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An Approach to Treating Radionuclide Decay Heating for Use in the MELCOR Code System

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ABSTRACT

A new code system is being developed for use in assessment of nuclear reactor accident risks. The code system, termed MELCOR, will treat thermal-hydraulic and fission product behavior jointly. As part of its treatment of thermal-hydraulic processes, the code system will evaluate decay heating from fission product inventories contained within the reactor core debris and compartments that are defined for the reactor system and containment. A simple approach to treating radionuclide decay heating is proposed for use in MELCOR. The proposed approach uses a table-lookup to estimate element decay powers as a function of time after reactor shutdown (start of accident). Decay power for each element in a compartment of the reactor system is found by multiplying the mass of the element in the compartment by the element's decay-heat rate per unit mass which is a function of time after reactor scram. The approach assumes that daughter products are transported along with the parent radionuclide during the accident. The validity of this assumption is discussed. In addition, methods for apportioning the decay energy between the walls and the gases in a compartment are also discussed. The proposed approach is based on SANDIA-ORIGEN calculations for a 3412 MWt PWR and a 3578 MWt BWR.

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1.0 Introduction

A new code system is being developed for the U.S. Nuclear Regulatory Commission (NRC) to replace MARCH, CORRAL, and CRAC (Sp-83) for assessing the risk of severe reactor accidents. The code system, termed MELCOR, will treat thermal-hydraulic and fission product behavior jointly. As part of its treatment of thermal-hydraulic processes, MELCOR will consider decay heating resulting from fission product inventories contained within the reactor core debris and the compartments that are defined for the reactor system and containment.

The heat that drives the thermal-hydraulic progression of severe LWR accidents results from decay of the radionuclide inventory present at reactor scram. Radionuclide decay heating is equivalent to about 6% of reactor operating power at shutdown and to about 0.5% of reactor operating power after one day. Accordingly, MELCOR must be capable of realistically modeling the time-dependent production of decay heat. In principal, decay heat can be exactly calculated by following the radioactive decay chains of all of the radionuclides present at reactor scram. However, such calculations are not practical because of the computation time required. Thus, a simplified approach to treating decay heating was sought for the MELCOR code.

Radionuclide decay heating is treated in most currently available risk codes with precalculated data. Thermal-hydraulic codes such as MARCH (Wo-80) use the ANS standard whole core decay heat curve with an approximate correction for the release of radionuclides from overheated or molten core materials. Fission product behavior codes such as CORRAL (US-75) or TRAP-MELT (Jo-79) use externally calculated thermal-hydraulic data from codes such as MARCH, and assume that the effects of radionuclide decay heat are embedded in the data. Codes that couple fission product behavior and thermal-hydraulic calculations either use decay heat curves precalculated for important decay heat producing elements (e.g., CORCON (Mu-81)) or use ANS standard decay heat data which is corrected by the code for a selected set of radioactive decay chains (e.g., CONTAIN (Se-82)). The element-specific decay heat curves are for the total inventories of the elements. These total inventories include inventories of the element's isotopes that were present at reactor scram as well as inventories of the element's isotopes that are produced as a result of decay processes after scram. These data ignore possible differences in the physical location of the initial inventories and the decay product inventories of elements that might result during accident progression. Since it is desired 1) that MELCOR be fast running and 2) that MELCOR be capable of treating decay heating within different compartments, a new approach to treating radionuclide decay heating was sought.

This paper describes a simple approach to calculating decay heat generation for reactor primary system and containment compartments following severe LWR accidents that should be suitable for use in MELCOR. The approach uses precalculated tables of decay heat data for elements that are present in the reactor core at shutown. Following reactor shutdown, radioactive decay products are assumed to retain the physical characteristics of their parent elements and thus are assumed to transport with their parents (e.g., the daughter product of an isotope in aerosol form is assumed to remain in aerosol form). The decay power due to each element in a compartment is determined by multiplying the mass of the element in that compartment by the element's decay heat rate per unit mass. This latter value is a function of the time after reactor shutdown.

This approach is given in Section 2.0 for a standard pressurized water reactor (PWR) and a standard boiling water reactor (BWR). Section 3.0 discusses the validity of assuming that daughter products are transported with their parent radionuclides during accident progression. Also discussed in Section 3.0 is a method for the apportionment of decay energy from gas-borne or deposited aerosols and vapors to compartment bulk gases and surfaces.

2.0 Calculations and Results

This section illustrates the proposed method for evaluating radionuclide decay power for a standard PWR and a standard BWR. The Sandia National Laboratories version of the ORIGEN computer code (SANDIA-ORIGEN (Be-79)) was used to perform the time- and element-specific decay heat calculations. Data were calculated for a 3412 MWt Westinghouse PWR and a 3578 MWt General Electric BWR. The PWR calculations assumed a three-year refueling cycle with the core charged with uranium enriched to 3.3 w/o U-235 and with respective end-of-year burnups of 11,800, 22,400 and 33,600 MWt per metric ton of uranium, a constant specific power of 38.3 MWt per metric ton of uranium charged, and an 80% capacity factor. The BWR calculations were performed for each of five "assembly groups." Each group is defined by a set of assumptions regarding fuel enrichment and burnup history. Initial enrichment assumed for assemblies was either 2.83 or 2.66 w/o U-235 with the number of years that a fuel assembly remains in the core being 3 or 4 years depending on the assembly group. The SANDIA-ORIGEN code determines the BWR core inventories by summing the results for the assembly groups. The BWR was also assumed to be refueled annually and to be operated at an 80% capacity factor. Further details on calculation assumptions are contained in Be-79.

The SANDIA-ORIGEN code provides estimates of decay power as a function of time after reactor scram. To facilitate the incorporation of decay heat information into the MELCOR code, decay power per MWt operating power for each element was calculated by dividing the total decay power due to the inventory of the element at reactor shutdown (watts) by the operating power of the reactor before shutdown. The total decay power for an element at any time after scram is the sum of the decay power of the isotopes of the element and their radioactive daughters. Element decay powers are for fuel constituents (i.e., fission products and actinides). Table 1 gives tabulations of "normalized" decay power (watts/kWt) for each of 29 elements initially present in the 3412 MWt PWR fuel. These elements collectively produce more than 99% of the reactor core decay heat at reactor scram. In addition, the decay heats of these elements plus their daughter products produce at a minimum 99% of the total decay heat up to 2 days after the accident.

The results for the 3578 MWt BWR are nearly the same as those for the PWR during the first few days after reactor shutdown (Table 5 provides a comparison of selected normalized decay powers for the PWR and BWR). This similarity results because 1) both reactors use thermal fission of U-235 as the power source and 2) decay power during the first few days after shutdown results principally from short-lived radionuclides. Inventories of short-lived radionuclides are proportional to reactor operating power and are relatively insensitive to reactor design and fuel management. Thus, the normalized decay powers given in Table 1 can be considered representative of both the 3412 MWt PWR and the 3578 MWt BWR.

Decay heating within a compartment can be determined by multiplying the masses of elements in the compartment by "specific powers" appropriate to the time elapsed since scram. That is,

$$DP_i(t) = \sum_j M_{ij}(t) * NDP_j(t)/NM_j \qquad (1)$$

where $DP_i(t) \equiv$ Total decay power in compartment i at time t after reactor scram (watts),

$M_{ij} \equiv$ Mass of element j from fuel that is present in compartment i at time t after reactor scram (gram),

$NDP_j \equiv$ Normalized decay power for element j at time t after scram (watt/Kwatt) and

$NM_j \equiv$ Normalized mass inventory of element j present in the reactor core at scram (gram/Kwatt).

Normalized decay powers, NDP_j, are given in Table 1 for 29 elements. Estimates of decay power at times in the accident progression other than those given in Table 1 can be determined by logarithmic interpolation. Normalized mass inventories, NM_j, for 37 elements are given in Table 2 for the 3412 MWt PWR and in Table 3 for the 3578 MWt BWR. These inventories are for fuel constituents. Mass inventories for core structural materials are given in Table 4.

Decay power during the first few days after reactor scram is relatively insensitive to time in the annual operating cycle. During the first few days, decay power results mostly from decay of short-lived radionuclides. Core inventories of short-lived radionuclides will reach equilibrium levels within a few days after reactor startup and will remain at these levels until reactor shutdown. Therefore, the decay powers given in Table 1 can be considered representative of any time in the annual operating cycle. The decay heat data may overpredict decay power if reactor scram occurs during the first few days after startup.

Mass inventories of elements in the reactor fuel vary during the annual operating cycle. Mass inventories of fission product elements result primarily from long-lived or stable isotopes. Inventories of these isotopes are sensitive to fuel burnup and to fuel management history. Although short-lived isotopes dominate the short-term decay power, these isotopes contribute negligibly to mass inventories. Mass inventories of elements in fuel at various times during the annual operating cycle can be obtained by linear interpolation between the beginning- and end-of-cycle figures given in Tables 2 and 3.

Decay power associated with the reactor core debris can be estimated with the use of total core decay power data given in Table 6 corrected for release of any of the 29 elements in Table 1. That is,

$$DDP(t) = TDP(t) - \sum_j RM_j(t) * NDP_j(t)/NM_j \qquad (2)$$

where $DDP(t) \equiv$ Core debris decay power at time t into accident progression (watt),

$TDP(t) \equiv$ Total decay power at time t into accident (watt), and

$RM_j \equiv$ Mass inventory of fuel element j removed from the core debris by time t into the accident progression (gram).

The other variables are as defined for equation 1. Estimates of decay power at times in the accident progression other than those given in Table 6 can be determined by logarithmic interpolation.

A comparison of the ANS standard decay heat curve as implemented by MARCH 2 (Wo-84) and the decay heat curve calculated by the SANDIA-ORIGEN code is given in Figure 1. The curves are in terms of decay power per unit of reactor operating power. The decay powers predicted by SANDIA-ORIGEN are shown to be from 10 to 30% smaller than those predicted by the MARCH 2 implementation of the ANS standard. A major cause of this difference is that MARCH 2 uses the "simplified method" for estimating radionuclide decay heat power. This method, which is described in AN-79, assumes that the decay power from fissioning isotopes other than U-235 is identical to that of U-235. The method overestimates decay heat power, especially with respect to LWR cores containing an appreciable amount of plutonium (AN-79).

The SANDIA-ORIGEN results have uncertainty due to 1) simplifying assumptions that are incorporated into the code and 2) uncertainty in the nuclear data library. Uncertainty in the SANDIA-ORIGEN results is probably less than a few ten's of percent (Be-79). The importance of this uncertainty for MELCOR results is probably small relative to other uncertainties in the modeling problem (Sp-84).

3.0 Summary and Discussion

Section 2.0 illustrated a method for evaluating the decay heat associated with reactor core debris and with gas-borne and deposited aerosols and vapors. This section discusses 1) the appropriateness of the assumption that element daughter products are transported along with their parent radioisotopes during accident progression, 2) the apportionment of decay energy between compartment bulk gases and surfaces, and 3) the application of the data discussed in Section 2.0 to other reactors.

The approach to treating decay heat assumes that daughter products of the element isotopes retain the physical characteristics of their parents and thus transport with the parents--an assumption that may not be appropriate in some

situations (e.g., consider the decay of fission product iodine present in an aerosol to radioactive isotopes of xenon). Although clearly sometimes in error, the impact of this simplifying assumption will, in most situations, not have a substantial impact on decay heat estimates. The analyses performed with the SANDIA-ORIGEN computer code show that, for all elements examined, the collective decay heats of parent isotopes are generally much greater than those of their daughter products. For example, during the first few hours after reactor shutdown, decay heat from radioiodine daughter products (principally Xe isotopes) is minor (<2% of the decay heat from parent iodine isotopes). Radioiodines provide approximately 10% of the total reactor decay heat during the first few hours after reactor shutdown; thus, the contribution of xenon daughters of iodine to total decay heat is less than about 0.2%. The above is also true for other important decay heat contributors such as La and Cs. In general, the decay heat produced by an element and its daughters will be largely due to decay of that element's initial isotopic inventory. Accordingly, the errors caused by assuming that the daughter products transport identically with their parents (an assumption that is most often correct) will be small.

The MELCOR system will model the movement of radioactive aerosols and vapors between compartments defined for the reactor primary system and containment (Sp-83). The medium into which decay energy is deposited (i.e., compartment bulk gasses or surfaces) can be important for thermal-hydraulic calculations. Decay heat results from emissions of gamma rays and beta and alpha particles. These radiations are emitted isotropically and thus, radionuclides deposited onto a surface will emit about 50% of their radiation into the surface and 50% away from the surface.

The thicknesses of solid surface materials required to absorb the energy of nuclear radiation will vary depending on particle type and energy. An analysis of the SANDIA-ORIGEN results suggests that about 50% of total core decay heating results from charged particle emissions (i.e., alpha and beta particles). Most of the remainder results from gamma-rays with energies between 0.5 and 2.0 MeV. Charged particles that are emitted into solid surfaces will dissipate their energy within a very small thickness of the surface material. For example, a 2 MeV beta particle would have a stopping distance of about 0.1 cm in steel and 0.5 cm in concrete. Alpha particles would have smaller ranges. Gamma rays would require greater thicknesses of materials to dissipate their energy with the thicknesses being dependent on gamma-ray energy. About 90% of 2 MeV gamma radiation would be absorbed within about 4 cm of steel. One MeV gamma radiation would require concrete thicknesses of up to 10 cm to absorb 90% of the gamma radiation while two MeV gamma radiation would require up to 15 cm of concrete. Most concrete surfaces in a reactor containment will have thicknesses greater than 15 cm. Thus, the energy from radiation emitted into a surface will for the most part be absorbed by the surface or the bulk material behind the surface.

Radiations emitted away from a surface would be absorbed principally by gasses over that surface or by other surfaces across the gas volume depending on the types of radiation emitted and the geometry of the compartment. The stopping distance for 2 MeV beta particles in air is about 8 m while that of a 1 MeV beta is about 3 m. An 0.5 MeV beta particle would have a stopping distance of about 1.5 m in air. Increasing the mass density over that of air at STP due to either increased pressure or addition of steam or aerosol particles would reduce the maximum range of beta particles. Analysis of aerosol behavior in reactor compartments (Sp-84) suggests

that aerosol mass loadings in compartment atmospheres could be as large as 10s of grams per cubic meter. Since the mass density of air at standard temperature and pressure is on the order of 1.3 kg per cubic meter, aerosols suspended in the compartment bulk gasses would not have an appreciable impact on stopping distances for beta particles. The mass density of saturated steam at 50 psia is on the order of 1.5 kg per cubic meter and about 1.0 kg per cubic meter at 30 psia. A compartment atmosphere composed of bulk gasses and steam at a pressure of 50 psia would have a mass density of about 2.5 kg per cubic meter. At this mass density, the stopping distance of beta particles is about half that in air at standard temperature and pressure. Alpha particles have a range of about 7 cm in air at standard temperature and pressure.

Gamma rays would require a relatively large thickness of air for complete dissipation of energy. The mean free path of an 0.5 MeV gamma ray in air is on the order of 90 m while that of a 1.0 MeV gamma ray is on the order of 120 m. Generally, less than 10% of the energy of 0.5 MeV gamma radiation would be absorbed within a distance of 10 m from the source. Addition of steam to a compartment atmosphere could decrease the mean free path of gamma particles by a factor proportional to the change in mass density of the atmosphere. Thus, it is assumed that steam in the compartment atmosphere doubles the mass density over that of air at standard temperature and pressure, then the mean path of gamma rays in the atmosphere would be approximately halved. Under these conditions, an 0.5 MeV gamma ray would still have a mean free path on the order of 45 m. Even in steam laden compartments, gamma rays emitted into the compartment atmosphere will likely traverse that atmosphere and deposit their energy into a compartment wall.

In light of the above discussion, the following approach to apportioning decay heat is recommended. Fifty percent of all decay energy from material deposited on a compartment surface can be assumed to be absorbed by that surface. This 50% component includes the beta, gamma, and alpha emissions into the surface. The energy of the gamma emissions away from a compartment surface can be assumed to be deposited in the other compartment surfaces. Since about 50% of the total decay heating will result from gamma emissions, 25% of the decay energy from deposits to a surface is deposited in the other compartment surfaces. The remaining 25% of the decay heat (beta and alpha emissions away from the surface) can be assumed to be deposited in the compartment bulk gasses.

The energy of gamma rays from radionuclides suspended in compartment atmospheres can be assumed to be deposited uniformly into the compartment wall materials. The amount of decay energy deposited to each surface can be assigned according to that surface's fraction of total compartment surface area. The remaining 50% of the decay energy (alpha and beta particles) can be assumed to be deposited in the compartment bulk gasses. It must be kept in mind that the above apportionment of decay heat is approximate. The fraction of decay heat deposited in bulk gasses and in surfaces would actually depend on the radionuclide and on the geometry and size of the compartment.

The data discussed in Section 2.0 should not be taken as representative of all light water reactors. The decay power for typical LWR cores in the 1 hr to 10 day time-frame is nearly proportional to operating power. The decay powers given in Tables 1 and 6 can be considered to be representative of most LWR cores during the first few days after reactor scram. Mass inventories of elements are sensitive to

burnup and to reactor design. The mass inventories given in Tables 2, 3 and 4 are for a representative BWR and a representative PWR. These mass inventories can be used for other reactors if the reactor power densities and fuel burnups for these reactors are similar to those for the reactors treated in this report.

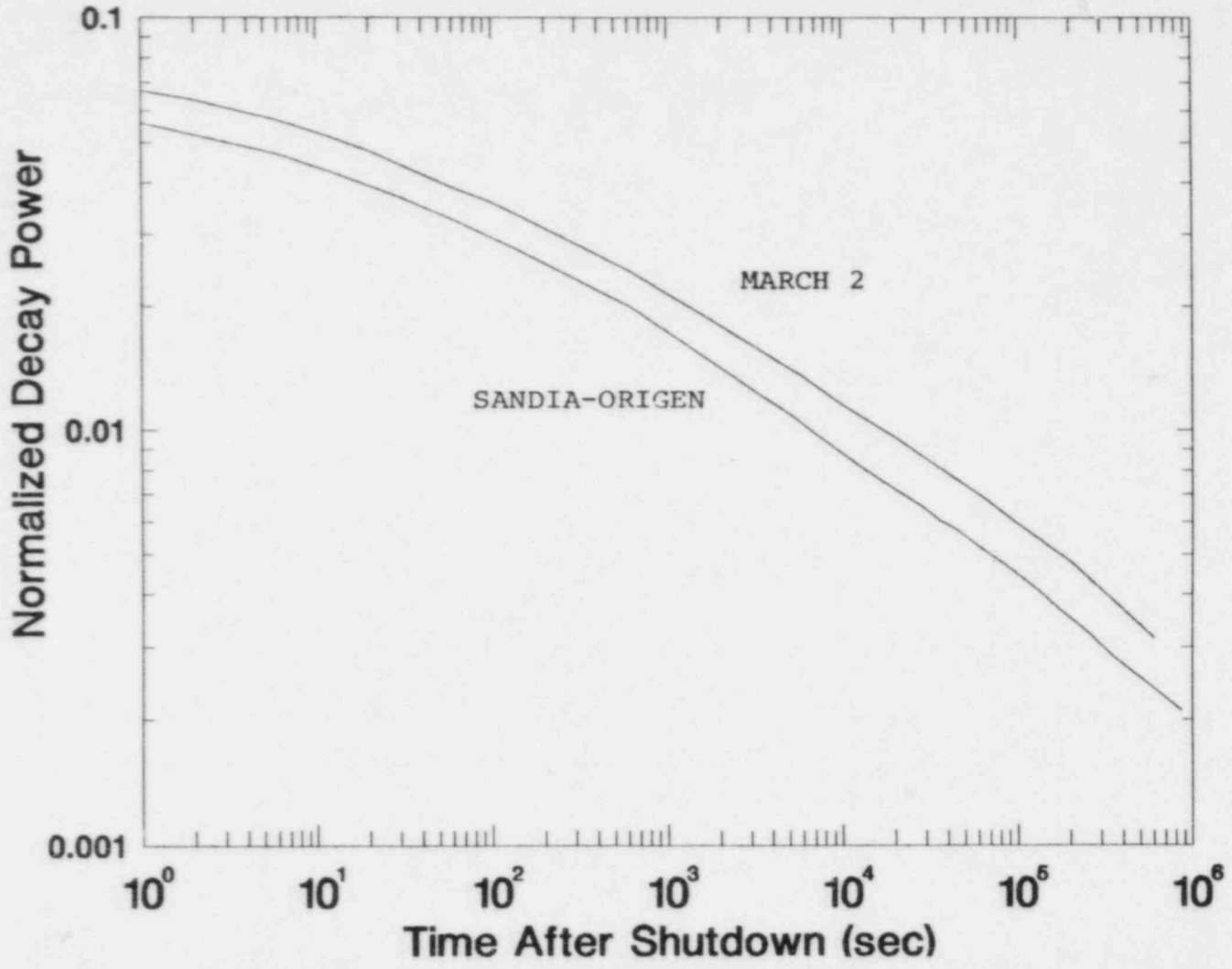

Figure 1 Comparison of SANDIA-ORIGEN and MARCH 2 Decay Heat Curves.

Table 1. Normalized Decay Powers as a Function of Time After Shutdown for the 3412 MWt PWR.

Element	Normalized Decay Power (Watt/Kwatt)						
	Time After Shutdown (hr)						
	0.0	0.0017	0.0033	0.005	0.0083	0.017	0.033
As	0.34	0.16	0.11	0.083	0.051	0.021	0.0089
Se	0.68	0.52	0.45	0.39	0.30	0.19	0.11
Br	2.0	1.3	1.0	0.88	0.70	0.47	0.30
Kr	2.4	1.9	1.7	1.6	1.5	1.3	1.1
Rb	4.4	3.2	2.8	2.6	2.3	2.1	1.6
Sr	3.4	2.6	2.4	2.3	2.1	1.8	1.5
Y	5.3	3.2	3.2	3.2	2.9	2.9	2.6
Zr	2.3	2.5	2.1	1.8	1.3	0.91	0.70
Nb	5.0	2.2	1.5	1.2	0.97	0.74	0.62
Mo	1.9	2.1	2.1	2.1	2.0	1.8	1.5
Tc	2.3	1.9	1.6	1.5	1.4	1.2	0.97
Ru	0.53	0.53	0.56	0.56	0.59	0.59	0.56
Rh	0.62	0.53	0.47	0.44	0.35	0.26	0.18
Pd	0.038	0.035	0.035	0.035	0.032	0.029	0.028
Ag	0.059	0.041	0.035	0.029	0.025	0.019	0.015
Sn	0.44	0.38	0.35	0.35	0.32	0.29	0.23
Sb	1.8	1.6	1.6	1.5	1.5	1.4	1.2
Te	2.1	1.8	1.7	1.6	1.4	1.3	1.2
I	4.7	4.1	4.1	3.8	3.5	3.3	2.9
Xe	2.4	2.1	1.9	1.8	1.7	1.5	1.2
Cs	4.4	3.2	3.2	2.9	2.8	2.3	1.9
Ba	2.6	2.2	2.0	1.8	1.6	1.4	1.3
La	4.4	4.1	3.8	3.5	3.2	2.9	2.6
Ce	1.0	0.88	0.85	0.85	0.82	0.73	0.64
Pr	1.4	1.3	1.3	1.2	1.2	1.1	1.0
Nd	0.17	0.17	0.17	0.17	0.17	0.16	0.16
Pm	0.12	0.12	0.11	0.11	0.11	0.097	0.085
U	1.2	1.2	1.2	1.2	1.2	1.2	1.2
Np	0.73	0.73	0.73	0.73	0.73	0.73	0.73

Table 1. Normalized Decay Powers as a Function of Time After Shutdown for the 3414 MWt PWR. (cont.)

Element	Normalized Decay Power (Watt/Kwatt) Time After Shutdown (hr)						
	0.067	0.17	0.33	1.0	1.5	2	4
As	0.0053	0.0038	0.0029	0.0017	0.0013	0.0011	0.00050
Se	0.067	0.041	0.029	0.0097	0.0047	0.0023	0.00038
Br	0.19	0.13	0.097	0.044	0.027	0.017	0.0056
Kr	0.97	0.79	0.76	0.70	0.62	0.53	0.32
Rb	1.2	0.70	0.41	0.073	0.021	0.0064	0.00067
Sr	1.3	1.1	0.88	0.76	0.73	0.70	0.62
Y	2.3	1.7	1.4	0.88	0.76	0.70	0.56
Zr	0.64	0.67	0.67	0.73	0.79	0.79	0.82
Nb	0.56	0.50	0.47	0.38	0.35	0.29	0.23
Mo	1.2	0.85	0.61	0.25	0.19	0.17	0.17
Tc	0.79	0.59	0.38	0.094	0.050	0.038	0.025
Ru	0.50	0.44	0.38	0.35	0.35	0.32	0.29
Rh	0.13	0.11	0.091	0.062	0.053	0.044	0.032
Pd	0.024	0.021	0.019	0.016	0.016	0.016	0.016
Ag	0.013	0.012	0.011	0.0097	0.0088	0.0082	0.0067
Sn	0.17	0.12	0.10	0.070	0.053	0.038	0.014
Sb	1.0	0.67	0.50	0.26	0.18	0.13	0.076
Te	1.1	1.1	1.0	0.85	0.79	0.76	0.64
I	2.8	2.6	2.4	1.9	1.6	1.5	1.0
Xe	0.94	0.70	0.53	0.29	0.20	0.15	0.091
Cs	1.6	1.2	0.91	0.38	0.23	0.15	0.073
Ba	1.2	1.0	0.79	0.47	0.38	0.35	0.27
La	2.4	2.3	2.1	1.6	1.5	1.3	1.0
Ce	0.56	0.50	0.50	0.53	0.50	0.47	0.47
Pr	0.91	0.73	0.56	0.29	0.23	0.20	0.17
Nd	0.16	0.15	0.13	0.094	0.085	0.082	0.070
Pm	0.073	0.053	0.044	0.041	0.041	0.041	0.038
U	1.1	0.91	0.70	0.22	0.10	0.053	0.018
Np	0.73	0.73	0.73	0.71	0.71	0.71	0.71

Table 1. Normalized Decay Powers as a Function of Time After Shutdown for the 3412 MWt PWR. (cont.)

Element	Normalized Decay Power (Watt/Kwatt) Time After Shutdown (hr)						
	6	8	10	12	15	20	24
As	0.00027	0.00017	0.00013	0.00011	0.00010	-	-
Se	0.00016	-	-	-	-	-	-
Br	0.0035	0.0024	0.0018	0.0015	0.0012	0.00097	0.00085
Kr	0.18	0.11	0.067	0.041	0.021	0.0067	0.0028
Rb	0.00056	0.00050	0.00044	0.00038	0.00032	0.00024	0.00020
Sr	0.53	0.44	0.39	0.32	0.27	0.21	0.18
Y	0.47	0.41	0.35	0.32	0.27	0.22	0.20
Zr	0.79	0.76	0.70	0.67	0.62	0.56	0.50
Nb	0.21	0.20	0.20	0.20	0.20	0.20	0.20
Mo	0.17	0.17	0.17	0.17	0.17	0.16	0.16
Tc	0.020	0.016	0.012	0.0097	0.0067	0.0038	0.0024
Ru	0.28	0.26	0.25	0.23	o.23	0.22	0.21
Rh	0.026	0.022	0.020	0.018	0.017	0.015	0.014
Pd	0.015	0.014	0.013	0.012	0.011	0.0088	0.0073
Ag	0.0056	0.0047	0.0041	0.0038	0.0035	0.0032	0.0029
Sn	0.0067	0.0044	0.0035	0.0029	0.0026	0.0026	0.0023
Sb	0.059	0.047	0.038	0.032	0.026	0.020	0.017
Te	0.64	0.64	0.64	0.64	0.64	0.62	0.59
I	0.82	0.70	0.62	0.53	0.47	0.38	0.32
Xe	0.082	0.076	0.073	0.070	0.067	0.062	0.059
Cs	0.062	0.062	0.062	0.062	0.062	0.059	0.059
Ba	0.25	0.26	0.27	0.29	0.32	0.35	0.41
La	0.88	0.79	0.73	0.70	0.64	0.59	0.53
Ce	0.44	0.44	0.44	0.41	0.41	0.41	0.41
Pr	0.15	0.13	0.12	0.11	0.11	0.094	0.088
Nd	0.064	0.062	0.059	0.059	0.059	0.056	0.056
Pm	0.038	0.035	0.035	0.032	0.032	0.029	0.027
U	0.017	0.016	0.016	0.016	0.016	0.015	0.015
Np	0.67	0.64	0.64	0.62	0.62	0.56	0.53

Table 1. Normalized Decay Powers as a Function of Time After Shutdown for the 3412 MWt PWR. (cont.)

Element	Normalized Decay Power (Watt/Kwatt) Time After Shutdown (hr)						
	36	48	72	96	144	192	240
As	-	-	-	-	-	-	-
Se	-	-	-	-	-	-	-
Br	0.0067	0.00053	0.00032	0.00021	-	-	-
Kr	0.00050	0.00032	0.00029	0.00029	0.00029	0.00029	0.00029
Rb	0.00013	0.00010	-	-	-	-	-
Sr	0.13	0.11	0.10	0.10	0.10	0.10	0.097
Y	0.16	0.14	0.13	0.12	0.12	0.11	0.11
Zr	0.38	0.32	0.26	0.24	0.23	0.23	0.23
Nb	0.19	0.19	0.19	0.18	0.18	0.17	0.16
Mo	0.14	0.12	0.094	0.073	0.044	0.027	0.016
Tc	0.00064	0.00020	-	-	-	-	-
Ru	0.21	0.21	0.20	0.20	0.19	0.19	0.18
Rh	0.011	0.0088	0.0056	0.0035	0.0013	0.00053	0.00021
Pd	0.0044	0.0026	0.00097	0.00038	-	-	-
Ag	0.0028	0.0027	0.0025	0.0024	0.0021	0.0019	0.0017
Sn	0.0022	0.0021	0.0019	0.0018	0.0016	0.0014	0.0013
Sb	0.014	0.012	0.010	0.0085	0.0062	0.0044	0.0032
Te	0.53	0.47	0.38	0.32	0.20	0.13	0.091
I	0.21	0.16	0.11	0.082	0.059	0.047	0.041
Xe	0.053	0.047	0.041	0.035	0.028	0.021	0.016
Cs	0.059	0.059	0.059	0.059	0.056	0.056	0.053
Ba	0.50	0.56	0.64	0.67	0.70	0.64	0.62
La	0.44	0.35	0.23	0.16	0.067	0.029	0.013
Ce	0.38	0.35	0.32	0.32	0.29	0.28	0.28
Pr	0.079	0.076	0.070	0.067	0.062	0.056	0.050
Nd	0.056	0.053	0.050	0.044	0.038	0.035	0.029
Pm	0.022	0.018	0.013	0.0091	0.0050	0.0029	0.0020
U	0.014	0.013	0.011	0.0097	0.0076	0.0059	0.0047
Np	0.47	0.41	0.29	0.22	0.12	0.067	0.038

Table 2 Mass of Elements in the 3412 MWt PWR Fuel at Different Times in the Annual Operating Cycle.

Element	Normalized Mass x 10^{-3} (g/Kwatt)			
	Beginning of Cycle	1/3 Cycle	2/3 Cycle	End of Cycle
As	0.0011	0.0015	0.0019	0.0023
Se	0.53	0.71	0.89	1.1
Br	0.21	0.28	0.35	0.42
Kr	3.7	4.9	6.1	7.3
Rb	3.5	4.6	5.8	6.9
Sr	9.7	13	16	19
Y	5.0	6.7	8.1	9.6
Zr	34	45	56	67
Nb	0.65	0.84	1.0	1.0
Mo	29	38	47	57
Tc	7.7	10	12	15
Ru	19	25	32	38
Rh	4.0	4.7	5.9	6.9
Pd	7.0	9.7	13	17
Ag	0.40	0.55	0.70	0.88
Cd	0.40	0.56	0.78	1.0
In	0.014	0.018	0.022	0.025
Sn	0.36	0.49	0.63	0.78
Sb	0.13	0.18	0.23	0.29
Te	3.7	5.1	6.4	7.7
I	1.8	2.6	3.2	3.9
Xe	47	62	78	93
Cs	25	33	41	49
Ba	12	17	21	25
La	11	15	19	22
Ce	27	37	45	53
Pr	10	13	16	20
Nd	32	43	54	65

Table 2 Mass of Elements in the 3412 MWt PWR Fuel at Different Times in the Annual Operating Cycle. (cont.)

Element	Normalized Mass x 10^{-3} (g/Kwatt)			
	Beginning of Cycle	1/3 Cycle	2/3 Cycle	End of Cycle
Pm	1.9	2.3	2.7	2.9
Sm	4.1	5.4	6.7	7.9
Eu	1.0	1.4	1.9	2.4
Gd	0.38	0.54	0.78	1.1
U	$2.6(10^4)$	$2.6(10^4)$	$2.5(10^4)$	$2.5(10^4)$
Np	2.9	5.9	7.2	8.8
Pu	95	120	140	160
Am	0.28	0.47	0.76	1.2
Cm	0.048	0.11	0.21	0.38

Table 3 Mass of Elements in the 3578 MWt BWR Fuel at Different Times in the Annual Operating Cycle.

Element	Normalized Mass x 10^{-3} (g/Kwatt)			
	Beginning of Cycle	1/3 Cycle	2/3 Cycle	End of Cycle
As	0.0019	0.0023	0.0027	0.0031
Se	0.88	1.1	1.2	1.4
Br	0.34	0.41	0.48	0.55
Kr	6.0	7.2	8.4	9.6
Rb	5.7	6.8	7.9	9.0
Sr	15	18	21	24
Y	7.8	9.4	11	12
Zr	55	66	77	87
Nb	0.80	0.89	1.0	1.0
Mo	48	57	67	77
Tc	13	15	17	20
Ru	31	38	44	51
Rh	6.7	7.5	8.6	9.8
Pd	14	17	21	25
Ag	0.79	0.94	1.1	1.3
Cd	0.83	1.0	1.3	1.6
In	0.024	0.028	0.032	0.035
Sn	0.64	0.78	0.93	1.1
Sb	0.23	0.29	0.33	0.39
Te	5.5	7.8	9.1	10
I	3.1	4.0	4.6	5.3
Xe	77	93	110	120
Cs	41	49	58	66
Ba	21	25	30	34
La	19	22	26	30
Ce	43	52	60	68
Pr	17	20	23	26
Nd	55	65	77	88

Table 3 Mass of Elements in the 3578 MWt BWR at Different Times in the Annual Operating Cycle. (cont.)

Element	Normalized Mass x 10^{-3} (g/Kwatt)			
	Beginning of Cycle	1/3 Cycle	2/3 Cycle	End of Cycle
Pm	2.8	3.1	3.4	3.6
Sm	7.0	8.3	9.6	11
Eu	1.9	2.3	2.8	3.3
Gd	0.78	0.99	1.2	1.6
U	$3.8(10^4)$	$3.7(10^4)$	$3.7(10^4)$	$3.7(10^4)$
Np	5.0	8.3	9.5	11
Pu	170	200	230	250
Am	0.86	1.2	1.6	2.1
Cm	0.19	0.28	0.41	0.59

Table 4 Mass of Elements Present in the Structural Materials for the 3412 MWt PWR and the 3578 MWt BWR.

Element	Normalized Mass (g/MWt)	
	3412 MWt PWR	3578 MWT BWR
Al	2.0	1.2
St	11	17
Tt	2.7	3.6
Cr	280	360
Mn	22	34
Fe	820	1200
Co	2.4	2.3
Nt	270	260
Cu	0.34	0.14
Zr	6,700	21,000
Nb	19	1.4
Mo	10	---
Sn	110	340

Table 5 Comparison of Normalized Decay Powers for the 3412 MWt PWR and the 3578 MWt BWR.

Element	Normalized Decay Power (Watt/Kwatt)[a]	
	3412 MWt PWR	3578 MWt BWR
Rb	4.4	4.3
Sr	3.4	3.4
Y	5.3	5.1
Nb	5.0	4.9
Tc	2.3	2.3
Ru	0.53	0.54
Rh	0.62	0.64
Sn	0.44	0.44
I	4.7	4.7
Xe	2.4	2.4
Cs	4.4	4.4
La	4.4	4.4

[a] At time of reactor shutdown

Table 6 Normalized Total Decay Power as a Function of Time After Reactor Shutdown for the 3412 MWt PWR

Time After Shutdown (hr)	Normalized Power (Watt/Kwatt)	Time After Shutdown (hr)	Normalized Power (Watt/Kwatt)
0.0	59	6.0	7.0
0.0017	47	8.0	6.4
0.0033	43	10	6.2
0.0050	40	12	5.6
0.0083	37	15	5.3
0.017	33	20	5.0
0.033	28	24	4.7
0.067	25	36	4.1
0.17	20	48	3.8
0.33	18	72	3.2
1.0	12	96	2.9
1.5	11	144	2.6
2.0	10	192	2.3
4.0	7.9	240	2.1

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