	0.3900	6.1	0.835	0.993
778.4	0.7816	3.9	0.4188	4.1	0.3628	6.2	0.778	0.995
857.2	0.7311	4.0	0.3902	4.2	0.3409	6.3	0.728	0.996
936	0.6860	4.0	0.3692	4.2	0.3168	6.4	0.685	0.998
1014.8	0.6448	4.0	0.3507	4.2	0.2942	6.5	0.646	1.002
1096.6	0.6098	4.1	0.3308	4.2	0.2790	6.6	0.610	1.000
1252.95	0.5496	4.2	0.2989	4.0	0.2508	6.5	0.550	1.001
1812.8	0.3903	4.4	0.2093	4.1	0.1810	6.8	0.401	1.028
2038.35	0.3494	4.5	0.1904	4.6	0.1591	7.3	0.359	1.028
2524.3	0.2762	4.4	0.1548	5.0	0.1213	7.5	0.290	1.050
2738.75	0.2620	4.2	0.1436	4.3	0.1184	6.6	0.266	1.016
3208.65	0.2110	4.4	0.1232	4.8	0.0878	7.6	0.224	1.062
3899.75	0.1741	4.5	0.1005	5.3	0.0736	8.3	0.180	1.032
4115.95	0.1587	4.6	0.0952	7.1	0.0635	10.9	0.169	1.062

4602.5	0.1356	4.9	0.0848	6.8	0.0508	11.8	0.148	1.091
4822.4	0.1332	5.8	0.0799	6.3	0.0533	12.1	0.140	1.050
5324	0.1181	5.2	0.0689	4.6	0.0492	9.2	0.124	1.052
5970	0.1011	4.6	0.0612	5.5	0.0399	9.1	0.108	1.070
8007.2	0.0725	4.7	0.0412	6.6	0.0313	9.8	0.075	1.038
9003.25	0.0619	5.3	0.0346	6.8	0.0272	10.7	0.065	1.049
10016.7	0.0556	5.6	0.0321	7.2	0.0235	12.1	0.057	1.019
11000	0.0492	6.0	0.0278	7.1	0.0214	12.4	0.050	1.020
14506.7	0.0327	5.2	0.0170	8.5	0.0157	11.1	0.035	1.064
17075.85	0.0267	5.4	0.0138	9.2	0.0129	12.0	0.028	1.047
19658.9	0.0215	6.6	0.0101	11.9	0.0114	13.9	0.023	1.078
22248	0.0181	7.6	0.0088	12.5	0.0093	16.2	0.020	1.088
24869.55	0.0163	7.7	0.0079	13.5	0.0084	17.0	0.017	1.047
27452.35	0.0154	7.3	0.0073	15.0	0.0082	17.0	0.015	0.978
32110.75	0.0114	9.3	0.0072	16.1	0.0041	32.6	0.012	1.086
37023.05	0.0104	9.5	0.0055	18.6	0.0049	25.6	0.010	0.997
42069.45	0.0080	11.5	0.0047	22.9	0.0034	38.0	0.009	1.104
47060.16	0.0077	11.7	0.0020	44.1	0.0058	21.7	0.008	0.992
52092.76	0.0067	12.1	0.0019	47.6	0.0048	25.0	0.007	0.998

Table 5. Measured and calculated ^{235}U decay heat power following a 20,000-second irradiation (Friesenhahn et al. 1979)

Cooling time (s)	Measured decay heat power						Calculated decay heat power	
	Total		Gamma		Beta		Total	Total
	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	C/M
0.95	11.0345	3.6	5.5572	3.7	5.4774	5.1	10.31	0.934
1.25	10.9029	3.5	5.5729	3.7	5.3300	5.1	10.09	0.925
1.59	10.6504	3.5	5.4210	3.7	5.2294	5.1	9.875	0.927
2.00	10.3939	3.5	5.2906	3.7	5.1033	5.1	9.646	0.928
2.44	10.1775	3.5	5.2076	3.7	4.9698	5.1	9.431	0.927
2.94	9.9514	3.5	5.1005	3.7	4.8430	5.1	9.215	0.926
3.5	9.6926	3.5	5.0127	3.7	4.6799	5.1	9.003	0.929
4.25	9.4443	3.5	4.8739	3.7	4.5708	5.1	8.755	0.927
5.00	9.2100	3.5	4.7917	3.7	4.4183	5.1	8.541	0.927
5.89	8.9577	3.5	4.7035	3.4	4.2543	4.9	8.319	0.929
6.95	8.6960	3.5	4.5557	3.4	4.1403	4.9	8.091	0.930
8.16	8.4483	3.5	4.4452	3.4	4.0031	5.0	7.866	0.931
9.55	8.2052	3.5	4.3325	3.4	3.8727	5.0	7.645	0.932
11.16	7.9706	3.5	4.2322	3.4	3.7384	5.0	7.426	0.932
13.05	7.7218	3.5	4.1140	3.4	3.6078	5.0	7.206	0.933
15.25	7.4903	3.5	4.0079	3.4	3.4824	5.0	6.989	0.933
17.7	7.2573	3.5	3.9155	3.4	3.3418	5.0	6.782	0.935
20.64	7.0320	3.5	3.8047	3.4	3.2273	5.0	6.572	0.935
24.00	6.7951	3.5	3.6858	3.4	3.1093	5.0	6.368	0.937
27.84	6.5817	3.6	3.5858	3.4	2.9960	5.0	6.169	0.937
32.3	6.3653	3.6	3.4753	3.4	2.8900	5.1	5.972	0.938
37.55	6.1483	3.6	3.3597	3.4	2.7886	5.1	5.773	0.939
43.55	5.9281	3.6	3.2438	3.4	2.6843	5.1	5.579	0.941
50.55	5.7200	3.6	3.1341	3.4	2.5859	5.1	5.385	0.941
58.59	5.5153	3.6	3.0239	3.4	2.4914	5.1	5.194	0.942
67.95	5.3078	3.6	2.9143	3.4	2.3935	5.1	5.003	0.943
78.8	5.0972	3.6	2.8052	3.4	2.2920	5.1	4.814	0.944
91.34	4.9002	3.6	2.6965	3.4	2.2037	5.1	4.629	0.945
105.89	4.6992	3.6	2.5908	3.4	2.1084	5.1	4.448	0.947
122.75	4.5029	3.6	2.4847	3.4	2.0182	5.2	4.272	0.949
142.3	4.3117	3.6	2.3826	3.4	1.9291	5.2	4.102	0.951
165.00	4.1358	3.6	2.2848	3.4	1.8510	5.2	3.937	0.952
191.41	3.9673	3.6	2.1909	3.4	1.7764	5.2	3.778	0.952
259.14	3.6310	3.6	2.0083	3.5	1.6227	5.3	3.472	0.956
300.14	3.4808	3.6	1.9267	3.5	1.5541	5.4	3.330	0.957
347.55	3.3298	3.6	1.8483	3.5	1.4815	5.4	3.193	0.959
402.45	3.1841	3.6	1.7703	3.5	1.4139	5.4	3.059	0.961
541.09	2.8985	3.6	1.6137	3.5	1.2848	5.5	2.793	0.964
619.59	2.7690	3.6	1.5437	3.5	1.2253	5.5	2.672	0.965
698.00	2.6557	3.7	1.4842	3.5	1.1715	5.5	2.566	0.966
776.39	2.5532	3.7	1.4297	3.6	1.1235	5.5	2.471	0.968
854.8	2.4605	3.7	1.3808	3.6	1.0797	5.5	2.385	0.969
933.2	2.3805	3.7	1.3340	3.6	1.0465	5.6	2.307	0.969
1011.55	2.3012	3.7	1.2956	3.6	1.0056	5.6	2.235	0.971
1089.84	2.2291	3.7	1.2567	3.6	0.9725	5.6	2.168	0.973
1249.05	2.1006	3.6	1.1867	3.6	0.9139	5.5	2.047	0.974
1796.5	1.7371	4.0	0.9949	3.5	0.7421	6.1	1.725	0.993
2012.95	1.6287	4.3	0.9305	3.5	0.6982	6.6	1.626	0.998
2472.00	1.4474	4.9	0.8315	3.5	0.6159	5.9	1.452	1.003
2687.2	1.3649	4.1	0.7879	3.5	0.5770	6.3	1.383	1.013
3162.55	1.2358	4.1	0.7112	3.6	0.5247	6.3	1.254	1.015
3862.09	1.0993	4.0	0.6311	3.8	0.4682	6.1	1.103	1.003
4079.3	1.0545	3.9	0.6007	3.6	0.4538	6.0	1.064	1.009

4559.34	0.9686	4.3	0.5556	4.0	0.4130	6.9	0.9864	1.018
4776.95	0.9361	4.2	0.5377	4.0	0.3984	6.7	0.9549	1.020
5259.3	0.8855	4.0	0.4954	3.3	0.3891	6.0	0.8919	1.007
5862.5	0.8183	4.0	0.4525	3.3	0.3657	5.8	0.8239	1.007
7816.2	0.6499	4.0	0.3538	3.4	0.2961	5.8	0.6597	1.015
8804.00	0.5949	4.0	0.3179	3.4	0.2770	5.8	0.5985	1.006
9778.2	0.5423	4.1	0.2859	3.5	0.2564	5.9	0.5478	1.010
10777.41	0.5052	4.1	0.2597	3.4	0.2455	5.8	0.5036	0.997
11754.5	0.4647	4.2	0.2365	3.4	0.2282	5.8	0.4665	1.004
14476.05	0.3854	4.2	0.1874	3.5	0.1980	5.8	0.3861	1.002
17018.2	0.3332	4.2	0.1573	3.6	0.1759	5.8	0.3320	0.996
19586.19	0.2905	4.2	0.1335	3.7	0.1570	5.8	0.2906	1.000
22130.3	0.2590	4.3	0.1155	3.7	0.1435	6.0	0.2585	0.998
24674.09	0.2353	4.3	0.1038	4.0	0.1315	6.1	0.2326	0.989
27218.95	0.2135	4.4	0.0939	4.0	0.1196	6.0	0.2112	0.989
31978.55	0.1794	4.4	0.0791	4.2	0.1003	6.2	0.1796	1.001
37004.44	0.1547	4.5	0.0680	4.3	0.0867	6.3	0.1542	0.997
41979.81	0.1356	4.5	0.0591	4.5	0.0765	6.4	0.1345	0.992
46992.84	0.1182	4.5	0.0502	5.0	0.0680	6.7	0.1184	1.002
52007.31	0.1069	4.7	0.0450	5.0	0.0619	6.7	0.1052	0.984
64933.41	0.0791	4.7	0.0365	5.5	0.0425	7.5	0.08035	1.016
74947.69	0.0659	4.9	0.0306	5.5	0.0353	7.9	0.06703	1.017
94942.84	0.0485	5.2	0.0238	6.3	0.0247	9.0	0.04927	1.016
104933.8	0.0416	5.0	0.0213	6.7	0.0282	9.6	0.04314	1.037

Table 6. Measured and calculated ^{235}U decay heat power following a 24-hour irradiation (Friesenhahn et al. 1979)

Cooling time (s)	Measured decay heat power						Calculated decay heat power	
	Total		Gamma		Beta		Total	Total
	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	C/M
0.85	11.7559	3.9	6.0556	3.7	5.7002	5.7	10.900	0.927
1.05	11.6298	3.7	5.9676	3.7	5.6622	5.3	10.740	0.923
1.35	11.4759	3.7	5.8330	3.7	5.6430	5.2	10.530	0.918
1.70	11.2331	3.6	5.7748	3.7	5.4583	5.2	10.320	0.919
2.10	11.0160	3.6	5.6888	3.7	5.3272	5.2	10.100	0.917
2.55	10.8643	3.6	5.5967	3.7	5.2676	5.1	9.890	0.910
3.05	10.5348	3.6	5.5080	3.6	5.0269	5.1	9.681	0.919
3.60	10.3065	3.6	5.3810	3.6	4.9255	5.1	9.477	0.920
4.30	10.0371	3.6	5.3046	3.6	4.7325	5.0	9.250	0.922
5.10	9.8274	3.6	5.1878	3.5	4.6396	5.0	9.024	0.918
6.00	9.6028	3.5	5.0509	3.5	4.5519	5.0	8.803	0.917
7.05	9.3190	3.5	4.9659	3.5	4.3531	5.0	8.581	0.921
8.25	9.0799	3.5	4.8393	3.5	4.2486	4.9	8.361	0.921
9.65	8.8672	3.5	4.7328	3.5	4.1344	4.9	8.140	0.918
11.25	8.5930	3.5	4.6552	3.5	3.9377	5.0	7.924	0.922
13.10	8.3566	3.5	4.5153	3.5	3.8413	5.0	7.710	0.923
15.25	8.1314	3.5	4.3956	3.5	3.7358	5.0	7.498	0.922
17.80	7.8848	3.5	4.3051	3.4	3.5796	5.0	7.284	0.924
20.65	7.6593	3.5	4.2025	3.4	3.4567	5.0	7.081	0.924
24.08	7.4255	3.5	4.1052	3.5	3.3202	5.0	6.873	0.926
27.90	7.2026	3.5	3.9849	3.4	3.2177	5.0	6.676	0.927
32.40	7.0051	3.5	3.8580	3.4	3.1471	5.0	6.477	0.925
37.55	6.7794	3.5	3.7435	3.4	3.0359	5.0	6.283	0.927
43.60	6.5336	3.5	3.6301	3.4	2.9235	5.0	6.087	0.932
50.55	6.3435	3.5	3.5262	3.4	2.8173	5.0	5.894	0.929
58.60	6.1381	3.5	3.4196	3.4	2.7186	5.0	5.702	0.929
68.00	5.9281	3.5	3.3022	3.4	2.6259	5.0	5.511	0.930
78.95	5.7192	3.5	3.1854	3.4	2.5339	5.0	5.320	0.930
91.55	5.5154	3.5	3.0609	3.4	2.4546	5.0	5.135	0.931
106.15	5.3023	3.5	2.9577	3.4	2.3446	5.0	4.953	0.934
123.15	5.1200	3.5	2.8538	3.4	2.2662	5.0	4.776	0.933
142.75	4.9283	3.5	2.7385	3.4	2.1897	5.0	4.605	0.934
165.50	4.7411	3.5	2.6360	3.4	2.1052	5.0	4.440	0.936
191.80	4.5746	3.5	2.5370	3.4	2.0376	5.0	4.282	0.936
259.50	4.2329	3.5	2.3541	3.4	1.8788	5.1	3.975	0.939
300.40	4.0763	3.5	2.2615	3.4	1.8148	5.1	3.833	0.940
347.80	3.9273	3.5	2.1786	3.4	1.7487	5.1	3.695	0.941
402.65	3.7835	3.5	2.0944	3.4	1.6891	5.1	3.560	0.941
541.30	3.4948	3.5	1.9338	3.4	1.5610	5.1	3.291	0.942
619.65	3.3564	3.5	1.8608	3.4	1.4956	5.1	3.169	0.944
698.00	3.2382	3.5	1.7982	3.4	1.4400	5.1	3.062	0.946
776.35	3.1409	3.5	1.7423	3.5	1.3986	5.2	2.966	0.944
854.60	3.0403	3.5	1.6898	3.5	1.3506	5.2	2.879	0.947
932.90	2.9569	3.6	1.6382	3.5	1.3187	5.2	2.799	0.947
1011.20	2.8721	3.2	1.5972	3.5	1.2749	4.8	2.726	0.949
1089.50	2.7994	3.6	1.5560	3.5	1.2434	5.2	2.658	0.949
1248.58	2.6596	3.6	1.4789	3.5	1.1808	5.2	2.534	0.953
1783.00	2.2760	4.2	1.2761	4.0	0.9998	6.7	2.211	0.971
1986.1	2.1933	4.0	1.2183	3.9	0.9750	6.2	2.115	0.964
2436.6	2.0296	4.3	1.0909	3.5	0.9387	6.5	1.935	0.953
2632.7	1.9270	4.2	1.0478	3.7	0.8792	6.3	1.869	0.970
3081.6	1.8055	4.1	0.9793	4.0	0.8262	6.4	1.737	0.962
3728.8	1.6359	4.0	0.8987	4.1	0.7372	6.2	1.585	0.969

3923.7	1.6185	4.6	0.8796	4.1	0.7389	7.5	1.545	0.955
4398.5	1.4837	4.3	0.8259	3.9	0.6577	6.8	1.459	0.983
4599.8	1.4735	4.4	0.8018	4.0	0.6717	6.7	1.426	0.968
5040.2	1.4016	4.7	0.7609	3.6	0.6407	7.3	1.359	0.970
5587.8	1.3301	4.1	0.7104	3.4	0.6258	6.0	1.287	0.968
6530.6	1.2230	4.2	0.6379	3.5	0.5851	6.0	1.182	0.966
8397.9	1.0625	4.2	0.5341	3.6	0.5283	6.0	1.025	0.965
9323.6	0.9814	4.2	0.4994	3.6	0.4820	6.1	0.963	0.982
10282.2	0.9453	4.2	0.4666	3.6	0.4786	5.9	0.908	0.961
11232.4	0.9027	4.2	0.4380	3.5	0.4646	6.0	0.860	0.952
14448.7	0.7604	4.2	0.3630	3.5	0.3975	6.0	0.732	0.963
16920.8	0.6860	4.3	0.3219	3.6	0.3642	6.0	0.660	0.962
19425.0	0.6211	4.3	0.2924	3.5	0.3287	6.0	0.601	0.967
21959.1	0.5732	4.3	0.2660	3.5	0.3071	6.0	0.552	0.962
24467.5	0.5281	4.3	0.2450	3.4	0.2831	6.1	0.511	0.967
26947.8	0.4909	4.3	0.2274	3.5	0.2635	6.1	0.476	0.970
31920.3	0.4351	4.3	0.2034	3.6	0.2317	6.1	0.418	0.962
36942.4	0.3824	4.3	0.1799	3.8	0.2025	6.3	0.372	0.973
41949.7	0.3406	4.4	0.1617	3.8	0.1789	6.3	0.334	0.982
46932.7	0.3084	4.3	0.1501	4.0	0.1583	6.5	0.303	0.982
51935.2	0.2817	4.3	0.1387	3.9	0.1430	6.5	0.276	0.980
64986.6	0.2259	4.4	0.1143	4.0	0.1116	6.8	0.223	0.986
74896.0	0.1916	4.4	0.0991	4.2	0.0925	6.9	0.193	1.007
94923.59	0.1491	4.5	0.0792	4.4	0.0699	7.7	0.150	1.008
100883.5	0.1412	4.6	0.0736	4.3	0.0676	7.8	0.141	0.996
106580.0	0.1302	4.6	0.0698	4.8	0.0611	8.1	0.132	1.017
129893.0	0.1025	4.7	0.0574	5.1	0.0451	8.9	0.106	1.034
154877.59	0.0850	5.1	0.0466	5.6	0.0384	9.7	0.087	1.020

Table 7. Measured and calculated ^{235}U decay heat power following a 35-day irradiation (Friesenhahn et al. 1979)

Cooling time (s)	Measured decay heat power						Calculated decay heat power	
	Total		Gamma		Beta		Total	Total
	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	σ (%)	(MeV/fiss)	C/M
1.3	12.1806	3.9	6.5526	3.8	5.6280	5.9	11.140	0.915
1.5	12.0488	3.9	6.5397	3.8	5.5101	5.9	11.010	0.914
1.75	11.9211	3.9	6.4843	3.8	5.4367	5.9	10.870	0.912
2.15	11.7087	3.9	6.3200	3.8	5.3886	5.9	10.660	0.910
2.50	11.5499	3.9	6.2100	3.8	5.3399	5.9	10.490	0.908
2.95	11.2990	3.9	6.1537	3.8	5.1453	6.0	10.300	0.912
3.45	11.0996	3.9	6.0281	3.8	5.0715	5.9	10.110	0.911
4.1	10.8966	3.9	5.9166	3.8	4.9800	5.9	9.895	0.908
4.75	10.6526	3.9	5.8183	3.8	4.8344	6.0	9.704	0.911
5.5	10.4077	3.9	5.7168	3.8	4.6908	6.0	9.510	0.914
6.45	10.2010	3.9	5.6113	3.8	4.5897	6.0	9.293	0.911
7.45	9.9492	3.9	5.5308	3.8	4.4183	6.0	9.095	0.914
8.7	9.7217	3.9	5.4356	3.8	4.2861	6.1	8.879	0.913
10.1	9.4971	3.9	5.3143	3.8	4.1827	6.1	8.671	0.913
11.75	9.2713	3.9	5.2141	3.8	4.0572	6.1	8.460	0.912
13.68	9.0339	3.9	5.0905	3.8	3.9434	6.1	8.249	0.913
15.75	8.8101	3.9	4.9576	3.8	3.8525	6.2	8.054	0.914
18.25	8.5781	3.9	4.8591	3.8	3.7190	6.2	7.853	0.915
21.15	8.3306	3.9	4.7509	3.8	3.5797	6.2	7.653	0.919
24.5	8.1100	3.9	4.6432	3.8	3.4668	6.3	7.456	0.919
28.4	7.9116	3.9	4.5184	3.8	3.3932	6.3	7.260	0.918
32.9	7.6860	3.9	4.3957	3.8	3.2903	6.3	7.067	0.919
38.1	7.4688	3.9	4.2984	3.8	3.1704	6.3	6.875	0.920
44.1	7.2587	3.9	4.1768	3.8	3.0819	6.3	6.685	0.921
61.2	6.7805	3.9	3.9100	3.9	2.8705	6.4	6.263	0.924
70.55	6.5780	3.9	3.8007	3.9	2.7774	6.4	6.082	0.925
81.4	6.3777	3.9	3.6897	3.9	2.6880	6.4	5.902	0.925
93.9	6.1868	3.9	3.5790	3.9	2.6078	6.5	5.725	0.925
108.45	5.9907	3.9	3.4616	3.9	2.5298	6.5	5.550	0.926
125.3	5.8160	3.9	3.3509	3.9	2.4651	6.5	5.380	0.925
144.75	5.6116	3.9	3.2465	3.9	2.3651	6.6	5.215	0.929
167.3	5.4318	4.0	3.1454	4.0	2.2864	6.6	5.055	0.931
193.45	5.2593	4.0	3.0398	4.0	2.2195	6.6	4.900	0.932
223.7	5.0988	4.0	2.9480	4.0	2.1508	6.6	4.751	0.932
258.75	4.9355	4.0	2.8566	4.0	2.0785	6.6	4.606	0.933
299.65	4.7709	4.0	2.7608	4.0	2.0101	6.7	4.465	0.936
347.1	4.6103	4.0	2.6715	4.0	1.9388	6.7	4.327	0.939
402.1	4.4701	4.0	2.5862	4.0	1.8839	6.7	4.191	0.938
541.05	4.1564	4.0	2.4115	4.1	1.7449	6.9	3.922	0.944
619.55	4.0170	4.0	2.3331	4.1	1.6839	6.9	3.799	0.946
698.05	3.8911	4.1	2.2678	4.1	1.6232	7.0	3.692	0.949
776.55	3.7899	4.1	2.2011	4.1	1.5888	7.0	3.595	0.949
855.00	3.6947	4.1	2.1411	4.1	1.5536	7.1	3.507	0.949
933.45	3.6125	4.1	2.0958	4.1	1.5168	7.1	3.427	0.949
1011.9	3.5387	4.1	2.0499	4.1	1.4888	7.1	3.354	0.948
1090.3	3.4520	4.1	2.0116	4.2	1.4403	7.2	3.285	0.952
1248.9	3.3180	4.2	1.9291	4.2	1.3889	7.2	3.161	0.953
1792.1	3.0248	5.6	1.6771	5.1	1.3477	10.1	2.829	0.935
1997.1	2.9599	4.9	1.6619	5.5	1.2979	9.1	2.731	0.923
2788.3	2.5449	5.2	1.4561	4.5	1.0888	9.2	2.437	0.958
2995.05	2.4371	4.6	1.4296	4.6	1.0076	8.2	2.376	0.975
3636.7	2.2788	5.6	1.3423	4.5	0.9366	10.3	2.216	0.972
4109.5	2.2375	5.2	1.2728	4.2	0.9646	9.1	2.119	0.947
4318.1	2.1060	10.0	1.2427	4.3	0.8633	16.9	2.080	0.988

4767.95	2.0733	4.4	1.2065	4.1	0.8668	7.4	2.005	0.967
4967.6	2.1051	5.2	1.1761	4.1	0.9290	9.1	1.975	0.938
5577.95	1.9743	5.0	1.1231	5.0	0.8512	9.3	1.891	0.958
6207.05	1.8837	5.4	1.0833	4.5	0.8004	9.8	1.816	0.964
7461.65	1.7894	5.8	0.9860	5.3	0.8034	10.6	1.693	0.946
8092.55	1.7099	4.8	0.9475	4.7	0.7625	8.3	1.641	0.960
8692.65	1.6703	5.2	0.9240	4.3	0.7462	8.9	1.596	0.956
9571.00	1.6257	5.2	0.8838	6.4	0.7419	10.2	1.538	0.946
10554.55	1.5471	4.8	0.8576	3.9	0.6896	8.1	1.480	0.957
13944.6	1.3736	4.7	0.7359	5.6	0.6379	8.7	1.327	0.966
16168.95	1.3004	4.4	0.7027	5.5	0.5977	8.3	1.253	0.964
18691.05	1.2392	5.3	0.6605	5.3	0.5787	9.5	1.184	0.955
21150.65	1.1776	5.2	0.6309	4.2	0.5467	8.7	1.128	0.958
23641.65	1.1235	4.5	0.6040	4.3	0.5195	7.5	1.080	0.961
26146.4	1.0733	5.0	0.5705	4.3	0.5027	8.3	1.038	0.967
31131.75	1.0001	4.7	0.5450	5.6	0.4552	8.9	0.968	0.968
36155.55	0.9400	4.6	0.5112	4.4	0.4288	8.0	0.911	0.969
41119.55	0.8838	4.6	0.4865	4.3	0.3973	8.0	0.864	0.977
46154.45	0.8389	4.5	0.4724	4.0	0.3666	7.8	0.823	0.981
51122.3	0.8001	5.0	0.4613	6.0	0.3389	10.6	0.788	0.985
61121.4	0.7412	4.6	0.4296	4.9	0.3116	9.0	0.730	0.985
71149.86	0.6979	4.6	0.4018	4.1	0.2961	8.4	0.684	0.980
91126.81	0.6150	7.8	0.3886	6.1	0.2264	18.9	0.615	1.000
101147.7	0.5976	5.4	0.3567	3.8	0.2409	10.5	0.588	0.984
126148.6	0.5316	5.4	0.3243	4.1	0.2073	10.9	0.535	1.007
151138.2	0.4981	5.5	0.3073	4.3	0.1909	11.3	0.496	0.996
170530.2	0.4744	4.5	0.2922	4.7	0.1821	9.6	0.472	0.994
181650.9	0.4636	4.4	0.2859	4.5	0.1777	9.1	0.459	0.991
202656.6	0.4409	4.4	0.2795	5.0	0.1614	10.3	0.439	0.996

Table 8. Measured and calculated ^{235}U total decay heat power following a 1-hour irradiation (Schrock et al. 1978)

Cooling time (sec)	Measurements			Calculations	
	Run 1 (MeV/fiss)	Run 12 (MeV/fiss)	σ (%)	Run 1 & 12 (MeV/fiss)	Run 12 C/M
11	6.130	7.069	22.7	6.344	0.8974
20	5.849	6.633	17.5	5.518	0.8319
40	5.303	5.883	11.1	4.603	0.7824
50	5.065	5.562	7.3	4.316	0.7760
70	4.650	5.008	6.4	3.89	0.7768
100	4.148	4.356	5.7	3.454	0.7929
200	3.116	3.122	4.0	2.696	0.8635
400	2.194	2.231	3.4	2.069	0.9274
600	1.673	1.803	3.5	1.739	0.9645
720		1.724	3.4	1.595	0.9252
1020		1.368	3.4	1.328	0.9708
1980		0.8725	3.4	0.856	0.9811
3000		0.6277	3.4	0.606	0.9654
4980		0.3902	3.4	0.3728	0.9554
7020		0.2834	3.4	0.2594	0.9153
10020		0.2256	3.4	0.1731	0.7673

Table 9. Measured and calculated ^{235}U total decay heat power following a 4-hour irradiation (Schrock et al. 1978)

Cooling time (sec)	Measurements				Calculations	
	Run 3 (MeV/fiss)	Run 4 (MeV/fiss)	Run 9 (MeV/fiss)	σ (%)	Run 3, 4 & 9 (MeV/fiss)	Run 9 C/M
11	8.922	9.372	8.777	22.7	7.290	0.831
20	8.216	8.586	8.247	17.5	6.459	0.783
40	7.003	7.237	7.343	11.1	5.536	0.754
50	6.540	6.722	6.959	7.3	5.245	0.754
70	5.821	5.925	6.302	6.4	4.811	0.763
100	5.100	5.130	5.539	5.7	4.365	0.788
200	4.024	3.977	4.127	4	3.582	0.868
400	3.149	3.106	3.120	3.4	2.917	0.935
600	2.524	2.497	2.607	3.5	2.555	0.980
720	2.474	2.456	2.377	3.4	2.394	1.007
1020	2.113	2.103	2.058	3.4	2.086	1.014
1980	1.500	1.582	1.532	3.4	1.506	0.983
3000	1.210	1.220	1.191	3.4	1.169	0.982
4980	0.818	0.904	0.862	3.4	0.816	0.946
7020	0.626	0.609	0.677	3.4	0.619	0.914
10020	0.468	0.447	0.498	3.4	0.451	0.907
19980	0.274	0.265	0.280	3.4	0.228	0.813
30000	0.148	0.136	0.201	9.6	0.150	0.746
49980		0.098	0.103	9.6	0.084	0.816
70020		0.082	0.049	21	0.055	1.122

Table 10. Measured and calculated ^{235}U total decay heat power following a 22.35-hour irradiation (Schrock et al. 1978)

Cooling time (sec)	Measurements			Calculations	
	Run 6 (MeV/fiss)	Run 8 (MeV/fiss)	σ (%)	Run 6 & 8 (MeV/fiss)	Run 8 C/M
11	10.19	9.25	22.7	7.958	0.860
20	9.353	8.623	17.5	7.126	0.826
40	7.933	7.602	11.1	6.201	0.816
50	7.401	7.188	7.3	5.910	0.822
70	6.588	6.509	6.4	5.475	0.841
100	5.796	5.771	5.7	5.027	0.871
200	4.682	4.564	4	4.239	0.929
400	3.808	3.722	3.4	3.567	0.958
600	3.162	3.193	3.5	3.199	1.002
720	3.006	3.139	3.4	3.035	0.967
1020	2.697	2.746	3.4	2.719	0.990
1980	2.131	2.117	3.4	2.118	1.000
3000	1.773	1.777	3.4	1.760	0.990
4980	1.319	1.384	3.4	1.369	0.989
7020	1.150	1.156	3.4	1.136	0.983
10020	0.9291	0.9453	3.4	0.923	0.976
19980	0.5903	0.6121	3.4	0.590	0.963
30000		0.4544	9.6	0.439	0.967
49980		0.3024	9.6	0.286	0.947
70020		0.2159	21	0.207	0.958

Table 11. Comparison of decay heat measurements and calculations for ^{239}Pu and ^{233}U thermal fission (Fiche 1976)

	Measured				Calculated			
	^{239}Pu		^{233}U		^{239}Pu		^{233}U	
Cooling time (s)	(MeV/s)	σ (%)	(MeV/s)	σ (%)	(MeV/s)	C/E	(MeV/s)	C/M
50	1.94E-02	14			2.18E-02	1.122	2.08E-02	
60			1.52E-02	25	1.82E-02		1.74E-02	1.146
70	1.50E-02	13	1.33E-02	21	1.55E-02	1.034	1.50E-02	1.125
100	1.06E-02	12	9.33E-03	18	1.05E-02	0.991	1.03E-02	1.101
150	6.68E-03	11	6.04E-03	15	6.49E-03	0.971	6.46E-03	1.070
200	4.88E-03	10	4.42E-03	13	4.58E-03	0.939	4.61E-03	1.042
300	3.12E-03	9	2.84E-03	12	2.88E-03	0.922	2.90E-03	1.021
500	1.89E-03	9	1.69E-03	11	1.71E-03	0.904	1.71E-03	1.012
700	1.40E-03	8	1.23E-03	11	1.25E-03	0.890	1.23E-03	1.003
1000	1.01E-03	8	8.80E-04	11	8.90E-04	0.882	8.74E-04	0.994
1500	6.73E-04	8	5.92E-04	11	5.92E-04	0.880	5.85E-04	0.988
2000	4.91E-04	8	4.36E-04	10	4.31E-04	0.878	4.32E-04	0.991
3000	2.99E-04	8	2.73E-04	10	2.62E-04	0.877	2.72E-04	0.996
5000	1.46E-04	8	1.43E-04	10	1.30E-04	0.893	1.45E-04	1.013
7000	8.82E-05	8	9.36E-05	10	8.06E-05	0.913	9.51E-05	1.016
10000	5.31E-05	8	5.97E-05	11	4.83E-05	0.910	6.09E-05	1.020
15000	2.96E-05	8	3.52E-05	11	2.71E-05	0.915	3.63E-05	1.031
20000	1.97E-05	8	2.42E-05	11	1.82E-05	0.924	2.49E-05	1.031
30000	1.18E-05	8	1.40E-05	11	1.09E-05	0.925	1.46E-05	1.046
50000	6.81E-06	8	7.15E-06	12	6.01E-06	0.882	7.29E-06	1.019
70000	4.42E-06	8	4.47E-06	13	3.96E-06	0.895	4.46E-06	0.997
100000	2.82E-06	8	2.50E-06	13	2.49E-06	0.882	2.63E-06	1.051

Table 12. Comparison of measured and calculated fission product decay heat for Pu fuel samples irradiated in the Dounraey reactor (Johnson 1965)

Sample	Decay time (days)	Measured fission product decay heat (μW)	Experimental error (%)	Calculated fission product decay heat (μW)	C/M
2	46	625	5%	519.1	0.831
	67	450	5%	377.9	0.840
	76	399	5%	339.7	0.851
	90	351	5%	294.0	0.838
	110	289	5%	245.9	0.851
	144	221	5%	189.2	0.856
3	54	491	5%	462.4	0.942
	62	429	5%	405.4	0.945
	75	358	5%	339.7	0.949
	97	281	5%	268.7	0.956
	125	231	5%	211.8	0.917
4	53	521	5%	470.8	0.904
	61	462	5%	411.7	0.891
	74	385	5%	343.9	0.893
	96	303	5%	271.3	0.895
	124	248	5%	213.4	0.860

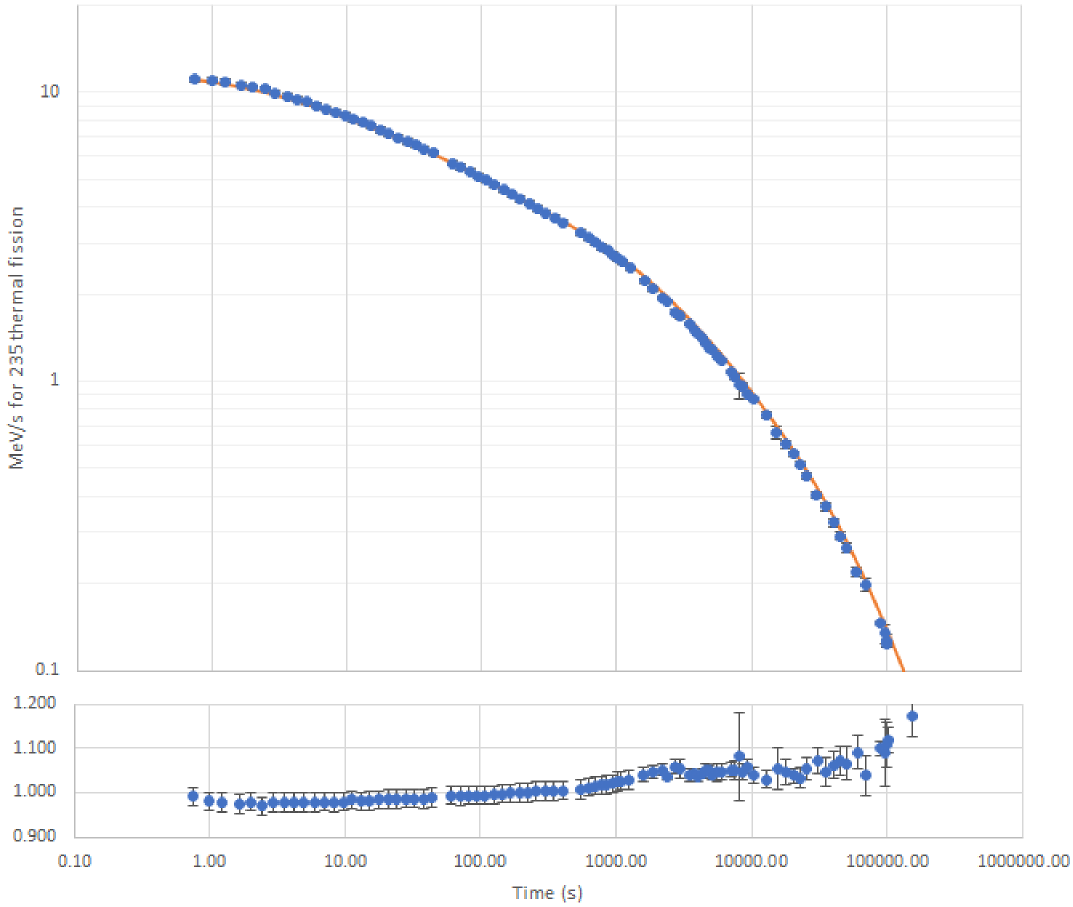

Figure 1. Measured (points) and calculated (solid line) total decay heat power following thermal ^{235}U fission. The calculated to measured ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1976.

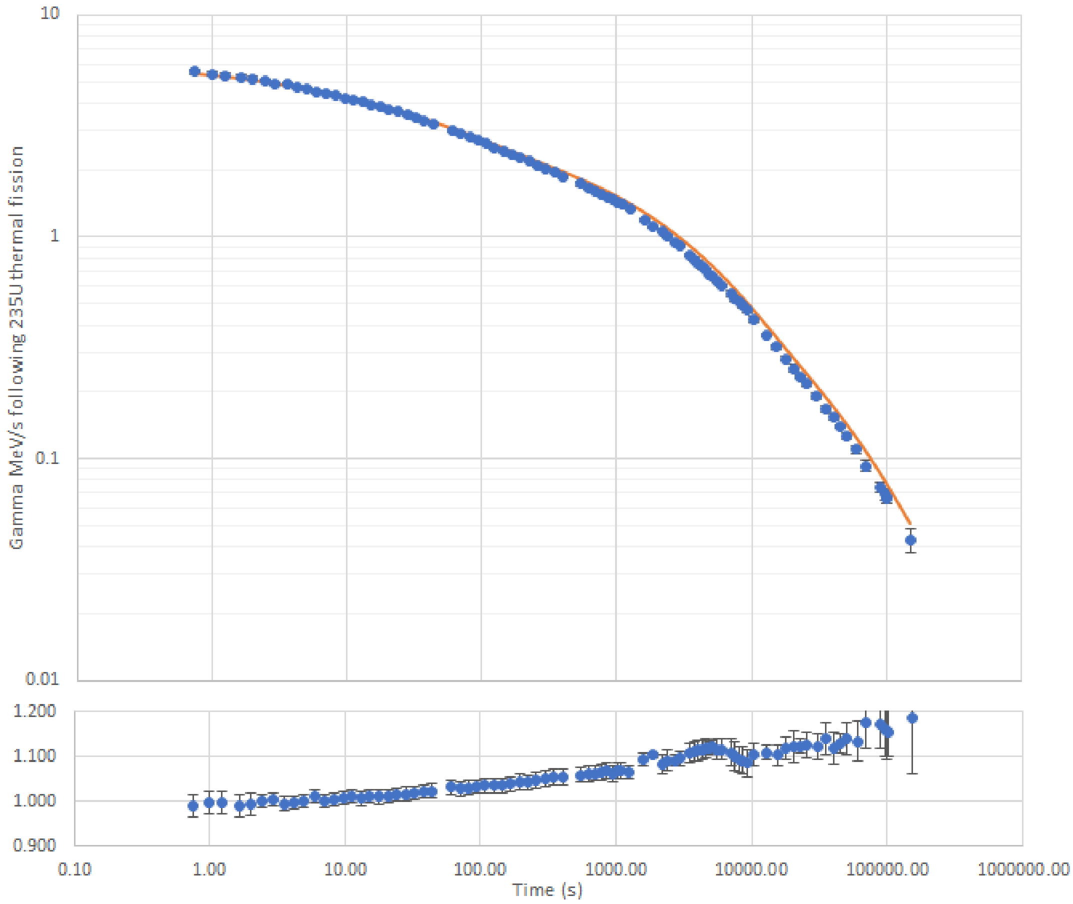

Figure 2. Measured (points) and calculated (solid line) gamma component of decay heat power following thermal ^{235}U fission. The calculated to measured ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1976.

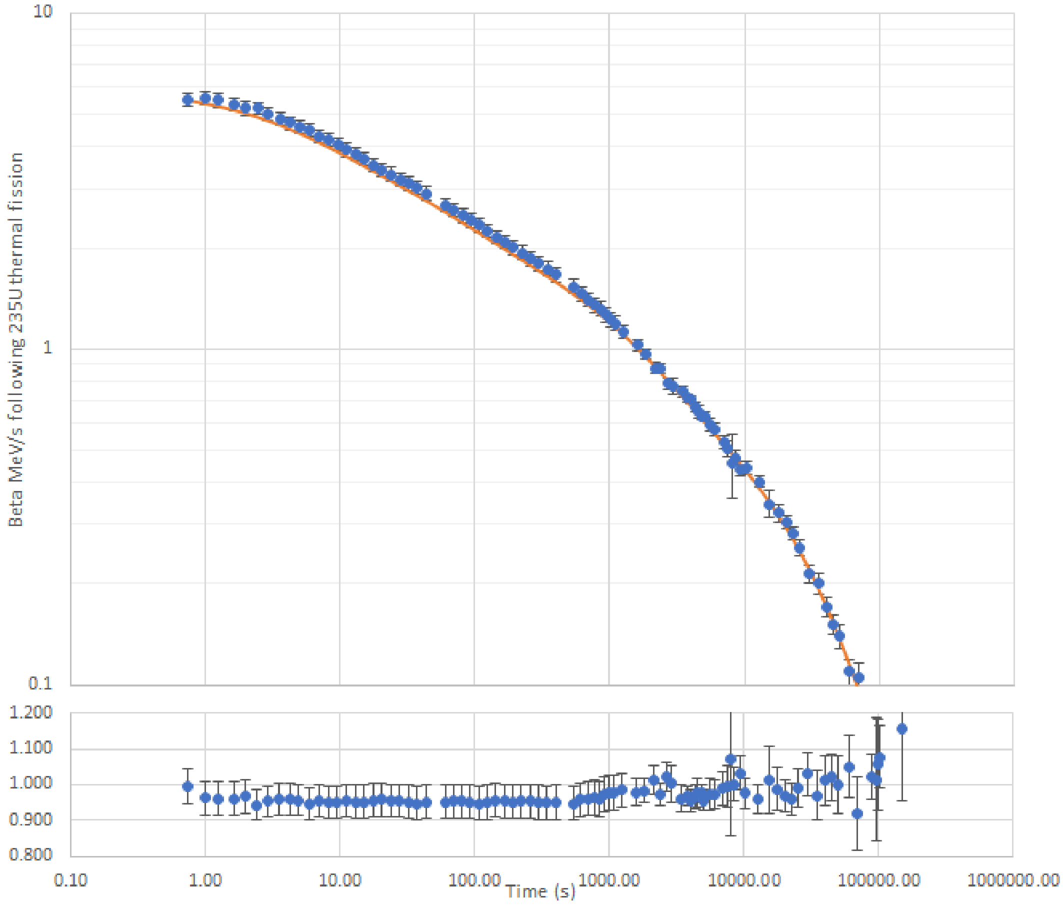

Figure 3. Measured (points) and calculated (solid line) beta component of decay heat power following thermal ^{235}U fission. The calculated to measured ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1976.

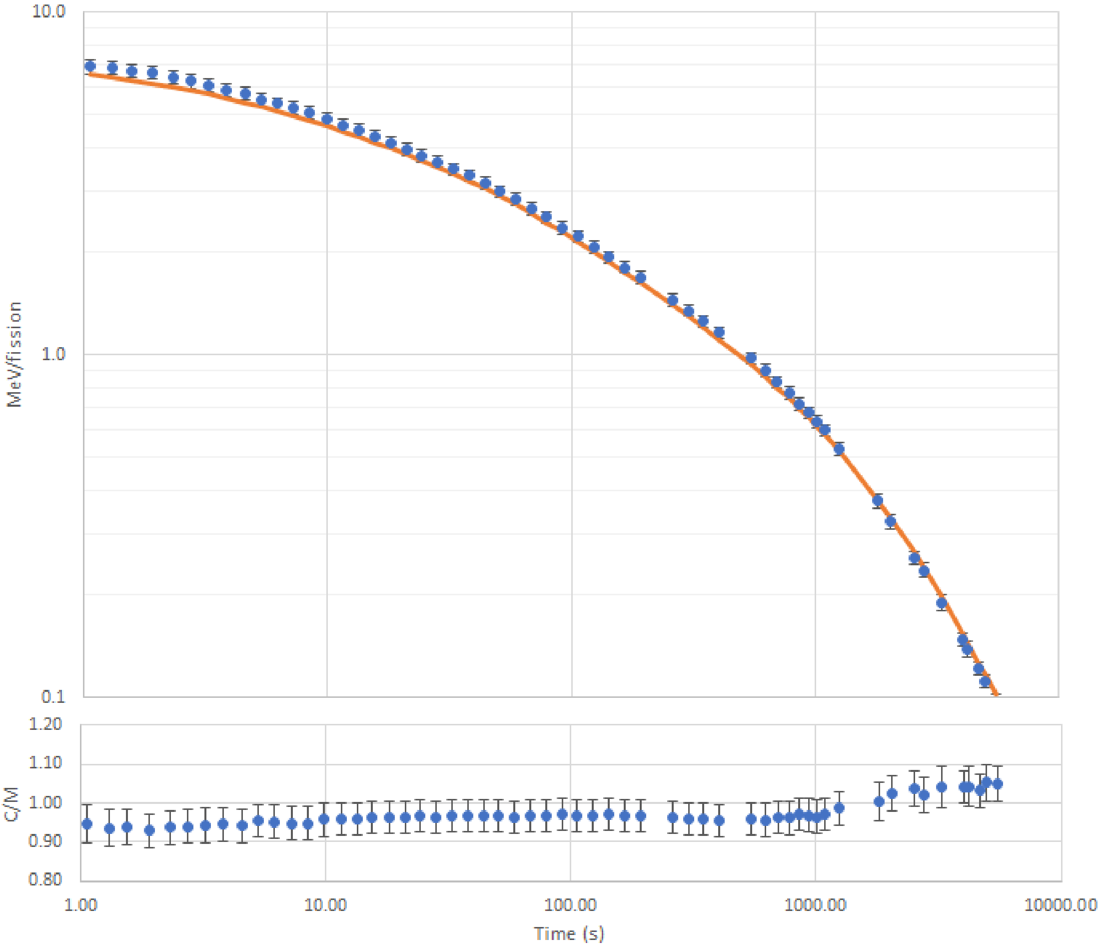

Figure 4. Measured (points) and calculated (solid line) total decay heat power following 1000-second irradiation of ^{239}Pu. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1979.

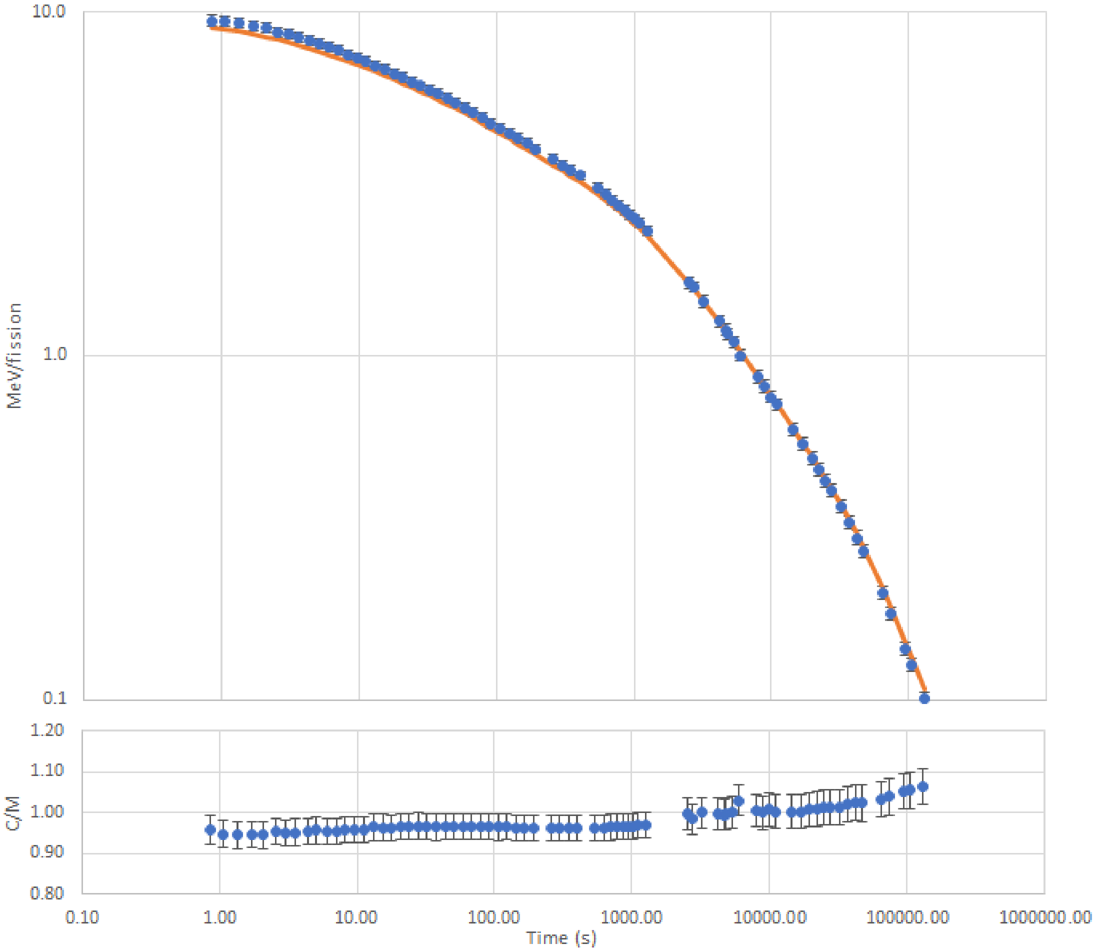

Figure 5. Measured (points) and calculated (solid line) total decay heat power following 24-hour irradiation of ^{239}Pu. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1979.

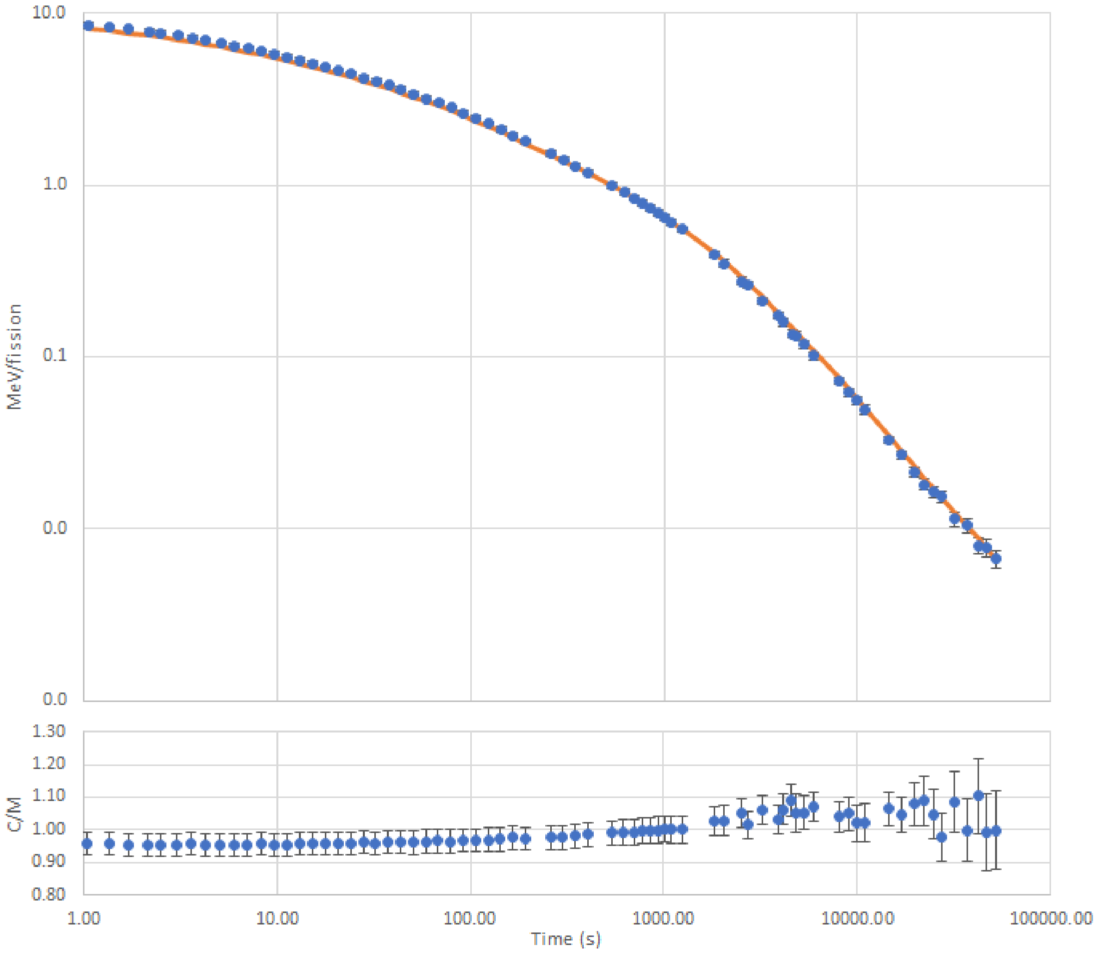

Figure 6. Measured (points) and calculated (solid line) total decay heat power following 1000-second irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1979.

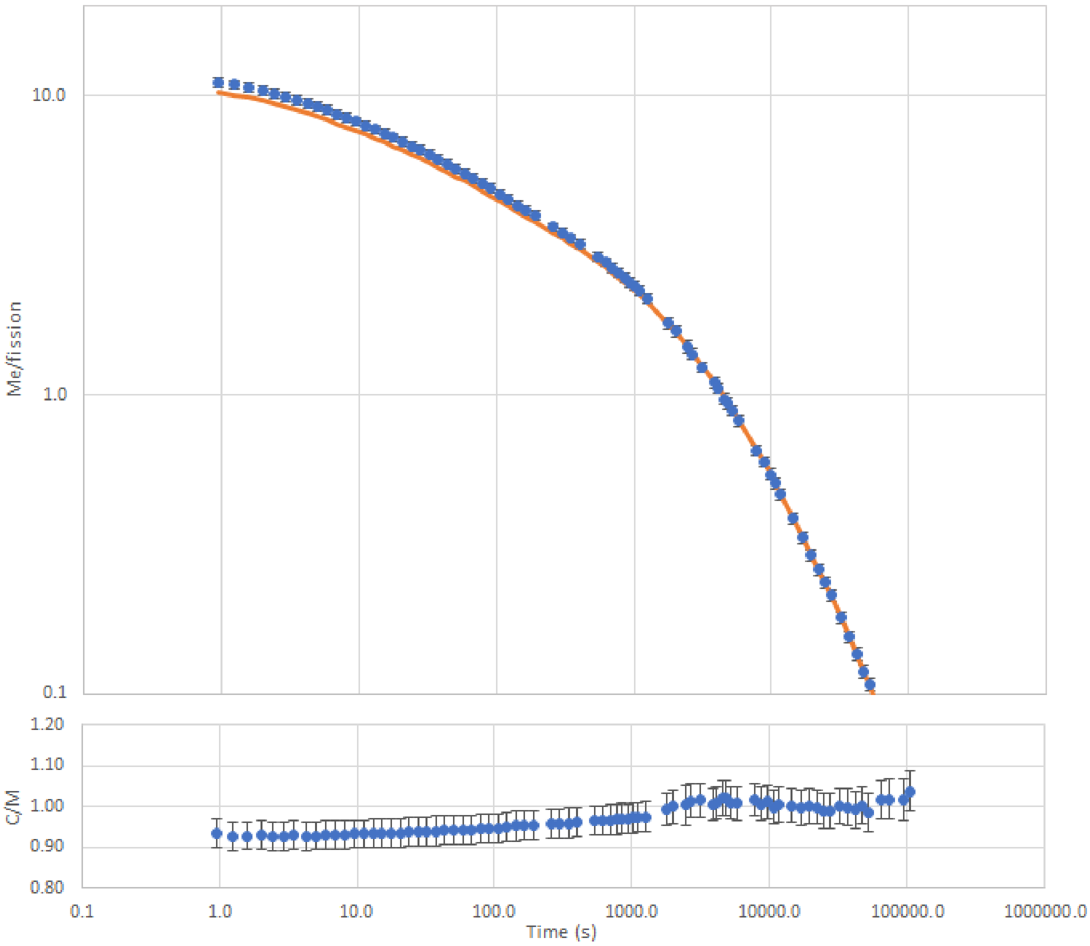

Figure 7. Measured (points) and calculated (solid line) total decay heat power following 20,000-second irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1979.

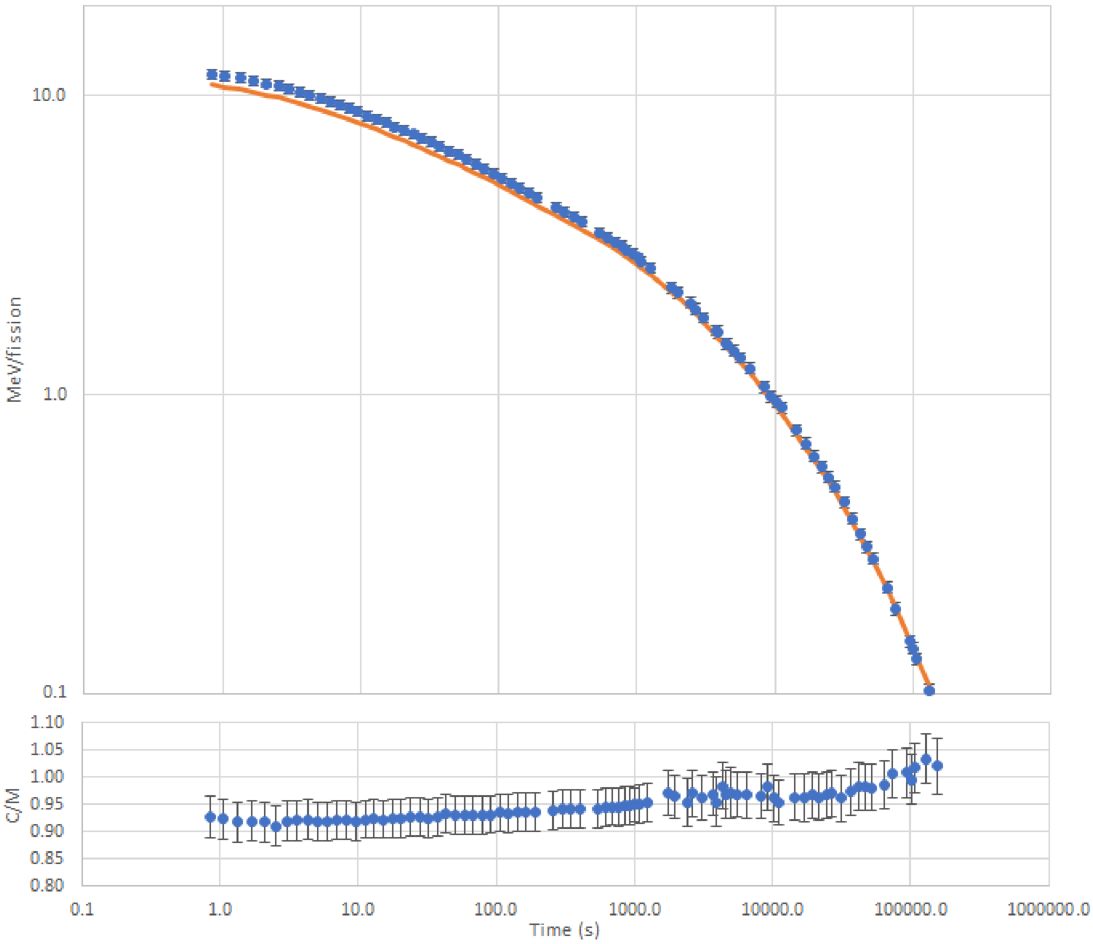

Figure 8. Measured (points) and calculated (solid line) total decay heat power following 24-hour irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1979.

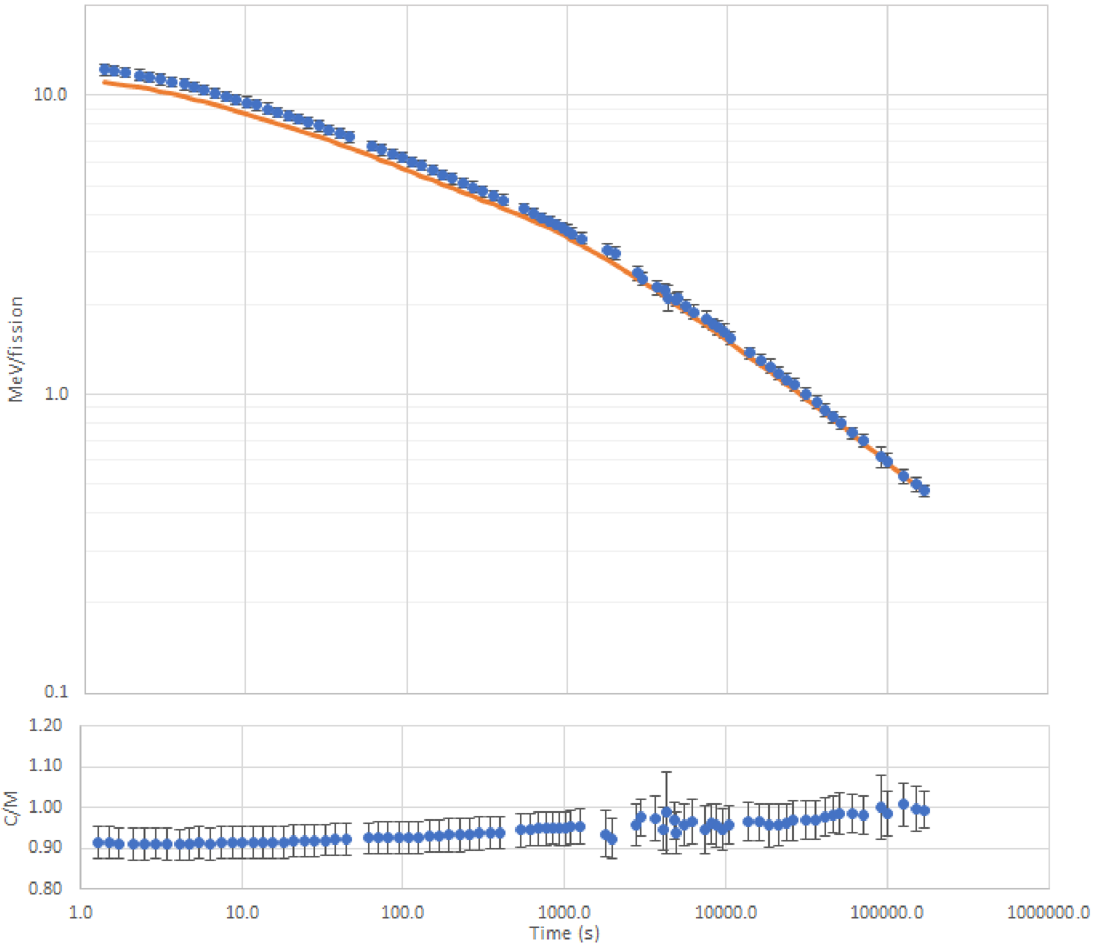

Figure 9. Measured (points) and calculated (solid line) total decay heat power following 35-day irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Friesenhahn et al. 1979.

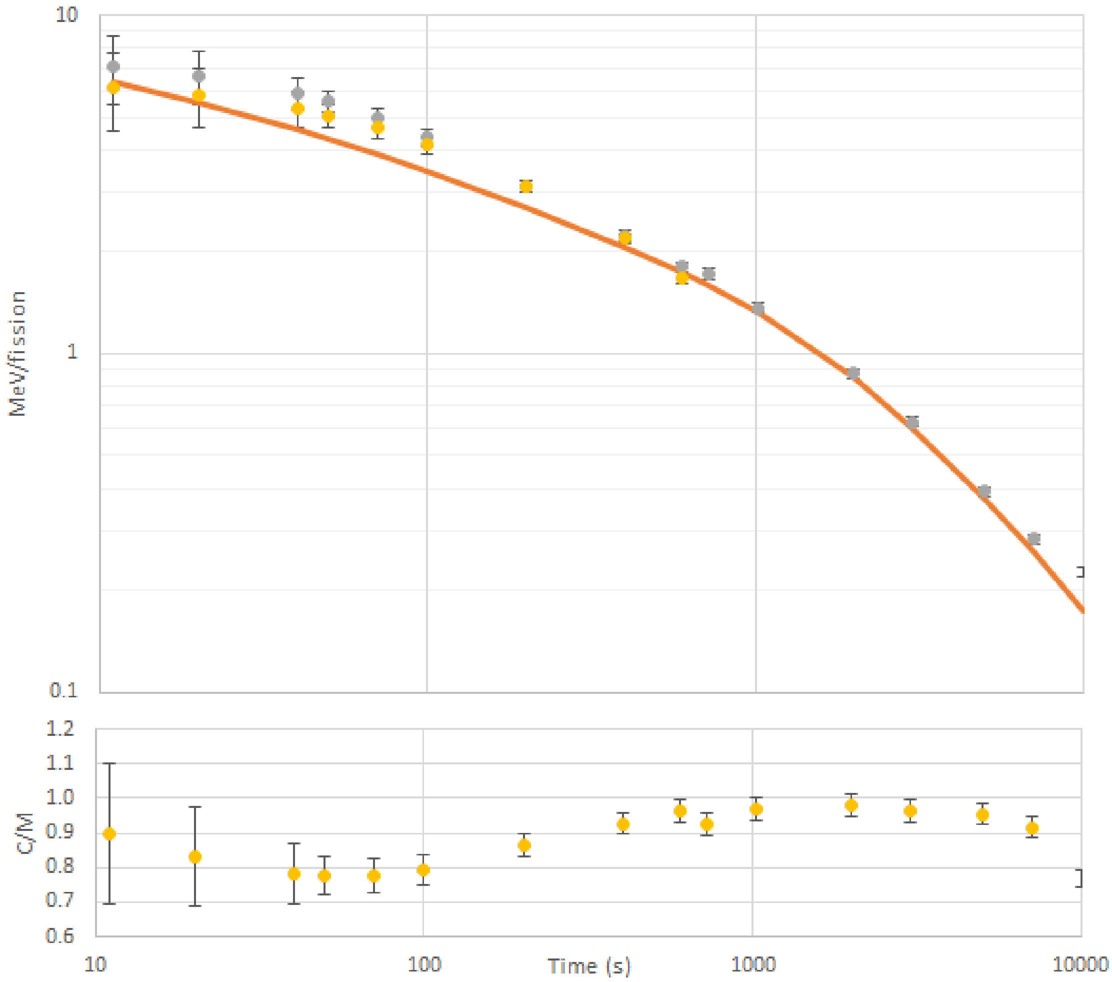

Figure 10. Measured (points) and calculated (solid line) total decay heat power following 1-hour irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Schrock et al. 1978.

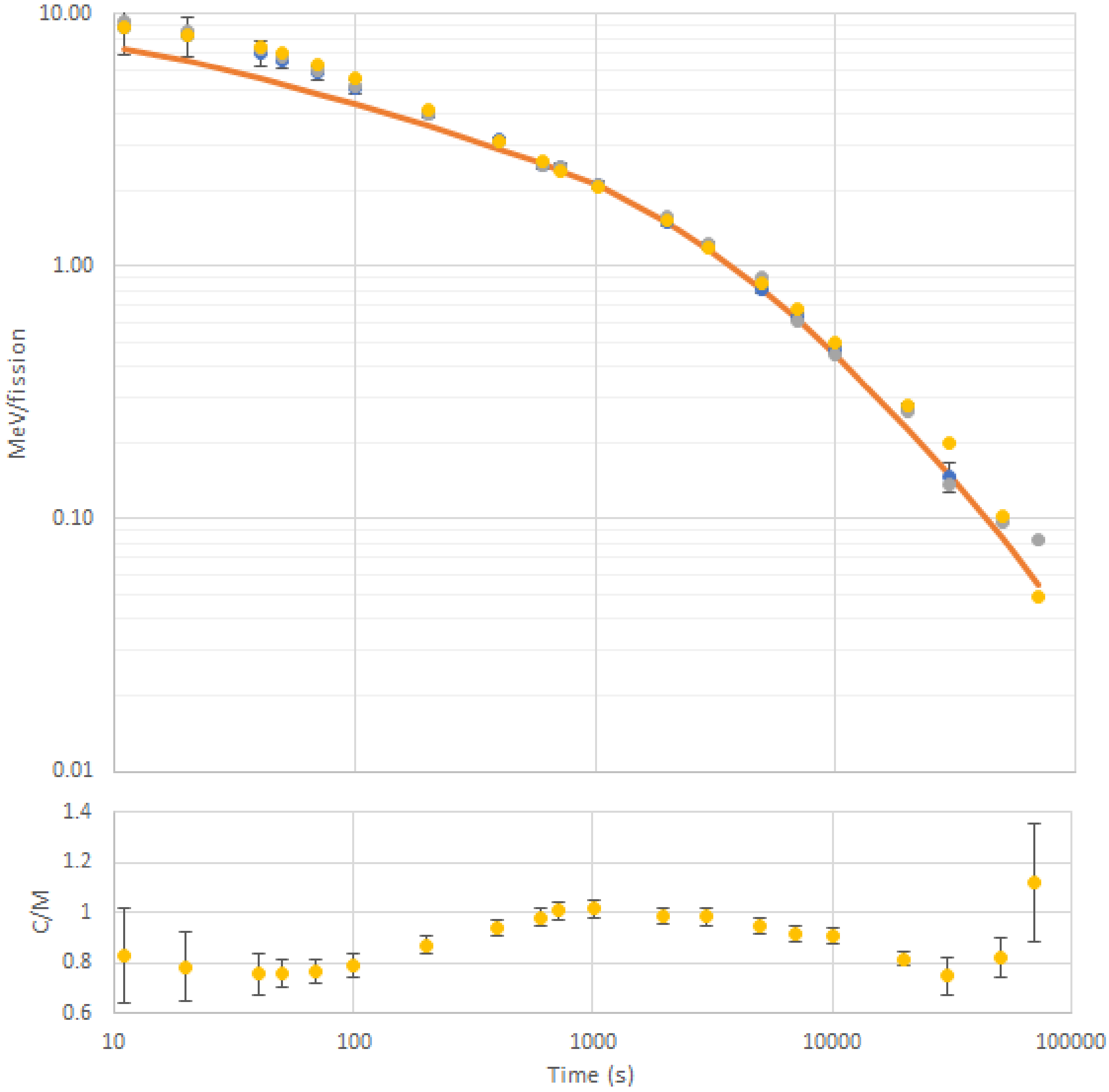

Figure 11. Measured (points) and calculated (solid line) total decay heat power following 4-hour irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Schrock et al. 1978.

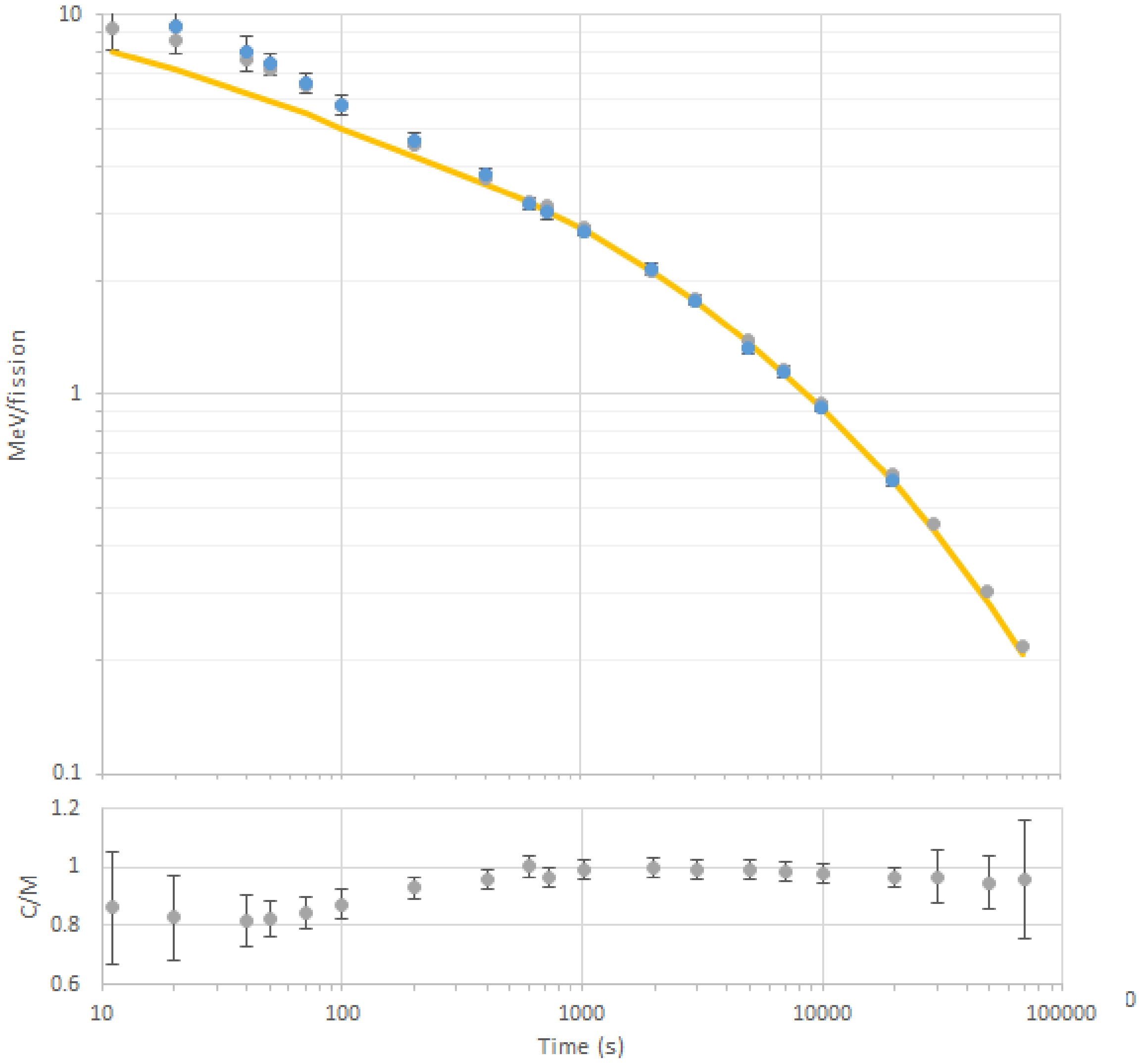

Figure 12. Measured (points) and calculated (solid line) total decay heat power following 22.35-hour irradiation of ^{235}U. The calculated to measured (C/M) ratio is shown on the bottom graph with experimental uncertainties. Measurements are from Schrock et al. 1978.

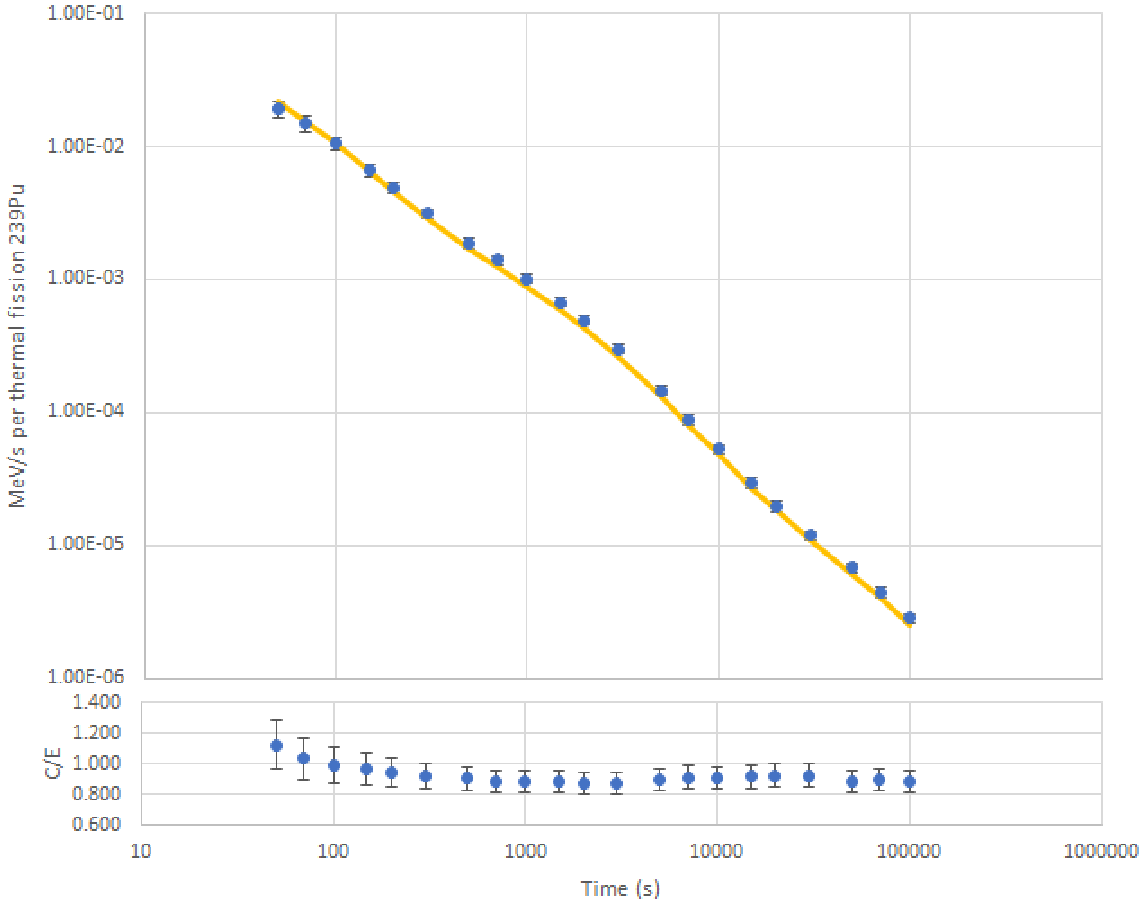

Figure 13. Comparison of measured and calculated decay heat for thermal ^{239}Pu fission. Measurements are from Fiche et al. 1976.

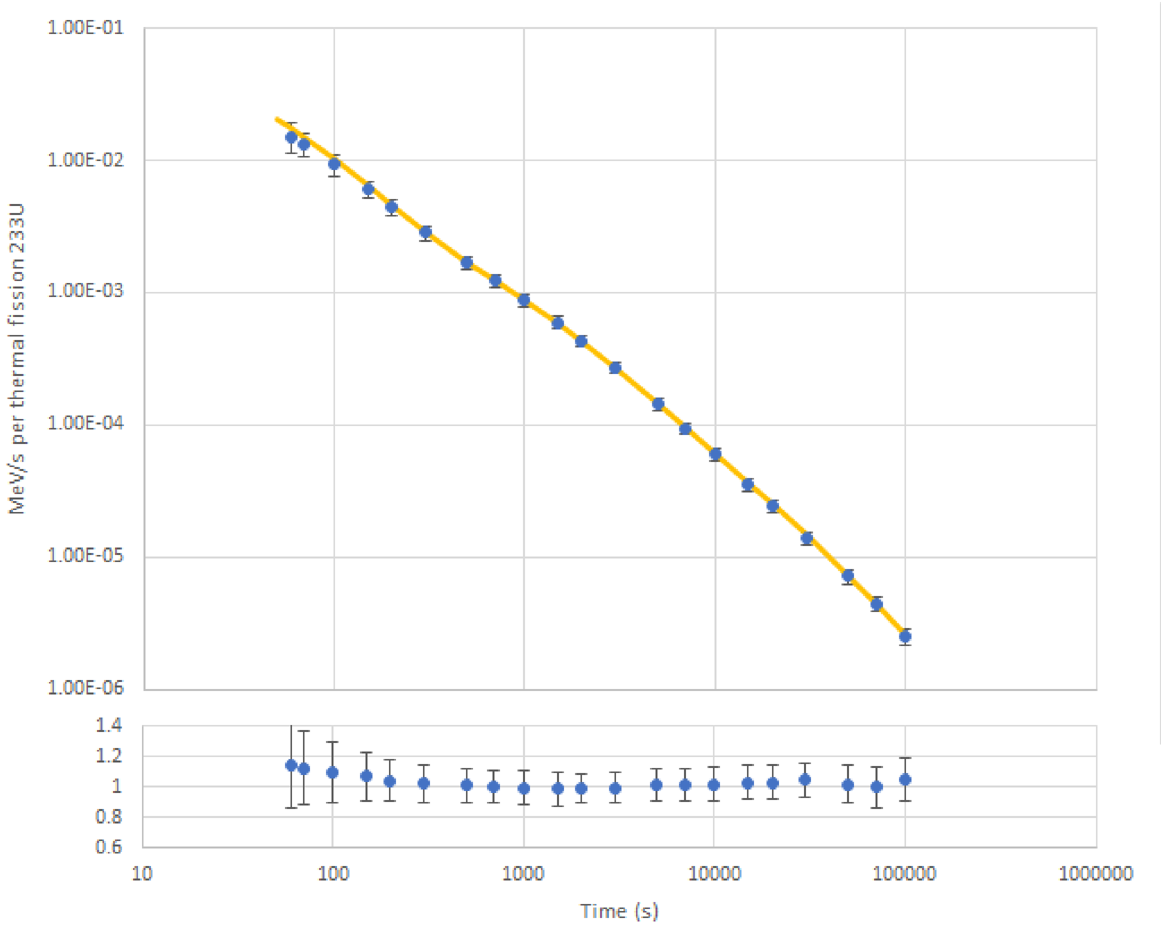

Figure 14. Comparison of measured and calculated decay heat for thermal ^{233}U fission. Measurements are from Fiche et al. 1976.

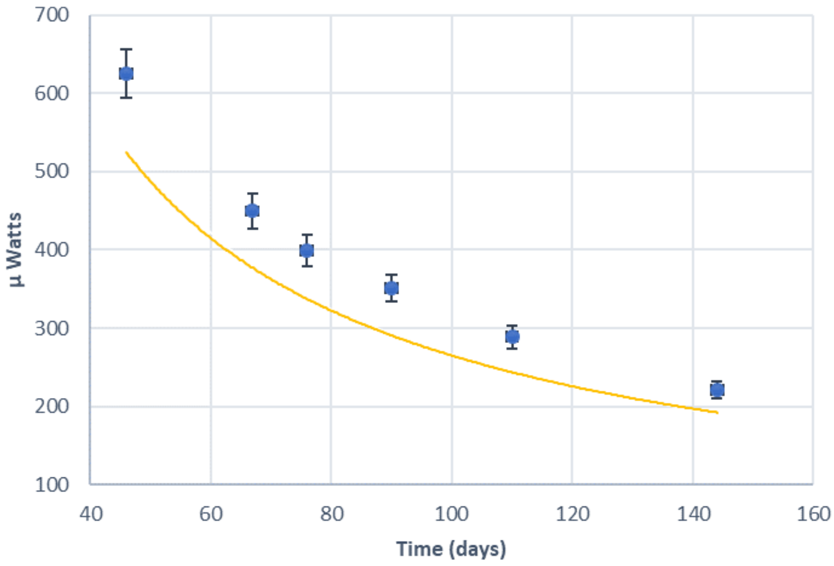

Figure 15. Comparison of measurements (points and error bars) and calculations (solid line) of fission product decay heat for plutonium sample 2. Measurements are from Johnson 1965.

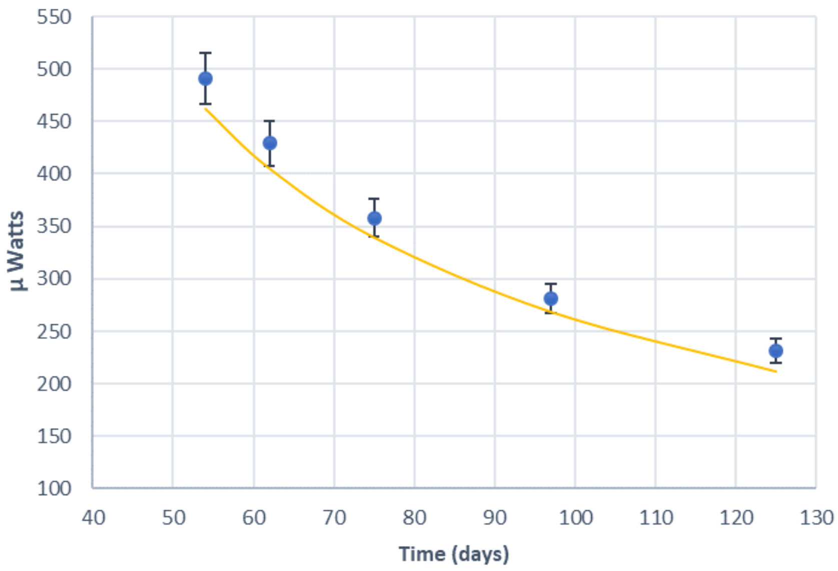

Figure 16. Comparison of measurements (points and error bars) and calculations (solid line) of fission product decay heat for plutonium sample 3. Measurements are from Johnson 1965.

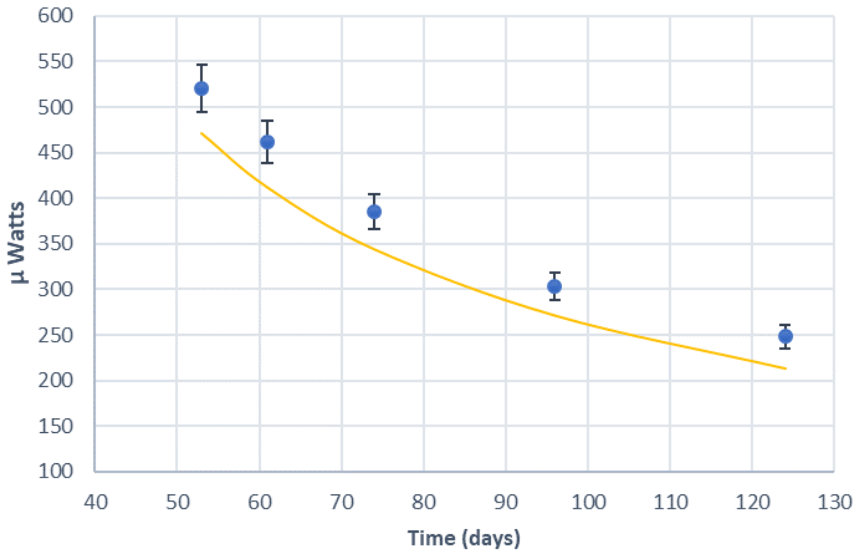

Figure 17. Comparison of measurements (points and error bars) and calculations (solid line) of fission product decay heat for plutonium sample 4. Measurements are from Johnson 1965.

Appendix A

Experiment identifiers and associated electronic data files

Table A.1 Summary of experiment identifiers and associated electronic data files

#	Author (First)	Institute	Nuclide	Year	Spec	Data file	Reference
1	Friesenhahn	IRT	235U	1976	T	U235-FR76-24h-total	EPRI-NP-180
2	Friesenhahn	IRT	235U	1976	T	U235-FR76-24h-gamma	EPRI-NP-180
3	Friesenhahn	IRT	235U	1976	T	U235-FR76-24h-beta	EPRI-NP-180
4	Friesenhahn	IRT	239Pu	1979	T	239PU-FR79-1000s	EPRI-NP-998
5	Friesenhahn	IRT	239Pu	1979	T	239PU-FR79-24h	EPRI-NP-998
6	Friesenhahn	IRT	235U	1979	T	235U-FR79-1000s	EPRI-NP-998
7	Friesenhahn	IRT	235U	1979	T	235U-FR79-20000s	EPRI-NP-998
8	Friesenhahn	IRT	235U	1979	T	235U-FR79-24h	EPRI-NP-998
9	Friesenhahn	IRT	235U	1979	T	235U-FR79-35d	EPRI-NP-998
10	Schrock	Berkeley	235U	1978	T	235U-SC76-1h-run1	EPRI-NP-616
11	Schrock	Berkeley	235U	1978	T	235U-SC76-1h-run12	EPRI-NP-616
12	Schrock	Berkeley	235U	1978	T	235U-SC76-4h-run3	EPRI-NP-616
13	Schrock	Berkeley	235U	1978	T	235U-SC76-4h-run4	EPRI-NP-616
14	Schrock	Berkeley	235U	1978	T	235U-SC76-4h-run9	EPRI-NP-616
15	Schrock	Berkeley	235U	1978	T	235U-SC76-22h-run6	EPRI-NP-616
16	Schrock	Berkeley	235U	1978	T	235U-SC76-22h-run8	EPRI-NP-616
17	Fiche	CEA	239Pu	1976	T	239PU-FI76	NEARCP-L-212
18	Fiche	CEA	233U	1976	T	233U-FI76	NEARCP-L-212
19	Johnson	UK AWRE	239Pu	1965	F	239PU-JO65-sample2	J. Nucl. En. 19
20	Johnson	UK AWRE	239Pu	1965	F	239PU-JO65-sample3	J. Nucl. En. 19
21	Johnson	UK AWRE	239Pu	1965	F	239PU-JO65-sample4	J. Nucl. En. 19
22	Gunst	Bettis	235U	1975	T	235U-GU75-sample25	Nuc Sci En 56:3
23	Gunst	Bettis	233U	1975	T	233U-GU75-sample45	Nuc Sci En 56:3
24	Gunst	Bettis	239Pu	1975	T	239PU-GU75-sample19	Nuc Sci En 56:3
25	Gunst	Bettis	232Th	1975	T	232TH-GU75-sample41	Nuc Sci En 56:3

Spec: T= thermal fission, F= fast fission