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SPENT FUEL HEAT GENERATION IN AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION

A. INTRODUCTION

Paragraph (h)(1) of § 72.72, "Overall Requirements," of 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI), requires that the fuel cladding be protected against degradation and gross ruptures. It has been shown that, under certain environmental conditions, high storage temperatures can cause degradation and gross rupture of the fuel rods to occur very rapidly. It is necessary to know what storage temperatures are anticipated during the life of the storage installation and that these temperatures will not significantly degrade the cladding to a point of causing gross ruptures. The temperature environment in an ISFSI is a function of the heat generated by the stored fuel assemblies. Paragraph (a)(4) of § 72.75 requires that the spent fuel storage system be designed with a heat removal capability consistent with its importance to safety. This regulatory guide presents a method acceptable to the NRC staff for calculating conservative values of heat generation rates for use as design input for an independent spent fuel storage installation (ISFSI).

This regulatory guide contains no information collection requirements and therefore is not subject to the requirements of the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.).

B. DISCUSSION

A number of methods have been developed for calculating the long-term heat generation rates of fuel assemblies from light-water-cooled power reactors as a function of burnup and decay times. The NRC staff considers that the SAS2 control module and ORIGEN-S code of SCALE updated with ENDF/B-V cross section data represent an appropriate method for computing these values. Results obtained by the use of these codes are listed in Tables 1 through 7. This range of values is considered adequate to cover ISFSI design for the foreseeable future. Detailed calculations are shown in Appendices A and B. In addition, it has been shown that afterheat power data predicted by the SAS2/ORIGEN-S method are in good agreement with data predicted by the method of ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors,"* when the conservatism of the fission product neutron capture correction of the ANSI method is discounted.

C. REGULATORY POSITION

The SAS2 control module and ORIGEN-S code of the SCALE code system in conjunction with the standard ORIGEN-S data libraries updated with ENDF/B-V cross section data are acceptable for use in predicting afterheat power in uranium-fueled pressurized water reactor (PWR) fuel assemblies. The calculations performed in developing Tables 1 through 7 were not designed to be representative of the boiling water reactor (BWR) fuel assemblies, but the use of additional safety factors with the tabulated values of afterheat power will ensure conservatism in the application to BWR assemblies.** (See Regulatory Position 2, ORIGEN-S BWR Afterheat Calculations.)

This method is designed for predicting conservative values of afterheat power. The tabulations should be used only for uranium-fueled assemblies in a once-through fuel cycle. They are not applicable to assemblies using recycled fuel. In addition, they are not designed for use with more than one assembly at a time and the spatial distribution of afterheat power within an assembly is not considered.

In all prescriptions that follow, the term "2σ uncertainty" is not used in a strictly statistical sense but is

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

**See NUREG/CR-2397 (ORNL/CSD-90), "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," September 1982. Available from the National Technical Information Service, Springfield, Va. 22161.

taken to mean twice the uncertainty due to random errors plus an estimate of any nonconservative bias.

For the purpose of following the prescriptions, the operating history will be divided into three operating cycles of equal burnup. Note that the lengths of the three cycles are not necessarily equal.

1. ORIGEN-S PWR AFTERHEAT CALCULATIONS

The following method will yield predictions of afterheat power that are conservative:

a. If the third cycle of the operating history has at least 10% downtime in the last 30% of the cycle time and at least 20% downtime in the last 70% of the cycle time, use the tabulated values of afterheat power for a typical operating history. Here, the 10% and 20% downtimes mean 10% and 20% of the time for the entire last cycle. Otherwise, use the tabulated values for a conservative operating history. Both sets of values are contained in Tables 1 through 7.

b. Calculate the burnup of the assembly plus a 2σ uncertainty ($B_{max} = B + 2\sigma_B + \Delta B_{bias}$) and then compute the afterheat power at the tabulated cooling times by linear interpolation in burnup between the tabulated values.

c. Compute the interpolated enrichment E_I corresponding to the burnup B_{max} by linear interpolation between the initial enrichments given for the tabulated cases. If the interpolated enrichment E_I is greater than the actual enrichment E, increase the values of afterheat power from step b by 0.5% for each difference of 0.1 in the weight-percent ^{235}U enrichment.

d. Calculate the average specific power of the assembly plus a 2σ uncertainty (in MW/MTU) for the following time intervals: (1) the last 10 days of operation, (2) the last 30 days of operation, (3) the last 60 days of operation, (4) the time equivalent to the last one-half (in burnup) of the last operating cycle, (5) the last operating cycle, and (6) the entire operating history. When computing the average specific power for each interval, do not include any downtime in the calculation. For example, if the last 30 days include 5 days downtime, the average power is the burnup for the last 30 days divided by 25 days. In addition, compute a specific power equal to 88% of the maximum specific power (including a 2σ uncertainty) during the last half of the operating cycle. Then, find the maximum of these seven values. This value will be called $P_{max,avg}$, the maximum average specific power.

e. If $P_{max,avg}$ is greater than 37.5 MW/MTU, multiply the values of afterheat power from step c by the ratio ($P_{max,avg}/37.5$). If the assembly specific power (plus a 2σ uncertainty) immediately before discharge (P_{dis}) is greater than 37.5 MW/MTU, the afterheat power at discharge should be multiplied by the ratio ($P_{dis}/37.5$) instead of the ratio ($P_{max,avg}/37.5$). If the specific power for the assembly never exceeds 37.5 MW/MTU, go to step f.

f. After the corrections of step e, afterheat power for cooling times not in the table should be computed by interpolation that is logarithmic in afterheat power and linear in time.

g. Multiply the values of afterheat power from step f by a safety factor of (1) 1.15 at discharge or (2) 1.08 for cooling times greater than one day. For cooling times between discharge and one day, determine the safety factor by linear interpolation between 1.15 and 1.08.

h. If the fuel assembly can be shown to have an initial effective ^{59}Co content (see Appendix B) of no more than 142.96 g/MTU, including uncertainty, the calculation is finished. Otherwise, an additional safety factor of 1% of the afterheat power from step g should be added for every 10% excess of ^{59}Co over 142.96 g/MTU. The calculation is now complete. (To derive the effective amount of ^{59}Co to use for the afterheat power calculations, the total ^{59}Co content of an assembly zone is multiplied by the correction factors for that zone and the effective amounts from all three zones are summed. The 142.96 g/MTU of ^{59}Co include an allowance for 100% uncertainty in the initial amount of ^{59}Co of a Westinghouse 17x17 fuel assembly. When fuel assemblies other than Westinghouse 17x17 are used, the licensee should provide documentation of how the amounts used with the afterheat prescription were determined.)

2. ORIGEN-S BWR AFTERHEAT CALCULATIONS

In applying Tables 1 to 7 to BWR assemblies, follow steps a to h of Regulatory Position 1 with the understanding that the enrichment correction step 1.c should be made on the basis of the lowest enrichment and add the following step:

For cooling times of 30 years or less, multiply the final values of afterheat power obtained from the prescription for PWR assemblies by 1.15. This is in addition to the multiplication by 1.08 required for PWR assemblies. For cooling times between 30 and 110 years, determine this additional factor by linear interpolation between 1.15 and 1.5. Then multiply the value obtained for PWR assemblies by this factor.

3. COMBINATION OF ORIGEN-S LIGHT ELEMENTS AND ACTINIDES WITH FISSION PRODUCT AFTERHEATS FROM ANSI/ANS 5.1-1979

ANSI/ANS 5.1-1979 computes afterheat power for only ^{239}U and ^{239}Np heavy actinides. The following method is presented for combining values of fission product afterheat power from the standard with the tabulated values of actinide and light element afterheat power to predict conservative values:

a. Select the tabulated values in Tables 1 through 7 for either the typical or conservative operating history as prescribed in step a of Regulatory Position 1.

b. Calculate the burnup of the assembly plus a 2σ uncertainty ($B_{max} = B + 2\sigma_B + \Delta B_{bias}$) and then compute the light element plus actinide afterheat power at the tabulated cooling times by linear interpolation in burnup between the tabulated values.

c. Compute the interpolated enrichment E_I corresponding to the burnup B_{max} by linear interpolation between the initial enrichments given for the tabulated cases. If the interpolated enrichment E_I is greater than the actual enrichment E, increase the values of afterheat power from step b by 2% for each difference of 0.1 in the weight-percent ^{235}U enrichment.

d. Compute the maximum average specific power $P_{max,avg}$ as in step d of Regulatory Position 1.

e. If $P_{max,avg}$ is greater than 37.5 MW/MTU, multiply the values of afterheat power from step c by the ratio ($P_{max,avg}/37.5$). Otherwise, no correction is made.

f. After any corrections in step e, afterheat power for cooling times not in the table should be computed by interpolation that is logarithmic in afterheat power and linear in time.

g. Compute the total afterheat power by adding the values of light element and actinide afterheat power from step f to the fission product afterheat power (plus 2σ uncertainty) from the method in the standard. An increase in downtime during irradiation that is followed by further irradiation may lead to an increase in actinide afterheat power. Since exceptionally long shutdown times during the irradiation history can significantly increase the actinide afterheat power, the tabulated values of actinide afterheat power may not be used if the downtime used in the ANSI method calculations exceeds 8 months in any cycle. If the actual downtime exceeds 8 months in any cycle, the tabulated values of actinide afterheat power may still be used if the user limits the downtime for the ANSI method calculations to 8 months per cycle.

h. Multiply the values of total afterheat power from step g by a safety factor of 1.06.

i. If the initial effective ^{59}Co content of the fuel assembly, including uncertainty, exceeds 142.96 g/MTU, add an additional 1% to the values of afterheat power in step h for every 10% excess of ^{59}Co over 142.96 g/MTU. The calculation is now complete. (See step h of Regulatory Position 1.)

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plan for using this regulatory guide.

Except in those areas for which an applicant or licensee proposes an acceptable alternative method for calculating heat generation rates needed for design input for an ISFSI, the method presented in this guide will be used in the evaluation of the heat removal capability of the spent fuel storage system.

Table 1

AFTERHEAT POWER (W/MTU)
Burnup: 18 GWD/MTU; Enrichment: 2.50 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.2753E+03	1.2229E+05	2.0986E+06	2.2222E+06
10 d	5.2087E+02	3.9377E+03	6.7374E+04	7.1832E+04
30 d	4.5181E+02	4.7992E+02	3.9419E+04	4.0350E+04
60 d	3.7112E+02	3.6990E+02	2.6126E+04	2.6867E+04
90 d	3.0702E+02	3.3068E+02	2.0176E+04	2.0814E+04
120 d	2.5631E+02	2.9850E+02	1.6464E+04	1.7019E+04
180 d	1.8638E+02	2.4554E+02	1.1811E+04	1.2243E+04
1 yr	1.0715E+02	1.4694E+02	6.0846E+03	6.3387E+03
2 yr	8.1250E+01	8.4174E+01	2.9349E+03	3.1004E+03
5 yr	5.2769E+01	7.5924E+01	8.5211E+02	9.8081E+02
10 yr	2.6851E+01	8.8430E+01	5.0423E+02	6.1951E+02
20 yr	7.2189E+00	1.0518E+02	3.6922E+02	4.8162E+02
30 yr	2.0426E+00	1.1430E+02	2.8767E+02	4.0401E+02
40 yr	6.5354E-01	1.1880E+02	2.2584E+02	3.4529E+02
50 yr	2.7477E-01	1.2051E+02	1.7787E+02	2.9865E+02
60 yr	1.6721E-01	1.2056E+02	1.4034E+02	2.6107E+02
70 yr	1.3289E-01	1.1964E+02	1.1085E+02	2.3062E+02
80 yr	1.1863E-01	1.1817E+02	8.7608E+01	2.0590E+02
90 yr	1.1015E-01	1.1640E+02	6.9267E+01	1.8578E+02
100 yr	1.0356E-01	1.1448E+02	5.4780E+01	1.6937E+02
110 yr	9.7792E-02	1.1251E+02	4.3329E+01	1.5593E+02
		Conservative Irradiation History		
0 d	1.3598E+03	1.2221E+05	2.1079E+06	2.2315E+06
10 d	5.9970E+02	3.9310E+03	7.5784E+04	8.0315E+04
30 d	5.1945E+02	4.6877E+02	4.5240E+04	4.6228E+04
60 d	4.2426E+02	3.5915E+02	3.0087E+04	3.0870E+04
90 d	3.4816E+02	3.2113E+02	2.3101E+04	2.3771E+04
120 d	2.8784E+02	2.9006E+02	1.8694E+04	1.9272E+04
180 d	2.0467E+02	2.3892E+02	1.3176E+04	1.3620E+04
1 yr	1.1122E+02	1.4376E+02	6.5438E+03	6.7987E+03
2 yr	8.2635E+01	8.3268E+01	3.0955E+03	3.2614E+03
5 yr	5.3570E+01	7.5682E+01	8.7096E+02	1.0002E+03
10 yr	2.7248E+01	8.8280E+01	5.0729E+02	6.2282E+02
20 yr	7.3228E+00	1.0515E+02	3.7083E+02	4.8331E+02
30 yr	2.0706E+00	1.1435E+02	2.8889E+02	4.0531E+02
40 yr	6.6133E-01	1.1889E+02	2.2679E+02	3.4634E+02
50 yr	2.7713E-01	1.2063E+02	1.7861E+02	2.9952E+02
60 yr	1.6810E-01	1.2071E+02	1.4092E+02	2.6180E+02
70 yr	1.3336E-01	1.1980E+02	1.1131E+02	2.3124E+02
80 yr	1.1898E-01	1.1834E+02	8.7972E+01	2.0643E+02
90 yr	1.1045E-01	1.1658E+02	6.9555E+01	1.8624E+02
100 yr	1.0383E-01	1.1466E+02	5.5007E+01	1.6977E+02
110 yr	9.8050E-02	1.1269E+02	4.3509E+01	1.5630E+02

Table 2

AFTERHEAT POWER (W/MTU)
Burnup: 27 GWD/MTU; Enrichment: 2.50 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.3893E+03	1.3125E+05	2.0319E+06	2.1646E+06
10 d	5.9940E+02	5.1932E+03	7.1804E+04	7.7597E+04
30 d	5.2462E+02	1.3709E+03	4.2658E+04	4.4554E+04
60 d	4.3738E+02	1.1525E+03	2.8836E+04	3.0426E+04
90 d	3.6804E+02	1.0318E+03	2.2726E+04	2.4126E+04
120 d	3.1309E+02	9.2854E+02	1.8930E+04	2.0171E+04
180 d	2.3702E+02	7.5794E+02	1.4129E+04	1.5124E+04
1 yr	1.4914E+02	4.3885E+02	7.9167E+03	8.5047E+03
2 yr	1.1669E+02	2.3080E+02	4.0713E+03	4.4188E+03
5 yr	7.6091E+01	1.8435E+02	1.2922E+03	1.5527E+03
10 yr	3.8756E+01	1.9675E+02	7.5153E+02	9.8703E+02
20 yr	1.0431E+01	2.1201E+02	5.3722E+02	7.5966E+02
30 yr	2.9586E+00	2.1817E+02	4.1591E+02	6.3704E+02
40 yr	9.5263E-01	2.1910E+02	3.2548E+02	5.4553E+02
50 yr	4.0521E-01	2.1711E+02	2.5589E+02	4.7341E+02
60 yr	2.4938E-01	2.1355E+02	2.0172E+02	4.1552E+02
70 yr	1.9932E-01	2.0921E+02	1.5925E+02	3.6866E+02
80 yr	1.7827E-01	2.0455E+02	1.2584E+02	3.3056E+02
90 yr	1.6557E-01	1.9981E+02	9.9491E+01	2.9947E+02
100 yr	1.5564E-01	1.9514E+02	7.8686E+01	2.7398E+02
110 yr	1.4693E-01	1.9060E+02	6.2244E+01	2.5299E+02
		Conservative Irradiation History		
0 d	1.4893E+03	1.3115E+05	2.0408E+06	2.1734E+06
10 d	6.9227E+02	5.1699E+03	7.9376E+04	8.5238E+04
30 d	6.0370E+02	1.3501E+03	4.8510E+04	5.0464E+04
60 d	4.9885E+02	1.1339E+03	3.3153E+04	3.4786E+04
90 d	4.1537E+02	1.0153E+03	2.6048E+04	2.7479E+04
120 d	3.4934E+02	9.1386E+02	2.1546E+04	2.2809E+04
180 d	2.5836E+02	7.4628E+02	1.5843E+04	1.6847E+04
1 yr	1.5477E+02	4.3286E+02	8.6204E+03	9.2080E+03
2 yr	1.1919E+02	2.2856E+02	4.3490E+03	4.6968E+03
5 yr	7.7570E+01	1.8319E+02	1.3273E+03	1.5881E+03
10 yr	3.9489E+01	1.9570E+02	7.5660E+02	9.9179E+02
20 yr	1.0623E+01	2.1113E+02	5.3938E+02	7.6113E+02
30 yr	3.0099E+00	2.1740E+02	4.1752E+02	6.3793E+02
40 yr	9.6655E-01	2.1842E+02	3.2672E+02	5.4611E+02
50 yr	4.0913E-01	2.1650E+02	2.5686E+02	4.7377E+02
60 yr	2.5060E-01	2.1300E+02	2.0248E+02	4.1572E+02
70 yr	1.9980E-01	2.0870E+02	1.5985E+02	3.6875E+02
80 yr	1.7854E-01	2.0408E+02	1.2631E+02	3.3057E+02
90 yr	1.6578E-01	1.9938E+02	9.9865E+01	2.9941E+02
100 yr	1.5582E-01	1.9474E+02	7.8981E+01	2.7387E+02
110 yr	1.4710E-01	1.9023E+02	6.2477E+01	2.5286E+02

Table 3

AFTERHEAT POWER (W/MTU)
Burnup: 33 GWD/MTU; Enrichment: 3.30 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.2991E+03	1.2453E+05	2.0467E+06	2.1725E+06
10 d	5.8755E+02	5.6055E+03	7.4364E+04	8.0557E+04
30 d	5.1581E+02	1.8079E+03	4.4528E+04	4.6852E+04
60 d	4.3205E+02	1.5297E+03	3.0305E+04	3.2266E+04
90 d	3.6540E+02	1.3711E+03	2.4013E+04	2.5750E+04
120 d	3.1252E+02	1.2351E+03	2.0100E+04	2.1648E+04
180 d	2.3919E+02	1.0105E+03	1.5139E+04	1.6388E+04
1 yr	1.5390E+02	5.9003E+02	8.6624E+03	9.4064E+03
2 yr	1.2126E+02	3.1492E+02	4.5816E+03	5.0178E+03
5 yr	7.9164E+01	2.4980E+02	1.5649E+03	1.8939E+03
10 yr	4.0343E+01	2.6046E+02	9.3263E+02	1.2334E+03
20 yr	1.0858E+01	2.7238E+02	6.6510E+02	9.4834E+02
30 yr	3.0753E+00	2.7519E+02	5.1421E+02	7.9248E+02
40 yr	9.8614E-01	2.7295E+02	4.0208E+02	6.7602E+02
50 yr	4.1633E-01	2.6802E+02	3.1597E+02	5.8440E+02
60 yr	2.5443E-01	2.6178E+02	2.4899E+02	5.1103E+02
70 yr	2.0266E-01	2.5500E+02	1.9654E+02	4.5174E+02
80 yr	1.8109E-01	2.4812E+02	1.5528E+02	4.0358E+02
90 yr	1.6820E-01	2.4136E+02	1.2275E+02	3.6428E+02
100 yr	1.5816E-01	2.3483E+02	9.7075E+01	3.3207E+02
110 yr	1.4937E-01	2.2860E+02	7.6784E+01	3.0553E+02
		Conservative Irradiation History		
0 d	1.3952E+03	1.2442E+05	2.0551E+06	2.1809E+06
10 d	6.7672E+02	5.5797E+03	8.1765E+04	8.8022E+04
30 d	5.9148E+02	1.7855E+03	5.0481E+04	5.2858E+04
60 d	4.9068E+02	1.5097E+03	3.4832E+04	3.6832E+04
90 d	4.1053E+02	1.3533E+03	2.7562E+04	2.9326E+04
120 d	3.4719E+02	1.2193E+03	2.2939E+04	2.4506E+04
180 d	2.5992E+02	9.9780E+02	1.7058E+04	1.8315E+04
1 yr	1.6004E+02	5.8329E+02	9.5116E+03	1.0255E+04
2 yr	1.2434E+02	3.1212E+02	4.9298E+03	5.3662E+03
5 yr	8.1007E+01	2.4814E+02	1.6113E+03	1.9404E+03
10 yr	4.1259E+01	2.5896E+02	9.3998E+02	1.2402E+03
20 yr	1.1097E+01	2.7110E+02	6.6823E+02	9.5042E+02
30 yr	3.1393E+00	2.7406E+02	5.1652E+02	7.9372E+02
40 yr	1.0035E+00	2.7193E+02	4.0387E+02	6.7680E+02
50 yr	4.2118E-01	2.6710E+02	3.1736E+02	5.8488E+02
60 yr	2.5591E-01	2.6093E+02	2.5009E+02	5.1128E+02
70 yr	2.0323E-01	2.5422E+02	1.9740E+02	4.5182E+02
80 yr	1.8140E-01	2.4740E+02	1.5596E+02	4.0354E+02
90 yr	1.6842E-01	2.4069E+02	1.2329E+02	3.6415E+02
100 yr	1.5835E-01	2.3421E+02	9.7499E+01	3.3187E+02
110 yr	1.4954E-01	2.2802E+02	7.7120E+01	3.0529E+02

Table 4

AFTERHEAT POWER (W/MTU)
Burnup: 40 GWD/MTU; Enrichment: 4.00 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.2647E+03	1.2222E+05	2.0468E+06	2.1703E+06
10 d	5.9546E+02	6.3431E+03	7.6679E+04	8.3617E+04
30 d	5.2432E+02	2.4525E+03	4.6294E+04	4.9271E+04
60 d	4.4115E+02	2.0946E+03	3.1734E+04	3.4269E+04
90 d	3.7491E+02	1.8809E+03	2.5290E+04	2.7546E+04
120 d	3.2231E+02	1.6972E+03	2.1281E+04	2.3301E+04
180 d	2.4929E+02	1.3935E+03	1.6188E+04	1.7831E+04
1 yr	1.6384E+02	8.2458E+02	9.4816E+03	1.0470E+04
2 yr	1.2999E+02	4.5079E+02	5.1672E+03	5.7480E+03
5 yr	8.4979E+01	3.5635E+02	1.8823E+03	2.3237E+03
10 yr	4.3331E+01	3.6161E+02	1.1372E+03	1.5421E+03
20 yr	1.1665E+01	3.6441E+02	8.0706E+02	1.1831E+03
30 yr	3.3028E+00	3.5951E+02	6.2280E+02	9.8562E+02
40 yr	1.0578E+00	3.5075E+02	4.8649E+02	8.3829E+02
50 yr	4.4561E-01	3.4027E+02	3.8207E+02	7.2278E+02
60 yr	2.7178E-01	3.2927E+02	3.0098E+02	6.3052E+02
70 yr	2.1629E-01	3.1836E+02	2.3752E+02	5.5610E+02
80 yr	1.9324E-01	3.0785E+02	1.8764E+02	4.9568E+02
90 yr	1.7950E-01	2.9787E+02	1.4832E+02	4.4638E+02
100 yr	1.6883E-01	2.8847E+02	1.1729E+02	4.0593E+02
110 yr	1.5949E-01	2.7964E+02	9.2768E+01	3.7256E+02
		Conservative Irradiation History		
0 d	1.3571E+03	1.2210E+05	2.0548E+06	2.1783E+06
10 d	6.8105E+02	6.3147E+03	8.3875E+04	9.0871E+04
30 d	5.9675E+02	2.4283E+03	5.2246E+04	5.5271E+04
60 d	4.9720E+02	2.0730E+03	3.6372E+04	3.8943E+04
90 d	4.1815E+02	1.8617E+03	2.8994E+04	3.1274E+04
120 d	3.5575E+02	1.6799E+03	2.4296E+04	2.6332E+04
180 d	2.6975E+02	1.3796E+03	1.8297E+04	1.9946E+04
1 yr	1.7076E+02	8.1692E+02	1.0489E+04	1.1476E+04
2 yr	1.3388E+02	4.4727E+02	5.5969E+03	6.1780E+03
5 yr	8.7338E+01	3.5405E+02	1.9436E+03	2.3850E+03
10 yr	4.4506E+01	3.5949E+02	1.1478E+03	1.5518E+03
20 yr	1.1973E+01	3.6256E+02	8.1155E+02	1.1861E+03
30 yr	3.3849E+00	3.5785E+02	6.2611E+02	9.8734E+02
40 yr	1.0801E+00	3.4923E+02	4.8904E+02	8.3935E+02
50 yr	4.5181E-01	3.3888E+02	3.8406E+02	7.2339E+02
60 yr	2.7364E-01	3.2798E+02	3.0254E+02	6.3080E+02
70 yr	2.1698E-01	3.1717E+02	2.3875E+02	5.5613E+02
80 yr	1.9360E-01	3.0674E+02	1.8861E+02	4.9554E+02
90 yr	1.7976E-01	2.9684E+02	1.4909E+02	4.4611E+02
100 yr	1.6905E-01	2.8751E+02	1.1789E+02	4.0557E+02
110 yr	1.5969E-01	2.7874E+02	9.3246E+01	3.7214E+02

Table 5

AFTERHEAT POWER (W/MTU)
Burnup: 46 GWD/MTU; Enrichment: 4.00 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.3344E+03	1.2855E+05	2.0187E+06	2.1486E+06
10 d	6.3834E+02	7.4332E+03	7.8277E+04	8.6349E+04
30 d	5.6348E+02	3.3081E+03	4.7552E+04	5.1423E+04
60 d	4.7588E+02	2.8615E+03	3.2813E+04	3.6151E+04
90 d	4.0612E+02	2.5765E+03	2.6310E+04	2.9293E+04
120 d	3.5073E+02	2.3302E+03	2.2272E+04	2.4953E+04
180 d	2.7380E+02	1.9229E+03	1.7136E+04	1.9333E+04
1 yr	1.8333E+02	1.1590E+03	1.0287E+04	1.1630E+04
2 yr	1.4637E+02	6.5429E+02	5.7420E+03	6.5427E+03
5 yr	9.5810E+01	5.1602E+02	2.1589E+03	2.7708E+03
10 yr	4.8875E+01	5.0641E+02	1.2944E+03	1.8497E+03
20 yr	1.3167E+01	4.8595E+02	9.1049E+02	1.4096E+03
30 yr	3.7348E+00	4.6368E+02	7.0103E+02	1.1684E+03
40 yr	1.2019E+00	4.4177E+02	5.4697E+02	9.8994E+02
50 yr	5.1081E-01	4.2120E+02	4.2930E+02	8.5101E+02
60 yr	3.1419E-01	4.0229E+02	3.3808E+02	7.4068E+02
70 yr	2.5112E-01	3.8506E+02	2.6676E+02	6.5207E+02
80 yr	2.2466E-01	3.6940E+02	2.1072E+02	5.8035E+02
90 yr	2.0874E-01	3.5514E+02	1.6657E+02	5.2191E+02
100 yr	1.9630E-01	3.4209E+02	1.3172E+02	4.7401E+02
110 yr	1.8540E-01	3.3010E+02	1.0418E+02	4.3447E+02
		Conservative Irradiation History		
0 d	1.4275E+03	1.2840E+05	2.0265E+06	2.1563E+06
10 d	7.2467E+02	7.4046E+03	8.5255E+04	9.3384E+04
30 d	6.3647E+02	3.2844E+03	5.3416E+04	5.7337E+04
60 d	5.3242E+02	2.8403E+03	3.7466E+04	4.0839E+04
90 d	4.4993E+02	2.5575E+03	3.0086E+04	3.3094E+04
120 d	3.8485E+02	2.3132E+03	2.5396E+04	2.8094E+04
180 d	2.9518E+02	1.9091E+03	1.9390E+04	2.1595E+04
1 yr	1.9144E+02	1.1512E+03	1.1436E+04	1.2779E+04
2 yr	1.5132E+02	6.5044E+02	6.2504E+03	7.0521E+03
5 yr	9.8845E+01	5.1331E+02	2.2362E+03	2.8484E+03
10 yr	5.0389E+01	5.0385E+02	1.3085E+03	1.8627E+03
20 yr	1.3564E+01	4.8359E+02	9.1632E+02	1.4135E+03
30 yr	3.8407E+00	4.6149E+02	7.0530E+02	1.1706E+03
40 yr	1.2306E+00	4.3973E+02	5.5027E+02	9.9122E+02
50 yr	5.1880E-01	4.1928E+02	4.3187E+02	8.5167E+02
60 yr	3.1661E-01	4.0049E+02	3.4009E+02	7.4090E+02
70 yr	2.5201E-01	3.8338E+02	2.6834E+02	6.5198E+02
80 yr	2.2513E-01	3.6783E+02	2.1197E+02	5.8002E+02
90 yr	2.0908E-01	3.5366E+02	1.6755E+02	5.2143E+02
100 yr	1.9659E-01	3.4071E+02	1.3250E+02	4.7341E+02
110 yr	1.8566E-01	3.2881E+02	1.0480E+02	4.3380E+02

Table 6

AFTERHEAT POWER (W/MTU)
Burnup: 50 GWD/MTU; Enrichment: 4.50 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.3046E+03	1.2650E+05	2.0240E+06	2.1518E+06
10 d	6.3673E+02	7.8363E+03	7.9413E+04	8.7886E+04
30 d	5.6251E+02	3.6663E+03	4.8418E+04	5.2647E+04
30 d	4.7558E+02	3.1760E+03	3.3502E+04	3.7154E+04
90 d	4.0634E+02	2.8623E+03	2.6912E+04	3.0181E+04
120 d	3.5137E+02	2.5912E+03	2.2818E+04	2.5760E+04
180 d	2.7502E+02	2.1429E+03	1.7607E+04	2.0025E+04
1 yr	1.8511E+02	1.3019E+03	1.0655E+04	1.2143E+04
2 yr	1.4804E+02	7.4568E+02	6.0272E+03	6.9209E+03
5 yr	9.6955E+01	5.9108E+02	2.3358E+03	3.0238E+03
10 yr	4.9473E+01	5.7698E+02	1.4131E+03	2.0395E+03
20 yr	1.3330E+01	5.4904E+02	9.9324E+02	1.5556E+03
30 yr	3.7792E+00	5.2067E+02	7.6433E+02	1.2888E+03
40 yr	1.2148E+00	4.9371E+02	5.9617E+02	1.0911E+03
50 yr	5.1518E-01	4.6892E+02	4.6782E+02	9.3726E+02
60 yr	3.1626E-01	4.4642E+02	3.6837E+02	8.1511E+02
70 yr	2.5254E-01	4.2611E+02	2.9063E+02	7.1699E+02
80 yr	2.2589E-01	4.0775E+02	2.2957E+02	6.3755E+02
90 yr	2.0990E-01	3.9111E+02	1.8145E+02	5.7277E+02
100 yr	1.9743E-01	3.7593E+02	1.4349E+02	5.1962E+02
110 yr	1.8650E-01	3.6203E+02	1.1349E+02	4.7570E+02
		Conservative Irradiation History		
0 d	1.3934E+03	1.2634E+05	2.0316E+06	2.1594E+06
10 d	7.1900E+02	7.8009E+03	8.6295E+04	9.4815E+04
30 d	6.3204E+02	3.6365E+03	5.4247E+04	5.8516E+04
60 d	5.2951E+02	3.1495E+03	3.8170E+04	4.1849E+04
90 d	4.4828E+02	2.8386E+03	3.0731E+04	3.4018E+04
120 d	3.8421E+02	2.5700E+03	2.6002E+04	2.8956E+04
180 d	2.9593E+02	2.1257E+03	1.9939E+04	2.2361E+04
1 yr	1.9359E+02	1.2923E+03	1.1878E+04	1.3364E+04
2 yr	1.5342E+02	7.4113E+02	6.5761E+03	7.4706E+03
5 yr	1.0026E+02	5.8797E+02	2.4221E+03	3.1103E+03
10 yr	5.1125E+01	5.7402E+02	1.4295E+03	2.0546E+03
20 yr	1.3762E+01	5.4632E+02	1.0001E+03	1.5602E+03
30 yr	3.8950E+00	5.1813E+02	7.6934E+02	1.2914E+03
40 yr	1.2461E+00	4.9134E+02	6.0003E+02	1.0926E+03
50 yr	5.2389E-01	4.6669E+02	4.7083E+02	9.3805E+02
60 yr	3.1888E-01	4.4433E+02	3.7073E+02	8.1538E+02
70 yr	2.5350E-01	4.2415E+02	2.9249E+02	7.1689E+02
80 yr	2.2639E-01	4.0591E+02	2.3103E+02	6.3717E+02
90 yr	2.1026E-01	3.8938E+02	1.8261E+02	5.7221E+02
100 yr	1.9773E-01	3.7432E+02	1.4440E+02	5.1892E+02
110 yr	1.8677E-01	3.6052E+02	1.1421E+02	4.7492E+02

Table 7

AFTERHEAT POWER (W/MTU)
Burnup: 55 GWD/MTU; Enrichment: 4.50 wt-% ^{235}U

Cooling Time	Light Elements	Actinides	Fission Products	Total
		Typical Irradiation History		
0 d	1.3609E+03	1.3171E+05	2.0036E+06	2.1366E+06
10 d	6.7109E+02	8.7636E+03	8.0673E+04	9.0108E+04
30 d	5.9363E+02	4.4117E+03	4.9399E+04	5.4405E+04
60 d	5.0289E+02	3.8505E+03	3.4332E+04	3.8686E+04
90 d	4.3065E+02	3.4788E+03	2.7689E+04	3.1599E+04
120 d	3.7333E+02	3.1569E+03	2.3570E+04	2.7100E+04
180 d	2.9372E+02	2.6243E+03	1.8326E+04	2.1244E+04
1 yr	1.9976E+02	1.6243E+03	1.1274E+04	1.3098E+04
2 yr	1.6032E+02	9.6013E+02	6.4794E+03	7.5999E+03
5 yr	1.0508E+02	7.6502E+02	2.5600E+03	3.4301E+03
10 yr	5.3635E+01	7.3170E+02	1.5395E+03	2.3249E+03
20 yr	1.4459E+01	6.7405E+02	1.0757E+03	1.7642E+03
30 yr	4.1058E+00	6.2407E+02	8.2669E+02	1.4549E+03
40 yr	1.3251E+00	5.8121E+02	6.4438E+02	1.2269E+03
50 yr	5.6610E-01	5.4451E+02	5.0547E+02	1.0505E+03
60 yr	3.4996E-01	5.1295E+02	3.9794E+02	9.1124E+02
70 yr	2.8042E-01	4.8558E+02	3.1394E+02	7.9980E+02
80 yr	2.5110E-01	4.6162E+02	2.4797E+02	7.0984E+02
90 yr	2.3337E-01	4.4042E+02	1.9600E+02	6.3665E+02
100 yr	2.1948E-01	4.2146E+02	1.5499E+02	5.7667E+02
110 yr	2.0729E-01	4.0434E+02	1.2259E+02	5.2714E+02
		Conservative Irradiation History		
0 d	1.4490E+03	1.3152E+05	2.0111E+06	2.1440E+06
10 d	7.5283E+02	8.7218E+03	8.7391E+04	9.6866E+04
30 d	6.6275E+02	4.3765E+03	5.5141E+04	6.0181E+04
60 d	5.5665E+02	3.8192E+03	3.8987E+04	4.3363E+04
90 d	4.7268E+02	3.4509E+03	3.1545E+04	3.5468E+04
120 d	4.0651E+02	3.1319E+03	2.6823E+04	3.0362E+04
180 d	3.1533E+02	2.6042E+03	2.0762E+04	2.3682E+04
1 yr	2.0928E+02	1.6134E+03	1.2604E+04	1.4427E+04
2 yr	1.6665E+02	9.5529E+02	7.0915E+03	8.2134E+03
5 yr	1.0900E+02	7.6182E+02	2.6608E+03	3.5317E+03
10 yr	5.5594E+01	7.2856E+02	1.5593E+03	2.3434E+03
20 yr	1.4973E+01	6.7101E+02	1.0839E+03	1.7699E+03
30 yr	4.2431E+00	6.2113E+02	8.3264E+02	1.4580E+03
40 yr	1.3623E+00	5.7839E+02	6.4897E+02	1.2287E+03
50 yr	5.7645E-01	5.4182E+02	5.0905E+02	1.0515E+03
60 yr	3.5308E-01	5.1040E+02	4.0075E+02	9.1150E+02
70 yr	2.8158E-01	4.8317E+02	3.1615E+02	7.9960E+02
80 yr	2.5170E-01	4.5935E+02	2.4971E+02	7.0931E+02
90 yr	2.3380E-01	4.3828E+02	1.9738E+02	6.3589E+02
100 yr	2.1985E-01	4.1945E+02	1.5608E+02	5.7575E+02
110 yr	2.0762E-01	4.0245E+02	1.2345E+02	5.2612E+02

APPENDIX A

SAMPLE PROBLEMS USING TABULATED AFTERHEAT POWER DATA

1. SAMPLE PROBLEM 1

The problem is to predict a conservative value of afterheat power given the following:

a. The cooling time for a PWR assembly was 3 years (for operating history see Table A-1).

b. The initial fuel loading was 0.475 MTU.

c. The initial ^{235}U enrichment was 3.2 wt-%.

d. The specific power in each time interval has a 1σ statistical uncertainty of 0.4 MW/MTU.

e. The 1σ statistical uncertainty for each time interval is 0.05 day.

f. The maximum nonconservative bias for any specific power is estimated to be 0.25 MW/MTU. (The use of constant uncertainties is just one example; uncertainties could vary with interval, or they could be expressed as a constant percentage of the values for each interval.)

g. The initial ^{59}Co concentration is $\leq$ 142.96 g/MTU per assembly.

The steps in following the prescription of Regulatory Positions 1, 2, and 3 are outlined below.

Step 1. The total time for cycle 3 from Table A-1 is 452.299 days (including downtime). There are 81 days (17.9% of cycle time) downtime in the last 70% (316.6 days) of cycle 3. There are 37 days (8.2% of cycle time) downtime in the last 30% (135.7 days) of cycle 3. Therefore, the tabulated values of afterheat power for a conservative operating history must be used.

Step 2. The burnup of the assembly is 32081.2 MWD/MTU. The 1σ statistical uncertainty of the burnup can be found (by standard propagation of error) from

$$\sigma_B = \left[\sum_{i=1}^{N} \left(P_i^2 \sigma_{t_i}^2 + t_i^2 \sigma_{P_i}^2\right)\right]^{1/2} \qquad \text{(A-1)}$$

where N = number of time intervals
t_i = length of interval i
P_i = specific power during interval i
σ_{t_i} = standard deviation of t_i
σ_{P_i} = standard deviation of P_i

It was assumed that σ_{P_i} was zero for all downtime (zero power) intervals. For this example, σ_B is 75.5 MWD/MTU. The bias in the burnup is 0.25 MW/MTU x 941 days irradiation time, or ΔB_{bias} = 235.25 MWD/MTU. The maximum possible burnup is given by:

$$B_{max} = B + 2\sigma_B + \Delta B_{bias} \qquad \text{(A-2)}$$

For this example, B_{max} = 32467.4 MWD/MTU.

Since B_{max} lies between the tabulated values of 27 and 33 GWD/MTU, the afterheat power for this case is found from:

$$A = A_{27} + (A_{33} - A_{27})\left[\frac{B_{max} - 27}{33 - 27}\right] \qquad \text{(A-3)}$$

where A_{27} = afterheat power for B = 27 GWD/MTU
A_{33} = afterheat power for B = 33 GWD/MTU
B_{max} = burnup (GWD/MTU)

The interpolated values of afterheat power at cooling times of 2 and 5 years are shown in the second column of Table A-2.

Step 3. The interpolated enrichment corresponding to B_{max} is found from:

$$E_I = E_{27} + (E_{33} - E_{27})\left[\frac{B_{max} - 27}{33 - 27}\right] \qquad \text{(A-4)}$$

where E_{27} = enrichment for 27 GWD/MTU case (2.5 wt-%)
E_{33} = enrichment for 33 GWD/MTU case (3.3 wt-%)

For this example, E_I = 3.2290 wt-%. Since E_I > 3.2 wt-%, a correction factor is needed. Its value is:

$$C = 1 + 0.01\left[\frac{0.5}{0.1}\right](3.2290 - 3.2) = 1.00145$$

The values of afterheat power at cooling times of 2 and 5 years after multiplication by C are shown in column 3 of Table A-2.

Step 4. The average specific power for a time period t is given by:

$$\bar{P} = \sum_{i=1}^{N} P_i t_i / t \qquad \text{(A-5)}$$

The standard deviation of $\overline{P}$ is given by:

$$\sigma_{\overline{P}} = \left\{ \sum_{i=1}^{N} \left[\left(\frac{t_i}{t}\right)^2 \sigma_{P_i}^2 + \left(\frac{P_i - \overline{P}}{t}\right)^2 \sigma_{t_i}^2 \right] \right\}^{1/2} \qquad \text{(A-6)}$$

where $$t = \sum_{i=1}^{N} t_i \qquad \text{(A-7)}$$

and all other symbols were defined earlier. The summations of Equations A-5, A-6, and A-7 do not include downtime (zero power) intervals.

The maximum power for a time period is:

$$P_m = \overline{P} + 2\sigma_{\overline{P}} + \Delta P_{bias} \qquad \text{(A-8)}$$

The six values of P_m needed to calculate $P_{max,avg}$ are shown in Table A-3. $\overline{P}$ is the maximum value for $\overline{P}$ in the last half of cycle 3. The maximum specific power during the last half of cycle 3 is:

$$P_{max} = 30.9 + 2(0.4) + 0.25 = 31.95 \text{ MW/MTU}$$

Since 88% of P_{max} is 28.112 MW/MTU, the value of $P_{max,avg}$ is 35.935 MW/MTU.

Step 5. No correction is needed because $P_{max,avg} < 37.5$.

Step 6. The value of afterheat power at 3 years cooling time from log-linear interpolation is 3.7798×10^3 W/MTU.

Step 7. The afterheat power, after multiplication by a safety factor of 1.08, is 4.0822×10^3 W/MTU.

Step 8. Since the initial ^{59}Co content (43 g/0.475 MTU = 90.53 g/MTU) is less than 142.96 g/MTU, no further correction is needed. The total afterheat power for the assembly is 1939 W (4082.2 W/MTU x 0.475 MTU) at 3 years cooling time.

2. SAMPLE PROBLEM 2

The problem is to predict a conservative value of afterheat power at 180 days cooling time for the PWR assembly whose operating history is given in Table A-4. It is known that:

a. The initial fuel loading was 0.395 MTU,

b. The initial ^{235}U enrichment was 3.6 wt-%,

c. The initial effective amount of ^{59}Co (including uncertainty) is 58 g per assembly.

The standard deviation of the specific power is 0.32 MW/MTU for any time interval, and the maximum nonconservative bias (ΔP_{bias}) is 0.31 MW/MTU. The standard deviation of the time for any interval is 0.025 days.

The steps in the calculation are:

Step 1. The last 70% of cycle 3 (287.1 days) has 88 days downtime (21.5% of cycle time). The last 30% of cycle 3 (123.0 days) has 42 days downtime (10.2% of cycle time). The afterheat power data tabulations for a typical operating history can be used.

Step 2. The burnup is 35111.5 MWD/MTU. The standard deviation and bias for this burnup are 6.221 and 302.56 MWD/MTU respectively. For this example, B_{max} = 35538.5 MWD/MTU. The interpolated afterheat power at 180 days cooling time is:

$$\begin{aligned} A &= 16388 + (17831 - 16388)(35.5385 - 33)/(40 - 33) \\ &= 16911 \text{ W/MTU} \end{aligned}$$

Step 3. The interpolated enrichment is:

$$\begin{aligned} E_I &= 3.3 + (4.0 - 3.3)(35.5385 - 33)/(40 - 33) \\ &= 3.5539 \text{ wt-\%} \end{aligned}$$

Since $E_I < 3.6$ wt-% (the actual enrichment), no correction is needed.

Step 4. The values of P_m necessary to compute $P_{max,avg}$ are shown in Table A-5. The value of 0.88 P_{max} is given by:

$$0.88(37.2 + 2 \times 0.32 + 0.31) = 33.572 \text{ MW/MTU}$$

It is seen that $P_{max,avg} = 38.150$.

Step 5. A correction factor of 38.15/37.5 is used, and the corrected afterheat power becomes 17204 W/MTU.

Step 6. There is no need to interpolate for cooling time.

Step 7. Using a safety factor of 1.08, the corrected afterheat power becomes 18580 W/MTU.

Step 8. The initial effective ^{59}Co content is 146.84 g/MTU (58 g/0.395 MTU). The percent excess ^{59}Co is (146.84/142.96 – 1) x 100, or 2.78%. The final correction factor is:

$$1 + 0.005 \left(\frac{2.78}{10}\right) = 1.0014$$

The final afterheat power is 18606 W/MTU, and the total afterheat power for the assembly (0.395 MTU) is 7349 W.

Table A-1

ASSEMBLY OPERATING HISTORY FOR EXAMPLE 1

Interval	Δt (days)	$\Sigma \Delta t$ (days)	P (MW/MTU)	$P\Delta t$ (MWD/MTU)	$\Sigma P\Delta t$ (MWD/MTU)
1	14	14	32.1	449.4	449.4
2	16	30	37.2	595.2	1044.6
3	13	43	37.1	482.3	1526.9
4	2	45	0.0	0.0	1526.9
5	32	77	37.2	1190.4	2717.3
6	24	101	37.1	890.4	3607.7
7	11	112	37.0	407.0	4014.7
8	2	114	0.0	0.0	4014.7
9	47	161	37.0	1739.0	5753.7
10	51	212	37.1	1892.1	7645.8
11	41	253	36.9	1512.9	9158.7
12	3	256	0.0	0.0	9158.7
13	18	274	36.8	662.4	9821.1
14	2	276	0.0	0.0	9821.1
15a	23.842	299.842	36.6	872.6	10693.7
15b	7.158	307	36.6	262.0	10955.7
16	31	338	36.4	1128.4	12084.1
17	27	365	0.0	0.0	12084.1
18	29	394	35.2	1020.8	13104.9
19	47	441	35.7	1677.9	14782.8
20	35	476	36.1	1263.5	16046.3
21	2	478	0.0	0.0	16046.3
22	25	503	34.7	867.5	16913.8
23	2	505	0.0	0.0	16913.8
24	40	545	34.2	1368.0	18281.8
25	51	596	33.6	1713.6	19995.4
26	4	600	0.0	0.0	19995.4
27c	42.701	642.701	32.6	1392.1	21387.5
27d	0.299	643	32.6	9.7	21397.2
28	58	701	32.8	1902.4	23299.6
29	29	730	0.0	0.0	23299.6
30	25	755	31.2	780.0	24079.6
31	40	795	32.2	1288.0	25367.6
32	5	800	0.0	0.0	25367.6
33	22	822	31.3	688.6	26056.2
34	3	825	0.0	0.0	26056.2
35	15	840	31.1	466.5	26522.7
36	9	849	0.0	0.0	26522.7
37e	6.849	855.849	30.9	211.6	26734.3
37f	33.151	889	30.9	1024.4	27758.7
38	33	922	30.7	1013.1	28771.8
39	27	949	0.0	0.0	28771.8
40	37	986	30.6	1132.2	29904.0
41	26	1012	0.0	0.0	29904.0
42	43	1055	30.4	1307.2	31211.2
43	11	1066	0.0	0.0	31211.2
44	29	1095	30.0	870.0	32081.2

a Last period of cycle 1.
b First period of cycle 2.
c Last period of cycle 2.
d First period of cycle 3.
e Last period of first half of cycle 3.
f First period of last half of cycle 3.

Table A-2

AFTERHEAT POWER AT VARIOUS STEPS IN THE CALCULATION

Cooling Time (Y)	Afterheat Power (W/MTU) After Step 2	After Step 3	After Step 5	After Step 6	After Step 7
2	5.3068E+3	5.3145E+3	5.3145E+3	–	–
3	–	–	–	3.7798E+3	4.0822E+3
5	1.9091E+3	1.9119E+3	1.9119E+3	–	–

Table A-3

SPECIFIC POWER VALUES FOR CALCULATING $P_{max,avg}$

Time Interval	$\overline{P}$ (MW/MTU)	$\sigma_{\overline{P}}$ (MW/MTU)	ΔP_{bias} (MW/MTU)	P_m (MW/MTU)
Last 10 days	30	0.4	0.25	31.05
Last 30 days	30	0.4	0.25	31.05
Last 60 days	30.163	0.288	0.25	30.989
Last half cycle	30.527	0.181	0.25	31.139
Last cycle	31.241	0.133	0.25	31.757
All 3 cycles	34.093	0.796	0.25	35.935

Table A-4

ASSEMBLY OPERATING HISTORY FOR EXAMPLE 2

Interval	Δt (days)	ΣΔt (days)	P (MW/MTU)	PΔt (MWD/MTU)	ΣPΔt (MWD/MTU)
1	14	14	32.1	449.4	449.4
2	16	30	37.2	595.2	1044.6
3	13	43	37.1	482.3	1526.9
4	2	45	0.0	0.0	1526.9
5	32	77	37.2	1190.4	2717.3
6	24	101	37.1	890.4	3607.7
7	11	112	37.0	407.0	4014.7
8	2	114	0.0	0.0	4014.7
9	47	161	37.0	1739.0	5753.7
10	51	212	37.1	1892.1	7645.8
11	41	253	36.9	1512.9	9158.7
12	3	256	0.0	0.0	9158.7
13	18	274	36.8	662.4	9821.1
14	2	276	0.0	0.0	9821.1
15	31	307	36.6	1134.6	10955.7
16a	20.553	327.553	36.4	748.1	11703.8
16b	10.447	338	36.4	380.3	12084.1
17	27	365	0.0	0.0	12084.1
18	29	394	35.2	1020.8	13104.9
19	47	441	35.7	1677.9	14782.8
20	35	476	36.1	1263.5	16046.3
21	2	478	0.0	0.0	16046.3
22	25	503	34.7	867.5	16913.8
23	2	505	0.0	0.0	16913.8
24	40	545	34.2	1368.0	18281.8
25	51	596	33.6	1713.6	19995.4
26	4	600	0.0	0.0	19995.4
27	43	643	32.6	1401.8	21397.2
28	58	701	32.8	1902.4	23299.6
29	29	730	0.0	0.0	23299.6
30c	2.913	732.913	37.1	108.1	23407.7
30d	50.087	783	37.1	1858.2	25265.9
31	40	823	37.0	1480.0	26745.9
32	6	829	0.0	0.0	26745.9
33	27	856	36.9	996.3	27742.2
34	10	866	0.0	0.0	27742.2
35	17	883	36.8	625.6	28367.8
36	9	892	0.0	0.0	28367.8
37e	24.102	916.102	37.0	891.8	29259.6
37f	15.898	932	37.0	588.2	29847.8
38	33	965	37.1	1224.3	31072.1
39	27	992	0.0	0.0	31072.1
40	37	1029	36.9	1365.3	32437.3
41	26	1055	0.0	0.0	32437.3
42	43	1098	37.1	1595.3	34032.7
43	16	1114	0.0	0.0	34032.7
44	29	1143	37.2	1078.8	35111.5

a Last period of cycle 1.
b First period of cycle 2.
c Last period of cycle 2.
d First period of cycle 3.
e Last period of first half of cycle 3.
f First period of last half of cycle 3.

Table A-5

SPECIFIC POWER VALUES FOR CALCULATING $P_{max,avg}$

Time Interval	$\bar{P}$ (MW/MTU)	$\sigma_{\bar{P}}$ (MW/MTU)	ΔP_{bias} (MW/MTU)	P_m (MW/MTU)
Last 10 days	37.2	0.32	0.31	38.150
Last 30 days	37.2	0.32	0.31	38.150
Last 60 days	37.166	0.237	0.31	37.950
Last half cycle	37.061	0.149	0.31	37.669
Last cycle	37.027	0.110	0.31	37.557
All 3 cycles	35.975	0.0635	0.31	36.412

APPENDIX B

CALCULATION OF EFFECTIVE QUANTITY OF ^{59}Co

This appendix shows how the effective amount of ^{59}Co used in the ORIGEN-S afterheat power calculations was derived. To determine if an additional safety factor for excess ^{60}Co afterheat power is needed, a similar calculation should be performed for the fuel assembly in question.

Table B-1 shows the amount of ^{59}Co (g/MTU) present in the three major regions of a Westinghouse 17x17 fuel assembly. The fuel zone of the assembly has no correction for flux depression or spectral differences. The fuel-gas plenum zone has an activation ratio correction of 0.042 to account for the lower flux density in this zone. The end fitting zone has an activation ratio correction of 0.011. Both the fuel-gas plenum zone and end fitting zone have a correction factor of 0.67 to account for changes in the ^{59}Co capture cross section due to differences in the neutron energy spectrum from that in the fuel zone. Both the activation ratio correction and the spectrum correction are derived from axial flux calculations for a PWR fuel assembly.*

To derive the effective amount of ^{59}Co to use for the afterheat power calculation, the total ^{59}Co content of an assembly zone is multiplied by the correction factors for that zone and the effective amounts from all three zones are summed. For the afterheat power calculations discussed in Regulatory Position 1, 71.48 g/MTU of ^{59}Co is given by $71.22 + 3.48 \times 0.042 \times 0.67 + 21.76 \times 0.011 \times 0.67$.

Note that an additional safety factor for excess ^{60}Co afterheat power is to be applied only if the total amount of ^{59}Co (including any uncertainty) for the assembly in question is greater than 142.96 g/MTU. The safety factors discussed in Regulatory Position 1 already include an allowance for a 100% uncertainty in the initial amount of ^{59}Co in a Westinghouse 17x17 fuel assembly.

*A. G. Croff et al., "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1978.

Table B-1

^{59}Co CONTENT OF A WESTINGHOUSE 17x17 FUEL ASSEMBLY

Assembly Zone	^{59}Co Content in Constituent Materials (g/MTU)					
	Zircaloy-4	Inconel-718	SS304	SS302	Nicrobraze 50	Total
Fuel	2.23	60.08	7.92	0.00	0.99	71.22
Fuel-gas plenum	0.12	0.00	0.00	3.36	0.00	3.48
End fitting	0.00	0.00	21.76	0.00	0.00	21.76

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Spent fuel may be stored in either a wet or dry mode. Water basins have largely been used since the mid-1940s for the handling, transfer, and storage of spent fuel at reactor sites. There is a strong indication that spent fuel can be stored under water for several decades without serious degradation. Dry storage has not been used for spent fuel from commercial light-water reactors but has been used for a number of years for other types of spent fuel. One of the principal design parameters for a spent fuel storage installation that is applicable to all modes of storage is the heat generation rate for use as design input for an Independent Spent Fuel Storage Installation (ISFSI). The proposed action would present a method acceptable to the NRC staff for calculating conservative values of these heat generation rates.

1.2 Need

A number of methods have been developed for calculating the long-term heat generation rates of spent fuel from power reactors as a function of burnup and decay times. No standard method for calculating this long-term heat generation has been published by NRC, nor has the practice of applicants or the staff in this respect been uniform. A definite need for criteria covering this parameter exists.

1.3 Value/Impact

1.3.1 NRC

The value of the proposed action to the NRC staff will be the consistent use of standardized methodology to determine heat generation values. Since the accepted values of heat generation and the methodology proposed to determine such values will be the same as the values and methodology often used now by the staff, there should be little impact.

1.3.2 Other Government Agencies

This proposed action would provide a standardized set of values for use by government agencies such as the Department of Energy in the design and construction of ISFSI facilities.

1.3.3 Industry

The value/impact on licensees will be the same as that on the NRC staff. The time required to determine heat generation values by the proposed methodology is believed to compare favorably with the time required for use of the previously existing approaches. Further, adoption of the proposed methodology would provide both a standardized set of values and a standardized methodology for determining such values.

1.3.4 Public

No impact on the public can be foreseen.

1.4 Decision

Standard methodology should be used for calculating the long-term heat generation rates as a function of burnup and decay times.

2. TECHNICAL APPROACH

2.1 Technical Alternatives

The proposed action requires the specification of burnup, specific power, enrichment, and ^{59}Co content, and the methodology employs the use of the ORIGEN-S code that is a part of the SCALE code system developed for the NRC through the Office of Nuclear Material Safety and Safeguards or a combination of ORIGEN-S with ANS/ANSI-5.1-1979. No technical alternatives are needed.

2.2 Decision on Technical Approach

The proposed action uses the SAS2 control module within the SCALE system to provide time-dependent cross section libraries for the actinide elements and fission products. These libraries, in conjunction with the ORIGEN-S code, yield decay heat generation rates as a function of decay time. An acceptable range of values derived in this manner can be established by the NRC staff.

3. PROCEDURAL APPROACH

3.1 Procedural Alternatives

Procedures that may be used to carry out the proposed action and technical approach include the following:

- Regulation
- Regulatory guide
- Branch position
- NUREG-series report

3.2 Value/Impact of Procedural Alternatives

The matter is not of sufficient importance to justify issuance of a regulation. No branch position has been prepared or is anticipated. NUREG/CR-2397, a contractor report, was published in September 1982.

3.3 Decision on Procedural Approach

A regulatory guide should be prepared.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

The authority for this guide is derived from the safety requirements of the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as implemented by the Commission's regulations, in particular, 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation."

4.2 Need for NEPA Assessment

The proposed action is not a major Federal action significantly affecting the quality of the human environment and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

Additional guidance on ISFSIs is contained in the following regulatory guides:

1. Regulatory Guide 3.44, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type)," deals with wet-mode storage.

2. Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)," deals with dry-mode storage.

3. Regulatory Guide 3.50, "Guidance on Preparing a License Application To Store Spent Fuel in an Independent Spent Fuel Storage Installation," deals with the format to be used in the license application for an ISFSI.

These guides identify the need to calculate the heat generated by spent fuel but do not provide guidance on methodology to be used in the calculation.

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