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TECHNICAL SUPPORT FOR A PROPOSED DECAY HEAT GUIDE USING SAS2H/ORIGEN-S DATA

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ABSTRACT

This report was prepared to provide support for major revisions to the current U.S. Nuclear Regulatory Commission decay heat rate guide entitled "Regulatory Guide 3.54, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," using a new data base produced by the SAS2H analysis sequence of the SCALE-4 system. The new data base of heat generation rates provides a significant improvement by increasing the number and range of parameters that generally characterize pressurized-water-reactor (PWR) and boiling-water-reactor (BWR) spent fuel assemblies. Using generic PWR and BWR assembly models, calculations were performed with each model for six different burnups at each of three separate specific powers to produce heat rates at 20 cooling times in the range of 1 to 110 y. A procedure that includes proper interpolation formulae for the tabulated heat generation rates is specified. Adjustment formulae for the interpolated values are provided to account for differences in initial ^{235}U enrichment and changes in the specific power of a cycle from the average value. Finally, safety factor formulae were derived as a function of burnup, cooling time, and type of reactor. The procedure included in this report was developed with the intention of providing one that was easier to use than that in the current Regulatory Guide. Also, the complete data base and procedure is incorporated into an interactive code called LWRARC which can be executed on a personal computer.

The report shows adequate comparisons of heat rates computed by SAS2H/ORIGEN-S and measurements for 10 BWR and 10 PWR fuel assemblies. The average differences of the computed minus the measured heat rates of fuel assemblies were –0.7 ± 2.6% for the BWR and 1.5 ± 1.3% for the PWR. In addition, a detailed analysis of the proposed procedure indicated the method and equations to be valid.

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FOREWORD

This report will provide technical support for major revisions proposed to the current U.S. Nuclear Regulatory Commission (NRC) decay heat rate guide entitled "Regulatory Guide 3.54, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation." A proposed revised guide is now under development by the NRC staff. The proposed procedure applies computed results of the SAS2H/ORIGEN-S analyses sequence of the SCALE-4 system, a more recent version of the software than that used for the current guide. The calculated decay heat rate data base proposed here has a broader application and is designed to be easier to use than that in the current guide.

This report is not a substitute for NRC regulation, and compliance is not required. The approaches and/or methods described in this document are provided for information only. Publication of this report does not necessarily constitute NRC approval or agreement with the information contained herein.

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1 INTRODUCTION

Heat is generated during the radioactive decay of discharged fuel from nuclear power reactors. The assurance of proper methods of storing the spent fuel assemblies requires knowledge of their decay heat generation rates (also, known as decay heats or afterheat powers). Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," that was issued in September 1984, addresses acceptable methods for calculating long-term heat generation rates. Recently, improved nuclear data libraries and computational models incorporated into ORIGEN-S[1] and the SAS2H control module[2] of the SCALE-4 system[3] have been used to develop a basis for a substantial revision to the current decay heat rate guide. The purpose of this report is to present the data and analysis performed to support a proposed revision to the regulatory guide.

1.1 Background of Current Guide

The current version of Regulatory Guide 3.54 (issued in 1984) was developed upon the concept of providing a procedure that specifies proper interpolation and adjustment formulae for a data base of computed heat generation rates. The technical basis for the data and safety factors used in the current guide is reported in Ref. 4. The current guide relies on a decay heat data base calculated only for pressurized-water-reactor (PWR) fuel. With no measured heat generation data or validated calculations for boiling-water-reactor (BWR) fuel, the guide incorporated large safety factors to prevent the possibility of specifying nonconservative heat generation rates. In addition, only a single maximum specific power (rather than a range of specific power values) was used in the analyses. The current guide provides decay heat rates that are fairly accurate (within several percent) for PWR assemblies that were operated at or near the maximum power and decayed for relatively short cooling times. However, for BWR assemblies and PWR assemblies with more typical power densities (commonly with average specific power levels near half the maximum used for the guide basis), conservative heat rates are produced by the current guide. The main cause for this overestimation of heat rates is the result of using an upper envelope of the possible operating powers and is not the result of the computational model.

Since completion of the technical basis for the current guide, a number of decay heat measurements have been performed for PWR and BWR spent fuel. Thus, the NRC decided to study the possibility of revising the current guide to reduce the conservatism by (1) developing separate decay heat data bases for PWR and BWR fuel and (2) increasing the decay heat data base to encompass a broader range of parameters selected to characterize the PWR and BWR spent fuel.

1.2 Improvements in the Proposed Guide

In developing a proposed revision to the current regulatory guide, the goal is to provide significant technical improvements while also providing an easier-to-use format and/or formulae. The technical improvements discussed in this report were made by adding a data base for BWR assemblies and increasing the number and range of parameters selected to characterize the spent fuel (i.e., burnup, specific power, initial enrichment). This subsection briefly discusses these improvements.

Analyses performed to provide a basis for the current guide used the SAS2 analysis sequence provided in the SCALE-2 and SCALE-3 releases of the SCALE code system. This earlier SAS2 procedure used a unit-fuel-pin-cell model at each depletion time step to obtain the flux spectrum required to obtain burnup- dependent cross sections for the fuel depletion analysis. This simple neutronics model was shown[4] to produce slightly conservative actinide inventories for PWR spent fuel and did not provide the flexibility required to model BWR fuel. After the release of the current guide, the neutronic analysis capabilities of SAS2 were significantly enhanced to form an updated sequence called SAS2H that was released as a module of the SCALE-4 system. The new SAS2H sequence was used to calculate the PWR and BWR decay heat rates used in preparing this revision to the current guide procedure. For each depletion time step, SAS2H performs one-dimensional (1-D) neutron transport analyses of the reactor fuel assembly using a two-part procedure with two separate unit-cell-lattice models. The first model considered in the sequence is a unit fuel-pin cell from which cell-weighted cross sections are obtained for use in the second model that represents a larger unit cell (e.g., an assembly) within an infinite lattice. The larger unit cell zones can be structured for different types of BWR or PWR assemblies. The neutron flux spectrum obtained from the large unit cell model is used to determine the appropriate nuclide cross sections for the specified burnup-dependent fuel composition.

A more detailed description of the improved analysis method is given in Sects. S2.2.2 through S2.2.5 of the SCALE-4.0 documentation.[3] Essentially, this expanded depletion model removes most of the conservatism from the computed actinide decay heat rates and, also, provides a procedure for calculating decay heat rates of spent fuel from BWRs as well as PWRs. Prior to generating the data base of decay heat rates, the SAS2H analysis procedure was validated using measured decay heat data obtained for PWR and BWR spent fuel assemblies.

The current regulatory guide is formatted to provide a set of tables containing decay heat rates as a function of parameter values that characterize a particular assembly. Using the appropriate table and interpolation guidelines, an appropriate decay heat value can be obtained from the tabular data base. This basic concept of interpolating a reference data base to obtain the decay heat value has been continued in preparing the proposed revisions to the current guide. However, the revised data base has been improved significantly by incorporating computed decay heat rates at six different burnups for each of three specific powers (compared with one maximum specific power in the current guide). Within each case, final decay heat generation rates were computed at 20 different cooling times in the range of 1 to 110 years. The ranges of the BWR burnup and power were 20 to 45 MWd/kgU and 12 to 30 kW/kgU, respectively. The PWR burnup and power ranges were 25 to 50 MWd/kgU and 18 to 40 kW/kgU, respectively. Also, additional cases were computed in which the ^{235}U enrichment was either decreased or increased by one-third from that of the standard case. Thus, the calculated decay heat rate data were produced as a function of burnup, specific power, cooling time, initial fuel ^{235}U enrichment, and assembly type (i.e., BWR or PWR).

An example demonstrating the significance of the improvement in using the actual specific power as opposed to a single maximum power can be seen in the following comparison. Consider a PWR assembly that has a burnup of 30 MWd/kgU and a specific power of 18 kW/kgU. At a cooling time of 2 years, the computed heat rate is 3.632 W/kgU. Had the heat rate result been determined from calculations using the maximum power of 40 kW/kgU, for which the computed heat rate is 5.129 W/kgU, the result would have been excessively conservative by 41%. Of course the differences between the heat rates at these two powers is decreased considerably at increased decay times.

The data base for a proposed guide revision has been developed to encompass the defining characteristics of the vast majority of spent fuel that is discharged from the mainstream of normal reactor operations. It was decided not to include assemblies with atypical characteristics because it would force the guide to be overly conservative for typical assemblies and/or significantly increase the computational effort and/or guide procedure.

1.3 Overview of Report

This introduction has provided a brief background of the current guide and a discussion giving justification for a proposed revision. The sources of the data used for the analyses are presented in Sect. 2. Section 3 presents the validation of the decay heat rate computational model performed by comparison with measured calorimetric data. A description of the cases producing heat rate data for a guide revision is given in Sect. 4. The tabulated data and complete procedure proposed for a revised guide are presented in Sect. 5. This section is followed by a detailed analysis of the method and equations. Finally, Sect. 7 provides a brief description of the LWRARC code (for use on a personal computer), which is an easy-to-use code applying the data base and procedures presented in Sect. 5. Also included are the addresses of two code centers from which the LWRARC code may be requested.

The Appendix contains assembly design and operation data, examples of the input for two of the tabulated cases, a sample problem using the proposed procedure, plots showing heat rates of major isotopes, and examples of the LWRARC code printouts.

2 SOURCES OF DATA FOR COMPUTING HEAT RATES

The ORIGEN-S nuclear data library[5] provided with the SCALE-4 system was the source of data for the half-lives, decay branching fractions, and recoverable energy per decay for all fission products and significant actinide and light element nuclides. These data were taken from either ENDF/B-V[6] or ENSDF.[7] A more detailed description of the source of all nuclear data is presented in Sect. M6.2.6 of Ref. 5. The fission product yield data were taken entirely from ENDF/B-V.[6] A convenient formatted listing of these yield data is presented in Ref. 8.

The SCALE 27-energy-group depletion cross-section library (27BURNUPLIB)[9] was used in all the SAS2H cases. This library contains ENDF/B-IV data for the major actinides and pre-release ENDF/B-V data for the fission products. For each reactor type (i.e., BWR or PWR), a preliminary SAS2H case was performed to produce an ORIGEN-S library (sometimes called a "preSAS library") to be used as the initial library in all subsequent SAS2H cases. Each preliminary SAS2H case performed one pass through the neutronics portion of the sequence in order to produce updated cross sections for 181 fission products, and provided a base PWR or BWR library to replace the outdated default ORIGEN-S library (made with circa 1960 data). The subsequent SAS2H cases used the BWR or PWR preSAS library as the base library and generated updated cross sections for 38 to 39 significant nuclides (plus 6 gadolinium isotopes for the BWR) as a function of burnup. A list of the nuclides that were updated as a function of burnup is provided in Table 2.1. The cross-section update was obtained from the neutronics analysis of the larger unit cell model which simulates the fuel assembly. The fuel spectrum from this analysis also provides new values for the spectral parameters THERM, RES, and FAST used by ORIGEN-S to model energy dependence within the depletion calculation.

The above data sources were applied to all the standard cases used to produce data for the proposed decay heat guide revision. Other sources of data used to evaluate the validity of the SAS2H data will be referred to later in this report.

Table 2.1 List of nuclides updated as a function of burnup in the SAS2H analyses

^{1}H	^{143}Pr	$^{158}Gd^{b}$
^{10}B	^{143}Nd	$^{160}Gd^{b}$
$^{11}B^{a}$	^{145}Nd	^{234}U
^{16}O	^{147}Nd	^{235}U
^{59}Co	^{147}Pm	^{236}U
^{94}Zr	^{149}Sm	^{238}U
^{99}Tc	^{151}Sm	^{237}Np
^{106}Ru	^{152}Sm	^{238}Pu
^{103}Rh	^{153}Eu	^{239}Pu
^{105}Rh	^{154}Eu	^{240}Pu
^{131}Xe	^{155}Eu	^{241}Pu
^{135}Xe	$^{154}Gd^{b}$	^{242}Pu
^{133}Cs	$^{155}Gd^{b}$	^{241}Am
^{134}Cs	$^{156}Gd^{b}$	^{243}Am
^{144}Ce	$^{157}Gd^{b}$	^{244}Cm

[a]Only in PWR cases.
[b]Only in BWR cases.

3 COMPARISONS OF COMPUTED AND MEASURED HEAT RATES

The reliability of a computer code to calculate decay heat generation rates can be demonstrated by comparing calorimetric measurements of heat rates of spent fuel assemblies with values computed by using code input similar to the design and operating characteristics of the fuel assemblies. In this study, results were compared for ten PWR and ten BWR spent fuel assemblies. In the comparison benchmarks, the fuel came from three reactors—Point Beach Unit 2 PWR, Turkey Point Unit 3 PWR, and Cooper Nuclear Station BWR. The heat rates of the Turkey Point assemblies were measured[10] at the Engine Maintenance Assembly and Disassembly Facility at the Nevada Test Site. The measurements of the Point Beach and Cooper Station assemblies were performed[11,12] at General Electric's Morris Facility. Decay heat results from these sets of measurements were taken from Refs. 13 through 15.

This section presents a description of the design and reactor operating characteristics of the spent fuel used in the measurements. Then, the comparison of measured and calculated heat rates is listed and summarized.

3.1 Assembly Design and Operating Characteristics

The design and operating history data of each measured assembly from the three types of reactors is listed in Tables 3.1 through 3.12, inclusive. The data in these tables are sufficient for the required input to the SAS2H and ORIGEN-S modules used for computing heat rates of all the fuel assemblies in this comparison study. However, more detailed operating histories[15] than the data in Table 3.9 were used for some of the Cooper Nuclear Station assemblies when considered necessary to account for large power fluctuations over a fuel cycle. Initial uranium isotopic contents were determined from Table 3.12, as in a previous similar procedure.[23] The isotopic ratio factors in Table 3.12 were simply taken from those derived from mass spectrometer analyses[24] of initial fuel for the Yankee Reactor Core V. Part of the assembly data (e.g., some of the temperatures) were not available in the references on the experiments and assembly designs. In those cases, the values were taken from Ref. 17 for the generic case of the reactor. Table 3.11 lists the element contents (excluding that of uranium) of the measured assemblies along with the amounts applied in the PWR and BWR cases used to create the decay heat data base presented in Sect. 5.

The greatest uncertainty in the calculated decay heat rates caused by inaccurate input data is probably that resulting from the uncertainty in the cobalt contents listed in Table 3.11. Measurements[25,26] of cobalt in clad and structural materials known to the authors indicate amounts significantly less than that used here. In order to be consistent, the same cobalt content per kgU for each reactor type was applied in both the cases for measurement comparisons and the cases for the new decay heat data base of Sect. 5.

From the tabulated data on the measured assemblies, note that some of the assemblies were in the reactor during the same cycles (i.e., had similar overall operating histories) and had approximately equal burnups (within 5%). These similarities in assembly characteristics permitted a more efficient use of computational time because a separate SAS2H calculation was not needed for each assembly. Instead, for each set of assemblies with similar characteristics, the SAS2H sequence was used to perform a reactor depletion analysis with an approximate operating history. The burnup-dependent cross-section libraries created by the SAS2H case were then accessed in separate stand- alone ORIGEN-S cases that modeled the specific operating history and decay time for each applicable assembly. In these cases, the heat rate difference from not using SAS2H for individual assemblies was estimated to be <1%.

3.2 Discussion of Comparisons

Comparisons of measured and SAS2H/ORIGEN-S calculated decay heat rates of spent fuel assemblies from the three reactors in this study are listed in Tables 3.13, 3.14, and 3.15. The major parameters of total burnup, ^{235}U enrichment, and cooling time of each assembly are shown. The measured and computed heat rates for each experiment are listed. Percentage differences between the measured and calculated values are presented to provide a measure of the comparison. In addition to the percentage difference for each measurement, the average percentage difference of all the measurements on each assembly is indicated. Averages of both types of percentage differences and the standard deviations of the average differences are also provided.

There is at least one excessively high percentage difference in each of the three tables. These data were not excluded because it was decided to use comparisons for **all** reported measurements for which pertinent parameters were available. Each of the calculated heat rates reported for the Point Beach PWR assemblies in Table 3.13 was higher than the measured value. However, except for the

Table 3.1 Point Beach Unit 2 PWR assembly description

Parameter	Data	Reference
Assembly general data		
Designer	Westinghouse	16
Lattice	14 × 14	16
Fuel weight, kg U	386	14
Water temperature, K	579	16
Water pressure, psia (12/73)	2000	16
Water density, avg, g-cm^{-3}	0.7115[a]	
Soluble boron, cycle avg, ppm (wt)	550	17
Number of fuel rods	179	18
Number of guide tubes[b]	16	18
Number of instrument tubes	1	18
Fuel rod data		
Type fuel pellet	UO_2	16
	9.467[a]	
Pellet stack density, g-cm^{-3}	1.412 (0.556)	18
Rod pitch, cm (in.)	1.0719 (0.422)	18
	0.9484 (0.3734)	18
Rod OD, cm (in.)	365.8 (144)	18
	Zircaloy-4	18
Rod ID, cm (in.)	0.030[a]	
	3.397	14
Active fuel length, cm (in.)	0.016[a]	
Clad material	96.557[a]	
	811	17
^{234}U wt %	620	17
^{235}U wt %		
^{236}U wt %		
^{238}U wt %	0.6845 (0.539)	18
Effective fuel temperature, K	0.6414 (0.505)	18
Clad temperature, K	Zircaloy-4	18
Guide tube data[b]		
Inner radius, cm (ID, as in.)		
Outer radius, cm (OD, as in.)		
Tube material		

[a]These data were calculated from other data in the table.
[b]Control rods were considered to be fully withdrawn during reactor uptime.

Table 3.2 Point Beach Unit 2 PWR operating history[a]

Cycle	Startup	Shutdown	Uptime, d	Downtime, d	Core-avg burnup, MWd/kgU
1A	8/01/72[b]		273	0	1.070
1B	5/01/73[b]	10/16/74[b]	533	65	15.993
2	12/20/74	2/26/76	433	32	11.806
3	3/29/76	3/03/77	339		
					10.040

[a]See Ref. 14.
[b]Cycle 1 was split in order to apply significantly different powers more correctly.

Table 3.3 Point Beach Unit 2 assembly burnups[a] and powers

	Fuel assembly ID					
Burnup, MWd/kgU	C-52	C-56	C-64	C-66	C-67	C-68
Cycle 1	10.801	16.475	16.920	11.668	16.600	13.034
Cycle 2	12.316	12.881	12.844	13.256	12.801	13.908
Cycle 3	8.797	9.561	9.620	10.509	9.545	10.115
Power,[b] MW/assembly						
Cycle 1A	0.958	1.461	1.500	1.035	1.472	1.156
Cycle 1B	7.332	11.183	11.485	7.920	11.268	8.847
Cycle 2	10.979	11.483	11.450	11.817	11.412	12.398
Cycle 3	10.017	10.887	10.954	11.966	10.868	11.517

[a]See Ref. 14.
[b]Computed from the uptimes and core-averaged burnups in Table 3.2, 386 kgU/assembly, and the above burnups.

Table 3.4 Turkey Point Unit 3 PWR assembly description

Parameter	Data	Reference
Assembly general data		
Designer	Westinghouse	19
Lattice	15 × 15	19
Fuel weight of B-43, kgU	447.8[a]	
Fuel weight of D-15, kgU	456.1[a]	
Fuel weight of D-22, kgU	458.0[a]	
Fuel weight of D-34, kgU	455.2[a]	
Water temperature, K	570	4
Water density, g-cm^3	0.7311	4
Soluble boron, cycle avg, ppm (wt)	450	4
Number of fuel rods	204	4,18
Number of guide tubes[b]	20	18
Number of instrument tubes	1	18
Fuel rod data		
Type fuel pellet	UO_2	4,18
Stack density (B-43), % TD	91.53[c]	
Stack density (D-15), % TD	93.23[c]	
Rod pitch, cm (in.)	1.4300 (0.563)	4,18
Rod OD, cm (in.)	1.0719 (0.422)	4,18
Rod ID, cm (in.)	0.9484 (0.3734)	4,18
Pellet OD, cm (in.)	0.9296 (0.366)	4
Active fuel length	365.8 (144)	4,18
Clad material	Zircaloy-4	4,18
^{234}U wt %	0.023[c]	
^{235}U wt % (B-assemblies)	2.559[a]	
^{235}U wt % (D-assemblies)	2.557[a]	
^{236}U wt %	0.012[c]	
^{238}U wt % (B-assemblies)	97.406[c]	
^{238}U wt % (D-assemblies)	97.408[c]	
Effective fuel temperature, K	922	4
Clad temperature, K	595	4
Guide tube data[b]		
Inner radius, cm (ID, as in.)	0.6502 (0.512)	18
Outer radius, cm (OD, as in.)	0.6934 (0.546)	18
Tube material	Zircaloy-4	18

[a]Florida Power and Light Co. data provided by E. R. Knuckles.
[b]Control rods were considered to be fully withdrawn.
[c]These data were calculated from other data in the table.

Table 3.5 Turkey Point Unit 3 PWR operating history[a]

Cycle	Startup	Shutdown	Uptime, d	Downtime, d
1	10/20/72	10/04/74	714	73
2	12/16/74	10/26/75	314	58
3	12/23/75	11/15/76	327	62
4	1/16/77	11/24/75	312	

[a]Florida Power and Light Co. data provided by E. R. Knuckles.

Table 3.6 Turkey Point Unit 3 assembly burnups[a] and powers

	Fuel assembly ID			
Burnup, MWd/kgU	B-43	D-15	D-22	D-34
Cycle 1	15.998			
Cycle 2	8.829	9.480	9.826	9.488
Cycle 3		9.752	8.867	9.338
Cycle 4		8.920	7.253	8.794
Power,[b] MW/assembly				
Cycle 1	10.033			
Cycle 2	12.591	13.771	14.332	13.756
Cycle 3		13.603	12.419	13.000
Cycle 4		13.040	10.647	12.831

[a]Florida Power and Light Co. data obtained from PDQ-7 analyses and provided by E. R. Knuckles.
[b]Computed from the uptimes in Table 3.5, the uranium weights per assembly in Table 3.4 and the above burnups.

Table 3.7 Cooper Nuclear Station BWR assembly description

Parameter	Data	Reference
Assembly general data		
Designer	General Electric	20
Lattice	7 × 7	20
Fuel weight, kgU	190.2[a]	15
Water temperature, K	558	17
Water vol-avg density, g-cm^{-3}	0.4323	17
Number of fuel rods	49	20
Burnable poison element	Gd	20
Number containing poison	4	17
Assembly pitch, cm (in.)	15.24 (6.0)	15
Shroud (tube) thickness, cm (in.)	0.2032 (0.08)	15
Shroud inside flat-to-flat, cm (in.)	13.406 (5.278)	15
Shroud material	Zircaloy-4	17
Shroud temperature, K	558	17
Channel water density, g-cm^{-3}	0.669[b]	17
Channel water temperature, K	552	17
Channel avg ^{10}B content, atom/b-cm	7.15 × 10^{-6} (see footnote c)	
Fuel rod data		
Type fuel pellet	UO_2	18
Pellet stack density, g-cm^{-3}	9.96[d]	
Rod pitch, cm (in.)	1.8745 (0.738)	15
Rod OD, cm (in.)	1.4300 (0.563)	15
Rod ID, cm (in.)	1.2421 (0.489)	15
Active fuel length, cm (in.)	365.76 (144)	20
Clad material	Zircaloy-2	20
Gadolinia bearing rods, Gd wt %	3.5[e]	
Assembly CZ102 average U content:		
^{234}U wt %	0.010[d]	
^{235}U wt %	1.100	15
^{236}U wt %	0.005[d]	
^{238}U wt %	98.885[d]	
Average U content, all except CZ102:		
^{234}U	0.022[d]	
^{235}U	2.500	15
^{236}U	0.012[d]	
^{238}U	97.466[d]	
Effective fuel temperature, K	840	17
Clad temperature, K	620	17

[a]Some assemblies had 190.5 kgU. However, the 190.2 value was used in the analyses.
[b]Reduced the 0.743 g-cm^{-3} bottom node density by 10% to account for control cruciform displacement.
[c]Applied in channel region for boron cruciform; used content producing average k_{eff} of approximately unity.
[d]These data were calculated from other data in the table.
[e]Used the average of 3 and 4 wt % Gd, each the content of two rods.

Table 3.8 Cooper Nuclear Station BWR operating history[a]

Cycle	Cycle start	Since startup, d	Shutdown	Since startup, d	Uptime, d	Downtime, d
1	7/03/74	0	9/17/76	807	807	59
2	11/15/76	866	9/17/77	1172	306	31
3	10/18/77	1203	3/31/78	1367	164	35
4	5/05/78	1402	4/17/79	1749	347	23
5	5/10/79	1772	3/01/80	2068	296	98
6	6/07/80	2166	4/20/81	2483	317	48
7	6/07/81	2531	5/21/82	2879	348	

[a]See Ref. 15.

Table 3.9 Cooper Nuclear Station adjusted cycle burnups[a]

Assembly	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5	Cycle 6	Cycle 7
CZ102	9.394	2.273					
CZ205	10.298	7.414	2.987	0	0	1.864	2.781
CZ209	10.651	7.669	3.110	0	0	2.296	1.657
CZ259	6.026	4.339	2.555	8.511	2.165	2.870	
CZ331	12.875	5.495	2.962				
CZ369	11.162	8.035	2.481	0	0	1.982	2.916
CZ429	10.878	7.833	2.899	0	0	3.232	2.799
CZ515	11.003	7.922	0	0	2.691	4.121	
CZ526	10.939	7.875	2.734	0	0	3.239	2.809
CZ528	10.996	7.917	0	0	2.692	4.110	

[a]See Ref. 15.

Table 3.10 Cooper Nuclear Station assembly powers

	Powers[a] by cycles, MW/assembly						
Assembly	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5	Cycle 6	Cycle 7
CZ102	2.214	1.413					
CZ205	2.427	4.608	3.464	0	0	1.118	1.520
CZ209	2.510	4.767	3.607	0	0	1.378	0.906
CZ259	1.420	2.697	2.963	4.665	1.391	1.722	
CZ331	3.034	3.416	3.435				
CZ369	2.631	4.994	2.877	0	0	1.189	1.594
CZ429	2.564	4.869	3.362	0	0	1.939	1.530
CZ515	2.593	4.924	0	0	1.729	2.473	
CZ526	2.578	4.895	3.171	0	0	1.943	1.535
CZ528	2.592	4.921	0	0	1.730	2.466	

[a]Computed from uptimes in Table 3.8, burnups in Table 3.9 and 190.2 kgU/assembly.

Table 3.11 Element contents[a] from clad, structure, and water (for BWR)

Element[b]	BWR g/kgU	PWR g/kgU	Cooper Station kg/assembly	Point Beach kg/assembly	Turkey Point kg/assembly
H	16.4		3.1		
B	0.068		0.013		
O	265.0	135.0	50.5	52.0	62.0
Cr	2.4	5.9	0.45	2.3	2.7
Mn	0.15	0.33	0.029	0.13	0.15
Fe	6.6	12.9	1.2	5.0	5.9
Co	0.024	0.075	0.0046	0.029	0.034
Ni	2.4	9.9	0.45	3.8	4.5
Zr	516.0	221.0	98.2	85.0	101.0
Nb	0	0.71	0	0.27	0.32
Sn	8.7	3.6	1.6	1.4	1.6
Gd	c		0.544		

[a]Calculated from data and factors in Ref. 21, except for spectral correction factors in Ref. 22 for PWRs.
[b]Included only elements with contents exceeding 0.5 g/kgU plus Mn, Co, and B (for BWR only).
[c]The Gd in BWR standard cases varied with wt % Gd in pins.

Table 3.12 Uranium isotope dependence[23] on X wt % ^{235}U enrichment

Isotope	Assay, wt %
^{234}U	0.0089 X
^{235}U	1.0000 X
^{236}U	0.0046 X
^{238}U	100 1.0135 X

Table 3.13 Point Beach PWR measured[a] and computed decay heat rates

Assembly ID	Burnup, MWd/kgU	Initial ^{235}U wt %	Cooling time, d	Heat rate, W Meas.	Heat rate, W Calc.	% Difference (C/M-1)100%	% Difference assembly-avg
C-52	31.914	3.397	1635	724[b]	732.2	1.1	
			1635	723[c]	732.2	1.3	1.2
C-56	38.917	3.397	1634	921	943.3	2.4	2.4
C-64	39.384	3.397	1633	931[b]	959.0	3.0	
			1633	825[c]	959.0	16.2	9.6
C-66	35.433	3.397	1630	846	852.2	0.7	0.7
C-67	38.946	3.397	1629	934	946.5	1.3	1.3
C-68	37.057	3.397	1630	874	898.0	2.7	2.7
Average						3.6	3.0
Standard deviation						±2.3	±1.9

[a]See Ref. 14.
[b]Static test.
[c]Recirculation test.

16.2% value, the differences did not exceed 3%. This reactor was the only reactor for which the average difference exceeded the standard deviation (i.e., 3.0 ± 1.9%) and, therefore, it indicates there is a systematic bias to calculate decay heat rates higher than the measured data. The burnups and ^{235}U enrichments also were higher than the other assemblies compared. The 3% difference for the C-64 assembly was the result of a comparison with a measurement by a static test, whereas the 16.2% resulted from a comparison with a measurement that was determined by a recirculation test. Had the assembly been excluded from consideration, the average percentage difference of the other assemblies would have been 1.7 ± 0.9%.

The computed heat rates of the Turkey Point PWR assemblies in Table 3.14 were both higher and lower than measured values. A previous comparison[4] of SAS2 results with measurements applied equal burnups and specific powers[13] for the three cycles of the D-assemblies. This rather rough estimate of operating history was improved in the present calculations by using more complete data given by the operating utility (see Tables 3.5–3.6). The results in Table 3.14 show that three of the assembly average differences were within 2.3%. The remaining assembly, B-43 (which had a –4.5% difference), was the only one of the four that was in the reactor during the first cycle. The lower calculated value for assembly B-43 could be caused by extremely low operating powers during initial reactor startup. This extremely low power during the early period of the first cycle lowered the average cycle power below that used for most of the cycle-1 burnup. Thus, the measured decay heat rate of assembly B-43 is greater than it would have been for the use of a constant power in cycle 1, because the decay time of part of the fission products is less. The average assembly percentage difference, however, of –0.7 ± 1.7% indicates good agreement.

The percentage differences in the comparisons in Table 3.15 for the Cooper Nuclear Station BWR assemblies extended through a much wider range than those for the PWRs. However, the decay heat measurements were much lower values (62.3 to 395.4 W) than those measured for the PWRs (625 to 1550 W). Because measurement precision tends to be represented as a constant heat rate instead of a percentage of the total heat rate, the percentage uncertainty in the measured data would increase as the measured value decreases. Thus, the broader range of percentage differences in measured and calculated decay heat rates is expected. The increase in the number of measurements and assemblies, however, has somewhat reduced the final standard deviation. The average assembly difference of –0.7 ±2.6% shows good agreement between calculated and measured values for the BWR assemblies.

A summary of percentage differences in comparisons of measured and calculated spent fuel decay heat rates for all cases and assemblies is presented in Table 3.16. The average heat rate computed was less than the measured value for the BWR assemblies and the opposite was true for the PWR assemblies. The final average difference for all 20 LWR spent fuel assemblies was 0.4 ± 1.4%. Then at the confidence level associated with 2 standard deviations the percentage differences should lie in the range –2.4 to 3.2%. Thus, at the 2σ confidence level and for the design and operating parameters of the given assemblies, the nonconservative error in computed decay heat rates should not exceed 2.4% plus any nonconservative bias in the measurements. The comparisons of measured and calculated decay heat rates shown in this section provide the basis for the calculational bias that will be used in development of a proposed regulatory guide.

Table 3.14 Turkey Point PWR measured[a] and computed decay heat rates

Assembly ID	Burnup, MWd/kgU	Initial ^{235}U wt %	Cooling time, d	Heat rate, W Meas.	Heat rate, W Calc.	% Difference (C/M-1)100%	% Difference assembly-avg
B-43	24.827	2.559	1782	637	608.1	–4.5	–4.5
D-15	28.152	2.557	962	1423	1436.0	0.9	
			1144	1126	1172.0	4.1	
			2077	625	628.4	0.5	1.8
D-22	25.946	2.557	963	1284	1255.0	–2.3	–2.3
D-34	27.620	2.557	864	1550	1582.0	2.1	2.1
Average						0.1	–0.7
Standard deviation						±1.3	±1.7

[a]See Ref. 14.

Table 3.15 Cooper Nuclear Station BWR measured[a] and computed decay heat rates

Assembly ID	Burnup, MWd/kgU	Initial ^{235}U wt %	Cooling time, d	Heat rate, W Meas.	Heat rate, W Calc.	% Difference (C/M-1)100%	% Difference assembly-avg
CZ102	11.667	1.1	2565	62.3	78.9	26.6	
			2645	70.4	77.8	10.5	18.6
CZ205	25.344	2.5	857	324.0	328.3	1.3	
			867	361.0	325.3	–9.9	
			871	343.5	324.1	–5.6	
			872	353.2	323.8	–8.3	
			886	331.8	319.8	–3.6	
			887	338.7	319.5	–5.7	
			892	327.5	318.1	–2.9	
			896	313.1	316.9	1.2	
			899	311.4	316.1	1.5	
			930	314.0	307.8	–2.0	
			936	331.2	306.2	–7.5	
			946	317.1	303.7	–4.2	–3.8
CZ209	25.383	2.5	891	279.5	290.1	3.8	3.8
CZ259	26.466	2.5	1288	247.6	285.7	15.4	
			1340	288.5	278.5	–3.5	6.0
CZ331	21.332	2.5	2369	162.8	161.6	–0.7	
			2457	180.1	158.2	–12.2	–6.5
CZ369	26.576	2.5	888	347.6	340.4	–2.1	–2.1
CZ429	27.641	2.5	889	385.6	366.5	–5.0	–5.0
CZ515	25.737	2.5	1254	294.0	282.3	–4.0	
			1285	296.0	276.7	–6.5	–5.3
CZ526	27.596	2.5	864	395.4	374.7	–5.2	–5.2
CZ528	25.715	2.5	1286	297.6	275.4	–7.5	–7.5
Average						–1.4	–0.7
Standard deviation						±1.7	±2.6

[a]See Ref. 15.

Table 3.16 Summary of decay heat rate comparisons

Type of summary	Number	% Difference[a] ± std dev
Summary by cases:		
Average Point Beach case	8	3.6 ± 2.3
Average Turkey Point case	6	0.1 ± 1.3
Average Cooper case	25	1.4 ± 1.7
Average PWR case	14	2.1 ± 1.4
Average BWR case	25	1.4 ± 1.7
Average, PWR and BWR avg-case		0.3 ± 1.1
Summary by assemblies:		
Average Point Beach assembly 6	3.0 ± 1.9	
Average Turkey Point assembly	4	0.7 ± 1.7
Average Cooper assembly	10	0.7 ± 2.6
Average PWR assembly	10	1.5 ± 1.3
Average BWR assembly	10	0.7 ± 2.6
Final average, all assemblies	20	0.4 ± 1.4

[a](Calculated/measured − 1)100%.

4 HEAT RATE DATA COMPUTED FOR PROPOSED GUIDE

This section provides a few summary remarks about the SAS2H/ORIGEN-S cases used to calculate the decay heat rates. For each reactor type, combinations of six different burnup values and three different specific powers were considered. The ranges of the BWR burnup and power were 20 to 45 MWd/kgU and 12 to 30 kW/kgU, respectively. The PWR burnup and power ranges were 25 to 50 MWd/kgU and 18 to 40 kW/kgU, respectively. Final decay heat generation rates were calculated in each case at 20 different cooling times ranging from 1 to 110 years. A total of 720 decay heat generation rates were calculated from the 18 PWR and 18 BWR cases.

The PWR and BWR assembly design and operating characteristics applied in the SAS2H/ORIGEN-S cases are taken from the generic data provided in Ref. 17. The specific data used for the cases are provided in detail in Appendix A. For the BWR cases, the contents of gadolinium in the fuel, the boron in the cruciform control assemblies and the coolant density between the assembly shrouds were changed from the generic-case values to represent more realistic data. The specific powers, burnup, and initial ^{235}U fuel enrichments were changed from that provided in Ref. 17 to those needed to span the desired range for each parameter. The cycle times were changed to produce the proper burnup and power. Uptimes of 80% (which includes the effect of reload downtimes) were used for all cycles except the last one was considered to be 100% uptime. Three cycles were used for the two lowest burnup cases, four cycles for the next two higher in burnup and five cycles for the two highest in burnup.

In the procedure provided in Sect. 5, the heat rate corresponding to the conditions given for a particular assembly is first determined by interpolating tabulated values linearly between powers and burnups and logarithmically between cooling times. However, this interpolated value corresponds to the computed heat rate at only the power, burnup, and cooling time specified. The interpolated heat rate value must be corrected for significant changes between other conditions used in the calculations and those of the given assembly (e.g., ^{235}U enrichment or operating history). Most of these different parameter variations cause small enough changes in the results that their effects could be conveniently included in the safety factor. However, explicit factors are derived for deviations from the calculations in parameters of the assembly such as the initial ^{235}U enrichment and the last two operating cycle powers. These factors are then applied as adjustments to the interpolated value. An additional safety factor is applied as a function of reactor type, burnup, and cooling time. A more detailed analysis and discussion of the factors and the method in general are given in Sect. 6.

5 PROPOSED REGULATORY GUIDE PROCEDURE

This section of the report presents the proposed procedure for use in a revision to the NRC regulatory guide on spent fuel heat generation in an independent spent fuel storage installation. Section 5.1 contains the definitions, as used here, of parameters needed in the determination of the heat generation rate of a fuel assembly. Section 5.2.1 contains the procedure for interpolating tables to derive the uncorrected heat rate of an assembly. Sections 5.2.2-5.2.6 include the final evaluation method that uses simple adjustment factors for cases that are somewhat nontypical, in addition to the specified safety factor.

There may be fuel assemblies with characteristics that lie sufficiently outside the mainstream of typical plant operations as to require a separate method for predicting the heat generation rate. Assemblies whose parameters lie outside the range of values used in the guide may be considered atypical for the purposes of using the proposed guide revision. A discussion of the characteristics of assumed typical reactor operations is in Sect. 5.3. A glossary of terms is given in Sect. 5.4.

5.1 Definitions and Derivations of Required Parameters

The following definitions are used in the proposed guide procedure.

5.1.1 Heat Generation Rate of the Assembly (*p*)

The **heat generation rate** of the spent fuel assembly is the recoverable thermal energy (from radioactive decay) of the assembly per unit time per unit fuel mass. The units for heat generation rate used in this guide are W per kg U, where U is the initial uranium loaded. Heat generation rate has also been referred to as decay heat rate, afterheat, or afterheat power.

5.1.2 Cycle and Cycle Times of the Assembly (T_i)

A **cycle** of the operating history for a fuel assembly is the duration between the time criticality is obtained for the initially loaded or reloaded reactor to the time at which the next reloaded core becomes critical. The exception is for the last cycle where the cycle ends with the last reactor shutdown before discharge of the assembly. T_i denotes the elapsed time during cycle i for the assembly. Specifically, the first and last cycles are denoted by $i = s$ (for start) and $i = e$ (for end), respectively. T_{res}, the total residence time of the assembly, is the sum of all T_i for $i = s$ through e, inclusive. Except for the last cycle for an assembly, the cycle times include the downtimes during reload. Cycle times, in this guide, are in **days**.

5.1.3 Fuel Burnup of the Assembly (B_i and B_{tot})

The fuel burnup of cycle i, B_i, is the recoverable thermal energy per unit fuel mass during the cycle in units of megawatt days per metric ton (tonne) initial uranium (MWd/tU) or in the SI units* of mass used in the guide, megawatt day per kilogram U (MWd/kgU). B_i is the maximum estimate of the fuel assembly burnup during cycle i. B_{tot} is the total operating history burnup:

$$B_{tot} = \sum_{i=s}^{e} B_i \, . \qquad (1)$$

5.1.4 Specific Power of the Fuel (P_i, P_e, and P_{ave})

Specific power has a unique meaning in the guide. The reason for developing this definition is to take into account the differences between the actual operating history of the assembly and that used in the computation of the tabulated heat generation rates. The calculational model applied an uptime (time at power) of 80% of the cycle time in all except the last cycle (of the discharged fuel assembly), which had no downtime. The definition of specific power, used here, has two basic characteristics. First, when the actual uptime experienced by the assembly exceeds the 80% applied in the SAS2H/ORIGEN-S calculations, the heat rate changes by less than 1%. Second, when the actual uptime experienced is lower than the 80% applied in the calculations, the heat rate is reduced.

*The adopted International System of units.

The technical basis for these characteristics is presented in Sect. 6.1.

The specific power of cycle *i*, or *e* (last cycle), in kW/kgU, using burnup in MWd/kgU, is defined as

$$P_i = \frac{1000\,B_i}{0.8\,T_i} \text{ for } i < e\,;$$
$$P_e = \frac{1000\,B_e}{T_e} \text{ for } i = e\,. \tag{2}$$

The average specific power over the entire operating history of a fuel assembly, using the same units as in Eq. (2), is defined as

$$P_{ave} = \frac{1000\,B_{tot}}{T_e + 0.8\sum_{i=s}^{e-1} T_i}\,. \tag{3}$$

The average specific power through the next to last cycle is used in applying the adjustment factor for short cooling time (see Sect. 5.2.2). This parameter is defined as

$$P_{ave,e-1} = \frac{1000\,(B_{tot} - B_e)}{0.8\,(T_{res} - T_e)}\,. \tag{4}$$

Note that B_{tot} and P_{ave}, as derived in these definitions, are used in determining the heat generation rate from the guide. Also, for cooling times ≤7 years, P_e is used in an adjustment formula. The method applied here accommodates storage of a fuel assembly outside the reactor during one or two cycles and returning it to the reactor. Then, $B_i = 0$ may be set for all intermediate storage cycles. If the cooling time is short (i.e., <10 years), the results derived here may be excessively high for cases in which the fuel was temporarily discharged. Other evaluation methods that include the incorporation of storage cycles in the power history may be preferable.

5.1.5 Assembly Cooling Time (T_c)

The cooling time, T_c, of an assembly is the time elapsed from the last downtime of the reactor prior to its discharge (at end of T_e) to the time at which the heat generation rate is desired. Cooling times, in the guide, are in **years**.

5.1.6 Assembly Initial Fuel Enrichment (E_s)

The initial enrichment, E_s, of the fuel assembly is considered to be the average wt % ^{235}U in the uranium when it is first loaded into the reactor. Heat generation rates vary with initial enrichment for fuel having the same burnup and specific power; the heat rate increases with lower enrichment. If the enrichment is different than that used in the calculations at a given burnup and specific power, a correction factor is applied.

5.2 Determination of Heat Generation Rates

Directions for determining the heat generation rates of light-water-reactor (LWR) fuel assemblies from Tables 5.1-5.8 are given in this section. First, a heat rate, p_{tab}, is found by interpolation from Tables 5.1–5.3 or Tables 5.5–5.7. Then, a safety factor and all the necessary adjustment factors are applied to determine the final heat generation rate, p_{final}. There are three adjustment factors (see Sects. 5.2.2–5.2.4) plus a safety factor (see Sect. 5.2.5) that are applied in computing the final heat generation rate, p_{final}, from p_{tab}. In many cases, the adjustment factors are unity and thus are not required. An alternative to these directions is the use of the LWRARC code on a personal computer (see Sect. 7). This code evaluates p_{tab} and p_{final} using the data and procedures established in this section.

5.2.1 Computing Heat Rate Provided by Tables

Use Tables 5.1–5.3 for BWR fuel or Tables 5.5–5.7 for PWR fuel. The heat rates in each table pertain to a single specific power and are listed as a function of total burnup and cooling time. After determining P_{ave}, B_{tot}, and T_c, as defined above, select the next lower (L-index) and next higher (H-index) heat rate values from the tables so that:

$$\boldsymbol{P_L \# P_{ave} \# P_H;}$$
$$\boldsymbol{B_L \# B_{tot} \# B_H;}$$

and

$$\boldsymbol{T_L \# T_c \# T_H.}$$

Table 5.1 BWR spent fuel heat generation rates, watts per kilogram U, for specific power = 12 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU					
	20	25	30	35	40	45
1.0	4.147	4.676	5.121	5.609	6.064	6.531
1.4	3.132	3.574	3.955	4.370	4.760	5.163
2.0	2.249	2.610	2.933	3.281	3.616	3.960
2.8	1.592	1.893	2.174	2.472	2.764	3.065
4.0	1.111	1.363	1.608	1.865	2.121	2.384
5.0	0.919	1.146	1.371	1.606	1.844	2.087
7.0	0.745	0.943	1.142	1.349	1.562	1.778
10.0	0.645	0.819	0.996	1.180	1.369	1.561
15.0	0.569	0.721	0.876	1.037	1.202	1.370
20.0	0.518	0.656	0.795	0.940	1.088	1.240
25.0	0.477	0.603	0.729	0.861	0.995	1.132
30.0	0.441	0.556	0.672	0.792	0.914	1.039
40.0	0.380	0.478	0.576	0.678	0.781	0.886
50.0	0.331	0.416	0.499	0.587	0.674	0.764
60.0	0.292	0.365	0.438	0.513	0.589	0.666
70.0	0.259	0.324	0.387	0.454	0.520	0.587
80.0	0.233	0.291	0.347	0.405	0.464	0.523
90.0	0.212	0.263	0.313	0.365	0.418	0.470
100.0	0.194	0.241	0.286	0.333	0.380	0.427
110.0	0.179	0.222	0.263	0.306	0.348	0.391

Table 5.2. BWR spent fuel heat generation rates, watts per kilogram U, for specific power = 20 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU					
	20	25	30	35	40	45
1.0	5.548	6.266	6.841	7.455	8.000	8.571
1.4	4.097	4.687	5.173	5.690	6.159	6.647
2.0	2.853	3.316	3.718	4.142	4.540	4.950
2.8	1.929	2.296	2.631	2.982	3.324	3.673
4.0	1.262	1.549	1.827	2.117	2.410	2.705
5.0	1.001	1.251	1.501	1.760	2.024	2.292
7.0	0.776	0.985	1.199	1.420	1.650	1.882
10.0	0.658	0.838	1.023	1.215	1.413	1.616
15.0	0.576	0.731	0.890	1.056	1.227	1.403
20.0	0.523	0.663	0.805	0.954	1.107	1.263
25.0	0.480	0.608	0.737	0.871	1.009	1.150
30.0	0.444	0.560	0.678	0.800	0.925	1.053
40.0	0.382	0.481	0.579	0.682	0.786	0.893
50.0	0.332	0.417	0.501	0.588	0.677	0.767
60.0	0.292	0.365	0.438	0.513	0.589	0.666
70.0	0.259	0.324	0.386	0.452	0.518	0.585
80.0	0.233	0.290	0.345	0.403	0.460	0.519
90.0	0.211	0.262	0.311	0.362	0.413	0.465
100.0	0.193	0.239	0.283	0.329	0.375	0.421
110.0	0.178	0.220	0.260	0.302	0.343	0.385

Table 5.3. BWR spent fuel heat generation rates, watts per kilogram U, for specific power = 30 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU					
	20	25	30	35	40	45
1.0	6.809	7.786	8.551	9.337	10.010	10.706
1.4	4.939	5.721	6.357	7.006	7.579	8.169
2.0	3.368	3.958	4.463	4.979	5.453	5.938
2.8	2.211	2.651	3.050	3.460	3.855	4.256
4.0	1.381	1.705	2.016	2.339	2.663	2.991
5.0	1.063	1.335	1.605	1.885	2.172	2.462
7.0	0.797	1.015	1.239	1.471	1.713	1.958
10.0	0.666	0.850	1.039	1.237	1.443	1.653
15.0	0.579	0.737	0.898	1.067	1.242	1.422
20.0	0.525	0.667	0.811	0.962	1.117	1.276
25.0	0.482	0.611	0.741	0.877	1.017	1.160
30.0	0.445	0.563	0.681	0.805	0.931	1.061
40.0	0.382	0.482	0.581	0.685	0.790	0.898
50.0	0.332	0.418	0.502	0.589	0.678	0.769
60.0	0.292	0.366	0.438	0.513	0.589	0.666
70.0	0.259	0.323	0.386	0.451	0.517	0.584
80.0	0.232	0.289	0.344	0.401	0.459	0.517
90.0	0.210	0.261	0.310	0.361	0.411	0.463
100.0	0.192	0.238	0.282	0.327	0.372	0.418
110.0	0.177	0.219	0.259	0.300	0.340	0.382

Table 5.4 BWR enrichments for burnups in tables

Fuel burnup, MWd/kgU	Average initial enrichment, wt % U-235
20	1.9
25	2.3
30	2.7
35	3.1
40	3.4
45	3.8

Table 5.5. PWR spent fuel heat generation rates, watts per kilogram U, for specific power = 18 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU					
	25	30	35	40	45	50
1.0	5.946	6.574	7.086	7.662	8.176	8.773
1.4	4.485	5.009	5.448	5.938	6.382	6.894
2.0	3.208	3.632	4.004	4.411	4.793	5.223
2.8	2.253	2.601	2.921	3.263	3.595	3.962
4.0	1.551	1.835	2.108	2.398	2.685	2.997
5.0	1.268	1.520	1.769	2.030	2.294	2.576
7.0	1.008	1.223	1.439	1.666	1.897	2.143
10.0	0.858	1.044	1.232	1.430	1.633	1.847
15.0	0.744	0.905	1.068	1.239	1.414	1.599
20.0	0.672	0.816	0.963	1.116	1.272	1.437
25.0	0.615	0.746	0.879	1.018	1.159	1.308
30.0	0.566	0.686	0.808	0.934	1.063	1.197
40.0	0.487	0.588	0.690	0.797	0.904	1.017
50.0	0.423	0.510	0.597	0.688	0.780	0.875
60.0	0.372	0.447	0.522	0.601	0.680	0.762
70.0	0.330	0.396	0.462	0.530	0.599	0.670
80.0	0.296	0.355	0.413	0.473	0.534	0.596
90.0	0.268	0.321	0.372	0.426	0.480	0.536
100.0	0.245	0.293	0.339	0.387	0.436	0.486
110.0	0.226	0.270	0.312	0.356	0.399	0.445

Table 5.6. PWR spent fuel heat generation rates, watts per kilogram U, for specific power = 28 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU					
	25	30	35	40	45	50
1.0	7.559	8.390	9.055	9.776	10.400	11.120
1.4	5.593	6.273	6.836	7.441	7.978	8.593
2.0	3.900	4.432	4.894	5.385	5.838	6.346
2.8	2.641	3.054	3.435	3.835	4.220	4.642
4.0	1.724	2.043	2.352	2.675	2.999	3.346
5.0	1.363	1.637	1.911	2.195	2.486	2.793
7.0	1.045	1.271	1.500	1.740	1.987	2.248
10.0	0.873	1.064	1.261	1.465	1.677	1.900
15.0	0.752	0.915	1.083	1.257	1.438	1.627
20.0	0.677	0.823	0.973	1.128	1.289	1.457
25.0	0.619	0.751	0.886	1.027	1.171	1.322
30.0	0.569	0.690	0.813	0.941	1.072	1.208
40.0	0.488	0.590	0.693	0.800	0.909	1.023
50.0	0.424	0.511	0.599	0.689	0.782	0.877
60.0	0.372	0.447	0.523	0.601	0.680	0.762
70.0	0.330	0.396	0.461	0.529	0.598	0.668
80.0	0.295	0.354	0.411	0.471	0.531	0.593
90.0	0.267	0.319	0.371	0.424	0.477	0.531
100.0	0.244	0.291	0.337	0.385	0.432	0.481
110.0	0.225	0.268	0.310	0.352	0.396	0.440

Table 5.7. PWR spent fuel heat generation rates, watts per kilogram U, for specific power = 40 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU					
	25	30	35	40	45	50
1.0	8.946	10.050	10.900	11.820	12.580	13.466
1.4	6.514	7.400	8.111	8.863	9.514	10.254
2.0	4.462	5.129	5.692	6.284	6.821	7.418
2.8	2.947	3.441	3.884	4.346	4.787	5.267
4.0	1.853	2.212	2.554	2.910	3.265	3.647
5.0	1.429	1.728	2.021	2.327	2.639	2.970
7.0	1.067	1.304	1.543	1.793	2.052	2.325
10.0	0.881	1.078	1.278	1.488	1.705	1.936
15.0	0.754	0.921	1.091	1.268	1.452	1.645
20.0	0.678	0.827	0.978	1.136	1.298	1.469
25.0	0.619	0.754	0.890	1.032	1.178	1.331
30.0	0.570	0.693	0.816	0.945	1.077	1.215
40.0	0.488	0.592	0.695	0.803	0.912	1.026
50.0	0.423	0.512	0.599	0.691	0.783	0.879
60.0	0.371	0.448	0.522	0.601	0.680	0.762
70.0	0.329	0.396	0.461	0.529	0.597	0.668
80.0	0.294	0.353	0.410	0.470	0.530	0.592
90.0	0.266	0.319	0.369	0.422	0.475	0.530
100.0	0.243	0.290	0.336	0.383	0.430	0.479
110.0	0.224	0.267	0.308	0.351	0.393	0.437

Table 5.8 PWR enrichments for burnups in tables

Fuel burnup, MWd/kgU	Average initial enrichment, wt % U-235
25	2.4
30	2.8
35	3.2
40	3.6
45	3.9
50	4.2

Compute p_{tab}, the heat generation rate, at P_{ave}, B_{tot}, and T_c, by proper interpolation between the tabulated values of heat rates at the parameter limits of Eqs. (5) through (7). A linear interpolation should be used between heat rates for either burnup or specific power interpolations. In computing the heat rate at T_c, the interpolation should be logarithmic in heat rate and linear in cooling time. Specifically, the interpolation formulae for interpolating in specific power, burnup, and cooling time, respectively, are

$$p = p_L + \frac{p_H - p_L}{P_H - P_L} (P_{ave} - P_L) \ , \tag{5}$$

$$p = p_L + \frac{p_H - p_L}{B_H - B_L} (B_{tot} - B_L) \ , \tag{6}$$

$$p = p_L \exp\left[\frac{\ln(p_H / p_L)}{T_H - T_L} (T_c - T_L) \right] \ , \tag{7}$$

where p_L and p_H represent the tabulated or interpolated heat rates at the appropriate parameter limits corresponding to the L and H index. If applied in the sequence given above, Eq. (5) would need to be used four times to obtain p values that correspond to B_L and B_H at values of T_L and T_H. A mini-table of four p values at P_{ave} is now available to interpolate on burnup and cooling time. Equation (6) would then be applied to obtain two values of p at T_L and T_H. One final interpolation of these two p values (at P_{ave} and B_{tot}) using Eq. (7) is needed to calculate the final p_{tab} value corresponding to P_{ave}, B_{tot}, and T_c. The optional Lagrangian interpolation scheme offered by the LWRARC code is also considered an acceptable method for interpolating the decay heat data, but is not discussed in this section.

If P_{ave} or B_{tot} falls below the minimum table value range, the minimum table specific power or burnup, respectively, may be used conservatively. If P_{ave} exceeds the maximum table value, the table with the maximum specific power (Table 5.3 for BWR fuel and Table 5.7 for PWR fuel) may be used in addition to the adjustment factor, f_p, described in Sect. 5.3.2. The tables should not be applied if B_{tot} exceeds the maximum burnup in the tables, or if T_c is less than the minimum (1 year) or exceeds the maximum (110 years) cooling time of the tables.

5.2.2 The Short Cooling Time Factors f_7 and f'_7

The heat rates presented in Tables 5.1–5.3 and Tables 5.5–5.7 were computed from operating histories in which a constant specific power and an uptime of 80% of the cycle time were applied. Expected variations from these assumptions cause only minor changes (<1%) in decay heat rates beyond approximately 7 years of cooling. However, if the specific power near the end of the operating history is significantly different than the average specific power, P_{ave}, then p_{tab} needs to be adjusted if $T_c \leq 7$. The ratios P_e/P_{ave} and $P_{e-1}/P_{ave,e-1}$ are used, respectively, to determine the adjustment factors f_7 and f'_7. The factors reduce the heat rate p_{tab} if the corresponding ratio is less than 1 and increase the heat rate p_{tab} if the corresponding ratio is greater than 1. The formulae for the factors are defined below.

$$\begin{array}{ll} f_7 = 1 & \text{when } T_c > 7 \text{ years or } e \\ & \quad \text{(i.e., 1 cycle only)} \\ f_7 = 1 + 0.35R/\sqrt{T_c} & \text{when } 0 \leq R \leq 0.3 , \\ f_7 = 1 + 0.25R/T_c & \text{when } -0.3 \leq R < 0 , \\ f_7 = 1 - 0.075/T_c & \text{when } R < -0.3 , \end{array} \tag{8}$$

where

$$R = \frac{P_e}{P_{ave}} - 1 \ . \tag{9}$$

$$\begin{array}{ll} f'_7 = 1 & \text{when } T_c > 7 \text{ years (} \\ & \quad e < 3 , \\ f'_7 = 1 + 0.10R'/\sqrt{T_c} & \text{when } 0 \leq R' \leq 0.6 , \\ f'_7 = 1 + 0.08R'/T_c & \text{when } -0.5 \leq R' < (\\ f'_7 = 1 - 0.04/T_c & \text{when } R' < -0.5 , \end{array} \tag{10}$$

where

$$R' = \frac{P_{e-1}}{P_{ave,e-1}} - 1 \ . \tag{11}$$

It is recommended **not** to use the decay heat values of this report if any of the following conditions occur:

$$\text{if } T_c \leq 10 \text{ years and } P_e/P_{ave} > 1.3,$$
$$\text{if } 10 \text{ years} < T_c \leq 15 \text{ years and } P_e/P_{ave} > 1.7,$$
$$\text{if } T_c \leq 10 \text{ years and } P_{e-1}/P_{ave,e-1} > 1.6. \qquad (12)$$

Although it is safe to use the procedures herein, the heat rate values for p_{final} may be excessively high when

$$T_c \leq 7 \text{ years and } P_e/P_{ave} < 0.6,$$
$$T_c \leq 7 \text{ years and } P_{e-1}/P_{ave,e-1} < 0.4. \qquad (13)$$

5.2.3 The Excess Power Adjustment Factor f_p

The maximum specific power, P_{max}, used to generate the data in Tables 5.1–5.3 and Tables 5.5–5.7 is 40 kW/kgU for a PWR and 30 kW/kgU for a BWR. If P_{ave}, the average cumulative specific power, is more than 35% higher than P_{max} (i.e., 54 kW/kgU for PWR fuel and 40.5 kW/kgU for BWR fuel), then the guide should not be used. When $1 < P_{ave}/P_{max} < 1.35$, the guide can still be used, but an excess power adjustment factor, f_p, must be applied. The excess power adjustment factor is

$$f_p = \sqrt{P_{ave}/P_{\max}}\,. \qquad (14)$$

For P_{ave} \# P_{max}, $f_p = 1$.

5.2.4 The Enrichment Factor f_e

The decay heat rates of Tables 5.1–5.3 and Tables 5.5–5.7 were calculated using initial enrichments of Tables 5.4 and 5.8. The enrichment factor f_e is used to adjust the value p_{tab} for the actual initial enrichment of the assembly E_s. To calculate f_e, the data in Tables 5.4 (BWR) or 5.8 (PWR) should be interpolated linearly to obtain the enrichment value E_{tab} that corresponds to the assembly burnup, B_{tot}. If $E_s/E_{tab} < 0.6$, it is recommended **not** to use the guide. Otherwise, set the enrichment factor as follows:

$$f_e = 1 + 0.01[a + b(T_c - d)][1 - E_s/E_{tab}]$$

$$\text{when } E_s/E_{tab} \; \# \; 1.5$$

$$f_e = 1 \quad 0.005\,[a + b(T_c - d)]$$

$$\text{when } E_s/E_{tab} > 1.5 \qquad (15)$$

where the parameters a, b, and d vary with reactor type, E_s, E_{tab}, and T_c. These variables are defined in Tables 5.9 and 5.10.

5.2.5 Safety Factor S

Before obtaining the final heat rate p_{final}, an appropriate estimate of a percentage safety factor S must be determined. Evaluations of uncertainties performed as part of this project indicate the safety factor should vary with burnup and cooling time.

For BWR assemblies:

$$S = 6.4 + 0.15\,(B_{tot} \quad 20) + 0.044\,(T_c \quad 1)\,. \qquad (16)$$

For PWR assemblies:

$$S = 6.2 + 0.06\,(B_{tot} \quad 25) + 0.050\,(T_c \quad 1)\,. \qquad (17)$$

The purpose of deriving spent fuel heat generation rates is usually to apply the heat rates in the computation of the temperatures for storage systems. A preferred engineering practice may be to calculate the temperatures prior to application of a final safety factor. This practice is acceptable if S is accounted for in the more comprehensive safety factors applied to the calculated temperatures.

5.2.6 Final Heat Generation Rate Evaluation

The equation for converting p_{tab}, determined in Sect. 5.2.1, to the final heat generation rate of the assembly, is

$$p_{final} = (1 + 0.01S)\, f_7\, f\text{N}_7\, f_p\, f_e\, p_{tab}\,, \qquad (18)$$

where f_7, $f\text{N}_7$, f_p, f_e, and S are determined by the procedures given in Sects. 5.2.2 through 5.2.5.

5.3 Acceptability and Limits of the Guide

Inherent difficulties arise in attempting to prepare a heat rate guide that has appropriate safety factors, is not excessively conservative, is easy to use, and applies to all commercial reactor spent fuel assemblies. In the endeavor to increase the value of the guide an effort was made to ensure that safe but not overly conservative heat rates were computed. The procedures and data recommended in the guide should be appropriate for the mainstream of power reactor operations with only minor limitations in the range of applicability.

Table 5.9 Enrichment factor parameter values for BWR assemblies

Parameter in Eq. (15)	Parameter value			
	$E_s / E_{tab} < 1$		$E_s / E_{tab} > 1$	
	$1 \leq T_c \leq 40$	$T_c > 40$	$1 \leq T_c \leq 15$	$T_c > 15$
a	5.7	5.7	0.6	0.6
b	0.525	0.184	0.72	0.06
d	40	40	15	15

Table 5.10 Enrichment factor parameter values for PWR assemblies

Parameter in Eq. (15)	Parameter value			
	$E_s / E_{tab} < 1$		$E_s / E_{tab} > 1$	
	$1 \leq T_c \leq 40$	$T_c > 40$	$1 \leq T_c \leq 20$	$T_c > 20$
a	4.8	4.8	1.8	1.8
b	0.6	0.133	0.51	0.033
d	40	40	20	20

In general, the guide should not be applied outside the ranges of the parameters of Tables 5.1 through 5.8. These restrictions, in addition to certain limits on adjustment factors, are given in the text. The major table limits are summarized in Table 5.11.

In using the guide, the range in cooling time, T_c, and the upper limit on burnup, B_{tot} should never be extended. An adjustment factor, f_p, can be applied if the specific power, P_{ave}, does not exceed the maximum value of the tables by more than 35%. Thus, if P_{ave} is greater than 54 kW/kgU for PWR fuel or 40.5 kW/kgU for a BWR fuel, then the guide should not be applied. The minimum table value of specific power or burnup can be used for values below the table range; however, if the real value is considerably less than the table minimum, the heat rate derived can be excessively conservative.

In preparing generic depletion/decay analyses for use in specific applications, the most difficult condition to model is the power operating history of the assembly. Although a power history variation (other than the most extreme) does not significantly change the decay heat rate after a cooling time of approximately 7 years, it can have significant influence on the results in the first few years. Cooling time adjustment factors, f_7 and f'_7, are applied to correct for variations in power history from that used in the generation of the tables. For example, the heat rate at 1 year is increased substantially if the power in the last cycle is twice the average power of the assembly. The limits in Eqs. (12) and (13) on ratios of cycle to average specific power are required, first, to derive cooling time adjustment factors that are valid and, second, to exclude cases that are extremely atypical. Although these limits were determined so that the factors are safe, a reasonable degree of discretion should be used in the considerations of atypical assemblies— particularly with regard to their power histories.

Another variable that requires attention is the ^{59}Co content of the clad and structural materials. Cobalt-59 is partly transformed to ^{60}Co in the reactor and subsequently contributes to the decay heat rate. The ^{59}Co content used in deriving the tables here should apply only to assemblies containing Zircaloy-clad fuel pins. The ^{60}Co contribution can become excessive for ^{59}Co contents found in stainless-steel clad. Thus, the use of the guide for stainless-steel-clad assemblies should be limited only to cooling times that exceed 20 years. Because ^{60}Co has a 5.27-y half-life, the heat rate contribution from ^{60}Co is reduced by the factor of 13.9 in 20 years.

Table 5.11 Parameter ranges for applicability of the proposed regulatory guide

Parameter	BWR	PWR
$T_c(y)$	1 110	1 110
B_{tot} (MWd/kgU)	20 45	25 50
P_{ave} (kW/kgU)	12 30	18 40

In addition to parameters used here, 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 indicated that the expected differences (approximately 0.2% heat rate change per 1% change in water density, during first 30-year decay) are not sufficient to require additional corrections. The PWR decay heat rates in the tables were calculated for fuel assemblies containing water holes. Computed decay heat rates for assemblies containing burnable poison rods (BPRs) did not change significantly (<1% during first 30-year decay) from fuel assemblies containing water holes.

Several conditions were considered in deriving the safety factors [Eqs. (16) and (17)] that were developed for use in the guide. Partial uncertainties in the heat generation rates were computed for selected cases (see Sect. 6.6.1) applying the known standard deviations of half-lives, Q-values, and fission yields of all the fission product nuclides that have a significant contribution to decay heat rates. This calculation did not account for uncertainties in contributions produced by the neutron absorption in nuclides in the reactor flux (see Sect. 6.6.2), or from variations in other parameters (see Sect. 6.6.3). In addition to the standard deviations in neutron cross sections, much of the uncertainty from neutron absorption was found to derive upon approximations in the model used in the depletion analysis. In developing the safety factors, these more indirect uncertainties were determined from comparisons of the calculated total or individual nuclide decay heat rates with those determined by independent computational methods, as well as heat rate measurements obtained for a variety of reactor spent fuel assemblies. Note from the equations that the safety factors increase with both burnup and cooling time. This increase in the safety factor is a result of the increased importance of the actinides to the decay heat with increased burnup and cooling time, together with the larger uncertainty in actinide predicitons caused by model approximations and limited experimental data.

Whenever there is a unique difference in either the design or operating conditions of a spent-fuel assembly that is more extreme than that accepted here, another well-qualified method of analysis that accounts for the difference should be used.

5.4 Glossary of Terms and Units Used in Guide

B_e - burnup in last cycle, MWd/kgU
B_{e-1} - burnup in next-to-last cycle, MWd/kgU
B_i - fuel burnup increase for cycle i, MWd/kgU
B_{tot} - total burnup of discharged fuel, MWd/kgU
E_s - initial fuel enrichment, wt % ^{235}U
P - specific power of fuel as in Eqs. (2) and (3), kW/kgU
P_{ave} - average cumulative specific power during 80% uptime, kW/kgU
$P_{ave,e-1}$ - average cumulative specific power (at 80%) through cycle *e-1*, the next-to-last cycle
P_e - fuel specific power during the last cycle *e*
P_{e-1} - fuel specific power during cycle *e-1*, the next-to-last cycle
S - percentage safety factor applied to decay heat rates, p_{tab}
T_c - cooling time of an assembly, years
T_e - cycle time of last cycle before discharge, days
T_{e-1} - cycle time of next-to-last cycle, days
T_i - cycle time of ith reactor operating cycle including downtime for all but last cycle of assembly history, days
T_{res} - reactor residence time of assembly, from first loading to discharge, days
f_7 - last-cycle short cooling time modification factor
f'_7 - next-to-last cycle short cooling time factor
f_e - ^{235}U initial enrichment modification factor
f_p - excess power adjustment factor
p - heat generation rate of spent fuel assembly, W/kgU

6 DISCUSSION OF THE PROPOSED PROCEDURE

The purpose of this section is to describe and discuss the work performed to develop the procedures and formulae presented in Sect. 5. Section 6.1 discusses the investigation to determine the sensitivity of the decay heat rate to variations in the reactor operating history. Section 6.2 discusses and demonstrates the accuracy of the techniques recommended for interpolation of the data in Tables 5.1–5.3 and Tables 5.5–5.7. The remaining sections provide the basis for the adjustment factor and safety factor formulae.

6.1 Variations in Operating History

The decay heat rates presented in Tables 5.1–5.3 and Tables 5.5–5.7 were calculated by applying different total burnups and average specific powers to a "standard" operating history profile. The distribution of uptime and downtime in the operating history of an actual assembly could be considerably different from that used in the calculations. The purpose of this section is to present the work performed to determine which types of variations have significant effects upon the decay heat rates of reactor spent fuel.

As noted in Sect. 4, the standard power history profile used to generate the tabulated heat rate data had either three, four, or five power cycles. Commonly, cycle time refers only to the time between cycle startup of a loaded or a partially reloaded core and shutdown for another partial core reloading. In order to simplify operating histories, the definition in the procedure of Sect. 5 would extend the cycle time, except for the last cycle of an assembly, to include the downtime for reloading.

The first cycle of the standard profile had a downtime (i.e., nonpower operation time) of 20% at the middle of the cycle. The second cycle (and optionally third and fourth cycles) had a 10% downtime both near the midpoint and at the end of the cycle, thus producing an 80% uptime (i.e., power operation time). The last cycle had an uptime of 100%. The power was held constant during all the time the reactor was in operation and the burnups in each cycle were equal.

Note again the objective of this analysis was to determine if normal differences in the actual operating experiences of fuel assemblies from that assumed in the standard case have significant effects upon their decay heat rates. In essence, comparisons in the results of these differences are needed to support the premise, mathematically stated in Eq. (3) of Sect. 5.1.4, that the specific power to be used in determining the heat rate from the tabulated data, can be properly defined from only the total burnup and cycle times. The three-cycle PWR operating histories shown in Figure 6.1 were developed to investigate operating history changes under the conditions that the total burnup, the cycle times, and the average power are unchanged.

All the cases illustrated in Figure 6.1 have the same total burnup of 30 MWd/kgU and cycle times of 400, 400, and 320 days. Thus, the average specific power of 31.25 kW/kgU [computed from Eq. (3) of Sect. 5.1.4] is the same for all of the cases. The "standard case" of Figure 6.1(A) has the general operating history described above and was used for all the three-cycle cases used in producing Tables 5.1–5.3 and 5.5–5.7.

If the total burnup, cycle times, and average power are unchanged, the only possible changes in the operating history of Figure 6.1(A) pertain to the uptime and downtime during a cycle, the distribution of power within a cycle (accounting for within-cycle changes) and the burnup distribution to the various cycles (between cycle changes).

The change in downtime during a cycle is illustrated in Figure 6.1(B) and (C). In order to keep the cycle burnups the same as the standard case [Figure 6.1(A)] and in order to reduce the uptime by the factor 7/8, a power equal to $8\overline{P}/7$ is required in the case shown in Figure 6.1(B). Similarly, in Figure 6.1(C), the illustrated uptime change by the factor 9/8 requires a power of $8\overline{P}/9$. Table 6.1 gives a list of the decay heat rates for several cooling times as calculated by ORIGEN-S using the basic LWR ORIGEN-S library. The table also lists the percentage differences in decay heat rates as compared with results for the standard case. As a general criterion, consider here that a 1% difference can adequately be covered in a safety factor. The conclusion from studying the results from operating histories A through C is that differences in the cycle downtime (which is a very common difference) produce only small conservative changes in decay heat rate results.

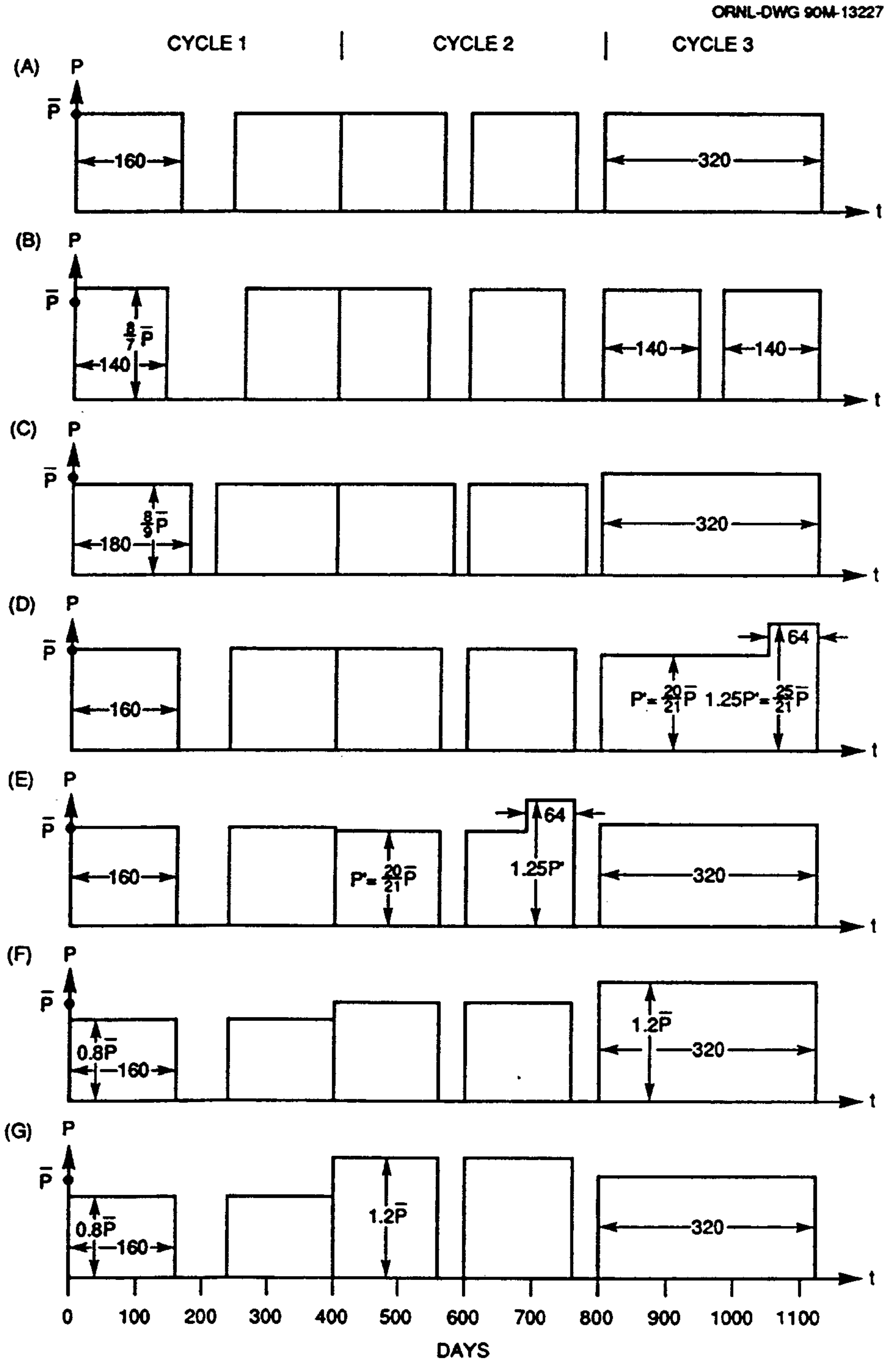


Figure 6.1 Diagrammatic illustration of operating history variations having same total burnup and cycle times: (A) standard case, 20% downtime; (B) 30% downtime; (C) 10% downtime; (D) power increase in an end interval of last cycle; (E) power increase in an end interval of middle cycle; (F) power increase during all of last cycle; and (G) power increase during all of middle cycle

Table 6.1 Comparison of heat rates from operating history variations

	Decay heat rates (W/kgU) Cooling time, years						% difference[a] of (A) and (X) Cooling time, years					
Case	1	2	4	10	30	110	1	2	4	10	30	110
(A) Std case 8.888	4.644	2.095	1.069	0.6909	0.2675							
(B) 30% downtime	8.824	4.601	2.079	1.064	0.6882	0.2666	–0.7	–0.9	–0.8	–0.5	–0.4	–0.3
(C) 10% downtime	8.879	4.635	2.087	1.063	0.6872	0.2661	–0.1	–0.2	–0.4	–0.6	–0.5	–0.5
(D) $\Delta P > P_e$	8.995	4.674	2.102	1.069	0.6911	0.2676	1.2	0.6	0.3	0.0	0.1	0.1
(E) $\Delta P > P_{e-1}$	8.912	4.656	2.098	1.069	0.6908	0.2675	0.3	0.3	0.2	0.0	<0.1	0.0
(F) $P_e = 1.2\ \overline{P}$	**9.567**	**4.917**	**2.159**	**1.071**	**0.6915**	**0.2686**	**7.6**	**5.9**	**3.1**	**0.2**	**0.1**	**0.4**
(G) $P_{e-1} = 1.2\ \overline{P}$	**9.040**	**4.716**	**2.111**	**1.069**	**0.6909**	**0.2680**	**1.8**	**1.6**	**0.8**	**0.0**	**0.0**	**0.2**

[a]The percentage differences of case (A) heat rate, H_A, from case (X) heat rate, H_X: 100% $(H_X - H_A)/H_X$.

The changes in power distribution within a cycle were investigated using the operating histories shown in Fig. 6.1(D) and (E) where the last and middle cycles, respectively, differ from the standard case in Figure 6.1(A). The power during the last 20% of the cycle uptime is 1.25 times that of the first 80%. The average cycle power is unchanged from that of Figure 6.1(A). Table 6.1 shows the percentage differences in heat rate (from the standard case) for these two cases to be less than 1% in magnitude for all comparisons except the 1-year cooling time for case D. The heat rate difference for case D at 1-year cooling is 1.2% greater than the corresponding heat rates from case A. Using a 1% difference as a safe criterion, it would appear that the procedure of Sect. 5 should not be used for cooling times less than 2 years if there is a 25% increase in the power in the last one-fifth of the last cycle operation. Typically, however, the specific power decreases (or at most remains constant) during the end interval of a cycle. An example showing the operating history of the first seven cycles of the Cooper Nuclear Station BWR[27] is presented in Figure 6.2. This case with the large power increase is used for amplifying the effects of small increases only and should actually be considered to be an atypical case, outside the mainstream of commercial power operations.

It is a fairly common occurrence for operating histories that produce the same total burnup to have differences in the average power or burnup of a cycle. Examples of a last-cycle power increase and decrease from the average power are given in Figures 6.3 and 6.4, respectively, for two Cooper Nuclear Station spent fuel assemblies.[15] Changes in the average power or burnup of a cycle were studied using the operating histories shown in Figure 6.1(F) and (G). In Figure 6.1(F) the first-cycle power and burnup are decreased by 20% and the last-cycle power and burnup are increased by 20% from the similar cycle data of the standard case. In the final case, Figure 6.1(G), the increased values are in the next-to-last cycle instead of the last cycle. The first five cases (A–E) of Figure 6.1 have equal burnups within each cycle, whereas the final two cases have cycle burnups that are not equal. For cases A–E, burnup was redistributed from the standard case within the same cycle only. As might be expected, the more significant redistribution of burnup from one cycle to another causes greater differences in the decay heat rates than observed for cases A–D. The magnitudes of the largest differences from the standard case for cases F and G, listed in Table 6.1, are 7.6% and 1.8%, respectively. These magnitudes are significantly nonconservative. However, the differences are considerably reduced after 10-year cooling time. This smaller reduction at longer decay times is caused by the predominance of isotopes with half-lives that are long with respect to cycle times. Thus, from the results of these comparisons, it was concluded that a proper adjustment factor needs to be applied to the tabulated data for short cooling times (e.g., the first 7 years) to adjust for increased specific power in the last two cycles. This short cooling time adjustment factor was presented in Sect. 5.2.2 and will be discussed further in Sect. 6.3.

In summarizing the comparisons of Table 6.1, it is clear that differences in the distribution among cycles of the burnup between operating histories has a significantly greater effect on heat rate than the other two types of changes which were studied. The basic difference in the operating history changes illustrated in Figure 6.1 is that burnup is redistributed (from the burnup of the standard case) within a cycle in Figures 6.1(B)–(E), whereas a larger magnitude of burnup is moved to an entirely different cycle in Figures 6.1(F)– (G). The quantity or the percentage difference in the decay heat rate is increased as the amount of burnup moved to a later time is increased and as the shift in time becomes greater. Thus, it is concluded that the effects from within-cycle changes [Figures 6.1(B)–(E)] produce differences from the standard case that are satisfactory (i.e., <1% different), while a proper short cooling-time factor is required for conditions where the cycle burnups [Figures 6.1(F)–(G)] vary from the standard case. It is further concluded that Eqs. (1) through (4) are appropriate definitions to allow accurate use of the decay heat rates in Tables 5.1–5.3 and 5.5–5.7.

6.2 Interpolation Accuracy

The effective tabulated heat generation rate, p_{tab}, is derived by interpolation of tabulated heat rates for the given parameters P_{ave}, B_{tot}, and T_c. A linear interpolation is used between heat rates for either burnup or power. The decay time interpolation is logarithmic in heat rate and linear in cooling time. Estimated magnitudes of the interpolation error are presented in this section.

The initial effort of this study was to determine if the tabulated decay heat rates were computed for intervals of burnup, power, and cooling time that are sufficiently fine to produce an acceptable accuracy with areasonable interpolation scheme. Two different methods could be used to estimate the interpolation errors. One method requires execution of a number of SAS2H/ORIGEN-S cases at numerous intermediate parameter values to produce results for comparisons. However, the number of SAS2H/ORIGEN-S cases needed would be too numerous

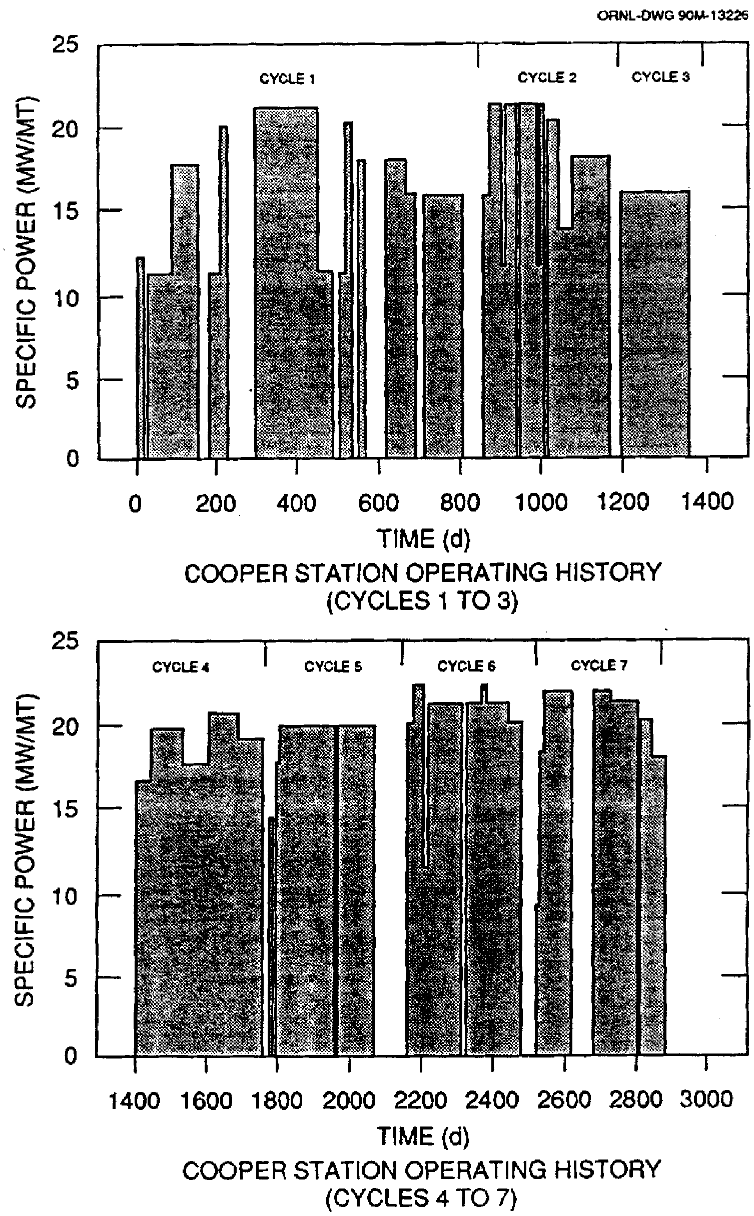


Figure 6.2 Cooper Nuclear Station operating history (from Ref. 27)

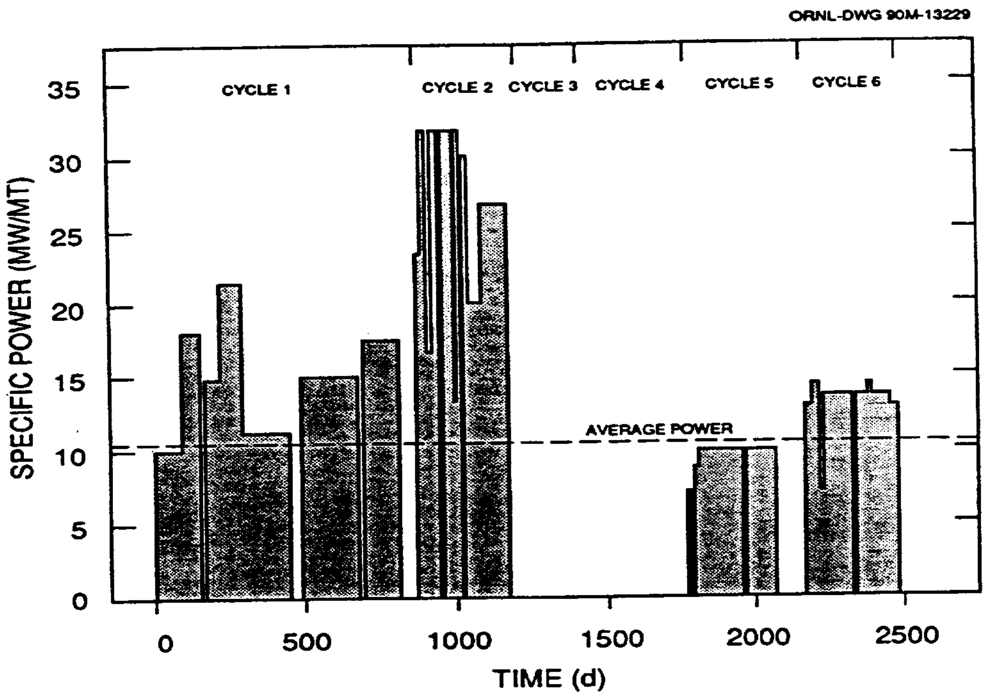


Figure 6.3 Cooper Nuclear Station assembly CZ515 operating history (from Ref. 15).

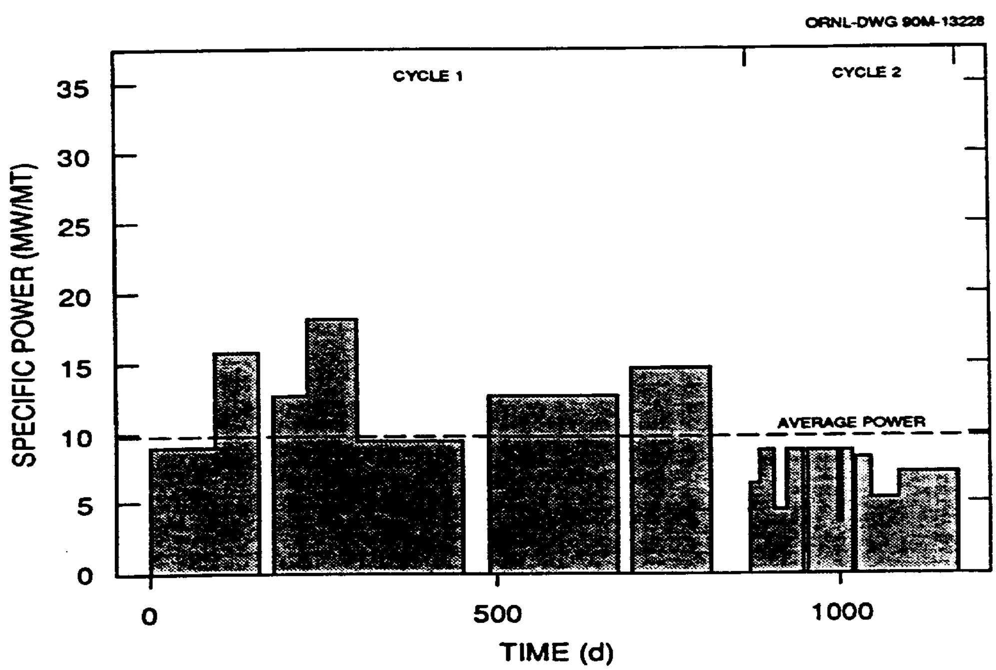


Figure 6.4 Cooper Nuclear Station assembly CZ102 operating history (from Ref. 15).

to sufficiently cover the parameter ranges. The other method uses a polynomial fit to the tabulated heat rates as a function of one of the variables with the other two parameters held constant. The polynomial fit is applied to derive intermediate results for comparison. This latter method was chosen because it was both easy to implement and appeared adequate to assess the accuracy of the simpler interpolation schemes proposed in Sect. 5. It was necessary to limit the equation to a quadratic fit for the specific powers since only three different powers were used. A quadratic fit was also considered adequate for the burnups because the tabulated heat rates as a function of the burnup exhibited significantly less variation than heat rates as a function of power. The errors vary from a zero error when exactly at a parameter value to the maximum error near the midpoint between two values. Interpolated values using Eqs. (5) through (7) were compared with those computed from the quadratic equation at ten equal intervals between adjacent values in Tables 5.1–5.3 and Tables 5.5–5.7. The plot of the logarithm of decay heat rates as a function of cooling time is a curve that is concave upward. Thus, the interpolated value between two cooling times would be greater than the corresponding value on the log curve.

The nonconservative percentage differences between the interpolated and the more correctly estimated values of decay heat rates are shown in Table 6.2. It can be seen that after approximately 10 years, the error ($\leq$0.3%) is small. The maximum percentage difference of 1.1% at 1 year is considered to be acceptable.

6.3 Discussion of the Short Cooling-Time Factors

The standard cases producing the tabulated heat rates (i.e., Tables 5.1–5.3 and Tables 5.5–5.7) for the proposed guide use operating histories with constant specific power during the entire operational uptime as shown in Figure 6.1 (A). Differences in heat rates caused by operating history variations of the types discussed in Sect. 6.1 and listed in Table 6.1 appear to be satisfactory at 10 years and longer cooling times. However, unacceptably large (>1%) differences in the results were produced over a shorter cooling-time range for cases in which large quantities of burnup and power were moved from one cycle to another, as in Figure 6.1 (F) and (G). It was noted in Sect. 6.1 that short cooling-time adjustment factors would be required to account for these larger redistributions in burnup.

The equations and limits of the short cooling-time factors, f_7 and f'_7, are specified in Eqs. (8) through (13) of Sect. 5.2.2. The factors are dependent on the variables T_c, R, and R'. The parameters R and R', respectively, are the fractional changes in the last and next-to-last cycle specific power from the average specific power of the fuel.

The short cooling-time factors were derived and tested in 22 cases using widely different conditions. The cases were computed for no downtime because only the effect of moving burnup between cycles was analyzed. Also, it was expected that there would be insignificant differences between PWRs and BWRs in regard to operating history effects. The reason for this expectation is that at shorter cooling times there is significantly less decay heat rate from the neutron absorption-dependent actinides (which vary considerably with reactor type) than from the fission products (which are essentially dependent only upon fission yields).[4,18] Also, the results from similar sets of cases using both a BWR and a PWR library indicated that the type of reactor library was not significant to these short cooling-time analyses. Thus, the PWR library was used in all of the other cases of this sensitivity study.

Table 6.3 contains an evaluation of the ability of the short cooling-time factors to adjust the tabulated data properly for cases having large fractions of redistributed burnup. There are 22 cases, A through V, listed in the table. There is one reference case for each set of cases having similar residence time (number of cycles times cycle length), total burnup, and average specific power. Each reference case has a constant specific power, similar (except for downtime) to the cases that were used to produce the tabulated data. The other cases listed are compared with its defined reference case. The power history diagram shows for each cycle the ratio of power to the average assembly power. From the diagram information and Eqs. (9) and (11) of Sect. 5.2.2, the values listed under R and R' were calculated. Then, R and R' were used in Eqs. (8) and (10) to compute f_7 and f'_7, respectively. The heat rates of the reference case, corrected by factors f_7 and f'_7, were compared with heat rates from the case computed by ORIGEN-S for the operating history shown. Comparisons of the percentage difference between the adjusted (reference case) heat rate and the computed heat rate value (of case X) are shown at four cooling times in Table 6.3. Additional comparisons of uncorrected heat rates are listed as percentage differences at 8 and 10 years.

Table 6.2 Evaluation of accuracy in table interpolations

	Nonconservative % differences[a] — Independent variables of the decay heat rate				
	Specific power		Burnup		Cooling time
Cooling time, d	BWR	PWR	BWR	PWR	BWR or PWR
1	1.1	0.9	0.4	0.3	<0
2	1.1	0.9	0.4	0.2	<0
4	0.7	0.6	0.1	0.1	<0
7	0.4	0.3	<0.1	<0.1	<0
10	0.3	0.2	<0.1	<0.1	<0
20	0.2	0.1	<0.1	<0.1	<0
40	0.1	0.1	<0.1	<0.1	<0
70	<0.1	<0.1	0.1	0.1	<0
110	<0.1	<0.1	0.1	0.1	<0

[a](Correct/interpolated − 1)100%.

Cases for two, three, four, and five cycles are shown in Table 6.3. Case A is a reference two-cycle case. Note that for a two-cycle case, the factor f'_7 = 1 [see Eq. (10) of Sect. 5.2.2], and so R' does not need to be used. The results computed for case B, in which the last cycle power was 1.2 times the average, indicated the corrected reference case heat rates to be conservative by 0.3 to 2.0%. The value of R in case C is increased to 0.3, the limit specified in Eq. (12) of Sect. 5.2.2 for cooling times less than 10 years. The percentage differences of 0.4 to 3.1% for this case are an increase from those in case B by a scale factor very similar to the increase in R, as would be expected. Note that throughout the cases listed in Table 6.3, numerous changes are made in the number of cycles, cycle length, total burnup, and average specific power. In some cases, either R, R', or both are made positive to demonstrate increasing power. Also, cases having negative values of R or R' test decreasing powers that cause the corresponding f_7 or f'_7 to be less than unity. Cases E and F were computed using a BWR library, whereas similar cases G and H used a PWR library. The percentage differences listed for cases F and H are sufficiently close to indicate that the type of reactor library does not appear to be significant. Note that R, in most of the cases, is set to either positive or negative 0.2. The differences for these cases can be scaled up to the magnitude for R of 0.3, the limit. Extremes in increases or decreases of specific power in both the last and next-to-last cycles, in addition to other operating history parameter variations, appear to be adequately covered in the 22 cases. Even an out-of-reactor cycle case is shown in case Q, although it is more conservative than that obtained by dividing the first cycle burnup between the first two cycles.

In summary, the results shown in Table 6.3 demonstrate that the short cooling-time factors f_7 and f'_7 ensure conservative decay heat values are obtained by the new procedure. The cases used to develop Table 6.3 span a burnup range of 25 to 49.5 MWd/kgU, cycle times from 300 to 450 days, and average powers from 24 to 50 kW/kgU. Only three of the corrected values were nonconservative (5% of the corrections) and the largest in magnitude was –0.2%. The largest nonconservative discrepancy beyond the 7-year cooling time when f_7 and f'_7 are applicable was –0.7% (at 8-year cooling). Although a small nonconservative error (<0.5%) may be produced in computing f_7 and f'_7, it can be easily incorporated into the safety factor. Table 6.3 shows that typically the factors f_7 and f'_7 yield conservative errors of 3 to 4% maximum and 1 to 2% on the average.

6.4 Discussion of the Excess Power Adjustment Factor

The maximum specific powers of the decay heat rates provided in Tables 5.1–5.3 and Tables 5.5–5.7 are probably greater than the average power derived by Eq. (3) for all current U.S. commercial reactors. Although it is not expected that the average specific power will exceed the maximum tabulated values of 30 kW/kgU for the BWR and 40 kW/kgU for the PWR, it was determined that a

Table 6.3 Evaluation of heat rates after adjustments for extreme power history changes

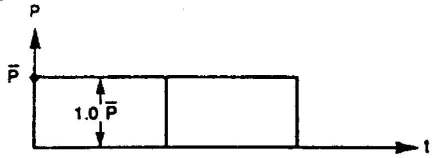
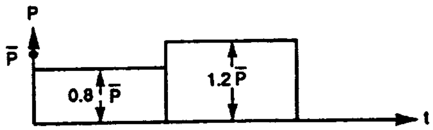
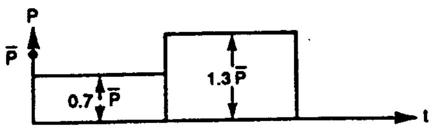
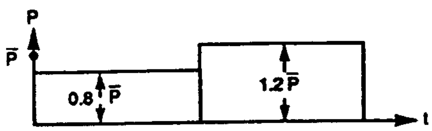
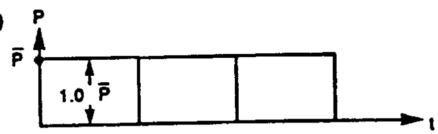
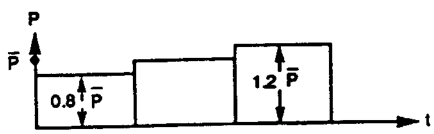
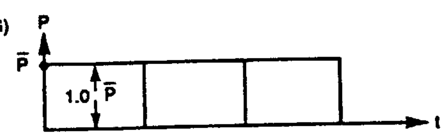
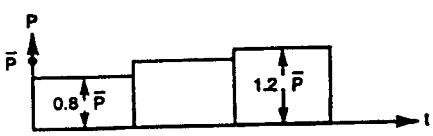
Case	Power history diagram, P vs t	T_{cyc}, days	B_{tot}, $\frac{MWd}{kgU}$	$\bar{P}$, $\frac{kW}{kgU}$	R	R'	Ref. case	% difference[a] at T_c(years) 1	3	5	7	8	10
(A)		350	28	40	0.0	0.0	(A)	Reference case					
(B)		350	28	40	0.2	0.0	(A)	0.4	0.3	2.0	2.0	-0.4	-0.1
(C)		350	28	40	0.3	0.0	(A)	0.6	0.4	2.4	3.1	-0.5	-0.2
(D)		450	45	50	0.2	0.0	(L)	0.0	-0.2	1.3	2.1	-0.6	-0.1
(E)		300	25	27.78	0.0	0.0	(E)	Reference case					
(F)		300	25	27.78	0.2	0.111	(E)	1.0	0.8	2.2	2.4	-0.3	-0.2
(G)		300	33	36.67	0.0	0.0	(G)	Reference case					
(H)		300	33	36.67	0.2	0.111	(G)	1.5	0.5	2.2	2.6	-0.6	-0.2

Table 6.3 (continued)

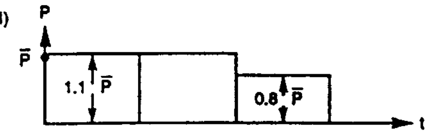
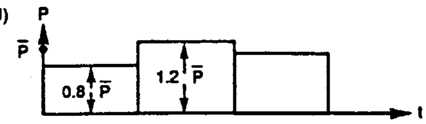
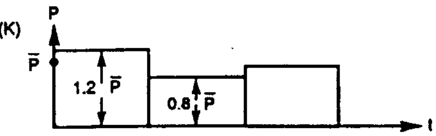
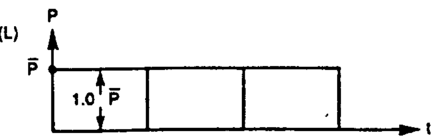
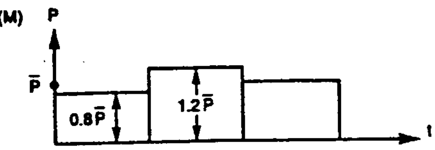
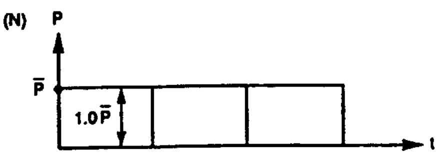
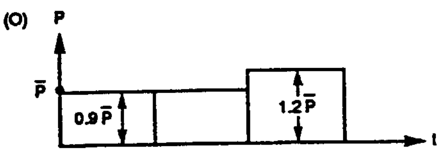
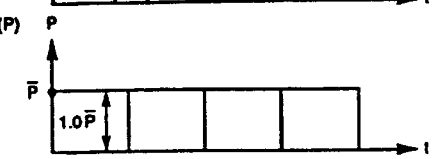
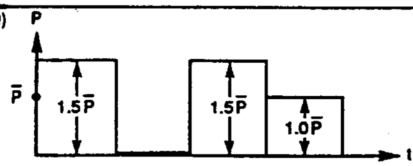
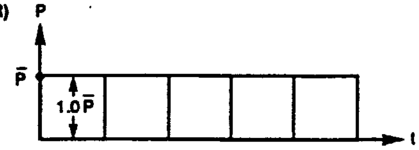
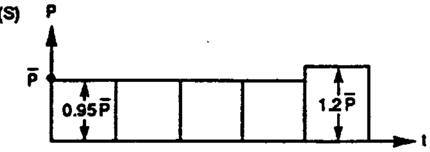
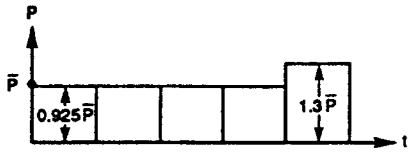
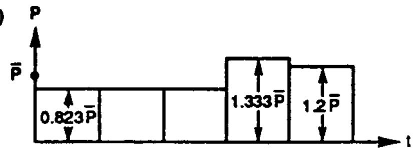
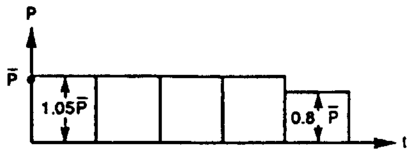
Case	Power history diagram, P vs t	T_{cyc}, days	B_{tot}, $\frac{MWd}{kgU}$	$\bar{P}$, $\frac{kW}{kgU}$	R	R'	Ref. case	% difference[a] at T_c(years) 1	3	5	7	8	10
(I)		300	33	36.67	-0.2	0.0	(G)	1.3	1.7	0.6	-0.1	0.6	-0.2
(J)		300	33	36.67	0.0	0.2	(G)	0.3	0.3	1.3	0.7	0.2	0.0
(K)		300	33	36.67	0.0	-0.2	(G)	0.1	0.6	0.1	0.4	-0.2	-0.1
(L)		300	45	50	0.0	0.0	(L)	Reference case					
(M)		300	45	50	0.0	0.2	(L)	0.2	0.1	0.8	0.6	-0.2	-0.3
(N)		450	49.5	36.67	0.0	0.0	(N)	Reference case					
(O)		450	49.5	36.67	0.2	0.0	(N)	0.0	0.1	1.5	2.1	-0.3	-0.1
(P)		300	33	27.5	0.0	0.0	(P)	Reference case					

Table 6.3 (continued)

Case	Power history diagram, P vs t	T_{cyc}, days	B_{tot}, $\frac{MWd}{kgU}$	$\bar{P}$, $\frac{kW}{kgU}$	R	R'	Ref. case	% difference[a] at T_c(years) 1	3	5	7	8	10
(Q)		300	33	27.5	0.0	0.5	(P)	2.6	1.1	1.5	1.0	-0.2	-0.2
(R)		300	36	24	0.0	0.0	(R)	Reference case					
(S)		300	36	24	0.2	0.0	(R)	0.9	0.8	1.5	2.0	-0.4	-0.2
(T)		300	36	24	0.3	0.0	(R)	1.4	1.2	2.4	2.8	-0.7	-0.3
(U)		300	36	24	0.2	0.4	(R)	1.3	0.7	2.4	3.1	-0.7	-0.3
(V)		300	36	24	-0.2	0.0	(R)	1.2	1.7	0.6	-0.2	0.7	0.3

[a]Percentage difference in decay heat rate between value calculated by ORIGEN-S and adjusted (using f_r and f_r') reference case value: (adjusted reference case/case X value – 1)100%. No adjustment to the reference case value was required for cooling times of 8 and 10 years.

simple adjustment factor f_P could be applied to extend the specific power range. This factor, computed by Eq. (14) of Sect. 5.2.3, is simply the square root of the ratio of the average power of the assembly to the maximum used in computing the tabulated data. A user should be aware that other characteristics of the assembly and operating conditions may force the assembly to be considered as atypical if the excess specific power significantly exceeded the maximum values used to generate the tables.

The formula for the excess power factor was validated by performing select calculations using SAS2H/ ORIGEN-S. The maximum excess power allowed by the procedure of Sect. 5 is 35% greater than the maximum tabulated powers, or 40.5 kW/kgU for the BWR and 54 kW/kgU for the PWR. SAS2H cases were then calculated using these specific powers and the lowest, the highest, and the second from the highest burnups used in determining the tabulated data. The decay heat rates derived using the adjustment factor of Eq. (14), and those computed by SAS2H are listed in Table 6.4. The comparisons show that the adjusted heat rates are conservative in all of the PWR cases and in all cases at cooling times greater than 1 year for the BWR. The nonconservative differences (<1.4%) in the range less than 2-year cooling time for the BWR are judged to be small enough to be appropriately accounted for in the final safety factor.

6.5 Discussion of the Initial Enrichment Factor

The average initial ^{235}U enrichments considered as typical for present and extended burnup reactor fuel are listed in Tables 5.4 and 5.8. These ^{235}U enrichments are selected so that the reactor has sufficient reactivity to maintain criticality throughout the operation that produces the corresponding burnup specified in the tables. Lower enrichments may be insufficient, whereas higher enrichments may be less economical. However, because commercial reactor data exhibit significant variations from the burnup and enrichment sets tabulated and used in computing the standard cases, an enrichment adjustment factor is applied to correct the decay heat rates. This enrichment factor f_e is given by Eq. (15) in Sect. 5.2.4. This section describes the method used in determining the formula for f_e.

In addition to the 18 standard cases computed for each reactor type, ten cases were calculated using different enrichments. At the minimum and maximum burnups and specific powers (i.e., for the parameters in Tables 5.1–5.8) of each type of reactor, SAS2H/ ORIGEN-S cases were computed with all the data unchanged except for an increase and a decrease by one-third in the initial ^{235}U enrichment from that of the standard cases. Also, at a one-third decrease of the initial enrichment, two cases were computed at a middle-value burnup and the two highest specific powers for each reactor. Then, for each of these altered enrichment cases, the maximum percentage heat rate change was used in deriving f_e. The extreme variation of the burnup and specific power in each set of cases appeared to be sufficient to provide a conservative envelope of the percentage heat rate changes needed to produce a conservative formula for f_e. The maximum difference was not always in the same case, but it was always in one of the cases having a maximum or minimum burnup and power. Thus, the two middle burnup cases at the lower enrichments were not computed at the higher enrichments.

Data showing the percentage heat rate changes for cases using a one-third decrease in initial enrichment are presented in Table 6.5. For both the BWR and PWR, the average and maximum percentage changes are given at cooling times from 1 to 110 years. The percentage changes for (f_e – 1) as derived from Eq. (15) and Tables 5.9–5.10 are shown under the column labeled "Equation." Table 6.5 shows that the decrease in initial enrichment causes an increase in the decay heat that is conservatively bounded by the initial enrichment factor of Eq. (15) (one exception is the 0.1% difference at 4 years for the PWR). Similarly, Table 6.6 shows that an increase in initial enrichment decreases the decay heat and that Eq. (15) again provides an adequately conservative estimate of the decay heat change. In summary, Tables 6.5 and 6.6 provide sufficient evidence that the formula for f_e provides an adequately conservative method for adjusting the tabulated decay heat rate for different initial enrichment values.

6.6 Formulation of the Safety Factor Equations

The safety factor, *S*, applied in the final equation for heat generation rate is computed from either Eq. (16) or Eq. (17). The development of the safety factor formulae is presented in this section.

Table 6.4. Excess power adjustment of decay heat, in W/kgU, using Eq. (14) compared with actual SAS2H calculations as percentage differences[a]

Reactor type	Burnup MWd/kgU	Guide procedures	1 year SAS2H	% diff.	Guide procedures	2 years SAS2H	% diff.	Guide procedures	20 years SAS2H	% diff.	Guide procedures	110 years SAS2H	% diff.
BWR	20	7.911E+00	7.716E+00	2.5	3.913E+00	3.706E+00	5.6	6.100E-01	5.218E-01	16.9	2.057E-01	1.755E-01	17.2
BWR	35	1.085E+01	1.093E+01	-0.8	5.785E+00	5.663E+00	2.2	1.118E+00	9.657E-01	15.7	3.486E-01	2.985E-01	16.8
BWR	45	1.244E+01	1.260E+01	-1.3	6.899E+00	6.786E+00	1.7	1.483E+00	1.284E+00	15.5	4.438E-01	3.798E-01	16.9
PWR	25	1.039E+01	1.009E+01	3.1	5.184E+00	4.888E+00	6.1	7.878E-01	6.747E-01	16.8	2.603E-01	2.220E-01	17.2
PWR	40	1.373E+01	1.361E+01	0.9	7.301E+00	7.030E+00	3.9	1.320E+00	1.133E+00	16.5	4.078E-01	3.476E-01	17.3
PWR	50	1.565E+01	1.557E+01	0.5	8.619E+00	8.329E+00	3.5	1.707E+00	1.463E+00	16.7	5.077E-01	4.321E-01	17.5

[a](Guide procedure value/SAS2H calculated value − 1)100%.

Table 6.5 Evaluation of adjustments for decreased initial enrichments

Cooling time, days	% heat rate change[a] for a 1/3 decrease in enrichment: BWR Average	BWR Maximum	BWR Equation	PWR Average	PWR Maximum	PWR Equation
1	6.6	7.6	8.7	6.8	7.7	9.4
2	7.1	8.4	8.6	7.7	8.7	9.2
4	6.1	8.0	8.2	7.1	8.9	8.8
7	3.9	6.5	7.7	5.2	7.4	8.2
15	2.2	4.6	6.3	2.9	5.1	6.6
25	1.6	3.2	4.5	1.9	3.4	4.6
40	1.3	1.9	1.9	1.2	1.6	1.6
60	1.4	2.8	3.1	0.9	2.2	2.5
80	1.9	4.3	4.4	1.1	3.1	3.4
110	2.9	6.2	6.2	1.7	4.7	4.7

[a](Decreased case/standard − 1)100%.

Table 6.6 Evaluation of adjustments for increased initial enrichments

Cooling time, days	% heat rate change[a] for a 1/3 increase in enrichment					
	BWR			PWR		
	Average	Maximum	Equation	Average	Maximum	Equation
1	5.3	3.8	3.6	5.3	4.0	3.8
2	-5.5	-4.5	-3.3	-5.8	-4.8	-3.7
4	-4.4	-3.1	-2.8	-5.0	-4.0	-3.3
7	-2.6	-1.2	-2.1	-3.3	-2.1	-2.8
15	-1.5	-0.2	-0.2	-1.9	-0.7	-1.5
25	-1.4	-0.5	-0.4	-1.5	-0.7	-0.7
40	-1.7	-1.3	-0.7	-1.6	-1.3	-0.8
60	-2.4	-1.4	-1.1	-2.0	-1.2	-1.0
80	-3.2	-1.7	-1.5	-2.7	-1.3	-1.3
110	-4.3	-2.1	-2.0	-3.6	-1.6	-1.6

[a](Increased case/standard – 1)100%.

A large quantity of data is input to a rather complex computational model to compute the tabulated heat rates in the set of 36 standard cases. Also, procedures involving both interpolations and adjustment factors are applied to the tabulated data. An appropriate safety factor to be applied to the final results should account for both random and systematic errors, computational model bias, procedural guide inaccuracies, and any significant parameter variations that have not already been taken into account. Examples of random data errors in the heat rate calculation are the standard deviations in fission yields, half-lives, recoverable energies (*Q*-values), and neutron cross sections. The overriding systematic data error and computational bias is in the calculation of neutron cross sections. All the actinides and light-element activation products, plus a few fission products, are mainly dependent on cross sections. The procedural inaccuracies are caused by interpolations and adjustment factor errors. Also, there are several minor parameter variations that are not taken into account.

It appears that there is a natural division of the errors into the following four types:

1. error from random data uncertainty;
2. error in cross sections resulting from data uncertainty and computational model bias;
3. procedural inaccuracy and extra parameter variation error; and
4. other contingency errors.

A discussion of these four error categories is given in Sects. 6.6.1–6.6.4.

6.6.1 Error From Random Data Uncertainty

The random errors considered here are the standard deviations in fission product yields, half-lives, and *Q*-values. Both fission products and light element activation products are included in this analysis. The conventional type of quadratic propagation of standard deviations, described below, is applied to determine the final standard deviation. First, the equations for final uncertainty will be derived. Then tables listing the input standard deviations and the final heat rate standard deviations at different cooling times for several of the standard cases are presented.

Given a general equation for z as a function of variables x_i

$$z = F(x_1, x_2, ..., x_i, ...) \;, \qquad (19)$$

the standard deviation in z from standard deviations in x_i may be derived by the partial differential equation:

$$\sigma_z = \left[\sum_i \left(\frac{F}{x_i} \sigma_{x_i} \right)^2 \right]^{\frac{1}{2}} \;, \qquad (20)$$

where,
σ_z is the standard deviation in z,
σ_{x_i} is the standard deviation in x_i.

Equation (20) will be applied to several equations later in this section. First, the effective yield, y, of a fission product nuclide from the individual yields due to ^{235}U, ^{239}Pu, and ^{238}U fissions is determined by

$$y = \sum_{i=1}^{3} f_i y_i \;, \qquad (21)$$

where

y = the fraction of all fissions yielding the fission product nuclide,
f_i = the fraction of fissions produced from fissile isotope i,
y_i = the fraction of fissions from isotope i yielding the fission product,
i = 1, 2, and 3 for ^{235}U, ^{239}Pu, and ^{238}U, respectively.

Note that in the ORIGEN-S code,[1] the calculation of y from Eq. (21) is never executed explicitly. Instead, a transition constant equal to y_i times the fission cross section of isotope i is used as an element of the transition data matrix. The fission yield rates of a fission product nuclide are added for each isotope i, and the nuclide is accumulated and depleted over the entire irradiation period. However, with the correct effective values of f_i Eq. (21) will represent a good approximation for the propagation of uncertainties of y_i. The error in f_i is not considered as a random uncertainty because it results from the cross-section and computational bias. Thus, f_i is given no standard deviation here, and its error will be considered as part of the cross-section bias in the next section.

Fission-product yields in this project were taken from ENDF/B-V[6] files. The individual and accumulated (by mass number) fission product yields and their standard deviations have been conveniently listed in Ref. 8. The accumulated yields and percentage standard deviations for the dominant (in decay heat rate) 20 fission product nuclides are given in Table 6.7. There are three

Table 6.7 Percentage fission-product yields for fissile isotopes

	^{235}U		^{239}Pu		^{238}U		Total[a,b]	
Nuclide	y_1, %	$\frac{100\,F_{y_1}}{y_1}$	y_2, %	$\frac{100\,F_{y_2}}{y_2}$	y_3, %	$\frac{100\,F_{y_3}}{y_3}$	y, %	$\frac{100\,F_y}{y}$
Kr-85	1.310	0.35	0.570	0.50	0.730	1.40	0.938	0.284
Sr-89	4.900	1.40	1.700	2.80	2.800	2.00	3.324	1.182
Sr-90	5.900	0.70	2.100	2.00	3.200	2.00	4.012	0.687
Y-90	5.900	0.70	2.100	2.00	3.200	2.00	4.012	0.687
Y-91	5.900	0.50	2.500	1.40	4.100	2.80	4.260	0.536
Zr-95	6.500	0.70	4.900	2.00	5.100	1.00	5.684	0.853
Nb-95	6.500	0.70	4.900	2.00	5.100	1.00	5.684	0.853
Ru-106	0.400	1.00	4.300	2.80	2.500	4.00	2.284	2.347
Rh-106	0.400	1.00	4.300	2.80	2.500	4.00	2.284	2.347
Ag-109	0.034	11.00	1.880	8.00	0.270	11.00	0.865	7.657
Sb-125	0.029	4.00	0.111	8.00	0.053	8.00	0.067	5.912
Te-125m	0.029	4.00	0.111	8.00	0.053	8.00	0.067	5.912
Cs-133	6.700	0.35	7.000	0.70	6.600	1.40	6.824	0.373
Cs-137	6.200	0.35	6.700	0.50	6.000	1.00	6.404	0.292
Ba-137m	6.200	0.35	6.700	0.50	6.000	1.00	6.404	0.292
Ce-144	5.500	0.50	3.700	0.50	4.500	1.00	4.628	0.344
Pr-144	5.500	0.50	3.700	0.50	4.500	1.00	4.628	0.344
Pm-147	2.250	1.00	2.040	1.40	2.530	1.40	2.180	0.771
Eu-153	0.161	2.80	0.360	6.00	0.411	2.80	0.269	3.646
Eu-155	0.032	4.00	0.166	11.00	0.133	16.00	0.099	8.316

[a] y is computed from Eq. (21).
[b] Percentage standard deviations in y are computed from Eq. (22).

exceptions in the dominant nuclides. The yields are given for ^{109}Ag, ^{133}Cs, and ^{153}Eu because these isotopes appear after neutron absorption in the path to essentially all of the dominant nuclides ^{110m}Ag, ^{134}Cs, and ^{154}Eu, respectively. Then, one can apply Eq. (20) to compute the standard deviation in y as given by Eq. (21),

$$\frac{F_y}{y} = \frac{1}{y}\left(\sum_{i=1}^{3}(f_i y_i F_{y_i}/y_i)^2\right)^{\frac{1}{2}} . \qquad (22)$$

Values of f_i, the fraction of fissions from isotope i, were computed from the concentrations and fission cross sections of fissile isotope i for a typical SAS2H case. Then, the fraction for ^{239}Pu was increased by 10% and that for ^{235}U was reduced similarly. This was done to account for the increase of ^{239}Pu fissions later in the irradiation period. Applying f_1 = 0.48, f_2 = 0.44, and f_3 = 0.08, the total yields and the percentage standard deviations in the yields were determined from Eq. (22) and listed in Table 6.7. Data for the Q-values and decay constants (λ) were taken from ENSDF.[7] Standard deviations of the half-lives and the β and γ energies and intensities for ENSDF data have been tabulated in Ref. 28. The equation for Q is

$$Q = \sum_i E_i I_i / 100 , \qquad (23)$$

where,

E_i = γ line or average β energy,
I_i = intensity of E_i as percentage of the total disintegrations.

The standard deviation in Q can be obtained using Eq. (20) to be

$$F_Q = 0.01\left(\sum_i (E_i^2 F_{I_i}^2 + I_i^2 F_{E_i}^2)\right)^{\frac{1}{2}} . \qquad (24)$$

Now, the percentage standard deviation in the decay heat rate of a nuclide is to be determined. First, the equation for heat rate is required. Consider d to be the number of atoms of a nuclide at discharge. The depleted number of atoms at cooling time t is

$$x_t = de^{-\lambda t} . \qquad (25)$$

Then, by substituting $d = Cy$, where C is the total number of fissions times the fraction of the nuclide atoms that are not depleted and y is the applicable cumulative yield from Eq. (21), Eq. (25) becomes

$$x_t = Cye^{-\lambda t} . \qquad (26)$$

The constant C cancels out in the final error equation. Note that y is not applicable to the daughter nuclide of two nuclides in secular equilibrium. In this case, it is required that the rate in which both parent and daughter atoms disintegrate are identical:

$$x_1 8_1 = x_2 8_2 , \qquad (27)$$

where x_1 and x_2 are the parent and daughter nuclide concentrations, respectively. Thus, the number of atoms of the daughter nuclide is

$$x_t = x_2 = \frac{8_1}{8_2} x_1 = C' y_1 e^{-8_1 t} , \qquad (28)$$

where

$$C' = C8_1/8_2 . \qquad (29)$$

Then, the most general equation for heat rate of the nuclide is

$$h = x_t 8Q = C' 8_2 Q y_1 e^{-8_1 t} . \qquad (30)$$

If the nuclide is not the daughter in a secular equilibrium pair, then $8 = 8_1 = 8_2$, and

$$h = C8Qye^{-8t} . \qquad (31)$$

Applying Eq. (20) to Eq. (31) yields

$$\begin{aligned}
\frac{Mh}{MQ} &= C\lambda ye^{-8t} = \frac{h}{Q} ; \\
\frac{Mh}{My} &= \frac{h}{y} ; \\
\frac{Mh}{M8} &= CQye^{-8t} - tC8Qye^{-8t} \\
&= \frac{h(1 - 8t)}{8} \\
F_h/h &= [(F_Q/Q)^2 + (F_y/y)^2 + (F_8/8)^2 \\
&\quad (1 - 8t)^2]^{\frac{1}{2}} .
\end{aligned} \qquad (32)$$

Similarly, the uncertainty propagated through Eq. (30) for the secular equilibrium daughter is

$$F_h/h = [(F_Q/Q)^2 + (F_y/y)^2 + (F_{8_2}/8_2)^2 + \tag{33}$$

At discharge (t = 0), the fractional standard deviation in the heat rate h_o is computed from only the first three terms of Eq. (33), and, then, Eqs. (32) and (33) are identical.

The percentage standard deviations in Q, 8_1, 8_2, y, and h_o, as computed from Eqs. (32) and (33) for $t = 0$, are listed in Table 6.8 for the dominant (i.e., in heat rate) 20 fission products and dominant 5 light-element nuclides. Note that the yield data for ^{109}Ag, ^{133}Cs, and ^{153}Eu were substituted for that of ^{110m}Ag, ^{134}Cs, and ^{154}Eu, respectively.

Associated with data for half-lives were standard deviations. Some of the half-lives were taken from a more recent version of ENSDF (see Ref. 28) than that used in the ORIGEN-S libraries.[5] If the decay constant (derived from the half-life) were larger in ORIGEN-S than in Ref. 28, it produced a more conservative decay heat rate for the nuclide. However, if the decay constant were smaller in ORIGEN-S than in Ref. 28, one-half the difference between the two values was added to the standard deviation. Thus, when 2F is used in applying the contribution of this error to the safety factor, the full bias is taken into account.

If h_i and F_{h_i} are the heat rate and the standard deviation in the heat rate for nuclide i, then the total heat rate, H, is

$$H = \sum_i h_i \ , \tag{34}$$

and its standard deviation is

$$F_H = \left(\sum_i F_{h_i}^2 \right)^{\frac{1}{2}} \ . \tag{35}$$

Thus, Eqs. (32) through (35) may be used with Table 6.8 data to compute the approximate standard deviation due to random data uncertainty (excluding cross sections) for the sum of heat rates from the dominant fission products and light elements. The increase in computed uncertainty from applying t > 0 instead of t = 0 in Eqs. (32) and (33) is sufficiently cancelled by the increase in the long cooling time heat rates calculated from using a larger half-life for ^{90}Sr in ORIGEN-S than that in the more recent ENSDF data base[28] that was used in the uncertainty analysis. The adequacy of the selection of dominant nuclides can be seen in comparisons of their totals with totals listed by ORIGEN-S. It would require a project that is both too extensive and unnecessary to calculate F_H for all 20 cooling times of all of the 36 standard cases computed here. The reason this computation is considered unnecessary is because F_H was determined to be small in comparison with other errors that contribute to the safety factors. Data from four of the standard cases (see Tables 5.1–5.8) were used in this study. A case having a power and burnup that are typical and a case having an extended burnup for both the BWR and PWR were used. The calculation of F_H was performed for a total of 24 separate cases comprising cooling times of 1, 1.4, 4, 10, 50, and 110 years. Table 6.9 (for the typical BWR case) shows the individual percentage standard deviations, 100 F_{h_i}/h_i from Eqs. (32) or (33), h_i from the SAS2H/ ORIGEN-S case, and the resulting standard deviations in h_i. Also, the standard deviation in the total heat rate of the 25 nuclides and its conversion to percentage of the total heat rate, including that from actinide nuclides, is summarized in the bottom line of Table 6.9. It is seen that the percentage standard deviation of the total heat rate from the above uncertainties is 0.70% at 1-year decay for the typical BWR case. Also, it is seen in Table 6.10 that the percentage standard deviation (from the random errors) in the total heat rate is 0.10% at the cooling time of 110 years. The reason for the significant reduction to 0.10% is due to the essential vanishing of heat rate from nuclides having the larger standard deviations. Note that 0.0 W denotes the 10^{-78} limit of the floating-point numbers on the IBM 3090 mainframe.

Finally, in order to reduce the number of tables listing detailed data, as in Tables 6.9 and 6.10, a summary of the percentage standard deviations in the total heat rates is listed in Table 6.11. Note that the maximum percentage of the total is 0.72% for the typical PWR case at 1.4-year cooling time. These data will be used in Sect. 6.6.4 in the determination of safety factors.

6.6.2 Error From Cross Sections and Computational Model Bias

This section includes a discussion of error in the cross sections resulting from both data uncertainty and computational model bias. The reason for combining these two types of errors is that it would be extremely complex to propagate standard deviations of each energy

Table 6.8 Percentage standard deviations in data for dominant fission products and light elements

Type	Nuclide	$\frac{100F_Q}{Q}$	$\frac{100F_{8_1}}{8_1}$	$\frac{100F_{8_2}}{8_2}$	$\frac{100F_y}{y}$	$\frac{100F_{h_o}}{h_o}$
Fission <u>products</u>						
	Kr-85	**0.320**	**0.090**	**0.090**	**0.284**	**0.437**
	Sr-89	**0.260**	**0.180**	**0.180**	**1.182**	**1.223**
	Sr-90	**0.410**	**1.920**	**1.920**	**0.687**	**2.080**
	Y-90	**0.130**	**1.920**	**0.160**	**0.687**	**0.718**
	Y-91	**0.160**	**0.100**	**0.100**	**0.536**	**0.569**
	Zr-95	**1.190**	**0.060**	**0.060**	**0.853**	**1.466**
	Nb-95	**0.020**	**0.060**	**0.380**	**0.853**	**0.934**
	Ru-106	**0.800**	**0.330**	**0.330**	**2.347**	**2.502**
	Rh-106	**0.420**	**0.330**	**0.800**	**2.347**	**2.515**
	Ag-110m	**0.410**	**0.040**	**0.040**	**7.657**	**7.668**
	Sb-125	**0.550**	**1.440**	**1.440**	**5.912**	**6.110**
	Te-125m	**1.020**	**1.440**	**1.720**	**5.912**	**6.241**
	Cs-134	**0.220**	**0.242**	**0.242**	**0.373**	**0.496**
	Cs-137	**0.290**	**0.099**	**0.099**	**0.292**	**0.423**
	Ba-137m	**0.320**	**0.099**	**0.080**	**0.292**	**0.440**
	Ce-144	**1.120**	**0.106**	**0.106**	**0.344**	**1.176**
	Pr-144	**0.150**	**0.106**	**0.170**	**0.344**	**0.412**
	Pm-147	**1.400**	**0.006**	**0.006**	**0.771**	**1.598**
	Eu-154	**1.340**	**1.160**	**1.160**	**3.646**	**4.054**
	Eu-155	**2.030**	**0.202**	**0.202**	**8.316**	**8.562**
Light <u>elements</u>						
	Co-60	**0.010**	**0.020**	**0.020**	**0.000**	**0.022**
	Zr-95	**1.190**	**0.060**	**0.060**	**0.000**	**1.192**
	Nb-95	**0.020**	**0.060**	**0.380**	**0.000**	**0.381**
	Sb-125	**0.550**	**1.440**	**1.440**	**0.000**	**1.541**
	Eu-154	**1.340**	**1.160**	**1.160**	**0.000**	**1.772**

Table 6.9 Computed standard deviation in nuclides and their total heat rate at 1 year for typical BWR

Nuclide or total	$\frac{100 F_i}{h_i}$	h_i, W	F_{h_i}, W
Co-60	0.022	3.668E-02	8.202E-06
Kr-85	0.437	1.099E-02	4.803E-05
Sr-89	1.223	1.007E-02	1.231E-04
Sr-90	2.080	7.235E-02	1.505E-03
Y-90	0.718	3.456E-01	2.480E-03
Y-91	0.569	2.779E-02	1.580E-04
Zr-95(FP)	1.466	8.234E-02	1.207E-03
Nb-95(FP)	0.934	1.754E-01	1.639E-03
Ru-106	2.502	1.137E-02	2.844E-04
Rh-106	2.515	1.840E+00	4.627E-02
Ag-110m	7.668	1.516E-02	1.163E-03
Sb-125(FP)	6.110	1.504E-02	9.189E-04
Te-125m	6.241	9.848E-04	6.146E-05
Cs-134	0.496	8.349E-01	4.139E-03
Cs-137	0.423	1.016E-01	4.296E-04
Ba-137m	0.440	3.410E-01	1.501E-03
Ce-144	1.176	1.797E-01	2.113E-03
Pr-144	0.412	1.991E+00	8.202E-03
Pm-147	1.598	3.610E-02	5.770E-04
Eu-154(FP)	4.054	7.733E-02	3.135E-03
Eu-155	8.562	3.809E-03	3.262E-04
Zr-95(LE)	1.192	6.576E-03	7.835E-05
Nb-95(LE)	0.381	1.401E-02	5.330E-05
Sb-125(LE)	1.541	3.922E-03	6.046E-05
Eu-154(LE)	1.772	1.277E-03	2.263E-05
Sum of above		6.234	4.751E-02
All FP and LE[a]		6.251	
Total, incl. actinides[b]		6.841	
Percentage std. dev. in total			0.70%

[a]This is sum of heat rates from all fission products (FP) and light elements (LE) computed by ORIGEN-S.
[b]Total heat rate as tabulated in Table 5.2.

Table 6.10 Computed standard deviation in nuclides and their total heat rate at 110 years for typical BWR

Nuclide	$\frac{100 F_{h_i}}{h_i}$	h_i, W	F_{h_i}, W
Co-60	0.022	2.182E-08	4.880E-12
Kr-85	0.437	9.547E-06	4.173E-08
Sr-89	1.223	0.000E+00	0.000E+00
Sr-90	2.080	5.402E-03	1.124E-04
Y-90	0.718	2.580E-02	1.852E-04
Y-91	0.569	0.000E+00	0.000E+00
Zr-95(FP)	1.466	0.000E+00	0.000E+00
Nb-95(FP)	0.934	0.000E+00	0.000E+00
Ru-106	2.502	3.185E-35	7.968E-37
Rh-106	2.515	5.154E-33	1.296E-34
Ag-110m	7.668	1.653E-50	1.267E-51
Sb-125(FP)	6.110	2.142E-14	1.309E-15
Te-125m	6.241	1.405E-15	8.769E-17
Cs-134	0.496	1.018E-16	5.045E-19
Cs-137	0.423	8.298E-03	3.510E-05
Ba-137m	0.440	2.787E-02	1.227E-04
Ce-144	1.176	1.250E-43	1.470E-45
Pr-144	0.412	1.385E-42	5.706E-45
Pm-147	1.598	1.119E-14	1.789E-16
Eu-154(FP)	4.054	1.182E-05	4.792E-07
Eu-155	8.562	9.229E-10	7.902E-11
Zr-95(LE)	1.192	0.000E+00	0.000E+00
Nb-95(LE)	0.381	0.000E+00	0.000E+00
Sb-125(LE)	1.541	5.585E-15	8.609E-17
Eu-154(LE)	1.772	1.953E-07	3.461E-09
Sum of above		0.06739	2.514E-04
All FP and LE[a]		0.06750	
Total, incl. actinides[b]		0.260	
Percentage std. dev. in total			0.10%

[a]Sum of heat rates from all fission products (FP) and light elements (LE) computed by ORIGEN-S.

[b]Total heat rate as tabulated in Table 5.2.

group (or point) cross section through the computations of the resonance cross sections and the final neutron transport calculation. Furthermore, it would be difficult to estimate the bias in the resonance, transport, and point-depletion types of computations.

Instead of analyzing the uncertainty of data, as used in standard deviation calculations of Sect. 6.6.1, a more global type of technique is applied to determine the error from cross sections and computational bias. Although certain approximations were necessary in parts of the evaluation, a somewhat larger than expected quantity of data were available. The major global technique used here is the comparisons of SAS2H/ORIGEN-S results with other code computations and measurements. In referring to cross-section error here, it is meant to include that error due to the cross-section data uncertainty and code computational bias of the actinides, the light-element activation products, and three fission products (^{133}Cs, ^{153}Eu, and ^{109}Ag). Two additional error types—decay data uncertainties in the actinide production and any other code biases—are also included as a very small part of the total error discussed as cross-section error.

The use of previously documented[14,17] comparisons of decay heat rates calculated by different codes is inadequate to evaluate the uncertainty in the present work. The comparison of results from ORIGEN2[29] and ORIGEN-S in Ref. 14 involved the 1978 ORIGEN2 library[21] and the earlier SCALE-3 version of SAS2H. The multicode comparisons of Ref. 17, although more recent, are not sufficiently current because the ORIGEN2 library[21] has been substantially improved,[30] the ORIGEN-S fission product yields have been updated, the BWR cases have been changed with more realistic data, and the EPRI-CELL code[31] (used with CINDER-2[32]) has been enhanced significantly.[33]

For this uncertainty study, a comparison of heat rates computed by ORIGEN2 using improved BWR and PWR libraries[30] with heat rates computed by SAS2H/ORIGEN-S was made. Note that the improved PWR and BWR libraries available for ORIGEN2 were prepared for only two specific burnups (typical and extended burnups of 27.5 and 40 MWd/kgU for BWR and 33 and 50 MWd/kgU for PWR) and corresponding initial enrichment. The sum of the heat rates at 10 years from the actinides, the light-element activation products and fission products ^{134}Cs and ^{154}Eu computed by ORIGEN2, was determined from Ref. 30. The two fission product heat rates were converted from curies, as listed, to W and added to total heat rates of the other two types of nuclides. The quantity of ^{110m}Ag is insignificant at the decay time of 10 years. Similar cases were computed by SAS2H/ORIGEN-S, where only burnups, powers, and enrichments were changed from the standard cases used to create the decay heat data base provided in Sect. 5. The heat rates from the same set of nuclides as used for ORIGEN2 were also determined for SAS2H/ORIGEN-S. The comparisons for these cross-section-dependent heat rates as computed by SAS2H/ORIGEN-S and ORIGEN2 and their percentage differences are listed in Table 6.12 for the two typical and extended burnup BWR and PWR cases. The agreement for the typical burnup cases was relatively good, while differences of 7.2 and 15.9% for the extended burnup PWR and BWR cases, respectively, were considerably more. A larger fraction of the heat rate for the extended burnup cases is produced from nuclides farther down the various transition chains of the actinides as the burnup is increased. The effect of propagating uncertainty or bias from cross sections of these additional nuclides correspondingly increases the final uncertainty in the heat rate.

The code system that produced data for the ORIGEN2 libraries is denoted as GPRCYCLE.[30] This code system uses a more complex method of computing cross sections than that used by SAS2H in part because it performs multidimensional depletion calculations. Axial variations, changes in fuel pin enrichments, and different fuel zones within the core may be explicitly included in the reactor model.

Reactor calculations performed by the GPRCYCLE system can compute the soluble boron in the moderator of a PWR through a criticality search. Repeated soluble boron computations[34] during a fuel exposure cycle produced data fitting closely the "boron letdown curve." Changes in boron content is a form of determining variations in the k_{eff} of the core because moderator boron content may be converted into reactivity worth. In turn, although cross-section error can be partially cancelled in deriving a k_{eff} (due to similar data in numerator and denominator), the cancellation can never be complete at all points on the boron let-down curve.

In the application of the cross-section data provided by GPRCYCLE, the ORIGEN2 library was designed to include burnup-dependent cross sections for ^{239}Pu through ^{242}Pu and ^{241}Am and initial or midpoint burnup cross sections for the other nuclides. In deriving the SAS2H/ORIGEN-S BWR extended burnup heat rate listed in Table 6.12 (the case having the greatest difference) only midpoint burnup cross sections were applied for all nuclides besides plutonium and ^{241}Am. Executing SAS2H/ORIGEN-S in this fashion for the extended BWR

Table 6.11 Percentage standard deviation in fission-product and light-element heat rate applying *Q*, *8*, and fission yield uncertainties

Reactor type, final uncertainty	Power, kW/kgU	Burnup, MWd/kgU	Percentage standard deviation in total heat rate — Cooling time, y: 1	1.4	4	10	50	110
BWR	20	30	0.70	0.70	0.40	0.30	0.21	0.10
BWR	30	40	0.69	0.69	0.40	0.30	0.21	0.10
PWR	28	30	0.71	0.72	0.42	0.30	0.21	0.10
PWR	40	50	0.68	0.67	0.38	0.29	0.20	0.10
Uncertainty: $2F_{max}$			1.4	1.4	0.8	0.6	0.4	0.2

Table 6.12 Code comparisons of heat rate at 10-year cooling from actinides, light-element activation products plus two fission products (^{134}Cs and ^{154}Eu)

Reactor type	Burnup, MWd/kgU	Cross-section-dependent heat rate, W: SAS2H/ ORIGEN-S	GPRCYCLE/ ORIGEN2	% difference $\left(\frac{\mathrm{ORS}}{\mathrm{OR2}} - 1\right)$ 100%
BWR	27.5	257.9	265.7	2.9
BWR	40.0	508.9	605.2	15.9
PWR	33.0	368.4	360.4	2.2
PWR	50.0	729.3	785.5	7.2

case increased the heat rate provided by normal execution of SAS2H/ ORIGEN-S. The 15.9% difference provided in Table 6.12 is most likely caused by multidimensional effects such as axial variation in the moderator density, partial insertion of boron cruciform control rods, and different fuel enrichments within an assembly. Another reason for the difference may be the use of different ENDF versions of cross sections, in addition to the application of data having different energy group structures. The significant differences in the methodology and approach used to generate the ORIGEN2 and ORIGEN-S cross sections provide a basis for using the heat rate results of Table 6.12 to estimate a cross-section bias for use in formulating the safety factor. The differences provided in Table 6.12 are used in Table 6.13, which summarizes the contribution to the total safety factor.

Now consider comparisons of SAS2H/ORIGEN-S calculations of heat rates with measurements. Note that a comparison of computed total heat rates with measurements for ten BWR and ten PWR assemblies was presented in Sect. 3. The results were summarized in Table 3.16. Also, it is seen that the random type of error or standard deviation computed, as listed in Table 6.11, is approximately 0.5% in the 3– to 4-year range in which the measurements were performed. The error due to cross-section-dependent nuclides should be at least several times the random type error. For the purpose of estimating a measurement of the cross-section-dependent error, the assumption is made that the entire difference between computed and measured decay heat rates is completely due to cross-section bias. Assuming other biases are small, the difference is actually a good estimate of the cross-section error. If there were no systematic bias, the standard deviations of the percentage differences in computed and measured values are large enough to indicate that these data are simply not a very precise estimate of the cross-section bias. Nevertheless, it may be considered to be one of the best available global measurements of cross-section-dependent heat rate error. The fraction of the total heat rate that is produced from the cross-section-dependent nuclides was computed as 0.311 and 0.315 of the averages of two typical BWR and PWR cases, respectively. Dividing the differences of –0.7 ± 2.6% and 1.5 ± 1.3% (from Table 3.16) by the above fractions produces cross-section bias estimates of –2.3 ± 8.4% and 4.8 ± 4.1% for the BWR and PWR, respectively. Thus, the second type of consideration used in estimating the cross-section error and the safety factor is that derived from comparisons of SAS2H/ ORIGEN-S and measured heat rates, as listed on the third line in Table 6.13.

In addition to the two major methods discussed above for determining cross-section biases, it appeared reasonable to consider other information pertaining to the reliability of the calculated decay heat values. During the development of SAS2H for the SCALE-4 system, cases were executed producing isotopic contents that could be compared with experimental determinations of the isotopic inventories of various PWR assemblies. The comparison of measured and computed contents of uranium and plutonium isotopes for a Turkey Point Unit 3 assembly is given in Ref. 35. The largest difference was the nonconservative 8.9% for ^{240}Pu; however, it is not a significant contributor to heat rate. The next largest difference was the conservative 6.1% for ^{241}Pu. No large nonconservative bias in heat rate is indicated. However, the measurement of a significant decay heat rate contributor, ^{244}Cm, was not given. Comparisons were also made for PWR assemblies from the H. B. Robinson Unit 2 Reactor[4] in South Carolina and the Obrigheim (KWO) Reactor[36] in the Federal Republic of Germany. These comparisons included measurements of ^{244}Cm. The H. B. Robinson measurements included ^{134}Cs. Although the α spectrometer standard deviations of ^{244}Cm results are quoted as ± 20 to 30%, there were also measurements of the Obrigheim samples in which the much more precise method of isotope dilution analysis was used for ^{244}Cm. Although it does not appear to be within the scope of this project or report to give detailed comparisons, note that agreement in the total cross-section-dependent heat rate would be well below 10% and would possibly be conservative. Note that this applies only to PWR assemblies in the typical burnup region.

Isotopic analyses of BWR spent fuel were performed using SAS2H/ORIGEN-S as part of a study on using actual spent fuel isotopics in the criticality safety analyses of transport casks.[37] Average assembly isotopic concentrations for some of the most significant nuclides (e.g., ^{244}Cm and ^{133}Cs) were calculated using multi-axial-node models and SAS2H. The isotopic differences between the multi-node and single-node models using SAS2 were less than half the differences observed in the GPRCYCLE/ORIGEN2 and SAS2H comparison.

Some select cases were also run to investigate the effect of the energy grouping in the SCALE 27-group cross sections used by SAS2H to calculate the tabulated

Table 6.13 Summary of cross-section bias estimates and safety factors

	BWR		PWR	
Burnup, MWd/kgU	**27.5**	**40.0**	**33.0**	**50.0**
Difference in codes, %[a]	**2.9**	**15.9**	**2.2**	**7.2**
Difference in measurements, %[b]	**2.3 ± 8.4**		**4.8 ± 4.1**	
Average difference, %	**2.6**	**15.9**	**3.5**	**7.2**
Safety factor for σ-error, %[c]	**11.0**	**16.0**	**10.0**	**11.0**

[a]Comparisons shown in Table 6.12.
[b]Assuming differences between computed and measured heat rates (Table 3.16) caused by cross-section error.
[c]Safety factors, applied as adequate, for cross-section uncertainty and computation model bias in heat rate of nuclides produced primarily by neutron absorption.

decay heat values. The PWR case for 28 kW/kgU and 45 MWd/kgU was executed using the 218-energy- group ENDF/B-IV library in the SCALE system. The total heat rates computed by the 218-group case were greater by 0.3% at 1 year, about equal at 5 years, and 1.5% smaller than those of the 27-group case at 110 years. In another study, the typical and extended burnup BWR cases (27.5 and 40 MWd/kgU, respectively) were executed on the CRAY X-MP using ENDF/B-V cross sections, which resulted in an increase in the computed actinide and total heat rates.

The actinide heat rates increased by 3 and 6% and the total heat rates increased by 0.2 and 1.6% in the typical and extended burnup cases, respectively. All of these values are within the uncertainty envelope provided by the code comparison of Table 6.12.

Another comparison study[38] was performed on an international scale to determine differences between the mathematical models of codes computing decay heat rates. Contributions to the study were received from China, France, Germany (FRG), Japan, Sweden, UK, USA, and USSR, applying the following codes: AFPA, CINDER-10, CINDER, DCHAIN, FISP6, INVENT, PEPIN, FISPIN, KORIGEN, MECCYCO, and ORIGEN-S. By starting with identical model and library data, each code computed decay heat rates for each decade from 1 to 10^{13} s, applying a ^{235}U fission pulse and a long irradiation (3×10^7 s) in the two benchmark cases. The total heat rates for the 13 decay times computed by all codes are within 0.7% from the average for the pulse case and within 1.6% for the long irradiation case. The total heat rates from ORIGEN-S are within 0.5% from the average for the pulse case and within 0.4% for the long irradiation case. A significant part of the ORIGEN-S difference can arise from the roundoff in the three-place printout. The agreement shows good validation of the mathematical model of ORIGEN-S.

The major discussion in this section has been concerned with the error in cross sections resulting from both data uncertainty and computational model bias. The effect of these has been demonstrated via code comparison and the differences between computed and measured cross-section-dependent heat rates. Using the percentage differences derived for Table 6.12 and the comparisons with measurements, a summary of estimates of the cross-section bias are presented in Table 6.13. For the same cases, the differences between the code computations (GPRCYCLE/ ORIGEN2 and SAS2H/ORIGEN-S) and the differences between the calculations and measurements (after assuming they may be converted to cross-section bias) are in moderate agreement. The average differences for the typical BWR and PWR are –2.6 and 3.5%, respectively. After consideration of these data and other information discussed above, it was decided to apply an 11% safety factor to the lower burnup BWR cross-section-dependent heat rates. A 10% safety factor was given to the lower burnup PWR values. Although the PWR results appear to have significantly less nonconservative bias, the 10% minimum is reasonably liberal and not greatly overconservative. Indications were seen that the extended burnup PWR results are approximately the same as the lower burnup BWR and should have a similar safety factor. Also, from all comparisons of calculations for the extended burnup BWR, it appeared that a 16% safety factor for cross-section bias was both sufficient and reasonable. These factors will be used in determining the contribution from cross-section bias to the safety factors of the total heat rate. Examples of these contributions are given in Tables 6.14 and 6.15, which show the itemized calculation of the safety factor in the total heat rate due to cross-section bias for four cases at 1 and 110 years,

Table 6.14 Contribution of cross-section (σ) bias to safety factor of total heat rate at 1 year

Type of data	Cases			
Reactor type	BWR	BWR	PWR	PWR
Specific power, MW/kgU	20	20	28	28
Burnup, MWd/kgU	20	45	25	50
Actinide heat rate, W/kgU	296	1129	387	1282
Light-element heat rate, W/gU	62	77	136	177
^{134}Cs heat rate, W/gU	487	1356	707	1773
^{154}Eu heat rate, W/gU	43	133	59	158
^{110m}Ag heat rate, W/gU	10	23	16	43
Sum σ-dependent heat rate, W/gU	898	2718	1305	3433
Total heat rate for case, W/gU	5548	8571	7559	11120
Percentage of total from σs[a]	16.2	31.7	17.3	30.9
Safety factor for σ-error, %	11.0	16.0	10.0	11.0
Safety factor in total from σs, %[a]	1.8[b]	5.1	1.7	3.4

[a]The term "from σs," as used here, means the heat rate of nuclides produced primarily by neutron absorption.

[b]This part of the total heat rate safety factor due to cross-section bias is the product of fraction of heat rate from σ-dependent nuclides and their safety factor (see also Table 6.13) or 16.2 × 11.0/100 = 1.8%.

Table 6.15 Contribution of cross-section (σ) bias to safety factor of total heat rate at 110 years

Type of data	Cases			
Reactor type	BWR	BWR	PWR	PWR
Specific power, MW/kgU	20	20	28	28
Burnup, MWd/kgU	20	45	25	50
Actinide heat rate, W/kgU	132	287	167	328
Other σ-dependent heat rates	0	0	0	0
Sum σ-dependent heat rate, W/gU	132	287	167	328
Total heat rate for case, W/gU	178	385	225	440
Percentage of total from σs[a]	74.2	74.5	74.2	74.5
Safety factor for σ-error, %	11.0	16.0	10.0	11.0
Safety factor in total from σs, %[a]	8.2[b]	11.9	7.4	8.2

[a]The term "from σs," as used here, means the heat rate of nuclides produced primarily by neutron absorption.

[b]This part of the total heat rate safety factor due to cross-section bias is computed as the product of values on last two prior lines as 74.2 × 11.0/100 = 8.2%.

respectively. The lowest and highest burnup cases for the middle specific power of the tabulated data of each reactor type are used in the example shown. The portion of the total heat rate safety factors due to cross-section bias listed on the last line of these tables is applied in the final safety rate determination in Sect. 6.6.4.

Consideration was given to comparing cross-section-dependent decay heat rates computed by SAS2H/ ORIGEN-S with other accepted procedures for computing decay heat rates. The American National Standard for Decay Heat Power in Light Water Reactors[39] (ANSI 5.1) computes heat rates from ^{239}U and ^{239}Np (significant in loss-of-coolant accidents), but it does not compute other actinide heat rates that have significant contributions at 1 year and later cooling times. Thus, no comparison was made in this study. A draft document (from the International Organization for Standardization) of a standard on decay heat power[40] (which is not referred to officially as the international standard until publication) applies a contribution of the actinide heat rate in addition to that from ^{239}U and ^{239}Np. The proposed method multiplies an actinide factor, $A(t)$, times the summed heat rate of fission product decays, P_S, to determine the actinide contribution, P_A. Values of $A(t)$ are tabulated as a function of time only. Thus, $A(t)$ does not vary with burnup. A value with the same definitions as $A(t)$ may be computed from the SAS2H data in Table 6.14. The summed heat rates of fission products, P_S, for the SAS2H data are simply the difference given by the "total heat rate for the case" minus the "sum σ-dependent heat rate." Then, for the BWR cases at 1-year decay, the SAS2 values of $A(1)$ are 296/(5548-898) = 0.064 in the 20-MWd/kgU case and 0.193 in the 45-MWd/kgU case. Similarly, for the low- and high-burnup cases of the PWR, the $A(1)$ values are 0.062 and 0.167, respectively. The value at 1 year, $A(1)$, in the proposed standard is 0.214. The simple formalism used by the proposed standard necessitates the conservatism shown by the comparison with the SAS2H values.

6.6.3 Error in Procedure of Guide and Extra Parameter Variation

This section deals with inaccuracies in the proposed procedure of Sect. 5 in addition to bias that can arise from variations in parameters not previously taken into account. An example of a procedural guide error is that resulting from the interpolation of heat rate data in Tables 5.1–5.3 and Tables 5.5–5.7. An example of a parameter variation not taken into account is the application of the revised procedure to a PWR fuel assembly containing BPRs instead of guide tubes for control rods.

Inaccuracies in the proposed procedure of Sect. 5 are discussed in Sects. 6.1–6.5. Summaries listing the results of several types of comparisons indicating these inaccuracies are shown in Tables 6.1–6.6. In general, the nonconservative differences do not exceed 1%. The only exceptions to this limit were the errors in the interpolations of heat rate with respect to specific power for short cooling times that are in error by 1.1%. The error diminishes for longer cooling times. Other parameter variations not taken into account in the previous discussions are (1) different reactor designs, (2) fuel assemblies (PWR) containing BPRs, and (3) PWR moderator density variations.

An earlier version of SAS2H was used to compare heat rates[4] for PWR fuel assemblies of four different array designs from three different vendors. The cooling time range was 0–10 years. The largest difference was 0.6% for the cooling time of 10 years. The average differences were 0.3 and 0.4% at 1 and 10 years, respectively. These differences, although small, were considered in developing the safety factor.

Cases with fuel assemblies containing BPRs for one of the cycles were computed by SAS2H/ORIGEN-S for both a typical (33 MWd/kgU) and an extended (50 MWd/kgU) burnup of a PWR. Although there were no significant differences for short cooling times, the maximum was a nonconservative 2.2% for 110 years in the typical case. Likewise, reasonable moderator density changes in a PWR provided a slight heat rate change for short cooling and increased to about 2% for long decay times. Changes in the actinide production caused by BPRs and/or water density variations causes the differences in decay heat to appear only at long cooling times where actinides become important to the total decay heat.

It was concluded that an adequate magnitude of safety factor for the procedural inaccuracy and extra parameter variation is 1.5% at 1 year and linearly increased to 2.0% at 110 years. This factor would tend to cover the interpolation error plus other small bias at 1 year, in addition to somewhat larger differences from the above considerations at long cooling times.

6.6.4 Total Safety Factors

This section uses data from Sects. 6.6.1–6.6.3 to determine the total safety factors in the decay heat rates. The total safety factors for 16 cases are compared with those computed by formulae given in Sect. 5. Then additional comments are given on other comparisons not explicitly shown.

The determination and summary of the safety factors for the BWR and PWR are presented in Tables 6.16 and 6.17, respectively. The data on the first line under the case parameter heading are the 2σ random data uncertainties discussed in Sect. 6.6.1 and given in Table 6.11. The values on the next line are the contributions of the cross-section bias to the total safety factor as derived in Sect. 6.6.2 and given in the last lines of Tables 6.14 and 6.15 for cooling times of 1 and 110 years. Similar data were calculated for the cooling times of 4 and 10 years. The safety factor contributions on the next line are for errors in the procedure and extra parameter variation. A discussion of these data is given in Sect. 6.6.3. On the next line a contingency safety factor of 1% was arbitrarily chosen for all the cases. It is simply a small, although significant, increase in the safety factor to cover any other or unexpected error not adequately taken into account. The 1% magnitude, although a matter of judgment, appears to be commensurate with the relative completeness of the safety factor formulation. The totals of the first four lines of Tables 6.15 and 6.16, shown on the fifth line, are of total required safety factors. The final line of each table contains the safety factors for BWR and PWR as produced by Eqs. (16) and (17), respectively. In all of the 16 cases, except the 2 cases at 110 years for the BWR, the equations sufficiently cover the safety factor. In these two cases, the differences are 0.1 and 0.2%. In addition to the cases in Tables 6.16 and 6.17, the safety factors and equation values were calculated for a variety of other cases. The required safety factor and equation value for the lower specific power cases tended to compare more closely than other cases. For only a few PWR cases, the equation value was lower by a magnitude as large as 0.1%. Only two BWR cases had equation values below the required safety factor. These two BWR 110-year cooling time cases had differences of 0.3 and 0.4% below the equation values. Note that the total safety factors for these two cases exceeded 11%. Thus, because there are only rare exceptions in which the equations are even slightly nonconservative, Eqs. (16) and (17) are considered to appropriately envelop all required safety factors.

The final equation to be analyzed here is Eq. (18). This equation simply applies all adjustment factors and the safety factor to the heat rate determined through interpolation of tabulated data.

Table 6.16 Summary of percentage safety factors for BWR

Type of error or safety factor;	Section discussed	$20\ \frac{\text{kW}}{\text{kgU}}, 20\ \frac{\text{MWd}}{\text{kgU}}$				$20\ \frac{\text{kW}}{\text{kgU}}, 45\ \frac{\text{MWd}}{\text{kgU}}$			
		Cooling time, years				Cooling time, years			
		1	4	10	110	1	4	10	110
$2\sigma(Q,\vartheta,y)$, FP + LE[a]	6.6.1	1.4	0.8	0.6	0.2	1.4	0.8	0.6	0.2
Cross-section bias	6.6.2	1.8	3.0	2.9	8.2	5.1	6.4	5.7	11.9
Procedure and extra bias	6.6.3	1.5	1.5	1.6	2.0	1.5	1.5	1.6	2.0
Contingency	6.6.4	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Total		5.7	6.3	6.1	11.4	9.0	9.7	8.9	15.1
S, Eq. (16)	5.2.5	6.4	6.5	6.8	11.2	10.2	10.3	10.6	15.0

[a]FP: fission products; LE: light elements.

Table 6.17 Summary of percentage safety factors for PWR

Type of error or safety factor;	Section discussed	$28\ \frac{\text{kW}}{\text{kgU}}, 25\ \frac{\text{MWd}}{\text{kgU}}$				$28\ \frac{\text{kW}}{\text{kgU}}, 50\ \frac{\text{MWd}}{\text{kgU}}$			
		Cooling time, years				Cooling time, years			
		1	4	10	110	1	4	10	110
$2\sigma(Q,\vartheta,y)$, FP + LE[a]	6.6.1	1.4	0.8	0.6	0.2	1.4	0.8	0.6	0.2
Cross-section bias	6.6.2	1.7	3.0	2.9	7.4	3.4	4.6	4.2	8.2
Procedure and extra bias	6.6.3	1.5	1.5	1.6	2.0	1.5	1.5	1.6	2.0
Contingency	6.6.4	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Total		5.6	6.3	6.1	10.6	7.3	7.9	7.4	11.4
S, Eq. (17)	5.2.5	6.2	6.4	6.7	11.1	7.7	7.9	8.2	12.7

[a]FP: fission products; LE: light elements.

7 THE LWRARC CODE

The LWRARC (Light Water Reactor Afterheat Rate Calculation) code is a Microsoft BASIC program that follows the complete procedure as described in Sect. 5. It runs on an IBM-compatible personal computer. LWRARC features a pulldown menu system with sophisticated data entry screens containing context-sensitive help messages. The menu system organizes the major command categories as menu titles and pulldown commands. The program performs checking for each input screen and presents warning and error message boxes and yes/no dialog boxes to verify that the user wants to perform certain functions (e.g., write over files that have been saved previously). The menus may be used with either a keyboard or a mouse.

7.1 Using the Menu System

The pulldown menu system contains a menu bar across the top of the screen. Under each menu bar option there is a unique pulldown menu. The menu bar options are summarized below.

Files	Options to retrieve files, save files, and exit the program
Data	Options to input data and review a list of case titles
Run	Options to execute cases and browse and print output

When LWRARC begins, it presents the **Files** pulldown menu. The user may scan across the menu bar by using the left and right arrow keys. The user may select an option from a pulldown menu by using the up and down arrow keys. A description of the highlighted option appears on the bottom line of the screen. To execute the option, simply press the <Enter> key or the highlighted letter (known as the "hot key") of the desired option.

The mouse may also be used to select an option by moving it left or right along the menu bar or up and down on a pulldown menu. To choose a pulldown menu command, simply move the mouse cursor over the desired option and click the left mouse button. Some users may prefer to "drag" the mouse (holding the left mouse button) over the menus and then release the left button when the cursor is over the desired command. One difference between the two methods of using the mouse is that dragging the mouse causes the description of the highlighted option to be displayed on the bottom of the screen as when using the keyboard.

Some pulldown menu commands are black with a hot key, while others are gray. The black commands are active; the gray commands are inactive and produce no effect if selected. Commands are activated as necessary data are entered. For example, once cases have been entered or files have been retrieved, the **Review Cases** and the **Execute Cases** options are active.

7.2 Menu System Options

This section presents an overview of the menu system commands. Menu bar options are presented next to round bullets; corresponding pulldown commands are listed next to diamond bullets. The underlined letter in commands discussed below represents the hot key for the command; these hot keys appear on screen as bright letters, and they allow immediate access to an item by pressing the highlighted key.

• **Files** – The **Files** menu allows the user to retrieve and save files and to exit the program.

Retrieve Files – This option allows the user to retrieve files which the user has previously saved with LWRARC. The user selects the desired input file from a list of all LWRARC input files in the current directory. LWRARC then loads the input file and the output file (if it exists) with that file name. The input and output files for a set of cases have the same file name specified by the user, but a different file extension designated by LWRARC (".INP" for input and ".OUT" for output). The program copies the files to a set of temporary files, ARCXXYYZ.∗ (i.e., file extensions remain the same). A set of sample files has been provided on the distribution diskette. The input and output files are SAMPLE.INP and SAMPLE.OUT, respectively.

Save Input File – LWRARC automatically saves the input file when cases are executed, so this option is only needed when a user wants to save the input file without executing cases. This option is designed to allow the user to occasionally save input data while entering a large number of cases, so that it will not be lost if the program terminates abnormally.

Note that if LWRARC does fail, the data entered still exist in the ARCXXYYZ.∗ files. When LWRARC is subsequently run, it will notify the user that it has detected these files and will give him an opportunity to rename and save them.

Quit – This option terminates the program. If the input file has not been saved and cases have not been executed since the user last selected the **Enter Data** option, the program verifies that this is what the user wants to do. Since this option does not save files, the user may also use this option to cancel any changes he has made since the last time the input file was saved or cases were executed.

• **Data** – The Data menu provides the user with selections to enter and review the fuel assembly data.

Enter Data – This is the only active option other than **Retrieve Files** when the user enters the program. When this option is selected, the fuel assembly data input screen is displayed. The cursor appears at the first data field. To move from one field to the next, press <Enter> or <Tab>. To move backward to the previous field, press <Shift+Tab> or <Backspace>. If a field is protected, the cursor skips that field and moves to the next unprotected field. The cursor must be at the beginning of the field for <Backspace> to move to the previous field, because within a data field it moves back one space and deletes the previous character. The user may also change fields with the up and down arrow keys, but the movement is somewhat erratic as the cursor skips fields if there is more than one field per screen line. <Home> moves the cursor to the beginning of the field, and <End> moves to the end of the field. The <Ins> and <Del> keys perform their normal functions for editing a field. <Ctrl+Home> moves the cursor to the first field on the screen, and <CTRL+End> moves the cursor to the last field. <PgUp> and <PgDn> performs the same respective functions. Most fields have a help message associated with them. Pressing <F1> displays the message. Press <Enter> or <Esc> to remove the message from the screen.

Some fields have a multiple choice menu associated with them. Pressing <Enter> or the Space Bar at one of these fields activates the multiple choice menu. When this menu is displayed, all other processing and functions keys are disabled until a choice is selected and the menu disappears from the screen. Pressing a letter on the keyboard moves the cursor to the next choice which begins with that letter. If there is not a choice that begins with that letter, the cursor moves to the beginning of the list. The up and down arrow keys, <PgUp>, <PgDn>, or the mouse may be used to make a selection. <Home> moves the cursor to the first choice, and <End> moves the cursor to the last choice. Once the desired choice is highlighted, press <Enter> or double click the left mouse button to select it and remove the menu.

The date fields accept dates in the form MM-DD-YYYY from 01-01-1900 to 01-01-2065. The default for the first two digits of the year are "19." The cursor automatically skips the first two characters of the year. To modify them, use the backspace key to delete them and then type the new values.

Pressing <Ctrl+B> while the cursor is in a date field will blank that field. The data for a case are saved by pressing <F10> to advance to the next case, or <F9> to go back to the previous case. The <Ctrl+D> key combination allows the user to duplicate a case by changing the case number to the next available case number. Any unsaved changes on the screen when <Ctrl+D> is pressed are not saved for the original case number, but are carried to the new case number. Thus, the user may press <Ctrl+D> either before or after making changes to the old case he wants to copy. Once he is finished with the changes to the new copy, he should press <F9> or <F10> to save them. To avoid saving changes on the screen at any time, press <F4> to return to the main menu system. LWRARC asks the user if he wants to save the changes on the screen before returning.

The <Ctrl+E> key combination erases an unwanted case. LWRARC verifies that the user wants to delete the case. To review existing cases, press <Ctrl+R> to view a list of case numbers and titles.

The input parameters are described below. The variable names in parentheses refer to those used in Sect. 5.

1. Title – 56-character maximum.

2. Reactor type –BWR or PWR.

3. Fuel type – Select from the multiple-choice menu. Based on the fuel assembly type selected, the code automatically supplies the generic kilograms of uranium per assembly for that type (Tables 7.1 and 7.2) in the next field and protects the field. These generic values are only typical loadings taken from Ref. 41. If the user wishes to input a more exact value for the next parameter, he should select "Other." LWRARC then unprotects the next field and allows the user to input his own value.

4. Fuel loading – the kgU/assembly is used to convert the decay heat from watts per kgU (as given in the tables in Sect. 5.2) to watts per assembly. Enter 0 if this calculation is not desired.

5. Interpolation – Log-linear (duplicates hand calculation) or Lagrangian[42] (typically a more precise interpolation scheme).

6. Enrichment (E_s) – initial fuel enrichment, wt % ^{235}U.

7. Total burnup (B_{tot}) – for the assembly, MWd/kgU or GWd/MTU.

8. Cycle N burnup (B_e) – last cycle burnup, MWd/kgU or GWd/MTU.

9. Cycle N-1 burnup (B_{e-1}) – next-to-last cycle burnup, MWd/kgU or GWd/MTU.

10. Initial startup date – cycle startup date for the initial cycle for this assembly, used to calculate the total residence time parameter.

11. Cycle N-1 startup date – next-to-last cycle startup date used to calculate the next-to-last cycle time parameter.

12. Cycle N startup date – Last cycle startup date, used to calculate the next-to-last and last cycle time parameters.

13. Final shutdown date – last cycle shutdown date, used to calculate the last cycle time, the total residence time, and the cooling time parameters.

14. End of cooling date – used to calculate the cooling time parameter.

15. Cooling time (T_c) – time since final discharge of assembly in years.

16. Total residence time (T_{res}) – assembly residence time from first loaded to final discharge in days.

17. Cycle N time (T_e) – last cycle time, startup to discharge, days.

18. Cycle N-1 time (T_{e-1}) – next-to-last cycle time, startup to startup, days.

    The date parameters are optional. If they are input, the code calculates the time parameters automatically and protects the time fields. Otherwise, the user inputs the time parameters. The time and burnup for the last and next-to-last cycles may be set to zero if the cooling time is greater than 15 years.

Review Cases (Ctrl+R) – This option displays a list of case numbers and titles for all cases that have been created. This function may also be performed by pressing <Ctrl+R> at the main menu or while entering data. To remove the list from the screen, press <Enter> or <Esc>.

• **Run** – The **Run** menu provides the user with selections to execute cases and view the output.

Execute Cases – This option performs the decay heat calculations based on the entire procedure given in Sect. 5 for the cases entered. A message box is displayed when execution is completed. When cases are executed, the input file and the output file are saved to the file name specified by the user.

Browse Output File – This option allows the user to browse the output file generated when the cases are executed. The top line of the screen shows the file name and line number. The bottom line of the screen shows the keys that may be used to move within the file while browsing.

Table 7.1 BWR fuel assembly loadings

Assembly manufacturer	Array size	Class	Version	Kilograms U per assembly
ANF	7 × 7	GE BWR/2,3		184
ANF	8 × 8	GE BWR/2,3		174
ANF	9 × 9	GE BWR/2,3		168
ANF	8 × 8	GE BWR/4-6		177
ANF	9 × 9	GE BWR/4-6		173
General Electric	7 × 7	BWR/2,3	2a	196
General Electric	7 × 7	BWR/2,3	2b	193
General Electric	8 × 8	BWR/2,3	4	183
General Electric	8 × 8	BWR/2,3	5	177
General Electric	8 × 8	BWR/2,3		176
General Electric	7 × 7	BWR/4-6	2	193
General Electric	7 × 7	BWR/4-6	3a	188
General Electric	7 × 7	BWR/4-6	3b	190
General Electric	8 × 8	BWR/4-6	4a	184
General Electric	8 × 8	BWR/4-6	4b	186
General Electric	8 × 8	BWR/4-6	5	183
General Electric	8 × 8	BWR/4-6		183
Plant-specific designs				
General Electric	9 × 9	Big Rock Pt.		138
ANF	11 × 11	Big Rock Pt.		132
ANF	6 × 6	Dresden 1		95
General Electric	6 × 6	Humboldt Bay		76
ANF	10 × 10	LaCrosse		108

Table 7.2 PWR fuel assembly loadings

Assembly manufacturer	Array size	Version	Kilograms U per assembly
General designs			
ANF	14 × 14	WE	379
ANF	14 × 14	CE	381
ANF	14 × 14	Top Rod	365
ANF	15 × 15	WE	432
ANF	17 × 17	WE	401
Babcock & Wilcox	15 × 15		464
Babcock & Wilcox	17 × 17	Mark C	456
Combustion Engineering	14 × 14	Std	386
Combustion Engineering	16 × 16		426
Westinghouse	14 × 14	Std/LOPAR	389
Westinghouse	14 × 14	OFA	336
Westinghouse	14 × 14	Model C (CE)	397
Westinghouse	15 × 15	Std/LOPAR	456
Westinghouse	15 × 15	OFA	463
Westinghouse	17 × 17	Std/LOPAR	464
Westinghouse	17 × 17	OFA	426
Westinghouse	17 × 17	Vantage 5	423
Plant-specific designs			
Combustion Engineering	14 × 14	Fort Calhoun	376
Westinghouse	15 × 15	Haddam Neck	413
Babcock & Wilcox	15 × 15	Haddam Neck	409
ANF	15 × 15	Palisades	401
Combustion Engineering	15 × 15	Palisades	413
Combustion Engineering	16 × 16	St. Lucie 2	390
Westinghouse	14 × 14	San Onofre 1	373
ANF	15 × 16	Yankee Rowe	236
Combustion Engineering	15 × 16	Yankee Rowe	231

Print Output File – This option allows the user to print the output file. There is one page of output per case.

7.3 The LWRARC Code Distribution Diskette

Files included on the LWRARC distribution diskette are listed in Table 7.3. LWRARC.EXE is the executable program. LWRARC.QSL is the screen library. This library contains all the input screens displayed by LWRARC. The files with the ".FRM" extension contain the form definition for each of the screens in the library. The files with the ".BSV" extension were written with Microsoft BASIC's BSAVE command and contain the regulatory guide decay heat data from the tables in Sect. 5.2 and the fuel assembly kgU loadings. In addition to these files, LWRARC input and output files for the sample cases in Appendix C are included on the diskette. These files are SAMPLE.INP and SAMPLE.OUT. The user may retrieve them in LWRARC.

No matter how carefully a computer code is written, it is inevitable that some option combination or set of data requires modifications or corrections to the code. In order to be certain that a calculated case was performed with an approved version of the code, the creation date of the version used is printed in the case output heading. The current approved creation date, at the publication time of this report, is 5/30/94.

Table 7.3 Files required by LWRARC

File name	Extension
ARCDATA1	BSV
ARCDATA2	BSV
ARCDATE1	FRM
FUELASSY	BSV
LWRARC	EXE
LWRARC	QSL
RENAMFIL	FRM
SAMPLE	INP
SAMPLE	OUT
SAVEFIL	FRM
SAVEINP	FRM

The latest LWRARC code version may be requested from either the Radiation Shielding Information Center (RSIC) or the Energy Science and Technology Software Center (ESTSC):

Radiation Shielding Information Center
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6362
Telephone: (615) 574-6176
FAX: (615) 574-6182

Energy Science and Technology
Software Center
P.O. Box 1020
Oak Ridge, TN 37831-1020
Telephone: (615) 576-2606
FAX: (615) 576-2865

For inquiries about the latest version date, the user may contact RSIC or one of the following:

C. V. Parks (615) 574-5280
O. W. Hermann (615) 574-5256
S. M. Bowman (615) 574-5263

8 SUMMARY

Proper methods of storing spent fuel from nuclear power plants require knowledge of their decay heat generation rates. Presently, the NRC has issued the decay heat guide entitled "Regulatory Guide 3.54, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation." A significant revision to the heat generation rate guide will be developed by expanding the data base, simplifying the procedure, and using improved computational methods. This report was written to provide the necessary technical support for a proposed decay heat guide.

The primary purposes in revising the guide are to use accurate heat rates that are adjusted with adequate safety factors and, at the same time, avoid excess conservatism. There are three reasons that the current guide is overly conservative. First, the SAS2 calculations used in producing the guide were conservative because not all of the moderator (i.e., the water in and surrounding the guide tubes) was taken into account. Second, the current guide is conservative because the decay heat data base was calculated at several burnups but only for one specific power. Although the power was large enough to envelop most operating LWR reactors, the use of an excessive power will decrease the fuel exposure time which, in turn, increases predicted decay heat rates computed for the range of the cooling times considered. The final reason for excess conservatism in the present guide is that no data base was provided for BWR fuel and a simple conservative safety factor was applied instead.

These reasons for excess conservatism will be eliminated in the proposed guide revision discussed in this report. The SCALE-4 version of SAS2 (also known as SAS2H) applies a second pass in the depletion analysis that simulates water holes or BPRs more correctly and provides substantially better calculations of isotopic contents. Also, the channel water in a BWR can be better simulated in the current SAS2H version. A data base of decay heat is provided for PWR and BWR fuel by computing cases for six different burnups at each of three separate specific powers. Decay heat values at twenty different cooling times from 1 to 110 years were tabulated for each case. A prescribed interpolation procedure gives accurate heat rates over all ranges of burnup, specific power, and cooling time within the limits of the tabulated data. Also, adjustments in the interpolated heat rate take into account variations in the initial ^{235}U enrichment and the short cooling time effects from having an average power within a cycle that is different than the average constant power over all cycles.

The validation of using SAS2H/ORIGEN-S for computing heat rates was provided by comparing predicted results with measurements performed on discharged fuel assemblies. The fuel was taken from Cooper Nuclear Station BWR, Point Beach Unit 2 PWR, and Turkey Point Unit 3 PWR. All design and operating history data for the 20 measured assemblies are tabulated in the report. The measured and computed results are also listed, along with the average of all case differences and the average of differences by assemblies. The average differences of the calculated minus the measured heat rates of fuel assemblies were –0.7 ± 2.6% for the BWR and 1.5 ± 1.3% for the PWR. These comparisons are considered to be acceptable.

The complete procedure to be included in a proposed regulatory guide is included in Sect. 5. Attention is given to special definitions of cycle time and specific power. A total of 720 decay heat rates computed by SAS2H/ ORIGEN-S as a function of the fuel assembly's total burnup, specific power, cooling time, and reactor type (BWR or PWR) are tabulated in a data base in Sect. 5. The interpolation of the heat rate using the specified values of the above four parameters for the assembly is then used with any of the required adjustment factors and the safety factor to determine the final decay heat rate.

A detailed analysis of the proposed procedure for determining decay heat rates is presented in Sect. 6. The derivation and/or discussion of each equation of Sect. 5 is given in sequential order. The definitions of parameters applied and the reasons behind the use of certain variables are extensively discussed in order to more clearly explain why the procedure of the proposed guide revision is appropriate. Also, the accuracies of both the interpolations and the adjustment factor equations are properly analyzed. The formulation of the safety factors is presented in detail. The discussion and analysis of the safety factor is separated into several natural divisions: random data uncertainty, cross-section-dependent error from both data uncertainty and computational model bias, procedural guide inaccuracy, extra parameter (i.e., those not previously applied) variation error, and contingency error. The report discusses and demonstrates why the safety factor ranges of 6.4 to 14.9% for the BWR and 6.2–13.2% for the PWR are considered adequate. The decay time-dependent standard deviations for fission products and light-element activation products (excluding cross-section-dependent error) are listed in Table 6.11. These uncertainties are derived from the standard deviations in data for the fission-product yields, the half-lives, and the recoverable energies of the nuclides.

Also, an extensive discussion of cross-section bias is summarized in Table 6.13. Finally, the four types of errors considered in the analysis are summarized, totaled, and compared with the safety factor equation values in Tables 6.16 and 6.17.

The report also includes a description of LWRARC, a BASIC PC code for easily applying the revised procedure presented in Sect. 5. The executable and data files of the LWRARC code are on the diskette, available at either RSIC or ESTSC (Sect. 7.3).

Heat generation rate tables listed separately for actinides, fission products, and light element activation products are included in Appendix D. These tables are for information purposes only and would not be used directly in the guide's method for determining heat rates.

Plots of the heat rates from dominant nuclides and the total for all of the tabulated cases in Sect. 5 are shown in Appendix E. These plots show more completely the major components to the computed decay heat rates.

Although it was not intended in this study to evaluate other methods of determining decay heat rates, the question may be asked concerning other suitable methods. The authors recognize that there are various other methods for deriving decay heat rates, at least in part or under certain conditions. Some of the appropriate codes or standards for calculating decay heat rates are the following: ORIGEN2,[29] CELL-2[33] or EPRI-CELL[31]/ CINDER2,[32] KORIGEN,[36] the American National Standard for Decay Heat Power in LWRs[39] (ANSI/ANS 5.1-1979), the international decay heat power standard[40] (draft ISO/DIS 10645), and SAS2H/ORIGEN-S. Of course, the use of any proper method requires adequate safety factors that envelop uncertainties such as those discussed in Sect. 6.6.

9 REFERENCES

1. O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," as described in Sect. F7 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I-III (draft November 1993). Available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-545.

2. C. V. Parks, O. W. Hermann, and B. L. Broadhead, "The SCALE Analysis Sequence for LWR Fuel Depletion," *Proc. of ANS/ENS International Topical Meeting*, Pittsburgh, PA, April 28-May 1, 1991, pp. 10.2 3.1–10.2 3-14. See also O. W. Hermann, "SAS2: A Coupled One-Dimensional Depletion and Shielding Analysis Module," as described in Sect. S2 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I-III (draft November 1993). Available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-545.

3. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I-III (draft November 1993). Available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-545.

4. J. C. Ryman et al., *Fuel Inventory and Afterheat Power Studies of Uranium-Fueled Pressurized Water Reactor Fuel Assemblies Using the SAS2 and ORIGEN-S Modules of SCALE with an ENDF/B-V Updated Cross Section Library*, NUREG/CR-2397 (ORNL-CSD-90), Union Carbide Corp., Nucl. Div., Oak Ridge Natl. Lab., September 1982.

5. J. C. Ryman, "ORIGEN-S Data Libraries," as described in Sect. M6 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/ CSD-2/R4), Vols. I-III (Version 4.0 released February 1990). Available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-545.

6. *The Evaluated Nuclear Data File*, Versions IV and V (ENDF/B-IV and -V), available from and maintained by the National Nuclear Data Center at Brookhaven National Laboratory.

7. W. B. Ewbank et al., *Nuclear Structure Data File: A Manual for Preparation of Data Sets*, ORNL-5054, Union Carbide Corp., Nucl. Div., Oak Ridge Natl. Lab., June 1975.

8. T. R. England et al., *Summary of ENDF/B-V Data for Fission Products and Actinides*, EPRI NP-3787 (LA-UR 83-1285, ENDF-322), Electric Power Research Institute, December 1984.

9. W. C. Jordan "SCALE Cross-Section Libraries," as described in Sect. M4 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I-III (Version 4.0 released February 1990). Available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-545.

10. J. M. Creer and J. W. Shupe, Jr., *Development of a Water-Boiloff Spent Fuel Calorimeter System*, PNL-3434, Pacific Northwest Laboratory, May 1981.

11. B. F. Judson et al., *In-Plant Test Measurements for Spent Fuel Storage at Morris Operation*, NEDG-24922-3, General Electric Company, February 1982.

12. M. A. McKinnon et al., *Decay Heat Measurements and Predictions of BWR Spent Fuel*, EPRI NP-4269, Electric Power Research Institute, 1985.

13. F. Schmittroth, G. J. Neely, and J. C. Krogness, *A Comparison of Measured and Calculated Decay Heat for Spent Fuel Near 2.5 Years Cooling Time*, TC-1759, Hanford Engineering Development Laboratory, August 1980.

14. F. Schmittroth, *ORIGEN2 Calculations of PWR Spent Fuel Decay Heat Compared With Calorimeter Data*, HEDL-TME 83-32, Hanford Engineering Development Laboratory, January 1984.

15. M. A. McKinnon et al., *Decay Heat Measurements and Predictions of BWR Spent Fuel*, EPRI NP-4619, Electric Power Research Institute, June 1986.

16. *Point Beach 1 and 2*, Section in **Nuclear Power Experience, Plant Descriptions and Histories**, Vol. PWR-1, March 1974.

17. O. W. Hermann et al., *Multicode Comparison of Selected Source Term Computer Codes*, ORNL/CSD/TM-251, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., April 1989.

18. J. W. Roddy et al., *Physical and Decay Characteristics of Commercial LWR Spent Fuel*, ORNL/TM-9591/V1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., October 1985.

19. *Turkey Point Units 3 and 4*, Section in **Nuclear Power Experience, Plant Descriptions and Histories**, Vol. PWR-1, December 1972.

20. *Cooper Nuclear Station*, Section in **Nuclear Power Experience, Plant Descriptions and Histories**, Vol. BWR-1, March 1974.

21. A. G. Croff et al., *Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code*, ORNL/TM-6051, Union Carbide Corp., Nucl. Div., Oak Ridge Natl. Lab., 1978.

22. A. T. Luksic, *Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal*, Pacific Northwest Laboratory, PNL-6906, Vol. I, June 1989.

23. S. P. Cerne, O. W. Hermann, and R. M. Westfall, *Reactivity and Isotopic Composition of Spent PWR Fuel as a Function of Initial Enrichment, Burnup, and Cooling Time*, ORNL/CSD/TM-244, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., October 1987.

24. J. B. Melehan, *Yankee Core Evaluation Program Final Report*, WCAP-3017-6094, Westinghouse Nuclear Energy Systems, January 1971.

25. R. W. Rasmussen, D. F. French, and H. T. Sneed, "Duke Power Spent Fuel Storage and Transportation Experience," *Trans. Am. Nucl. Soc.* **44**, 479 (1983).

26. *TN-REG Spent Fuel Package Safety Analysis Report for Transport*, Document No. E-7451, Transnuclear, Inc., October 1985.

27. L. E. Wiles et al., *BWR Spent Fuel Storage Cask Performance Test*, PNL-5777, Vol. II, Pacific Northwest Laboratory, June 1986.

28. D. C. Kocher, *Radioactive Decay Data Tables*, DOE/TIC-11026, U.S. Department of Energy, 1981.

29. A. G. Croff, *ORIGEN2 A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Union Carbide Corp., Nucl. Div., Oak Ridge Natl. Lab., July 1980.

30. S. B. Ludwig and J. P. Renier, *Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code*, ORNL/TM-11018, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1989.

31. T. R. England, W. B. Wilson, and M. G. Stamatelatos, *Fission Product Data for Thermal Reactors, Part 2: User's Manual for EPRI-CINDER Code and Data*, EPRI NP-356, Part 2 (LA-6746-MS), Electric Power Research Institute, 1976.

32. T. R. England, *CINDER A One-Point Depletion and Fission Product Program*, WAPD-TM-334, Westinghouse Electric Corp., 1962 (Rev. 1964).

33. M. L. Williams, *Validation of CELL-2 Calculations for Predicting the Isotopic Content of Exposed LWR Fuel*, EPRI NP-6440, Electric Power Research Institute, September 1986.

34. F. J. Sweeney and J. P. Renier, *Sensitivity of Detecting In-Core Vibrations and Boiling in Pressurized Water Reactors Using Ex-Core Neutron Detectors*, NUREG/CR-2996 (ORNL/TM-8549), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1984.

35. C. V. Parks, *Summary Description of the SCALE Modular Code System*, NUREG/CR-5033 (ORNL/CSD/TM-252), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1987.

36. U. Fischer and H. W. Wiese, *Improved and Consistent Determination of the Nuclear Inventory of Spent PWR Fuel on the Basis of Cell-Burnup Methods Using KORIGEN*, KFK 3014 (ORNL-tr-5043), Karlsruhe Nuclear Research Center, Federal Republic of Germany (January 1983). Available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-457.

37. B. L. Broadhead, *Feasibility Assessment of Burnup Credit in the Criticality Analysis of Shipping Casks with Boiling Water Reactor Spent Fuel*, ORNL/CSD/TM-268, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., August 1991.

38. B. Duchemin and C. Nordborg, *DECAY HEAT CALCULATION An International Nuclear Code Comparison*, Nuclear Energy Agency Report NEACRP-319"L," NEANDC-275"U," 1989.

39. *American National Standard for Decay Heat Power in Light Water Reactors*, ANSI/ANS-5.1-1979, American Nuclear Society, 1979.

40. *Nuclear Energy Light Water Reactors Calculation of the Decay Heat Power in Nuclear Fuels*, ISO/DIS 10645, Draft of an International Standard, International Organization for Standardization, May 1990.

41. *Characteristics of Potential Repository Wastes Appendix 2A. Physical Descriptions of LWR Fuel Assemblies*, DOE/RW-0184-R1, U.S. Department of Energy, July 1990.

42. F. D. Hammerling, *DLAG Lagrangian Interpolation for Function of Two Variables*, Chapter I **The Computing Technology Center Numerical Analysis Library**, G. W. Westley and J. A. Watts, Eds., CTC-39, Union Carbide Corp., Nucl. Div., October 1970.

APPENDIX A
DATA AND SAMPLE INPUT TO TABULATED CASES

Table A.1 BWR assembly design description for tabulated cases

Parameter	Data
Assembly general data	
Designer	General Electric
Lattice	8 × 8
Type	Burnable poison
Water temperature, K	558
Water vol-avg density, g-cm^3	0.4323
Number of fuel rods	63
Number of holes	1
Burnable poison element	Gd as Gd_2O_3
Number containing poison	4
Poison content as wt % Gd_2O_3	0.8 4.8[a]
Assembly pitch, cm (in.)	15.24 (6.0)
Shroud (tube) thickness, cm (in.)	0.3048 (0.12)
Shroud outside flat-to-flat, cm (in.)	13.40 (5.275)
Shroud material	Zircaloy
Shroud temperature, K	558
Channel water density, g-cm^{-3}	0.669[b]
Channel water temperature, K	552
Channel avg ^{10}B content,[c] atoms/b-cm	7.15×10^{-6}
Fuel rod data	
Type fuel pellet	UO_2[d]
Pellet stack density, g-cm^{-3}	9.871
Rod pitch, cm (in.)	1.6256 (0.640)
Rod OD, cm (in.)	1.25222 (0.493)
Rod ID, cm (in.)	1.0795 (0.425)
Active fuel length, cm (in.)	375.9 (148)
Effective fuel temperature, K	840
Clad temperature, K	620
Clad material[e]	Zircaloy

[a]Changed Gd_2O_3 content linearly with burnup over this range.
[b]Reduced the 0.743 g-cm^{-3} bottom node density[17] by 10% to account for control rod displacement.
[c]Applied in channel region for boron cruciform; used content producing average k_{eff} of approximately unity.
[d]The uranium isotopes were determined from Tables 5.4 and 3.12.
[e]Clad and other light-element data except for Gd were determined from Table 3.11.

Table A.2 PWR assembly design description for tabulated cases

Parameter	Data
Assembly general data	
Designer	Westinghouse
Lattice	17 × 17
Water temperature, K	570
Water density, g-cm^{-3}	0.7295
Soluble boron, cycle avg, ppm (wt)	550
Number of fuel rods	264
Number of guide tubes	24
Number of instrument tubes	1
Fuel rod data	
Type fuel pellet	UO_2[a]
Pellet stack density, % TD	94.5
Rod pitch, cm (in.)	1.25984 (0.496)
Rod OD, cm (in.)	0.94966 (0.374)
Rod ID, cm (in.)	0.83566 (0.329)
Pellet diameter, cm (in.)	0.81915 (0.3225)
Active fuel length, cm (in.)	365.8 (144)
Effective fuel temperature, K	811
Clad temperature, K	620
Clad material[b]	Zircaloy
Guide tube data[c]	
Inner radius, cm (ID, as in.)	0.5715 (0.45)
Outer radius, cm (OD, as in.)	0.61214 (0.482)
Tube material	Zircaloy

[a]The uranium isotopes were determined from Tables 5.8 and 3.12.
[b]Clad and other light-element data were determined from Table 3.11.
[c]Control rods were considered to be withdrawn during reactor uptime.

Table A.3 Operating history data and fuel isotopic content of BWR cases

Burnup, MWd/kgU	Specific power, kW/kgU	Cycles per case	Cycle time, d Uptime	Downtime	U-isotopic content, wt % ^{234}U	^{235}U	^{236}U	^{238}U
20	12	3	555.55	138.89	0.017	1.900	0.009	98.074
25	12	3	694.44	173.61	0.020	2.300	0.011	97.669
30	12	4	625.00	156.25	0.024	2.700	0.012	97.264
35	12	4	729.17	182.29	0.028	3.100	0.014	96.858
40	12	5	666.67	166.67	0.030	3.400	0.016	96.554
45	12	5	750.00	187.50	0.034	3.800	0.017	96.149
20	20	3	333.33	83.33	0.017	1.900	0.009	98.074
25	20	3	416.67	104.17	0.020	2.300	0.011	97.669
30	20	4	375.00	93.75	0.024	2.700	0.012	97.264
35	20	4	437.50	109.37	0.028	3.100	0.014	96.858
40	20	5	400.00	100.00	0.030	3.400	0.016	96.554
45	20	5	450.00	112.50	0.034	3.800	0.017	96.149
20	30	3	222.22	55.56	0.017	1.900	0.009	98.074
25	30	3	277.78	69.44	0.020	2.300	0.011	97.669
30	30	4	250.00	62.50	0.024	2.700	0.012	97.264
35	30	4	291.67	72.92	0.028	3.100	0.014	96.858
40	30	5	266.67	66.67	0.030	3.400	0.016	96.554
45	30	5	300.00	75.00	0.034	3.800	0.017	96.149

Table A.4 Operating history data and fuel isotopic content of PWR cases

Burnup, MWd/kgU	Specific power, kW/kgU	Cycles per case	Cycle time, d Uptime	Downtime	U-isotopic content, wt % ^{234}U	^{235}U	^{236}U	^{238}U
25	18	3	462.96	115.74	0.021	2.400	0.011	97.568
30	18	3	555.56	138.89	0.024	2.800	0.012	97.164
35	18	4	486.11	121.53	0.028	3.200	0.014	96.758
40	18	4	555.56	138.89	0.032	3.600	0.016	96.352
45	18	5	500.00	125.00	0.034	3.900	0.017	96.049
50	18	5	555.56	138.89	0.037	4.200	0.019	95.744
25	28	3	297.62	74.40	0.021	2.400	0.011	97.568
30	28	3	357.14	89.29	0.024	2.800	0.012	97.164
35	28	4	312.50	78.12	0.028	3.200	0.014	96.758
40	28	4	357.14	89.29	0.032	3.600	0.016	96.352
45	28	5	321.43	80.36	0.034	3.900	0.017	96.049
50	28	5	357.14	89.29	0.037	4.200	0.019	95.744
25	40	3	208.33	52.08	0.021	2.400	0.011	97.568
30	40	3	250.00	62.50	0.024	2.800	0.012	97.164
35	40	4	218.75	54.69	0.028	3.200	0.014	96.758
40	40	4	250.00	62.50	0.032	3.600	0.016	96.352
45	40	5	225.00	56.25	0.034	3.900	0.017	96.049
50	40	5	250.00	62.50	0.037	4.200	0.019	95.744

The following is the entire input for the 12-kW/kgU, 20-MWd/kgU BWR case using SAS2H to generate the burnup-dependent cross-section libraries, ORIGEN-S to provide detailed decay heat rate tables, and PLORIGEN to plot selected results.

```
=SAS2     PARM='HALT03,SKIPSHIPDATA'
BWR 12 KW/KGU 20 MWD/KGU, NRC SPENT-FUEL HEAT RATE REG-GUIDE 3.54, 1990
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF FUEL-PIN-UNIT-CELL:
'
27BURNUPLIB    LATTICECELL
UO2  1  DEN=9.871 1 840
 92234 0.017 92235 1.900 92236 0.009 92238 98.074  END
'HOT-DEN=10.96(THE THEOR.-DEN)*0.94(%-TD)*(.416/.425)**2 COLD/HOT DIAM
CO-59  3 0 1-20 558  END
ZR-94  1 0 1-20 840  END
TC-99  1 0 1-20 840  END
RU-106 1 0 1-20 840  END
RH-103 1 0 1-20 840  END
RH-105 1 0 1-20 840  END
XE-131 1 0 1-20 840  END
CS-134 1 0 1-20 840  END
CE-144 1 0 1-20 840  END
PR-143 1 0 1-20 840  END
ND-143 1 0 1-20 840  END
ND-145 1 0 1-20 840  END
ND-147 1 0 1-20 840  END
PM-147 1 0 1-20 840  END
SM-149 1 0 1-20 840  END
SM-151 1 0 1-20 840  END
SM-152 1 0 1-20 840  END
EU-153 1 0 1-20 840  END
EU-154 1 0 1-20 840  END
EU-155 1 0 1-20 840  END
ZIRCALLOY  2 1  620   END
H2O  3 DEN=0.4323  1  558 END
'
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF LARGER-UNIT-CELL:
'
UO2  9  DEN=9.871 1 840
 92234 0.017 92235 1.900 92236 0.009 92238 98.074  END
ARBM-GDBURN  9.871 7 0 1 1
           64154  2.18  64155 14.80  64156 20.47
           64157 15.65  64158 24.84  64160 21.86
            8016 150.0  9  0.008  840  END
'                ....ABOVE IS 0.8 WT % GADOLINIUM (AS GD2-OX3) IN THE
'                    BURNABLE POISON PINS OF BWR ASSEMBLY....
ZIRCALLOY  10  1  588 END
'                ....ABOVE IS ZIRCALLOY CASING AROUND ASSEMBLY
B-10    11 0 7.15-6  552  END
H2O        11 0.669  552 END
'                ....ABOVE IS CHANNEL MODERATOR AT HIGHER DENSITY
END COMP
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'      FUEL-PIN-CELL GEOMETRY:
'
SQUAREPITCH   1.6256 1.0795  1 3  1.25222  2  END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
```

```
'
'    ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=63  FUELNGHT=1993.26  NCYCLES=3  NLIB/CYC=1
PRINTLEVEL=4  LIGHTEL=10  INPLEVEL=2  NUMZONES=6   END
9 0.53975 2 0.62611 3 0.91715 500 3.6398 10 3.8103 11 4.3261
'           ..THESE MIXTURES & RADII PLACE GADOLINIUM PIN AT CENTER
'             OF 1/4 OF ASSEMBLY FUEL, CASING & CHANNEL MOD.
'(COMMENT)  POWER=12    BURN=555.55  DOWN=DDDDDD   END
'(COMMENT)  POWER=12    BURN=555.55  DOWN=DDDDDD   END
POWER=12    BURN=555.55  DOWN=138.89   END
POWER=12    BURN=555.55  DOWN=138.89   END
POWER=12    BURN=555.55 DOWN=3652.5    END
  H 16.4 B 0.068
  O 265  CR  2.4  MN  0.15
  FE 6.6 CO 0.024 NI  2.4
  ZR 516 SN  8.7
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
END
=ORIGENS
0$$ 58 A3 57 A11 19 E 1T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
54$$ 2 E 3$$ 33 0 1   A16 2 E 2T
4T
56$$ 10 A5 10 3 A9 1 A13 15 4 3 0 4 1 E 57** A3 10 0.33333  E 5T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
1000 KG U LOADED
58** 4R12 1-22 0 4R12
60** 8I69.444  694.44
66$$ A1 2 A5 2 E  76$$ 50100 77** 0.5963E-07 78$$ 1
73$$ 922340 922350 922360 922380 640000
 10000  50000  80000  240000  250000
260000  270000  280000  400000  500000
74** 0.017+4 1.900+4 0.009+4 98.074+4 0.50000+3
16.4+3  0.068+3  265+3  2.4+3  0.15+3
6.6+3  0.024+3  2.4+3  516+3  8.7+3
75$$ 4R2 11R4  6T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
'
'
'  COMMENT SPACES OVER -2ND SUBCASE OF CYC-5 CASES
'
'
'
'
'
3$$ 34 0 1   E 2T
4T
56$$ 10 A5 10 3 A9 1 10 A17 4 1 E 57** A3 10 0.33333  E 5T
44** A15 1.248  E
58** 4R12 0 4R12 1-22
60** 8I69.444  694.44
66$$ A1 2 A5 2 E 76$$ 50100 77** 0.6095E-07 78$$ 1 6T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
'
'
'  COMMENT SPACES OVER 4RTH SUBCASE OF CYC-5 CASES
'
'
'
'
'
3$$ 35 0 1   E 2T
4T
56$$ 10 A5 10 A9 1 10 A17 4 1 E 57** A3 10 0.33333  E 5T
```

```
44** A15 1.248  E
58** F12
60** 8I61.728  555.55
66$$ A1 2 A5 2 E 76$$ 50100 77** 0.6583E-07 78$$ 1 6T
56$$ A5 5 A10 -10 A14 5 3 57 4 E 54$$ A6 12 0 1 E 5T
HEAT RATE OF 12KW-20MWD/KGU BWR SPENT FUEL
1 KILOGRAM U (AS LOADED)
61** F0.01  65$$ A12 1 A33 1 A54 1 E 79** F1-3
60** 1 1.4 2 2.8 4   5 7 10 15 20 6T
56$$ 0 -10 A10 1 E 5T
56$$ A5 3 A10 10 A14 5 2 57 4 E 57** 20 E 5T
1 KILOGRAM U (AS LOADED)
61** F0.001 65$$ A12 1 A33 1 A54 1 E
60** 25 7I 30 110 6T
56$$ 0 -10 A10 1 E 5T
56$$ F0 5T
END
=PLORIGEN
PLOTDEF=NO   NUMUNIT=19   NPRINT=58   NCOMP=15 MINPOS=1 MAXP=20
TYXAXIS=TLOG  XHEAD=TIME SINCE DISCHARGED, YEARS
YHEADING=CURIES / KG URANIUM LOADED
TITLE=SPENT-FUEL RADIOACTIVITY / KGU   TMIN=1 TMAX=110  END
PLOTDEF=NO   NUMUNIT=19   NCOMP=15  TYPLOT=TOTWATTS
TYXAXIS=TLOG  XHEAD=TIME SINCE DISCHARGED, YEARS
YHEADING=WATTS / KG URANIUM LOADED
MINPOS=1 MAXP=20  TITLE=12KW-20MWD/KGU BWR FUEL AFTERHEAT
TMIN=1 TMAX=30  NPRINT=58   END
END
```

The following is the entire input for the 18-kW/kgU, 25-MWd/kgU PWR case using SAS2H to generate the burnup-dependent cross-section libraries, ORIGEN-S to provide detailed decay heat rate tables, and PLORIGEN to plot selected results.

```
=SAS2     PARM='HALT03,SKIPSHIPDATA'
PWR 18 KW/KGU 25 MWD/KGU, NRC SPENT-FUEL HEAT RATE REG-GUIDE 3.54, 1990
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF FUEL-PIN-UNIT-CELL:
'
27BURNUPLIB    LATTICECELL
UO2  1 0.945 811 92234 0.021 92235 2.4 92236 0.011 92238 97.568  END
CO-59  3 0 1-20 570  END
ZR-94  1 0 1-20 811  END
TC-99  1 0 1-20 811  END
RU-106 1 0 1-20 811  END
RH-103 1 0 1-20 811  END
RH-105 1 0 1-20 811  END
XE-131 1 0 1-20 811  END
CS-134 1 0 1-20 811  END
CE-144 1 0 1-20 811  END
PR-143 1 0 1-20 811  END
ND-143 1 0 1-20 811  END
ND-145 1 0 1-20 811  END
ND-147 1 0 1-20 811  END
PM-147 1 0 1-20 811  END
SM-149 1 0 1-20 811  END
SM-151 1 0 1-20 811  END
SM-152 1 0 1-20 811  END
EU-153 1 0 1-20 811  END
EU-154 1 0 1-20 811  END
EU-155 1 0 1-20 811  END
ZIRCALLOY  2 1  620   END
H2O  3 DEN=0.7295  1  570 END
ARBM-BORMOD  0.7295 1 1 0 0 5000 100 3 550.0E-6  570  END
'
'  550 PPM BORON (WT) IN MODERATOR
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
END COMP
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'      FUEL-PIN-CELL GEOMETRY:
'
SQUAREPITCH   1.25984 0.81915 1 3 0.94966 2 0.83566 0  END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'    ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=264 FUELNGHT=787.28 NCYCLES=3  NLIB/CYC=1
PRINTLEVEL=4 LIGHTEL=9  INPLEVEL=1 ORTUBE=0.61214 SRTUBE=0.5715
NUMINSTR=1  END
POWER=18  BURN=462.96  DOWN=115.74   END
POWER=18  BURN=462.96  DOWN=115.74   END
POWER=18  BURN=462.96  DOWN=3652.5   END
  O 135  CR  5.9  MN  0.33
  FE 13. CO 0.075 NI  9.9
  ZR 221 NB  0.71 SN  3.6
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
END
```

```
=ORIGENS
0$$ 58 A3 57 A11 19 E 1T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
3$$ 33 0 1 A5 58 A16 2 E 2T
4T
56$$ 10 A5 10 3 A13 13 4 3 0 4 1 E 57** A3 10 0.333333  E 5T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
1000 KG U LOADED
58** 4R18 1-22 0 4R18
60** 8I57.870  578.70
66$$ A1 2 A5 2 E
73$$ 922340 922350 922360 922380
80000 240000 250000 260000 270000 280000
400000 410000 500000
74** 0.021+4 2.400+4 0.011+4 97.568+4
135+3 5.9+3 0.33+3 13+3 75 9.9+3 221+3 0.71+3 3.6+3
75$$ 4R2 9R4 6T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
3$$ 34 0 1 A5 58 E 2T
4T
56$$ 10 A5 10 3 A10 10 A17 4 1 E 57** A3 10 0.333333  E 5T
58** 4R18 0 4R18 1-22
60** 8I57.870  578.70
66$$ A1 2 A5 2 E 6T
 NRC REGULATORY GUIDE 3.54 HEAT RATE PROJECT, O. W. HERMANN 1989-90
3$$ 35 0 1 A5 58 E 2T
4T
56$$ 10 A5 10 A10 10 A17 4 1 E 57** A3 10 0.333333  E 5T
58** F18
60** 8I51.440  462.96
66$$ A1 2 A5 2 E 6T
56$$ A5 5 A10 -10 A14 5 3 57 4 E 54$$ A6 12 0 1 E 5T
HEAT RATE OF 18KW-25MWD/KGU PWR SPENT FUEL
1 KILOGRAM U (AS LOADED)
61** F0.01  65$$ A12 1 A33 1 A54 1 E 79** F1-3
60** 1 1.4 2 2.8 4   5 7 10 15 20 6T
56$$ 0 -10 A10 1 E 5T
56$$ A5 3 A10 10 A14 5 2 57 4 E 57** 20 E 5T
1 KILOGRAM U (AS LOADED)
61** F0.001 65$$ A12 1 A33 1 A54 1 E
60** 25 7I 30 110 6T
56$$ 0 -10 A10 1 E 5T
56$$ F0 5T
END
=PLORIGEN
PLOTDEF=NO   NUMUNIT=19   NPRINT=58   NCOMP=15 MINPOS=1 MAXP=20
TYXAXIS=TLOG  XHEAD=TIME SINCE DISCHARGED, YEARS
YHEADING=CURIES / KG URANIUM LOADED
TITLE=SPENT-FUEL RADIOACTIVITY / KGU   TMIN=1 TMAX=110  END
PLOTDEF=NO   NUMUNIT=19   NCOMP=15  TYPLOT=TOTWATTS
TYXAXIS=TLOG  XHEAD=TIME SINCE DISCHARGED, YEARS
YHEADING=WATTS / KG URANIUM LOADED
MINPOS=1 MAXP=20  TITLE=18KW-25MWD/KGU PWR FUEL AFTERHEAT
TMIN=1 TMAX=30  NPRINT=58   END
END
```

APPENDIX B
SAMPLE CASE USING HEAT GENERATION RATE TABLES

A BWR fuel assembly with an average fuel enrichment of 2.6 wt % ^{235}U was in the reactor for four cycles. Determine its final heat generation rate with safety factors, using the method in the guide, at 4.2 years after discharge. Adequate details of the operating history associated with the fuel assembly are shown in Table B.1.

Table B.1 Sample case operating history

Relative fuel cycle	Time from startup of fuel, days		Accumulated burnup (best maximum estimate), MWd/kgU
	Cycle startup	Cycle shutdown	
1	0	300	8.1
2	340	590	14.7
3	630	910	20.9
4	940	1240	26.3

Note that the output of the LWRARC code for this case is shown in the first case of Appendix C.

Using Sect. 5.1:

The following were given in the sample case (see Sect. 5.1 for definitions):

$$T_{res} = 1240 \text{ d},$$

$$B_{tot} = 26.30 \text{ MWd/kgU},$$

$$T_c = 4.2 \text{ y},$$

$$E_s = 2.6 \text{ wt \% } {}^{235}\text{U}.$$

Compute T_e, B_e, P_e, $T_{e\text{-}1}$, $P_{e\text{-}1}$, and P_{ave} from Sect. 5.1 and Eqs. (2) through (4):

$$T_e = 1240 - 940 = 300 \text{ d},$$

$$B_e = 26{,}300 - 20{,}900 = 5{,}400 \text{ kWd/kgU},$$

$$P_e = (26{,}300 - 20{,}900)/300 = 18.00 \text{ kW/kgU},$$

$$T_{e\text{-}1} = 940 - 630 = 310 \text{ d},$$

$$B_{e\text{-}1} = 20{,}900 - 14{,}700 = 6{,}200 \text{ kWd/kgU},$$

$$P_{e\text{-}1} = 6{,}200/[0.8(310)] = 25.0 \text{ kW/kgU},$$

$$P_{ave,e\text{-}1} = 20{,}900/[0.8(940)] = 27.793 \text{ kW/kgU},$$

$$P_{ave} = 26{,}300/[300 + 0.8\,(940)] = 25.00 \text{ kW/kgU}.$$

Using Sect. 5.2

Now, p_{tab} should be determined from P_{ave}, B_{tot}, and T_c, as described in Sect. 5.2.1. First, select the nearest heat rate values in Tables 5.1 and 5.2 for the following limits:

$$P_L = 20 \le P_{ave} \le P_H = 30,$$

$$B_L = 25 \le B_{tot} \le B_H = 30,$$

$$T_L = 4 \le T_c \le T_H = 5.$$

Next, use the prescribed interpolation procedure for computing p_{tab} from the tabular data. Although the order is optional, the example here interpolates between specific powers, burnups, and, then, cooling times. Denote the heat rate, p, as a function of specific power, burnup, and cooling time by $p(P,B,T)$. Then, the table values at P_L and P_H for B_L and T_L are

$$p(P_L, B_L, T_L) = p(20,25,4) = 1.549,$$

$$p(P_H, B_L, T_L) = p(30,25,4) = 1.705.$$

First, interpolate the above heat rates to P_{ave} using

$$p(P_{ave},25,4) = p(20,25,4) + F_p[p(30,25,4) - p(20,25,4)],$$

where

$$F_p = (P_{ave} \quad P_L)/(P_H \quad P_L) = 0.5.$$

The result at $p(P_{ave},25,4)$ is

$$p(P_{ave},25,4) = 1.549 + 0.5\,(1.705 - 1.549) = 1.627.$$

The other three values at P_{ave} are computed with a similar method:

$$p(P_{ave},30,4) = 1.827 + 0.5\,(2.016 - 1.827) = 1.9215,$$

$$p(P_{ave},25,5) = 1.293,$$

$$p(P_{ave},30,5) = 1.553.$$

These are heat rates at the burnup and time limits.

Second, interpolate each of the above pairs of heat rates to B_{tot} from the values at B_L and B_H:

$$F_B = (B_{tot} \quad B_L)/(B_H \quad B_L) = 0.26,$$

$$p(P_{ave}, B_{tot}, 4) = 1.627 + 0.26\,(1.9215 - 1.627) = 1.7036,$$

$$p(P_{ave}, B_{tot}, 5) = 1.3606.$$

Third, compute the heat rate at T_c from the above values at T_L and T_H by an interpolation that is logarithmic in heat rate and linear in time:

$$F_T = (T_c \quad T_L)/(T_H \quad T_L) = 0.2$$

$$log[p(P_{ave}, B_{tot}, T_c)] = log\ 1.7036 + 0.2\,(log\ 1.3606 - log\ 1.7036) = 0.2118.$$

$$p_{tab} = p(P_{ave}, B_{tot}, T_c) = 10^{0.2118} = 1.629 \text{ W/kgU}.$$

With the value for p_{tab}, the formulae of Sects. 5.2.2– 5.2.6 can be used to determine p_{final}. Since $T_c < 7$ y, use Eqs. (8) through (11) to calculate the short cooling time factors:

$$R = P_e/P_{ave} - 1 = (18/25) - 1 = -0.28,$$

$$f_7 = 1 + [0.25(-0.28)]/4.2 = 0.983,$$

$$R' = P_{e\text{-}1}/P_{ave,e\text{-}1} - 1 = -0.1005,$$

$$f'_7 = 1 + [0.08(-0.1005)]/4.2 = 0.998.$$

Since $P_{ave} < P_H = P_{max}$, the excess power factor, f_p, is unity. Interpolating Table 5.4 enrichments to obtain the enrichment associated with the burnup yields

$$E_{tab} = 2.3 + (2.7 - 2.3)(26.3 - 25)/(30 - 25) = 2.404.$$

The enrichment factor f_e is then calculated using Eq. (15):

$$f_e = 1 + 0.01\,(8.376)(1 - 2.6/2.404) = 0.993,$$

because $E_s > E_{tab}$.

The safety factor, S, for a BWR is given in Eq. (16):

$$S = 6.4 + 0.15\,(26.3 - 20) + 0.044\,(4.2 - 1) = 7.49\%.$$

Then, using Eq. (18),

$$p_{final} = (1 + 0.01\,S)\, f_7 f'_7 f_p f_e p_{tab},$$

with the above adjustment factors and p_{tab} yields

$$p_{final} = 1.0749 \times 0.983 \times 0.998 \times 1 \times 0.993 \times 1.629$$

$$= 1.0749 \times 1.587 = 1.706 \text{ W/kgU}.$$

Thus, the final heat generation rate, including the safety factor, of the given fuel assembly is 1.706 W/kgU.

APPENDIX C
LWRARC CODE SAMPLE RESULTS

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          ***************************************************
          *                                                 *
          * HEAT GENERATION RATE OF BWR SPENT-FUEL ASSEMBLY *
          *        (PERTAINING TO USNRC GUIDE 3.54)         *
          *        LWRARC CODE CREATION DATE : 05/30/94     *
          *                                                 *
          ***************************************************
 TITLE: REG GUIDE SAMPLE CASE FOR BWR - AS HAND CALC.

                     .ASSEMBLY INPUT DESCRIPTION.
_______________________________________________________________________

PARAMETER     DATA, UNITS              DEFINITION, IN GUIDE

  E(S)       2.600 WT-% U-235   INITIAL FUEL ENRICHMENT
  KGU(S)     0.000 KGU/ASSY     ASSEMBLY FUEL LOADING
  T(C)       4.200 YEARS        ASSEMBLY COOLING TIME
  B(TOT)    26.300 MWD/KGU      BURNUP, BEST MAXIMUM ESTIMATE
  T(RES)  1240.000 DAYS         ASSEMBLY RESIDENCE TIME IN REACTOR
  T(E)     300.000 DAYS         LAST CYCLE TIME, STARTUP-DISCHARGE
  B(E)       5.400 MWD/KGU      LAST CYCLE BURNUP
  T(E-1)   310.000 DAYS         NEXT-TO-LAST-CYC TIME, STARTUP-STARTUP
  B(E-1)     6.200 MWD/KGU      NEXT-TO-LAST-CYCLE BURNUP
_______________________________________________________________________

                    .CORRECTION FACTORS COMPUTED.
_______________________________________________________________________

  F(P)              1.000         EXCESS POWER ADJUSTMENT
  F(E)              0.993         INITIAL U-235 ENRICHMENT CORRECTION
  F(7)              0.983         LAST CYCLE HISTORY CORRECTION
  F-PRIME(7)        0.998         NEXT-TO-LAST-CYC HISTORY CORRECTION
  F-SAFE            1.075         SAFETY FACTOR APPLIED TO RESULT
_______________________________________________________________________

 *CASE DUPLICATES HAND COMPUTATION OF GUIDE.

                   .HEAT GENERATION RESULTS, W/KGU.
           __________________________________________________

           AFTER TABLE INTERPOLATION                    1.629
           AFTER ALL CORRECTIONS EXCEPT SAFETY FACTOR   1.588
           AFTER SAFETY FACTOR INCLUDED                 1.706
           __________________________________________________

   --- FINAL HEAT GENERATION RATE IS    1.706 WATTS PER KILOGRAM U LOADED
```

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                   ***************************************************
                   *                                                 *
                   * HEAT GENERATION RATE OF PWR SPENT-FUEL ASSEMBLY *
                   *         (PERTAINING TO USNRC GUIDE 3.54)        *
                   *         LWRARC CODE CREATION DATE : 05/30/94    *
                   *                                                 *
                   ***************************************************
 TITLE: TRES=1944, T(E)=556, BU=30.001, T(C)=10.01, 2.8 WT%

                        .ASSEMBLY INPUT DESCRIPTION.
______________________________________________________________________

PARAMETER     DATA, UNITS               DEFINITION, IN GUIDE

  E(S)       2.800 WT-% U-235   INITIAL FUEL ENRICHMENT
  KGU(S)     0.000 KGU/ASSY     ASSEMBLY FUEL LOADING
  T(C)      10.010 YEARS        ASSEMBLY COOLING TIME
  B(TOT)    30.001 MWD/KGU      BURNUP, BEST MAXIMUM ESTIMATE
  T(RES)  1944.000 DAYS         ASSEMBLY RESIDENCE TIME IN REACTOR
  T(E)     556.000 DAYS         LAST CYCLE TIME, STARTUP-DISCHARGE
  B(E)      10.000 MWD/KGU      LAST CYCLE BURNUP
  T(E-1)   694.000 DAYS         NEXT-TO-LAST-CYC TIME, STARTUP-STARTUP
  B(E-1)    10.000 MWD/KGU      NEXT-TO-LAST-CYCLE BURNUP
______________________________________________________________________

                       .CORRECTION FACTORS COMPUTED.
______________________________________________________________________

  F(P)                1.000       EXCESS POWER ADJUSTMENT
  F(E)                1.000       INITIAL U-235 ENRICHMENT CORRECTION
  F(7)                1.000       LAST CYCLE HISTORY CORRECTION
  F-PRIME(7)          1.000       NEXT-TO-LAST-CYC HISTORY CORRECTION
  F-SAFE              1.070       SAFETY FACTOR APPLIED TO RESULT
______________________________________________________________________

 *CASE DUPLICATES HAND COMPUTATION OF GUIDE.

                      .HEAT GENERATION RESULTS, W/KGU.
           __________________________________________________

           AFTER TABLE INTERPOLATION                    1.044
           AFTER ALL CORRECTIONS EXCEPT SAFETY FACTOR   1.044
           AFTER SAFETY FACTOR INCLUDED                 1.116
           __________________________________________________

   --- FINAL HEAT GENERATION RATE IS    1.116 WATTS PER KILOGRAM U LOADED
```

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**************************************************
*                                                *
* HEAT GENERATION RATE OF PWR SPENT-FUEL ASSEMBLY *
*        (PERTAINING TO USNRC GUIDE 3.54)        *
*        LWRARC CODE CREATION DATE : 05/30/94    *
*                                                *
**************************************************
 TITLE: POINT BEACH 2, ASSY C-52, DECAY HEAT=723.5 W/ASSY MEAS

                     .ASSEMBLY INPUT DESCRIPTION.
______________________________________________________________________

PARAMETER     DATA, UNITS              DEFINITION, IN GUIDE

  E(S)       3.397 WT-% U-235   INITIAL FUEL ENRICHMENT
  KGU(S)   386.000 KGU/ASSY     ASSEMBLY FUEL LOADING
  T(C)       4.476 YEARS        ASSEMBLY COOLING TIME
  B(TOT)    31.914 MWD/KGU      BURNUP, BEST MAXIMUM ESTIMATE
  T(RES)  1675.000 DAYS         ASSEMBLY RESIDENCE TIME IN REACTOR
  T(E)     339.000 DAYS         LAST CYCLE TIME, STARTUP-DISCHARGE
  B(E)       8.797 MWD/KGU      LAST CYCLE BURNUP
  T(E-1)   465.000 DAYS         NEXT-TO-LAST-CYC TIME, STARTUP-STARTUP
  B(E-1)    12.316 MWD/KGU      NEXT-TO-LAST-CYCLE BURNUP
______________________________________________________________________

                     .CORRECTION FACTORS COMPUTED.
______________________________________________________________________

  F(P)                1.000       EXCESS POWER ADJUSTMENT
  F(E)                0.985       INITIAL U-235 ENRICHMENT CORRECTION
  F(7)                1.024       LAST CYCLE HISTORY CORRECTION
  F-PRIME(7)          1.025       NEXT-TO-LAST-CYC HISTORY CORRECTION
  F-SAFE              1.068       SAFETY FACTOR APPLIED TO RESULT
______________________________________________________________________

 *CASE USES MORE PRECISE INTERPOLATIONS THAN THAT OF METHOD IN GUIDE.

                    .HEAT GENERATION RESULTS, W/KGU.
           __________________________________________________

           AFTER TABLE INTERPOLATION                    1.837
           AFTER ALL CORRECTIONS EXCEPT SAFETY FACTOR   1.900
           AFTER SAFETY FACTOR INCLUDED                 2.029
           __________________________________________________

   --- FINAL HEAT GENERATION RATE IS    2.029 WATTS PER KILOGRAM U LOADED
                                 OR    783.1 WATTS PER ASSEMBLY
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          ***************************************************
          *                                                 *
          * HEAT GENERATION RATE OF BWR SPENT-FUEL ASSEMBLY *
          *         (PERTAINING TO USNRC GUIDE 3.54)        *
          *         LWRARC CODE CREATION DATE : 05/30/94    *
          *                                                 *
          ***************************************************
 TITLE: COOPER BWR 4-CYC, ASSY CZ528, 297.6 W/ASSEMBLY MEAS

                    .ASSEMBLY INPUT DESCRIPTION.
______________________________________________________________________

PARAMETER     DATA, UNITS               DEFINITION, IN GUIDE

  E(S)       2.500 WT-% U-235   INITIAL FUEL ENRICHMENT
  KGU(S)   190.500 KGU/ASSY     ASSEMBLY FUEL LOADING
  T(C)       3.521 YEARS        ASSEMBLY COOLING TIME
  B(TOT)    25.715 MWD/KGU      BURNUP, BEST MAXIMUM ESTIMATE
  T(RES)  2483.000 DAYS         ASSEMBLY RESIDENCE TIME IN REACTOR
  T(E)     317.000 DAYS         LAST CYCLE TIME, STARTUP-DISCHARGE
  B(E)       4.110 MWD/KGU      LAST CYCLE BURNUP
  T(E-1)   394.000 DAYS         NEXT-TO-LAST-CYC TIME, STARTUP-STARTUP
  B(E-1)     2.692 MWD/KGU      NEXT-TO-LAST-CYCLE BURNUP
______________________________________________________________________

                   .CORRECTION FACTORS COMPUTED.
______________________________________________________________________

  F(P)                1.000       EXCESS POWER ADJUSTMENT
  F(E)                0.995       INITIAL U-235 ENRICHMENT CORRECTION
  F(7)                1.006       LAST CYCLE HISTORY CORRECTION
  F-PRIME(7)          0.993       NEXT-TO-LAST-CYC HISTORY CORRECTION
  F-SAFE              1.074       SAFETY FACTOR APPLIED TO RESULT
______________________________________________________________________

 *CASE USES MORE PRECISE INTERPOLATIONS THAN THAT OF METHOD IN GUIDE.

                  .HEAT GENERATION RESULTS, W/KGU.
           __________________________________________________

           AFTER TABLE INTERPOLATION                    1.578
           AFTER ALL CORRECTIONS EXCEPT SAFETY FACTOR   1.568
           AFTER SAFETY FACTOR INCLUDED                 1.684
           __________________________________________________

   --- FINAL HEAT GENERATION RATE IS     1.684 WATTS PER KILOGRAM U LOADED
                                 OR     320.8 WATTS PER ASSEMBLY
```

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***************************************************
*                                                 *
* HEAT GENERATION RATE OF BWR SPENT-FUEL ASSEMBLY *
*         (PERTAINING TO USNRC GUIDE 3.54)        *
*         LWRARC CODE CREATION DATE : 05/30/94    *
*                                                 *
***************************************************
 TITLE: COOPER BWR 3-CYC, ASSY CZ331, 162.8 W/ASSEMBLY MEASURED

                      .ASSEMBLY INPUT DESCRIPTION.
______________________________________________________________________

PARAMETER     DATA, UNITS               DEFINITION, IN GUIDE

  E(S)       2.500 WT-% U-235   INITIAL FUEL ENRICHMENT
  KGU(S)   190.500 KGU/ASSY     ASSEMBLY FUEL LOADING
  T(C)       6.486 YEARS        ASSEMBLY COOLING TIME
  B(TOT)    21.332 MWD/KGU      BURNUP, BEST MAXIMUM ESTIMATE
  T(RES)  1367.000 DAYS         ASSEMBLY RESIDENCE TIME IN REACTOR
  T(E)     164.000 DAYS         LAST CYCLE TIME, STARTUP-DISCHARGE
  B(E)       2.962 MWD/KGU      LAST CYCLE BURNUP
  T(E-1)   337.000 DAYS         NEXT-TO-LAST-CYC TIME, STARTUP-STARTUP
  B(E-1)     5.495 MWD/KGU      NEXT-TO-LAST-CYCLE BURNUP
______________________________________________________________________

                     .CORRECTION FACTORS COMPUTED.
______________________________________________________________________

  F(P)              1.000        EXCESS POWER ADJUSTMENT
  F(E)              0.983        INITIAL U-235 ENRICHMENT CORRECTION
  F(7)              0.998        LAST CYCLE HISTORY CORRECTION
  F-PRIME(7)        1.003        NEXT-TO-LAST-CYC HISTORY CORRECTION
  F-SAFE            1.068        SAFETY FACTOR APPLIED TO RESULT
______________________________________________________________________

 *CASE USES MORE PRECISE INTERPOLATIONS THAN THAT OF METHOD IN GUIDE.

                    .HEAT GENERATION RESULTS, W/KGU.
           __________________________________________________

           AFTER TABLE INTERPOLATION                    0.865
           AFTER ALL CORRECTIONS EXCEPT SAFETY FACTOR   0.852
           AFTER SAFETY FACTOR INCLUDED                 0.910
           __________________________________________________

   --- FINAL HEAT GENERATION RATE IS    0.910 WATTS PER KILOGRAM U LOADED
                                 OR     173.3 WATTS PER ASSEMBLY
```

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***************************************************
*                                                 *
* HEAT GENERATION RATE OF PWR SPENT-FUEL ASSEMBLY *
*         (PERTAINING TO USNRC GUIDE 3.54)        *
*         LWRARC CODE CREATION DATE : 05/30/94    *
*                                                 *
***************************************************
 TITLE: TURKEY PT. 3, ASSY D-15, TC=2077 D, 625 W/ASSEMBLY MEAS

                      .ASSEMBLY INPUT DESCRIPTION.
______________________________________________________________________

PARAMETER     DATA, UNITS               DEFINITION, IN GUIDE

  E(S)       2.557 WT-% U-235   INITIAL FUEL ENRICHMENT
  KGU(S)   456.100 KGU/ASSY     ASSEMBLY FUEL LOADING
  T(C)       5.687 YEARS        ASSEMBLY COOLING TIME
  B(TOT)    28.152 MWD/KGU      BURNUP, BEST MAXIMUM ESTIMATE
  T(RES)  1073.000 DAYS         ASSEMBLY RESIDENCE TIME IN REACTOR
  T(E)     312.000 DAYS         LAST CYCLE TIME, STARTUP-DISCHARGE
  B(E)       8.920 MWD/KGU      LAST CYCLE BURNUP
  T(E-1)   389.000 DAYS         NEXT-TO-LAST-CYC TIME, STARTUP-STARTUP
  B(E-1)     9.752 MWD/KGU      NEXT-TO-LAST-CYCLE BURNUP
______________________________________________________________________

                     .CORRECTION FACTORS COMPUTED.
______________________________________________________________________

  F(P)              1.000         EXCESS POWER ADJUSTMENT
  F(E)              1.009         INITIAL U-235 ENRICHMENT CORRECTION
  F(7)              0.997         LAST CYCLE HISTORY CORRECTION
  F-PRIME(7)        1.000         NEXT-TO-LAST-CYC HISTORY CORRECTION
  F-SAFE            1.066         SAFETY FACTOR APPLIED TO RESULT
______________________________________________________________________

 *CASE USES MORE PRECISE INTERPOLATIONS THAN THAT OF METHOD IN GUIDE.

                    .HEAT GENERATION RESULTS, W/KGU.
           __________________________________________________

           AFTER TABLE INTERPOLATION                    1.386
           AFTER ALL CORRECTIONS EXCEPT SAFETY FACTOR   1.394
           AFTER SAFETY FACTOR INCLUDED                 1.487
           __________________________________________________

   --- FINAL HEAT GENERATION RATE IS    1.487 WATTS PER KILOGRAM U LOADED
                                 OR    678.2 WATTS PER ASSEMBLY

     * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
     *        CONGRATULATIONS ... YOU HAVE COMPLETED LWRARC             *
     *                                                                  *
     * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
```

APPENDIX D

ACTINIDE, FISSION PRODUCT, AND LIGHT-ELEMENT TABLES

Table D.1 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 12 kW/kgU, Set 1

Burnup = 20 MWd/kgU			Cooling Time, years	Burnup = 25 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
4.747E-02	3.374E-01	3.762E+00	1.0	4.993E-02	4.850E-01	4.141E+00
3.410E-02	2.293E-01	2.869E+00	1.4	3.642E-02	3.323E-01	3.205E+00
2.841E-02	1.542E-01	2.066E+00	2.0	3.054E-02	2.257E-01	2.354E+00
2.480E-02	1.210E-01	1.446E+00	2.8	2.670E-02	1.779E-01	1.688E+00
2.077E-02	1.123E-01	9.784E-01	4.0	2.239E-02	1.641E-01	1.177E+00
1.804E-02	1.132E-01	7.879E-01	5.0	1.946E-02	1.642E-01	9.626E-01
1.370E-02	1.178E-01	6.135E-01	7.0	1.480E-02	1.683E-01	7.595E-01
9.134E-03	1.243E-01	5.118E-01	10.0	9.896E-03	1.744E-01	6.349E-01
4.701E-03	1.328E-01	4.312E-01	15.0	5.121E-03	1.820E-01	5.340E-01
2.443E-03	1.390E-01	3.768E-01	20.0	2.680E-03	1.873E-01	4.660E-01
1.280E-03	1.433E-01	3.321E-01	25.0	1.417E-03	1.906E-01	4.105E-01
6.770E-04	1.463E-01	2.938E-01	30.0	7.571E-04	1.925E-01	3.629E-01
1.971E-04	1.491E-01	2.308E-01	40.0	2.266E-04	1.933E-01	2.849E-01
6.381E-05	1.493E-01	1.819E-01	50.0	7.571E-05	1.914E-01	2.245E-01
2.566E-05	1.480E-01	1.437E-01	60.0	3.099E-05	1.882E-01	1.772E-01
1.409E-05	1.459E-01	1.136E-01	70.0	1.681E-05	1.843E-01	1.401E-01
1.014E-05	1.432E-01	8.983E-02	80.0	1.176E-05	1.800E-01	1.108E-01
8.488E-06	1.404E-01	7.108E-02	90.0	9.608E-06	1.756E-01	8.767E-02
7.573E-06	1.375E-01	5.626E-02	100.0	8.447E-06	1.713E-01	6.939E-02
6.926E-06	1.345E-01	4.453E-02	110.0	7.666E-06	1.671E-01	5.493E-02

Table D.2 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 12 kW/kgU, Set 2

Burnup = 30 MWd/kgU			Cooling Time, years	Burnup = 35 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
5.162E-02	6.440E-01	4.425E+00	1.0	5.315E-02	8.103E-01	4.746E+00
3.790E-02	4.462E-01	3.471E+00	1.4	3.918E-02	5.694E-01	3.761E+00
3.187E-02	3.076E-01	2.594E+00	2.0	3.299E-02	4.003E-01	2.848E+00
2.790E-02	2.448E-01	1.901E+00	2.8	2.889E-02	3.229E-01	2.120E+00
2.343E-02	2.253E-01	1.359E+00	4.0	2.428E-02	2.976E-01	1.543E+00
2.037E-02	2.241E-01	1.127E+00	5.0	2.112E-02	2.950E-01	1.290E+00
1.551E-02	2.270E-01	8.994E-01	7.0	1.610E-02	2.962E-01	1.037E+00
1.039E-02	2.316E-01	7.538E-01	10.0	1.080E-02	2.985E-01	8.704E-01
5.400E-03	2.369E-01	6.334E-01	15.0	5.625E-03	3.005E-01	7.306E-01
2.841E-03	2.399E-01	5.523E-01	20.0	2.969E-03	3.004E-01	6.367E-01
1.511E-03	2.412E-01	4.863E-01	25.0	1.586E-03	2.989E-01	5.603E-01
8.140E-04	2.412E-01	4.297E-01	30.0	8.587E-04	2.964E-01	4.949E-01
2.484E-04	2.385E-01	3.372E-01	40.0	2.654E-04	2.893E-01	3.882E-01
8.490E-05	2.338E-01	2.656E-01	50.0	9.202E-05	2.808E-01	3.057E-01
3.528E-05	2.280E-01	2.096E-01	60.0	3.864E-05	2.719E-01	2.412E-01
1.908E-05	2.218E-01	1.657E-01	70.0	2.091E-05	2.630E-01	1.906E-01
1.316E-05	2.156E-01	1.310E-01	80.0	1.432E-05	2.543E-01	1.508E-01
1.061E-05	2.094E-01	1.037E-01	90.0	1.145E-05	2.460E-01	1.193E-01
9.240E-06	2.035E-01	8.206E-02	100.0	9.930E-06	2.382E-01	9.441E-02
8.344E-06	1.978E-01	6.495E-02	110.0	8.941E-06	2.309E-01	7.473E-02

Table D.3 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 12 kW/kgU, Set 3

Burnup = 40 MWd/kgU			Cooling Time, years	Burnup = 45 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
5.467E-02	1.008E+00	5.001E+00	1.0	5.559E-02	1.186E+00	5.289E+00
4.027E-02	7.180E-01	4.002E+00	1.4	4.089E-02	8.574E-01	4.265E+00
3.390E-02	5.141E-01	3.068E+00	2.0	3.439E-02	6.255E-01	3.300E+00
2.970E-02	4.198E-01	2.315E+00	2.8	3.013E-02	5.175E-01	2.517E+00
2.497E-02	3.875E-01	1.709E+00	4.0	2.533E-02	4.791E-01	1.880E+00
2.173E-02	3.827E-01	1.440E+00	5.0	2.205E-02	4.723E-01	1.593E+00
1.657E-02	3.810E-01	1.164E+00	7.0	1.681E-02	4.678E-01	1.293E+00
1.112E-02	3.797E-01	9.777E-01	10.0	1.129E-02	4.626E-01	1.087E+00
5.800E-03	3.760E-01	8.199E-01	15.0	5.892E-03	4.533E-01	9.110E-01
3.066E-03	3.711E-01	7.139E-01	20.0	3.118E-03	4.434E-01	7.930E-01
1.641E-03	3.653E-01	6.280E-01	25.0	1.671E-03	4.332E-01	6.973E-01
8.905E-04	3.590E-01	5.545E-01	30.0	9.086E-04	4.229E-01	6.155E-01
2.771E-04	3.456E-01	4.347E-01	40.0	2.841E-04	4.029E-01	4.824E-01
9.697E-05	3.320E-01	3.422E-01	50.0	1.001E-04	3.840E-01	3.797E-01
4.118E-05	3.188E-01	2.701E-01	60.0	4.286E-05	3.666E-01	2.996E-01
2.250E-05	3.064E-01	2.134E-01	70.0	2.357E-05	3.505E-01	2.367E-01
1.550E-05	2.948E-01	1.688E-01	80.0	1.629E-05	3.358E-01	1.872E-01
1.242E-05	2.840E-01	1.335E-01	90.0	1.307E-05	3.223E-01	1.481E-01
1.078E-05	2.739E-01	1.057E-01	100.0	1.135E-05	3.099E-01	1.172E-01
9.704E-06	2.646E-01	8.365E-02	110.0	1.022E-05	2.985E-01	9.279E-02

Table D.4 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 20 kW/kgU, Set 1

Burnup = 20 MWd/kgU			Cooling Time, years	Burnup = 25 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
6.181E-02	2.957E-01	5.190E+00	1.0	6.573E-02	4.366E-01	5.764E+00
4.084E-02	2.036E-01	3.853E+00	1.4	4.447E-02	3.012E-01	4.341E+00
3.278E-02	1.398E-01	2.680E+00	2.0	3.605E-02	2.068E-01	3.073E+00
2.832E-02	1.119E-01	1.789E+00	2.8	3.123E-02	1.648E-01	2.100E+00
2.362E-02	1.051E-01	1.133E+00	4.0	2.608E-02	1.531E-01	1.370E+00
2.047E-02	1.064E-01	8.744E-01	5.0	2.262E-02	1.536E-01	1.075E+00
1.551E-02	1.113E-01	6.493E-01	7.0	1.716E-02	1.582E-01	8.101E-01
1.032E-02	1.181E-01	5.298E-01	10.0	1.145E-02	1.646E-01	6.620E-01
5.299E-03	1.271E-01	4.431E-01	15.0	5.905E-03	1.730E-01	5.521E-01
2.748E-03	1.337E-01	3.866E-01	20.0	3.081E-03	1.788E-01	4.810E-01
1.436E-03	1.384E-01	3.406E-01	25.0	1.623E-03	1.826E-01	4.235E-01
7.570E-04	1.416E-01	3.012E-01	30.0	8.640E-04	1.849E-01	3.743E-01
2.182E-04	1.450E-01	2.366E-01	40.0	2.554E-04	1.865E-01	2.939E-01
6.936E-05	1.456E-01	1.865E-01	50.0	8.361E-05	1.853E-01	2.315E-01
2.712E-05	1.447E-01	1.473E-01	60.0	3.322E-05	1.826E-01	1.828E-01
1.449E-05	1.428E-01	1.164E-01	70.0	1.748E-05	1.791E-01	1.445E-01
1.027E-05	1.404E-01	9.208E-02	80.0	1.200E-05	1.753E-01	1.143E-01
8.551E-06	1.378E-01	7.286E-02	90.0	9.719E-06	1.713E-01	9.040E-02
7.618E-06	1.351E-01	5.767E-02	100.0	8.519E-06	1.673E-01	7.155E-02
6.966E-06	1.324E-01	4.565E-02	110.0	7.722E-06	1.635E-01	5.664E-02

Table D.5 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 20 kW/kgU, Set 2

Burnup = 30 MWd/kgU			Cooling Time, years	Burnup = 35 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
6.880E-02	5.899E-01	6.182E+00	1.0	7.175E-02	7.543E-01	6.629E+00
4.724E-02	4.101E-01	4.716E+00	1.4	4.971E-02	5.303E-01	5.110E+00
3.856E-02	2.844E-01	3.395E+00	2.0	4.074E-02	3.731E-01	3.728E+00
3.348E-02	2.276E-01	2.370E+00	2.8	3.541E-02	3.014E-01	2.645E+00
2.799E-02	2.103E-01	1.589E+00	4.0	2.964E-02	2.783E-01	1.809E+00
2.430E-02	2.097E-01	1.267E+00	5.0	2.574E-02	2.762E-01	1.458E+00
1.846E-02	2.130E-01	9.671E-01	7.0	1.956E-02	2.779E-01	1.123E+00
1.233E-02	2.181E-01	7.921E-01	10.0	1.309E-02	2.809E-01	9.207E-01
6.386E-03	2.243E-01	6.595E-01	15.0	6.794E-03	2.838E-01	7.654E-01
3.348E-03	2.280E-01	5.740E-01	20.0	3.574E-03	2.846E-01	6.656E-01
1.775E-03	2.299E-01	5.051E-01	25.0	1.902E-03	2.839E-01	5.855E-01
9.518E-04	2.305E-01	4.462E-01	30.0	1.025E-03	2.821E-01	5.170E-01
2.868E-04	2.289E-01	3.501E-01	40.0	3.127E-04	2.763E-01	4.055E-01
9.588E-05	2.250E-01	2.757E-01	50.0	1.060E-04	2.689E-01	3.193E-01
3.858E-05	2.200E-01	2.176E-01	60.0	4.300E-05	2.609E-01	2.520E-01
2.016E-05	2.144E-01	1.720E-01	70.0	2.240E-05	2.529E-01	1.991E-01
1.357E-05	2.087E-01	1.360E-01	80.0	1.490E-05	2.450E-01	1.575E-01
1.079E-05	2.031E-01	1.076E-01	90.0	1.172E-05	2.375E-01	1.246E-01
9.353E-06	1.977E-01	8.518E-02	100.0	1.009E-05	2.303E-01	9.859E-02
8.426E-06	1.925E-01	6.742E-02	110.0	9.049E-06	2.236E-01	7.804E-02

Table D.6 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 20 kW/kgU, Set 3

Burnup = 40 MWd/kgU			Cooling Time, years	Burnup = 45 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
7.481E-02	9.488E-01	6.976E+00	1.0	7.701E-02	1.129E+00	7.365E+00
5.211E-02	6.752E-01	5.432E+00	1.4	5.376E-02	8.133E-01	5.780E+00
4.281E-02	4.826E-01	4.015E+00	2.0	4.420E-02	5.908E-01	4.315E+00
3.726E-02	3.937E-01	2.893E+00	2.8	3.848E-02	4.873E-01	3.147E+00
3.120E-02	3.635E-01	2.015E+00	4.0	3.224E-02	4.507E-01	2.222E+00
2.711E-02	3.593E-01	1.638E+00	5.0	2.801E-02	4.444E-01	1.820E+00
2.062E-02	3.582E-01	1.271E+00	7.0	2.131E-02	4.405E-01	1.420E+00
1.380E-02	3.575E-01	1.042E+00	10.0	1.427E-02	4.361E-01	1.166E+00
7.174E-03	3.550E-01	8.652E-01	15.0	7.422E-03	4.280E-01	9.673E-01
3.780E-03	3.511E-01	7.517E-01	20.0	3.914E-03	4.191E-01	8.400E-01
2.016E-03	3.462E-01	6.608E-01	25.0	2.090E-03	4.099E-01	7.382E-01
1.089E-03	3.407E-01	5.834E-01	30.0	1.131E-03	4.006E-01	6.515E-01
3.342E-04	3.288E-01	4.573E-01	40.0	3.486E-04	3.823E-01	5.105E-01
1.142E-04	3.165E-01	3.600E-01	50.0	1.197E-04	3.650E-01	4.018E-01
4.668E-05	3.046E-01	2.840E-01	60.0	4.921E-05	3.490E-01	3.170E-01
2.442E-05	2.932E-01	2.244E-01	70.0	2.583E-05	3.342E-01	2.505E-01
1.626E-05	2.826E-01	1.775E-01	80.0	1.720E-05	3.207E-01	1.981E-01
1.278E-05	2.727E-01	1.404E-01	90.0	1.350E-05	3.084E-01	1.567E-01
1.098E-05	2.635E-01	1.111E-01	100.0	1.159E-05	2.970E-01	1.240E-01
9.843E-06	2.550E-01	8.796E-02	110.0	1.038E-05	2.866E-01	9.817E-02

Table D.7 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 30 kW/kgU, Set 1

Burnup = 20 MWd/kgU			Cooling Time, years	Burnup = 25 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
7.591E-02	2.580E-01	6.475E+00	1.0	8.104E-02	3.892E-01	7.316E+00
4.621E-02	1.816E-01	4.711E+00	1.4	5.080E-02	2.731E-01	5.397E+00
3.561E-02	1.289E-01	3.203E+00	2.0	3.970E-02	1.924E-01	3.726E+00
3.046E-02	1.062E-01	2.074E+00	2.8	3.405E-02	1.567E-01	2.460E+00
2.529E-02	1.012E-01	1.255E+00	4.0	2.830E-02	1.472E-01	1.529E+00
2.188E-02	1.028E-01	9.384E-01	5.0	2.450E-02	1.482E-01	1.162E+00
1.655E-02	1.079E-01	6.725E-01	7.0	1.855E-02	1.530E-01	8.438E-01
1.100E-02	1.148E-01	5.398E-01	10.0	1.235E-02	1.597E-01	6.776E-01
5.638E-03	1.241E-01	4.492E-01	15.0	6.356E-03	1.684E-01	5.619E-01
2.918E-03	1.308E-01	3.916E-01	20.0	3.309E-03	1.745E-01	4.890E-01
1.522E-03	1.357E-01	3.450E-01	25.0	1.738E-03	1.786E-01	4.304E-01
8.003E-04	1.391E-01	3.050E-01	30.0	9.222E-04	1.812E-01	3.804E-01
2.290E-04	1.427E-01	2.396E-01	40.0	2.701E-04	1.831E-01	2.986E-01
7.194E-05	1.436E-01	1.888E-01	50.0	8.718E-05	1.823E-01	2.352E-01
2.769E-05	1.428E-01	1.491E-01	60.0	3.402E-05	1.799E-01	1.857E-01
1.459E-05	1.411E-01	1.179E-01	70.0	1.764E-05	1.767E-01	1.468E-01
1.029E-05	1.389E-01	9.322E-02	80.0	1.203E-05	1.730E-01	1.161E-01
8.564E-06	1.364E-01	7.376E-02	90.0	9.736E-06	1.693E-01	9.185E-02
7.635E-06	1.338E-01	5.838E-02	100.0	8.539E-06	1.655E-01	7.269E-02
6.986E-06	1.312E-01	4.621E-02	110.0	7.745E-06	1.618E-01	5.754E-02

Table D.8 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 30 kW/kgU, Set 2

Burnup = 30 MWd/kgU			Cooling Time, years	Burnup = 35 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
8.512E-02	5.349E-01	7.931E+00	1.0	8.934E-02	6.945E-01	8.553E+00
5.454E-02	3.770E-01	5.925E+00	1.4	5.801E-02	4.935E-01	6.454E+00
4.307E-02	2.666E-01	4.153E+00	2.0	4.609E-02	3.525E-01	4.580E+00
3.704E-02	2.170E-01	2.796E+00	2.8	3.971E-02	2.883E-01	3.132E+00
3.084E-02	2.023E-01	1.783E+00	4.0	3.309E-02	2.680E-01	2.038E+00
2.672E-02	2.021E-01	1.376E+00	5.0	2.868E-02	2.664E-01	1.590E+00
2.025E-02	2.058E-01	1.013E+00	7.0	2.176E-02	2.685E-01	1.181E+00
1.351E-02	2.112E-01	8.144E-01	10.0	1.453E-02	2.718E-01	9.504E-01
6.976E-03	2.178E-01	6.736E-01	15.0	7.523E-03	2.753E-01	7.844E-01
3.648E-03	2.219E-01	5.855E-01	20.0	3.948E-03	2.766E-01	6.811E-01
1.928E-03	2.242E-01	5.151E-01	25.0	2.096E-03	2.763E-01	5.988E-01
1.031E-03	2.251E-01	4.550E-01	30.0	1.126E-03	2.749E-01	5.288E-01
3.076E-04	2.240E-01	3.569E-01	40.0	3.404E-04	2.697E-01	4.146E-01
1.014E-04	2.206E-01	2.811E-01	50.0	1.138E-04	2.629E-01	3.264E-01
4.003E-05	2.159E-01	2.219E-01	60.0	4.528E-05	2.555E-01	2.576E-01
2.056E-05	2.108E-01	1.753E-01	70.0	2.311E-05	2.479E-01	2.036E-01
1.369E-05	2.054E-01	1.387E-01	80.0	1.516E-05	2.404E-01	1.610E-01
1.085E-05	2.001E-01	1.097E-01	90.0	1.184E-05	2.332E-01	1.274E-01
9.393E-06	1.949E-01	8.682E-02	100.0	1.015E-05	2.264E-01	1.008E-01
8.461E-06	1.900E-01	6.873E-02	110.0	9.099E-06	2.200E-01	7.979E-02

Table D.9 BWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 30 kW/kgU, Set 3

Burnup = 40 MWd/kgU			Cooling Time,	Burnup = 45 MWd/kgU		
Light El	Actinides	Fis Prod	years	Light El	Actinides	Fis Prod
9.371E-02	8.840E-01	9.029E+00	1.0	9.716E-02	1.063E+00	9.546E+00
6.150E-02	6.345E-01	6.883E+00	1.4	6.410E-02	7.709E-01	7.334E+00
4.912E-02	4.590E-01	4.945E+00	2.0	5.132E-02	5.651E-01	5.322E+00
4.241E-02	3.780E-01	3.435E+00	2.8	4.435E-02	4.694E-01	3.742E+00
3.538E-02	3.507E-01	2.277E+00	4.0	3.701E-02	4.357E-01	2.518E+00
3.068E-02	3.470E-01	1.794E+00	5.0	3.211E-02	4.299E-01	2.000E+00
2.329E-02	3.463E-01	1.343E+00	7.0	2.438E-02	4.264E-01	1.507E+00
1.556E-02	3.460E-01	1.081E+00	10.0	1.629E-02	4.224E-01	1.214E+00
8.070E-03	3.441E-01	8.899E-01	15.0	8.457E-03	4.149E-01	9.982E-01
4.243E-03	3.407E-01	7.719E-01	20.0	4.451E-03	4.067E-01	8.653E-01
2.258E-03	3.363E-01	6.783E-01	25.0	2.372E-03	3.980E-01	7.600E-01
1.216E-03	3.313E-01	5.987E-01	30.0	1.281E-03	3.892E-01	6.707E-01
3.703E-04	3.202E-01	4.692E-01	40.0	3.919E-04	3.718E-01	5.255E-01
1.248E-04	3.086E-01	3.693E-01	50.0	1.328E-04	3.553E-01	4.136E-01
5.001E-05	2.973E-01	2.914E-01	60.0	5.343E-05	3.400E-01	3.263E-01
2.555E-05	2.866E-01	2.303E-01	70.0	2.730E-05	3.260E-01	2.578E-01
1.669E-05	2.765E-01	1.821E-01	80.0	1.777E-05	3.131E-01	2.039E-01
1.297E-05	2.671E-01	1.441E-01	90.0	1.376E-05	3.014E-01	1.613E-01
1.108E-05	2.583E-01	1.140E-01	100.0	1.173E-05	2.906E-01	1.276E-01
9.913E-06	2.501E-01	9.024E-02	110.0	1.047E-05	2.806E-01	1.010E-01

Table D.10 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 18 kW/kgU, Set 1

Burnup = 25 MWd/kgU			Cooling Time,	Burnup = 30 MWd/kgU		
Light El	Actinides	Fis Prod	years	Light El	Actinides	Fis Prod
1.198E-01	4.377E-01	5.389E+00	1.0	1.269E-01	5.918E-01	5.855E+00
1.062E-01	3.002E-01	4.079E+00	1.4	1.130E-01	4.091E-01	4.487E+00
9.579E-02	2.045E-01	2.908E+00	2.0	1.021E-01	2.814E-01	3.249E+00
8.550E-02	1.620E-01	2.006E+00	2.8	9.113E-02	2.241E-01	2.286E+00
7.262E-02	1.505E-01	1.328E+00	4.0	7.742E-02	2.072E-01	1.550E+00
6.351E-02	1.515E-01	1.053E+00	5.0	6.771E-02	2.071E-01	1.245E+00
4.869E-02	1.567E-01	8.031E-01	7.0	5.191E-02	2.116E-01	9.597E-01
3.275E-02	1.642E-01	6.609E-01	10.0	3.492E-02	2.182E-01	7.905E-01
1.696E-02	1.739E-01	5.530E-01	15.0	1.808E-02	2.264E-01	6.600E-01
8.806E-03	1.807E-01	4.822E-01	20.0	9.389E-03	2.318E-01	5.749E-01
4.589E-03	1.855E-01	4.247E-01	25.0	4.894E-03	2.351E-01	5.060E-01
2.406E-03	1.885E-01	3.755E-01	30.0	2.566E-03	2.367E-01	4.471E-01
6.872E-04	1.910E-01	2.948E-01	40.0	7.342E-04	2.366E-01	3.509E-01
2.235E-04	1.905E-01	2.323E-01	50.0	2.399E-04	2.336E-01	2.764E-01
9.681E-05	1.882E-01	1.834E-01	60.0	1.047E-04	2.290E-01	2.182E-01
6.071E-05	1.849E-01	1.450E-01	70.0	6.620E-05	2.238E-01	1.724E-01
4.912E-05	1.812E-01	1.147E-01	80.0	5.379E-05	2.183E-01	1.364E-01
4.425E-05	1.772E-01	9.072E-02	90.0	4.856E-05	2.127E-01	1.079E-01
4.133E-05	1.733E-01	7.180E-02	100.0	4.541E-05	2.072E-01	8.539E-02
3.904E-05	1.693E-01	5.684E-02	110.0	4.293E-05	2.019E-01	6.759E-02

Table D.11 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 18 kW/kgU, Set 2

Burnup = 35 MWd/kgU			Cooling Time,	Burnup = 40 MWd/kgU		
Light El	Actinides	Fis Prod	years	Light El	Actinides	Fis Prod
1.319E-01	7.602E-01	6.194E+00	1.0	1.365E-01	9.352E-01	6.590E+00
1.177E-01	5.305E-01	4.800E+00	1.4	1.221E-01	6.608E-01	5.155E+00
1.064E-01	3.696E-01	3.528E+00	2.0	1.105E-01	4.680E-01	3.833E+00
9.506E-02	2.965E-01	2.529E+00	2.8	9.870E-02	3.797E-01	2.785E+00
8.077E-02	2.736E-01	1.754E+00	4.0	8.386E-02	3.507E-01	1.963E+00
7.064E-02	2.721E-01	1.426E+00	5.0	7.335E-02	3.476E-01	1.609E+00
5.416E-02	2.751E-01	1.110E+00	7.0	5.623E-02	3.486E-01	1.261E+00
3.643E-02	2.798E-01	9.162E-01	10.0	3.783E-02	3.508E-01	1.041E+00
1.887E-02	2.852E-01	7.640E-01	15.0	1.959E-02	3.522E-01	8.670E-01
9.797E-03	2.879E-01	6.649E-01	20.0	1.017E-02	3.515E-01	7.540E-01
5.108E-03	2.888E-01	5.850E-01	25.0	5.305E-03	3.492E-01	6.630E-01
2.680E-03	2.882E-01	5.167E-01	30.0	2.784E-03	3.457E-01	5.854E-01
7.683E-04	2.841E-01	4.052E-01	40.0	7.997E-04	3.367E-01	4.590E-01
2.525E-04	2.778E-01	3.191E-01	50.0	2.641E-04	3.264E-01	3.614E-01
1.114E-04	2.705E-01	2.518E-01	60.0	1.175E-04	3.156E-01	2.852E-01
7.106E-05	2.628E-01	1.990E-01	70.0	7.559E-05	3.049E-01	2.253E-01
5.804E-05	2.551E-01	1.574E-01	80.0	6.201E-05	2.947E-01	1.782E-01
5.251E-05	2.476E-01	1.245E-01	90.0	5.621E-05	2.849E-01	1.410E-01
4.916E-05	2.404E-01	9.855E-02	100.0	5.267E-05	2.758E-01	1.116E-01
4.652E-05	2.335E-01	7.801E-02	110.0	4.989E-05	2.672E-01	8.831E-02

Table D.12 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 18 kW/kgU, Set 3

Burnup = 45 MWd/kgU			Cooling Time,	Burnup = 50 MWd/kgU		
Light El	Actinides	Fis Prod	years	Light El	Actinides	Fis Prod
1.410E-01	1.144E+00	6.891E+00	1.0	1.458E-01	1.354E+00	7.273E+00
1.264E-01	8.180E-01	5.438E+00	1.4	1.308E-01	9.824E-01	5.781E+00
1.144E-01	5.883E-01	4.090E+00	2.0	1.185E-01	7.198E-01	4.385E+00
1.022E-01	4.820E-01	3.011E+00	2.8	1.059E-01	5.973E-01	3.259E+00
8.685E-02	4.454E-01	2.153E+00	4.0	8.996E-02	5.533E-01	2.354E+00
7.597E-02	4.399E-01	1.778E+00	5.0	7.869E-02	5.452E-01	1.952E+00
5.824E-02	4.378E-01	1.401E+00	7.0	6.033E-02	5.393E-01	1.543E+00
3.918E-02	4.358E-01	1.158E+00	10.0	4.059E-02	5.322E-01	1.274E+00
2.029E-02	4.312E-01	9.628E-01	15.0	2.102E-02	5.200E-01	1.058E+00
1.054E-02	4.251E-01	8.367E-01	20.0	1.092E-02	5.074E-01	9.187E-01
5.498E-03	4.181E-01	7.354E-01	25.0	5.698E-03	4.947E-01	8.072E-01
2.887E-03	4.106E-01	6.491E-01	30.0	2.993E-03	4.821E-01	7.122E-01
8.315E-04	3.947E-01	5.087E-01	40.0	8.641E-04	4.580E-01	5.579E-01
2.766E-04	3.789E-01	4.004E-01	50.0	2.891E-04	4.356E-01	4.391E-01
1.246E-04	3.636E-01	3.159E-01	60.0	1.315E-04	4.152E-01	3.464E-01
8.104E-05	3.493E-01	2.496E-01	70.0	8.632E-05	3.966E-01	2.737E-01
6.684E-05	3.360E-01	1.974E-01	80.0	7.150E-05	3.797E-01	2.164E-01
6.072E-05	3.236E-01	1.562E-01	90.0	6.507E-05	3.643E-01	1.712E-01
5.696E-05	3.121E-01	1.236E-01	100.0	6.108E-05	3.502E-01	1.355E-01
5.398E-05	3.014E-01	9.783E-02	110.0	5.792E-05	3.373E-01	1.073E-01

Table D.13. PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 28 kW/kgU, Set 1

Burnup = 25 MWd/kgU			Cooling Time,	Burnup = 30 MWd/kgU		
Light El	Actinides	Fis Prod	years	Light El	Actinides	Fis Prod
1.361E-01	3.874E-01	7.036E+00	1.0	1.461E-01	5.340E-01	7.710E+00
1.181E-01	2.700E-01	5.205E+00	1.4	1.275E-01	3.734E-01	5.772E+00
1.058E-01	1.885E-01	3.606E+00	2.0	1.144E-01	2.615E-01	4.056E+00
9.420E-02	1.527E-01	2.394E+00	2.8	1.020E-01	2.115E-01	2.741E+00
7.992E-02	1.436E-01	1.500E+00	4.0	8.653E-02	1.973E-01	1.759E+00
6.986E-02	1.450E-01	1.148E+00	5.0	7.564E-02	1.977E-01	1.364E+00
5.353E-02	1.506E-01	8.410E-01	7.0	5.796E-02	2.026E-01	1.010E+00
3.600E-02	1.583E-01	6.791E-01	10.0	3.898E-02	2.096E-01	8.159E-01
1.864E-02	1.685E-01	5.645E-01	15.0	2.018E-02	2.184E-01	6.764E-01
9.672E-03	1.757E-01	4.916E-01	20.0	1.047E-02	2.243E-01	5.883E-01
5.038E-03	1.808E-01	4.329E-01	25.0	5.454E-03	2.280E-01	5.176E-01
2.638E-03	1.841E-01	3.826E-01	30.0	2.857E-03	2.300E-01	4.573E-01
7.497E-04	1.871E-01	3.004E-01	40.0	8.124E-04	2.306E-01	3.588E-01
2.405E-04	1.870E-01	2.367E-01	50.0	2.611E-04	2.281E-01	2.826E-01
1.015E-04	1.850E-01	1.869E-01	60.0	1.106E-04	2.241E-01	2.231E-01
6.210E-05	1.820E-01	1.477E-01	70.0	6.793E-05	2.193E-01	1.763E-01
4.961E-05	1.786E-01	1.168E-01	80.0	5.440E-05	2.141E-01	1.394E-01
4.450E-05	1.749E-01	9.243E-02	90.0	4.886E-05	2.089E-01	1.103E-01
4.150E-05	1.711E-01	7.315E-02	100.0	4.561E-05	2.038E-01	8.730E-02
3.919E-05	1.674E-01	5.791E-02	110.0	4.311E-05	1.988E-01	6.911E-02

Table D.14. PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 28 kW/kgU, Set 2

Burnup = 35 MWd/kgU			Cooling Time,	Burnup = 40 MWd/kgU		
Light El	Actinides	Fis Prod	years	Light El	Actinides	Fis Prod
1.539E-01	6.960E-01	8.205E+00	1.0	1.613E-01	8.674E-01	8.747E+00
1.351E-01	4.903E-01	6.211E+00	1.4	1.420E-01	6.170E-01	6.682E+00
1.214E-01	3.464E-01	4.426E+00	2.0	1.278E-01	4.412E-01	4.816E+00
1.082E-01	2.812E-01	3.046E+00	2.8	1.139E-01	3.609E-01	3.360E+00
9.182E-02	2.612E-01	1.999E+00	4.0	9.670E-02	3.348E-01	2.244E+00
8.028E-02	2.603E-01	1.570E+00	5.0	8.455E-02	3.322E-01	1.778E+00
6.151E-02	2.637E-01	1.175E+00	7.0	6.479E-02	3.338E-01	1.341E+00
4.137E-02	2.689E-01	9.505E-01	10.0	4.357E-02	3.365E-01	1.085E+00
2.141E-02	2.749E-01	7.863E-01	15.0	2.255E-02	3.387E-01	8.958E-01
1.111E-02	2.783E-01	6.832E-01	20.0	1.171E-02	3.387E-01	7.776E-01
5.789E-03	2.797E-01	6.008E-01	25.0	6.098E-03	3.371E-01	6.835E-01
3.033E-03	2.796E-01	5.306E-01	30.0	3.195E-03	3.342E-01	6.034E-01
8.632E-04	2.763E-01	4.161E-01	40.0	9.102E-04	3.263E-01	4.730E-01
2.782E-04	2.707E-01	3.276E-01	50.0	2.940E-04	3.168E-01	3.723E-01
1.185E-04	2.640E-01	2.586E-01	60.0	1.258E-04	3.068E-01	2.938E-01
7.316E-05	2.568E-01	2.043E-01	70.0	7.803E-05	2.969E-01	2.322E-01
5.877E-05	2.496E-01	1.616E-01	80.0	6.286E-05	2.873E-01	1.836E-01
5.287E-05	2.426E-01	1.278E-01	90.0	5.663E-05	2.782E-01	1.452E-01
4.940E-05	2.358E-01	1.012E-01	100.0	5.295E-05	2.695E-01	1.149E-01
4.673E-05	2.294E-01	8.008E-02	110.0	5.012E-05	2.614E-01	9.098E-02

Table D.15 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 28 kW/kgU, Set 3

Burnup = 45 MWd/kgU			Cooling Time, years	Burnup = 50 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
1.690E-01	1.073E+00	9.156E+00	1.0	1.767E-01	1.282E+00	9.661E+00
1.493E-01	7.710E-01	7.058E+00	1.4	1.565E-01	9.334E-01	7.503E+00
1.344E-01	5.585E-01	5.145E+00	2.0	1.410E-01	6.870E-01	5.518E+00
1.199E-01	4.603E-01	3.640E+00	2.8	1.258E-01	5.721E-01	3.944E+00
1.018E-01	4.267E-01	2.471E+00	4.0	1.068E-01	5.310E-01	2.708E+00
8.901E-02	4.218E-01	1.975E+00	5.0	9.341E-02	5.234E-01	2.176E+00
6.821E-02	4.202E-01	1.499E+00	7.0	7.158E-02	5.180E-01	1.658E+00
4.587E-02	4.189E-01	1.212E+00	10.0	4.814E-02	5.116E-01	1.340E+00
2.374E-02	4.151E-01	9.993E-01	15.0	2.492E-02	5.003E-01	1.102E+00
1.233E-02	4.098E-01	8.666E-01	20.0	1.294E-02	4.886E-01	9.552E-01
6.422E-03	4.036E-01	7.613E-01	25.0	6.741E-03	4.766E-01	8.387E-01
3.366E-03	3.967E-01	6.719E-01	30.0	3.534E-03	4.648E-01	7.400E-01
9.603E-04	3.820E-01	5.264E-01	40.0	1.009E-03	4.420E-01	5.796E-01
3.114E-04	3.672E-01	4.143E-01	50.0	3.285E-04	4.209E-01	4.560E-01
1.342E-04	3.529E-01	3.269E-01	60.0	1.424E-04	4.016E-01	3.598E-01
8.388E-05	3.394E-01	2.583E-01	70.0	8.954E-05	3.841E-01	2.842E-01
6.784E-05	3.269E-01	2.042E-01	80.0	7.264E-05	3.681E-01	2.247E-01
6.121E-05	3.152E-01	1.616E-01	90.0	6.563E-05	3.536E-01	1.778E-01
5.729E-05	3.044E-01	1.279E-01	100.0	6.146E-05	3.404E-01	1.407E-01
5.425E-05	2.943E-01	1.012E-01	110.0	5.823E-05	3.282E-01	1.114E-01

Table D.16 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 40 kW/kgU, Set 1

Burnup = 25 MWd/kgU			Cooling Time, years	Burnup = 30 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
1.485E-01	3.419E-01	8.456E+00	1.0	1.607E-01	4.800E-01	9.412E+00
1.259E-01	2.435E-01	6.145E+00	1.4	1.374E-01	3.420E-01	6.921E+00
1.119E-01	1.754E-01	4.175E+00	2.0	1.224E-01	2.459E-01	4.761E+00
9.944E-02	1.458E-01	2.702E+00	2.8	1.088E-01	2.033E-01	3.129E+00
8.428E-02	1.389E-01	1.630E+00	4.0	9.225E-02	1.917E-01	1.928E+00
7.365E-02	1.406E-01	1.215E+00	5.0	8.061E-02	1.926E-01	1.455E+00
5.641E-02	1.463E-01	8.646E-01	7.0	6.175E-02	1.977E-01	1.045E+00
3.793E-02	1.543E-01	6.886E-01	10.0	4.152E-02	2.049E-01	8.315E-01
1.963E-02	1.646E-01	5.700E-01	15.0	2.148E-02	2.140E-01	6.859E-01
1.019E-02	1.721E-01	4.960E-01	20.0	1.115E-02	2.203E-01	5.960E-01
5.304E-03	1.773E-01	4.367E-01	25.0	5.805E-03	2.242E-01	5.243E-01
2.776E-03	1.808E-01	3.859E-01	30.0	3.038E-03	2.265E-01	4.631E-01
7.867E-04	1.841E-01	3.030E-01	40.0	8.612E-04	2.274E-01	3.633E-01
2.505E-04	1.842E-01	2.387E-01	50.0	2.743E-04	2.253E-01	2.862E-01
1.042E-04	1.825E-01	1.884E-01	60.0	1.143E-04	2.216E-01	2.259E-01
6.287E-05	1.797E-01	1.489E-01	70.0	6.902E-05	2.170E-01	1.785E-01
4.985E-05	1.764E-01	1.178E-01	80.0	5.479E-05	2.120E-01	1.412E-01
4.459E-05	1.728E-01	9.320E-02	90.0	4.905E-05	2.070E-01	1.117E-01
4.155E-05	1.692E-01	7.376E-02	100.0	4.574E-05	2.020E-01	8.839E-02
3.923E-05	1.656E-01	5.839E-02	110.0	4.322E-05	1.972E-01	6.997E-02

Table D.17 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 40 kW/kgU, Set 2

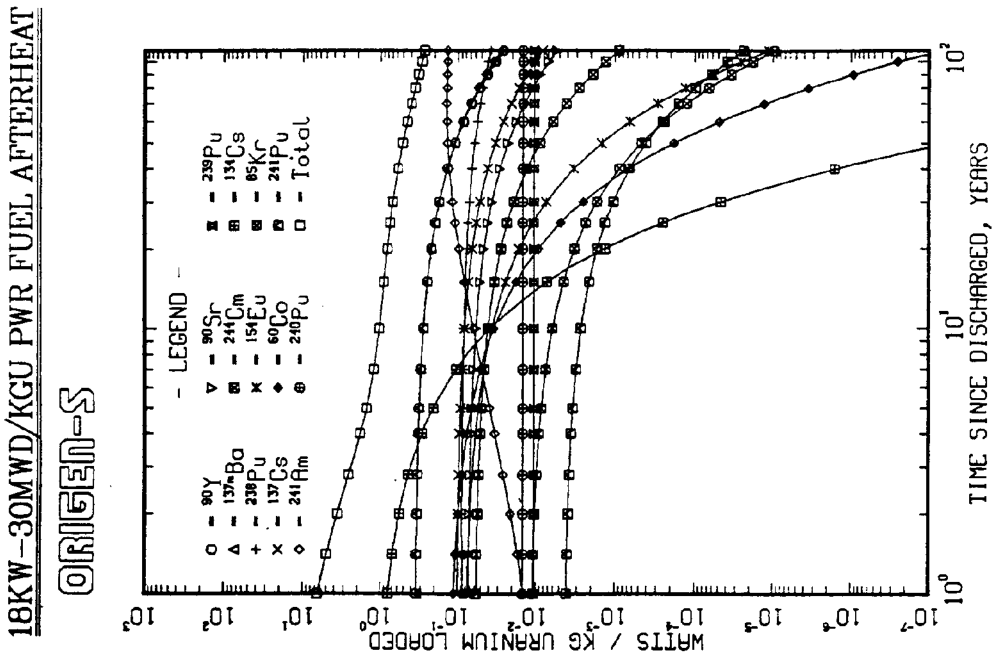
Burnup = 35 MWd/kgU			Cooling Time, years	Burnup = 40 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
1.706E-01	6.341E-01	1.010E+01	1.0	1.800E-01	7.998E-01	1.084E+01
1.469E-01	4.538E-01	7.510E+00	1.4	1.557E-01	5.766E-01	8.131E+00
1.311E-01	3.277E-01	5.233E+00	2.0	1.393E-01	4.201E-01	5.725E+00
1.167E-01	2.709E-01	3.496E+00	2.8	1.239E-01	3.487E-01	3.873E+00
9.891E-02	2.538E-01	2.201E+00	4.0	1.051E-01	3.258E-01	2.479E+00
8.644E-02	2.534E-01	1.681E+00	5.0	9.183E-02	3.238E-01	1.911E+00
6.621E-02	2.571E-01	1.220E+00	7.0	7.034E-02	3.257E-01	1.397E+00
4.452E-02	2.626E-01	9.713E-01	10.0	4.729E-02	3.287E-01	1.112E+00
2.304E-02	2.689E-01	7.990E-01	15.0	2.447E-02	3.314E-01	9.125E-01
1.195E-02	2.726E-01	6.934E-01	20.0	1.270E-02	3.318E-01	7.911E-01
6.224E-03	2.743E-01	6.096E-01	25.0	6.612E-03	3.306E-01	6.951E-01
3.258E-03	2.745E-01	5.382E-01	30.0	3.461E-03	3.281E-01	6.136E-01
9.238E-04	2.717E-01	4.220E-01	40.0	9.818E-04	3.207E-01	4.809E-01
2.946E-04	2.665E-01	3.323E-01	50.0	3.134E-04	3.117E-01	3.786E-01
1.230E-04	2.601E-01	2.622E-01	60.0	1.311E-04	3.022E-01	2.987E-01
7.449E-05	2.533E-01	2.072E-01	70.0	7.960E-05	2.927E-01	2.360E-01
5.924E-05	2.464E-01	1.639E-01	80.0	6.340E-05	2.834E-01	1.866E-01
5.310E-05	2.396E-01	1.296E-01	90.0	5.688E-05	2.746E-01	1.477E-01
4.956E-05	2.331E-01	1.026E-01	100.0	5.313E-05	2.663E-01	1.169E-01
4.686E-05	2.268E-01	8.121E-02	110.0	5.027E-05	2.585E-01	9.249E-02

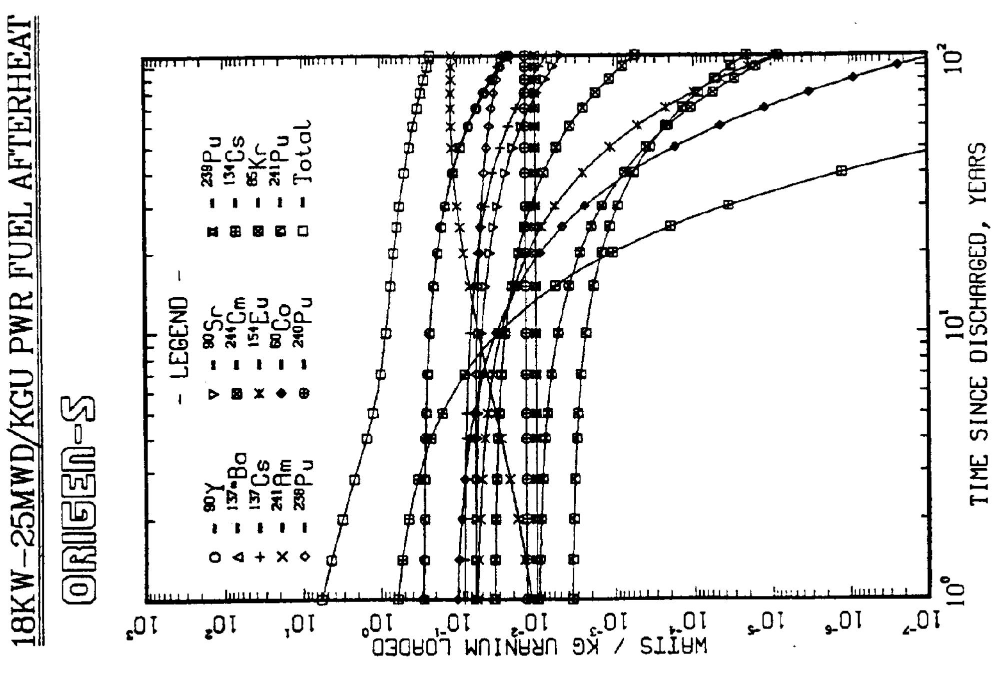
Table D.18 PWR decay heat rates (W/kgU) of light elements, actinides and fission products, for specific power = 40 kW/kgU, Set 3

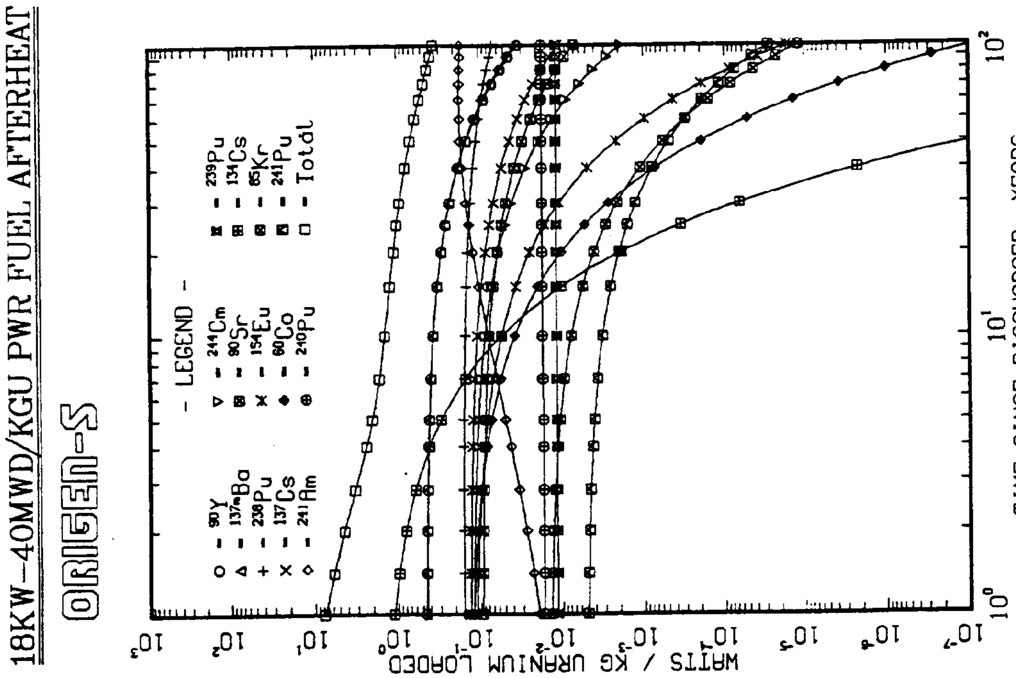
Burnup = 45 MWd/kgU			Cooling Time, years	Burnup = 50 MWd/kgU		
Light El	Actinides	Fis Prod		Light El	Actinides	Fis Prod
1.901E-01	9.994E-01	1.139E+01	1.0	2.001E-01	1.206E+00	1.206E+01
1.652E-01	7.266E-01	8.622E+00	1.4	1.746E-01	8.867E-01	9.193E+00
1.480E-01	5.346E-01	6.138E+00	2.0	1.565E-01	6.612E-01	6.600E+00
1.318E-01	4.460E-01	4.209E+00	2.8	1.394E-01	5.560E-01	4.572E+00
1.118E-01	4.157E-01	2.738E+00	4.0	1.182E-01	5.184E-01	3.010E+00
9.767E-02	4.114E-01	2.130E+00	5.0	1.033E-01	5.114E-01	2.355E+00
7.482E-02	4.102E-01	1.567E+00	7.0	7.917E-02	5.064E-01	1.739E+00
5.030E-02	4.092E-01	1.246E+00	10.0	5.323E-02	5.003E-01	1.382E+00
2.603E-02	4.059E-01	1.020E+00	15.0	2.754E-02	4.896E-01	1.128E+00
1.351E-02	4.010E-01	8.835E-01	20.0	1.429E-02	4.783E-01	9.761E-01
7.034E-03	3.951E-01	7.759E-01	25.0	7.443E-03	4.668E-01	8.568E-01
3.682E-03	3.886E-01	6.846E-01	30.0	3.897E-03	4.554E-01	7.557E-01
1.045E-03	3.746E-01	5.363E-01	40.0	1.107E-03	4.334E-01	5.918E-01
3.345E-04	3.604E-01	4.221E-01	50.0	3.549E-04	4.130E-01	4.657E-01
1.406E-04	3.466E-01	3.330E-01	60.0	1.497E-04	3.943E-01	3.673E-01
8.574E-05	3.336E-01	2.631E-01	70.0	9.166E-05	3.773E-01	2.902E-01
6.848E-05	3.215E-01	2.080E-01	80.0	7.336E-05	3.620E-01	2.295E-01
6.151E-05	3.103E-01	1.646E-01	90.0	6.596E-05	3.480E-01	1.815E-01
5.749E-05	2.998E-01	1.302E-01	100.0	6.168E-05	3.351E-01	1.437E-01
5.442E-05	2.902E-01	1.031E-01	110.0	5.841E-05	3.234E-01	1.137E-01

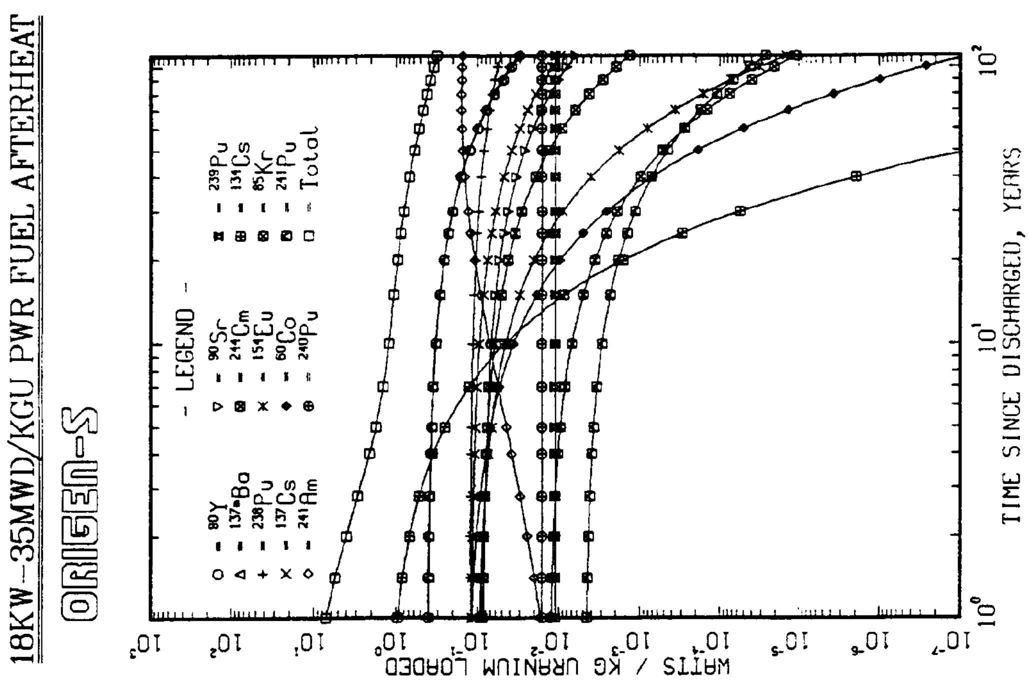
APPENDIX E

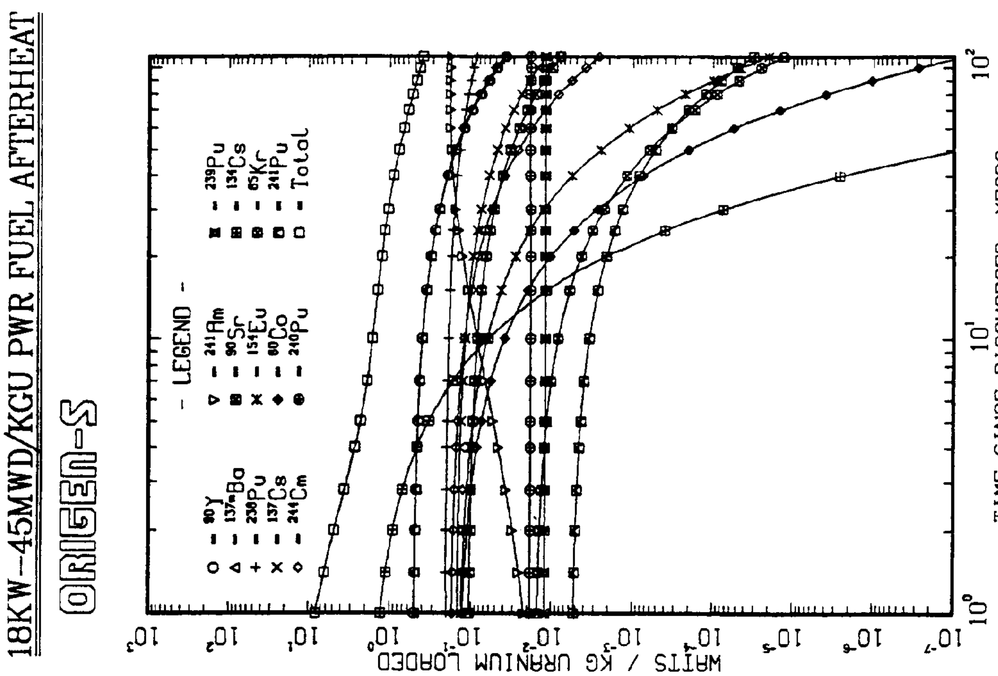
PLOTS OF MAJOR DECAY HEAT RATE NUCLIDES

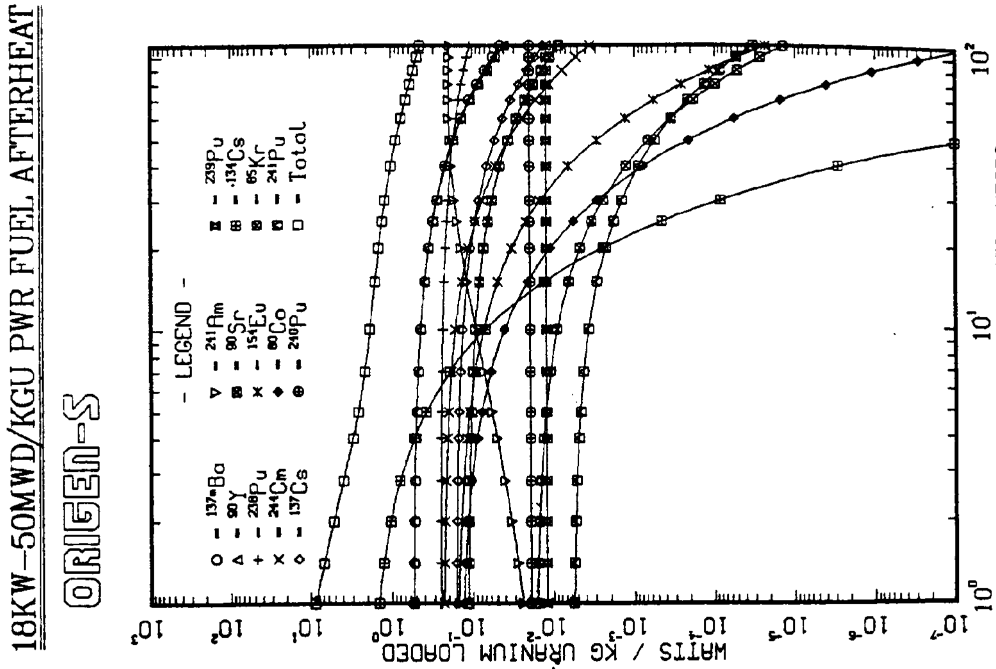


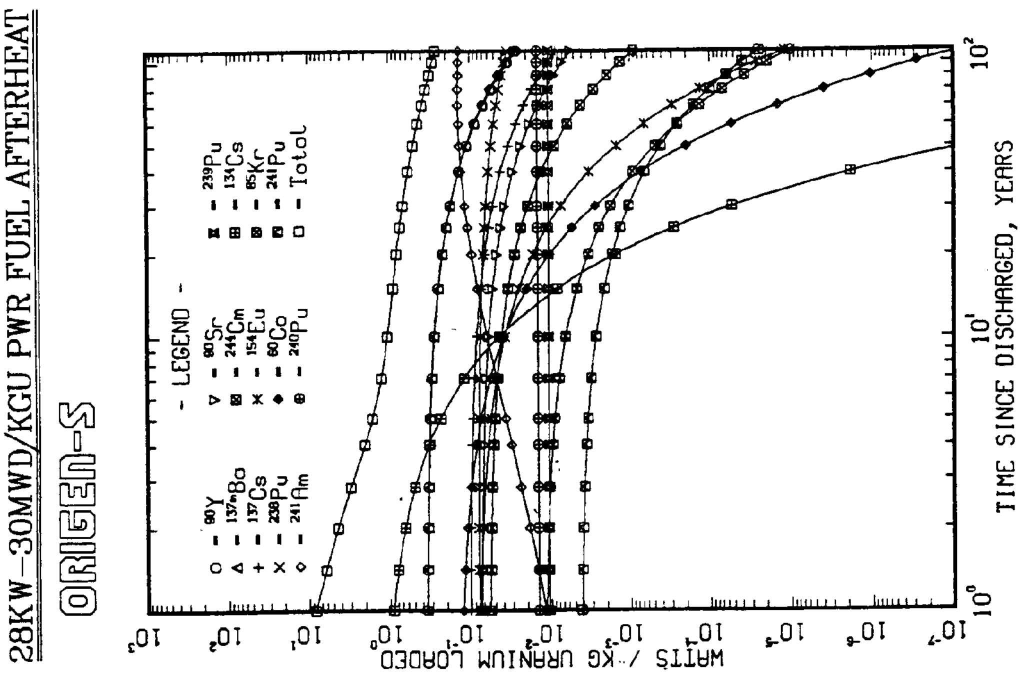


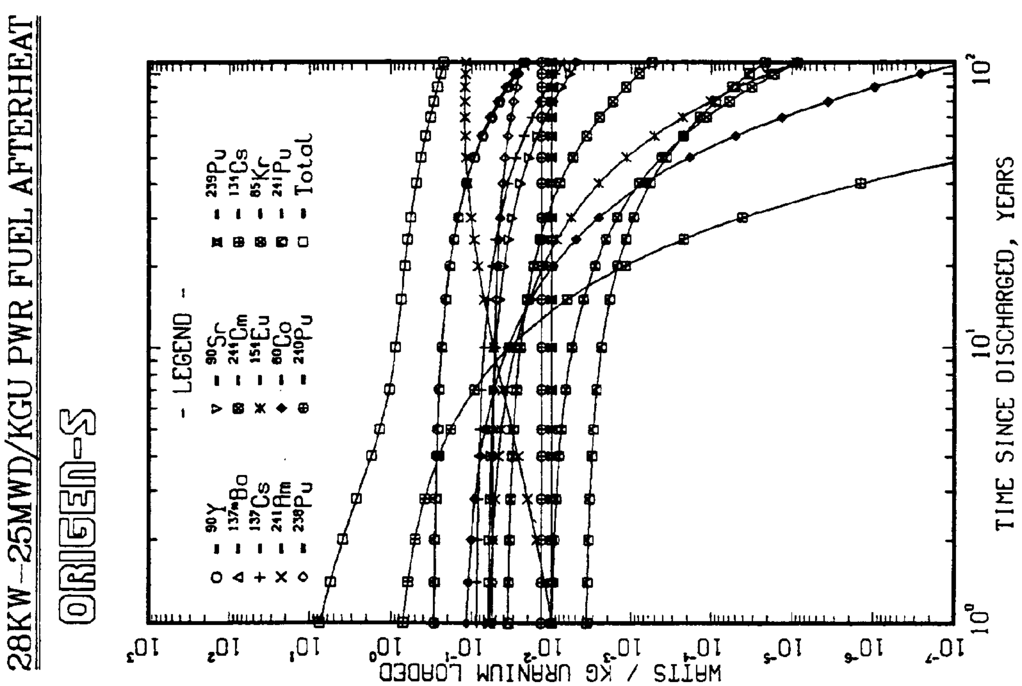

18KW—40MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
90Y
137mBa
238Pu
137Cs
241Am
244Cm
90Sr
154Eu
60Co
240Pu
239Pu
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


18KW—35MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
90Y
137mBa
238Pu
137Cs
241Am
90Sr
244Cm
154Eu
60Co
240Pu
239Pu
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


18KW–45MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
90Y
137mBa
238Pu
137Cs
244Cm
241Am
90Sr
154Eu
60Co
240Pu
239Pu
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


18KW–50MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
137mBa
90Y
238Pu
244Cm
137Cs
241Am
90Sr
154Eu
60Co
240Pu
239Pu
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


28KW-30MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
90Y
137mBa
137Cs
238Pu
241Am
90Sr
244Cm
154Eu
60Co
240Pu
239Pu
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


28KW-25MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
90Y
137mBa
137Cs
241Am
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90Sr
244Cm
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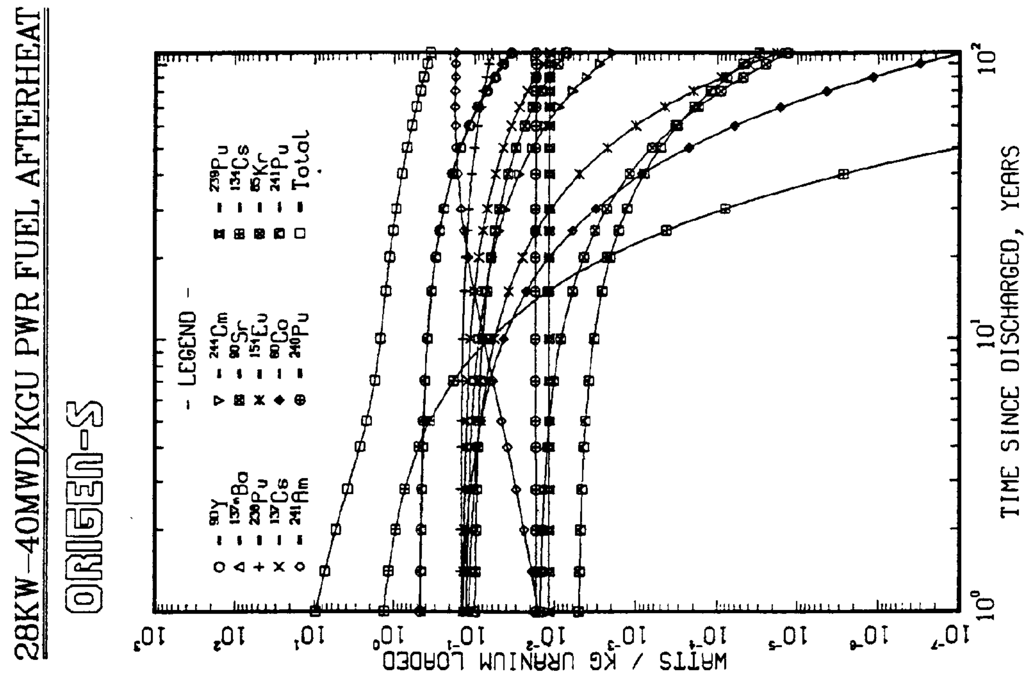
240Pu
239Pu
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS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ORIGEN-S
- LEGEND -
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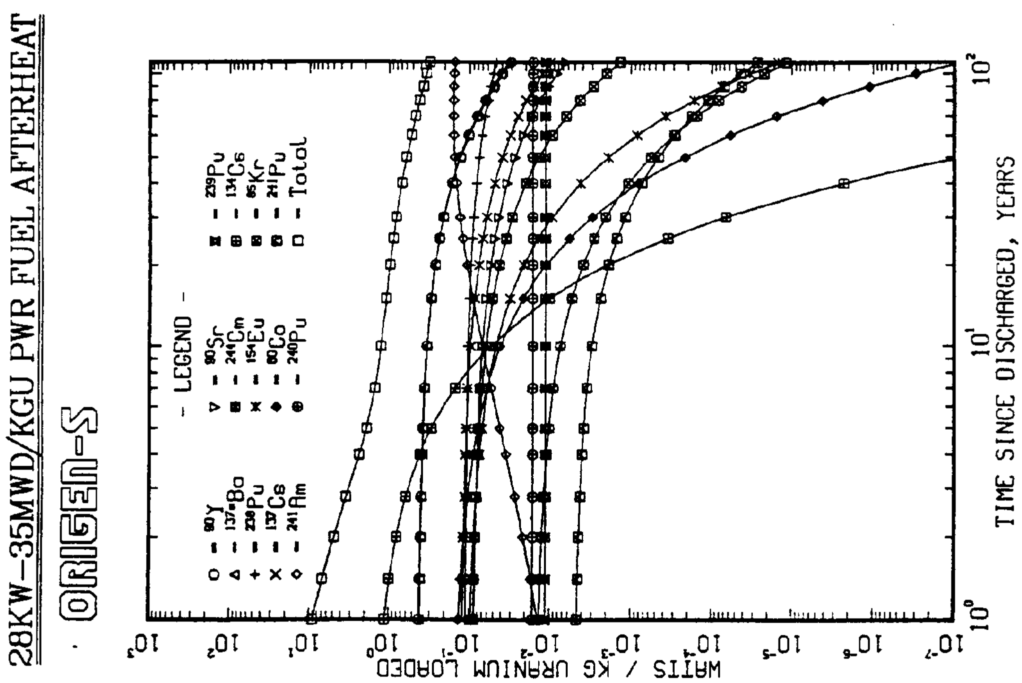
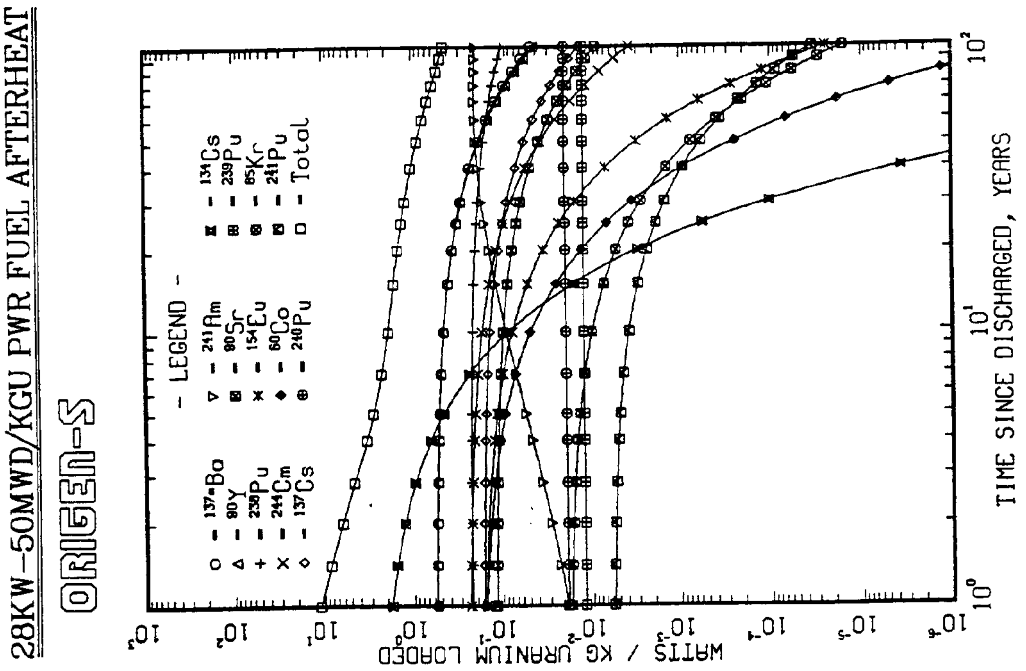
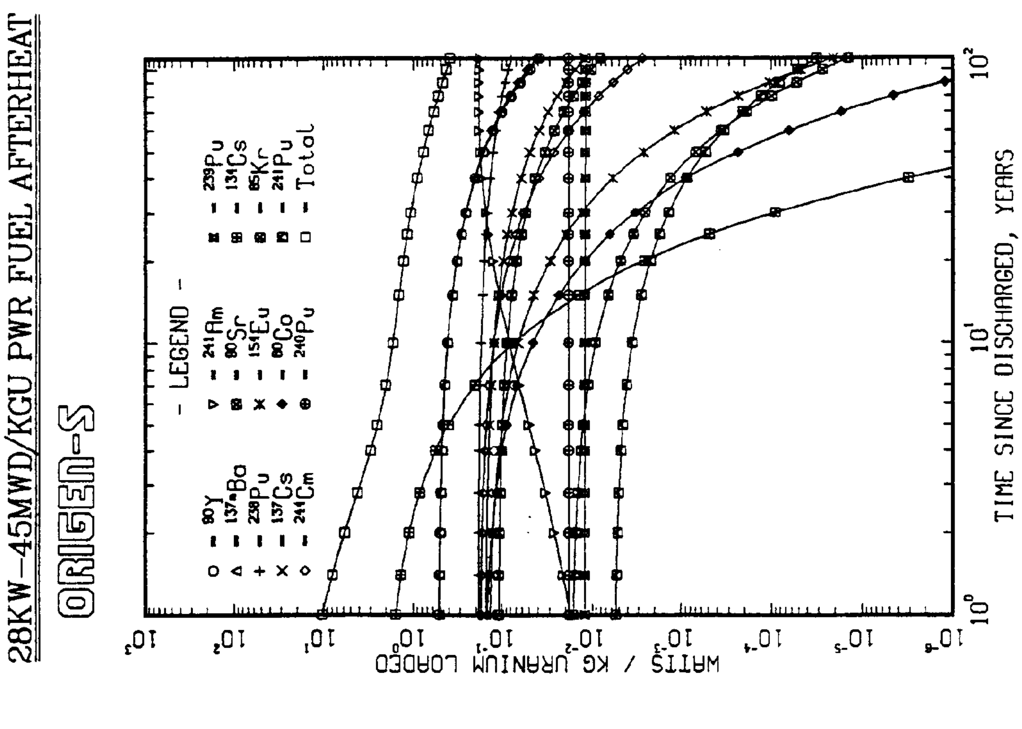
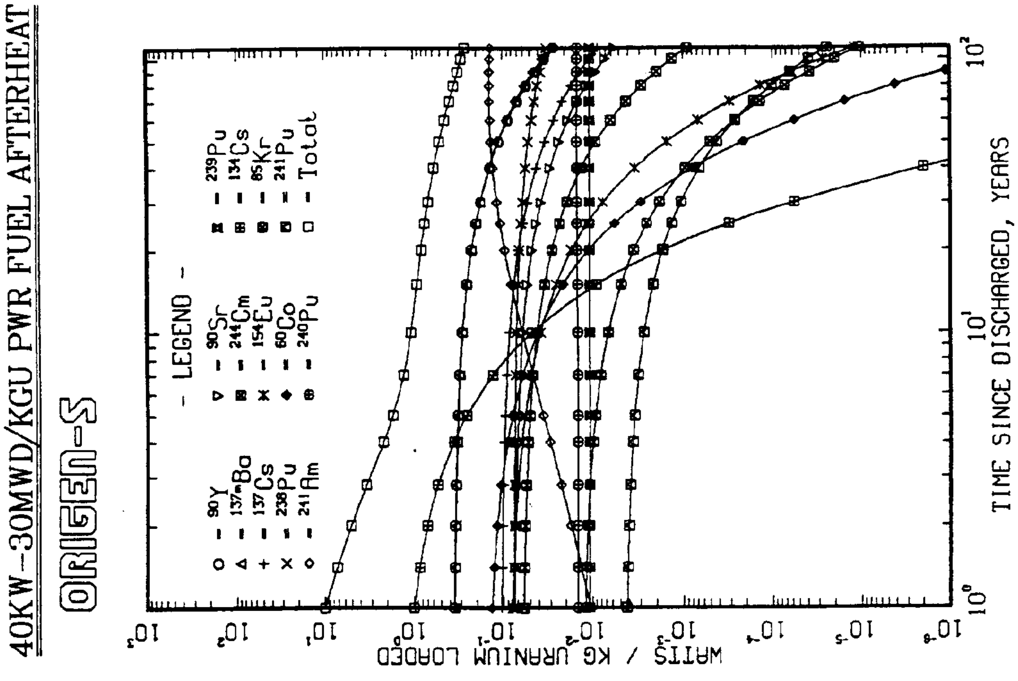
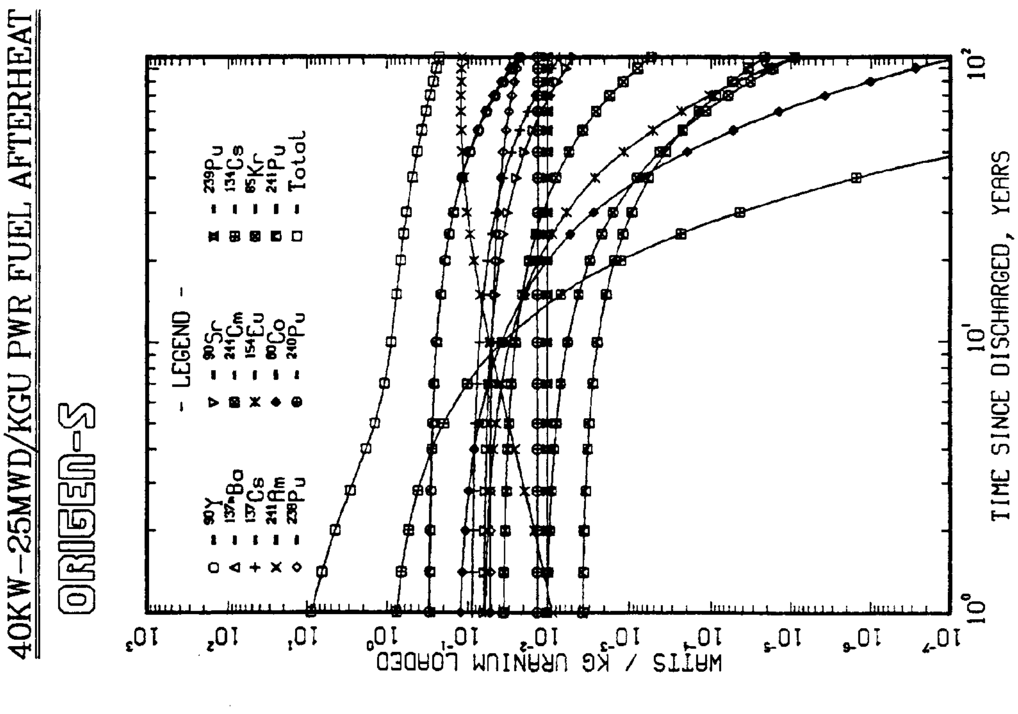
TIME SINCE DISCHARGED, YEARS

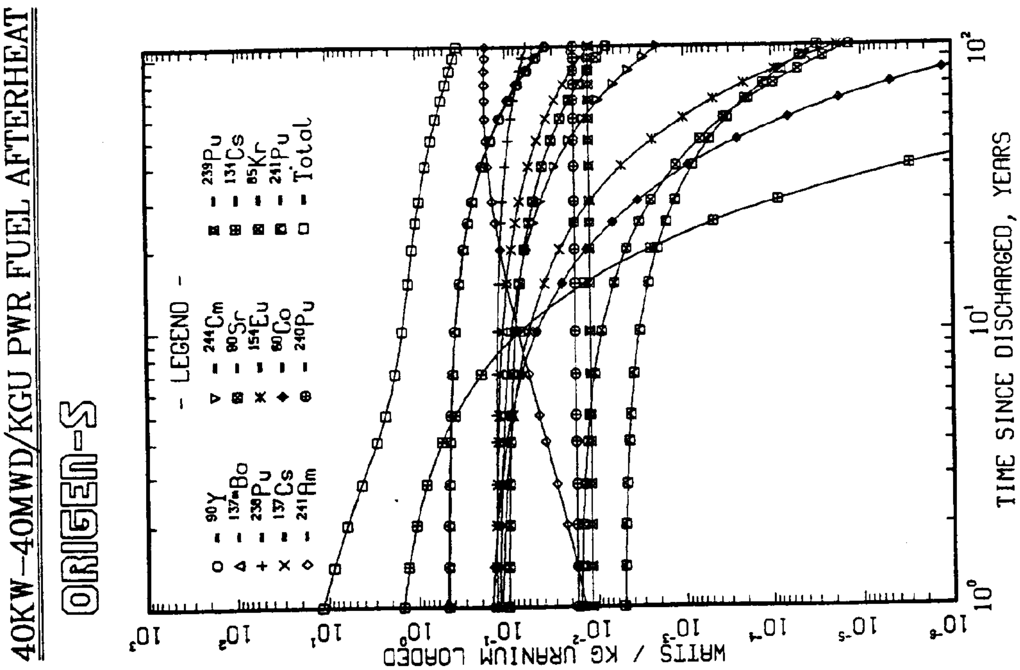

28KW–35MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


28KW-50MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
○ - 137mBa
△ - 90Y
+ - 238Pu
× - 244Cm
◇ - 137Cs
▽ - 241Am
90Sr
× - 154Eu
◆ - 60Co
⊕ - 240Pu
134Cs
239Pu
85Kr
241Pu
□ - Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


28KW-45MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
○ - 90Y
△ - 137mBa
+ - 238Pu
× - 137Cs
◇ - 244Cm
▽ - 241Am
90Sr
× - 154Eu
◆ - 60Co
⊕ - 240Pu
239Pu
134Cs
85Kr
241Pu
□ - Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


40KW–30MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
– LEGEND –
○ – 90Y
△ – 137mBa
+ – 137Cs
× – 238Pu
◇ – 241Am
▽ – 90Sr
▣ – 244Cm
× – 154Eu
◆ – 60Co
⊕ – 240Pu
– 239Pu
⊞ – 134Cs
– 85Kr
– 241Pu
□ – Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


40KW–25MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
– LEGEND –
○ – 90Y
△ – 137mBa
+ – 137Cs
× – 241Am
◇ – 238Pu
▽ – 90Sr
▣ – 244Cm
× – 154Eu
◆ – 60Co
⊕ – 240Pu
– 239Pu
⊞ – 134Cs
– 85Kr
– 241Pu
□ – Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


40KW–40MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
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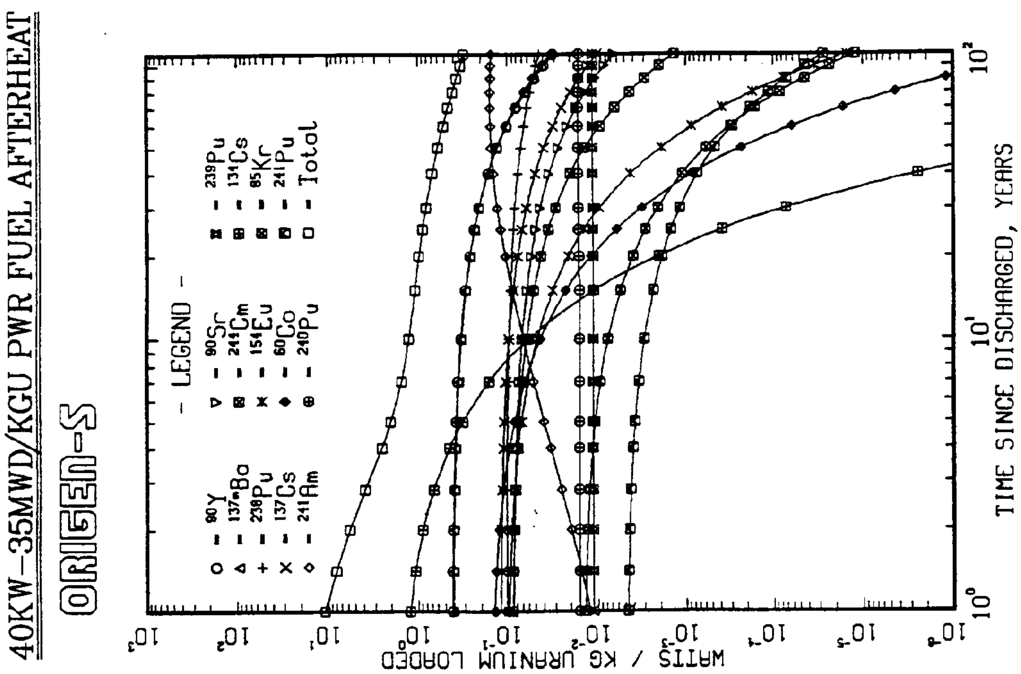
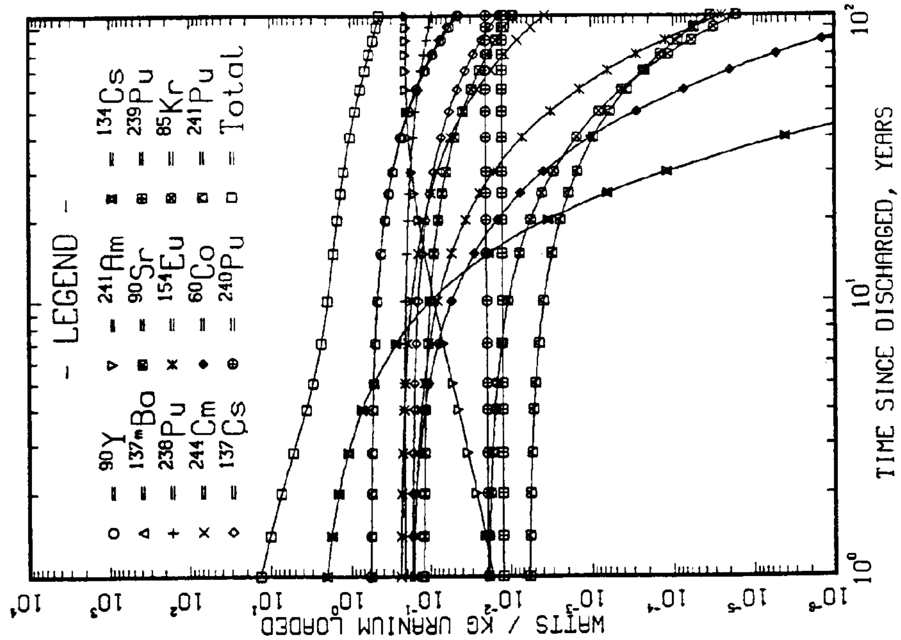
+ - 238Pu
× - 137Cs
◇ - 241Am
▽ - 244Cm
⊠ - 90Sr
✱ - 154Eu
◆ - 60Co
⊕ - 240Pu
⌘ - 239Pu
⊞ - 131Cs
▩ - 85Kr
▣ - 241Pu
□ - Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


40KW–35MWD/KGU PWR FUEL AFTERHEAT
ORIGEN-S
- LEGEND -
○ - 90Y
△ - 137mBa
+ - 238Pu
× - 137Cs
◇ - 241Am
▽ - 90Sr
⊠ - 244Cm
✱ - 154Eu
◆ - 60Co
⊕ - 240Pu
⌘ - 239Pu
⊞ - 131Cs
▩ - 85Kr
▣ - 241Pu
□ - Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS

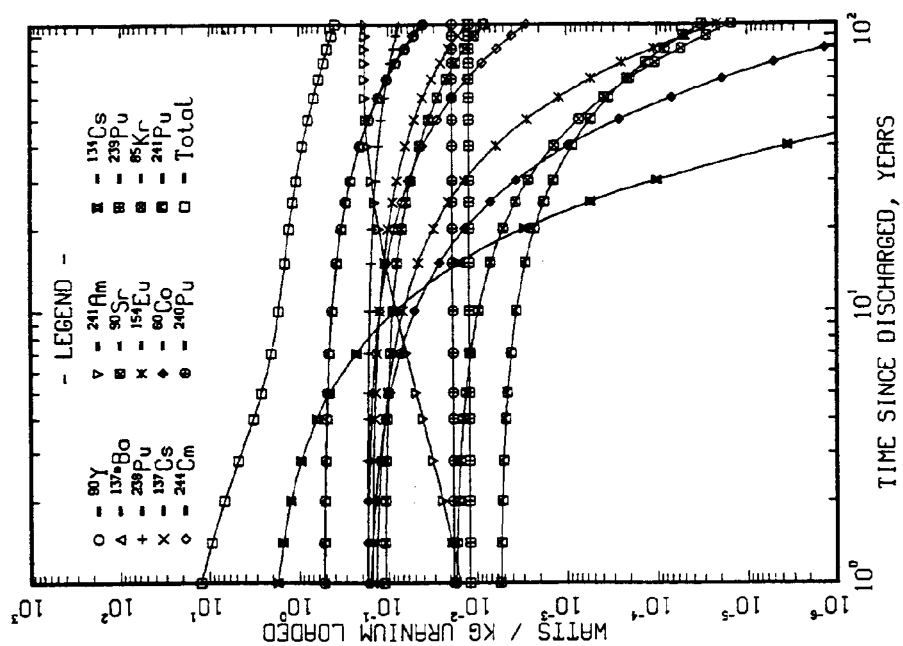
40KW–50MWD/KGU PWR FUEL AFTERHEAT

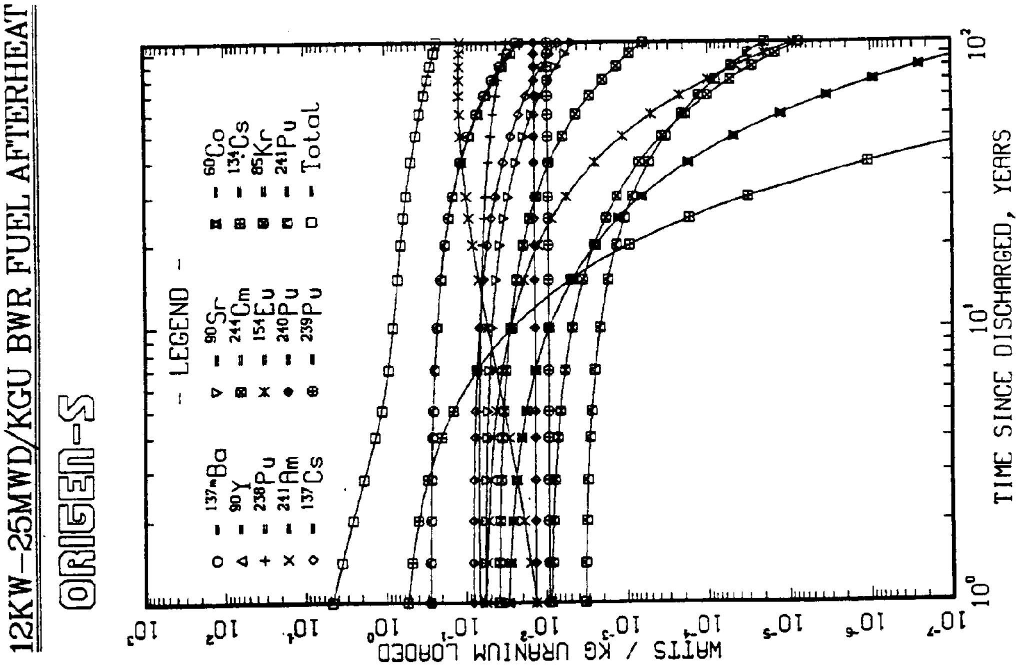
ORIGEN-S



40KW–45MWD/KGU PWR FUEL AFTERHEAT

ORIGEN-S




12KW–25MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
– LEGEND –
137mBa
90Y
238Pu
241Am
137Cs
90Sr
244Cm
154Eu
240Pu
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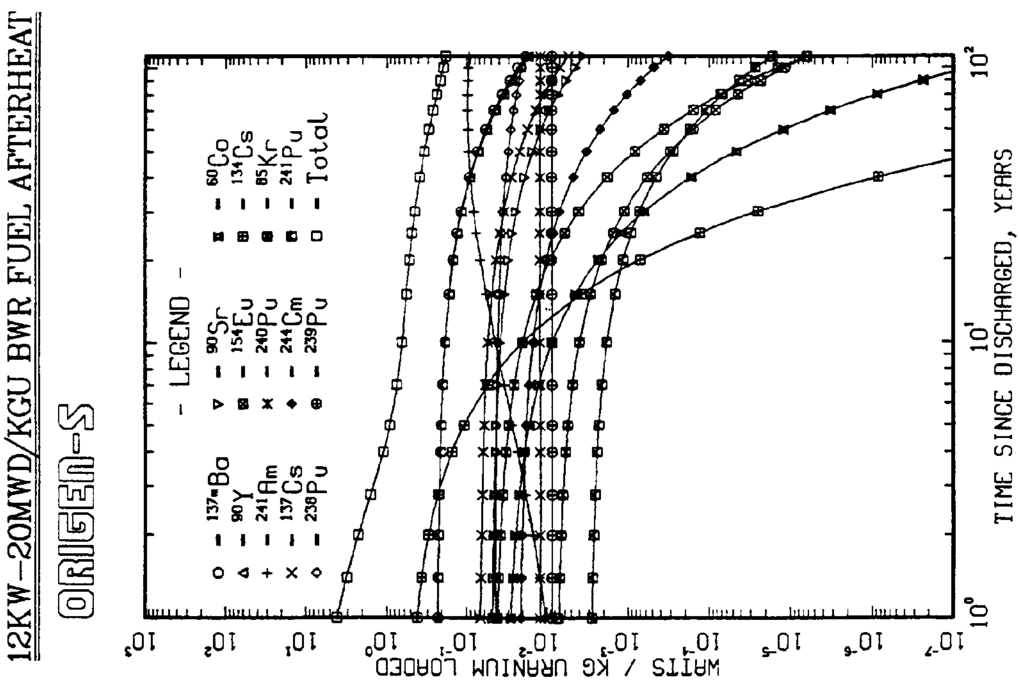
60Co
134Cs
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


12KW–20MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
– LEGEND –
137mBa
90Y
241Am
137Cs
238Pu
90Sr
154Eu
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239Pu
60Co
134Cs
85Kr
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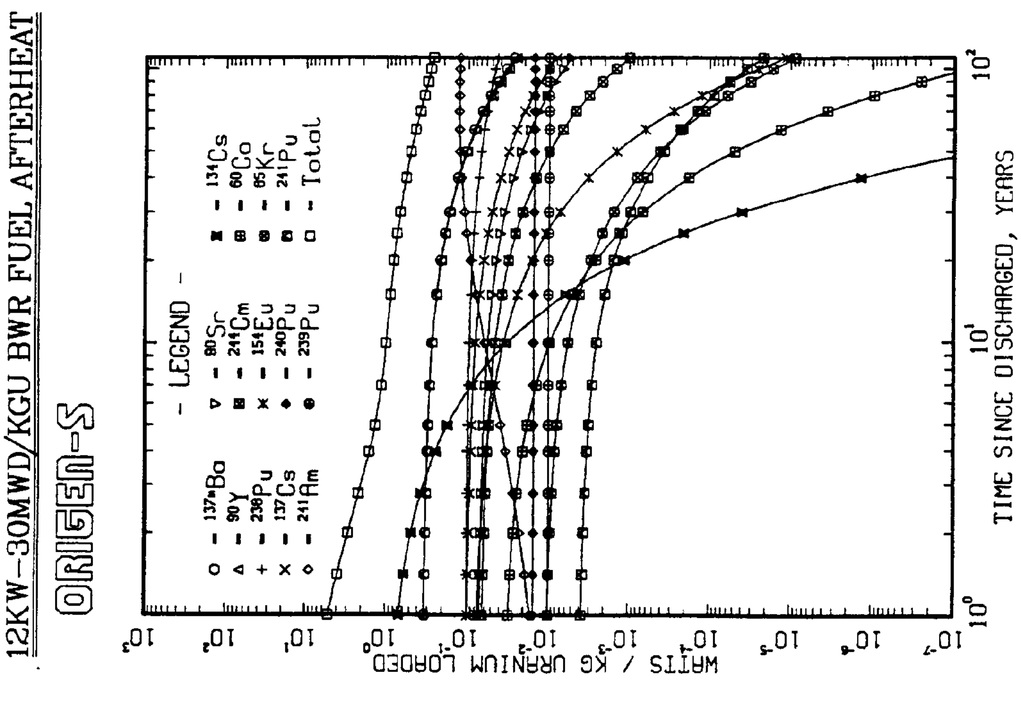
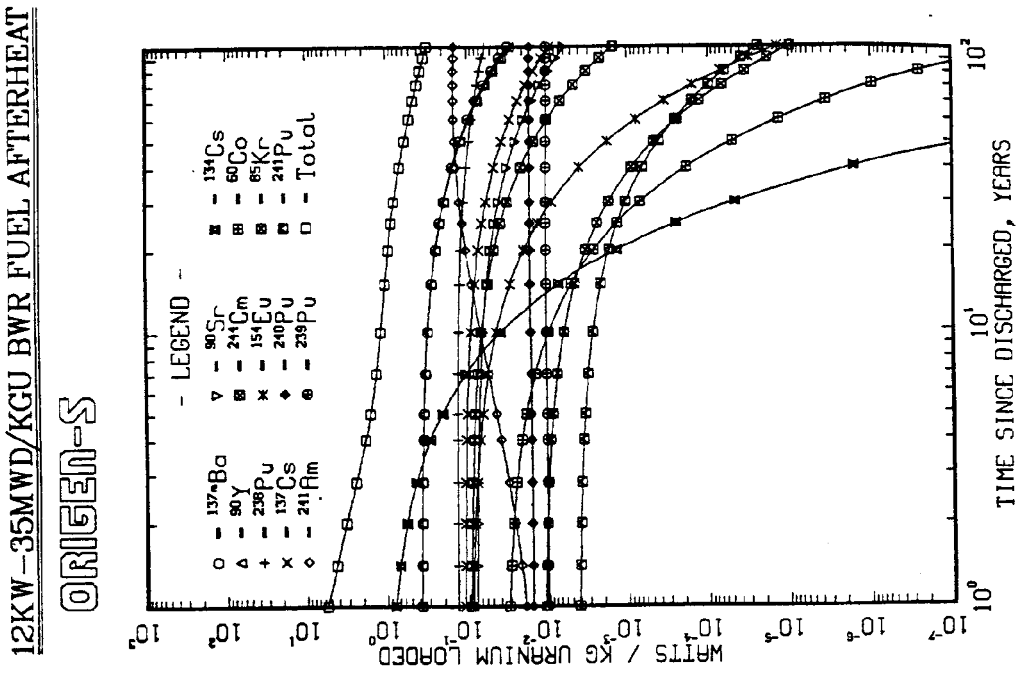
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


12KW–30MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
– LEGEND –
o – 137mBa
Δ – 90Y
+ – 238Pu
x – 137Cs
◇ – 241Am
▽ – 80Sr
– 244Cm
x – 154Eu
◆ – 240Pu
– 239Pu
– 134Cs
– 60Co
– 85Kr
– 241Pu
□ – Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS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ORIGEN-S
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+ – 238Pu
x – 137Cs
◇ – 241Am
▽ – 90Sr
– 244Cm
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◆ – 240Pu
– 239Pu
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□ – Total
WATTS / KG URANIUM LOADED
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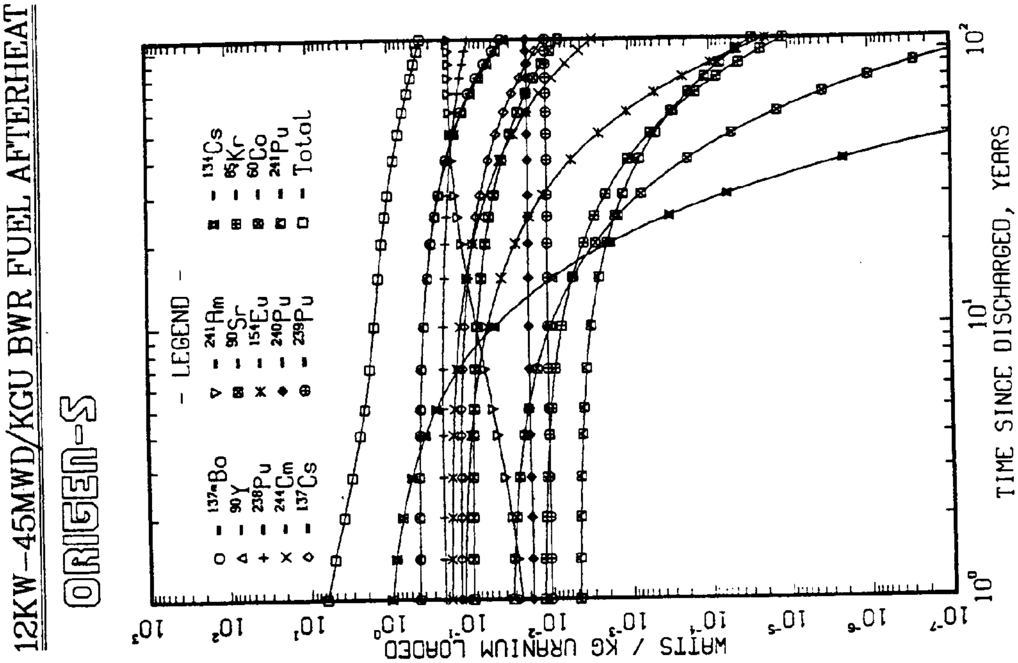
12KW–45MWD/KGU BWR FUEL AFTERHEAT
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244Cm
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85Kr
60Co
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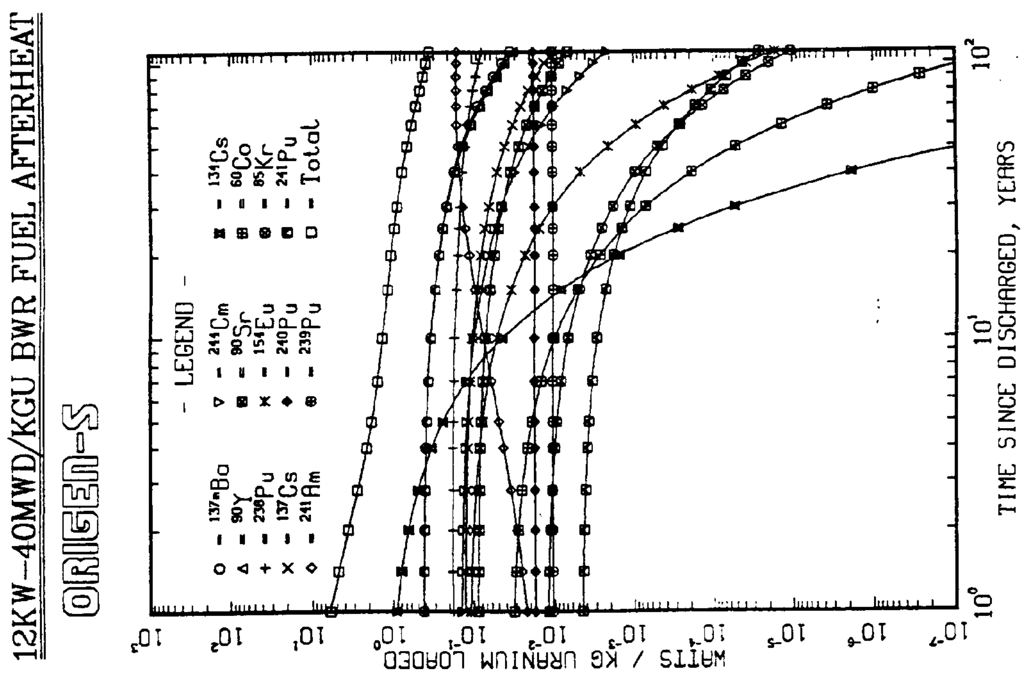
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


12KW–40MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
– LEGEND –
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90Y
238Pu
137Cs
241Am
244Cm
90Sr
154Eu
240Pu
239Pu
134Cs
60Co
85Kr
241Pu
Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS

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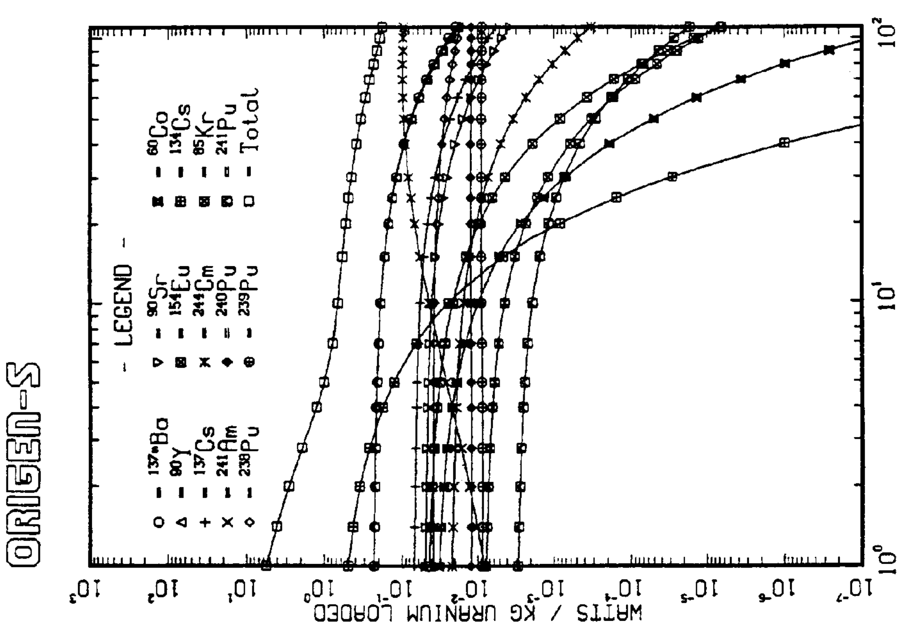

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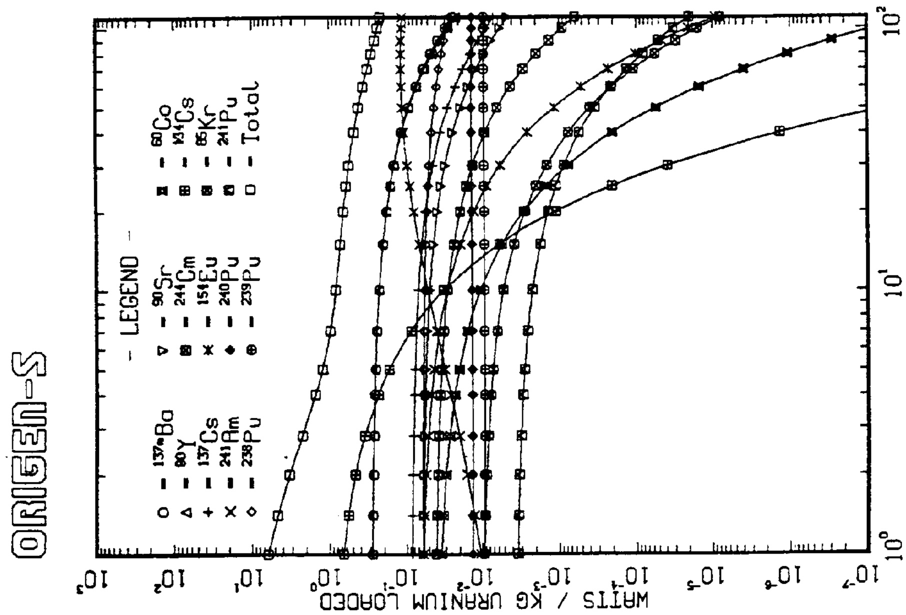

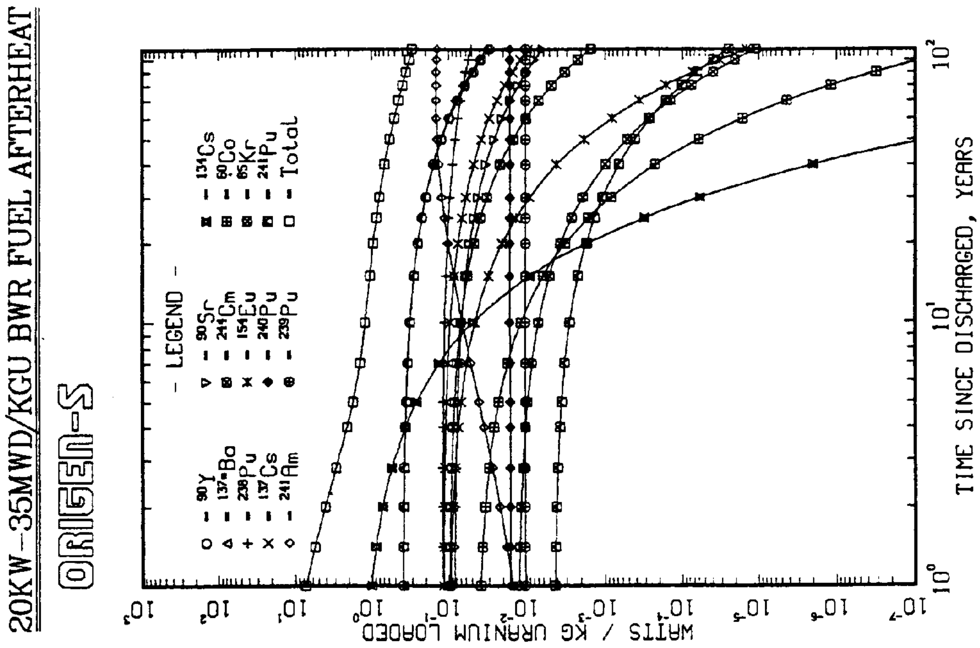
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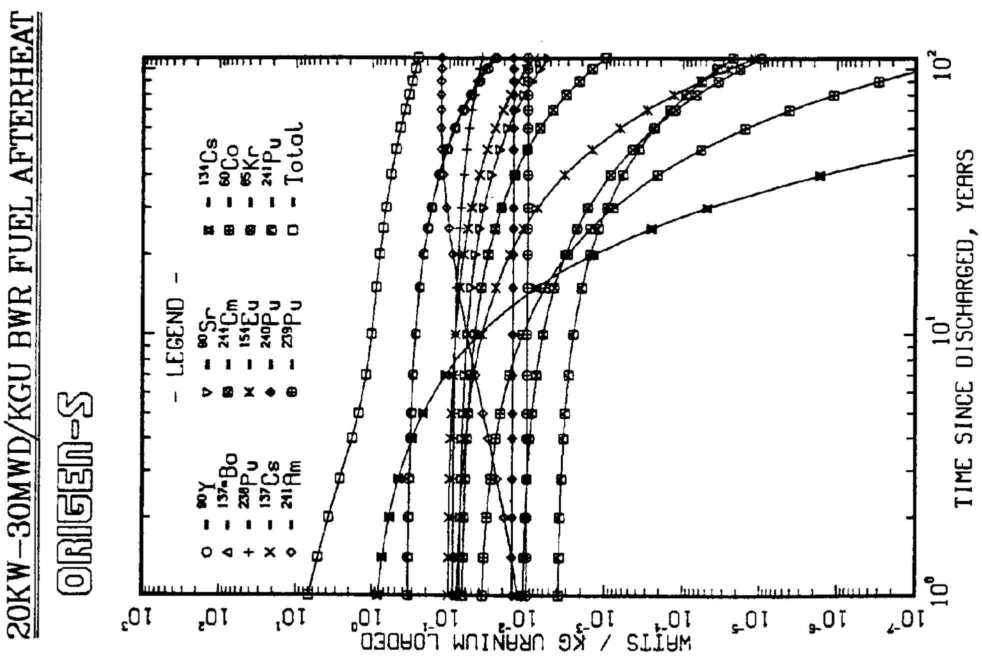
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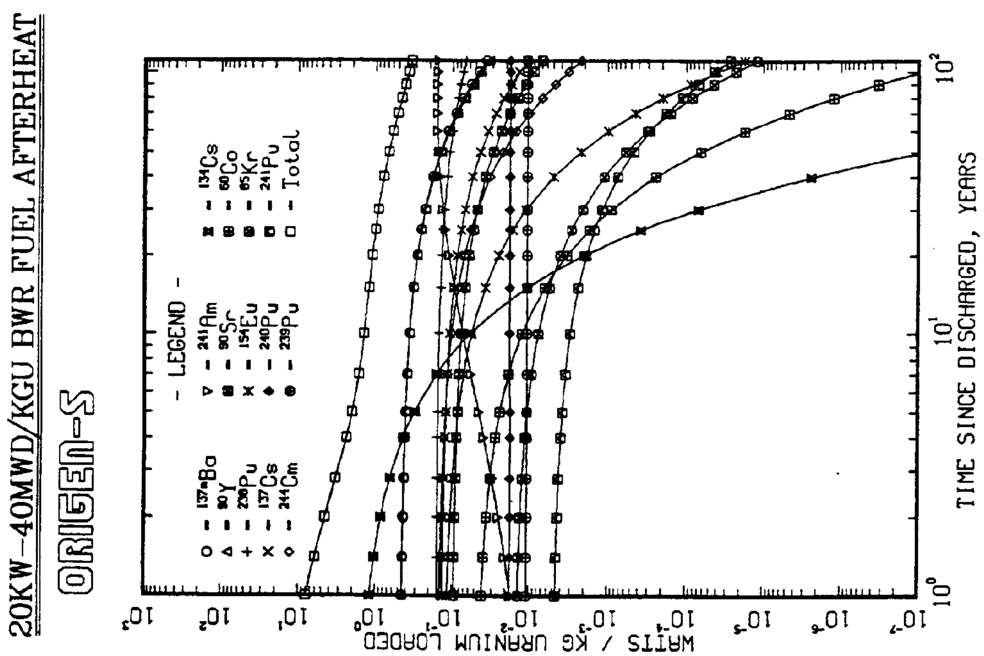
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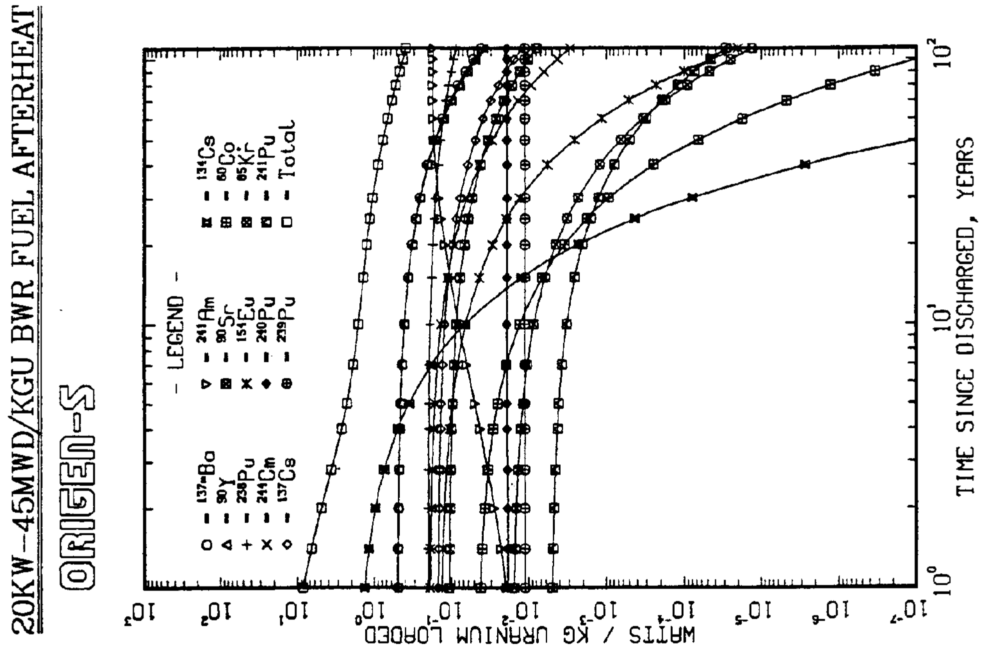
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WATTS / KG URANIUM LOADED
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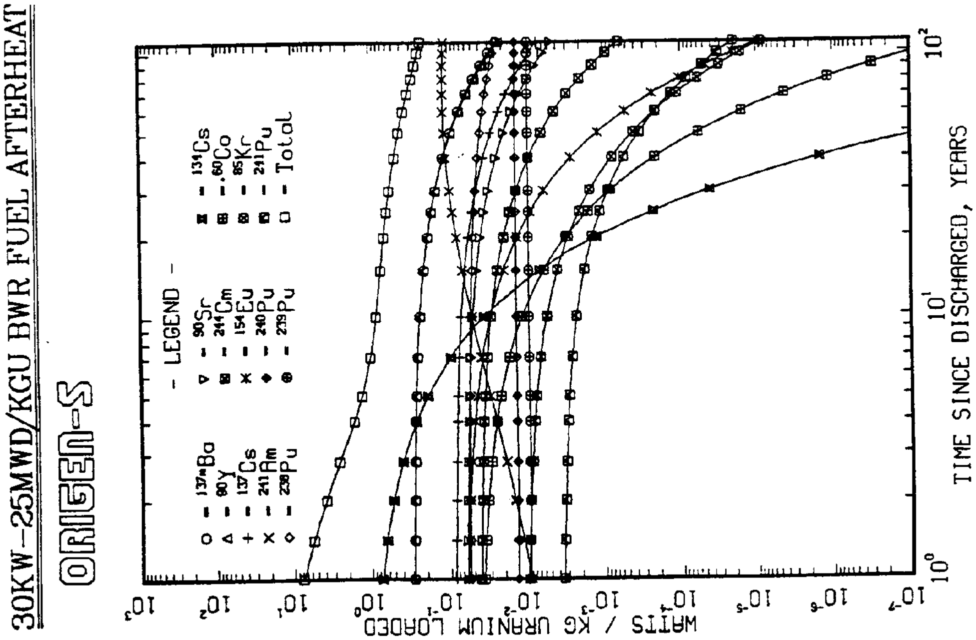
30KW-25MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
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137mBa
90Y
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244Cm
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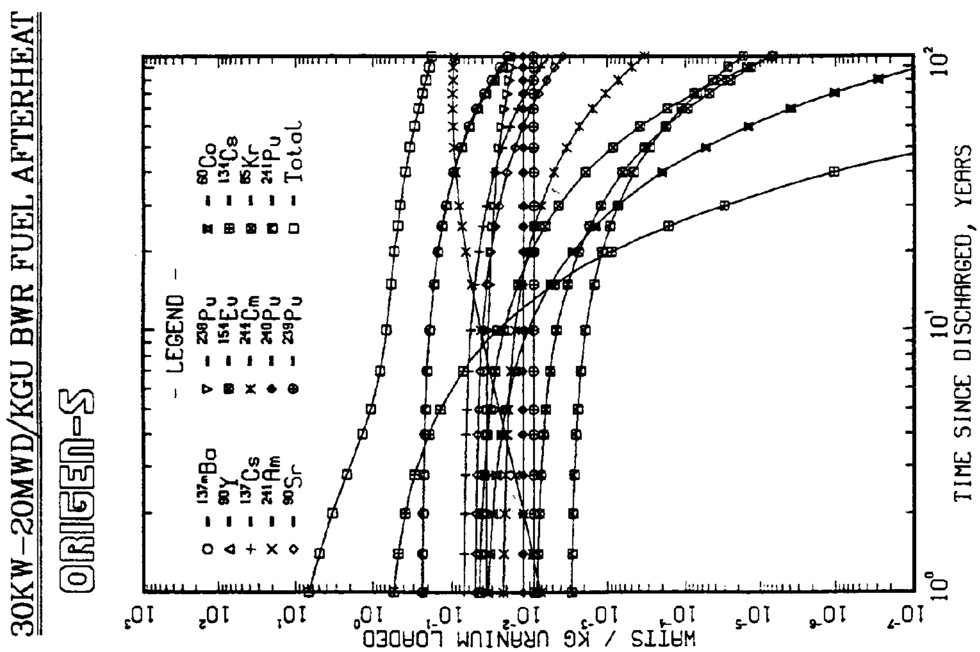
Total
TIME SINCE DISCHARGED, YEARS
WATTS / KG URANIUM LOADED


30KW-20MWD/KGU BWR FUEL AFTERHEAT
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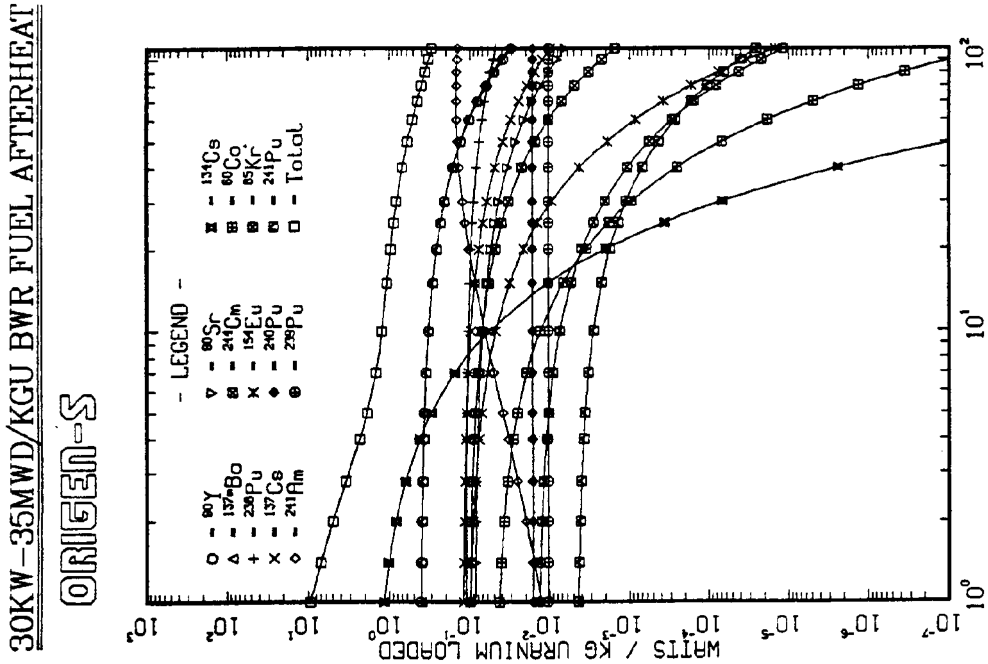
Total
TIME SINCE DISCHARGED, YEARS
WATTS / KG URANIUM LOADED


30KW–35MWD/KGU BWR FUEL AFTERHEAT
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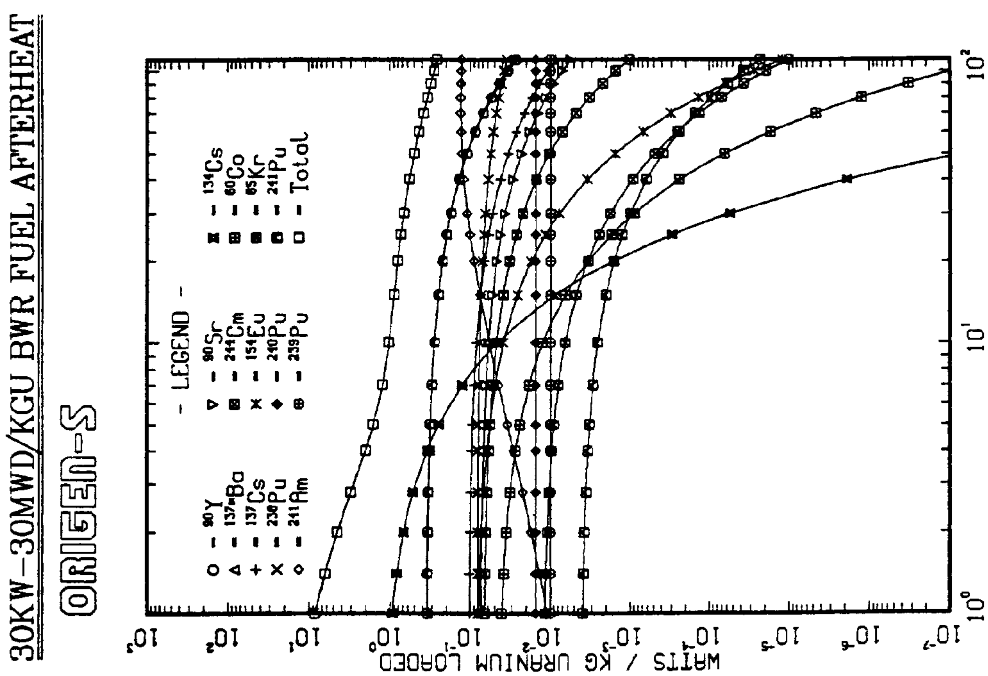
30KW–30MWD/KGU BWR FUEL AFTERHEAT
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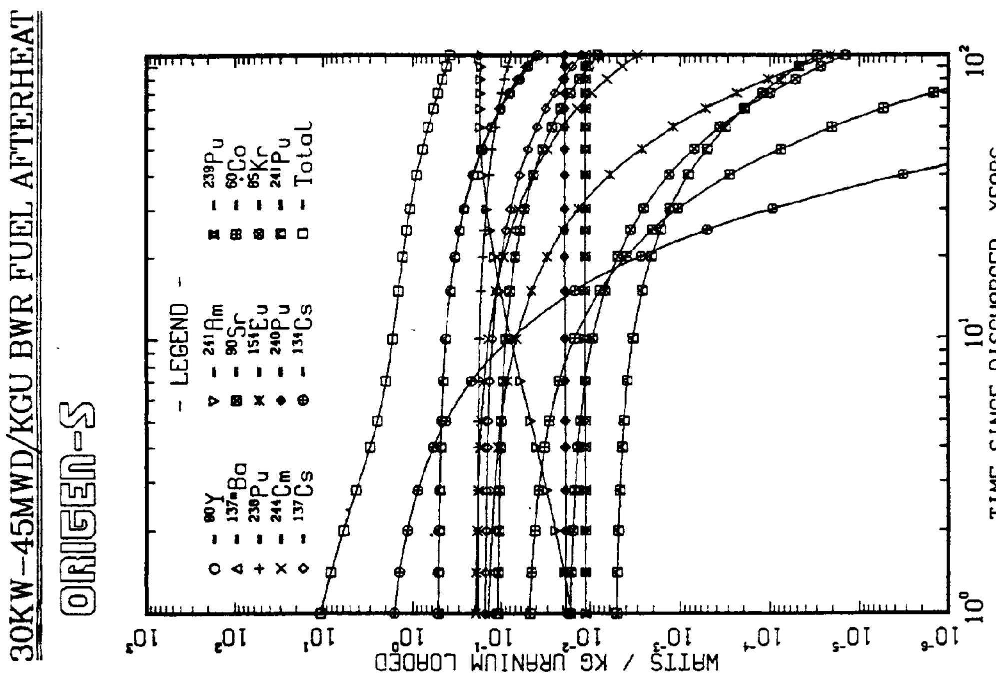
30KW–45MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
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◆ - 240Pu
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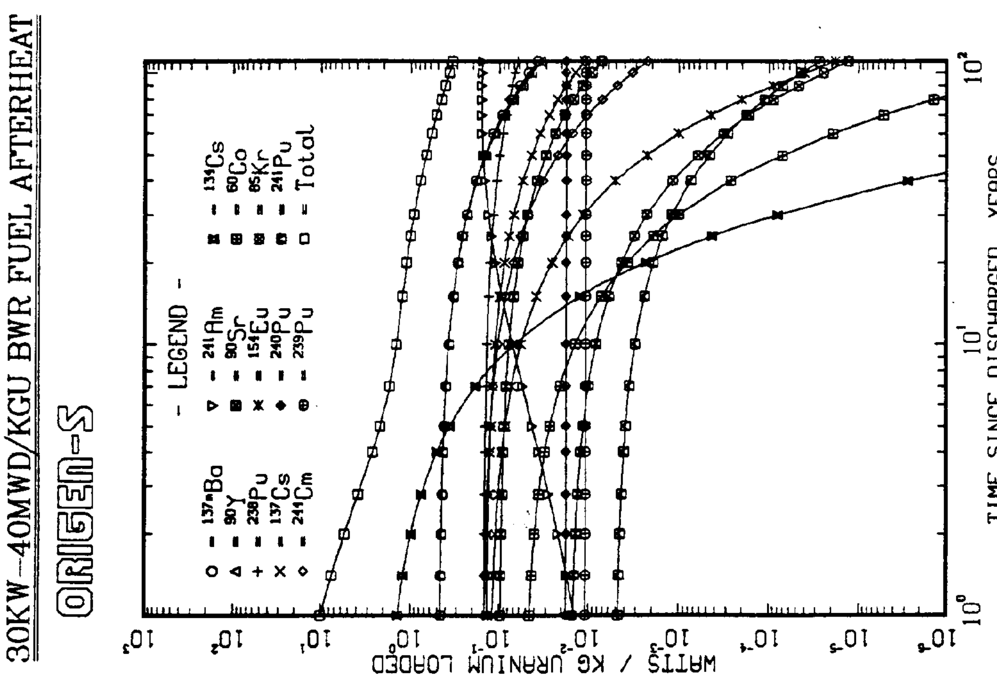
□ - Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS


30KW–40MWD/KGU BWR FUEL AFTERHEAT
ORIGEN-S
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+ - 238Pu
× - 137Cs
◇ - 244Cm
▽ - 241Am
90Sr
× - 154Eu
◆ - 240Pu
⊕ - 239Pu
134Cs
60Co
85Kr
241Pu
□ - Total
WATTS / KG URANIUM LOADED
TIME SINCE DISCHARGED, YEARS

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