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DRAFT REGULATORY GUIDE

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(Proposed Revision 2 of Regulatory Guide 3.54, dated January 1999)

SPENT FUEL HEAT GENERATION IN AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION

A. INTRODUCTION

Purpose

This regulatory guide (RG) provides methods that are acceptable to the Nuclear Regulatory Commission (NRC) staff for calculating spent nuclear fuel heat generation rates for use as design input for an independent spent fuel storage installation (ISFSI).

Applicability

This RG applies to all applicants and licensees that are subject to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste" (Ref. 1).

Applicable Regulations

- 10 CFR 72.122(h)(1), "Overall requirements," requires that spent fuel cladding be protected during storage against degradation that can lead to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage."
- 10 CFR 72.128(a)(4) "Criteria for spent fuel, high-level radioactive waste, reactor-related greater than class C waste, and other radiative waste storage and handling," requires that the spent fuel storage system be designed with a heat removal capability consistent with its importance to safety."

- 10 CFR 72.236(a)(f) states that the certificate holder and the applicant for a CoC shall ensure that the requirements of this section are met. 10 CFR 72.236(a) states that specification must be provided for the spent fuel to be stored in the spent fuel storage cask, such as, but not limited to, type of spent fuel, maximum allowable enrichment of the fuel prior to any irradiation, burnup, minimum acceptable cooling time of the spent fuel prior to storage in the spent fuel cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel, the inerting atmosphere requirements. 10 CFR 72.236(f) states that the spent fuel storage cask must be designed to provide adequate heat removal capacity without active cooling system.

Related Guidance

NRC, NUREG/CR 6999, "Technical Basis for the Proposed Expansion of the Regulatory Guide 3.54" (Ref. 2), describes the technical basis for expanding the range of the current NRC regulatory guide for calculating decay heat power in an independent spent fuel storage installation 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.

- NRC, NUREG/CR-5625, "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data" (Ref. 3), provides general guidance to licensees and applicants for ISFSIs on how to perform decay heat calculations using SAS2H/ORIGEN-S data.

- NRC, NUREG/CR-6971, "Spent Fuel Decay Heat Measurements Performed at the Swedish Central Interim Storage Facility" (Ref. 4), provides guidance for the review and evaluation of spent fuel decay heat measurements for licensees and applicants for ISFSIs.

- NRC, NUREG/CR-6972, "Validation of SCALE 5 Decay Heat Predictions for LWR Spent Nuclear Fuel" (Ref. 5), provides general guidance to validate decay heat predictions using the SCALE 5 depletion code for licensees and applicants for ISFSIs.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG contains and references information collections covered by 10 CFR Part 72 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), control number 3150-0132.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

This revision (Revision 2) presents an up-to-date methodology for determining heat generation rates for both pressurized-water reactor (PWR) and boiling water reactor (BWR) fuel and provides greater flexibility (fewer restrictions) than previous versions. It allows the loading of higher burnup fuel by using more accurate methods for decay heat calculations by covering a wider range of fuel characteristics, including operating history.

Background

The NRC issued Revision 1 in January 1999 to provide a standard methodology for calculating decay heat generation rates in light-water reactor (LWR) spent fuel.

Revision 1 provided calculations using specific fuel enrichment-burnup values that were available during the period that the RG was written. In addition, in Revision 1, the decay heat generation rates for a fuel assembly were determined with accuracies comparable to those of detailed isotopic depletion codes because they did not require complex calculations. Also, Revision 1 was restricted to boiling water reactor (BWR) fuel with a maximum burnup of 45 gigawatt days per metric ton of initial uranium (GWd/MTU), and pressurized water reactor (PWR) fuel with a maximum burnup of 50 GWd/MTU.

Revision 2 uses procedures and data from consensus standards developed for calculating decay heat by the American National Standards Institute/American Nuclear Society (ANSI/ANS-5.1-2005), "Decay Heat Power in Light Water Reactors" (Ref. 6), and the International Organization for Standardization (ISO) 10645:1992(en), "Nuclear Energy—Light Water Reactors—Calculation of the Decay Heat Power in Nuclear Fuels" (Ref. 7).

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 the temperatures within the fuel, cladding, shielding, structural, confinement, and criticality safety components will not exceed design and licensing specifications, potentially presenting operational safety problems.

The temperature is a function of the heat generated by the stored fuel assemblies. The spent fuel storage system is designed to protect its cladding during storage against degradation that leads to gross ruptures, or otherwise confine the fuel such that degradation of the fuel during storage will not pose operational safety problems if removed from storage.

The methodology used in this guide is appropriate for computing the heat generation rates of fuel assemblies from LWRs as a function of burnup, specific power, decay time, and enrichment up to 5 weight-percent uranium-235 (wt% U-235) (see NUREG/CR-6999, "Technical Basis for a Proposed Expansion of Regulatory Guide 3.54—Decay Heat Generation in an Independent Spent Fuel Storage Installation"). The computed heat generation results are used in the next section in a procedure for determining heat generation rates for PWR and BWR assemblies.

Calculations of decay heat have been verified by comparison with the existing database in NUREG/CR-6999 of experimentally measured decay heat rates for PWR and BWR spent fuel. The range

of parameter values in the procedure is considered to encompass the typical burnup, specific power, enrichment, and cooling time values.

Calculations of decay heat involve calculating individual contributions to decay heat from each of the following categories:

- fission products produced by the fission of U-235, uranium-238 (U-238), plutonium-239 (Pu-239), and plutonium-241 (Pu-241)
- nuclides that are produced by neutron capture by fission products
- actinides generated by neutron capture by initial uranium isotopes in the fuel
- activation products in the assembly structure and cladding materials

A detailed example calculation of decay heat power is shown in Appendix A. Appendix B includes terms and units that have been used in this guide.

Harmonization with International Standards

The NRC has a goal of harmonizing its guidance with international standards. The NRC staff reviewed guidance from the International Atomic Energy Agency (IAEA), and did not identify any standards that provided useful guidance to NRC staff, applicants, or licensees regarding calculations of the decay heat power in nuclear fuels. The German Institute for Standardization (DIN), the Japan Atomic Energy Research Institute (JAERI), and the International Organization for Standardization (ISO) have established a series of safety guides and standards for calculation of the decay power in nuclear fuels of LWRs. This guidance provides international good practices and, by providing calculations of the decay heat power in nuclear fuels, increasingly reflects best practices to help users striving to achieve high levels of safety. This RG incorporates similar guidelines and is consistent with the basic safety principles provided in the following documents:

- DIN, "Decay Heat Power in Nuclear Fuels of Light Water Reactors—Non-Recycled Nuclear Fuels," DIN 25463-1, Deutsches Institut für Normung, 1990 (Ref. 8), provides calculations of the decay heat power in nuclear fuels.
- JAERI, "Recommended Values of Decay Heat Power and Method to Utilize the Data," Appendix B, 1991 (Ref. 9), provides calculations of the decay heat power in nuclear fuels.
- ISO 10645:1992(en), "Nuclear Energy—Light Water Reactors—Calculation of the Decay Heat Power in Nuclear Fuels," provides calculations of the decay heat power in nuclear fuels.

C. STAFF REGULATORY GUIDANCE

The following method for determining heat generation rates of reactor spent fuel assemblies is considered acceptable for use to the NRC. The methods used in this guide define procedures and data for all constants and variables required to calculate decay heat. Limitations of the methods pertaining to each component of decay heat are also addressed. The following sections contain the methods and procedures for the calculation of the decay heat power. Appendix A presents an example of a decay heat calculation. Appendix B contains the terms and units used in this guide.

1. CALCULATION OF DECAY HEAT POWER

This section describes the method 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. The total decay heat generation rate without the safety factor is calculated as:

$$P_T(t,T) = P_F(t,T) + P_C(t,T) + P_E(t,T) + P_A(t,T) + P_S(t,T) \qquad \text{Equation (1)}$$

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

$$P'_T(t, T) = P_T(t, T)\, F_S(t) \qquad \text{Equation (2)}$$

1.1. Fission Products

Fission product decay heat (excluding neutron capture) should be calculated using methods and data developed in the American National Standard for decay heat, ANSI/ANS-5.1-2005. These procedures apply to the calculation of decay heat for irradiated PWR and BWR fuel assemblies.

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: U-235, Pu-239, U-238, and Pu-241. These four actinides account for more than 99 percent of the fission in typical LWR fuel. Fission of other isotopes is considered by treating them as U-235, 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:

$$f_i(t) = \sum_{j=1}^{23} \alpha_{ij} e^{-\lambda_{ij} t} \qquad \text{(MeV/s)} \qquad \text{Equation (3)}$$

where t is the time after fission and the coefficients αij and λij 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:

$$F_i(t,T) = \sum_{j=1}^{23} \frac{\alpha_{ij}}{\lambda_{ij}} (1 - e^{-\lambda_{ij} T})\, e^{-\lambda_{ij} t} \qquad \text{Equation (4)}$$

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:

$$P_F(t,T) = \sum_{i=1}^{4}\sum_{k=1}^{m}\left[\frac{S_{ik}}{Q_i}\sum_{j=1}^{23}\frac{\alpha_{ij}}{\lambda_{ij}}(1-e^{-\lambda_{ij}T_k})e^{-\lambda_{ij}t_k}\right] \quad \text{Equation (5)}$$

where the indices $i = 1, 2, 3,$ and 4 represent U-235 thermal fission, Pu-239 thermal fission, U-238 fast fission, and Pu-241 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 *mega electron volt per fission* (*MeV/fission*) is the recoverable energy per fission (energy generating heat in the system, see Table 2). 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 this 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 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 operation 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 W/kgU, is determined from the accumulated assembly burnup for the cycle divided by the irradiation time of the cycle:

$$S_k = \frac{8.64\times10^{10}\ (B_k - B_{k-1})}{T_k} \quad \text{Equation (6)}$$

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 megawatts (MW) to watts (W). The average specific power over the entire operating history of the fuel assembly is defined as:

$$S_{avg} = \frac{8.64\times10^{10}\ B_{tot}}{T} \quad \text{Equation (7)}$$

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 1 for each fissionable nuclide. Recommended values for Q_i for each fissionable nuclide are listed in Table 2 (see energy per fission values). The relative power fractions for each of the four fissionable nuclides (S_i/S) are given in Table 3 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 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.

The user should ensure that the fuel burnup obtained from the time-integrated specific power of the histogram in Figure 1 equals the actual burnup of the fuel. The user should also ensure that the sum of relative power fractions S_i/S for each irradiation interval is not less than unity (because of interpolation error). In this case, the user should increase the power fraction of U-235 to preserve the correct total operating power for the interval.

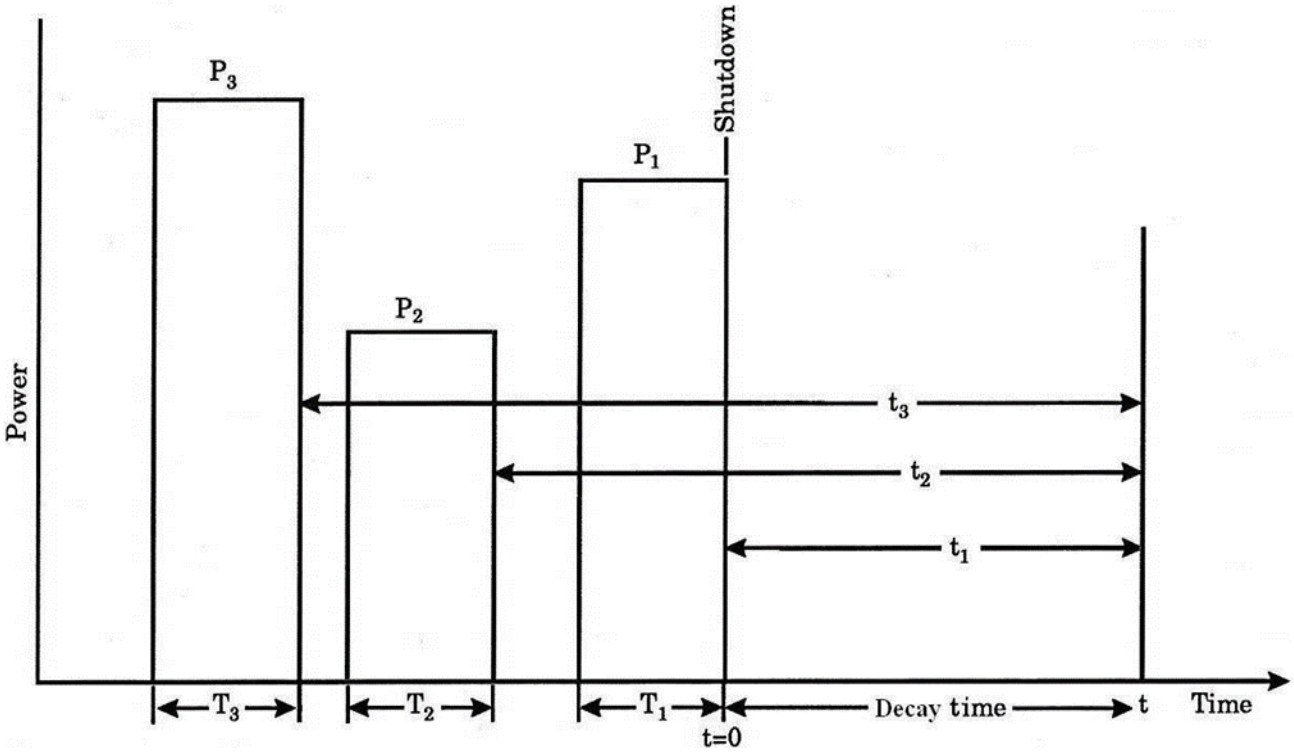

Figure 1. Example of a typical three-cycle reactor operating history

Table 1. Coefficients [a] for Thermal Fission of U-235, Pu-239, Pu-241, and Fast Fission of U-238

Term index j	U-235 (thermal [b])		Pu-239 (thermal)		U-238 (fast)		Pu-241 (thermal)	
	α_{1j}	λ_{1j}	α_{2j}	λ_{2j}	α_{3j}	λ_{3j}	α_{4j}	λ_{4j}
1	[c] 5.2800E-04	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 2. Recommended Fission Energy Values (Q) for Application to Decay Heat Analysis

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

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

Table 3. Power Fractions for Fission of U-235, Pu-239, U-238, and Pu-241

Burnup	2 wt% U-235				3 wt% U-235			
(MWd/kgU)	U-235	Pu-239	U-238	Pu-241	U-235	Pu-239	U-238	Pu-241
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	**4 wt% U-235**				**5 wt% U-235**			
(MWd/kgU)	**U-235**	**Pu-239**	**U-238**	**Pu-241**	**U-235**	**Pu-239**	**U-235**	**Pu-241**
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

1.2. Neutron Capture by Fission Products

The calculation of decay heat power from fission products described in Section 1.1 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 or longer-lived fission products and daughters resulting in an increased concentration of unstable nuclides. Because neutron absorption by fission products leads to production of nuclides farther from the line of stability, the net effect of absorption is an increase in decay heat generated.

At cooling times, in the region of 10^8 seconds (about 3 years), production of cesium-134 (Cs-134) through neutron capture by the stable fission product cesium-133 (Cs-133) can represent a significant contribution to decay heat. The only significant production route to Cs-134 is through neutron capture by Cs-133. The isotope Cs-134 is not produced by direct fission because the decay mass chain ends with stable xenon-134. Because Cs-134 is the dominant decay heat-generating nuclide resulting from neutron

capture, it is treated explicitly in this guide using the methodology developed in the ISO 10645:1992 standard. The contributions from other neutron capture products are conservatively treated as an aggregate in this guide using a bounding correction factor.

1.2.1. Contribution of Cs-134

Neutron capture by the stable fission product Cs-133 produces Cs-134, which has a half-life of 2.06 years. The concentration of stable Cs-133 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 Cs-134 resulting from neutron capture on Cs-133, in units of *W/kgU*, is given by the equation:

$$P_c(t,T) = Y\ E\ \lambda_4\ \frac{S}{Q}\left[\frac{1-e^{-(\lambda_4+\sigma_4\phi)T}}{\lambda_4+\sigma_4\phi}+\frac{e^{-\sigma_3\phi T}-e^{-(\lambda_4+\sigma_4\phi)T}}{\sigma_3\phi-(\lambda_4+\sigma_4\phi)}\right]e^{-\lambda_4 t} \qquad \text{Equation (8)}$$

where the quantity $\frac{S}{Q}=\sum_{i=1}^{4}\frac{S_i}{Q_i}$, t is the time after discharge in seconds, T is the irradiation interval time in seconds, and ϕ is the total neutron flux *(n/cm²/s)*. The other parameter constants used in Equation (6) are defined as follows.

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

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 10645:1992, but are generated from ENDF/B-VII data evaluations and yield improved estimates of Cs-134 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:

$$\phi = \frac{1}{T}\sum_{k=1}^{m}\phi_k T_k \qquad \text{Equation (9)}$$

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

$$\frac{S}{Q} = \frac{1}{T}\sum_{k=1}^{m}\sum_{i=1}^{4}\frac{S_{ik}}{Q_i}T_k \qquad \text{Equation (10)}$$

The neutron flux in the fuel is approximated using the relationship:

$$\phi_k = \frac{S_{avg}}{\alpha}\ 2.58\ x\ 10^{10} \qquad \text{Equation (11)}$$

where S_{avg} is the specific power density given in Equation (7), 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 U-235 wt% 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 Cs-134 contribution to decay heat that exceed the exact values by up to 5 percent. For lower burnup values, less than 25 MWd/kgU, the expression will overestimate the Cs-134 contribution by up to 15 percent.

1.2.2. Contribution of Other Neutron Capture Nuclides

The contribution from neutron capture on fission products, excluding Cs-133, is determined using the tabulated factors of *H(t)* given in Table 4 as a function of cooling time. The factors are multiplied by the decay heat power because of the direct fission products, $P_F(t,T)$, evaluated in Section 1.1 using Equation (5), according to the equation:

$$P_E(t, T) = H(t)\ P_F(t,T) \qquad \text{Equation (12)}$$

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% U-235.
- Burnup, in units of MWd/kgU, is less than 14 times the initial enrichment in wt% U-235.
- The power density, in units of kilowatts per kilogram of uranium, is less than 5 times the initial enrichment in wt% U-235.

This parameter range is adequate to cover most spent fuel assemblies discharged from commercial reactors operating in the United States (see discussion in Section 2).

Table 4. Correction Factors for Fission Product Neutron Capture (Excluding Cs-133), 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

1.3. **Actinides**

The decay heat from actinides is calculated as the sum of contributions from americium-241 (Am-241), curium-242 (Cm-242), Cm-244, Pu-238, Pu-239, Pu-240, and Pu-241. These seven actinides contribute more than 99.5 percent 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:

$$P_A^{'}(t) = \sum_{n=1}^{7} \hat{\beta}_n e^{-\lambda_n t} \qquad \text{Equation (13)}$$

where the index *n* corresponds to the actinides Am-241, Pu-241, Pu-240, Pu-239, Pu-238, Cm-244, and Cm-242, lambda sub n (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:

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

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

$$\hat{\beta}_n = \beta_n \qquad n = 3, \ldots\ldots, 7 \qquad \text{Equation (14)}$$

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 Am-241 ($5.629\ MeV$) and E_2 is the thermal energy released per decay for Pu-241 ($5.361 \times 10^{-3}\ MeV$).

The actinide coefficients β_n for PWR fuel are listed in Table 5 for tabulated values of initial U-235 enrichment, E_S, and final assembly burnup, B_{tot}. Coefficients for BWR fuel are given in Table 6. 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 βn 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 Equation (14), account for the decay of Am-241 that is produced from the decay of its parent Pu-241 (half-life of 14.4 years) after discharge. The energy released per decay of Am-241 is 103 times greater than its parent Pu-241, and the decay heat generated by in-growth of Am-241 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 of NUREG/CR-6999).

The average specific operating power used to generate the coefficients in Tables 5 and 6 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 Equation (13) to account for variations in operating specific power over the range of 12 to 50 kW/kgU using the equation

$$P_A(t) = P'_A(t) \times 1.82\left[S_{avg}\right]^{-0.06} \quad \text{Equation (15)}$$

Note that the units of specific power applied in Equation (15) 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 percent. However, in this cooling time range, actinides typically contribute less than 20 percent of the total decay heat power and the method does not result in undue conservatism in the total decay heat generation rate.

Table 5. Parameters and Coefficients for Calculating Actinide Decay Heat for PWR Fuel

Index *n*		**1**	**2**	**3**	**4**	**5**	**6**	**7**
Nuclide		**Am-241**	**Pu-241**	**Pu-240**	**Pu-239**	**Pu-238**	**Cm-244**	**Cm-242**
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 6. Parameters and Coefficients for Calculating Actinide Decay Heat for BWR Fuel

Index *n*		**1**	**2**	**3**	**4**	**5**	**6**	**7**
Nuclide		**Am-241**	**Pu-241**	**Pu-240**	**Pu-239**	**Pu-238**	**Cm-244**	**Cm-242**
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

1.4 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 (Zr-2), and Zr-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 and may contribute up to several percent of the total decay heat (see Figure 2.1 in NUREG/CR-6999). The decay heat contribution is determined from the formula:

$$P_S(t,T) = A(t)P_F((t,T) \qquad \text{Equation (16)}$$

where the values for $A(t)$ are listed in Table 4 and $P_F(t,T)$ is the direct fission product decay heat evaluated in Section 1.1 of this RG. 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 MWd/kgU does not exceed 14 times the initial enrichment.

1.5 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 4 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 Section 5 of NUREG/CR-6999.

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 percent of the measurement data at a 95 percent confidence level. Potential nonconservatism, because of 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 Section 4 of NUREG/CR-6999.

1.6 Final Decay Heat Power

As shown in Equation (1), the total decay heat generation rate without the safety factor is calculated as:

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

Where:

$P_F(t,T)$ is the fission product decay heat power (excluding neutron capture) from Section 1.1,
$P_C(t,T)$ is the decay heat power from neutron capture to product Cs-134 from Section 1.2.1,
$P_E(t,T)$ is the decay heat power from neutron capture on other fission products from Section 1.2.2,
$P_A(t,T)$ is the decay heat power from actinides from Section 1.3,
$P_S(t,T)$ is the decay heat power from activated structural materials from Section 1.4.

Per Equation (2), the final decay heat generation rate with the safety factor $F_S(t)$ included is determined as:

$$P'_T(t,T) = P_T(t,T)F_S(t)$$

2. LIMITS AND RANGE OF APPLICABILITY

This RG 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 this guide, the range of applicability has been extended to most of the existing spent fuel generated in the United States. The spent fuel characteristics applicable in this guide are summarized in Table 7. The cooling time T_C, is listed in units of years.

The range of application of the guide is illustrated in Figure 2 and compared with the existing and projected commercial spent fuel inventory of the United States, as published in U.S. Department of Energy (DOE) Form GC-859, "NUCLEAR FUEL DATA SURVEY FORM GC-859" at https://www.eia.gov/survey/form/gc_859/form.pdf.

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 Cs-134 (see section C.1.2 to this guide) and structural activation, which are based on a single conservative factor and generally bound irradiation history. The data developed for these components of decay heat limit the burnup, in MWd/kgU, to a maximum of 14 times the initial fuel enrichment, in units of wt% U-235. 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 7 regardless of the initial enrichment. The restrictions on applying the guide beyond the enrichment and burnup range of Table 7 are based mostly on the limited availability of experimental data for methods validation.

The procedures in this 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 (see section C.1.2 of this guide) 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 percent. Both variables are conservative with respect to typical reactor operations (the power range goes from 2 to 100 MWth). 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 7.

A detailed assessment found that beyond approximately one month cooling time, actinide decay heat power decreases as the specific operating power increases. 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 under predicted. At longer cooling times when the actinides become an increasingly important source of decay heat, such an under prediction could lead to nonconservative errors in the total decay heat power prediction. An assembly parameter that may restrict application of the guide is the cobalt-59 (Co-59) content of the clad and structural materials. Co-59 is partly transformed to Co-60 during irradiation and subsequently contributes to the decay heat rate. The Co-59 content used in deriving the activation tables here should be applied only to assemblies containing Zircaloy-clad fuel rods. The Co-59 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 Co-60 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 percent heat rate change per 1-percent 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 percent 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.

Table 7. Parameter Range for Applicability of the Regulatory Guide

Parameter	BWR	PWR
E_S (wt% U-235)	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

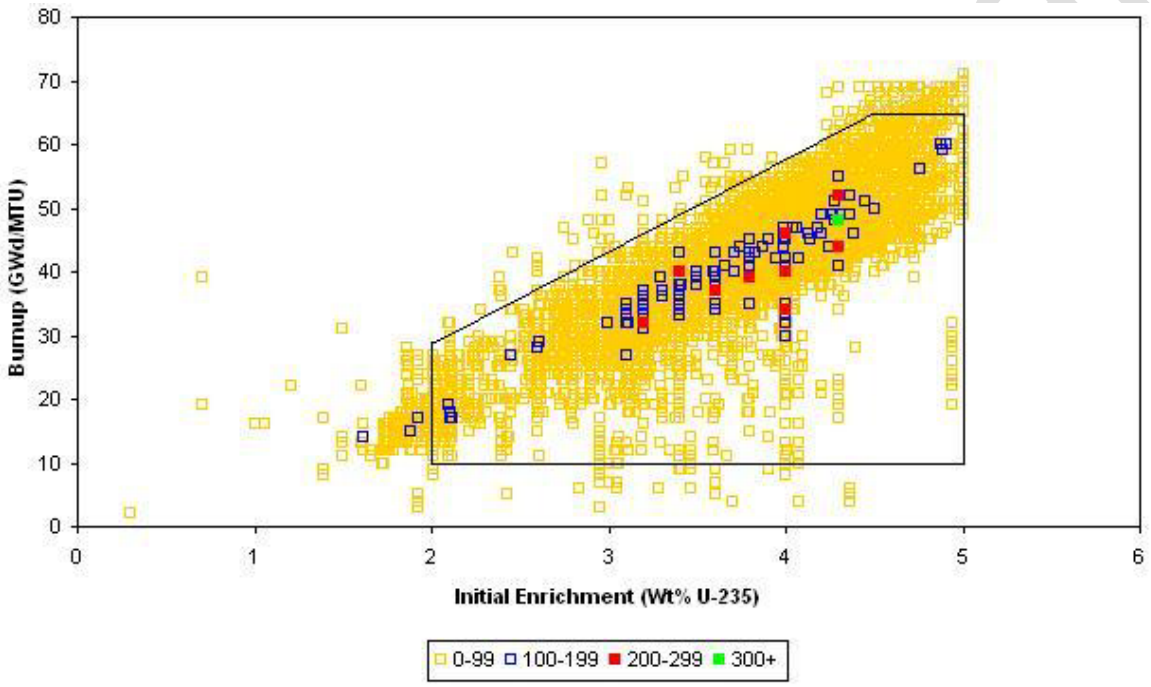

Figure 2. Nominal range of application of Regulatory Guide 3.54

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.

3. CALCULATIONS BY COMPUTER CODES

Data developed for this guide that is not adopted from decay heat standards is derived in large measure from calculations performed using the ORIGEN-S isotope generation and depletion code. 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 of NUREG/CR-6999 and against isotopic assay measurements for the key decay heat generating isotopes for the experiments summarized in Table 2.2 of NUREG/CR-6999.

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. The measured fuel samples included in this study and the experimental programs under which they were acquired are summarized in Table 2.2 of NUREG/CR-6999. 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 percent, and is somewhat larger (10-20 percent) for the curium isotopes. The PWR isotopic measurements include fuel samples with enrichments up to 4.6 wt% U-235 and burnup values up to 70 GWd/MTU. BWR isotopic validation is less extensive 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 this RG 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.

The ORIGEN-S calculations used to support the development of this guide were performed using standard LWR assembly cross-section libraries released with the Standardized Computer Analyses for Licensing Evaluation (SCALE) code system Version 5.1 (Ref. 10). 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 assembly designs such as as General Electric (GE) 7 × 7, GE 8 × 8, ASEA Brown Boveri 8 × 8, GE 9 × 9, ATRIUM 9, GE 10 × 10, ATRIUM 10, SVEA 64, SVEA 100, Combustion Engineering (CE) 14 × 14, Westinghouse (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 automatic rapid processing (ARP) 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.

The TRITON/NEWT depletion sequence, also included in the SCALE code system, is another computational tool available to determine burned fuel decay heat. This sequence couples the two-dimensional arbitrary polygonal mesh transport code NEWT with the point depletion and decay code ORIGEN to perform fuel burnup simulations. At each depletion step, the neutron transport flux solution from NEWT is used to generate cross sections for the ORIGEN calculation; the isotopic composition data resulting from each isotopic depletion step is employed in the subsequent transport calculation to obtain updated cross sections for the next depletion step in an iterative manner throughout the irradiation history. TRITON can simulate the depletion of multiple mixtures and regions in a fuel assembly model, which is useful for determining decay heat for fuel assemblies with axially and radially varying fuel enrichment, neutron absorber content, and structural features.

The most recent version of SCALE (SCALE 6.2.1) (Ref. 11), was released in August 2016 from the Radiation Safety Information Computational Center (RSICC).

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[1] may use this guide and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with the backfit rule in 10 CFR Part 72.62, "Backfitting."

Use by Applicants and Licensees

Applicants and licensees may voluntarily[2] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged**.**

Licensees may use the information in this RG for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 72.48, "Changes, Tests, and Experiments," Licensees may also use the information in this RG or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition of the guidance in this RG that would constitute backfitting. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action that would require the use of this RG without further backfit consideration. Examples of such NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 12) and NUREG-1409, "Backfitting Guidelines" (Ref. 13).

[1] In this section, "licensees" refers to licensees of ISFSIs under 10 CFR Part 72.

[2] In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

REFERENCES [3]

1. CFR, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," Part 72, Chapter I, Title 10, "Energy."

2. NRC, "Technical Basis for a Proposed Expansion of Regulatory Guide 3.54—Decay Heat Generation in an Independent Spent Fuel Storage Installation," NUREG/CR-6999, February 2010, ADAMS Accession No. ML100850213.

3. U.S. Nuclear Regulatory Commission (NRC), "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN S Data," NUREG/CR 5625, September 1994.

4. "Spent Fuel Decay Heat Measurements Performed at the Swedish Central Interim Storage Facility," NUREG/CR-6971, February 2010, Agencywide Documents Access and Management System (ADAMS) Accession No. ML100850212.

5. "Validation of SCALE 5 Decay Heat Predictions for LWR Spent Nuclear Fuel," NUREG/CR 6972, February 2010, ADAMS Accession No. ML100900229.

6. American National Standards Institute/American Nuclear Society, ANSI/ANS 5.1 2005, "Decay Heat Power in Light Water Reactors: An American National Standard," ANS, La Grange Park, IL. [4]

7. International Organization for Standardization, 10645:1992(en), "Nuclear Energy—Light Water Reactors—Calculation of the Decay Heat Power in Nuclear Fuels." [5]

8. German Institute for Standardization, DIN 25463 1, "Decay Heat Power in Nuclear Fuels of Light Water Reactors—Non Recycled Nuclear Fuels," Deutsches Institut für Normung. [6]

9. Japan Atomic Energy Research Institute, "Recommended Values of Decay Heat Power and Method To Utilize the Data," JAERI M 91 034, Appendix B, "Physical Constants and Conversion Factors," Ibaraki ken, Japan, March 1991. [7]

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[3] Publicly available NRC-published documents are available online through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone 301-415-4737 or 800-397-4209; fax 301-415-3548; and e-mail pdr.resource@nrc.gov.

[4] Copies of American National Standards Institute (ANSI) standards may be purchased from ANSI, 1819 L Street, NW, Washington, DC 20036, on its Web site at http://webstore.ansi.org/; telephone 202-293-8020; fax 202-293-9287; or e-mail storemanager@ansi.org.

[5] Copies of International Organization for Standardization (ISO) documents may be obtained by writing to the International Organization for Standardization, 1, ch. de la Voie-Creuse, CP 56, CH-1211 Geneva 20, Switzerland, telephone +41 22 749 01 11; fax +41 22 749 09 47; or e-mail sales@iso.org, or online at the ISO Store Web site: http://www.iso.org/iso/store.htm.

[6] This standard is available through the German National Standards organization at: DIN Deutsches Institut für Normung e.V., Am DIN-Platz, Burggrafenstrasse 6, DE-10787 Berlin, Germany, Tel: +49 30 26 01-0, Fax: +49 30 26 01 12 31, e-mail: directorate.international@din.de, Web: www.din.de, Webstore: www.beuth.de/.

[7] Available at: https://www-nds.iaea.org/publications/indc/indc-jpn-0149.pdf.

10. Oak Ridge National Laboratory (ORNL), "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," ORNL/TM-2005/39, Version 5.1, Vols. I-III, 2006. [8]

11. ORNL "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," ORNL/TM-2005/39, Version 6.2.1, 2011. [9]

12. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," October 9, 2013, ADAMS Accession No. ML12059A460.

13. NRC, "Backfitting Guidelines," NUREG-1409, July 1990, ADAMS Accession No. ML032230247.

[8] Available thru the Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785, Oak Ridge National Laboratory, Oak Ridge, TN. Radiation Safety Information Computational Center, P.O. Box 2008, 1 Bethel Valley Road, Oak Ridge, TN 37831-6003, Telephone: (865)574-6176, FAX: (865)241-4046, Electronic Mail: pdc@ornl.gov, Website: https://rsicc.ornl.gov/.

[9] Available thru the Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785, Oak Ridge National Laboratory, Oak Ridge, TN. Radiation Safety Information Computational Center, P.O. Box 2008, 1 Bethel Valley Road, Oak Ridge, TN 37831-6003, Telephone: (865)574-6176, FAX: (865)241-4046, Electronic Mail: pdc@ornl.gov, Website: https://rsicc.ornl.gov/.

APPENDIX A
EXAMPLE CALCULATION

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

A.1 Assembly C-64

The assembly selected in the example, designated C-64, is a Westinghouse (W) 14 × 14 design assembly irradiated for three consecutive cycles in a pressurized-water reactor (PWR). The assembly has an initial enrichment of 3.397 wt% U-235 and was discharged March 3, 1977, after achieving an average burnup of 39,384 megawatt-days per kilogram of uranium per metric ton of initial uranium (MWd/MTU). Decay heat measurements were performed at 1,633 days or about 4.5 years after discharge from the reactor. The uranium mass of the assembly is 386.63 kilogram (kg). The operating history and the accumulated burnup in each cycle are given in Table A.1. The average specific power for each cycle is derived from the accumulated assembly burnup and cycle time listed in the table according to Equation (6). 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 A.1 Irradiation Data for 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

$$P_T(t,T) = P_F(t,T) + P_C(t,T) + P_E(t,T) + P_A(t,T) + P_S(t,T) \qquad \text{(Equation A.1)}$$

A.1.1. Calculation of P_F

The fission product decay heat, $P_F(t,T)$, uncorrected for neutron capture, is calculated according to Equation (5). The fraction of the total specific operating power associated with the fission of U-235, Pu-239, U-238, and Pu-241 is determined by interpolating the data in Table 3 using the assembly enrichment of 3.397 wt% U-235 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 A.2. The fission product decay heat power is determined from the sum of the contributions of each fissionable isotope and irradiation cycle according to Equation (5) using the specific operating power for each isotope listed in Table A.2 and the Q values listed in Table 2. The irradiation time and decay time (T_k and t_k) for each cycle, in days, are listed in Table A.3.

Table A.2 Power Fractions and Specific Power during Operating Cycles

Cycle	Mid-cycle burnup (MWd/kgU)	U-235		Pu-239		U-238		Pu-241	
		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 A.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 (W/kgU), is listed in Table A.4 for each fissioning nuclide and each irradiation time step. The sum over all nuclides and cycles yields a total contribution because of fission product decay heat, uncorrected for neutron capture, of 1.505 W/kgU.

Table A.4 Results of Fission Product Decay Heat Power

Cycle	P_F (W/kgU)				
	U-235	Pu-239	U-238	Pu-241	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

A.1.2. Calculation of P_C

The calculation of the neutron capture correction for cesium-134 (Cs-134), $P_C(t,T)$, is calculated according to the procedures of Section 1.2.1. 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 Equation (11). The average power density of the fuel is determined from the final assembly burnup, 39.384 megawatt-days per kilogram of uranium (MWd/kgU), and the total irradiation time of 1,578 days, according to Equation (7) yielding a value of 24,958 W/kgU. The effective enrichment, α, is calculated to be 2.6985. These values, applied in Equation (11) 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 Cs-134. The average value of S/Q, used in Equation (8), is determined according to Equation (10) using

the specific power during operation generated by each nuclide (S_i) listed in Table A.2 and the recommended Q values for each isotope listed in Table 2 (units of million electron volts (MeV)). This value, applied in Equation (11), yields a calculated decay heat power for Cs-134 of 0.4365 W/kgU.

A.1.3. Calculation of P_E

The contribution because of neutron capture by other fission products is calculated by linear interpolation of the tabulated values in Table 4 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 Equation (12), yielding a contribution, $P_E(t,T)$, because of neutron capture effects of 0.0523 W/kgU.

A.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 4 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 Equation (13) using actinide coefficients determined from Equation (14). The interpolated values of β_n and derived values of $\hat{\beta}_n$ are listed in Table A-5.

Table A.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	Am-241	5.078×10^{-11}	1.023×10^{-2}	1.848×10^{-2}	1.834×10^{-1}
2	Pu-241	1.531×10^{-9}	4.845×10^{-3}	-1.697×10^{-1}	-1.367×10^{-1}
3	Pu-240	3.347×10^{-12}	1.841×10^{-2}	1.841×10^{-2}	1.840×10^{-2}
4	Pu-239	9.111×10^{-13}	1.100×10^{-2}	1.100×10^{-2}	1.100×10^{-2}
5	Pu-238	2.504×10^{-10}	1.619×10^{-1}	1.619×10^{-1}	1.563×10^{-1}
6	Cm-244	1.213×10^{-9}	1.676×10^{-1}	1.676×10^{-1}	1.412×10^{-1}
7	Cm-242	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.

A.1.5. Calculation of P_S

The contribution from structural activation products, $P_S(t,T)$, is calculated by interpolating the data in Table 4 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.

A.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. Total decay heat predicted by this guide, without the safety factor, exceeds the measured value by 28.6 W, or about 3 percent. The experimental uncertainty of the measurements is about 5 percent. Revision 1 of this guide predicted a value of 950.8 W with no safety factor included, 2.1 percent larger than measured. With the recommended safety factor in the current guide of 7.24 percent included, Revision 1 of this guide over predicted the assembly decay heat by about 10 percent. Revision 2 of this guide applies a smaller safety factor of 2 percent based on the direct evaluation of the methods against calorimeter measurements. The final decay heat generation rate calculated using this guide with the safety factor included is about 6 percent larger than the measured value.

APPENDIX B
TERMS AND UNITS USED IN GUIDE

T - Total reactor operating time, *s*

k - An index specifying an operating period at constant power.

T_k - Operating time of the k^{th} irradiation interval, *s*

t - Time after final shutdown to the desired decay time, *s*

t_k - Time after operating period k to the desired decay time, *s*

$f_i(t)$ - Decay heat power, *t* seconds after a fission pulse from fissionable nuclide *i*, (MeV/s)/fission

α_{ij} - Coefficients used to define decay heat power as 23 exponential terms

λ_{ij} - Exponent time constants used to define decay heat power as 23 exponential terms, *1/s*

$F_i(t,T)$ - 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, *MeV/fission*[a]

Q_i - Total recoverable energy associated with fission of nuclide *i*, *MeV/fission*

S_k - Total specific thermal power from fission during operational period *k*, *W/kgU*[b]

S_{ik} - Contribution from fission of nuclide *i* to the total specific thermal power during operational period *k*, *W/kgU*

$P_T(t,T)$ - Total decay heat power from all contributions at *t* seconds after shutdown from an operating history of *T* seconds duration, *W/kgU*

$P_F(t,T)$ - 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, *W/kgU*

$P_{Fi}(t,T)$ - Fission product decay heat power contribution from i^{th} fissionable nuclide, uncorrected for neutron capture in fission products, *W/kgU*

$P_A(t,T)$ - Contribution of actinides to the decay heat power, *W/kgU*

$P_C(t,T)$ - Contribution of Cs-134 to the decay heat power, *W/kgU*

$P_E(t,T)$ - Contribution to decay heat power from neutron capture by other fission products, *W/kgU*

$P_S(t,T)$ - Contribution to decay heat power from activated structural components, *W/kgU*

$F_S(t)$ - Safety factor applied to $P_T(t,T)$ to account for uncertainty in the methods

$P'_T(t,T)$ - Total decay heat power with the safety factor included, *W/kgU*

E_S - Average initial enrichment of the fuel assembly, *wt% U-235*

B_K - Average burnup of the fuel assembly following k^{th} irradiation interval, *MWd/kgU*

Subscripts

i Subscript referring to the fissionable isotopes U-235, PU-239, U-238, and PU-241 (*i* =1 to 4)

j Subscript for the expansion terms of the exponential function for the decay heat power following a pulse fission (*j*=l to 23),

k 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 $W = 6.243 \times 10^{12}$ *MeV/s*).