CONF-910435--79

DE91 012504

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Submitted to
1991 International
High-Level Radioactive Waste
Management Conference
April 28 - May 3, 1991

*Managed by Martin Marietta Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

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ABSTRACT

A proposed revision to Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation" has been developed for the U.S. Nuclear Regulatory Commission. The proposed revision includes a data base of decay heat rates calculated as a function of burnup, specific power, cooling time, initial fuel ^{235}U enrichment, and assembly type (i.e., PWR or BWR). Validation of the calculational method was done by comparison with existing measured decay heat rates. Procedures for proper use of the data base, adjustment formulae accounting for effects due to differences in operating history and initial enrichment, and a defensible safety factor were derived.

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). The U.S. Nuclear Regulatory Commission (NRC) has provided technical guidance in this area with the issuance of Regulatory Guide 3.54, entitled "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation." However, 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 recently been used to produce a data base of decay heat rates that serves as a basis for a substantial revision to the regulatory guide. The purpose of this paper is to discuss the technical work performed to develop the data and procedures incorporated in the proposed revision to the regulatory guide.

BACKGROUND

The current decay heat regulatory guide 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. However, the current guide relies on a decay heat data base calculated only for PWR fuel.[4] Thus, with no measured heat generation data or validated calculations for BWR fuel, the current guide incorporates increased safety factors for decay heat from BWR assemblies. In addition, the decay heat data base was developed for several burnup values but using only a single maximum specific power (rather than a range of specific power). The total effect of these limitations is that the current guide provides decay heat rates that are accurate (within a few 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, conservative heat rates are produced by the guide. Note that the main cause for this overestimation of heat rates (at least for PWR fuel) is the result of using an upper envelope of the possible operating powers and is not the result of the computational model used to produce the data base.

An example demonstrating the significance of using the actual specific power as opposed to an enveloping 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 two 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.

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, and some major improvements have been made in the computational tools[1,2,5-7] typically used to evaluate decay heat rates for spent fuel. Thus, the U.S. Nuclear Regulatory Commission decided to initiate a project to validate a selected computational tool against the measured data and then use the validated code in preparation of a revised decay heat guide. The goal of the revision was to reduce the conservatism in the current guide by (1) developing separate decay heat data bases for PWR and BWR fuel and (2) expanding the decay heat data base to encompass a broader range of parameters selected to characterize PWR and BWR spent fuel. Other objectives considered while developing the guide revision were ease-of-use and a more complete analysis of an adequate safety factor.

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CODE VALIDATION

The SAS2H/ORIGEN-S[1,2] analysis sequence within the SCALE-4 code system[3] was selected as the calculational tool to produce the data base of decay heat rates for the proposed guide revision. The SAS2H automated sequence produces burnup-dependent cross sections for the ORIGEN-S point-depletion calculation using spectrum data from one-dimensional neutronics analyses of a specified PWR or BWR assembly model. A preliminary SAS2H case was performed for each reactor type (i.e., BWR or PWR) to produce an ORIGEN-S library that was used as the starting library in all subsequent SAS2H cases. Each preliminary SAS2H case updated the cross sections for 181 fission products and provided a base PWR or BWR library. The subsequent SAS2H cases used the BWR or PWR starting library and updated cross sections for 38 to 39 significant actinides and fission products (plus 6 gadolinium isotopes for the BWR) as a function of burnup. The cross-section update was obtained from the neutronics analysis of the fuel assembly model. The fuel spectrum from this analysis also provides new values for the THERM, RES, and FAST parameters[1] used by ORIGEN-S. Use of the SAS2H/ORIGEN-S method allows a wide range of operating histories and assembly types to be considered in a reliable fashion and enables most of the conservatism to be removed from the computed decay heat rates.

Prior to computing the data base of decay heat rates for the proposed guide revision, the reliability of the SAS2H/ORIGEN-S sequence was demonstrated by comparing calorimetric measurements of spent fuel assembly heat rates with values computed by using code input representative of the design and operating characteristics of the assemblies. In this study, results were compared for ten PWR and ten BWR spent fuel assemblies obtained from three reactors: Point Beach Unit 2 (PWR), Turkey Point Unit 3 (PWR), and Cooper Nuclear Station (BWR). The assemblies with the highest burnups and initial ^{235}U enrichments are from the Point Beach reactor. The measured decay heat values along with the basic design and operating history data for each measured assembly is provided in Refs. 8-12. A more complete and compact description of this data is included in Ref. 13.

Comparisons of measured and calculated decay heat rates of spent fuel assemblies from the three reactors in this study are listed in Tables 1-3. There is at least one excessively high percent difference between the measured and calculated values in each of the three tables. These differences were not excluded because it was decided to use comparisons for all reported measurements for which pertinent parameters were available. All of the computed heat rates were higher than the measured values for the Point Beach PWR cases in Table 1. However, except for the 16.2% value, the differences did not exceed 3%. This was the only reactor for which the average difference exceeded the standard deviation (i.e., 3.0 ± 1.9%). 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 percent difference of the other assemblies would have been 1.7 ± 0.9%.

The computed heat rates of the Turkey Point PWR assemblies in Table 2 were both higher and lower than measured values. A previous comparison[4] of SAS2 results with measurements applied equal burnups and specific powers[10] for the three cycles of the D-assemblies. This rather rough estimate of operating history was replaced here with more complete data given by the utility. Three of the assembly average differences were within 2.3%. 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. Its lower calculated value could be due to exceptionally low operating powers during the first part of the first cycle. The average assembly percent difference, however, of -0.7 ± 1.7% indicates good overall agreement between measured and calculated values.

The percent differences in the decay heat comparisons in Table 3 for the Cooper Nuclear Station BWR assemblies extended over a much wider range than those for the PWR reactors. However, the measurement values were in the range of 62.3 to 395.4 watts, as opposed to the range of 625 to 1550 watts for the PWR reactors. Because measurement precision tends to be represented as a constant heat rate instead of a percent of the total heat rate, larger percent differences would result from the lower values being measured. 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% indicates good overall agreement between calculated and measured values.

A summary of percent differences in comparisons of measured and calculated spent fuel decay heat rates for all cases and assemblies is presented in Table 4. 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 two standard deviations the percent differences 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 does not exceed 2.4% plus any nonconservative bias in the measurements. It appears reasonable to conclude that with respect to the applications of computed heat rates in this project, the comparisons of measurements and calculations are in good agreement.

HEAT RATE DATA BASE

The data base of decay heat rates was produced using 36 SAS2H/ORIGEN-S cases (18 PWR and 18 BWR). The cases for each reactor were computed for six different burnups at three different specific powers. The burnup range (incremented by 5 MWd/kgU) was 20 to 45 MWd/kgU for the BWR cases and 25 to 50 MWd/kgU for the PWR cases. The specific powers considered were 12, 20, and 30 kW/kgU for the BWR cases and 18, 28, and 40 kW/kgU for the PWR cases. Within each case, final decay heat generation rates

Table 1. Point Beach PWR measured[a] and computed decay heat rates

Assembly ID	Burnup, MWd/kgU	Initial ^{235}U wt %	Cooling time, d	Heat rate, watts Meas.	Heat rate, watts 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.[illegible]	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
Std dev						±2.3	±1.9

[a] See Ref. 11.
[b] Static test.
[c] Recirculation test.

Table 2. Turkey Point PWR measured[a] and computed decay heat rates

Assembly ID	Burnup, MWd/kgU	Initial ^{235}U wt %	Cooling time, d	Heat rate, watts Meas.	Heat rate, watts 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
Std dev						±1.3	±1.7

[a] See Ref. 11.

were computed at 20 different cooling times in the range of 1 to 110 years. An example of the tabular data produced for the guide revision is shown in Table 5.

The assembly design (Westinghouse 17 x 17 and General Electric 8 x 8) and operating characteristics applied in the SAS2H/ORIGEN-S cases represent generic, yet realistic data, and are presented in detail in Ref. 13. Uptimes equal to 80% of the cycle time (which includes the reload downtimes) were used for all cycles except the last one in which it was considered to be 100%. The numbers of cycles used for the different burnup cases were three cycles for the two lowest burnup cases, four cycles for the next two higher in burnup, and five cycles for the two highest in burnup. Adjustment factors to account for effects due to differences in operating history and initial enrichment will be discussed in subsequent sections.

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

PROPOSED GUIDE PROCEDURE

In the proposed guide procedure, 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. The specific power value used to obtain an interpolated decay heat value is an average specific power defined uniquely for use in the guide procedure. The guide definition is

$$P_{ave} = \frac{B_{tot}}{T_N + 0.8 \sum_{n=1}^{N-1} T_n}, \qquad (1)$$

where B_{tot} = total assembly burnup,
T_n = cycle time between core reloads,
T_N = uptime of last cycle, and
N = number of cycles.

This definition accounts for the difference between the actual operating history of the assembly and the history (80% uptime for N-1 cycles and 100% uptime for last cycle) used in the computation of the data base of decay heat rates.

Table 3. 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, watts Meas.	Heat rate, watts 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
Std dev						±1.7	±2.6

[a]See Ref. 12.

Table 4. Summary of decay heat rate comparisons

Type of summary	Number	% difference[a] ± std dev
Summary by cases:		
Average Point Beach case	6	−3.6 ± 2.3
Average Turkey Point case	8	−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%.

As noted in the last section, the guide is most accurate only when P_{ave}, B_{tot}, and the cooling time T_c are within the ranges used to produce the decay heat data base. However, if P_{ave} or B_{tot} is below the minimum value used for the data base, then the decay heat data associated with the minimum specific power or burnup may conservatively be used. Also, if P_{ave} exceeds (by less than 35%) 30 kW/kgU for BWR fuel or 40 kW/kgU for PWR fuel, the table corresponding to the maximum specific power may be used, in addition to a power adjustment factor f_p in order to evaluate a decay heat rate. The data base (and thus the guide) should not be applied if B_{tot} exceeds the maximum burnup (45 MWd/kgU for BWR or 50 MWd/kgU for PWR) in the tables, or if T_c is less than the minimum (one year) or exceeds the maximum (110 years) cooling time of the tables.

The interpolated decay heat rate p_{tab} corresponds to the computed heat rate at the power, burnup, and cooling time specified and does not account for significant changes between other conditions (e.g., ^{235}U enrichment or operating history) used in the calculations and those of the actual assembly. 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 formulae are derived for adjustment factors that account for deviations from the calculations in parameters of the assembly such as the initial ^{235}U enrichment (factor f_e) and variations in the last two operating cycle powers (factors f_7 and f_7'). If necessary, these factors are applied as adjustments to the interpolated decay heat value p_{tab}. Finally, a safety factors—developed as a function of reactor type, burnup, and cooling time—is applied. Thus, the final decay heat rate is determined by

$$p_{final} = (1 + 0.01S) f_7 f_7' f_p f_e p_{tab} \,. \qquad (2)$$

To simplify application of the guide procedures, a personal computer (PC) code called ARC (Afterheat Rate Calculation) has been written (in FORTRAN 77) and tested. This interactive program prompts the user for the basic data needed to apply the guide procedures. The code performs the interpolation of the decay heat data base, calculates the adjustment factors and safety factors, and finally provides the user with values of p_{tab}, p_{final}, and the decay heat rate before application of the safety factor. The ARC code will not be issued with the regulatory guide but will be readily available from public code distribution centers.

ADJUSTMENT FACTORS

An extensive number of ORIGEN-S calculations were performed to develop a proper basis for the adjustment factor formulae noted in the previous section. Each of these formulae—f_p, f_e, f_7, and f_7'—will be discussed briefly in this section.

The maximum specific power P_{max} (equal to 30 kW/kgU for BWR assemblies and 40 kW/kgU for PWR assemblies) considered in producing the data base of decay heat rates is typically greater than the average specific power [defined by Eq. (1)] for all current commercial light water reactors. However, for cases where $1 \le P_{ave}/P_{max} \le 1.35$, the guide provides an adjustment factor f_p equal to the square root of P_{ave}/P_{max}. For any operating history the decay heat rate at a given cooling time increases at least proportionately with an increase in the number of fissions or specific power. However, f_p is not directly proportional to P_{ave}/P_{max} because the heat rate is associated with both a specified power and burnup such that both are increased similarly when the cycle times are unchanged. Thus, if the heat rate were increased in direct proportion to the power for a given burnup, the adjusted heat rate would not apply to the same burnup.

The initial ^{235}U enrichments used in production of the data base of decay heats are typical equilibrium reload enrichments that would ensure normal operation of the respective reactor type for the corresponding burnup (see Table 5 for the enrichments used for the PWR cases). 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.

Table 5. PWR spent fuel heat generation rates, watts per kilogram U, for specific power = 28 kW/kgU

Cooling time, years	Fuel burnup, MWd/kgU[a] 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	0.309	0.352	0.396	0.440

[a]From lowest to highest burnup, the assumed initial ^{235}U enrichment values are 2.4 wt %, 2.8 wt %, 3.2 wt %, 3.6 wt %, 3.9 wt %, and 4.2 wt %, respectively.

To derive the enrichment factor f_e, ten additional ORIGEN-S cases were calculated for each reactor type using different initial enrichments. At the minimum and maximum burnups and specific powers of each type of reactor, 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 the maximum percent heat rate change obtained from these cases was used in deriving formulae for f_e that are a function of the ratio of the actual (E_s) and tabulated enrichment (E_{tab}) for a particular burnup, the cooling time, and the reactor type. The value for f_e is greater than one if $E_s < E_{tab}$ and less than one if $E_s > E_{tab}$. The extreme variation of the burnup and specific power in each set of cases appeared to be sufficient that a conservative envelope of the percent heat rate changes would produce a conservative formula for f_e.

As noted in a previous section, the data base of decay heat rates was calculated by applying different total burnups and average specific powers to a "standard" operating history profile. However, the distribution of uptime and downtime in the operating history of an actual assembly could be considerably different from that used in the calculations. Three-cycle PWR operating histories were developed to investigate operating history changes under the conditions that the total burnup, the cycle times, and the average power are unchanged. These conditions restrict possible changes in the operating history 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 study showed that changes (from the "standard" cycle) in the cycle downtime produce only small (less than 1%) conservative changes in the calculated decay heat rates. For normal power distribution changes, the study also generally showed small effects on the decay heat rates. The most significant effect on decay heat rates was observed for history changes that produce the same total burnup but have different cycle burnups or specific power. Even for these changes the effect on decay heat was less than 1% except for cooling times less than seven years. Thus, short cooling time adjustment factors f_7 and f_7 were developed to adjust for differences between the cumulative average specific power and the specific power for the last two cycles.

The short cooling-time factors were derived and tested using 22 ORIGEN-S cases that modeled widely varying distributions of cycle burnup by changing the specific power. The resulting factor formulae for f_7 and f_7 are dependent, respectively, on the ratios P_N/P_{ave} and $P_{N-1}/P_{ave,N-1}$ where $P_{ave,N-1}$ is the average specific power through the N-1 cycle. The factors are developed to reduce the heat rate p_{tab} if the corresponding power ratio is less than one and increase p_{tab} if the corresponding ratio is greater than one. Restrictions limiting the range of acceptable power ratios are given in the guide.

FORMULATION OF THE SAFETY FACTOR

The final safety factors formulae included in the proposed guide revision is a function of cooling time, total burnup, and reactor type. The formulation is based on an extensive analysis of both random and systematic errors, computational model bias, procedural guide inaccuracies, and minor parameter variations that were not already taken into account. Reference 13 provides a complete analysis of the safety factor formulation developed for the proposed guide revision. Only a brief review is provided here.

Table 6 provides a summary of the error types that contribute to the safety factor included in the proposed guide for BWR assemblies. A similar table with slightly lower total values is provided in Ref. 13 for PWR assemblies. For both tables, the conventional type of quadratic propagation of standard deviations was applied to determine the final standard deviation of decay heat rates resulting from random data uncertainty. The random errors considered are the standard deviations in fission product yields, half-lives, and recoverable energies (Q-values). The procedural guide inaccuracies are a result of interpolation and adjustment factor errors. This error was determined from (1) evaluating the interpolation scheme against more accurate (and more complex) interpolation techniques and (2) testing performed during development of the adjustment factors.

The overriding systematic data error and computational bias is in the calculation of the burnup-dependent cross sections. All of the actinide and light-element activation products, plus a few important fission products, are strongly dependent on accurate cross-section preparation. To evaluate this error, an extensive study was performed to compare the SAS2H/ORIGEN-S heat rate results with the measured data[8-12] discussed earlier and one other improved computational model.[6,7] For the purpose of estimating the cross-section-dependent error, the conservative assumption was made that the entire difference between computed and measured decay heat rates is completely due to cross-section error. These two major comparisons were carefully evaluated to determine a reasonable upper limit value for the decay heat error due to uncertainties in the burnup-dependent cross sections.

It should be noted that the mathematical models in the ORIGEN-S code have been extensively evaluated via numerous national and international projects that compare the calculation of decay heat rates. These comparisons confirm that ORIGEN-S has a valid mathematical model and that, when coupled with available data libraries,[5] provides a very small bias (less than 0.5%) that is easily enveloped by the final safety factor formula.

COMPARISON WITH STANDARDS

During development of the current regulatory guide, extensive analysis was done to demonstrate[4] that the calculated decay heat rate from fission products and light elements compared well with the ANSI standard for Decay Heat Power in Light Water Reactors[14] (ANSI 5.1). The changes made to improve

Table 6. Summary of percentage safety factors for BWR

Type of error or safety factor	$20\ \frac{kW}{kgU}, 20\ \frac{MWd}{kgU}$				$20\ \frac{kW}{kgU}, 45\ \frac{MWd}{kgU}$			
	Cooling time, years				Cooling time, years			
	1	4	10	110	1	4	10	110
Random data uncertainty	1.4	0.8	0.6	0.2	1.4	0.8	0.6	0.2
Cross-section bias	1.8	3.0	2.9	8.2	5.1	6.4	5.7	11.9
Procedure inaccuracies	1.5	1.5	1.6	2.0	1.5	1.5	1.6	2.0
Contingency	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Sum of errors	5.7	6.3	6.1	11.4	9.0	9.7	8.9	15.1
Safety factor formula	6.4	6.5	6.8	11.2	10.2	10.3	10.6	15.0

SAS2H/ORIGEN-S and its libraries will only improve this comparison. However, in development of the proposed guide revision, this comparison effort was not duplicated explicitly. The major reason for not repeating this effort is that ANSI 5.1 computes only the ^{239}U and ^{239}Np contribution to the actinide heat rate. These two nuclides are significant contributors for very short cooling times (i.e., loss-of-coolant accident applications). However, several other actinides become considerably more important to the total actinide heat rate at cooling times greater than one year (the lower limit cooling time in the guide revision).

A draft document (from the International Organization for Standardization) of a standard on decay heat power[15] (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) was computed from the SAS2H/ORIGEN-S results obtained in this study. Using results from the BWR cases at one year decay, the SAS2H values of A(1) are 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 one year, A(1), in the proposed standard is 0.214. Although this value is not greatly different than the 0.193 maximum for the BWR, it appears to be conservative for the lower, more typical, burnups. Nevertheless, the inclusion of all the actinides into the proposed international standard is considered to be a commendable endeavor.

SUMMARY

Inherent difficulties arise in attempting to develop a regulatory guide for decay heat rates that has appropriate safety factors, is not excessively conservative, is easy to use, and applies to all reactor spent fuel assemblies. Work has been completed on a task to develop a revision to the current regulatory guide. The revision reduces the conservatism in the current guide by expanding the data base of decay heat rates to encompass a broader range of specific powers and burnups and adding a similar data base for BWR fuel. Ease of use has been addressed in the procedures and in development of a PC-based program that applies the revised guide procedures. The guide revision is currently being reviewed by the NRC staff and will be issued shortly for public comment.

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