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Technical Basis for a Proposed Expansion of Regulatory Guide 3.54—Decay Heat Generation in an Independent Spent Fuel Storage Installation

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ABSTRACT

This report describes the technical basis for expanding the range of the current U.S. Nuclear Regulatory Commission regulatory guide for calculating decay heat power in an independent spent fuel storage installation (Regulatory Guide 3.54, Rev. 1) to include current high burnup fuel. As part of the expansion of the guide, a revised methodology is proposed to improve flexibility, enable increased accuracy, and cover a wider range of reactor operating histories. The methods are based more directly on the physics of decay heat generation and implement, as part of the proposed guide, procedures and data from consensus standards developed for calculating decay heat by the American National Standards Institute and the International Standards Organization, ANSI/ANS-5.1-2005 and ISO 10645:1992(E), respectively. The proposed guide is validated using experimental calorimeter decay heat data for 68 spent fuel assemblies measured in the United States and for 64 assemblies measured more recently in Sweden at the Interim Storage Facility for Spent Nuclear Fuel. Validation of the methods beyond the range of the decay heat measurements is supported using experimental isotopic assay measurements of the dominant decay heat-generating isotopes, obtained from destructive radiochemical analysis of spent fuel samples.

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LIST OF ACRONYMS AND UNITS

ABB	Asea Brown Boveri
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARIANE	Actinides Research in a Nuclear Element (Belgonucleaire program)
ARP	Automatic Rapid Processing
ATM	Approved Testing Material
BPR	burnable poison rod
BWR	boiling water reactor
CE	Combustion Engineering (now Westinghouse Electric Company)
CLAB	Swedish Central Interim Storage Facility for Spent Nuclear Fuel
DIN	Deutsches Institut für Normung (German Institute for Standardization)
DOE	U.S. Department of Energy
ENDF/B	Evaluated Nuclear Data Files, Part B
GE	General Electric Company
GE-Morris	General Electric Morris Operation spent fuel storage facility
GE-VNC	General Electric Vallecitos Nuclear Center
GWd/MTU	gigawatt-days per metric ton of uranium (unit of fuel burnup)
Hanford	Hanford Engineering Development Laboratory
ISFSI	independent spent fuel storage installation
ISO	International Standards Organization
JAERI	Japan Atomic Energy Research Institute (now Japan Atomic Energy Agency)
LWR	light water reactor
MALIBU	MOX and UOX LWR Fuels Irradiated to High Burnup (Belgonucleaire program)
NRC	U.S. Nuclear Regulatory Commission
PWR	pressurized water reactor
REBUS	Reactivity Tests for a Direct Evaluation of the Burnup Credit on Selected Irradiated LWR fuel bundles (Belgonucleaire program)
RG 3.54	NRC Regulatory Guide 3.54, revision 1
RSD	relative standard deviation
SCALE	Standardized Computer Analyses for Licensing Evaluation
SCK-CEN	Studiecentrum voor Kernenergie-Centre d'étude de l'Energie Nucléaire (Belgian Nuclear Research Center)
SKB	Svensk Kärnbränslehuntering AB (Swedish Nuclear Fuel and Waste Management Company)
W	Westinghouse Electric Company

1 INTRODUCTION

Energy in the form of gamma rays and beta and alpha particles emitted by the radioactive decay process following nuclear fission contributes to heat deposition in the fuel and surrounding structures. The residual energy, known as decay heat, is an important factor in the safety and design of nuclear facilities for assessment of system performance, including emergency core cooling systems, post-irradiation nuclear fuel handling operations, wet pool storage, interim dry storage, transportation, and final disposal in a repository. One of the principal design parameters for a spent fuel storage facility is the decay heat generated by the fuel. Title 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste," requires that spent fuel cladding be protected during storage against degradation that can lead to gross failures. Under certain environmental conditions, high temperatures have been shown to accelerate degradation of the fuel cladding leading to failures. Decay heat generation rates are therefore needed to determine that temperatures within the fuel, cladding, and shielding components will not exceed design and licensing specifications.

U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.54, revision 1 (RG 3.54) issued in 1999 (Ref. 1), provides a standard methodology for calculating decay heat generation rates in light water reactor (LWR) spent nuclear fuel for use as design input for independent spent fuel storage installations (ISFSIs). The guide enables the decay heat generation rates for an assembly to be determined with accuracies comparable to those of detailed isotopic depletion codes, but without the need for complex calculations. RG 3.54 is presently restricted to boiling water reactor (BWR) fuel with a maximum burnup of 45 GWd/MTU and pressurized water reactor (PWR) fuel of 50 GWd/MTU.

Over the past decade, more efficient fuel management strategies and increased capacity factors have resulted in progressively higher burnup fuel being discharged from commercial reactors in the United States, and this trend is expected to continue in the future as enrichments increase and fuel assembly designs and core management are further optimized. Spent fuel now being discharged from commercial reactors has moved well beyond the range of the regulatory guidance and beyond the regime where the computer code predictions of decay heat have been validated. Since 2002, the majority of fuel discharged from nuclear power plants operating in the United States has exceeded an assembly average burnup of 45 GWd/MTU, and as of 2007, roughly 90% of spent fuel being discharged will have a burnup exceeding this value. Therefore, there is a need to expand the range of application of the current regulatory guide and provide a defensible technical basis for the extension, supported by verified and validated experimental data.

This report documents the development of a proposed revision to RG 3.54 and the technical basis necessary to support its expansion to include present high burnup fuel. Spent fuel decay heat measurements recently performed at the Interim Storage Facility for Spent Nuclear Fuel in Sweden and previous measurements performed in the United States were used for validation of the proposed methods.

As part of the proposed expansion to the range of applicability of RG 3.54, changes to the methodology are recommended that build on the work of expert working groups of the American National Standards Institute (ANSI) and the International Standards Organization (ISO) that have developed and approved standards on decay heat. New procedures are designed to provide users with a methodology that has greater flexibility, covers a wider range of fuel characteristics and operating histories, is based more fundamentally on the underlying physics, and provides more accurate estimates of decay heat.

2 BACKGROUND

There are currently 43 licensed ISFSIs in the United States, a number that is expected to increase significantly in the next decade as new dry storage facilities are licensed and constructed to deal with storage pools that have reached design capacity. As spent fuel storage pools reach their capacities, fuel must be moved to interim dry storage to maintain operational capability. Onsite interim dry storage at nuclear power plants is a recognized effective approach to spent fuel management while a program for long-term disposition of the fuel is implemented.

New storage facilities will have to be designed and certified for new types of spent fuel and fuel assembly designs. One of the challenges faced by storage facility designers and regulators is that spent fuel now being discharged from nuclear plants has significantly higher enrichments and correspondingly higher burnup than previously seen. Advances in fuel design and core management optimization over the past several decades have led to routine discharge of reactor assemblies with average burnups of 50 GWd/MTU or more. Currently, most of the fuel discharged from nuclear plants routinely exceeds the range of validation data and the limits of application of the current regulatory guide for decay heat. The range of application of the current regulatory guide is based in large part on analysis of decay heat measurements performed in the United States at the Hanford Engineering Development Laboratory (Hanford) and the General Electric (GE) Morris Operation spent fuel storage facility (GE-Morris). The measurements were made on assemblies discharged between 1977 and 1982, making the data more than 25 years old. The measured assemblies had a maximum assembly burnup of 39 GWd/MTU, average burnup of about 25 GWd/MTU, and involved assembly designs no longer widely used. Furthermore, these measurements involved decay times of less than 11 years. No new measurements have been made in the United States since that time. Lack of new experimental data for modern assembly designs and spent fuel properties available for validating computational methods has been the primary impediment to expanding the limits of the current guide.

Recently, the Central Interim Storage Facility for Spent Fuel (CLAB), located in Sweden, initiated an experimental program to measure decay heat for spent fuel assemblies under a project managed by the Swedish Nuclear Fuel and Waste Management Company, Svensk Kärnbränslehuntering AB (SKB). SKB has performed decay heat measurements on more than 86 fuel assemblies having relatively modern designs, higher enrichment and burnup, and longer cooling times than previously available. Oak Ridge National Laboratory, with support of the NRC Office of Regulatory Research, has collaborated with SKB to assist in the computational analysis of the measurements performed at CLAB using the ORIGEN-S isotope depletion and decay code, part of the SCALE (Standardized Computer Analyses for Licensing Evaluation) code system (Ref. 2). The new measurements performed at CLAB greatly increase the amount of data available for code validation and allow the range of code application to be accurately quantified over a wide range of characteristics that includes many modern design fuels. The cooling times of the measured Swedish fuel assemblies, up to 28 years, also enable the accuracy of decay heat predictions to be quantified for the extended cooling times relevant to spent fuel aging in a storage facility. The new SKB data and the previously measured data in the United States provide the technical basis necessary to support the expansion of RG 3.54.

2.1 CURRENT REGULATORY GUIDE

RG 3.54, last revised in 1999, was developed with the goal of providing a procedure that specifies proper interpolation and adjustment formulas for a tabulated database of total decay heat for BWR and PWR fuel assemblies. The tabulated data were generated using validated computational models for typical spent fuel characteristics and reactor operating conditions. Calculations were performed using specific fuel enrichment-burnup values selected as being representative of nuclear fuel at that time. Fuels that

experienced irradiation conditions outside the limited range of the tabulated data are managed using empirical correction factors. The tabulated decay heat data are developed for cooling times that range from 1 year to 110 years after discharge.

Experimental programs designed to measure spent fuel assembly decay heat were conducted in the United States in the 1980s using calorimeters operated at GE-Morris and Hanford. These programs measured approximately 80 fuel assemblies with assembly designs available at that time. The measurements included spent fuel with a maximum burnup of 39 GWd/MTU and 27 GWd/MTU for PWR and BWR fuels, respectively. Selected measurements from these experiments were used to validate computer code predictions (Ref. 3) that are the technical basis of the decay heat values in the current NRC regulatory guide (RG 3.54). The experimental data that was available for validation limited the range of application of RG 3.54 to PWR fuel with an assembly burnup less than 50 GWd/MTU and BWR fuel with burnup less than 45 GWd/MTU. These burnup limits rely on considerable extrapolation beyond the range of available experimental data, and any further extrapolation cannot be justified without applying large penalties for uncertainty.

The safety margins applied to the interpolated data to account for uncertainties and bias in the computational methods are based on uncertainties developed from comparisons of calculations and experimental data. Uncertainties in the nuclear data used in the calculations and results of code intercomparisons form the basis of additional safety margins applied to extrapolated regimes beyond the range of the measurement data.

The methodology implemented in RG 3.54 also limits the range of application. The use of precalculated tabulated data developed for assumed operating conditions restricts the ability to accurately predict decay heat for assemblies that have operated outside the range of assumptions. For example, assemblies can experience long and short cycles, with and without trips. Irradiated assemblies can also be stored temporarily in a storage pool before being reinserted into the core. Such nonroutine exposure histories cannot be accurately represented in RG 3.54. In addition, tabulated decay heat data are compiled for predefined combinations of enrichment and discharge burnup. In actual plant operations, an unloaded fuel batch with a given enrichment will have a wide range of burnup values. The differences between actual and tabulated burnup is treated using a correction factor. However, the current guide is not applicable to assemblies with an average burnup that exceeds the tabulated values by more than about 50% because such large differences cannot be accurately represented using correction factors alone.

The data are also tabulated as a function of reactor operating power (steady state specific power), using typical operating cycle times and refueling times. Empirical correction factors are again used to adjust for excess power that is greater than the maximum specific power of the tabulations and for variations in the operating power during the last cycle. Operating power variations, particularly near the end of irradiations, have a significant effect at short cooling times and the correction factors become less accurate. For this reason the guide is limited to a minimum cooling time of 1 year.

The reactor operating history data used to develop the current guide was based on plant data at that time. Modern plants now discharge higher burnup fuel, operate with longer cycles and shorter refueling times, and have increased capacity factors. Because the decay heat data developed for RG 3.54 are calculated based on a specific assumed operating history and reactor conditions, changes in operating history cannot be readily incorporated in the methodology of the current guide.

2.2 PROPOSED REGULATORY GUIDE EXPANSION

The primary motivation for revising the regulatory guide is to extend the range of application to include higher burnup fuel from currently operating nuclear plants. The target limits of the revised guide include

fuel with enrichments up to 5 wt % and burnup limits of 55 GWd/MTU for BWR fuel and 65 GWd/MTU for PWR fuel. The ability to extend the range beyond the current limits is predicated on the availability of new experimental data from CLAB in Sweden that can be used to validate the decay heat calculations. Commensurate with the expansion, there was a desire to increase the accuracy of the methods by representing the physics processes of the individual components of decay heat and thereby provide greater flexibility to simulate a wide range of reactor operating conditions and exposure histories. Such an approach would thus allow the guidance to adapt more readily to changes in reactor operations without the need to completely revise decay heat tables.

A revised methodology for calculating decay heat from irradiated fuel assemblies is proposed and evaluated in this report. The methods involve calculating individually the contributions to decay heat from each of the following categories:

- fission products that are produced as a direct result of fission,
- nuclides that are produced by neutron capture on fission products,
- actinides generated by neutron capture of initial uranium isotopes in the fuel, and
- activation products in the assembly structure and cladding materials.

The concept of calculating decay heat from the sum of its individual components is a departure from the current regulatory guide, which represents the decay heat as a single integrated quantity. That is, the current guide combines the contributions from actinides, fission products, and activation products into a single tabulated value. Separating the contribution of each component allows methods to be developed that are based more fundamentally on the physics of the isotope generation and decay processes, since each component behaves differently as a function of irradiation and cooling time.

Energy release by fission products represents the largest component of decay heat during the first 50 years of cooling time for typical irradiated LWR nuclear fuels. The integrated energy released by fission products between 1 and 10 years represents approximately 80% of the total delayed energy and is the dominant source of decay heat power when decay heat rates are at their largest and potentially most limiting values for licensing. An accurate and flexible methodology for representing fission product decay heat is therefore highly desirable in characterizing decay heat generation in spent fuel for cooling times important for interim fuel storage applications. At longer cooling times, the contribution from fission products decreases and actinides become an increasingly important source of decay heat. However, fission products still contribute more than 60% of the integrated energy release between 1 and 110 years cooling time. The relative contribution of fission products, actinides, and activated assembly components to the total decay heat from is illustrated in Figure 2.1 for typical current LWR fuel for cooling times from 1 to more than 100 years. The fractional contribution to the total decay from fission product nuclides is shown in Figure 2.2, and actinides are shown in Figure 2.3.

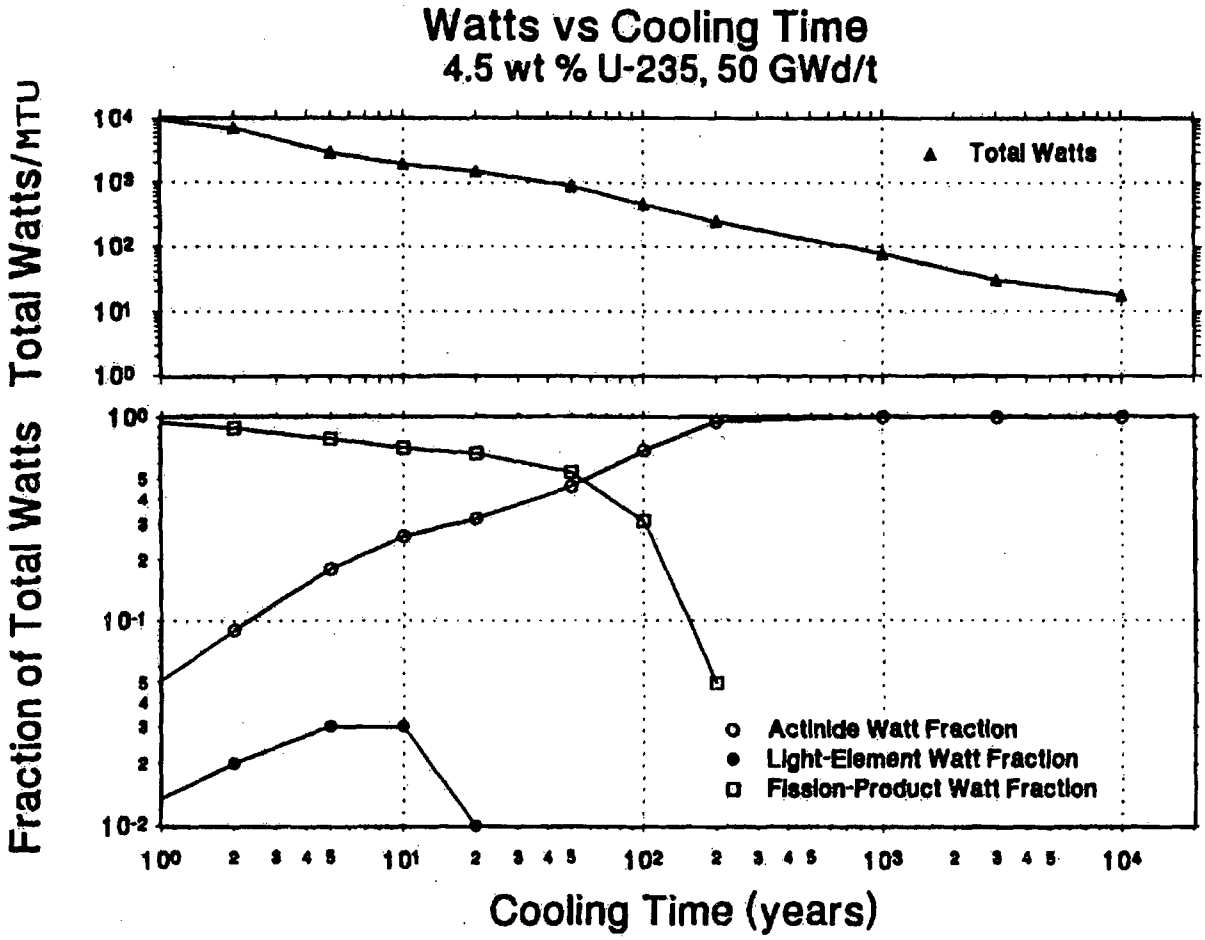

Figure 2.1. Total decay heat for spent nuclear fuel following discharge (top) and the fractional contributions of the actinides, fission products, and activated assembly structural components (bottom).

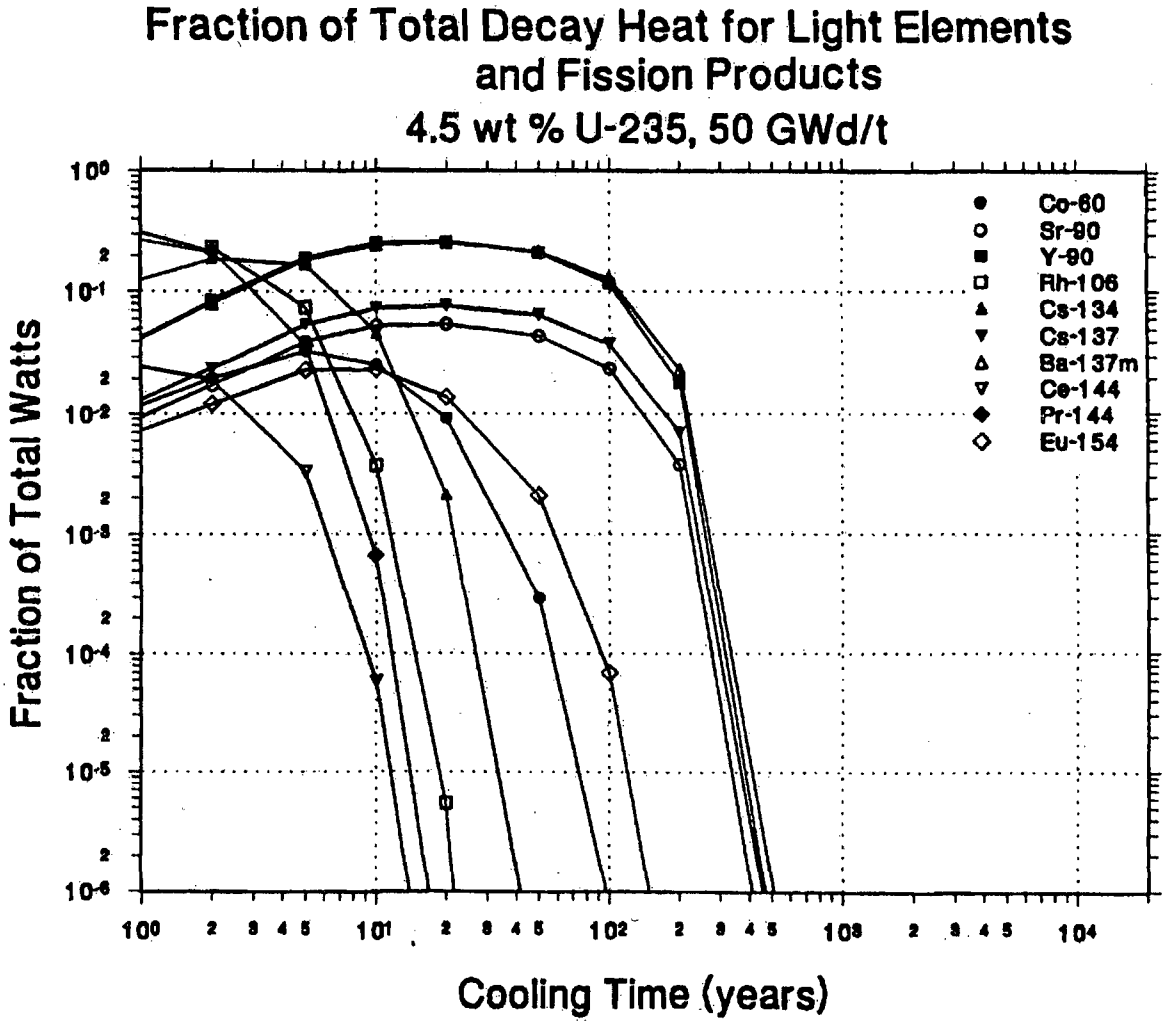

Figure 2.2. Fraction of the total decay heat due to the dominant fission products and activated structural nuclides.

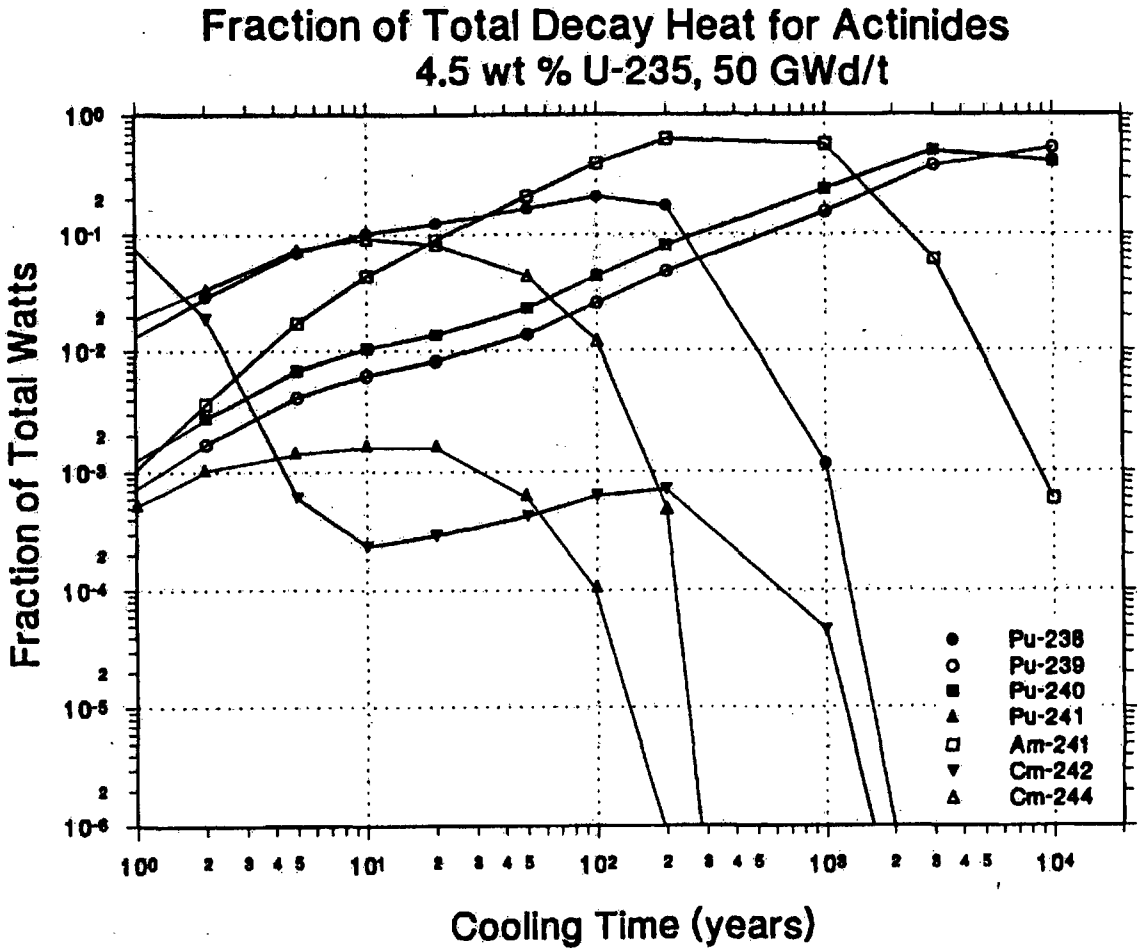

Figure 2.3. Fraction of the total decay heat due to the dominant actinides.

The fission product methodology proposed for the revised guide is based on methods widely adopted in domestic and international consensus decay heat standards, including ANSI/ANS-5.1 (Ref. 4), ISO 10645 (Ref. 5), and the Deutsches Institut für Normung (i.e., the German Institute for Standardization, DIN) standard DIN 25463 (Ref. 6). The proposed guide adopts the American National Standard for decay heat in LWRs, ANSI/ANS-5.1-2005, for the calculation of fission product decay heat power. These data have been subject to rigorous expert review and evaluation and are based directly on experimental data at short cooling times ($<10^5$ s) and on summation code calculations for the longer times that extend use of the standard to 10^{10} s, or about 300 years.

Standards are widely used to predict decay heat at short cooling times of interest to loss of coolant accidents and emergency core cooling system performance assessment. In this time frame the contribution from fission products represents most of the decay heat. Methods used to account for actinides and other components of decay heat are often either not explicitly prescribed in the procedures and are thus left to the user, or very conservative and easy-to-implement methods are developed. However, at longer cooling times where actinides can represent a large percentage of total decay heat, overly conservative methods can result in decay heat values that are drastically overestimated. Thus, current Standards-based approaches alone generally lack sufficient rigor or completeness to be accurately applied to all but the shortest cooling times, and a more complete methodology is required to accurately calculate decay heat generation rates for the range of times required for spent fuel storage applications.

To address the shortcomings of current standards for application to the longer cooling times of spent fuel storage, new methods and data are developed to more accurately address the nonfission product components of decay heat that become important in the time frames of the revised guide. ISO 10645 is used to guide the approach for some components of decay heat power addressed in the guide. However, a new methodology is required to analyze the actinides because of undue conservatism in the ISO 10645 methodology when applied to cooling times where actinides represent a significant contributing component to the total decay heat. The new methods are described and validated in this report as part of the technical basis for expanding and improving the regulatory guide.

Another important issue in developing a regulatory guide that implements procedures from current consensus standards is that standards typically require the user to make many choices pertaining to the implementation of the procedures and data for such things as irradiation history modeling, selection of fission energy values, determining the fraction of fission power for different fissioning nuclides, etc. Other components of decay heat, as noted, may be entirely omitted and left to the user to justify and include appropriately in the decay heat analysis. To implement a well-defined and structured approach for determining decay heat generation in the framework of a regulatory guide, the methods presented here involve a prescriptive set of procedures and define all data necessary to calculate all components of decay heat addressed in the guide.

2.3 DESCRIPTION OF VALIDATION STUDIES

Any expansion of the range of application of the current guide must have a defensible technical basis supported by sufficient verified and validated experimental data. Validation of the methods is based on comparisons of the proposed regulatory guide against decay heat measurements performed for more than 130 spent fuel assemblies. Experimental measurements of assembly decay heat have been made at the GE-Morris and Hanford in the United States and at CLAB, located in Oskarshamn, Sweden, and operated by SKB. These experiments involved decay heat measurements of full-length commercial fuel assemblies using a calorimeter.

Measurements performed in the United States between 1975 and 1986 have been widely reported and analyzed to validate computer model predictions of assembly decay heat (Refs. 7–11). A selected subset of these measurements that included 10 BWR and 10 PWR assemblies forms the validation database for the computational methods used to develop the data in the current version of RG 3.54 (Ref. 3). The complete set of measurements performed at GE-Morris and Hanford, used for validation of the proposed guide, includes the same 10 assemblies irradiated in PWR plants and a more extensive set of 58 assemblies irradiated in BWR plants.

Measurements performed in Sweden were made at CLAB in 2003 and 2004 (Ref. 12). The measurements used in this study included 30 assemblies irradiated in PWR plants and 34 assemblies irradiated in BWR plants (Ref. 13). Only measurements for standard commercial fuel assemblies were used for this analysis. The characteristics of the assemblies measured in the United States and Sweden that are used for methods validation in this report are summarized in Table 2.1. These same measurements have been used previously to benchmark the accuracy of the ORIGEN-S isotope generation and depletion code for spent fuel decay heat calculations (Ref. 14).

Table 2.1. Summary of decay heat measurements used for methods validation

Measurement facility[a]	Reactor	Assembly design	Number of assemblies	Enrichment (wt % ^{235}U)	Maximum burnup (GWd/MTU)
GE-Morris	Cooper	7 × 7	53	2.50	28.0
	Monticello	7 × 7	5	2.25	20.2
	Point Beach 2	14 × 14	6	3.40	39.4
Hanford	Turkey Point	15 × 15	4	2.56	28.6
CLAB	Ringhals 1	8 × 8	9	2.64, 2.91	44.9
	Ringhals 2	15 × 15	16	3.10–3.25	51.0
	Ringhals 3	17 × 17	14	2.10–3.40	47.3
	Barsebäck 1	8 × 8	2	2.92, 2.95	41.1
	Barsebäck 2	8 × 8	1	3.15	40.0
	Forsmark 1	8 × 8	2	2.09, 2.97	34.2
		9 × 9	3	2.94	37.9
	Forsmark 2	8 × 8	1	2.10	19.9
		SVEA 64 (8 × 8)	3	2.85, 2.92	32.8
	Forsmark 3	SVEA 100 (10 × 10)	2	2.77	31.3
	Oskarshamn 2	8 × 8	7	2.20–2.88	34.9
		SVEA 64 (8 × 8)	1	2.90	46.6
	Oskarshamn 3	8 × 8	1	2.58	35.6
		SVEA 100 (10 × 10)	2	2.71	40.4

[a] GE-Morris = General Electric Morris Operation spent fuel storage facility; Hanford = Hanford Engineering Development Laboratory; CLAB = the Swedish Central Interim Storage Facility for Spent Nuclear Fuel.

The measurements listed in Table 2.1 provide a relatively extensive database for validating the decay heat predictions of the proposed guide. The assemblies include a wide range of assembly design types, enrichments, burnup, and cooling times. The CLAB measurements include more modern assembly designs and higher burnup than the earlier data from the Hanford and GE-Morris measurements and also include assemblies with significantly longer cooling times. The Hanford and GE-Morris measurements cover cooling times from about 2.3 to 11.2 years after discharge. The CLAB measurements cover the range from 11.4 to 28 years. There are no measurements for fuel assemblies with cooling times less than 2 years.*

The calorimeter measurements provide direct experimental validation of the methods for cooling times from 2 years to about 28 years after shutdown. To provide validation beyond the cooling time range of the calorimeter data, measurements of isotopic concentrations made by destructive radiochemical analysis of spent nuclear fuel for the major actinides and fission products contributing to decay heat in spent nuclear fuel are evaluated. The variation in nuclide activity is governed in large measure by the nuclide half-lives

* Decay heat experiments involving smaller irradiated fuel samples have been performed for cooling times less than 1 day and used in the development of decay heat standards and for computer code validation.

in the cooling time domain of importance to interim storage. Therefore, the accuracy of calculations for individual isotopes can be used to establish the accuracy at decay times well beyond the time of the isotopic measurements themselves. Previous investigative studies (Refs. 15–17) have been undertaken to identify the major actinides and fission products that contribute to decay heat and their relative importance as a function of burnup and cooling time. For cooling times beyond the calorimeter data (28 years), the fission products ^{137}Cs and ^{90}Sr, and decay daughters ^{137m}Ba and ^{90}Y, contribute upwards of 90% of the fission product component of decay heat. The actinides ^{241}Am and ^{238}Pu are the major contributors after about 30 years, accounting for 70% to 90% of the total actinide decay heat. Therefore, validation of the predicted concentrations in spent fuel for a relatively small number of nuclides can provide indirect validation of decay heat calculations beyond the range where direct integral decay heat measurements are available.

The steady increase in enrichment and burnup of fuel assemblies means that the validated regime will invariably be surpassed as assembly designs and operating characteristics evolve. The maximum burnup assemblies in the database were those assemblies measured at CLAB. The highest burnup PWR assembly had a burnup of about 51 GWd/MTU. The highest burnup BWR assembly had a burnup of about 46 GWd/MTU. While the CLAB measurements greatly expand the database compared to previous measurements, the measured burnup range is below the target limits for a revised guide of 55 GWd/MTU and 65 GWd/MTU for BWR and PWR assemblies, respectively. To support validation of the proposed guide beyond the direct range of the calorimeter measurements, isotopic analysis results for the dominant decay heat generating nuclides are again applied to validate the proposed methods of the guide.

Evaluation of isotopic measurements for more than 40 PWR spent nuclear fuel samples with relatively extensive fission product and actinide measurements has recently been performed (Refs. 18–21). The measured fuel samples included in this study and the experimental programs under which they were acquired are summarized in Table 2.2. The accuracy of the isotopic measurements is dependent on the analytical methods used and the element/isotope. However, in general, the measurement accuracy for the major actinides (uranium and plutonium) and fission product nuclides is observed to be less than 5%, and is somewhat larger (10–20%) for the curium isotopes. The PWR isotopic measurements include fuel samples with enrichments up to 4.6 wt % ^{235}U and burnup values up to 70 GWd/MTU. BWR isotopic validation is less extensive (Ref. 22) and includes fewer measured isotopes than the PWR measurements. The results of the radiochemical isotopic measurements for the major decay heat isotopes are utilized in developing the data for the proposed guide and in deriving uncertainties in the calculated decay heat values and appropriate safety factors for application regimes of the guide that extend beyond the range of the calorimeter data.

Table 2.2. Summary of spent fuel nuclide concentration measurements

Reactor[a] (country)	Measurement facility[b] (country)	Experimental program[c]	Assembly design	Enrichment (wt % ^{235}U)	No. of samples	Nominal burnup (MWd/kgU)
TMI-1 (USA)	GE-VNC (USA)	DOE OCRWM (YMP)	15 × 15	4.66	8	22.8–29.9
Calvert Cliffs 1 (USA)	PNNL, KRI (USA, Russia)	DOE OCRWM (ATM)	14 × 14	3.04	3	27.4, 37.1, 44.3
Takahama 3 (Japan)	JAERI (Japan)	JAERI	17 × 17	4.11	9	24.4–47.3
Gösgen (Switzerland)	SCK-CEN, ITU (Belgium, Germany)	Belgonucleaire ARIANE	15 × 15	4.1	3	29.1, 52.5, 59.7
GKN II (Germany)	SCK-CEN	Belgonucleaire REBUS	18 × 18	3.8	1	54.0
Gösgen (Switzerland)	SCK-CEN (Belgium)	Belgonucleaire MALIBU	15 × 15	4.3	3	46.0, 50.8, 70.4

[a]TMI-1 = Three Mile Island Nuclear Power Plant, Unit 1; GKN II = Gemeinschaftskernkraftwerk Neckarwestheim Nuclear Power Plant, Unit II.

[b] GE-VNC = General Electric Vallecitos Nuclear Center; PNNL = Pacific Northwest National laboratory, KRI = Khlopin Radium Institute (St. Petersburg); JAERI = Japan Atomic Energy Research Institute (now Japan Atomic Energy Agency); SCK-CEN = Studiecentrum voor Kernenergie-Centre d'étude de l'Energie Nucléaire (Belgian Nuclear Research Center); ITU = Institute for Transuranium Elements.

[c]DOE OCRWM = U.S. Department of Energy, Office of Civilian Radioactive Waste Management; YMP = Yucca Mountain Project; ATM = Approved Testing Material; ARIANE = Actinides Research in a Nuclear Element; REBUS = Reactivity Tests for a Direct Evaluation of the Burnup Credit on Selected Irradiation Light Water Reactor Fuel Bundles; MALIBU = Mixed Oxide-Uranium Oxide Light Water Reactor Fuels Irradiated to High Burnup.

2.4 OUTLINE OF REPORT

This section has provided an overview of the technical issues and proposed methods for expanding and revising the guidance for predicting decay heat for times of interest to interim storage of spent nuclear fuel. The remainder of this report is organized as follows. Section 3 describes the methods and procedures for the proposed revision to RG 3.54. Section 4 discusses the development and technical basis for the revised procedures and data and the development of the safety factor. Section 5 presents comparisons of actual decay heat measurements for more than 130 assemblies with results obtained using the proposed guide. Finally, Sect. 6 presents a step-by-step example using the guide to calculate total decay heat for an assembly measured at GE-Morris. Tables containing information on the measured spent fuel assemblies, the measured decay heat values, and the calculations made using the proposed guide are summarized in Appendix A. Measured and calculated values of decay heat obtained using the current guide and the proposed revision of the guide are compared in Appendix B.

3 PROPOSED REGULATORY GUIDE

This section describes the methods and procedures for a proposed regulatory guide that can be used to evaluate the decay heat generation from nuclear fuel irradiated in LWRs. The range of application is restricted to cooling times appropriate to assessment of an ISFSI (i.e., >1 year). The methods apply only to LWR fuel initially containing uranium, mostly ^{235}U and ^{238}U, and application to recycled nuclear fuel containing initial uranium and plutonium mixtures is not permissible. The following components of decay heat generation are considered in the proposed guide:

- fission products produced by the fission of ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu;
- nuclides that are produced by neutron capture on fission products;
- actinides generated by neutron capture of initial uranium isotopes in the fuel; and
- activation products in the assembly structure and cladding materials.

The total decay heat is obtained from the sum of the individual components. Methods to evaluate each component are described separately. Section 3.1 provides a glossary of terms and definitions of variables and other parameters required to implement the guide. The procedures and data to calculate decay heat are given in Sect. 3.2. The subsections of 3.2 are organized according to each component of decay heat addressed by the proposed guide as follows: 3.2.1, fission products (excluding neutron capture); 3.2.2, neutron capture on fission products; 3.2.3, actinides; and 3.2.4, structural activation products. Section 3.3 discusses the limits and range of application for the proposed guide.

The proposed guide does not address decay heat from ^{239}U and ^{239}Np, which are included in consensus standards for decay heat because they are needed to accurately assess actinide decay heat at short cooling times of importance to postulated reactor accident analysis. These actinides and others, including ^{237}U and ^{238}Np, are not included in the analysis of actinide decay heat power in the guide because their half lives are sufficiently short that their contributions after 1 year of cooling are not significant.

Decay heat power from fission induced by delayed neutrons and spontaneous fission neutrons after shutdown is also not considered in the guide. Delayed neutrons emitted from the decay of short-lived fission products may induce fissions that continue to generate heat after shutdown. Although the fission rate from delayed neutrons is relatively small compared to operation, the prompt energy release by fission (~200 MeV) is much larger than the energy released by radioactive decay—about 8 MeV from beta particles and 7 MeV from gamma rays per fission. Delayed fission energy can represent a large component of decay heat at short times—in the time frame of seconds to minutes after fission, where the delayed neutron emission rate is large. Fission may also be caused by spontaneous fission and (α,n) neutron sources in the fuel after shutdown. Spontaneous fission and delayed neutrons are omitted from further consideration as their contribution to the total decay heat in the time frame covered by the guide is negligible.

3.1 GLOSSARY OF TERMS

The variables and terms used in this guide are defined in Table 3.1. The unit of time used throughout the guide is seconds to maintain consistency with the development of the standard for decay heat.

Table 3.1. Glossary of terms and variables

Variable[a]	Units	Description
T	s	Total reactor operating time
k	none	An index specifying an operating period at constant power
T_k	s	Operating time of the k^{th} irradiation interval
t	s	Time after final shutdown to the desired decay time
t_k	s	Time after operating period k to the desired decay time
$f_i(t)$	(MeV/s)/fission	Decay heat power, t seconds after a fission pulse from fissionable nuclide i
α_{ij}	none	Coefficients used to define decay heat power as 23 exponential terms
λ_{ij}	1/s	Exponent time constants used to define decay heat power as 23 exponential terms
$F_i(t,T)$	MeV/fission[a]	Decay heat power t seconds after an operating period of T seconds at constant fission rate of nuclide i in the absence of neutron capture in fission products
Q_i	MeV/fission	Total recoverable energy associated with fission of nuclide i
S_k	W/kgU[b]	Total specific thermal power from fission during operational period k
S_{ik}	W/kgU	Contribution from fission of nuclide i to the total specific thermal power during operational period k
$P_T(t,T)$	W/kgU	Total decay heat power from all contributions at t seconds after shutdown from an operating history of T seconds duration
$P_F(t,T)$	W/kgU	Total fission product decay heat power corresponding at t seconds after shutdown from an operating history of T seconds duration, uncorrected for neutron capture in fission products
$P_{Fi}(t,T)$	W/kgU	Fission product decay heat power contribution from i^{th} fissionable nuclide, uncorrected for neutron capture in fission products
$P_A(t,T)$	W/kgU	Contribution of actinides to the decay heat power
$P_C(t,T)$	W/kgU	Contribution of ^{134}Cs to the decay heat power
$P_E(t,T)$	W/kgU	Contribution to decay heat power from neutron capture by other fission products
$P_S(t,T)$	W/kgU	Contribution to decay heat power from activated structural components
$F_S(t)$	none	Safety factor applied to $P_T(t,T)$ to account for uncertainty in the methods
$P'_T(t,T)$	W/kgU	Total decay heat power with the safety factor included
E_s	wt % ^{235}U	Average initial enrichment of the fuel assembly
B_k	MWd/kgU	Average burnup of the fuel assembly following k^{th} irradiation interval
Subscripts		
i	none	Subscript referring to the fissionable isotopes ^{235}U, ^{239}Pu, ^{238}U, and ^{241}Pu
j	none	Subscript for the expansion terms of the exponential function for the decay heat power following a pulse fission (j=1 to 23)
k	none	Subscript denoting the individual irradiation time intervals, or cycles, in the power history

[a] Units are obtained from (MeV/s)/(fission/s).

[b] Units of watts per kilogram of uranium (W/kgU) are used for consistency throughout the guide for specific operational and decay heat power although in principle any unit of power may be used (1 W = 6.243×10^{12} MeV/s).

3.2 CALCULATION OF DECAY HEAT POWER

The proposed methodology defines procedures and data for all constants and variables required to calculate decay heat with this guide. This section describes the methods developed to calculate all components of decay heat necessary to obtain accurate estimates of total decay heat for the cooling times appropriate to this guide. Limitations of the methods pertaining to each component of decay heat are also addressed. The general limits and range of application of the proposed guide are discussed in Sect. 3.3.

3.2.1 Fission Products

Fission product decay heat (excluding neutron capture) is calculated using methods and data developed in the American National Standard for decay heat, ANSI/ANS-5.1-2005 (Ref. 4). These procedures apply to the calculation of decay heat for irradiated PWR and BWR assemblies. (A separate method to account for the effect of neutron capture by fission products is described in Sect. 3.2.2.)

The contribution of fission products to the decay heat power, uncorrected for neutron capture, is calculated from the individual contributions from fission of the four major fissionable isotopes in low-enriched uranium fuel: ^{235}U, ^{239}Pu, ^{238}U, and ^{241}Pu. These four actinides account for more than 99% of the fissions in typical LWR fuel. Fission of other isotopes is considered by treating them as ^{235}U, which is conservative for most cooling times. The method is based on a representation of fission product decay heat power following a single fission event. The time-dependent decay heat generation rate resulting from a single fission of nuclide i is represented as a summation series of 23 groups (exponential terms) of the form

$$f_i(t) = \sum_{j=1}^{23} \alpha_{ij} e^{-\lambda_{ij} t} \text{ (MeV/s)}, \tag{1}$$

where t is the time after fission and the coefficients α_{ij} and λ_{ij} are constants that depend on the fissionable isotope i. For an irradiation time interval of duration T and constant fission rate of 1 fission/s, the expression for $f(t)$ can be integrated analytically with the solution

$$F_i(t,T) = \sum_{j=1}^{23} \frac{\alpha_{ij}}{\lambda_{ij}} (1 - e^{-\lambda_{ij} T}) e^{-\lambda_{ij} t} \text{ (MeV/fission)}. \tag{2}$$

The units of $F_i(t,T)$ are MeV/fission (derived from MeV/s per fission/s).

For an irradiation history represented as a series of m irradiation time intervals, each interval having constant specific fission power S_{ik} over interval k and isotope i, the total fission product decay heat is determined from the sum over all irradiation intervals and fission isotopes using the expression

$$P_F(t,T) = \sum_{i=1}^{4} \sum_{k=1}^{m} \left[\frac{S_{ik}}{Q_i} \sum_{j=1}^{23} \frac{\alpha_{ij}}{\lambda_{ij}} (1 - e^{-\lambda_{ij} T_k}) e^{-\lambda_{ij} t_k} \right], \tag{3}$$

where the indices $i = 1, 2, 3,$ and 4 represent ^{235}U thermal fission, ^{239}Pu thermal fission, ^{238}U fast fission, and ^{241}Pu thermal fission. The quantity S_{ik} is the specific thermal operating power generated by fission of isotope i for irradiation interval k, and Q_i (MeV/fission) is the recoverable energy per fission (energy generating heat in the system). The units of decay heat power, $P_F(t,T)$, have the same units as the specific

operating power S. For purposes of consistency in the guide, the units of operating and decay heat power are defined as watts per kilogram of uranium (W/kgU).

A typical operating history consisting of three irradiation cycles with downtime included between each cycle is shown in Figure 3.1 to illustrate the relationship of the time variables t_k and T_k. Variations in reactor power during operation are taken into account by subdividing the operating history into intervals of constant power. For application of the methods to the range of cooling times applicable to this guide, it is adequate to represent each reactor operating cycle as an irradiation interval. The average specific power during cycle k, in units of watts per kilogram of uranium (W/kgU), is determined from the accumulated assembly burnup for the cycle divided by the irradiation time of the cycle

$$S_k = \frac{8.64 \times 10^{10} \ (B_k - B_{k-1})}{T_k} \ , \tag{4}$$

where B_k is the cumulative assembly burnup, in units of megawatt-days per kilogram of uranium (MWd/kgU), at the end of cycle k and T_k is the irradiation time of the cycle in seconds. The factor 8.64×10^{10} converts the time unit used to define the burnup from days to seconds and units of power from MW to W. The average specific power over the entire operating history of the fuel assembly is defined as

$$S_{avg} = \frac{8.64 \times 10^{10} \ B_{tot}}{T} \ , \tag{5}$$

where $T = \sum_{k=1}^{m} T_k$ is the total irradiation time, excluding downtime between operating cycles, and B_{tot} is the final discharge burnup of the assembly.

Tabulated 23-group coefficients for α_{ij} and λ_{ij} are listed in Table 3.2 for each fissionable nuclide. Recommended values for Q_i for each fissionable nuclide are listed in Table 3.3. The relative power fractions for each of the four fissionable nuclides (S_i/S) are given in Table 3.4 as a function of initial enrichment and burnup of the fuel. These values are applied to fuel from both PWR and BWR reactor types. The power fractions are obtained from the data in Table 3.4 by linear interpolation of the tabulated assembly average enrichment, E_s, and the accumulated assembly burnup calculated at the midpoint of each irradiation cycle k.

It is important to ensure that the fuel burnup obtained from the time-integrated specific power of the histogram in Figure 3.1 equals the actual burnup of the fuel. The user must also ensure that the sum of relative power fractions S_i/S for each irradiation interval is not less than unity (due to interpolation error). In this case the user should increase the power faction of ^{235}U to preserve the correct total operating power for the interval.

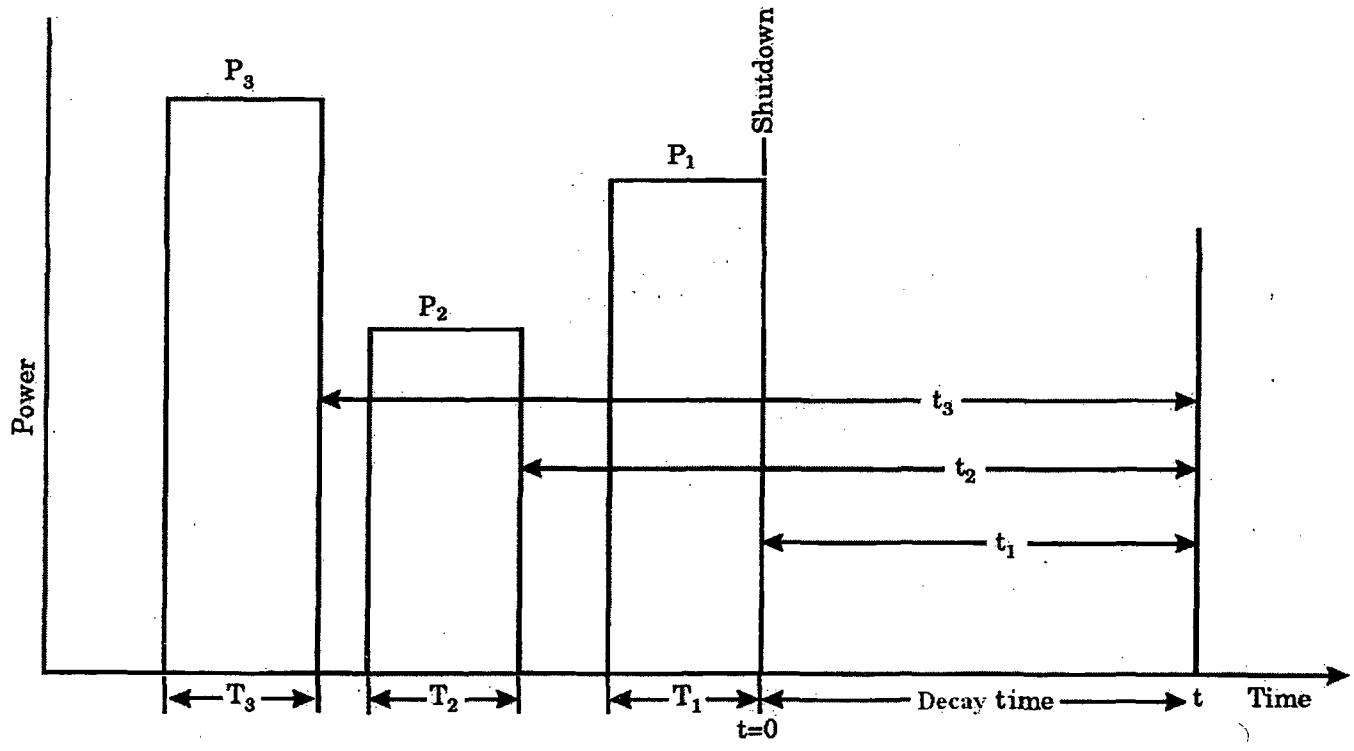

Figure 3.1. Example of a three-cycle reactor operating history.

Table 3.2. Coefficients[a] for thermal fission of ^{235}U, ^{239}Pu, ^{241}Pu, and fast fission of ^{238}U

Term index j	^{235}U (thermal[b])		^{239}Pu (thermal)		^{238}U (fast)		^{241}Pu (thermal)	
	α_{1j}	λ_{1j}	α_{2j}	λ_{2j}	α_{3j}	λ_{3j}	α_{4j}	λ_{4j}
1	5.2800E-04[c]	2.7216E+00	1.6540E-01	8.9246E+00	3.9368E-01	4.3427E+00	3.0934E-01	2.9049E+00
2	6.8588E-01	1.0256E+00	3.6928E-01	6.9005E-01	7.4588E-01	1.7114E+00	5.4434E-01	6.4911E-01
3	4.0752E-01	3.1419E-01	2.4006E-01	2.3618E-01	1.2169E+00	6.0572E-01	4.0782E-01	2.5569E-01
4	2.1937E-01	1.1788E-01	1.0269E-01	1.0118E-01	5.2820E-01	1.9429E-01	1.5828E-01	8.7123E-02
5	5.7701E-02	3.4365E-02	3.4916E-02	3.7193E-02	1.4805E-01	6.9788E-02	4.1577E-02	2.5068E-02
6	2.2530E-02	1.1762E-02	2.2961E-02	1.4319E-02	4.5980E-02	1.8809E-02	1.4818E-02	1.3323E-02
7	3.3392E-03	3.6065E-03	3.9070E-03	4.5094E-03	1.0406E-02	6.1265E-03	5.8176E-03	6.3772E-03
8	9.3667E-04	1.3963E-03	1.3080E-03	1.3211E-03	1.6991E-03	1.3799E-03	1.9482E-03	2.0221E-03
9	8.0899E-04	6.2608E-04	7.0265E-04	5.3481E-04	6.9102E-04	5.2799E-04	9.5196E-04	6.2933E-04
10	1.9572E-04	1.8924E-04	1.4297E-04	1.7297E-04	1.4736E-04	1.6145E-04	1.8208E-04	1.7462E-04
11	3.2609E-05	5.5074E-05	1.7642E-05	4.8918E-05	2.4049E-05	4.8419E-05	1.5310E-05	4.0172E-05
12	7.5827E-06	2.0971E-05	7.3646E-06	2.0155E-05	6.9288E-06	1.5644E-05	4.5039E-06	1.5289E-05
13	2.5189E-06	9.9940E-06	1.7720E-06	8.3687E-06	6.4927E-07	5.3610E-06	9.8277E-07	7.6113E-06
14	4.9836E-07	2.5401E-06	5.4945E-07	2.3620E-06	4.3556E-07	2.1689E-06	5.1832E-07	2.5083E-06
15	1.8523E-07	6.6332E-07	1.6736E-07	6.4594E-07	1.6020E-07	6.3343E-07	2.3018E-08	1.1312E-06
16	2.6592E-08	1.2281E-07	2.1160E-08	1.2822E-07	2.3089E-08	1.2879E-07	1.5817E-07	6.2987E-07
17	2.2356E-09	2.7163E-08	2.9388E-09	2.5166E-08	2.5481E-09	2.5604E-08	1.8074E-08	1.3149E-07
18	8.9582E-12	3.2955E-09	1.3659E-10	1.3176E-08	3.5071E-11	9.1544E-09	3.6922E-09	2.4237E-08
19	8.5968E-11	7.4225E-10	5.7450E-11	7.3568E-10	6.3399E-11	7.3940E-10	5.3843E-11	9.6433E-09
20	2.1072E-14	2.4681E-10	3.8422E-14	2.4663E-10	4.1599E-14	2.4731E-10	5.3003E-11	7.3467E-10
21	7.1219E-16	1.5596E-13	1.8030E-16	3.3490E-13	5.3295E-16	1.9594E-13	4.8358E-14	2.4827E-10
22	8.1126E-17	2.2573E-14	1.8342E-15	1.8761E-13	1.6695E-18	6.4303E-14	9.8516E-16	1.6873E-13
23	9.4678E-17	2.0503E-14	1.9884E-16	3.1544E-14	4.1058E-16	6.4229E-14	1.3076E-16	8.3639E-15

[a] Tabulated values from ANSI/ANS-5.1-2005.
[b] Energy of neutron-induced fission.
[c] Read as 5.2800×10^{-4}.

Table 3.3. Recommended fission energy values for application to decay heat analysis

Actinide	Energy of fission	Energy per fission[a] (MeV)
^{235}U	Thermal	202.2
^{238}U	Fast	205.5
^{239}Pu	Thermal	211.2
^{241}Pu	Thermal	213.7

[a] Values from ISO 10645:1992(E).

Table 3.4. Power fractions for fission of ^{235}U, ^{239}Pu, ^{238}U, and ^{241}Pu

Burnup (MWd/kgU)	2 wt % ^{235}U				3 wt % ^{235}U			
	^{235}U	^{239}Pu	^{238}U	^{241}Pu	^{235}U	^{239}Pu	^{238}U	^{241}Pu
0	0.931	0.000	0.069	0.000	0.939	0.000	0.061	0.000
5	0.652	0.264	0.074	0.010	0.755	0.176	0.065	0.004
10	0.498	0.384	0.079	0.039	0.632	0.278	0.070	0.020
15	0.386	0.456	0.084	0.074	0.535	0.348	0.074	0.043
20	0.298	0.505	0.089	0.108	0.453	0.401	0.078	0.068
25	0.229	0.540	0.093	0.138	0.381	0.443	0.082	0.094
30	0.174	0.567	0.096	0.163	0.316	0.479	0.086	0.119
35	0.131	0.586	0.099	0.184	0.260	0.508	0.090	0.142
40	0.097	0.601	0.102	0.200	0.211	0.534	0.093	0.162
45	0.072	0.612	0.103	0.213	0.169	0.555	0.096	0.180
50	0.053	0.620	0.105	0.222	0.132	0.573	0.099	0.196
55	0.039	0.626	0.106	0.229	0.103	0.588	0.101	0.208
60	0.028	0.630	0.107	0.235	0.079	0.600	0.103	0.218
65	0.020	0.633	0.108	0.239	0.061	0.609	0.104	0.226
Burnup (MWd/kgU)	**4 wt % ^{235}U**				**5 wt % ^{235}U**			
	^{235}U	^{239}Pu	^{238}U	^{241}Pu	^{235}U	^{239}Pu	^{238}U	^{241}Pu
0	0.943	0.000	0.057	0.000	0.946	0.000	0.054	0.000
5	0.808	0.129	0.061	0.002	0.842	0.100	0.057	0.001
10	0.711	0.213	0.064	0.012	0.762	0.170	0.060	0.008
15	0.630	0.275	0.068	0.027	0.694	0.225	0.063	0.018
20	0.560	0.324	0.071	0.045	0.634	0.269	0.066	0.031
25	0.496	0.365	0.074	0.065	0.578	0.307	0.069	0.046
30	0.436	0.401	0.078	0.085	0.524	0.341	0.072	0.063
35	0.380	0.433	0.081	0.106	0.476	0.371	0.074	0.079
40	0.329	0.462	0.084	0.125	0.428	0.399	0.077	0.096
45	0.280	0.488	0.088	0.144	0.382	0.425	0.080	0.113
50	0.235	0.512	0.091	0.162	0.337	0.450	0.083	0.130
55	0.195	0.533	0.094	0.178	0.293	0.474	0.086	0.147
60	0.160	0.552	0.096	0.192	0.253	0.496	0.089	0.162
65	0.128	0.569	0.099	0.204	0.214	0.517	0.092	0.177

3.2.2 Neutron Capture by Fission Products

The calculation of decay heat power from fission products described in Sect. 3.2.1 does not account for neutron capture by fission products during irradiation of fuel in the reactor. Neutron irradiation of the nuclides produced directly by fission can have two effects: (1) reduction of the concentration of direct-yield fission products with large cross sections and (2) activation of stable and/or longer-lived fission products and daughters resulting in an increased concentration of unstable nuclides. Because neutron absorption by fission products leads to product nuclides farther from the line of stability, the net effect of absorption is to increase the decay heat.

At cooling times in the region of 10^8 seconds (~3 years), production of ^{134}Cs through neutron capture by the stable fission product ^{133}Cs can represent a significant contribution to decay heat. The only significant production route to ^{134}Cs is via neutron capture by ^{133}Cs. The isotope ^{134}Cs is not produced by direct fission because the decay mass chain ends with stable ^{134}Xe. Because ^{134}Cs is the dominant decay heat generating nuclide resulting from neutron capture, it is treated explicitly in the guide using the methodology developed in the ISO 10645 standard. The contributions from other neutron capture products are conservatively treated as an aggregate in the proposed guide using a bounding correction factor.

3.2.2.1 Contribution of ^{134}Cs

Neutron capture by the stable fission product ^{133}Cs produces ^{134}Cs, which has a half-life of 2.06 years. The concentration of stable ^{133}Cs as a function of irradiation time is represented analytically by the production from fission and the rate of removal by neutron capture. The decay heat generated by the decay of ^{134}Cs resulting from neutron capture on ^{133}Cs, in units of W/kgU, is given by the equation

$$P_c(t,T) = Y\,E\,\lambda_4\,\frac{S}{Q}\left[\frac{1-e^{-(\lambda_4+\sigma_4\phi)T}}{\lambda_4+\sigma_4\phi}+\frac{e^{-\sigma_3\phi T}-e^{-(\lambda_4+\sigma_4\phi)T}}{\sigma_3\phi-(\lambda_4+\sigma_4\phi)}\right]e^{-\lambda_4 t}\,, \qquad (6)$$

where the average fission rate is $\frac{S}{Q}=\sum_{i=1}^{4}\frac{S_i}{Q_i}$, t is the time after discharge in seconds, T is the irradiation interval time in seconds, and ϕ is the total neutron flux (n/cm^2/s). The other parameter constants used in Eq. (6) are defined as follows.

Variable	Value	Description
Y	6.83 %	Effective cumulative ^{133}Cs yield per fission
λ_4	1.071 x 10^{-8} s^{-1}	Decay time constant of ^{134}Cs
σ_3	11.3 barns[†]	Spectrum average (n,γ) cross section of ^{133}Cs
σ_4	10.9 barns	Spectrum average absorption cross section of ^{134}Cs
E	1.720 MeV	Recoverable energy per decay for ^{134}Cs

[†] 1 barn = 10^{-24} cm^2

The one-group capture cross sections σ_3 and σ_4 are determined for a typical PWR fuel spectrum. When applied to BWR fuel they yield conservative results. The cross-section values are not those recommended in ISO 10645, but are generated from ENDF/B-VII data evaluations and yield improved estimates of ^{134}Cs production.

The analytical equation is exact for a single irradiation interval of flux ϕ_k and duration T_k. For an operating history with m irradiation intervals, the value for T is determined as the total irradiation time (conservatively excluding any downtime) and ϕ is determined as the average flux value over all time intervals such that

$$\varphi = \frac{1}{T}\sum_{k=1}^{m}\varphi_k T_k \ , \tag{7}$$

and the average fission rate over all irradiation intervals and isotopes is determined as

$$\frac{S}{Q} = \frac{1}{T}\sum_{k=1}^{m}\sum_{i=1}^{4}\frac{S_{ik}}{Q_i}T_k \ . \tag{8}$$

The neutron flux in the fuel is approximated using the relationship

$$\phi_k = \frac{S_{avg}}{\alpha}\ 2.58\times 10^{10}\ (\text{n/cm}^2\text{/s}) \ , \tag{9}$$

where S_{avg} is the specific power density given in Eq. (5), in units of watts per kilogram of initial uranium and α is the effective enrichment, calculated from the actual fuel enrichment E_s, expressed as initial ^{235}U weight percent in total uranium, using the equation $\alpha = (E_s/2) + 1$.

For enrichments and burnup values typical of LWRs, the flux calculated using these equations yields values of the ^{134}Cs contribution to decay heat that exceed the exact values by up to 5%. For lower burnup values, less than 25 MWd/kgU, the expression will overestimate the ^{134}Cs contribution by up to 15%.

3.2.2.2 Contribution of Other Neutron Capture Nuclides

The contribution from neutron capture on fission products, excluding ^{133}Cs, is determined using the tabulated factors of $H(t)$ given in Table 3.5 as a function of cooling time. The factors are multiplied by the decay heat power due to the direct fission products, $P_F(t,T)$, evaluated in Sect. 3.2.1 using Eq. (3), according to the equation

$$P_E(t,T) = H(t)\ P_F(t,T) \ . \tag{10}$$

The values of $H(t)$ are developed to yield conservative results provided that the following conditions are met:

- the initial enrichment is between 2.0 and 5.0 wt %;
- burnup, in units of megawatt-days per kilogram of uranium, is less than 14 times the initial enrichment in units of weight percent ^{235}U; and
- the power density, in units of kilowatts per kilogram of uranium, is less than 5 times the initial enrichment in units of weight percent ^{235}U .

This parameter range is adequate to cover most spent nuclear fuel assemblies discharged from commercial reactors operating in the United States (see discussion in Sect. 3.3).

Table 3.5. Correction factors for fission product neutron capture (excluding ^{133}Cs), activation products, and the safety factor

t (s)	*t (years)*[a]	*H(t)*	*A(t)*	*$F_s(t)$*
3.0×10^7	0.951	0.012	0.028	1.020
4.0×10^7	1.268	0.014	0.032	1.020
6.0×10^7	1.901	0.019	0.042	1.020
8.0×10^7	2.535	0.023	0.052	1.020
1.0×10^8	3.169	0.029	0.063	1.020
1.5×10^8	4.753	0.036	0.079	1.020
2.0×10^8	6.338	0.038	0.081	1.020
3.0×10^8	9.506	0.037	0.064	1.020
4.0×10^8	12.68	0.033	0.050	1.020
6.0×10^8	19.01	0.024	0.023	1.020
8.0×10^8	25.35	0.017	0.014	1.020
1.0×10^9	31.69	0.011	0.007	1.022
2.0×10^9	63.38	0.002	0.001	1.034
3.0×10^9	95.06	0.000	0.001	1.045
4.0×10^9	126.8	0.000	0.001	1.057

[a] 1 year = 3.1536×10^7 s

3.2.3 Actinides

The decay heat from actinides is calculated as the sum of contributions from ^{241}Am, ^{242}Cm, ^{244}Cm, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu. These seven actinides contribute more than 99.5% of the total actinide decay heat from 30 days to more than 200 years after discharge. The time-dependent contribution of the actinide decay heat component at time t after irradiation is calculated analytically according to the formula

$$P'_A(t) = \sum_{n=1}^{7} \hat{\beta}_n e^{-\lambda_n t} , \qquad (11)$$

where the index n corresponds to the actinides ^{241}Am, ^{241}Pu, ^{240}Pu, ^{239}Pu, ^{238}Pu, ^{244}Cm, and ^{242}Cm, λ_n is the physical decay constant (s^{-1}) of actinide n, t is the time after discharge (s), and $\hat{\beta}_n$ are coefficients calculated as

$$\hat{\beta}_1 = \beta_1 - \beta_2 \frac{E_1}{E_2} \frac{\lambda_1}{(\lambda_1 - \lambda_2)}$$

$$\hat{\beta}_2 = \beta_2 \left[1 + \frac{E_1}{E_2} \frac{\lambda_1}{(\lambda_1 - \lambda_2)} \right] \qquad (12)$$

$$\hat{\beta}_n = \beta_n \qquad n = 3,\ldots,7$$

The variables E_1 and E_2 are the values for recoverable energy per decay, where E_1 is the thermal energy released per decay for ^{241}Am ($5.629\ MeV$) and E_2 is the thermal energy released per decay for ^{241}Pu ($5.361 \times 10^{-3}\ MeV$).

The actinide coefficients β_n for PWR fuel are listed in Table 3.6 for tabulated values of initial ^{235}U enrichment, E_s, and final assembly burnup, B_{tot}. Coefficients for BWR fuel are given in Table 3.7. Intermediate values of enrichment and burnup are obtained by linear interpolation between the tabulated data. The actinide coefficients have standard units of decay heat power, i.e., watts per kilogram of initial uranium. The β_n coefficients physically represent the effective decay heat generation rate from each actinide, extrapolated to the time of discharge. The $\hat{\beta}_n$ coefficients, calculated in Eq. (12), account for the decay heat generated by the decay of ^{241}Am that is produced from the decay of its parent ^{241}Pu (half-life of 14.4 years) after discharge. The energy released per decay from ^{241}Am is more than 10^3 times greater than its parent ^{241}Pu, and the decay heat generated by in-growth of ^{241}Am rapidly becomes a dominant actinide source with increasing cooling time and is the major nuclide contributing to actinide decay heat power after about 50 years (see Figure 2.3).

The average specific operating power used to generate the coefficients in Tables 3.6 and 3.7 is 20 kW/kgU. The actinide decay heat power increases as the specific power decreases for cooling times greater than about 30 days. A correction factor is applied to the calculated actinide decay heat in Eq. (11) to account for variations in operating specific power over the range of 12 to 50 kW/kgU using the equation

$$P_A(t) = P_A'(t) \times 1.82 [S_{avg}]^{-0.06} \ . \qquad (13)$$

Note that the units of specific power applied in Eq. (13) are watts per kilogram of initial uranium. For cooling times less than 3 years and specific operating powers greater than 30 kW/kgU, the correction factor leads to conservative estimates of the actinide decay heat power contribution by up to 15%. However, in this cooling time range, actinides typically contribute less than 20% of the total decay heat power and the method does not result in undue conservatism in the total decay heat generation rate.

Table 3.6. Parameters and coefficients for calculating actinide decay heat for PWR fuel

Index n		1	2	3	4	5	6	7
Nuclide		^{241}Am	^{241}Pu	^{240}Pu	^{239}Pu	^{238}Pu	^{244}Cm	^{242}Cm
Decay constant λ_n (1/s)		5.078E-11	1.531E-09	3.347E-12	9.111E-13	2.504E-10	1.213E-09	4.923E-08
Enrichment	Burnup (MWd/kgU)	Actinide coefficients β_n (W/kgU)						
2 wt %	10	7.068E-04	1.166E-03	5.992E-03	7.102E-03	6.954E-03	8.070E-04	1.182E-01
	20	3.811E-03	2.982E-03	1.272E-02	9.070E-03	4.025E-02	2.275E-02	1.196E+00
	30	6.943E-03	4.273E-03	1.758E-02	9.722E-03	1.013E-01	1.295E-01	2.751E+00
	40	9.432E-03	5.053E-03	2.082E-02	1.002E-02	1.753E-01	3.821E-01	4.167E+00
	50	1.104E-02	5.529E-03	2.296E-02	1.021E-02	2.491E-01	8.029E-01	5.123E+00
	65	1.210E-02	5.832E-03	2.446E-02	1.037E-02	3.137E-01	1.374E+00	5.705E+00
3 wt %	10	4.968E-04	8.284E-04	4.395E-03	6.911E-03	5.455E-03	2.734E-04	6.253E-02
	20	3.321E-03	2.525E-03	1.034E-02	9.467E-03	3.299E-02	9.358E-03	7.886E-01
	30	6.968E-03	3.990E-03	1.543E-02	1.040E-02	9.066E-02	6.223E-02	2.159E+00
	40	1.022E-02	4.971E-03	1.932E-02	1.071E-02	1.720E-01	2.109E-01	3.728E+00
	50	1.202E-02	5.600E-03	2.202E-02	1.078E-02	2.654E-01	5.003E-01	4.945E+00
	65	1.345E-02	5.949E-03	2.395E-02	1.082E-02	3.538E-01	9.393E-01	5.814E+00
4 wt %	10	3.717E-04	6.235E-04	3.434E-03	6.708E-03	4.580E-03	1.204E-04	3.791E-02
	20	2.868E-03	2.147E-03	8.586E-03	9.721E-03	2.778E-02	4.563E-03	5.457E-01
	30	6.749E-03	3.673E-03	1.346E-02	1.103E-02	7.936E-02	3.300E-02	1.673E+00
	40	1.079E-02	4.816E-03	1.765E-02	1.151E-02	1.592E-01	1.212E-01	3.194E+00
	50	1.344E-02	5.626E-03	2.083E-02	1.159E-02	2.607E-01	3.108E-01	4.618E+00
	65	1.513E-02	6.102E-03	2.322E-02	1.151E-02	3.686E-01	6.301E-01	5.774E+00
5 wt %	10	2.902E-04	4.887E-04	2.802E-03	6.512E-03	4.010E-03	6.217E-05	2.509E-02
	20	2.477E-03	1.838E-03	7.301E-03	9.850E-03	2.410E-02	2.514E-03	3.951E-01
	30	6.348E-03	3.342E-03	1.185E-02	1.153E-02	6.989E-02	1.915E-02	1.308E+00
	40	1.094E-02	4.593E-03	1.604E-02	1.226E-02	1.444E-01	7.383E-02	2.679E+00
	50	1.452E-02	5.567E-03	1.949E-02	1.246E-02	2.459E-01	1.988E-01	4.150E+00
	65	1.710E-02	6.195E-03	2.227E-02	1.238E-02	3.633E-01	4.236E-01	5.522E+00

Table 3.7. Parameters and coefficients for calculating actinide decay heat for BWR fuel

Index n		1	2	3	4	5	6	7
Nuclide		^{241}Am	^{241}Pu	^{240}Pu	^{239}Pu	^{238}Pu	^{244}Cm	^{242}Cm
Decay constant λ_n (1/s)		5.078E-11	1.531E-09	3.347E-12	9.111E-13	2.504E-10	1.213E-09	4.923E-08
Enrichment	Burnup (MWd/kgU)	Actinide coefficients β_n (W/kgU)						
2 wt %	10	6.463E-04	1.029E-03	5.868E-03	6.474E-03	6.987E-03	7.945E-04	1.124E-01
	20	3.117E-03	2.451E-03	1.277E-02	7.861E-03	3.692E-02	1.999E-02	1.060E+00
	30	5.303E-03	3.451E-03	1.782E-02	8.126E-03	8.979E-02	1.167E-01	2.376E+00
	40	6.713E-03	3.993E-03	2.104E-02	8.162E-03	1.493E-01	3.539E-01	3.501E+00
	50	7.453E-03	4.292E-03	2.300E-02	8.197E-03	2.026E-01	7.566E-01	4.150E+00
	60	7.890E-03	4.463E-03	2.427E-02	8.249E-03	2.436E-01	1.305E+00	4.478E+00
3 wt %	10	4.587E-04	7.478E-04	4.230E-03	6.359E-03	5.531E-03	2.792E-04	5.971E-02
	20	2.793E-03	2.067E-03	1.021E-02	8.282E-03	3.010E-02	8.008E-03	6.925E-01
	30	5.524E-03	3.226E-03	1.547E-02	8.799E-03	7.984E-02	5.317E-02	1.850E+00
	40	7.606E-03	3.948E-03	1.948E-02	8.766E-03	1.479E-01	1.857E-01	3.167E+00
	50	8.340E-03	4.358E-03	2.211E-02	8.623E-03	2.208E-01	4.556E-01	4.101E+00
	60	8.789E-03	4.542E-03	2.383E-02	8.527E-03	2.824E-01	8.756E-01	4.649E+00
4 wt %	10	3.406E-04	5.629E-04	3.251E-03	6.168E-03	4.604E-03	1.217E-04	3.573E-02
	20	2.435E-03	1.755E-03	8.339E-03	8.548E-03	2.519E-02	3.854E-03	4.728E-01
	30	5.462E-03	2.969E-03	1.335E-02	9.435E-03	6.920E-02	2.719E-02	1.413E+00
	40	8.369E-03	3.845E-03	1.766E-02	9.545E-03	1.361E-01	1.017E-01	2.692E+00
	50	9.819E-03	4.417E-03	2.090E-02	9.337E-03	2.185E-01	2.696E-01	3.885E+00
	60	1.028E-02	4.686E-03	2.319E-02	9.065E-03	3.002E-01	5.656E-01	4.744E+00
5 wt %	10	2.625E-04	4.375E-04	2.611E-03	5.960E-03	3.971E-03	6.128E-05	2.319E-02
	20	2.108E-03	1.503E-03	6.984E-03	8.682E-03	2.174E-02	2.108E-03	3.381E-01
	30	5.190E-03	2.702E-03	1.158E-02	9.927E-03	6.030E-02	1.537E-02	1.088E+00
	40	8.702E-03	3.681E-03	1.587E-02	1.029E-02	1.223E-01	5.975E-02	2.227E+00
	50	1.109E-02	4.408E-03	1.944E-02	1.018E-02	2.053E-01	1.649E-01	3.481E+00
	60	1.230E-02	4.813E-03	2.224E-02	9.839E-03	2.980E-01	3.637E-01	4.609E+00

3.2.4 Structural Material Activation

Decay heat power is contributed by activation products from irradiated materials in fuel assembly structural components such as cladding, fuel rod spacers, water rods, tie plates, etc. Common assembly materials include Zircaloy-2, Zircaloy-4, Inconel, and stainless steel (typically used only in assembly end-region components). The decay heat contribution from activated assembly structural components is generally small relative to the fission products and actinides (see Figure 2.1) and may contribute up to several percent of the total decay heat.

The decay heat contribution is determined from the formula

$$P_S(t,T) = A(t)\, P_F(t,T)\,, \tag{14}$$

where the values for $A(t)$ are listed in Table 3.5 and $P_F(t,T)$ is the direct fission product decay heat evaluated in Sect. 3.2.1. The tabulated values of $A(t)$ yield conservative estimates of the decay heat power contributed by activated structural materials for typical fuel assembly designs, provided the burnup, in units of megawatt-days per kilogram of uranium, does not exceed 14 times the initial enrichment.

3.2.5 Safety Factor

An additional safety factor is applied to allow for uncertainties in the predicted values of the decay heat power obtained using the methods and data in this guide. The safety factor, $F_S(t)$, is tabulated in Table 3.5 as a function of decay time after discharge. Values applied for a cooling time, t, after discharge are obtained by linear interpolation of the tabulated data.

The uncertainty is determined primarily by comparison of predicted against calorimeter measurements of decay heat over the range of experimental data for the 132 assemblies described in Sect. 5. The uncertainty is found to be relatively small and largely independent of burnup over the range of the data. The methods are found to yield conservative estimates of decay heat on average. The safety factor includes additional statistical allowance to ensure that the values obtained using the guide are conservative with respect to 95% of the measurement data at a 95% confidence level. Potential nonconservatism due to the procedures of the guide and other approximations are also addressed by the safety factor.

Beyond the range of the calorimeter data, the safety factor is based on an extrapolation of the uncertainties and bias that is supported by isotopic analysis of spent fuel samples for the major actinides contributing to decay heat at the longer cooling times.

Development of the safety factor is described in Sect. 4.

3.2.6 Final Decay Heat Power

The total decay heat generation rate without the safety factor is calculated as

$$P_T(t,T) = P_F(t,T) + P_C(t,T) + P_E(t,T) + P_A(t,T) + P_S(t,T)\,, \tag{15}$$

where

$P_F(t,T)$ is the fission product decay heat power (excluding neutron capture) from Sect. 3.2.1,
$P_C(t,T)$ is the decay heat power from neutron capture to product ^{134}Cs from Sect. 3.2.2.1,
$P_E(t,T)$ is the decay heat power from neutron capture on other fission products from Sect. 3.2.2.2,
$P_A(t,T)$ is the decay heat power from actinides from Sect. 3.2.3,
$P_S(t,T)$ is the decay heat power from activated structural materials from Sect. 3.2.4.

The final decay heat generation rate with the safety factor $F_s(t)$ included is determined as

$$P'_T(t,T) = P_T(t,T)\, F_S(t)\,. \tag{16}$$

3.3 LIMITS AND RANGE OF APPLICABILITY

This proposed regulatory guide for decay heat has been developed to apply to the majority of commercial reactor spent fuel assemblies, to be easy to implement and use, and yield safe values that can be used for licensing evaluation but that are not excessively conservative. In the endeavor to increase the value of the proposed guide, the range of applicability has been extended to most of the existing spent fuel generated in the United States. The spent fuel characteristics for which the proposed guide is applicable are summarized in Table 3.8. The cooling time T_c is listed in units of years rather than units of seconds used in the actual implementation of this guide. The range of application of the guide is illustrated in

Figure 3.2 and compared with the existing and projected commercial spent fuel inventory in the United States published in U.S. Department of Energy (DOE) Form RW-859 (Ref. 23).

Table 3.8. Parameter range for applicability of the regulatory guide

Parameter	BWR	PWR
E_s (wt % ^{235}U)	2–5	2–5
B_{tot} (MWd/kgU)	10–55	10–65
T_c(year)	1–110	1–110
S_{avg} (kW/kgU)	12–50	12–50

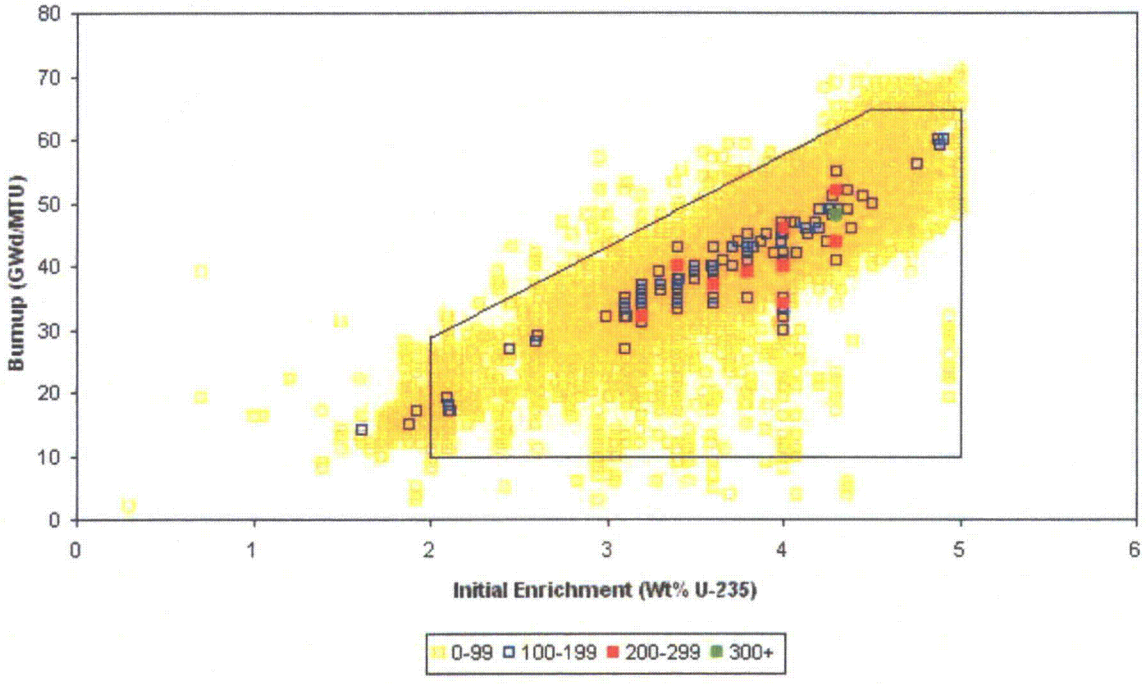

Figure 3.2. Plot showing the nominal range of application of the guide (limits shown for PWR fuel) compared with the current and projected inventory of commercial spent fuel assemblies in the United States, obtained from U.S. Department of Energy Form RW-859, displayed as a function of enrichment and final burnup. The number of assemblies is shown in the legend.

Additional limits are placed on the allowed combinations of enrichment and burnup. The restrictions are associated with the correction factors developed to address decay heat from components of neutron capture on fission products (excluding ^{134}Cs) and structural activation, which are based on a single conservative and generally bounding irradiation history. The data developed for these components of decay heat limit the burnup, in megawatt-days per kilogram of uranium, to a maximum of 14 times the initial fuel enrichment, in units of weight percent ^{235}U. The limit is sufficient to cover the vast majority of currently existing and future spent fuels. The maximum burnup should not exceed the limits of Table 3.8 regardless of the initial enrichment. The restrictions on applying the guide beyond the enrichment and burnup range of Table 3.8 are based for the most part on the limited availability of experimental data for methods validation.

The procedures in the revised guide are more flexible and thus enable accurate calculations for a wide range of assembly irradiation histories. The method implemented for analysis of fission products (excluding neutron capture effects) is not constrained by the operating history. However, the actinide coefficients were calculated using a constant specific operating power of 20 kW/kgU and average uptime of 80%. Both variables are conservative with respect to typical reactor operations. Correction factors are developed for the total actinide decay heat generation rate to account for variations in specific power over the range defined in Table 3.8. A detailed assessment found that beyond about 1 month cooling time actinide decay heat power decreases as the specific operating power increased.[‡] For conditions where the specific power is less than the allowable parameter range, the guide should not be used as this could result in actinide contributions that are underpredicted. At longer cooling times when the actinides become an increasingly important source of decay heat, such an underprediction could lead to nonconservative errors in the total decay heat power prediction.

An assembly parameter that may restrict application of the guide is the ^{59}Co content of the clad and structural materials. Cobalt-59 is partly transformed to ^{60}Co during irradiation and subsequently contributes to the decay heat rate. The ^{59}Co content used in deriving the activation tables here should be applied only to assemblies containing Zircaloy-clad fuel rods. The ^{59}Co content found in stainless-steel-clad fuel rods may result in a decay heat level that exceeds the tabulated values of the guide. Thus, application of the guide for stainless-steel-clad fuel should be limited to cooling times that exceed 20 years, after which the heat rate contribution from ^{60}Co has generally decayed to relatively insignificant levels. As modern assembly designs generally no longer use stainless-steel-clad fuel, this restriction is not expected to impact most fuel in storage.

In addition to the fuel parameters used to develop this guide, decay heat rates are a function of other variables to a lesser degree. Variations in moderator density (coolant pressure, temperature) can change decay heat rates, although calculations have shown that the expected differences (approximately 0.2% heat rate change per 1% change in water density, during the first 30 years of cooling) are not sufficient to require additional corrections. Other variations, such as the fuel assembly design, fuel diameter, pitch, number of guide tubes and water rods, and use of burnable poison rods (BPRs) and integral burnable poisons will also influence the decay heat of the assembly to a minor extent. The tables of fission fractions and actinide coefficients were calculated for fuel assemblies containing empty guide tubes. Computed decay heat rates for assemblies containing BPRs did not change significantly (<1% during the first 30 years of cooling) from fuel assemblies containing empty guide tubes or water rods.

Whenever the design or operating conditions for a spent fuel assembly exceed the parameter ranges developed in this guide, another well-qualified method of analysis should be used. A qualified method would be one that has been validated against measured decay data and demonstrated to provide accurate estimates of decay heat (i.e., with justified safety factors consistent with the measured data) for the design or operating conditions being evaluated.

[‡] At cooling times less than about 10 days, actinide decay heat increases with increasing specific power due to contributions from ^{238}Np, ^{239}Np, ^{236}U, and ^{239}U, which are not important or considered at the cooling times of the guide.

4 TECHNICAL BASIS

This section describes the rationale and technical basis for the procedures and data presented in Sect. 3. The accuracy of the procedures and data used in developing the procedures of the proposed guide are directly linked to experimental data.

4.1 CODE CALCULATIONS

Data developed for the proposed guide that are not adopted from decay heat standards are derived in large measure from calculations performed using the ORIGEN-S isotope generation and depletion code (Ref. 24). The ORIGEN-S code tracks the time-dependent concentrations of 129 actinides, 1,119 fission products, and 698 activation products produced during irradiation and decay. Calculation of integral quantities like decay heat is achieved by summing the contribution of each individual isotope to the total decay heat. This method (referred to as a summation method) is distinct from integral methods such as those widely used in standards that fit integral decay heat data rather than evaluate the decay heat contributions from individual nuclides. ORIGEN-S has been validated against integral measurements of decay heat for all assemblies listed in Table 2.1 (Refs. 14, 25) and against isotopic assay measurements for the key decay heat generating isotopes for the experiments summarized in Table 2.2 (Ref. 18–21).

The ORIGEN-S calculations used to support the development of this guide were performed using standard LWR assembly cross-section libraries released with the SCALE 5.1 code system (Refs. 26, 27). The libraries in SCALE were developed using two-dimensional neutron transport models of the fuel assembly to solve the neutron flux spectrum and effective reaction cross sections as a function of initial enrichment and burnup of the fuel. The commercial LWR fuel assembly libraries include selected Asea Brown Boveri (ABB), GE, Combustion Engineering (CE), Westinghouse Electric Company (W) and AREVA ATRIUM assembly designs, including GE 7 × 7, GE 8 × 8, ABB 8 × 8, GE 9 × 9, ATRIUM 9, GE 10 × 10, ATRIUM 10, SVEA 64, SVEA 100, CE 14 × 14, W 14 × 14, W 15 × 15, CE 16 × 16, and W 17 × 17 optimized fuel assembly designs. Cross sections in these libraries are tabulated as a function of initial fuel enrichment, burnup, and coolant density (for BWRs only), and are interpolated to the problem-dependent parameters by the ARP (automatic rapid processing) code in SCALE. The calculation sequence of cross-section generation, fuel depletion, and decay analysis is automated using the ORIGEN-ARP sequence, which has a Windows graphical user interface (Ref. 28).

4.2 FISSION PRODUCT METHODOLOGY

Fission product decay heat power is characterized as a function of time after fission for a single fission event by a series of exponential functions. The methodology and data, adopted from the ANS-5.1-2005 standard, enable the user to calculate the fission product component of decay heat with an accuracy comparable to that obtained from summation calculations for any operating history. Incorporation of this procedure to calculate fission product decay heat power represents a significant enhancement compared to the existing guide, which provides very limited flexibility to represent variations in operating power. However, the methods, as implemented in standards, generally require the user to select and justify a number of parameters and modeling approaches. To provide a fission product methodology that is complete and comprehensive, all data and modeling approaches are defined in the proposed guide. The technical bases for these data and approaches are described in this section.

4.2.1 Fission Product Decay Heat Functions

The function *f(t)* developed in the ANS-5.1-2005 standard represents the fission product decay heat power from 1 second to 10^{10} seconds (~300 years) after a fission pulse event. The coefficients are developed for thermal neutron induced fission of ^{235}U, ^{239}Pu, and ^{241}Pu and fast fission of ^{238}U. The decay heat power represented by *f(t)* includes the delayed energy released by fission product decay plus that released by decay progeny. The function is dependent only on the fission nuclide and the energy of induced fission. Nuclides generated by neutron capture on the fission products are represented separately as their effect is system dependent.

For an irradiation interval described by a constant fission rate with total irradiation time *T*, the decay heat power can be represented analytically from the *f(t)* function that describes the decay heat power (MeV/s) per fission as the convolution integral with the exact solution

$$F(t,T) = \int_{-T}^{0} f(t-t')\,dt' = \sum_{j=1}^{23} \frac{\alpha_j}{\lambda_j} e^{-\lambda_j t}\left[1 - e^{-\lambda_j T}\right], \qquad (17)$$

where the integral is performed over the irradiation time $-T$ to 0, and $t' = 0$ is the time at the end of irradiation, and t is the time after irradiation for which the total decay heat is calculated. The quantity $t - t'$ is the total time between a time point during irradiation t' and time t after irradiation, i.e., $t + (-t')$. If the operating history is represented as a series of irradiation intervals, with each interval having a constant fission rate as illustrated in Figure 3.1, the total decay heat power is evaluated by summing the contribution from each irradiation interval at the desired cooling time after irradiation according to Eq. (3).

The decay heat curve following a single fission of ^{235}U, calculated from the function *f(t)*, is illustrated in Figure 4.1. After 10 years cooling time, the decay heat power decreases with a characteristic half-life of about 29.5 years, consistent with the ^{137}Cs and ^{90}Sr half-lives of 30.0 years ± 0.7 % and 28.78 years ± 0.4%, respectively. These two fission products and their short-lived decay daughters ^{137m}Ba and ^{90}Y, with half-lives of 2.5 minutes and 2.67 days, respectively, collectively generate the majority of the fission product decay heat beyond about 5 years after discharge. These four isotopes contribute more than 80% of the total fission product decay heat after 10 years and more than 95% after 30 years. The dominance of ^{137}Cs and ^{90}Sr over the range of the decay heat calorimeter measurements, and moreover beyond the range of measurements up to the 110 year limit of the guide, is used as a basis for estimating the uncertainty in the methods and is discussed in the development of the safety factor in Sect. 4.5.

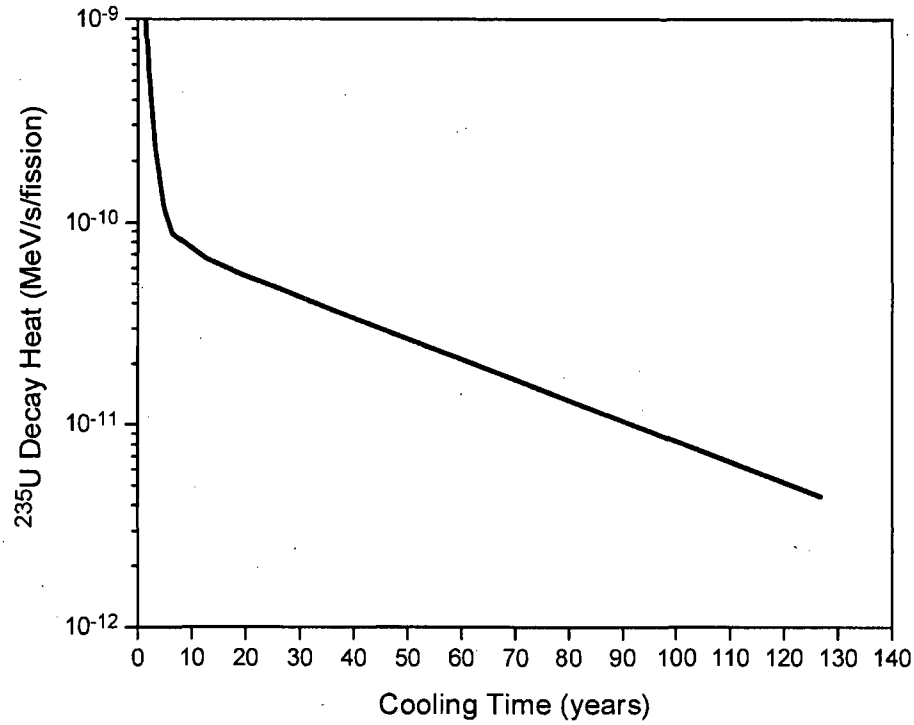

Figure 4.1. ANS-5.1-2005 fission product decay heat curve for a ^{235}U fission pulse. The decay heat beyond about 10 years decreases with a half-life of 29.5 years, characteristic of ^{137}Cs and ^{90}Sr decay.

4.2.2 Operating History Modeling

The operating power of the fuel is subject to change during the operating time. Such changes are taken into account when calculating the fission product decay heat power by subdividing the operating time into intervals of constant power. In principle, an arbitrary number of intervals, or histograms, may be used to approximate any operating history. An adequate number of intervals is required to minimize any systematic error due to the approximations, which decreases rapidly with increasing decay time. The error is also more sensitive to the power near the end of irradiation than at the beginning of irradiation.

Power changes in the current guide are accommodated using correction factors based on the ratio of the power during the last, and next-to-last, operating cycle before discharge and the average power during all irradiation cycles. This level of detail in representation of the operational power history has been demonstrated to ensure accurate results provided the cooling time is longer than one year (Ref. 3). Use of the existing guide for cooling times less than 1 year is not permitted because the methodology does not represent the operating history with a level of detail that is required for shorter times.

The proposed fission product methods, based directly on the ANS-5.1 standard, in principle enable the explicit representation of any irradiation history and thus the extension of the guide to much shorter cooling times than is currently allowed (i.e., 1 year). However, for spent fuel storage facility analysis, shorter cooling times are generally not required. The proposed guide therefore recommends representing the fission product power history using a representation that subdivides the operating time into intervals that correspond to the actual reactor operating cycles, an approach similar to the current guide. Unlike the

current guide, however, the operating power of the fuel during all cycles is represented explicitly, thereby eliminating the need for cycle power correction factors.

It is important to ensure that the fuel burnup obtained from the time-integrated power of the histogram equals the actual burnup of the fuel.

4.2.3 Energy per Fission

The total recoverable energy per fission, Q, is required to evaluate fission product decay heat power. This energy includes the recoverable prompt and delayed energy from fission (Q_f) and the gamma and beta energy released from neutron capture interactions in the fuel, structural materials, and coolant ($Q_{n,\gamma}$). The value of Q_f includes fission fragments, neutron kinetic energy, prompt gamma and beta radiation, and the total recoverable decay energy of the fission products (excluding neutrinos). Values of Q_f, and $Q_{n,\gamma}$ published in the ISO 10645 international standard are adopted for use in the guide. The capture energy is based on an average energy per capture of 6.1 MeV, which is characteristic of LWR systems. The average capture energy is dependent on the system characteristics and in particular on the neutron absorbing materials present. The estimated error of the Q values is 0.5% or less. A comparison of the ISO standard values with other evaluations is presented in Appendix I of ANS-5.1-2005 and is reproduced in Table 4.1, including the evaluations of Unik and Gindler (Ref. 29), DIN 25463 (Ref. 6), and values recommended by the Japan Atomic Energy Research Institute (JAERI) for decay heat calculations in Japan (Ref. 30). All evaluations agree to within 1%.

Table 4.1. Evaluated recoverable energy per fission pertaining to decay heat power calculations for LWR reactor systems

Actinide	ISO standard	Unik and Gindler	German standard	JAERI recommended
^{235}U	202.2 ± 0.5	201.7 ± 0.7	202.6 ± 0.9	202.2
^{238}U	205.5 ± 1.0	203.0 ± 1.1	205.9 ± 1.1	205.9
^{239}Pu	211.2 ± 0.7	210.6 ± 0.7	211.4 ± 1.1	210.9
^{241}Pu	213.7 ± 0.7	212.0 ± 0.8		213.2

Values obtained from Appendix I of ANS-5.1-2005.

While the total recoverable energy per fission values, Q_f, are constants of the fissionable nuclide (for a given incident neutron energy causing fission), the energy due to neutron capture is not constant but is both time dependent and fuel assembly material (system) dependent. The energy from neutron capture generally increases with irradiation time due to the buildup of fission products that contribute to increased absorption in the fuel. However, for LWR systems the materials and absorption in the fuel, coolant, and structural materials are relatively well defined and are observed to be consistent between the different evaluated sources.

4.2.4 Fission Power Fractions

Calculation of the fission product decay heat power requires the fraction of fission power contributed by each of the fissionable nuclides ^{235}U, ^{239}Pu, ^{238}U, and ^{241}Pu at each irradiation time step. The fractions are required because each fission nuclide has a unique decay heat power curve as represented by the coefficients in the expressions for $f(t)$. For uranium LWR, fuel the initial power comes mainly from ^{235}U and ^{238}U. During irradiation and the depletion of ^{235}U, fission power shifts from ^{235}U to the ^{239}Pu and

^{241}Pu isotopes. Neither the ANS-5.1-2005 standard nor the ISO 10645 standard provides guidance for determining the fission power fractions.

The proposed guide provides power fraction values that are tabulated as a function of burnup for initial enrichments between 2 and 5 wt %. The data were developed using ORIGEN-S burnup simulations for a Westinghouse 17 × 17 PWR fuel assembly. The data were derived from calculated fission fractions weighted by the energy per fission (Table 3.3) to obtain the relative power fractions listed in Table 3.4. These four fissionable nuclides represent more than 99% of the total fission power for LWR fuel over the range of the guide. Fission of other actinides is attributed to ^{235}U because the decay heat for ^{235}U is conservative for most cooling times after fission. The fission power curves provide slightly conservative results when applied to BWR fuels at most cooling times. The power fractions are determined from the value at the cycle midpoint and are assumed to be constant during each cycle. The effect of recalculating the fission power fractions on a more frequent time scale was found to be minor. The change in the calculated decay heat rate was less than ±0.2% on average and was less than 1% in all cases studies.

4.2.5 Corrections for Neutron Capture

The effect of neutron capture on fission products can be important at cooling times of interest to spent fuel storage, potentially contributing upwards of 20% of the total decay heat power. The peak effect occurs at 10^8 s (3.2 years) and is reduced significantly after about 10 years. Beyond 10^9 s (32 years) the neutron capture effect is not significant and no correction is required. Neutron capture increases the decay heat power mainly as a result of production of ^{134}Cs ($T_{1/2}$ = 2.06 years) from ^{133}Cs (stable). The cumulative fission yield of ^{134}Cs is very small ($<10^{-5}$) because the 134 mass chain ends with decay to ^{134}Xe, which is stable. Therefore, the only significant production route to ^{134}Cs is via ^{133}Cs capture. Other nuclides produced by capture that contribute to increasing decay heat power include ^{136}Cs, ^{148m}Pm, ^{148}Pm, and ^{154}Eu. Absorption by ^{135}Xe causes the opposite effect by producing stable ^{136}Xe instead of contributing to the decay heat by its own decay and that of its daughter, ^{135}Cs. The aggregate effect of neutron capture on fission products is to increase the decay heat power.

In the time range where the neutron capture effect is largest (~1–8 years), the contribution from ^{134}Cs typically accounts for more than 90% of the total neutron capture effect. An accurate representation of the decay heat contribution from ^{134}Cs is therefore required for analyses in the time frames of application for the proposed guide. The method used to account for neutron capture in the ANS-5.1-2005 standard uses a single set of correction factors (as a function of cooling time) developed using a bounding irradiation condition. The factors are independent of the actual enrichment or burnup of the fuel and can lead to decay heat values due to neutron capture that are excessively conservative for typical commercial fuel. The buildup and decay of ^{134}Cs can be accurately represented using a simple two-chain analytical model in Eq. (6) developed in the ISO 10645 decay heat standard. The neutron capture contribution from other fission products (all except ^{134}Cs) are treated as an aggregate using summation code methods to calculate the net effect of these nuclides using conservative irradiation assumptions.

The ^{133}Cs production rate is the cumulative fission yield, weighted for the different fissionable actinides, multiplied by the fission rate. The activity of ^{134}Cs, and its decay heat generation rate, is thereafter governed by the ^{133}Cs (n,γ) cross section and removal of ^{134}Cs by neutron absorption and decay.

The fuel flux is conservatively estimated according to Eq. (9), developed in ISO 10645. For an interval of constant power the initial flux level varies inversely with the enrichment level. For example, the flux required to achieve the same fission rate (power) for 2 wt % ^{235}U fuel is approximately twice that needed for 4 wt % enriched fuel. As fuel is irradiated, the net concentration of the fissile species decreases, requiring an increase in flux to maintain the same power level. The equation for the flux as a function of

initial enrichment and specific power yields flux values that are typical of LWR fuel. The resulting flux values are developed for PWR fuel and are conservative when applied to BWR fuel.

To provide independent confirmation of the procedures, developed directly from the ISO 10645 standard, validation of the equations and recommended nuclear data was performed using measured ^{133}Cs and ^{134}Cs nuclide concentrations obtained by experimental radiochemical isotopic analysis of spent fuel samples. The mass concentrations of ^{133}Cs and ^{134}Cs were calculated according to Eq. (6), adjusted to yield the content in grams instead of watts, for 27 spent fuel samples. The selected samples that included cesium isotopic measurements were

- three ATM-104 samples from the MKP109 rod measured by Pacific Northwest National Laboratory;
- nine Takahama fuel samples from rods SF95 and SF97 measured by JAERI;
- eight samples from Three Mile Island fuel rods O1, O12, and O13 measured by the GE-Vallecitos Nuclear Center (GE-VNC);
- samples GU1, GU3 and GU4 from the Belgonucleaire ARIANE program (Actinides Research in a Nuclear Element) measured by Studiecentrum voor Kernenergie-Centre d'étude de l'Energie Nucléaire (SCK-CEN), the Belgian Nuclear Research Center) the Institute for Transuranium Elements (ITU);
- samples GGU1 and GGU2 (1/2) from the Belgonucleaire MALIBU program (MOX and UOX LWR Fuels Irradiated to High Burnup) measured by SCK-CEN, the Paul Scherrer Institute (PSI), and the Commissariate à l'Énergie Atomic (CEA), the French Atomic Energy Commission, and
- one sample from rod M11 from the Belgonucleaire REBUS program (Reactivity Tests for a Direct Evaluation of the Burnup Credit on Selected Irradiated LWR fuel bundles) measured by SCK-CEN.

The concentration of ^{133}Cs, the capture precursor to ^{134}Cs, was measured in nine of the fuel samples. These measurements were used to verify the analytical expressions for the production of ^{134}Cs using the nuclear data defined in Sect. 3.2.2.1. Cesium concentrations calculated using these data and the analytical equations implemented in the proposed guide were compared directly to the experimental data.

The average cross-section values for ^{133}Cs and ^{134}Cs capture listed in Sect. 3.2.2.1 are derived in this study from ENDF/B-VII nuclear data evaluations. Spent fuel analyses performed using earlier versions of ENDF/B have exhibited a systematic underprediction of the ^{134}Cs concentrations of about 10–20% compared to measurements (Refs. 19–22). Average cross-section values recommended in the ISO 10645 standard were found to yield a similar bias in the predicted concentration of ^{134}Cs when applied to the analytical expression in Eq. (6). Re-evaluation of the average cross sections performed in this study using ENDF/B-VII-based data yielded a ^{133}Cs capture cross section that was about 6% larger than the value recommended in ISO 10645 and a ^{134}Cs capture cross section about 30% smaller than the ISO 10645 value. The combined effect of the cross section changes resulted in predicted ^{134}Cs concentrations that are significantly improved compared to values obtain using previous cross-section evaluations. The differences between calculation and experiment based on samples from the selected experiments are illustrated in Figure 4.2.

The concentrations of stable ^{133}Cs are predicted within 10% of measurement for all samples. The calculated concentrations of ^{134}Cs are similarly observed to be within 10% of experiment for many samples. Based on these comparisons with experimental data and the assembly decay heat validation results obtained using the guide (discussed in Sect. 5), the nuclear data and procedures used in the proposed guide to predict decay heat from ^{134}Cs are considered to be acceptable for the purposes of calculating decay heat.

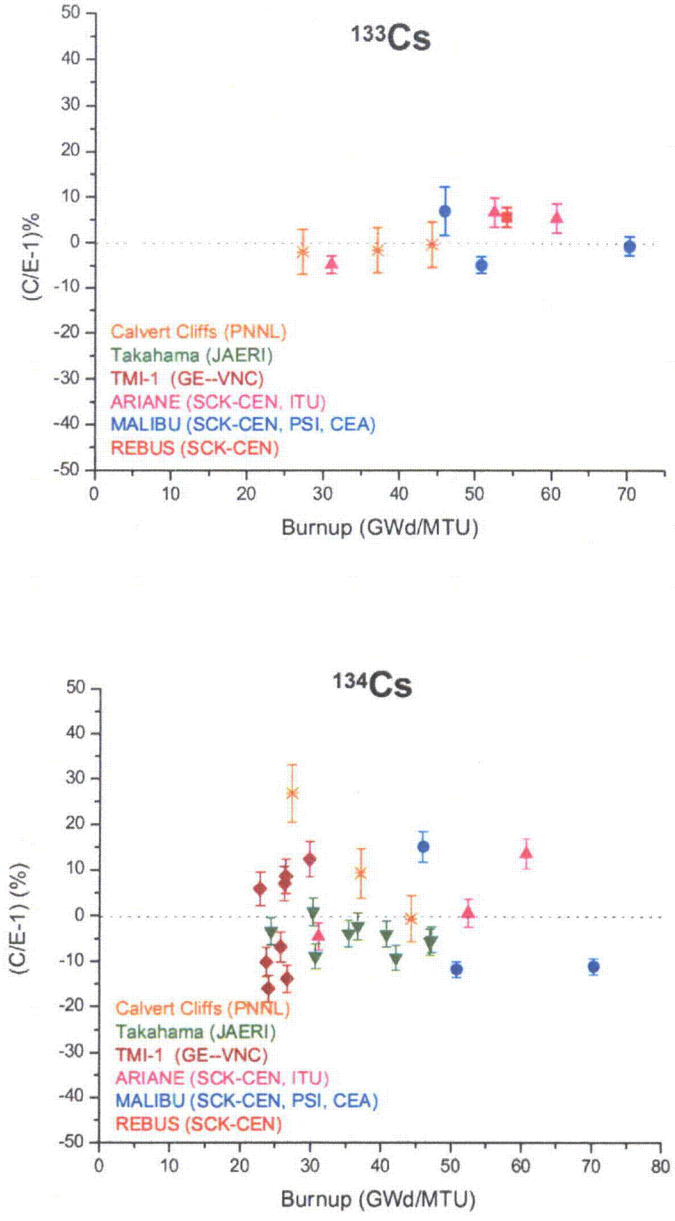

Figure 4.2. Comparison of ^{133}Cs and ^{134}Cs concentrations predicted using the methods of the proposed guide with experimental measurements – (C/E-1)% = (calculated/experiment – 1) × 100%.

4.3 ACTINIDE METHODOLOGY

Existing standards for decay heat do not include accurate methods to evaluate the actinides other than short-lived isotopes ^{239}Np and ^{239}U. The ANS-5.1 standard does not provide methods for any long-lived actinides. The ISO 10645 standard provides a bounding approach that would be excessively conservative for applications to cooling times of interest to spent fuel storage. The actinide methodology developed for the proposed guide addresses the contributions from ^{241}Am, ^{242}Cm, ^{244}Cm, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu. These nuclides contribute more than 99.5% of the total actinide decay heat over the time range of the proposed guide.

4.3.1 Numerical Framework

Developing an accurate methodology for actinides within the framework of an easy-to-use guide presents significant challenges. Production of the major decay heat generating actinides involves large transmutation chains, from ^{238}U to the isotopes of Np, Pu, Am, and Cm, involving multiple neutron capture and fission processes, and coupled decay chains that preclude solution with simple analytical equations. The reaction cross sections also change with burnup and operating conditions and depend on the assembly design and reactor type.

Early studies to develop an actinide methodology suitable for use within the framework of a standard were performed by Wilson et al. (Ref. 31), but the methods were not adopted. The approach effectively involved using correction factors applied to the calculated fission product decay heat component to obtain the total from actinides and fission products. The correction factors are dependent on initial enrichment of the fuel, burnup, specific power and operational history, reactor type, assembly design, and cooling time. Therefore, a large set of actinide correction factors is needed to cover the full range of commercial fuel. A similar actinide methodology based on correction factors is implemented in the ISO 10645 decay heat standard. However, the ISO standard uses only a single set of time-dependent factors that does not account for the enrichment or burnup of the fuel and is designed to yield very conservative estimates of the actinide decay heat power. Such an approach will yield acceptable results only for short cooling times where the actinide contribution is relatively small and a highly conservative methodology is therefore not overly restrictive. However, for cooling times beyond about 20 years actinides contribute more than 30% of the total decay heat. Therefore, in developing a guide that can be applied to interim storage and cooling times exceeding 100 years, a more rigorous and accurate methodology is highly desirable.

The method used in the existing regulatory guide basically involves a tabulated set of decay heat values tabulated over enrichment and burnup, specific power, and cooling time. However, actinides are not represented separately in the current guide, although such an approach is feasible and could be developed.

The methodology proposed for the revised guide represents the variation of actinide decay heat power with cooling time using the physical decay constants of the dominant decay heat generating actinides. The variation in activity, and therefore decay heat power, for the main actinides is observed to be exponentially decreasing according to the nuclide half-life in the time range of the guide where the respective actinides are important components of the total decay heat power, as shown in Figure 4.3. Several actinides exhibit a buildup shortly after discharge due to production from decay precursors. Specifically,

1. ^{239}Pu increases due to beta decay of ^{239}Np ($T_{1/2}$ = 2.36 days),
2. ^{238}Pu increases due to alpha decay of ^{242}Cm ($T_{1/2}$ = 162.8 days),
3. ^{240}Pu increases continuously over the range of the guide due to production from alpha decay of ^{244}Cm ($T_{1/2}$ = 18.1 years), and

4. ^{241}Am increases continuously over much of the range of the guide due to alpha decay of ^{241}Pu ($T_{1/2}$ = 14.4 years) and decreases at longer cooling times due to decay of ^{241}Am ($T_{1/2}$ = *432.7 years*).

Decay heat values (coefficients) for each actinide have been generated using depletion calculations for a wide range of conditions representative of typical fuel, taking account of the temporal variations noted above. Because of large neutron spectral differences and neutron cross sections between PWR and BWR systems, actinide coefficients are developed separately for these two reactor types. Within these reactor classes, a wide range of fuel assembly designs were evaluated to determine the design that yielded the most conservative estimates of actinide decay heat power for use in the guide.

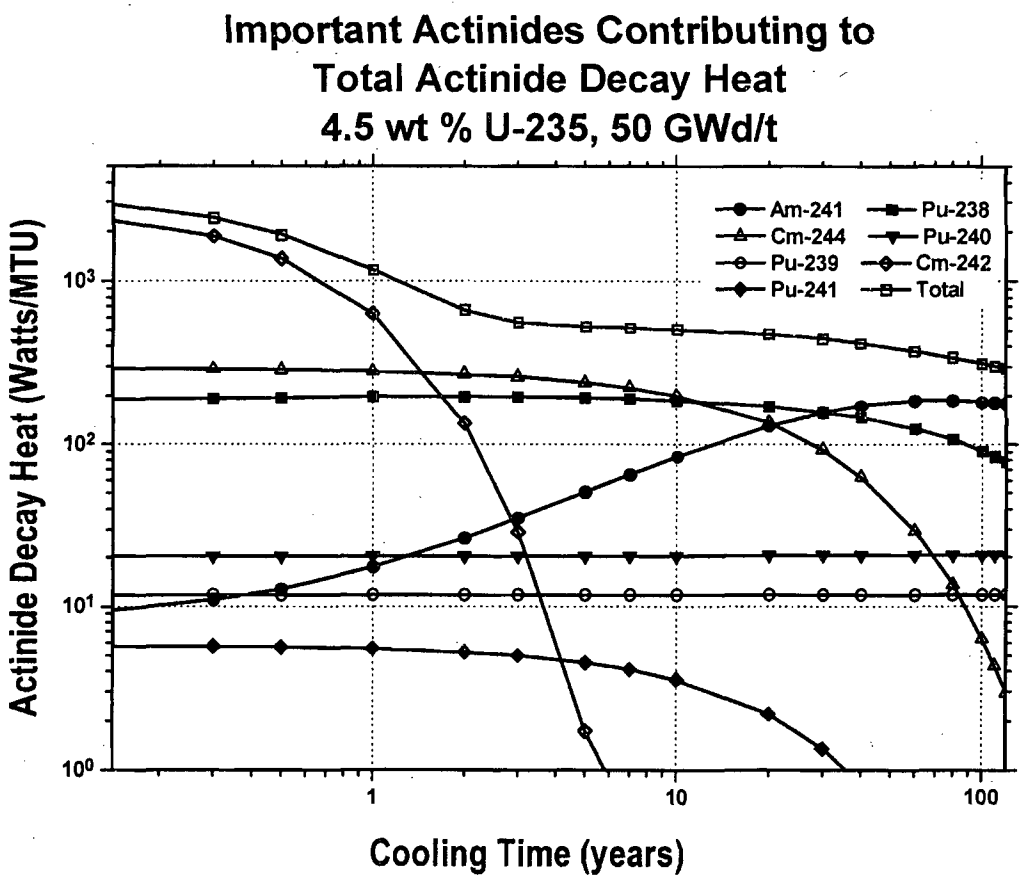

Figure 4.3. Decay heat per metric ton of uranium from dominant actinides (except for ^{239}Np and ^{239}U) in typical LWR spent fuel as a function of cooling time.

4.3.2 Development of Actinide Data

The actinide coefficients (β) tabulated in Tables 3.6 and 3.7 represent the specific decay heat generation rates, in units of watts per kilogram of uranium, derived by extrapolating the decay heat from the region of importance to the effective value at the time of discharge (t = *0*). The method is illustrated in Figure 4.4 for the example of ^{238}Pu. The concentration of ^{238}Pu increases initially after discharge due to the decay of ^{238}Np with a half-life of 2.12 days and the decay of ^{242}Cm with a half-life of 162.8 days and then decreases at a rate characteristic of the ^{238}Pu half-life of 87.7 years. The extrapolation procedure effectively includes contributions from both the actinide itself and its decay precursors. The method

ensures accurate predictions of each actinide in the region where it make a significant contribution to the total decay heat. In the example shown, the method overestimates decay heat at short cooling times. However, the absolute error in the total decay heat generation rates is insignificant because actinides as a group are relatively unimportant at short times compared to fission products.

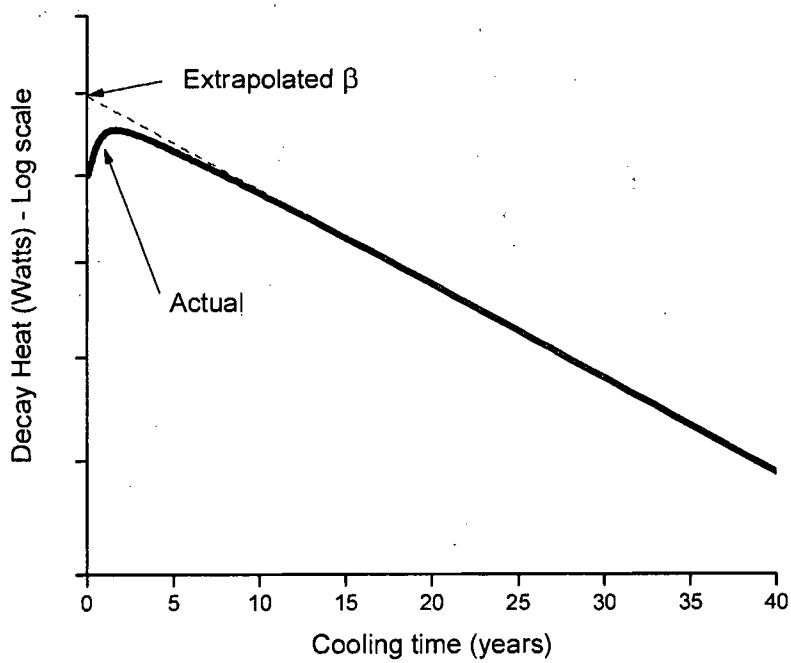

Figure 4.4. Extrapolation used to obtain the actinide coefficients (β), illustrated for the example of ^{238}Pu. The decay heat from ^{238}Pu initially increases after irradiation due to the buildup from ^{242}Cm decay ($T_{1/2}$ = 162.8 days) and subsequently decreases according to the half-life of ^{238}Pu ($T_{1/2}$ = 87.7 years).

Another example is ^{242}Cm, which decreases according to its half-life of 162.8 days for the initial 5 years following discharge. After this time the rate of decrease is much slower due to production by the decay of ^{242m}Am, with a half-life of 141 years. However, after 5 years cooling the contribution of ^{242}Cm to decay heat is insignificant and any production from ^{242m}Am can be neglected. Extrapolation of the heat generation rate from the decay of ^{242}Cm is performed in the region of importance in the guide, between 1 and 3 years.

The small increase in the decay heat generation rate of ^{240}Pu with time due to alpha decay of ^{244}Cm is conservatively addressed by assuming all ^{244}Cm decays instantaneously to ^{240}Pu (for the purposes of calculating ^{240}Pu only).

As discussed previously, the exception to this time-dependent behavior is ^{241}Am, which is produced from the decay of ^{241}Pu. The buildup of ^{241}Am activity with cooling time is represented analytically. The decay heat is related to activity by the recoverable energy per decay: ^{241}Am = 5.629 MeV and ^{241}Pu = 5.361 keV. Because the energy released by ^{241}Am decay is three orders of magnitude larger than ^{241}Pu, the decay heat generated by ^{241}Am rapidly becomes a dominant actinide source with increasing cooling time and a major component of the total decay heat. The time-dependent increase in the decay heat generated by the buildup of ^{241}Am following discharge is represented by the analytical expression

$$W_1 = W_2^0 \frac{E_1}{E_2} \frac{\lambda_1}{(\lambda_1 - \lambda_2)} (e^{-\lambda_2 t} - e^{-\lambda_1 t}) , \quad (18)$$

where subscript 1 refers to daughter ^{241}Am and 2 refers to the parent ^{241}Pu, W is the decay heat generation rate (W_2^0 is the decay heat from ^{241}Pu at $t = 0$), and E is the recoverable energy per decay. The buildup and decay are illustrated in Figure 4.5. The analytical expression is used to develop modified actinide coefficients $\hat{\beta}_n$ for ^{241}Am and ^{241}Pu in Eq. (12) that account for this buildup and decay.

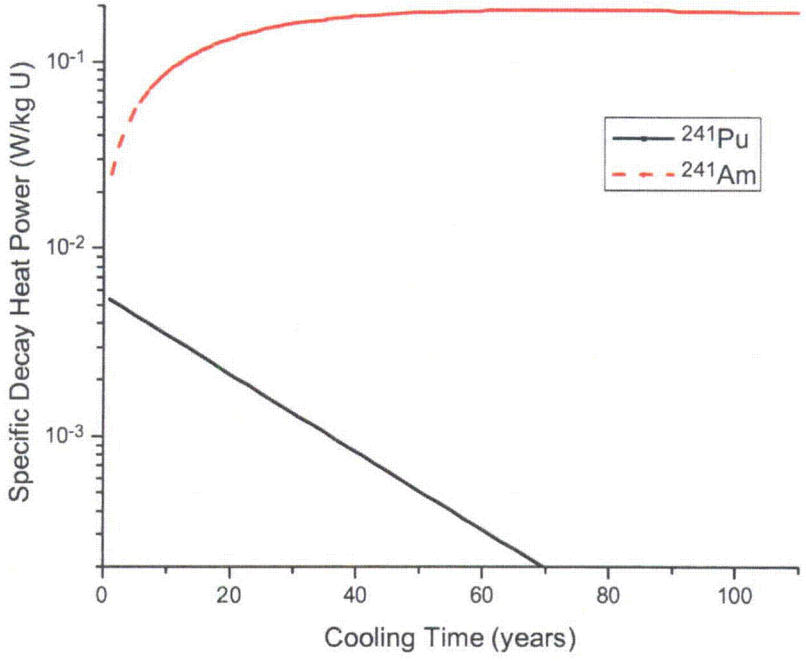

Figure 4.5. Decay heat contributions from ^{241}Pu and ^{241}Am predicted using the guide for PWR fuel with an enrichment of 4 wt % ^{235}U and discharge burnup of 50 MWd/kgU. The figure illustrates the buildup of ^{241}Am caused by decay of its precursor ^{241}Pu after discharge.

The PWR fuel assembly model used to calculate the actinide data coefficients β_n in Table 3.6 was a Westinghouse 17 × 17 assembly. This model produced the maximum actinide activities compared to other assembly designs evaluated. GE 7 × 7 and ATRIUM 10 assemblies, which produced the maximum activities (for different nuclides), were used to calculate the BWR actinide data coefficients in Table 3.7. The depletion calculations were performed using ORIGEN-S and a relatively low specific operating power of 20 kW/kgU that required longer irradiation times to achieve the discharge burnup than higher power and resulted in conservative values of actinide decay heat power, attributed mostly to increased production of ^{238}Pu. The operating history assumed a downtime fraction (for refueling between cycles) of 20%, again, a conservative value that increases actinide decay heat generation. A correction factor in Eq. (13) is applied to the calculated actinide decay heat for operating powers different from those used to generate the data.

Finally, the calculated actinide coefficients were adjusted to account for computational bias in the ORIGEN-S calculations, as determined from comparisons of calculated and experimental actinide concentrations in spent fuel (Ref. 18–21). The mean isotopic biases applied to correct the calculated

coefficients are listed in Table 4.2. The validation results from benchmarking the code against experiments are illustrated for the plutonium isotopes in Figure 4.6, showing the percent difference between calculated (C) and experimental (E) values. The results for curium isotopes are shown in Figure 4.7. The figures include results for the fuel samples listed in Table 2.2 that include burnup values covering the range of application for the guide. The coefficients were adjusted (conservatively) for the average bias determined for each actinide. Calculated values were only adjusted in cases where the calculated concentrations were less than measurements. Nuclides that were overpredicted (positive bias) were conservatively not adjusted. Because the activity of ^{241}Am is determined almost entirely by decay of its parent ^{241}Pu, the average bias in ^{241}Pu was assigned to ^{241}Am. Additional uncertainty associated with the evaluated half-life of ^{241}Pu is included in the development of the safety factor, discussed in Sect. 4.5. The maximum error due to the linear interpolation of the actinide coefficients, β_n, is determined to be less than 1%.

Table 4.2. Results of spent fuel isotopic validation studies

Nuclide	No. of samples evaluated	C/E-1 (%)[a]	$\sigma_{C/E-1}$ (%)
^{238}Pu	27	-8.7	7.0
^{239}Pu	27	3.7[b]	3.7
^{240}Pu	27	2.2[b]	3.7
^{241}Pu	27	-3.6	3.7
^{242}Cm	22	-17.4	18.9
^{244}Cm	24	-3.9	11.1

[a] Average bias in the code calculations, calculated as (Calculated/Experiment – 1) × 100%.
[b] Actinide coefficients not adjusted for conservative positive bias.

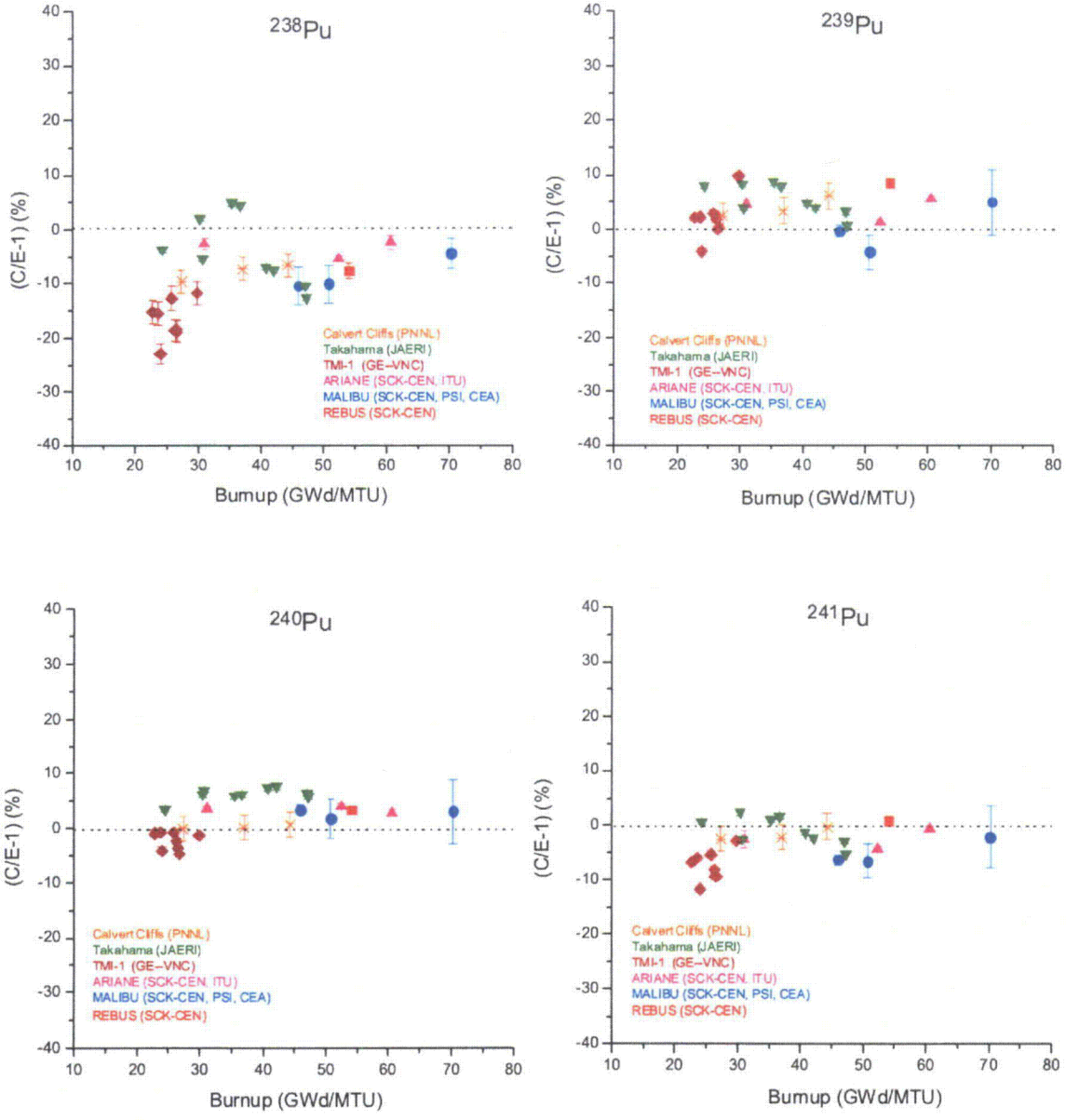

Figure 4.6. Comparisons of calculated and experimental concentrations of major plutonium isotopes in spent nuclear fuel samples – (C/E-1)(%) = (calculated/experiment – 1) × 100%.

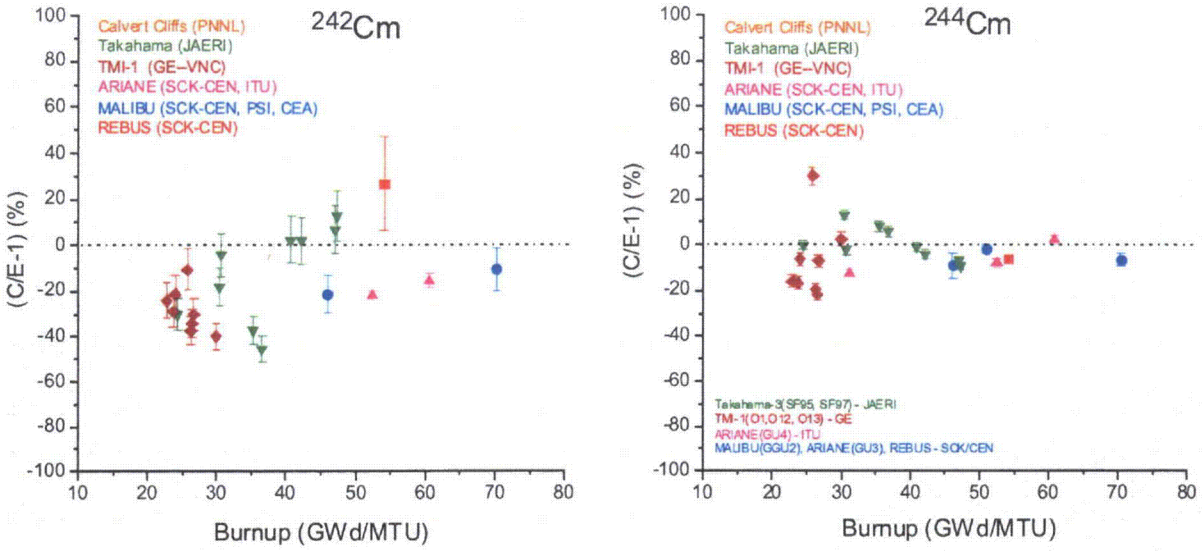

Figure 4.7. Comparisons of calculated and experimental concentrations of dominant curium isotopes in spent nuclear fuel samples – (C/E-1)(%) = (calculated/experiment – 1) × 100%.

4.4 STRUCTURAL ACTIVATION PRODUCTS

Decay heat from activated structure components can contribute up to several percent of the total decay heat power from an assembly. This component of decay heat is dominated at cooling times less than 50 years by ^{60}Co generated from the activated natural cobalt (^{59}Co) impurities in the structures. Activation products ^{63}Ni and ^{94}Nb are dominant at longer periods of time but contribute less than 0.1% of the total decay heat in the timeframe of the guide and are therefore not of practical importance. The decay heat from activation products is therefore very dependent on the amount of initial trace cobalt present in assembly components.

The initial quantities of structural and trace constituents used in the calculations to determine activation product decay heat are listed in Table 4.3. These values were used in developing RG 3.54 and are derived from material compositions specified in the DOE report *Characteristics of Potential Repository Wastes*, Volume 1 (Ref. 32). The component mass per assembly used in the activation calculations is weighted to reflect the difference in the neutron flux level and spectrum between the fuel and the activated components (Ref. 33). Hardware components include Zircaloy cladding, spacers, and assembly end-fitting components. The cobalt content based on upper limit specifications is 800 ppm for Type 304 stainless steel and 10 ppm for Zircaloy-4. Measured concentrations of cobalt (Refs. 34, 35) suggest significantly lower cobalt levels than used in the calculations for the guide, as fuel manufacturers currently use reduced-cobalt-impurity materials in assembly components with less than 100 ppm specifications to reduce radiation fields and disposal costs. The values in Table 4.3 therefore represent significant conservatism in the contribution of activated components that is dominated by cobalt. The overestimation will be most significant between about 2 and 10 years cooling, with the peak effect near 6 years cooling.

Table 4.3. Elemental contents of assembly hardware used in activation calculations

Element	Symbol	BWR g/kgU	PWR g/kgU
Oxygen	O	134.5	134.5
Chromium	Cr	2.4	5.9
Manganese	Mn	0.15	0.33
Iron	Fe	6.6	12.9
Cobalt	Co	0.024	0.075
Nickel	Ni	2.4	9.9
Zirconium	Zr	516	221
Niobium	Ni	0	0.71
Tin	Sn	8.7	3.6

The fuel assembly compositions also assume Zircaloy-clad fuel. Older assemblies have used stainless-steel-clad fuel rods that will have significantly larger amounts of activated ^{60}Co in the cladding than assumed in this work. Therefore, use of the guide for stainless-steel-clad fuel assemblies should be limited only to cooling times greater than 20 years. After 20 years, the activity of ^{60}Co will have decayed to less than 10% of the initial activity and its contribution will be minor. In most cases, stainless-steel clad fuels will not present a practical limitation for use of the guide for commercial fuel as steel cladding has not been used for many years.

The decay heat power generated by activation products is highly dependent on the irradiation fluence and cooling time. The values of $A(t)$ listed in Table 3.5 are developed for high fluence conditions and tabulated as a function of cooling time. The values in the guide are designed to be conservative for assemblies with an average burnup, in units megawatt-days per kilogram of uranium, which does not exceed 14 times the initial enrichment, in weight percent ^{235}U. These parameters are representative of the vast majority of commercial spent fuel assemblies.

4.5 DEVELOPMENT OF THE SAFETY FACTOR

The safety factor applied in the final decay heat generation rate is developed to provide additional margins for bias and uncertainty in the procedures and data used in the guide. The types of errors and uncertainties can be classified into the following groups.

- Errors in the nuclear data used to develop the data in the guide (including the standard data for fission products, effective fission yields, decay constants, energy per fission, and decay energy).
- Errors in the computational models used to develop data in the guide (e.g., assembly cross sections, use of a point model to represent the assembly).

- Associated procedural inaccuracies such as interpolation of data in the guide and representation of the irradiation history.

- Contingency for uncertainties not explicitly addressed.

Each component of decay heat evaluated in the guide has its own associated margin of conservatism and level of uncertainty. The actinide and structural material activation decay heat components, for example, are developed using a conservative irradiation history that will yield conservative results for the majority of commercial spent fuels. The fission products, on the other hand, are calculated using methods directly from the standard without added margins for conservatism. The combination of methods is expected to result in conservative estimates of decay heat for most fuel assemblies, without applying any additional safety factor. Validation of the methods against assembly calorimeter measurements, presented in Sect. 5, confirms that the vast majority of the predicted values indeed exceed measurement. Based on analysis of all measured assemblies, the guide overestimates the total decay heat power (without any safety factor included) with a relative mean bias of about 6%.

An appropriate safety factor that includes an additional margin for decay heat uncertainty is derived from direct comparisons of the decay heat rates predicted using the guide with experimental measurements of assembly decay heat for cooling times between 2 and 27 years. These comparisons are presented and discussed in Sect. 5 of this report. A summary of the results is discussed in this section as they pertain to the development of the safety factor. Beyond the range of experimental data, safety factors are developed using uncertainties in calculated isotopic concentrations of the major decay heat nuclides obtained from validation studies involving isotopic measurements of spent fuel samples.

The safety factor is derived from a statistical analysis of the validation data to determine the one-sided tolerance interval. A tolerance interval is the interval within which there is likelihood, with a defined level of confidence, that a specified fraction of the data values lie. The tolerance interval includes the influence of sample size. The ratio of the calculated-to-experimental (C/E) decay heat values for the 64 measured assemblies with a cooling time between 11 and 28 years (see Sect. 5.1) is 1.044 ± 0.024. That is, the calculations on average are about 4% greater than measurement. There is no discernable difference between the PWR and BWR results in this time range and the results are combined. The lower one-sided tolerance limit, above which 95% of the data reside with a confidence of 95% (95%/95% limit), is 0.996. Because the lower limit is slightly less than unity, an additional factor of 1.004, applied to the calculated decay heat values, will ensure that the results predicted using the guide will be conservative for 95% of the measured assemblies with a confidence of 95%. For comparison, the one-side tolerance limit for which 67% of the data exceed the limit with 95% confidence is 1.028, and no additional margin would be warranted for this statistical criteria. For the purposes of the guide, the safety factor is based on the one-sided 95%/95% lower tolerance limit.

Validation data for assemblies with less than 11 years cooling time is based entirely on the calorimeter measurements made at GE-Morris and Hanford. The calculated decay heat for the 10 PWR fuel assemblies, without any safety factor, is conservative with a mean C/E of 1.043 ± 0.030. This result is similar to the results for assemblies measured at CLAB that involve longer cooling times. The BWR data for cooling times less than 10 years are from measurements of 58 Cooper and Monticello reactor assemblies made at GE-Morris. These results show a larger bias and yield a C/E of 1.099 ± 0.059. The larger variance is consistent with the larger measurement uncertainty for the Cooper and Monticello measurements (5–10%) as compared to the CLAB measurements (2%). In particular, the Monticello measurements were problematic and many measurements showed poor reproducibility, leading, in part, to the larger variance. The larger positive bias observed for the analysis of BWR assemblies using the guide is attributed, in part, to a conservative estimate of the contribution from neutron capture to produce ^{134}Cs, which can be a significant decay heat component for cooling times less than about 8 years. The method

in the guide applies cross sections and a neutron flux level based on PWR fuel characteristics that are conservative when applied to BWR fuel analysis. Some small additional conservatism is attributed to the operating history for many of the Cooper assemblies that experienced removal from the reactor and storage for two cycles before being reintroduced into the core. Because the method used in the guide to evaluate neutron capture effects does not include provision to represent extended decay times between irradiation cycles (the methods do not include the decay of activated ^{134}Cs during this time), the results are expected to be conservative.

Statistical analysis of the 58 Cooper and Monticello BWR assemblies yields a lower one-sided tolerance interval of 0.980. A factor of 1.020, applied to the calculated BWR results, is therefore required to ensure a conservative estimate of decay heat with a 95%/95% confidence level. The larger safety factor for the BWR assemblies is a direct consequence of the larger variance between predictions and measurements caused in part by the larger experimental uncertainty of the GE-Morris measurements, and the trend of the guide to overpredict decay heat rates for some assemblies with extended out-of-core storage times at shorter cooling times. Based on the experimental data and the trend of the guide to overpredict the BWR results due to application of conservative methods and data, a safety factor of 1.02 is recommended for both PWR and BWR assembly types. This value, applied to the results for all assemblies yielded decay heat values with the guide that exceeded the measured values for all assemblies.

Beyond 28 years there are no available decay heat measurements for spent fuel assemblies. In this cooling time range, the fission product contribution decreases due primarily to the decay of the nuclides ^{137}Cs and ^{90}Sr (and their short-lived daughters ^{137m}Ba and ^{90}Y, respectively). Therefore, the relative contribution of the fission product error to the total decay heat error will decrease as fission products decrease in relative importance and the actinide importance increases. The relative increase in the actinide contribution is attributed in large measure to longer lived ^{238}Pu and ^{241}Am isotopes that make up most of the actinide decay heat beyond about 30 years. Any increase in the relative uncertainty in total decay heat power calculations beyond the range of calorimeter data is therefore attributed primarily to errors in the contributions for the main actinides as determined using the guide.

The accuracy of the actinide coefficients (and decay energy release rates), calculated by the ORIGEN-S code and adjusted for code bias, can be estimated from the analysis of destructive radiochemical isotopic assay measurements for spent nuclear fuel, discussed in Sect. 4.2. As the relative importance of the actinides increases with decay time, so will the margin of uncertainty. The actinide uncertainties in Table 4.2 are listed as the relative standard error expressed as the percent difference between calculation and experiment for each actinide. At cooling times beyond 5 years ^{241}Am is produced almost entirely from the decay of ^{241}Pu, and therefore any bias in the calculated ^{241}Am concentration will the same as the bias observed for ^{241}Pu. The effect of uncertainty in the ^{241}Pu half-life (14.35 ± 0.10 years) on the calculated ^{241}Am contribution increases the relative uncertainty for ^{241}Am by less than 1%.

Based on conservative predictions of actinide content for high burnup PWR fuel, the safety factors beyond 28 years were developed by weighting the uncertainties for the calculated actinide concentration by their relative importance to estimate the total uncertainty between 28 and 110 years. Computational analysis of the measured spent fuel assemblies with cooling times between 11 and 28 years using ORIGEN-S indicates that the actinides contributed an average of 27% (±4% standard deviation) of the total decay heat in this range. The maximum actinide contribution was calculated for high-burnup Ringhals 2 assembly F32, which had an actinide component that was about 39% of the total decay heat power. As cooling time increases, actinides represent a larger fraction of the total decay heat; as much as 75% after 110 years of cooling. The error analysis assumes that the uncertainty is dominated by the predicted compositions of the actinides and not the nuclear decay data (decay constants and energy per decay). A review of the data uncertainties in evaluated nuclear data files confirms that the decay data are well known relative to the actinide composition uncertainties listed in Table 4.2.

The relative uncertainty in the total decay heat contributed by the uncertainty in the actinides was estimated from the actinide isotopic validation data as

$$\sigma = \frac{1}{W_T}\sqrt{\sum_{n=1}^{7}(\omega_n \sigma_n)^2}\ , \tag{19}$$

where ω_n is the absolute time-dependent decay heat power contributed by actinide n, conservatively calculated for high burnup fuel and cooling times up to 110 years; W_T is the total decay heat power; and σ_n is the relative standard error associated with the contribution from each actinide listed in Table 4.2 plus ^{241}Am. This procedure results in a predicted uncertainty in the total decay, due to the actinides only, that varies between 1 and 2% over the range of the measurement data (less than 28 years cooling). The relative uncertainty (one standard deviation) increases nearly linearly with increasing cooling time to about 2.7% after 110 years. The increased uncertainty is caused by the relative increase in the actinide contribution which is dominated by the two isotopes ^{238}Pu and ^{241}Am. The safety factor developed for the proposed guide beyond 28 years is based on two standard deviations of the actinide uncertainty, increasing from a value of 1.020 at 25 years (range of the calorimeter data) to about 1.053 at 110 years.

Other sources of potential bias and uncertainty were evaluated. Previous studies of decay heat calculations found that the effect of using assembly average characteristics in spent fuel calculations was small, although not insignificant (Ref. 14). In particular, use of the assembly average burnup was found to produce decay heat generation rates that were about 1% different from those obtained by simulating the axial burnup profile of the assembly. The effect was found to be nonconservative for cooling times less than about 50 years and conservative thereafter. The effect of using assembly average burnup and axial void conditions was found to have a larger influence of up to 3% that is most pronounced in the first 10 years of cooling time. The evaluation of decay heat measurements for BWR assemblies in this time regime found that the guide overpredicts the decay heat due to conservatism in the methods, the potential nonconservative effect of using assembly averaged parameters is adequately addressed in the safety margin. Uncertainties associated with factors not explicitly addressed by the guide including potential differences between the assembly models used in preparing the guide and actual assemblies, variations in BPR configurations, differences with the assemblies used for validation, and variations in plant operation, are also deemed to be bounded by the safety factors developed for the guide and, thus, no additional margin is warranted. The many factors that may contribute to uncertainty in the guide are bounded by the safety factor developed using validation data over the range of the measurements which cover a large part of the application range.

The safety factor, applied to both PWR and BWR fuel, is illustrated in Figure 4.8.

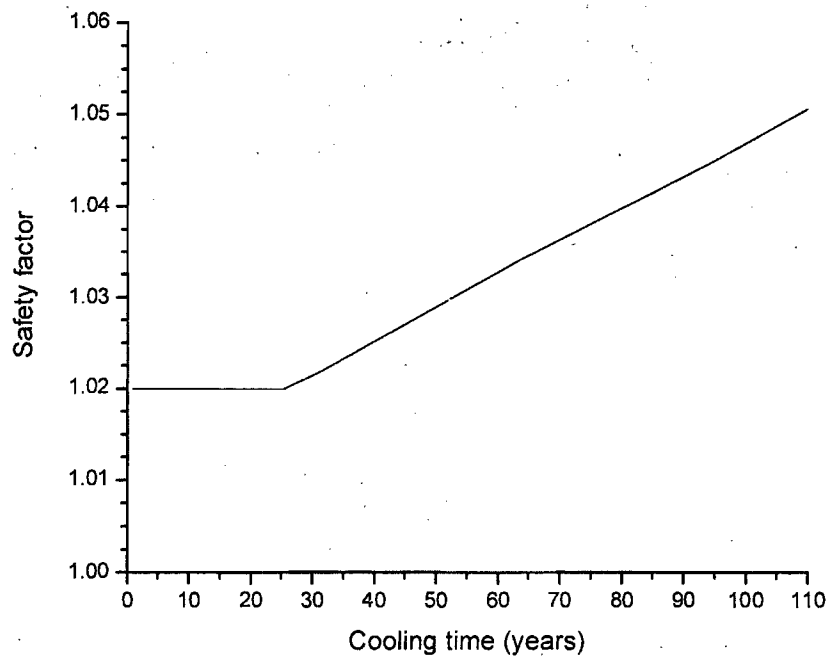

Figure 4.8. Safety factor over the range of cooling times of the guide.

4.6 RANGE OF APPLICATION

The development of the safety factor, discussed in Sect. 4.5, includes a margin for uncertainty over the range of the calorimeter measurements used to validate the methods and an additional margin for uncertainty for cooling times beyond the range of direct measurements. As discussed in Sect. 2.3, the target burnup limits of the proposed guide exceed the validated range established by the calorimeter measurements. As established in previous studies on nuclide importance (Refs. 15 and 16), the most significant impact of increasing burnup on decay heat is the increasing contribution of the actinides, mainly due to ^{244}Cm and ^{238}Pu.

Experimental isotopic assay measurements have been used extensively in the development of the guide and support the extension of the proposed guide beyond the range of direct calorimeter measurements. The details of the measurements are reported in Refs. 18–21, and the experimental data have been evaluated against the predictions of the ORIGEN-S code, used in the development of actinide data for use in the proposed guide. The isotopic measurements, summarized in Table 2.2, include spent fuel samples with enrichments higher than those of the assembly calorimeter measurements and burnup values that exceed the target ranges of the proposed guide. Evaluation of the validation results for the major actinides important in high burnup fuel decay heat, shown in Figures 4.6 and 4.7, do not indicate any significant trends in the accuracy of the ORIGEN-S calculations beyond about 50 GWd/MTU. Therefore, the methods developed for the proposed guide are judged to be equally applicable to the range of application beyond the limits of the calorimeter measurements, and no additional margin in the safety factor is deemed to be required.

Further justification for this position can be found in sensitivity and uncertainty calculations documented in Appendix A of Ref. 17. The global sensitivity of the calculated decay heat to the cross-section values

is shown to be highly similar between fuels with seemingly very different properties. Low enrichment fuel with burnup values consistent with the typical range for the enrichment was found to be very similar to higher enrichment fuel with a correspondingly higher burnup that is consistent with the enrichment. In other words, it is not just the burnup value that is important to the accuracy of decay heat calculations, it is the combination of enrichment and burnup. In the reported study (Appendix A, Ref. 17), the sensitivity of the decay heat to the cross-section values used in a calculation for fuel with an enrichment of 5 wt % ^{235}U irradiated to 65 GWd/MTU is effectively the same as for fuel with an enrichment of 3 wt % irradiated to about 45 GWd/MTU, a regime that is validated by direct calorimeter measurement. These studies lend further support justifying the applicability of the methods over the full range of the proposed enrichment and burnup limits of the guide.

5 METHODS EVALUATION AND VALIDATION

The quality and accuracy of the proposed methods for the expanded guide are validated by direct comparisons with experimental measurements. This section describes the validation of the proposed methods using available experiment data that include calorimeter measurements of full-length fuel assemblies with cooling times of up to 28 years. Validation beyond 28 years is supported by measurements of the isotopic composition for the dominant decay heat nuclides in spent fuel made by destructive radiochemical assay of irradiated fuel samples. The data are applied to benchmark the methods and provide estimates of bias and uncertainty over the full range of application of the guide, which were used to develop the safety factors described in Sect. 4.5.

5.1 CALORIMETER MEASUREMENTS

The methods for the proposed regulatory guide were used to calculate decay heat for 132 different spent fuel assemblies measured at GE-Morris and Hanford in the United States and CLAB in Sweden. Complete descriptions of the assemblies, operating histories, and calorimeter measurement results are provided in Ref. 14. Measurements made at GE-Morris and Hanford included 58 BWR assemblies from the Cooper and Monticello reactors, all with similar operating characteristics, and 10 PWR assemblies from the Point Beach 2 and Turkey Point 3 reactors. Cooling times were between 2 years and 11 years after discharge. The measurements made at CLAB included 30 PWR assemblies from the Ringhals 2 and 3 reactors and 34 BWR assemblies from the Ringhals, Barsebäck, Forsmark, and Oskarsham reactors. The cooling times of assemblies measured at CLAB ranged from 11 to 28 years after discharge.

The measured assembly designs included:

- CE 14 × 14
- W 14 × 14
- W 15 × 15
- W 17 × 17
- GE 7 × 7
- ABB 8 × 8
- SVEA 64 (8 × 8)
- ABB 9 × 9
- SVEA 100 (10 × 10)

Several assemblies had multiple measurements made at different cooling times. In most cases, these cooling times of repeated measurements were similar and were designed to provide a measure of experimental uncertainty, but they do not significantly expand the validation database. In this study, only one measurement for each assembly was selected for the validation study. The following assemblies were excluded because they had characteristics that were beyond the range of application of the proposed guide.

- San Onofre assemblies measured at GE-Morris were not used because the assemblies used stainless steel cladding with high cobalt content that was a dominant decay heat source at the time of measurement.

- Cooper assembly CZ102 was excluded because the enrichment was less than 2 wt % ^{235}U.
- Dresden assembly DN212 and Monticello assembly MT264 had a burnup less than 10 GWd/MTU, the minimum range of the guide.

The integral decay heat results for each assembly, calculated using the procedures described in Sect. 3, are listed in Appendix A, Tables A.1 and A.2. The results, illustrated in Figures 5.1 and 5.2, show that the decay heat power predicted using the proposed guide, without the safety factor included, is in good agreement with the measurements. Furthermore, no significant trends with cooling time are observed. The calculated results for all assemblies are on average 6.8% greater than the measurements, with a relative standard deviation (RSD) of 4.8%. Analysis of the results by measurement facility (Figure 5.3) shows that the CLAB assemblies are overpredicted on average by 4.4% using the proposed guide, with an RSD of 2.3%. The decay heat values calculated for the assemblies measured at GE-Morris and Hanford are overpredicted by 9.1% on average, with an RSD of 5.4%. The predicted decay heat values for all evaluated assemblies, without any added safety factor, are greater than the measurements for 98% of the assemblies. For the two assemblies that were nonconservative with respect to the measurements, the calculated decay heat was within 1% of measurement. When evaluating the results it is important to consider the error associated with the measurements. The estimated relative standard error of the GE-Morris, Hanford, and CLAB calorimeter measurements is ± 5%, 10%, and 2%, respectively.

Analysis of the results from each measurement facility indicates that GE-Morris results (dominated by the large number of Cooper BWR GE 7 × 7 assembly measurements) exhibit larger variability that other measurements. The values calculated using the proposed guide are overpredicted by about 9% on average, a value similar to the estimated experimental error. Analysis of the CLAB results shows smaller variability and a mean positive bias of about 4%, a value again close to the experimental error. These results are obtained prior to use of the safety factor.

The results obtained using the guide are compared with the experimental data plotted as a function of assembly average burnup in Figure 5.4. The results show no apparent trends with increasing burnup, further supporting the applicability of the guide to the extended burnup range.

To further evaluate the guide, the ORIGEN-S code was used to calculate the contribution for each component of decay heat addressed by the guide to provide additional insight into the source of differences. The calculations were performed using enrichment- and burnup-dependent cross-section libraries and simulated details of the assembly irradiation history. To calculate the fission product contribution without neutron capture for purposes of comparison with the fission product methods adopted from the ANS-5.1 standard, a modification to the ORIGEN code was implemented to remove neutron capture by fission products to calculate a decay heat value that is equivalent to that calculated by the standard (i.e., without neutron capture effects).

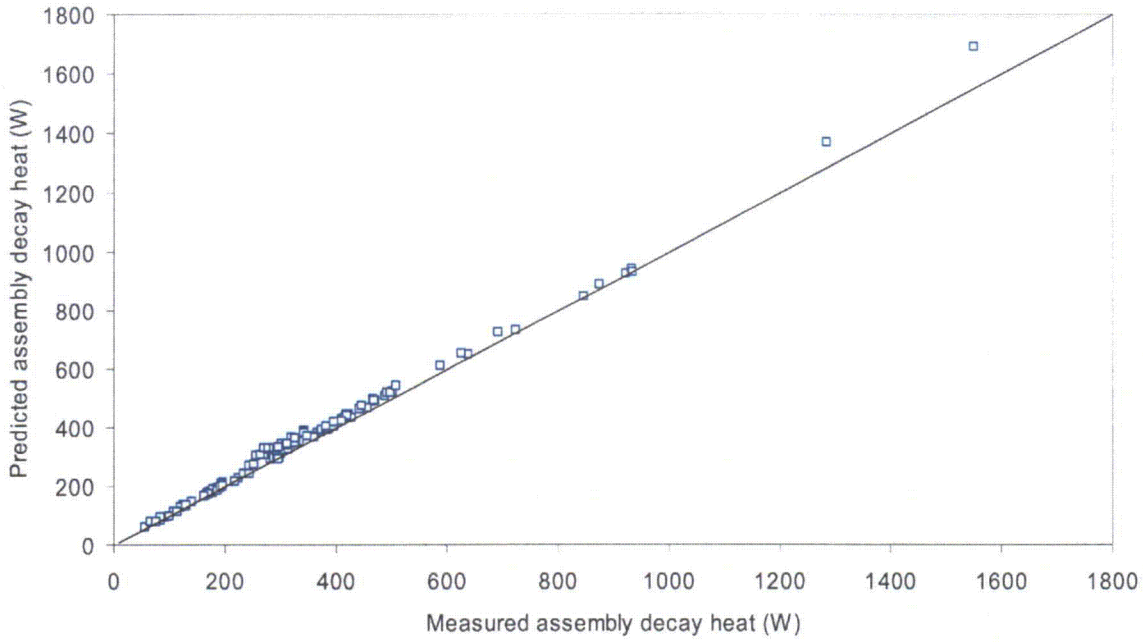

Figure 5.1. Comparison of measured decay heat and calculated decay heat obtained using the proposed guide excluding the safety factor for all evaluated assemblies.

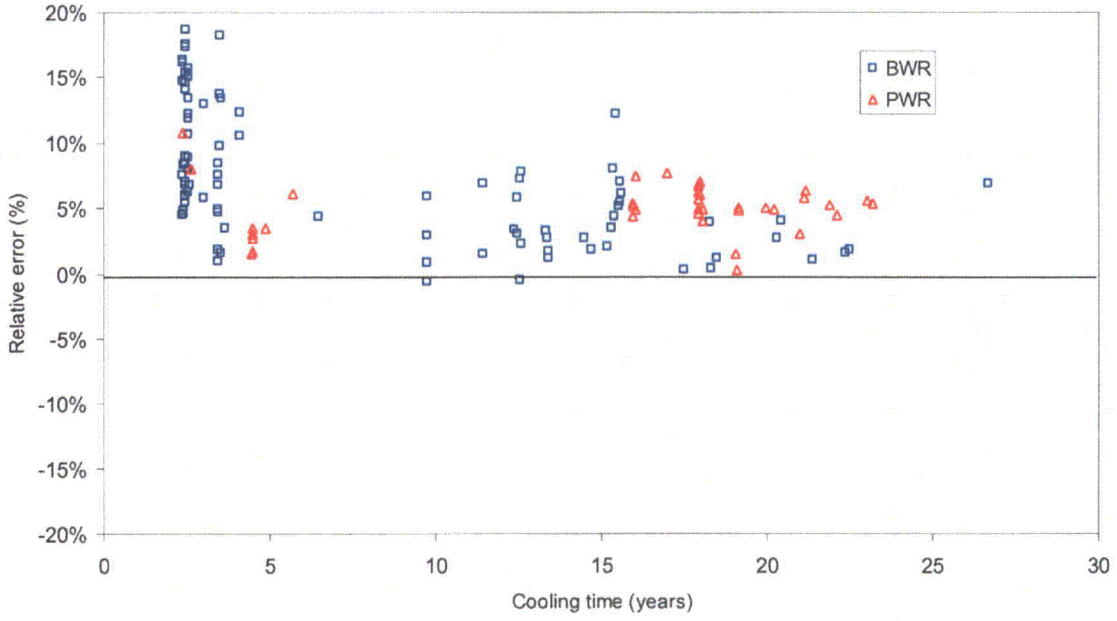

Figure 5.2. Comparison of measured assembly decay heat calculated decay heat using the proposed guide excluding the safety factor by reactor type as a function of assembly cooling time.

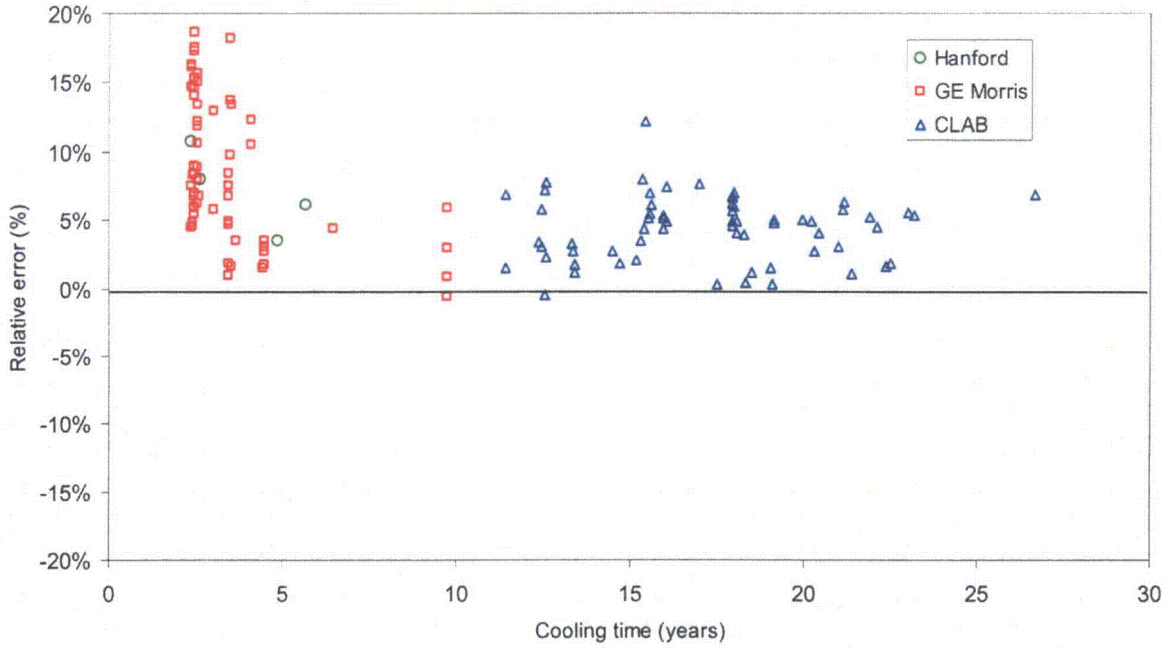

Figure 5.3. Residual error in the assembly decay heat calculated using the proposed guide, excluding the safety factor, shown as a function of cooling time for each measurement facility.

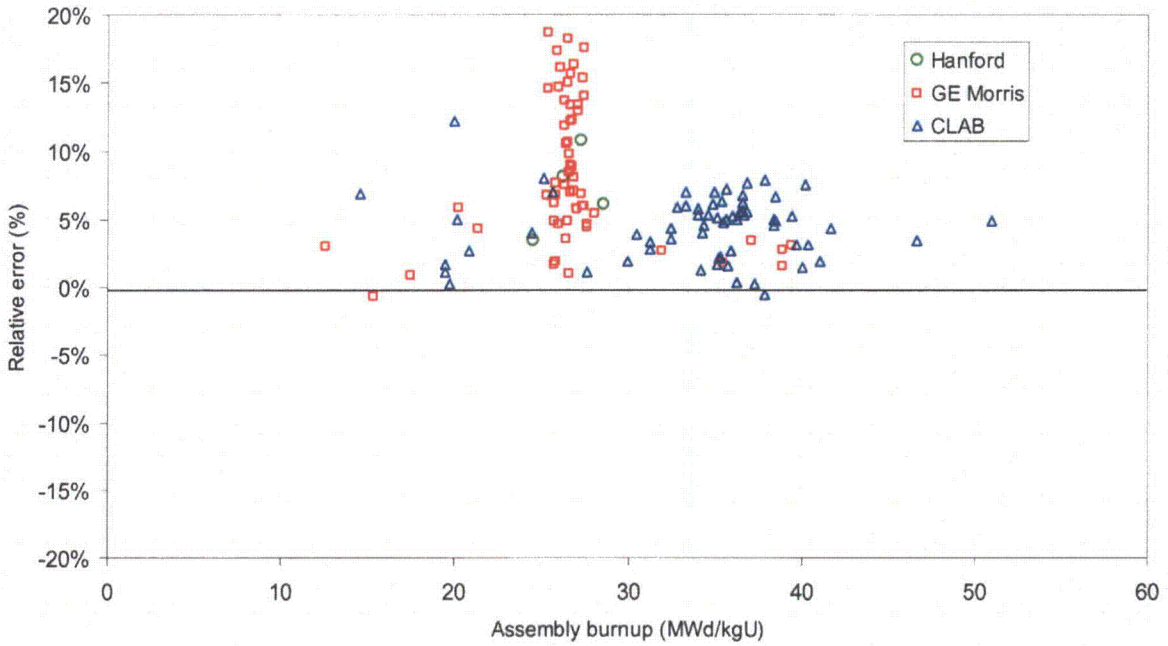

Figure 5.4. Relative error in the assembly decay heat calculated using the proposed guide, excluding the safety factor, shown as a function of average assembly discharge burnup for each measurement facility.

The guide (without addition of the safety factor) predicts, on average, decay heat values for the PWR assemblies that are about 3% greater than detailed ORIGEN-S calculations. The results for BWR assemblies were overestimated relative to ORIGEN-S for cooling times less than 10 years. The maximum overprediction of 14% was observed for BWR assemblies with cooling times of 2–3 years.

The BWR measurement data for cooling times less than 11 years is obtained for Cooper reactor assemblies (CZ assembly designations). The majority of Cooper assemblies were irradiated in the reactor core three consecutive cycles, removed from the core and stored for two cycles, and then reinserted in the core for two more cycles before being discharged. The extended storage time between cycles is explicitly simulated in the calculation of fission product decay heat power; however, the correction for neutron capture (mainly ^{134}Cs) is conservatively treated without considering decay time. Further comparison of the proposed guide calculations with results from ORIGEN-S indicate that most of the difference may be attributed to the calculation of the ^{134}Cs capture correction and the contribution from activated structural materials. Both of these components are important at cooling times from about 1 to 8 years. The overprediction using the proposed guide is attributed to the use of PWR fuel parameters in calculating activation of structural materials and fission product effects for both PWR and BWR fuel. The neutron flux levels in PWR fuel are higher than BWR fuel and result in larger capture rates (fission product and activated components). The use of PWR parameters thus produces conservative results when applied to BWR fuel.

5.2 COMPARISON WITH THE CURRENT GUIDE

The proposed guide discussed in this report is developed to improve the accuracy of decay heat calculations and provide increased flexibility to represent a wider range of fuel types and operating conditions than the current guide it is designed to replace. The previous section described the validation of the revised methodology of the proposed guide against experimental measurements of decay heat. This section compares the results obtained using the current regulatory guide for decay heat for the same set of assembly measurements to compare the new methodology against the existing approach.

The decay heat results obtained using the current regulatory guide, RG 3.54, for the 132 measured assemblies are summarized in Appendix B. Two assemblies could not be evaluated because the burnup exceeded the range of the current guide. Two other assemblies could not be evaluated because the specific power in the last cycle was higher than the average specific power level by more than 30%, which exceeded the recommended range of application of the current guide. The deviations, given as the percent difference between the values predicted by the guide and the measurements, are illustrated in Figure 5.5 for the evaluated assemblies. The results are shown with and without the added safety factor. For comparison, results obtained using the proposed new guide are shown for the same assemblies. The results are plotted as a function of cooling time of the assembly. Assemblies with cooling times less than 11 years were measured in the United States, and assemblies with longer cooling times were measured in Sweden.

The results obtained using the proposed guide are in better agreement with the measurements than the current guide, but are still conservative for most measured assemblies. The results calculated with the proposed guide, before application of the safety factor, are on average 6.8% greater than the measurements, with a standard deviation of 4.8%. The results calculated with the current guide for the same assemblies are on average 10.9% greater than the measurements, with a standard deviation of 9.4%. The safety factor in the proposed guide is 1.02 for all measured assemblies, compared to values that ranged from 1.06 to 1.10 in the current guide.

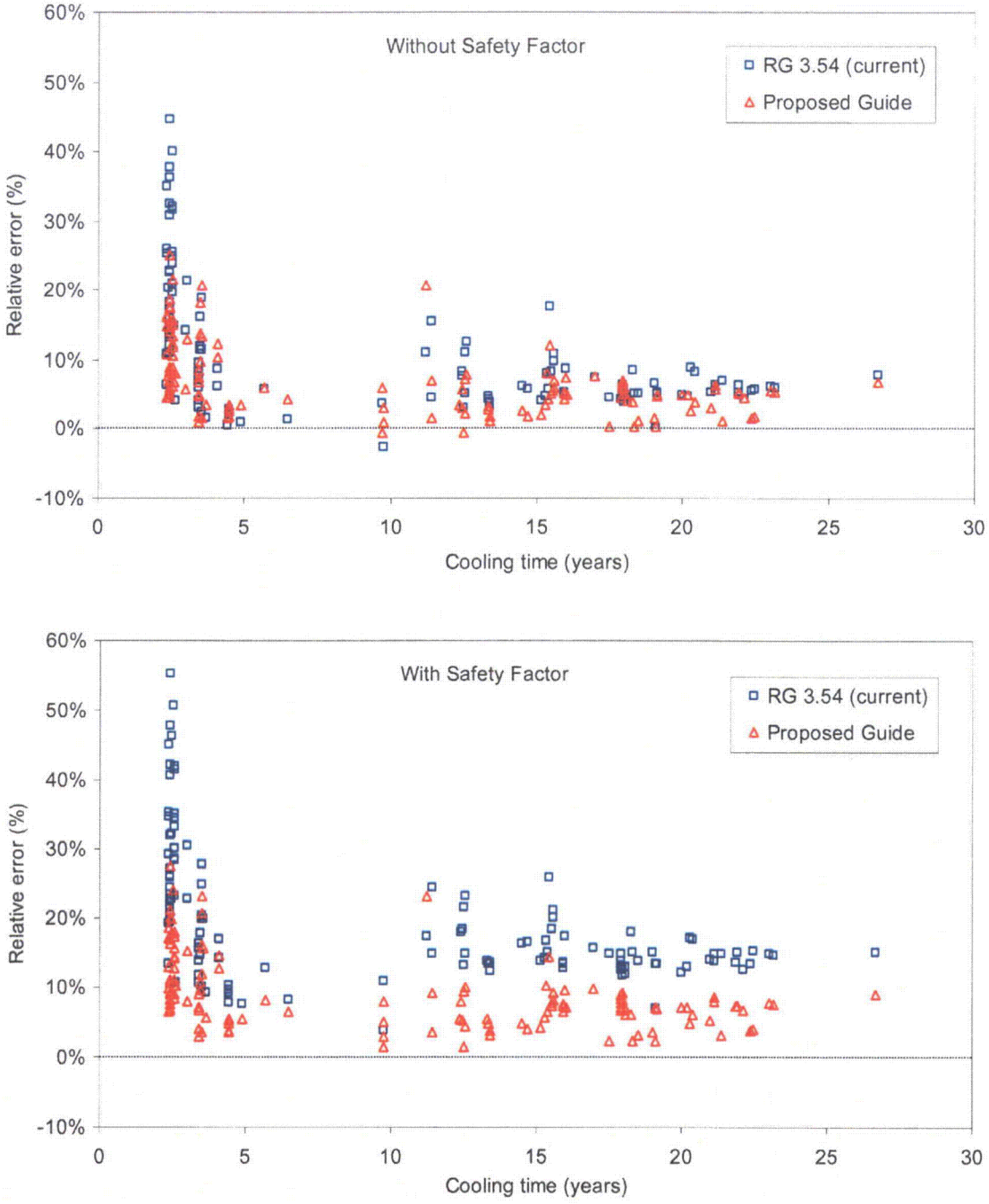

Figure 5.5. Comparison of differences between assembly decay heat values calculated using the current decay heat regulatory guide and the proposed new guide with measured values. Calculated values are shown without added safety factor (top) and with the safety factor included (bottom).

Some of the largest differences between the new and current approaches are seen for the assemblies with cooling times near 2.5 years. These assemblies are predominantly BWR assemblies from the Cooper reactor. Many of the Cooper assemblies were removed from the core for several cycles before being reintroduced in later cycles for further irradiation. Extended cooling times during the irradiation history cannot be accounted for in the current guide, leading to conservative decay heat predictions. The largest discrepancies are observed for assemblies that experienced such an operating history. The proposed guide does not have such operating history limitations because the fission product component of decay heat, which is dominant for the cooling times of the Cooper assemblies, is calculated using a detailed operating history that takes into account both the variations in operating power during irradiation cycles and extended downtimes between cycles. When the results for Cooper assemblies that resided in storage for one or more cycles during their irradiation history are removed from the data, the comparisons using the current guide are improved, with an average overprediction of 6.2% and standard deviation of 3.5%. However, even with the most discrepant assemblies removed from the comparison, the new proposed guide yields results that are more accurate and exhibit a smaller variance than the results obtained using the current guide.

The calculated values of decay heat using the proposed guide with the safety factor applied are on average 8.9% greater than the measurements. Calculated values of decay heat are greater than the measured values for all assemblies. The current guide results in decay heat values that are on average 19.5% greater than the measurements with safety factors applied.

5.3 SUMMARY

In summary, the proposed revision to the regulatory guide has been validated against measurements of decay heat for 132 assemblies that cover a wide range of fuel assembly designs, enrichments, and burnup. The measurements expand the validated cooling time range from 11 years, using previously available calorimeter measurements (GE-Morris and Hanford), to 28 years, using new measurements performed at CLAB. Validation beyond the range of cooling times of the calorimeter measurements is supported by spent fuel isotopic measurements for the actinides that become increasingly important as the fission products decay with cooling time. Isotopic validation results for the dominant decay heat generating actinides have been applied to ensure that actinide predictions are consistent with experimental observations. Uncertainties, developed from comparisons of calculations using the guide with calorimeter measurements, are applied in determining appropriate safety factors that ensure the decay heat values obtained using the guide are conservative with a statistical confidence level of 95%. Beyond the range of the calorimeter data, code calculations are used to develop the appropriate actinide data with uncertainties in the data established from spent fuel isotopic benchmarks and propagated to estimate the combined uncertainty in the decay heat power from the sum of the individual contributing nuclides.

The maximum burnup of the measured BWR assemblies was 46.6 GWd/MTU and the maximum burnup for the PWR assemblies was 50.9 GWd/MTU. The range of assembly burnup covered by the proposed guide extends beyond the measured assemblies by 8.4 GWd/MTU, or about 20%, for BWR assemblies and by 14.0 GWd/MTU, or 28%, for PWR assemblies. Again, validation in this extended high burnup regime is supported by isotopic measurement data for the actinides that become increasingly important as burnup increases. The range of the enrichments for samples with isotopic measurements (Table 2.2) adequately covers the range of the guide. The burnup of the measured fuel samples extends well beyond the proposed range of application of the guide.

The procedures and data proposed in the guide are applicable to a wide range of commercial nuclear power plant operations and most current spent fuel in storage with only minor limitations in applicability. The limitations specifically pertain to the development of the contributions from neutron capture by

fission products (except for ^{134}Cs) and activated assembly structural materials. Limitations in the proposed methods may be addressed by revising or extending the components that are affected without changing other components of the guide.

The methods are also amenable to further extension of the range of application to include longer and shorter cooling times. The methods in the proposed revision of the guide are applicable to calculating decay heat in the time range from 1 to 110 years after discharge from the reactor. The procedures implemented for the fission products, adopted from the ANS-5.1-2005 standard, are developed to be accurate at cooling times immediately after fission. The methods, as implemented in the guide, could be extended in the future to include cooling times less than 1 year to cover fuel handling and interim storage applications. However, such a revision would require appropriate reevaluation of the other components of decay heat calculated in the guide to ensure that they continue to produce safe values over any extended time domain.

6 EXAMPLE CALCULATION

The following example illustrates the use of the proposed guide for calculating the decay heat generation rate for a spent fuel assembly.

6.1 POINT BEACH ASSEMBLY C-64

The assembly selected in the example, designated C-64, is a W 14 × 14 design assembly irradiated for three consecutive cycles in the Point Beach PWR. The assembly has an initial enrichment of 3.397 wt % ^{235}U and was discharged March 3, 1977, after achieving an average burnup of 39,384 MWd/MTU. Decay heat measurements were performed at GE-Morris at 1,633 days or about 4.5 years after discharge from the reactor. The uranium mass of the assembly is 386.63 kg. The operating history and the accumulated burnup in each cycle are given in Table 6.1. The specific power for each cycle is derived from the accumulated assembly burnup and cycle time listed in the table according to Eq. (4). Note that cycle 1 was divided into two parts corresponding to an initial phase of low-power commissioning operation followed later by full-power operation.

Table 6.1. Irradiation data for Point Beach assembly C-64

Operational data	Cycle 1	Cycle 2	Cycle 3	Cycle 4
Startup date	8/1/1972	5/1/1973	12/20/1974	3/29/1976
Shutdown date	5/1/1973	10/16/1974	2/26/1976	3/3/1977
Operating days	273	533	433	339
Downtime (days)	0	65	32	1633
Cumulative burnup (MWd/kgU)	1.061	16.920	29.764	39.384
Power (W/kgU)	3,886	29,754	29,663	28,378

The decay heat generation rate, without the safety factor, is calculated according to the guide as the sum of each component

$$P_T(t,T) = P_F(t,T) + P_C(t,T) + P_E(t,T) + P_A(t,T) + P_S(t,T). \quad (20)$$

6.1.1 Calculation of P_F

The fission product decay heat, $P_F(t,T)$, uncorrected for neutron capture, is calculated according to Eq. (3). The fraction of the total specific operating power associated with the fission of ^{235}U, ^{239}Pu, ^{238}U, and ^{241}Pu is determined by interpolating the data in Table 3.4 using the assembly enrichment of 3.397 wt % ^{235}U and the midpoint burnup of each cycle. The fraction of assembly power, S_i/S, associated with each fissionable nuclide and the specific power from each nuclide, S_i, are listed in Table 6.2.

The fission product decay heat power is determined from the sum of the contributions of each fissionable isotope and irradiation cycle according to Eq. (3) using the specific operating power for each isotope listed in Table 6.2 and the Q values listed in Table 3.3. The irradiation time and decay time (T_k and t_k) for each cycle, in days, are listed in Table 6.3.

Table 6.2. Power fractions and specific power during operating cycles

Cycle	Mid-cycle burnup (MWd/kgU)	^{235}U		^{239}Pu		^{238}U		^{241}Pu	
		S_i/S	S_i ($\times 10^{-3}$)	S_i/S	S_i ($\times 10^{-3}$)	S_i/S	S_i ($\times 10^{-3}$)	S_i/S	S_i ($\times 10^{-3}$)
1	0.531	0.9231	3.5873	0.0167	0.0649	0.0598	0.2325	0.0003	0.0013
2	8.991	0.6861	20.415	0.2330	6.9339	0.0668	1.9866	0.0141	0.4188
3	23.342	0.4495	13.333	0.3982	11.813	0.0776	2.3027	0.0747	2.2145
4	34.574	0.3124	8.8656	0.4757	13.498	0.0861	2.4439	0.1258	3.5704

Table 6.3. Irradiation and decay times for operating cycles

$T_1 = 273$ days	$t_1 = t + 1402$ days
$T_2 = 533$ days	$t_2 = t + 869$ days
$T_3 = 433$ days	$t_3 = t + 371$ days
$T_4 = 339$ days	$t_4 = t$

The values of t_k are determined as the total time from the end of cycle k to the desired cooling time of $t = 1{,}633$ days after discharge. All times are to be expressed in seconds. The calculated fission product decay heat power, in units of watts per kilogram of uranium, is listed in Table 6.4 for each fissioning nuclide and each irradiation time step. The sum over all nuclides and cycles yields a total contribution due to fission product decay heat, uncorrected for neutron capture, of 1.505 W/kgU.

Table 6.4. Results of fission product decay heat power

Cycle	P_F (W/kgU)				
	^{235}U	^{239}Pu	^{238}U	^{241}Pu	Sum
1	0.0314	0.0004	0.0015	0.0000	0.0334
2	0.3705	0.0933	0.0281	0.0056	0.4974
3	0.2271	0.1786	0.0338	0.0353	0.4748
4	0.1481	0.2408	0.0395	0.0714	0.4998
Total	0.7772	0.5130	0.1029	0.1123	1.5054

6.1.2 Calculation of P_C

The calculation of the neutron capture correction for ^{134}Cs, $P_C(t,T)$, is calculated according to the procedures of Sect. 3.2.2.1. The required input parameters are the total irradiation time, the initial enrichment, the cooling time, and the average power density of the fuel. The total irradiation time is $T_1 + T_2 + T_3 + T_4 = 1{,}578$ days, and the cooling time is 1,633 days. Again, all time units must be converted to seconds.

The neutron flux in the fuel is determined from the initial enrichment and the specific power density according to Eq. (9). The average power density of the fuel is determined from the final assembly

burnup, 39.384 MWd/kgU, and the total irradiation time of 1,578 days, according to Eq. (5), yielding a value of 24,958 W/kgU. The effective enrichment, α, is calculated to be 2.6985. These values, applied in Eq. (9), yield a neutron flux of 2.386×10^{14} n/cm^2/s, a value that is appropriate for use in computing the production rate of ^{134}Cs. The average value of S/Q, used in Eq. (6), is determined according to Eq. (8) using the specific power during operation generated by each nuclide (S_i) listed in Table 6.2 and the recommended Q values for each isotope listed in Table 3.3 (units of MeV). This value, applied in Eq. (6), yields a calculated decay heat power for ^{134}Cs of 0.4365 W/kgU.

6.1.3 Calculation of P_E

The contribution due to neutron capture by other fission products is calculated by linear interpolation of the tabulated values in Table 3.5 for a cooling time of 1,633 days, yielding a value for $H(t)$ of 0.0348. This factor is multiplied by the fission product decay heat power $P_F(t,T)$ of 1.505 W/kgU, according to Eq. (10), yielding a contribution, $P_E(t,T)$, due to neutron capture effects of 0.0523 W/kgU.

6.1.4 Calculation of P_A

The actinide contribution, $P_A(t,T)$, is calculated by interpolating tabulated values of β for each actinide listed in Table 3.6 according to the average enrichment and total burnup of the assembly at discharge and determining the total actinide heating from the sum of all components according to Eq. (11) using actinide coefficients determined from Eq. (12). The interpolated values of β_n and derived values of $\hat{\beta}_n$ are listed in Table 6.5.

Table 6.5. Power fractions and specific power during operating cycles

n	Actinide	λ (1/s)	β_n (W/kgU)	$\hat{\beta}_n$ (W/kgU)	$\hat{\beta}_n e^{-\lambda_n t}$
1	^{241}Am	5.078×10^{-11}	1.023×10^{-2}	1.848×10^{-2}	1.834×10^{-1}
2	^{241}Pu	1.531×10^{-9}	4.845×10^{-3}	-1.697×10^{-1}	-1.367×10^{-1}
3	^{240}Pu	3.347×10^{-12}	1.841×10^{-2}	1.841×10^{-2}	1.840×10^{-2}
4	^{239}Pu	9.111×10^{-13}	1.100×10^{-2}	1.100×10^{-2}	1.100×10^{-2}
5	^{238}Pu	2.504×10^{-10}	1.619×10^{-1}	1.619×10^{-1}	1.563×10^{-1}
6	^{244}Cm	1.213×10^{-9}	1.676×10^{-1}	1.676×10^{-1}	1.412×10^{-1}
7	^{242}Cm	4.923×10^{-8}	3.421×10^{0}	3.421×10^{0}	3.292×10^{-3}

The total actinide decay heat is calculated to be 0.377 W/kgU. A small correction factor of 0.989 is applied to this value to account for the operating power of 24.96 kW/kgU, yielding a final actinide decay heat power contribution of 0.373 W/kgU.

6.1.5 Calculation of P_S

The contribution from structural activation products, $P_S(t,T)$, is calculated by interpolating the data in Table 3.5 for the desired cooling time, yielding a value for $A(t)$ of 0.0762. This factor is multiplied by the fission product decay heat power $P_F(t,T)$ of 1.505 W/kgU to yield a contribution from activation products of 0.115 W/kgU.

6.1.6 Final Result with Safety Factor

The total decay heat generation rate is calculated according to the guide as the sum of each component:

$$P_T(t,T) = P_S(t,T) + P_C(t,T) + P_E(t,T) + P_A(t,T) + P_F(t,T)$$

or

$$P_T(t,T) = 1.505 + 0.437 + 0.052 + 0.373 + 0.115 = 2.482 \text{ W/kgU}.$$

The assembly initial uranium mass of 386.63 kgU, is multiplied by the specific decay heat generation rate for the assembly of 2.482 W/kgU to yield the assembly decay heat generation rate of 959.6 W/assembly, without the safety factor included. With the safety factor of 1.02 added, the final assembly decay heat rate, $P'_T(t,T)$, is calculated to be 978.8 W.

The measured decay heat generation rate of assembly C-64 is 931.0 W. The total decay heat predicted by the proposed guide, without the safety factor, exceeds the measured value by 28.6 W, or about 3%. The experimental uncertainty of the GE-Morris measurements is about 5%. The current guide (Ref. 1) predicts a value of 950.8 W with no safety factor included, 2.1% larger than measured. With the recommended safety factor in the current guide of 7.24% included, the current guide overpredicts the assembly decay heat by about 10%. The proposed guide applies a smaller safety factor of 2% based on the direct evaluation of the methods against calorimeter measurements. The final decay heat generation rate calculated with the proposed guide with the safety factor included is about 6% larger than the measured value.

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APPENDIX A

SPENT FUEL ASSEMBLY BENCHMARK DATA SUMMARY OF EXPERIMENTAL AND CALCULATED RESULTS OBTAINED USING PROPOSED GUIDE

Table A.1 Measured and calculated decay heat obtained using the proposed guide for PWR fuel assemblies

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated P[a] (W)	Percent difference[c] (C/M–1)%	Calculated P'[b] (W)	Percent difference[c] (C/M–1)%
Point Beach	14 × 14	C-52	3.397	386.54	31,914	3/3/1977	1635	724.0	743.2	2.7%	758.1	4.7%
Point Beach	14 × 14	C-56	3.397	386.80	38,917	3/3/1977	1634	921.0	946.0	2.7%	964.9	4.8%
Point Beach	14 × 14	C-64	3.397	386.63	39,384	3/3/1977	1633	931.0	959.4	3.1%	978.8	5.1%
Point Beach	14 × 14	C-66	3.397	386.54	35,433	3/3/1977	1630	846.0	860.9	1.8%	878.1	3.8%
Point Beach	14 × 14	C-67	3.397	386.45	38,946	3/3/1977	1629	934.0	948.3	1.5%	967.3	3.6%
Point Beach	14 × 14	C-68	3.397	386.36	37,059	3/3/1977	1630	874.0	904.1	3.4%	922.2	5.5%
Turkey Point	15 × 15	B-43	2.559	447.79	24,554	9/11/1980	1782	637.0	658.8	3.4%	672.0	5.5%
Turkey Point	15 × 15	D-15	2.557	456.12	28,588	7/28/1983	2072	625.0	662.9	6.1%	676.2	8.2%
Turkey Point	15 × 15	D-22	2.557	458.00	26,291	7/9/1980	958	1284.0	1387.0	8.0%	1414.7	10.2%
Turkey Point	15 × 15	D-34	2.557	455.24	27,313	4/1/1980	859	1550.0	1716.0	10.7%	1750.3	12.9%
Ringhals 2	15 × 15	C01	3.095	455.79	36,688	4/4/1981	8468	415.8	438.0	5.4%	446.8	7.5%
Ringhals 2	15 × 15	C12	3.095	453.74	36,385	4/4/1981	8403	410.3	433.2	5.6%	441.9	7.7%
Ringhals 2	15 × 15	C20	3.095	454.76	35,720	4/4/1985	6952	428.9	435.4	1.5%	444.1	3.5%
Ringhals 2	15 × 15	D27	3.252	432.59	39,676	4/28/1983	7669	456.1	470.1	3.1%	479.5	5.1%
Ringhals 2	15 × 15	D38	3.252	434.21	39,403	5/6/1982	8005	442.3	465.5	5.2%	474.8	7.3%
Ringhals 2	15 × 15	E38	3.199	433.59	33,973	5/6/1982	8000	374.3	394.1	5.3%	402.0	7.4%
Ringhals 2	15 × 15	E40	3.199	434.24	34,339	5/6/1982	8075	381.3	398.4	4.5%	406.4	6.6%
Ringhals 2	15 × 15	F14	3.197	436.38	34,009	4/28/1983	7722	381.8	403.7	5.7%	411.8	7.8%
Ringhals 2	15 × 15	F21	3.197	435.94	36,273	4/13/1984	7376	420.9	441.6	4.9%	450.4	7.0%
Ringhals 2	15 × 15	F25	3.197	437.29	35,352	4/28/1983	7725	396.7	421.9	6.3%	430.3	8.5%
Ringhals 2	15 × 15	F32	3.197	436.99	50,962	5/12/1988	5860	692.0	725.8	4.9%	740.3	7.0%
Ringhals 2	15 × 15	G11	3.188	436.18	35,463	4/4/1985	6990	416.4	436.2	4.8%	444.9	6.9%
Ringhals 2	15 × 15	G23	3.206	436.13	35,633	4/4/1985	6984	420.6	441.5	5.0%	450.3	7.1%
Ringhals 2	15 × 15	I09	3.203	437.35	40,188	5/12/1988	5849	507.9	545.9	7.5%	556.8	9.6%
Ringhals 2	15 × 15	I24	3.203	429.60	34,294	4/30/1986	6601	410.1	426.4	4.0%	434.9	6.1%
Ringhals 2	15 × 15	I25	3.203	433.06	36,859	4/25/1987	6198	445.8	479.9	7.7%	489.5	9.8%
Ringhals 3	17 × 17	0E2	3.103	463.60	41,628	7/7/1988	5823	587.9	613.2	4.3%	625.5	6.4%
Ringhals 3	17 × 17	0E6	3.103	461.77	35,993	7/7/1988	5829	487.8	513.0	5.2%	523.3	7.3%
Ringhals 3	17 × 17	1E5	3.103	463.90	34,638	7/7/1988	5818	468.8	493.6	5.3%	503.5	7.4%
Ringhals 3	17 × 17	0C9	3.101	457.64	38,442	5/30/1986	6551	491.2	523.7	6.6%	534.2	8.8%

Table A.1. Measured and calculated decay heat obtained using the proposed guide for PWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated P[a] (W)	Percent difference[c] (C/M–1)%	Calculated P′[b] (W)	Percent difference[c] (C/M–1)%
Ringhals 3	17 × 17	1C2	3.101	459.05	33,318	5/30/1986	6559	417.7	446.7	7.0%	455.6	9.1%
Ringhals 3	17 × 17	1C5	3.101	457.99	38,484	5/30/1986	6593	499.2	523.5	4.9%	534.0	7.0%
Ringhals 3	17 × 17	2A5	2.100	462.03	20,107	5/11/1984	7297	233.8	245.4	5.0%	250.3	7.1%
Ringhals 3	17 × 17	2C2	3.101	459.49	36,577	5/30/1986	6550	466.5	498.0	6.7%	508.0	8.9%
Ringhals 3	17 × 17	3C1	3.101	458.43	36,572	5/30/1986	6545	470.2	496.9	5.7%	506.8	7.8%
Ringhals 3	17 × 17	3C5	3.101	458.87	38,373	5/30/1986	6543	501.4	524.3	4.6%	534.8	6.7%
Ringhals 3	17 × 17	3C9	3.101	459.14	36,560	5/30/1986	6552	468.4	497.5	6.2%	507.5	8.3%
Ringhals 3	17 × 17	4C4	3.101	459.05	33,333	5/30/1986	6572	422.0	447.3	6.0%	456.2	8.1%
Ringhals 3	17 × 17	4C7	3.101	458.26	38,370	5/30/1986	6549	498.7	523.4	4.9%	533.9	7.0%
Ringhals 3	17 × 17	5A3	2.100	461.48	19,699	5/11/1984	6977	243.4	244.0	0.2%	248.9	2.2%

[a]Calculated decay heat without the safety factor.
[b]Calculated decay heat with the safety factor.
[c](C/M – 1)% = (calculated/measured – 1) × 100%.

Table A.2. Measured and calculated decay heat obtained using the proposed guide for BWR fuel assemblies

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated P[a] (W)	Percent difference[c] (C/M–1)%	Calculated P′ [b] (W)	Percent difference[c] (C/M–1)%
Cooper	GE 7 × 7	CZ147	2.500	190.31	26,709	4/20/1981	1294	276.7	313.6	13.3%	319.9	15.6%
Cooper	GE 7 × 7	CZ148	2.500	190.22	26,310	4/20/1981	1282	273.5	311.0	13.7%	317.2	16.0%
Cooper	GE 7 × 7	CZ182	2.500	190.09	26,823	5/21/1982	860	342.6	398.5	16.3%	406.5	18.6%
Cooper	GE 7 × 7	CZ195	2.500	190.68	26,391	4/20/1981	1288	255.5	308.5	20.7%	314.7	23.2%
Cooper	GE 7 × 7	CZ205	2.500	190.72	25,344	5/21/1982	946	317.1	338.5	6.7%	345.3	8.9%
Cooper	GE 7 × 7	CZ209	2.500	190.38	25,383	5/21/1982	891	279.5	331.6	18.6%	338.2	21.0%
Cooper	GE 7 × 7	CZ211	2.500	190.82	26,679	4/20/1981	1260	296.0	321.1	8.5%	327.5	10.6%
Cooper	GE 7 × 7	CZ222	2.500	190.90	26,692	5/21/1982	898	355.7	380.4	6.9%	388.0	9.1%
Cooper	GE 7 × 7	CZ225	2.500	190.51	25,796	5/21/1982	865	333.5	358.8	7.6%	366.0	9.7%
Cooper	GE 7 × 7	CZ239	2.500	189.57	27,246	5/21/1982	893	366.5	391.5	6.8%	399.3	9.0%
Cooper	GE 7 × 7	CZ246	2.500	189.81	27,362	5/21/1982	899	341.7	394.3	15.4%	402.2	17.7%
Cooper	GE 7 × 7	CZ259	2.500	190.20	26,466	4/20/1981	1340	288.5	298.6	3.5%	304.6	5.6%
Cooper	GE 7 × 7	CZ264	2.500	190.89	26,496	4/20/1981	1282	263.8	311.9	18.2%	318.1	20.6%
Cooper	GE 7 × 7	CZ277	2.500	189.49	26,747	4/20/1981	1497	243.0	272.8	12.3%	278.3	14.5%
Cooper	GE 7 × 7	CZ286	2.500	189.95	27,141	5/21/1982	1103	284.2	320.9	12.9%	327.3	15.2%
Cooper	GE 7 × 7	CZ296	2.500	190.50	26,388	4/20/1981	1492	251.9	278.3	10.5%	283.9	12.7%
Cooper	GE 7 × 7	CZ302	2.500	190.00	26,594	4/20/1981	1283	285.6	313.4	9.7%	319.7	11.9%
Cooper	GE 7 × 7	CZ308	2.500	189.78	25,815	5/21/1982	895	269.7	337.4	25.1%	344.1	27.6%
Cooper	GE 7 × 7	CZ311	2.500	189.91	27,392	5/21/1982	890	356.9	378.3	6.0%	385.9	8.1%
Cooper	GE 7 × 7	CZ315	2.500	189.96	26,881	5/21/1982	932	328.0	354.3	8.0%	361.4	10.2%
Cooper	GE 7 × 7	CZ318	2.500	189.32	26,568	5/21/1982	931	277.6	337.2	21.5%	343.9	23.9%
Cooper	GE 7 × 7	CZ331	2.500	190.36	21,332	3/31/1978	2369	162.8	169.8	4.3%	173.2	6.4%
Cooper	GE 7 × 7	CZ337	2.500	189.90	26,720	5/21/1982	895	347.7	378.8	8.9%	386.4	11.1%
Cooper	GE 7 × 7	CZ342	2.500	190.16	27,066	5/21/1982	1101	300.0	317.4	5.8%	323.7	7.9%
Cooper	GE 7 × 7	CZ346	2.500	190.23	28,048	5/21/1982	890	388.7	409.9	5.5%	418.1	7.6%
Cooper	GE 7 × 7	CZ348	2.500	190.38	27,480	5/21/1982	894	342.8	390.9	14.0%	398.7	16.3%
Cooper	GE 7 × 7	CZ351	2.500	190.02	25,753	5/21/1982	934	313.8	333.3	6.2%	340.0	8.3%
Cooper	GE 7 × 7	CZ355	2.500	190.60	25,419	5/21/1982	891	290.5	332.8	14.6%	339.5	16.9%
Cooper	GE 7 × 7	CZ357	2.500	190.19	27,140	5/21/1982	932	320.3	363.1	13.4%	370.4	15.6%

Table A.2. Measured and calculated decay heat obtained using the proposed guide for BWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated P[a] (W)	Percent difference[c] (C/M–1)%	Calculated P′[b] (W)	Percent difference[c] (C/M–1)%
Cooper	GE 7 × 7	CZ369	2.500	190.20	26,575	5/21/1982	888	347.6	376.7	8.4%	384.2	10.5%
Cooper	GE 7 × 7	CZ370	2.500	190.23	26,342	4/20/1981	1257	293.6	315.7	7.5%	322.0	9.7%
Cooper	GE 7 × 7	CZ372	2.500	190.01	25,848	4/20/1981	1256	294.3	299.8	1.9%	305.8	3.9%
Cooper	GE 7 × 7	CZ379	2.500	190.18	25,925	5/21/1982	898	287.4	337.4	17.4%	344.1	19.7%
Cooper	GE 7 × 7	CZ398	2.500	189.83	27,478	5/21/1982	890	372.0	394.0	5.9%	401.9	8.0%
Cooper	GE 7 × 7	CZ415	2.500	189.72	25,863	4/20/1981	1255	289.3	308.9	6.8%	315.1	8.9%
Cooper	GE 7 × 7	CZ416	2.500	189.43	27,460	5/21/1982	894	319.8	376.1	17.6%	383.6	20.0%
Cooper	GE 7 × 7	CZ429	2.500	190.07	27,641	5/21/1982	889	385.6	403.3	4.6%	411.4	6.7%
Cooper	GE 7 × 7	CZ430	2.500	189.93	26,824	5/21/1982	894	353.3	378.1	7.0%	385.7	9.2%
Cooper	GE 7 × 7	CZ433	2.500	190.02	25,977	4/20/1981	1255	287.4	300.8	4.7%	306.8	6.8%
Cooper	GE 7 × 7	CZ460	2.500	190.18	26,511	5/21/1982	933	313.5	346.7	10.6%	353.6	12.8%
Cooper	GE 7 × 7	CZ466	2.500	189.86	26,077	5/21/1982	861	302.1	350.8	16.1%	357.8	18.4%
Cooper	GE 7 × 7	CZ468	2.500	189.78	26,757	5/21/1982	935	325.3	354.0	8.8%	361.1	11.0%
Cooper	GE 7 × 7	CZ472	2.500	190.12	25,957	5/21/1982	859	325.0	372.9	14.7%	380.4	17.0%
Cooper	GE 7 × 7	CZ473	2.500	189.76	26,517	5/21/1982	934	293.2	337.3	15.0%	344.0	17.3%
Cooper	GE 7 × 7	CZ498	2.500	189.69	26,482	5/21/1982	888	359.4	376.8	4.8%	384.3	6.9%
Cooper	GE 7 × 7	CZ508	2.500	190.68	26,357	5/21/1982	933	310.0	346.7	11.8%	353.6	14.1%
Cooper	GE 7 × 7	CZ515	2.500	190.48	25,737	4/20/1981	1254	294.0	308.4	4.9%	314.6	7.0%
Cooper	GE 7 × 7	CZ526	2.500	190.54	27,596	5/21/1982	864	395.4	412.9	4.4%	421.2	6.5%
Cooper	GE 7 × 7	CZ528	2.500	190.81	25,714	4/20/1981	1284	297.6	302.4	1.6%	308.4	3.6%
Cooper	GE 7 × 7	CZ531	2.500	189.90	26,699	5/21/1982	893	347.2	376.6	8.5%	384.1	10.6%
Cooper	GE 7 × 7	CZ536	2.500	190.17	26,589	4/20/1981	1256	295.2	298.1	1.0%	304.1	3.0%
Cooper	GE 7 × 7	CZ542	2.500	189.99	26,691	5/21/1982	932	311.9	349.9	12.2%	356.9	14.4%
Cooper	GE 7 × 7	CZ545	2.500	190.47	26,668	5/21/1982	935	295.2	341.5	15.7%	348.3	18.0%
Monticello	GE 7 × 7	MT116	2.250	193.53	17,482	9/12/1975	3559	114.9	115.9	0.9%	118.2	2.9%
Monticello	GE 7 × 7	MT123	2.250	193.53	14,152	3/15/1974	4100	66.8	80.6	20.6%	82.2	23.0%
Monticello	GE 7 × 7	MT133	2.250	193.53	20,189	9/12/1975	3555	129.0	136.6	5.9%	139.3	8.0%
Monticello	GE 7 × 7	MT190	2.250	193.53	15,312	9/12/1975	3557	99.2	98.6	-0.6%	100.6	1.4%
Monticello	GE 7 × 7	MT228	2.250	193.53	12,570	9/12/1975	3560	76.4	78.7	3.0%	80.2	5.0%
Barsebäck 1	8 × 8–1	9329	2.920	178.77	41,094	9/17/1988	5373	224.4	228.6	1.9%	233.2	3.9%

Table A.2. Measured and calculated decay heat obtained using the proposed guide for BWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated P[a] (W)	Percent difference[c] (C/M–1)%	Calculated P′[b] (W)	Percent difference[c] (C/M–1)%
Barsebäck 1	8 × 8–1	10288	2.950	179.16	35,180	9/17/1988	5534	185.8	189.6	2.0%	193.4	4.1%
Barsebäck 2	8 × 8–1	14076	3.150	179.57	40,010	7/2/1992	4177	240.3	243.8	1.5%	248.7	3.5%
Forsmark 1	8 × 8–1	3838	2.086	177.90	25,669	7/10/1992	4170	126.8	135.6	6.9%	138.3	9.1%
Forsmark 1	8 × 8–2	KU0100	2.976	174.92	34,193	8/17/1990	4893	185.3	187.5	1.2%	191.3	3.2%
Forsmark 1	9 × 9–5	KU0269	2.938	177.02	35,113	8/17/1990	4903	192.7	196.0	1.7%	199.9	3.7%
Forsmark 1	9 × 9–5	KU0278	2.939	177.13	35,323	5/24/1991	4595	195.4	199.8	2.3%	203.8	4.3%
Forsmark 1	9 × 9–5	KU0282	2.939	177.10	37,896	5/24/1991	4574	218.5	217.4	-0.5%	221.7	1.5%
Forsmark 2	8 × 8–1	5535	2.095	177.69	19,944	7/15/1988	5634	84.6	94.9	12.1%	96.8	14.4%
Forsmark 2	SVEA 64	11494	2.920	181.09	32,431	7/15/1988	5618	166.0	173.2	4.3%	176.7	6.4%
Forsmark 2	SVEA 64	11495	2.910	181.07	32,431	7/15/1988	5593	167.6	173.5	3.5%	177.0	5.6%
Forsmark 2	SVEA 64	13775	2.850	181.34	32,837	7/12/1991	4543	178.4	188.8	5.8%	192.6	7.9%
Forsmark 3	SVEA 100	13847	2.769	180.67	31,275	7/14/1990	4871	169.6	175.6	3.3%	179.1	5.4%
Forsmark 3	SVEA 100	13848	2.769	180.67	31,275	7/13/1990	4882	170.7	175.4	2.8%	178.9	4.8%
Ringhals 1	8 × 8–1	1177	2.642	180.59	36,242	8/2/1985	6690	177.9	178.5	0.3%	182.1	2.3%
Ringhals 1	8 × 8–1	1186	2.640	180.52	30,498	8/2/1985	6674	140.8	146.3	3.9%	149.2	6.0%
Ringhals 1	8 × 8–1	6423	2.900	177.70	35,109	8/5/1988	5669	174.2	183.1	5.1%	186.8	7.2%
Ringhals 1	8 × 8–1	6432	2.894	177.52	36,861	8/5/1988	5422	185.5	195.4	5.5%	199.3	7.6%
Ringhals 1	8 × 8–1	6454	2.898	177.68	37,236	8/15/1986	6395	186.3	186.8	0.3%	190.5	2.3%
Ringhals 1	8 × 8–1	8327	2.904	177.54	37,851	8/6/1991	4600	196.9	212.3	7.8%	216.5	10.0%
Ringhals 1	8 × 8–1	8331	2.910	177.69	35,903	9/15/1989	5291	187.0	192.0	2.7%	195.8	4.7%
Ringhals 1	8 × 8–1	8332	2.895	177.52	34,977	8/5/1988	5690	168.1	179.8	7.0%	183.4	9.1%
Ringhals 1	8 × 8–1	8338	2.911	177.60	34,830	8/5/1988	5695	169.5	179.8	6.1%	183.4	8.2%
Oskarshamn 2	8 × 8–1	1377	2.201	183.58	14,546	5/13/1977	9750	56.2	60.1	6.9%	61.3	9.0%
Oskarshamn 2	8 × 8–1	1389	2.201	183.65	19,481	7/15/1981	8171	83.9	85.3	1.6%	87.0	3.7%
Oskarshamn 2	8 × 8–1	1546	2.201	183.97	24,470	8/19/1983	7455	108.1	112.4	4.0%	114.6	6.1%
Oskarshamn 2	8 × 8–1	1696	2.201	184.25	20,870	8/19/1983	7411	92.4	94.9	2.7%	96.8	4.7%
Oskarshamn 2	8 × 8–1	1704	2.201	184.02	19,437	7/23/1982	7808	84.0	84.9	1.1%	86.6	3.1%
Oskarshamn 2	8 × 8–1	2995	2.699	179.38	29,978	7/15/1981	8211	130.5	132.9	1.8%	135.6	3.9%
Oskarshamn 2	8 × 8–1	6350	2.875	179.00	27,675	6/7/1985	6755	129.4	130.9	1.2%	133.5	3.2%

Table A.2. Measured and calculated decay heat obtained using the proposed guide for BWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated P[a] (W)	Percent difference[c] (C/M–1)%	Calculated P′[b] (W)	Percent difference[c] (C/M–1)%
Oskarshamn 2	SVEA 64	12684	2.902	182.32	46,648	8/2/1991	4519	282.7	292.3	3.4%	298.1	5.5%
Oskarshamn 3	8 × 8–1	12078	2.577	177.36	25,160	7/8/1988	5611	120.2	129.8	8.0%	132.4	10.1%
Oskarshamn 3	SVEA 100	13628	2.711	180.77	35,619	6/24/1991	4581	194.0	207.9	7.2%	212.1	9.3%
Oskarshamn 3	SVEA 100	13630	2.711	180.78	40,363	6/24/1991	4554	235.7	242.9	3.1%	247.8	5.1%

[a]Calculated decay heat without the safety factor.
[b]Calculated decay heat with the safety factor.
[c](C/M – 1)% = (calculated/measured – 1) × 100%.

APPENDIX B

SPENT FUEL ASSEMBLY BENCHMARK DATA COMPARISON OF EXPERIMENTAL AND CALCULATED RESULTS OBTAINED USING THE CURRENT AND THE PROPOSED GUIDE FOR DECAY HEAT

Table B.1. Comparison of decay heat obtain obtained using the proposed guide and RG 3.54 for PWR fuel assemblies

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated RG 3.54 (W)	Percent difference (C/M–1)%	Calculated New Guide (W)	Percent difference (C/M–1)%
Point Beach	14 × 14	C-52	3.397	386.54	31,914	3/3/1977	1635	724.0	793.4	9.6%	758.1	4.7%
Point Beach	14 × 14	C-56	3.397	386.80	38,917	3/3/1977	1634	921.0	1003.0	8.9%	964.9	4.8%
Point Beach	14 × 14	C-64	3.397	386.63	39,384	3/3/1977	1633	931.0	1019.6	9.5%	978.8	5.1%
Point Beach	14 × 14	C-66	3.397	386.54	35,433	3/3/1977	1630	846.0	923.3	9.1%	878.1	3.8%
Point Beach	14 × 14	C-67	3.397	386.45	38,946	3/3/1977	1629	934.0	1006.0	7.7%	967.3	3.6%
Point Beach	14 × 14	C-68	3.397	386.36	37,059	3/3/1977	1630	874.0	963.3	10.2%	922.2	5.5%
Turkey Point	15 × 15	B-43	2.559	447.79	24,554	9/11/1980	1782	637.0	684.2	7.4%	672.0	5.5%
Turkey Point	15 × 15	D-15	2.557	456.12	28,588	7/28/1983	2072	625.0	705.0	12.8%	676.2	8.2%
Turkey Point	15 × 15	D-22	2.557	458.00	26,291	7/9/1980	958	1284.0	1420.3	10.6%	1414.7	10.2%
Turkey Point	15 × 15	D-34	2.557	455.24	27,313	4/1/1980	859	1550.0	1756.5	13.3%	1750.3	12.9%
Ringhals 2	15 × 15	C01	3.095	455.79	36,688	4/4/1981	8468	415.8	476.3	14.6%	446.8	7.5%
Ringhals 2	15 × 15	C12	3.095	453.74	36,385	4/4/1981	8403	410.3	470.7	14.7%	441.9	7.7%
Ringhals 2	15 × 15	C20	3.095	454.76	35,720	4/4/1985	6952	428.9	492.9	14.9%	444.1	3.5%
Ringhals 2	15 × 15	D27	3.252	432.59	39,676	4/28/1983	7669	456.1	519.7	14.0%	479.5	5.1%
Ringhals 2	15 × 15	D38	3.252	434.21	39,403	5/6/1982	8005	442.3	508.4	14.9%	474.8	7.3%
Ringhals 2	15 × 15	E38	3.199	433.59	33,973	5/6/1982	8000	374.3	424.7	13.5%	402.0	7.4%
Ringhals 2	15 × 15	E40	3.199	434.24	34,339	5/6/1982	8075	381.3	428.9	12.5%	406.4	6.6%
Ringhals 2	15 × 15	F14	3.197	436.38	34,009	4/28/1983	7722	381.8	434.1	13.7%	411.8	7.8%
Ringhals 2	15 × 15	F21	3.197	435.94	36,273	4/13/1984	7376	420.9	475.2	12.9%	450.4	7.0%
Ringhals 2	15 × 15	F25	3.197	437.29	35,352	4/28/1983	7725	396.7	455.1	14.7%	430.3	8.5%
Ringhals 2	15 × 15	F32	3.197	436.99	50,962	5/12/1988	5860	692.0	---	---	740.3	7.0%
Ringhals 2	15 × 15	G11	3.188	436.18	35,463	4/4/1985	6990	416.4	472.0	13.4%	444.9	6.9%
Ringhals 2	15 × 15	G23	3.206	436.13	35,633	4/4/1985	6984	420.6	477.1	13.4%	450.3	7.1%
Ringhals 2	15 × 15	I09	3.203	437.35	40,188	5/12/1988	5849	507.9	595.4	17.2%	556.8	9.6%
Ringhals 2	15 × 15	I24	3.203	429.60	34,294	4/30/1986	6601	410.1	458.7	11.8%	434.9	6.1%
Ringhals 2	15 × 15	I25	3.203	433.06	36,859	4/25/1987	6198	445.8	515.7	15.7%	489.5	9.8%
Ringhals 3	17 × 17	0E2	3.103	463.60	41,628	7/7/1988	5823	587.9	668.1	13.6%	625.5	6.4%
Ringhals 3	17 × 17	0E6	3.103	461.77	35,993	7/7/1988	5829	487.8	553.0	13.4%	523.3	7.3%
Ringhals 3	17 × 17	1E5	3.103	463.90	34,638	7/7/1988	5818	468.8	528.5	12.7%	503.5	7.4%
Ringhals 3	17 × 17	0C9	3.101	457.64	38,442	5/30/1986	6551	491.2	563.7	14.8%	534.2	8.8%

Table B.1. Comparison of decay heat obtain obtained using the proposed guide and RG 3.54 for PWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated RG 3.54 (W)	Percent difference (C/M–1)%	Calculated New Guide (W)	Percent difference (C/M–1)%
Ringhals 3	17 × 17	1C2	3.101	459.05	33,318	5/30/1986	6559	417.7	470.8	12.7%	455.6	9.1%
Ringhals 3	17 × 17	1C5	3.101	457.99	38,484	5/30/1986	6593	499.2	563.6	12.9%	534.0	7.0%
Ringhals 3	17 × 17	2A5	2.100	462.03	20,107	5/11/1984	7297	233.8	262.0	12.1%	250.3	7.1%
Ringhals 3	17 × 17	2C2	3.101	459.49	36,577	5/30/1986	6550	466.5	530.7	13.8%	508.0	8.9%
Ringhals 3	17 × 17	3C1	3.101	458.43	36,572	5/30/1986	6545	470.2	529.5	12.6%	506.8	7.8%
Ringhals 3	17 × 17	3C5	3.101	458.87	38,373	5/30/1986	6543	501.4	564.2	12.5%	534.8	6.7%
Ringhals 3	17 × 17	3C9	3.101	459.14	36,560	5/30/1986	6552	468.4	530.1	13.2%	507.5	8.3%
Ringhals 3	17 × 17	4C4	3.101	459.05	33,333	5/30/1986	6572	422.0	471.2	11.6%	456.2	8.1%
Ringhals 3	17 × 17	4C7	3.101	458.26	38,370	5/30/1986	6549	498.7	563.2	12.9%	533.9	7.0%
Ringhals 3	17 × 17	5A3	2.100	461.48	19,699	5/11/1984	6977	243.4	260.0	6.8%	248.9	2.2%

--- Assembly properties outside the range of current regulatory guide RG 3.54, Rev. 1.

Table B.2. Comparison of decay heat obtain obtained using the proposed guide and current RG 3.54 Rev. 1 for BWR fuel assemblies

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated RG 3.54 (W)	Percent difference (C/M–1)%	Calculated New Guide (W)	Percent difference (C/M–1)%
Cooper	GE 7 × 7	CZ147	2.500	190.31	26,709	4/20/1981	1294	276.7	331.7	19.9%	319.9	15.6%
Cooper	GE 7 × 7	CZ148	2.500	190.22	26,310	4/20/1981	1282	273.5	328.7	20.2%	317.2	16.0%
Cooper	GE 7 × 7	CZ182	2.500	190.09	26,823	5/21/1982	860	342.6	463.5	35.3%	406.5	18.6%
Cooper	GE 7 × 7	CZ195	2.500	190.68	26,391	4/20/1981	1288	255.5	326.1	27.6%	314.7	23.2%
Cooper	GE 7 × 7	CZ205	2.500	190.72	25,344	5/21/1982	946	317.1	390.4	23.1%	345.3	8.9%
Cooper	GE 7 × 7	CZ209	2.500	190.38	25,383	5/21/1982	891	279.5	412.8	47.7%	338.2	21.0%
Cooper	GE 7 × 7	CZ211	2.500	190.82	26,679	4/20/1981	1260	296.0	339.4	14.7%	327.5	10.6%
Cooper	GE 7 × 7	CZ222	2.500	190.90	26,692	5/21/1982	898	355.7	436.5	22.7%	388.0	9.1%
Cooper	GE 7 × 7	CZ225	2.500	190.51	25,796	5/21/1982	865	333.5	430.6	29.1%	366.0	9.7%
Cooper	GE 7 × 7	CZ239	2.500	189.57	27,246	5/21/1982	893	366.5	452.2	23.4%	399.3	9.0%
Cooper	GE 7 × 7	CZ246	2.500	189.81	27,362	5/21/1982	899	341.7	451.2	32.1%	402.2	17.7%
Cooper	GE 7 × 7	CZ259	2.500	190.20	26,466	4/20/1981	1340	288.5	315.1	9.2%	304.6	5.6%
Cooper	GE 7 × 7	CZ264	2.500	190.89	26,496	4/20/1981	1282	263.8	329.2	24.8%	318.1	20.6%
Cooper	GE 7 × 7	CZ277	2.500	189.49	26,747	4/20/1981	1497	243.0	284.1	16.9%	278.3	14.5%
Cooper	GE 7 × 7	CZ286	2.500	189.95	27,141	5/21/1982	1103	284.2	370.9	30.5%	327.3	15.2%
Cooper	GE 7 × 7	CZ296	2.500	190.50	26,388	4/20/1981	1492	251.9	287.8	14.2%	283.9	12.7%
Cooper	GE 7 × 7	CZ302	2.500	190.00	26,594	4/20/1981	1283	285.6	342.4	19.9%	319.7	11.9%
Cooper	GE 7 × 7	CZ308	2.500	189.78	25,815	5/21/1982	895	269.7	418.7	55.3%	344.1	27.6%
Cooper	GE 7 × 7	CZ311	2.500	189.91	27,392	5/21/1982	890	356.9	453.7	27.1%	385.9	8.1%
Cooper	GE 7 × 7	CZ315	2.500	189.96	26,881	5/21/1982	932	328.0	421.8	28.6%	361.4	10.2%
Cooper	GE 7 × 7	CZ318	2.500	189.32	26,568	5/21/1982	931	277.6	417.9	50.5%	343.9	23.9%
Cooper	GE 7 × 7	CZ331	2.500	190.36	21,332	3/31/1978	2369	162.8	176.2	8.2%	173.2	6.4%
Cooper	GE 7 × 7	CZ337	2.500	189.90	26,720	5/21/1982	895	347.7	438.0	26.0%	386.4	11.1%
Cooper	GE 7 × 7	CZ342	2.500	190.16	27,066	5/21/1982	1101	300.0	368.0	22.7%	323.7	7.9%
Cooper	GE 7 × 7	CZ346	2.500	190.23	28,048	5/21/1982	890	388.7	469.1	20.7%	418.1	7.6%
Cooper	GE 7 × 7	CZ348	2.500	190.38	27,480	5/21/1982	894	342.8	452.3	31.9%	398.7	16.3%
Cooper	GE 7 × 7	CZ351	2.500	190.02	25,753	5/21/1982	934	313.8	402.9	28.4%	340.0	8.3%
Cooper	GE 7 × 7	CZ355	2.500	190.60	25,419	5/21/1982	891	290.5	412.9	42.1%	339.5	16.9%
Cooper	GE 7 × 7	CZ357	2.500	190.19	27,140	5/21/1982	932	320.3	430.2	34.3%	370.4	15.6%
Cooper	GE 7 × 7	CZ369	2.500	190.20	26,575	5/21/1982	888	347.6	437.6	25.9%	384.2	10.5%

Table B.2. Comparison of decay heat obtain obtained using the proposed guide and current RG 3.54 Rev. 1 for BWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated RG 3.54 (W)	Percent difference (C/M–1)%	Calculated New Guide (W)	Percent difference (C/M–1)%
Cooper	GE 7 × 7	CZ370	2.500	190.23	26,342	4/20/1981	1257	293.6	345.4	17.6%	322.0	9.7%
Cooper	GE 7 × 7	CZ372	2.500	190.01	25,848	4/20/1981	1256	294.3	328.7	11.7%	305.8	3.9%
Cooper	GE 7 × 7	CZ379	2.500	190.18	25,925	5/21/1982	898	287.4	420.1	46.2%	344.1	19.7%
Cooper	GE 7 × 7	CZ398	2.500	189.83	27,478	5/21/1982	890	372.0	454.7	22.2%	401.9	8.0%
Cooper	GE 7 × 7	CZ415	2.500	189.72	25,863	4/20/1981	1255	289.3	336.6	16.3%	315.1	8.9%
Cooper	GE 7 × 7	CZ416	2.500	189.43	27,460	5/21/1982	894	319.8	449.9	40.7%	383.6	20.0%
Cooper	GE 7 × 7	CZ429	2.500	190.07	27,641	5/21/1982	889	385.6	460.6	19.5%	411.4	6.7%
Cooper	GE 7 × 7	CZ430	2.500	189.93	26,824	5/21/1982	894	353.3	439.7	24.4%	385.7	9.2%
Cooper	GE 7 × 7	CZ433	2.500	190.02	25,977	4/20/1981	1255	287.4	332.3	15.6%	306.8	6.8%
Cooper	GE 7 × 7	CZ460	2.500	190.18	26,511	5/21/1982	933	313.5	417.1	33.0%	353.6	12.8%
Cooper	GE 7 × 7	CZ466	2.500	189.86	26,077	5/21/1982	861	302.1	438.0	45.0%	357.8	18.4%
Cooper	GE 7 × 7	CZ468	2.500	189.78	26,757	5/21/1982	935	325.3	422.8	30.0%	361.1	11.0%
Cooper	GE 7 × 7	CZ472	2.500	190.12	25,957	5/21/1982	859	325.0	437.4	34.6%	380.4	17.0%
Cooper	GE 7 × 7	CZ473	2.500	189.76	26,517	5/21/1982	934	293.2	415.8	41.8%	344.0	17.3%
Cooper	GE 7 × 7	CZ498	2.500	189.69	26,482	5/21/1982	888	359.4	436.7	21.5%	384.3	6.9%
Cooper	GE 7 × 7	CZ508	2.500	190.68	26,357	5/21/1982	933	310.0	412.4	33.0%	353.6	14.1%
Cooper	GE 7 × 7	CZ515	2.500	190.48	25,737	4/20/1981	1254	294.0	334.6	13.8%	314.6	7.0%
Cooper	GE 7 × 7	CZ526	2.500	190.54	27,596	5/21/1982	864	395.4	471.4	19.2%	421.2	6.5%
Cooper	GE 7 × 7	CZ528	2.500	190.81	25,714	4/20/1981	1284	297.6	327.1	9.9%	308.4	3.6%
Cooper	GE 7 × 7	CZ531	2.500	189.90	26,699	5/21/1982	893	347.2	437.8	26.1%	384.1	10.6%
Cooper	GE 7 × 7	CZ536	2.500	190.17	26,589	4/20/1981	1256	295.2	326.6	10.6%	304.1	3.0%
Cooper	GE 7 × 7	CZ542	2.500	189.99	26,691	5/21/1982	932	311.9	420.9	34.9%	356.9	14.4%
Cooper	GE 7 × 7	CZ545	2.500	190.47	26,668	5/21/1982	935	295.2	417.4	41.4%	348.3	18.0%
Monticello	GE 7 × 7	MT116	2.250	193.53	17,482	9/12/1975	3559	114.9	119.2	3.7%	118.2	2.9%
Monticello	GE 7 × 7	MT123	2.250	193.53	14,152	3/15/1974	4100	66.8	78.4	17.4%	82.2	23.0%
Monticello	GE 7 × 7	MT133	2.250	193.53	20,189	9/12/1975	3555	129.0	142.9	10.8%	139.3	8.0%
Monticello	GE 7 × 7	MT190	2.250	193.53	15,312	9/12/1975	3557	99.2	---	---	100.6	1.4%
Monticello	GE 7 × 7	MT228	2.250	193.53	12,570	9/12/1975	3560	76.4	---	---	80.2	5.0%
Barsebäck 1	8 × 8–1	9329	2.920	178.77	41,094	9/17/1988	5373	224.4	261.4	16.5%	233.2	3.9%

Table B.2. Comparison of decay heat obtain obtained using the proposed guide and current RG 3.54 Rev. 1 for BWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated RG 3.54 (W)	Percent difference (C/M–1)%	Calculated New Guide (W)	Percent difference (C/M–1)%
Barsebäck 1	8 × 8–1	10288	2.950	179.16	35,180	9/17/1988	5534	185.8	211.2	13.7%	193.4	4.1%
Barsebäck 2	8 × 8–1	14076	3.150	179.57	40,010	7/2/1992	4177	240.3	275.6	14.7%	248.7	3.5%
Forsmark 1	8 × 8–1	3838	2.086	177.90	25,669	7/10/1992	4170	126.8	157.6	24.3%	138.3	9.1%
Forsmark 1	8 × 8–2	KU0100	2.976	174.92	34,193	8/17/1990	4893	185.3	208.2	12.4%	191.3	3.2%
Forsmark 1	9 × 9–5	KU0269	2.938	177.02	35,113	8/17/1990	4903	192.7	218.7	13.5%	199.9	3.7%
Forsmark 1	9 × 9–5	KU0278	2.939	177.13	35,323	5/24/1991	4595	195.4	224.3	14.8%	203.8	4.3%
Forsmark 1	9 × 9–5	KU0282	2.939	177.10	37,896	5/24/1991	4574	218.5	247.0	13.0%	221.7	1.5%
Forsmark 2	8 × 8–1	5535	2.095	177.69	19,944	7/15/1988	5634	84.6	106.4	25.8%	96.8	14.4%
Forsmark 2	SVEA 64	11494	2.920	181.09	32,431	7/15/1988	5618	166.0	191.0	15.1%	176.7	6.4%
Forsmark 2	SVEA 64	11495	2.910	181.07	32,431	7/15/1988	5593	167.6	191.3	14.1%	177.0	5.6%
Forsmark 2	SVEA 64	13775	2.850	181.34	32,837	7/12/1991	4543	178.4	210.2	17.8%	192.6	7.9%
Forsmark 3	SVEA 100	13847	2.769	180.67	31,275	7/14/1990	4871	169.6	193.5	13.8%	179.1	5.4%
Forsmark 3	SVEA 100	13848	2.769	180.67	31,275	7/13/1990	4882	170.7	193.3	13.3%	178.9	4.8%
Ringhals 1	8 × 8–1	1177	2.642	180.59	36,242	8/2/1985	6690	177.9	204.7	15.0%	182.1	2.3%
Ringhals 1	8 × 8–1	1186	2.640	180.52	30,498	8/2/1985	6674	140.8	166.2	18.0%	149.2	6.0%
Ringhals 1	8 × 8–1	6423	2.900	177.70	35,109	8/5/1988	5669	174.2	206.0	18.3%	186.8	7.2%
Ringhals 1	8 × 8–1	6432	2.894	177.52	36,861	8/5/1988	5422	185.5	---	---	199.3	7.6%
Ringhals 1	8 × 8–1	6454	2.898	177.68	37,236	8/15/1986	6395	186.3	213.7	14.7%	190.5	2.3%
Ringhals 1	8 × 8–1	8327	2.904	177.54	37,851	8/6/1991	4600	196.9	242.6	23.2%	216.5	10.0%
Ringhals 1	8 × 8–1	8331	2.910	177.69	35,903	9/15/1989	5291	187.0	217.4	16.2%	195.8	4.7%
Ringhals 1	8 × 8–1	8332	2.895	177.52	34,977	8/5/1988	5690	168.1	203.5	21.1%	183.4	9.1%
Ringhals 1	8 × 8–1	8338	2.911	177.60	34,830	8/5/1988	5695	169.5	203.4	20.0%	183.4	8.2%
Oskarshamn 2	8 × 8–1	1377	2.201	183.58	14,546	5/13/1977	9750	56.2	64.6	15.0%	61.3	9.0%
Oskarshamn 2	8 × 8–1	1389	2.201	183.65	19,481	7/15/1981	8171	83.9	95.0	13.2%	87.0	3.7%
Oskarshamn 2	8 × 8–1	1546	2.201	183.97	24,470	8/19/1983	7455	108.1	126.4	16.9%	114.6	6.1%
Oskarshamn 2	8 × 8–1	1696	2.201	184.25	20,870	8/19/1983	7411	92.4	108.1	17.0%	96.8	4.7%
Oskarshamn 2	8 × 8–1	1704	2.201	184.02	19,437	7/23/1982	7808	84.0	96.4	14.7%	86.6	3.1%
Oskarshamn 2	8 × 8–1	2995	2.699	179.38	29,978	7/15/1981	8211	130.5	150.2	15.1%	135.6	3.9%

Table B.2. Comparison of decay heat obtain obtained using the proposed guide and current RG 3.54 Rev. 1 for BWR fuel assemblies (continued)

Reactor	Assembly type	Assembly ID	Enrichment (wt % ^{235}U)	Uranium mass (kg)	Final burnup (MWd/t)	Discharge date	Cooling time (days)	Measured (W)	Calculated RG 3.54 (W)	Percent difference (C/M–1)%	Calculated New Guide (W)	Percent difference (C/M–1)%
Oskarshamn 2	8 × 8–1	6350	2.875	179.00	27,675	6/7/1985	6755	129.4	147.2	13.8%	133.5	3.2%
Oskarshamn 2	SVEA 64	12684	2.902	182.32	46,648	8/2/1991	4519	282.7	---	---	298.1	5.5%
Oskarshamn 3	8 × 8–1	12078	2.577	177.36	25,160	7/8/1988	5611	120.2	140.1	16.6%	132.4	10.1%
Oskarshamn 3	SVEA 100	13628	2.711	180.77	35,619	6/24/1991	4581	194.0	235.4	21.4%	212.1	9.3%
Oskarshamn 3	SVEA 100	13630	2.711	180.78	40,363	6/24/1991	4554	235.7	279.1	18.4%	247.8	5.1%

--- Assembly properties outside the range of current regulatory guide RG 3.54, Rev. 1.

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10. SUPPLEMENTARY NOTES

M. Aissa, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report describes the technical basis for expanding the range of the current U.S. Nuclear Regulatory Commission regulatory guide for calculating decay heat power in an independent spent fuel storage installation (Regulatory Guide 3.54, Rev. 1) to include current high burnup fuel. As part of the expansion of the guide, a revised methodology is proposed to improve flexibility, enable increased accuracy, and cover a wider range of reactor operating histories. The methods are based more directly on the physics of decay heat generation and implement, as part of the proposed guide, procedures and data from consensus standards developed for calculating decay heat by the American National Standards Institute and the International Standards Organization, ANSI/ANS 5.1-2005 and ISO 10645:1992(E), respectively. The proposed guide is validated using experimental calorimeter decay heat data for 68 spent fuel assemblies measured in the United States and for 64 assemblies measured more recently in Sweden at the Interim Storage Facility for Spent Nuclear Fuel. Validation of the methods beyond the range of the decay heat measurements is supported using experimental isotopic assay measurements of the dominant decay heat-generating isotopes, obtained from destructive radiochemical analysis of spent fuel samples.

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