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A RADIOLOGICAL CONSEQUENCE ANALYSIS WITH HEU AND LEU FUELS

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ABSTRACT

A model for estimating the radiological consequences from a hypothetical accident in HEU and LEU fueled research and test reactors is presented. Simple hand calculations based on fission product yield table inventories and non-site specific dispersion data may be adequate in many cases. However, more detailed inventories and site specific data on meteorological conditions and release rates and heights can result in substantial reductions in the dose estimates. LEU fuel gives essentially the same doses as HEU fuel. The plutonium buildup in the LEU fuel does not significantly increase the radiological consequences. The dose to the thyroid is the limiting dose.

INTRODUCTION

A common approach in Safety Analysis Reports (SARs) for research and test reactors is to assume that a hypothetical accident results in the release of some portion of the inventory of radioactive materials from the fuel to the containment/reactor building and, eventually, in the release of a portion of these materials to the atmosphere. The consequences to the surrounding population are usually evaluated in terms of estimated radiological doses from the materials released.

A uniform model and methodology based on U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guides for estimating radiological doses from hypothetical accidents in research and test reactors is presented. The method incorporates fission product inventories and dose conversion factor data to calculate doses. The model accounts for the containment/reactor building leakage and for the decay of the fission products before leakage to the atmosphere. It also includes the dispersion of airborne material by diffusion factors (χ/Q) based on release height, wind velocity, atmospheric stability, and diffusion parameters.

The following sections describe the model and provide results of sample calculations for the generic 10 MW reactor described in IAEA-TECDOC-233.[1] Various approximate methods are considered with varying degrees of conservatism. The sensitivity of the results to the isotopic content of the inventory (including plutonium) is also considered.

Calculational Model

The calculation of dose estimates may be split into two categories, an internal (inhalation) dose and an external (immersion) dose. The internal dose to organ k from isotope i may be expressed as

$$D_i^k = \chi/Q(t)\ Q_i(t)\ BR(t)\ DCF_i^k, \tag{1}$$

and the external whole body dose from isotope i may be given by

$$D_i^{Ext} = \chi/Q(t)\ Q_i(t)\ DCF_i^{Ext}, \tag{2}$$

where

$\chi/Q(t)$ is the atmospheric diffusion factor, s/m^3,
$Q_i(t)$ is the inventory of isotope i released over time t, Curies (Ci),
$BR(t)$ is the breathing rate for the receptor during the time t, m^3/s,
DCF_i^k is the dose conversion factor for organ k and isotope i, rem/Ci,, and
DCF_i^{Ext} is the dose conversion factor for external beta and gamma radiation from isotope i, rem/Ci per s/m^3.

The total dose to organ k or to whole body is then the sum over all isotopes in the inventory released in the time interval t. The diffusion factor, the inventory released, and the breathing rate may vary with time. The components in Eqs. (1) and (2) may be determined with varying degrees of detail and conservatism to fit the safety requirements of a given reactor.

The total activity of isotope i released over time τ, $Q_i(\tau)$, is obtained from the following equation, given in Ref. 2,

$$Q_i(\tau) = F_P F_B\ q_i \frac{\lambda_\ell}{\lambda_\ell + \lambda_r}\left[1 - e^{-(\lambda_\ell + \lambda_r)\tau}\right], \tag{3}$$

where

F_P = fractional release from fuel to building,
F_B = fraction remaining airborne and available to be released from the building to the atmosphere,
q_i = inventory of isotope i in reactor core at time of accident, Curies,
λ_ℓ = leak rate parameter, sec^{-1}, and
λ_r = radioactive decay constant, sec^{-1}.

Isotope release rates depend on reactor fission product inventory, paths and rates of leakage from the primary system to the containment, and paths and rates of leakage from the containment or reactor building to the atmosphere. The leakage rates depend strongly on system design and containment or reactor building design. In addition, the leak fraction of a given radionuclide depends on its chemical form. The noble gases, krypton and xenon, will be free to escape completely; solid, non-volatile fission products will remain in place. Iodine is volatile, and it is normally assumed that a significant fraction escapes.

The inventory of fission products and other radionuclides in a reactor depends on a number of parameters including the fissile material, the reactor design and materials, the operating neutron flux levels and distributions, the power history, and the fuel management scheme. In the past, many SARs for research and tests reactors have relied mainly on fission yield tables, conservative estimates of total fissions at the end of a fuel cycle, and hand calculations to determine fission product inventories. The inventory of fission products and other radionuclides in a ^{235}U-fueled reactor can be estimated by

$$q_i = 0.84^* \gamma_i P_o(1-e^{-\lambda_r T_o}), \quad (4)$$

in which

q_i is the amount of isotope i contained in the fuel after T_o, Curies,
P_o is the fuel power level, watts,
γ_i is the ^{235}U fission yield of isotope i,
λ_r is the radiological decay constant for the isotope, s^{-1}, and,
T_o is the time interval during which the fuel has been at power P_o, s.

Various computer codes are available which can provide more accurate q_i results. The ORIGEN computer code[3] provides such an improvement utilizing extensive libraries of cross sections, yields, decay constants and branches for many nuclides. A power history (consisting of any number of arbitrarily sized time steps at constant power) may be imposed to the given reactor core composition, and the code computes the concentrations of all significant nuclides as a function of irradiation time. The code then estimates the concentrations of nuclides remaining after specific decay times (as supplied by the user) following irradiation.

With holdup in the containment, this q_i source may not be the only source of the i^{th} radionuclide. Many nuclides are produced by precursors at some time after the time of release. The precursor source to the iodine isotopes is approximated in the following fashion: The inventories of the precursors for each iodine isotope are simply summed, e.g.

$$^{131}In \rightarrow {}^{131}Sn \rightarrow {}^{131}Sb \rightarrow {}^{131m}Te \rightarrow {}^{131}Te \xrightarrow{\lambda} {}^{131}I,$$

where the In, Sn, Sb, and ^{m}Te are assumed to instantly decay to Te to form the iodine precursor source, q_{po}, and

$$q_p(\tau) = q_{po}\ (1 - e^{-\lambda\tau}), \quad (5)$$

where λ is the decay constant for Te and τ is the elapsed time after release. No credit is taken for holdup in decay of the precursors to Te. This contribution is then lumped in with the fission product inventory of iodine.

Diffusion factors (χ/Q) are determined based on release height, wind velocity, air stability, and the distance from release to receptor locations. The method used is credited to O. G. Sutton and is outlined in Ref. 4. Tables

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* $\frac{3.1 \times 10^{10} \text{ fissions/s/W}}{3.7 \times 10^{10} \text{ Disintegrations/s/Ci}} \approx 0.84$ Ci/W

and curves of χ/Q factors, as functions of these variables, are compiled in USNRC Regulatory Guides 1.3[5] and 1.4.[6] A more recent guide for site specific χ/Q values is provided in USNRC Regulatory Guide 1.145.[7] Values used in the remainder of this report were taken from those tables. The breathing rate data for man in these analyses are also taken from USNRC Regulatory Guides 1.3 and 1.4. The first eight (8) hours of the exposure are taken to be an active period with a breathing rate of 3.47×10^{-4} m^3/s. The interval from eight to twenty four (8-24) hours is considered to be a resting period with a rate of 1.75×10^{-4} m^3/s. For time periods greater than one day, a breathing rate of 2.32×10^{-4} m^3/s is used.

DCF data are available from a number of sources for external whole body immersion and for internal whole body, bone, lung, thyroid, etc. An extensive tabulation of DCFs for a large selection of isotopes and an assortment of internal organs is available in Ref. 8. These data are used in subsequent calculations.

APPLICATION OF METHODS

Some applications of the models and methods are provided for illustration and for the comparison of LEU and HEU fuel. Guidelines of the USNRC 10 CFR part 100[9] are used for this study, consisting of values of 25 rem for whole body and 300 rem for thyroid exposure, along with values of 150 rem for bone and 75 rem for lung doses. These values should be considered only as reference values for these evaluations.

Ideally one would like to use a simple yet conservative method which would demonstrate that even with conservative assumptions the reactor would not present a significant risk to the public. In some cases, however, these simple methods may be too conservative and it will be necessary to do more detailed calculations with more complex models. The sensitivity of some of the options are considered in the following discussion.

For illustrative purposes the generic 10 MW reactor has been selected for dose computations. No attempt will be made to postulate the accident conditions that might lead to the assumed releases for the cases considered. An attempt is made, however, to choose realistic examples that can be scaled to other conditions. In all cases only the fuel element with the highest power level at the beginning of equilibrium cycle is considered, and in most cases the "worst case" inventory for 100 full power days of irradiation is used as shown in Table 1. The HEU case has 280 g of ^{235}U in the element, and the LEU element contains 390 g of ^{235}U. Although only the inventory for a single element is considered, the doses computed can be scaled up or down as more or less damage to the fuel is postulated. This single element can also be thought of as representing partial releases from several elements.

The inventory of fission products is strongly influenced by the irradiation history of the reactor. The maximum inventory for a given isotope may not be at the end of the maximm irradiation time for the fuel. The duty cycle of the reactor can also have a substantial effect on this inventory. Table 1 illustrates the inventory of a selection of isotopes under varying irradiation conditions. The approximate half lives of these isotopes are also provided. The table shows many of the kryton and xenon isotopes peaking early in the irradiation history (~4 days), while the iodine isotopes have a peak inventory at about 100 full power days of irradiation. The inventories corresponding to

Table 1. Volatile Fission-Product Activities after Full-Power-Operation Runs of 1) 5d at 8h/d; 2) 4d; 3) 21d; 4) 100d, and 5) 300d

Nuclide	$T_{1/2}$ (Knolls Chart) 1977	ORIGEN-Calculated Activities at shutdown, 10^3 Curies 5-d,8h/d After 5d 0.74 MWD	5-d,8h/d After 6 weeks 4.34 MWD	4d 1.77 MWD	21d 9.31 MWD	100d 44.34 MWD	300d 133.0 MWD
^{83m}Kr	1.86 h	0.0200	0.0200	2.80	2.77	2.74	2.38
^{85m}Kr	4.48 h	0.0500	0.0496	6.87	6.84	6.77	5.84
^{85}Kr	10.72 y	1.08×10^{-5}	3.23×10^{-5}	9.71×10^{-4}	0.00539	0.0258	0.0744
^{87}Kr	76 m	0.132	0.132	13.4	13.3	13.1	11.3
^{88}Kr	2.84 h	0.163	0.163	18.8	18.7	18.5	15.9
^{89}Kr	3.15 m	0.246	0.246	24.6	24.4	24.1	20.6
^{131m}Xe	11.92 d	3.72×10^{-5}	2.28×10^{-4}	0.00317	0.0443	0.103	0.103
^{133m}Xe	2.19 d	0.00692	0.00281	0.577	1.04	1.04	0.896
^{133}Xe	5.25 d	0.125	0.0898	10.4	32.7	35.3	31.0
^{135m}Xe	15.3 m	0.0406	0.0389	5.76	5.71	5.68	5.15
^{135}Xe	9.09 h	0.0232	0.0211	3.83	3.62	2.84	0.742
^{138}Xe	14.2 m	0.329	0.329	32.8	32.6	32.3	28.0
^{131}I	8.041 d	0.0442	0.0388	4.08	12.4	14.8	14.1
^{132}I	2.29 h	0.131	0.0636	13.1	22.2	22.3	20.4
^{133}I	20.8 h	0.287	0.143	33.8	35.3	35.1	30.6
^{134}I	52.6 m	0.355	0.355	40.1	39.8	39.5	34.3
^{135}I	6.585 h	0.218	0.206	33.3	33.1	32.8	28.6

a five day week with eight hours per day of operation are shown after one week and after six (6) weeks (~10 days at full power total). These inventories are substantially below even the four day continuous irradiation at full power case. Accurately accounting for the operating history of the reactor can result in an important reduction in the peak inventory for a reactor.

Now that the inventory is established, further assumptions must be made regarding the release to the containment, the fraction available for release to the atmosphere, and the leakage rate to the atmosphere. One common set of assumptions is to assume that 100% of the noble gases, 50% of the halogens, and 1% of the solids in the inventory are released into the containment, and to assume that only 50% of the halogens remain available for release to the atmosphere. A leakage rate to the atmosphere of 1%/day is assumed for most of the sample cases. The variation in leakage rate with the change in pressure within the building is not included. No credit is taken for filtering, washdown, or other engineered safety features that might be included in some designs.

After release to the atmosphere, other assumptions may be made in order to describe the dispersion of the plume in the atmosphere before reaching the receptor site. Without site specific data on wind speed and atmospheric stability conditions, the conditions suggested in USNRC Regulatory Guide 1.4 for a ground level release may be assumed. This χ/Q data assumes a wind speed of 1 m/s and Pasquill type F (stable) atmospheric conditions for the first eight hours following release. χ/Q values at 500, 1000, and 5000 m distances are listed:

τ	BR, m^3/s	χ/Q, s/m^3 500 m	1000 m	5000 m
0-8h	3.47×10^{-4}	2.08×10^{-3}	6.40×10^{-4}	6.00×10^{-5}
8-24h	1.75×10^{-4}	4.95×10^{-4}	1.50×10^{-4}	1.20×10^{-5}
1-4d	2.32×10^{-4}	1.72×10^{-4}	5.30×10^{-5}	4.10×10^{-6}
4-30d	2.32×10^{-4}	4.00×10^{-5}	1.20×10^{-5}	8.80×10^{-7}

Doses for the generic 10 MW reactor with HEU and LEU fuel are compared in Table 2. The doses for release from the highest power element after 100 full power days at a 500 m site boundary are given at 2 hours and after 30 days. When the differences in the power levels (the highest power element in the LEU case is ~2% higher than the HEU) are discounted, the LEU and HEU doses are very similar. The LEU inhalation dose for bone and whole body are slightly higher due to the presence of more plutonium. This difference is considered in more detail later. In either case the largest dose is to the thyroid, and 5 rem after 2 hours or 27 rem after 30 days are both well below the 300 rem reference value for the thyroid.

The isotopes that make significant contributions to the dose in specific organs are tabulated in Table 3 for the HEU case. Only the iodines contribute to the thyroid dose with 66% of the dose contributed by ^{131}I. The iodine contributions from precursors is ~5% of the total for this case. The bone dose contributors are dominated by ^{144}Ce, ^{89}Sr, and ^{91}Y. The lung and internal whole body doses are similar and each show ^{144}Ce and ^{131}I as major contributors. The largest contributors to the external whole body dose are ^{132}I and ^{88}Kr. A large percentage of the doses are contributed by a limited number of isotopes. This makes the use of source data generated from fission product yield tables a reasonable approximation. The inventory from experiments and the operating history of the reactor may shift the importance of the various isotopes to dose.

Yield Table Approximation

The inventory of fission products in the fuel can be approximated by using Eq. (4). If adequate fission product yield data are available and a constant reactor power is an acceptable approximation for the irradiation, then data for only a few selected isotopes should provide a reasonable estimate for the release source. For long irradiation times the exponential term in Eq. (4) can be neglected and the resulting inventory is the maximum corresponding to an infinite irradiation time. Such an estimate should give a conservative approximation for the source.

The yield fractions and infinite irradiation fission product inventory data for selected isotopes are provided in Table 4 based on yield tables from Ref. 10. Only the first 29 isotopes are considered as important contributors to the direct inventory based on the results from the previous section. The last 19 isotopes of In, Sn, Sb, and Te are included primarily to account for iodine precursor contributions. Equations (1)-(3) and (5) may be used to estimate the dose directly without the need for computer codes. Dose estimates based on these cumulative yield data for the HEU core are compared with dose estimates from ORIGEN inventories after 100 and 300 FPD of irradiation in Table 5. These cases show the trends with increased irradiation time. The bone and lung doses increase slightly along with the internal whole body dose. The external whole body dose decreases slightly. The thyroid dose dominates the doses received and decreases slightly as shown. Two cases with cumulative yield data are shown in the table. The first case for infinite irradiation time uses the source data from Table 4 directly. The second case limits the irradiation time to 100 days, and the isotopic source is reduced by the factor $(1-e^{-\lambda_i T_0})$. The cumulative yield results with a 100 day irradiation time agree

Table 2. Doses at 500 m Site Boundary for 10 MW Generic Reactor 100 FPD Peak Element With HEU and LEU Fuel*

	HEU	LEU
Bone Dose, rem		
2 h	1.384-1	1.504-1
30 d	1.305	1.425
Lung Dose, rem		
2 h	1.987-1	2.031-1
30 d	1.600	1.634
Thyroid Dose, rem		
2 h	4.419	4.519
30 d	26.38	26.98
Whole Body (internal), rem		
2 h	1.549-2	1.604-2
30 d	1.171-1	1.217-1
Whole Body (external), rem		
2 h	5.305-2	5.461-2
30 d	1.812-1	1.874-1
Burnup, MWD	44.34	45.11

*Assuming 100% of noble gases, 25% of halogens, and 1% of other are available for release from the containment, and a leakage rate from the containment of 1%/day (using Regulatory Guide 1.4 χ/Q values).

Table 3. Isotopic Contributions to the Thyroid, Bone, and Lung Doses, the Whole Body Internal (Inhalation) Dose, and the Whole Body External (Immersion) Dose.

Isotope	Dose, %	Isotope	Dose, %	Isotope	Dose, %
		Thyroid Dose			
I-131	66.	I-135	4.8	I-134	0.4
I-133	25.	I-132	3.5		
		Bone Dose			
Ce-144	25.	Zr-95	4.3	Pu-239	0.5
Sr-89	24.	Pr-143	1.9	Nd-147	0.5
Y-91	23.	Ce-141	1.6	Pm-147	0.4
Ba-140	7.8	Pu-238	1.1		
Sr-90	7.8	Nb-95	0.6		
		Lung Dose			
Ce-144	17.	Ce-141	2.8	I-132	1.0
I-131	16.	Ru-103	2.6	La-140	0.8
Y-91	12.	Pr-143	2.4	Nd-147	0.7
Zr-95	11.	Nb-95	2.2	Sr-90	0.7
Ba-140	11.	I-135	1.3	Mo-99	0.6
Sr-89	8.5	Te-132	1.1	Ce-143	0.4
I-133	5.4	Ru-106	1.1		
		Whole Body Internal (Inhalation) Dose			
I-131	27.	Y-91	6.7	Pr-143	1.0
Ce-144	15.	I-135	2.5	Te-132	1.0
Zr-95	13.	Nb-95	2.1	La-140	0.5
I-133	8.4	I-132	1.9	Mo-99	0.5
Sr-89	7.5	Sr-90	1.7	Nd-147	0.4
Ba-140	7.0	Ce-141	1.5		
		Whole Body External (Immersion) Dose			
I-131	27.	I-134	6.3	Kr-85m	1.4
Kr-88	15.	Kr-87	4.9	La-140	1.1
Xe-133	11.	I-131	4.1	Xe-137	0.6
I-135	10.	Xe-135	1.7	Zr-95	0.5
I-133	9.8	Xe-138	1.6	Ba-140	0.3

Table 4. Isotopic Yields and Source Data

Isotope, i	Yield, Y_i	Decay Const., s^{-1}	$Q_i^*/MW \times 10^{-6}$	Precusor, i	Yield, Y_i
Kr-85m	0.01289	4.30-05	1.084	In-131	0.00029
Kr-87	0.02556	1.50-04	2.147	Sn-131	0.00932
Kr-88	0.03541	6.90-05	2.974	Sb-131	0.02577
Sr-89	0.04822	1.60-07	4.050	Te-131m	0.00391
Sr-90	0.05772	7.58-10	4.848	Te-131	0.02567
Y-91	0.05931	1.70-07	4.982	Sn-132	0.00576
Zr-95	0.06519	1.20-07	5.476	Sb-132m	0.01165
Nb-95	0.06520	2.30-07	5.477	Sb-132	0.01633
Mo-99	0.06074	2.90-06	5.102	Sn-133	0.00180
Ru-103	0.03029	2.00-07	2.544	Sb-133	0.02208
Ru-106	0.00402	2.17-08	0.338	Te-133m	0.04054
I-131	0.02892	1.00-06	2.429	Te-133	0.02999
Te-132	0.04297	2.50-06	3.609	I-133m	0.00153
I-132	0.04313	8.40-05	3.623	Sb-134m	0.00228
I-133	0.06693	9.30-06	5.622	Sb-134	0.00468
Xe-133	0.06696	1.50-06	5.625	Te-134	0.06932
I-134	0.07794	2.20-04	6.547	I-134m	0.00365
I-135	0.06293	2.90-05	5.286	Sb-135	0.00148
Xe-135	0.06535	2.10-05	5.489	Te-135	0.03247
Xe-137	0.06122	3.00-03	5.142		
Xe-138	0.06238	8.10-04	5.240		
Ba-140	0.06202	6.30-07	5.210		
La-140	0.06207	4.80-06	5.214		
Ce-141	0.05787	2.50-07	4.861		
Ce-143	0.05963	5.80-06	5.009		
Pr-143	0.05963	5.90-07	5.009		
Ce-144	0.05493	2.82-08	4.614		
Nd-147	0.02270	7.30-07	1.907		
Pm-147	0.02270	8.37-09	1.907		

*$Q_i = 0.84\ Y_i \times 10^6$, Ci/MW (infinite irradiation time).

Table 5. Comparison of Doses with Inventories from ORIGEN Code vs. Yield Tables

		Dose at 500 m Site Boundary, rem*				
		Inhalation			Whole Body	
Case	Time	Bone	Lung	Thyroid	Internal (Inhalation)	External (Immersion)
100 FPD	2 h	0.1384	0.1987	4.419	1.549-02	5.305-02
ORIGEN	30 d	1.305	1.600	26.38	0.1171	0.1812
300 FPD	2 h	0.3121	0.2726	4.305	2.217-02	5.117-02
ORIGEN	30 d	3.065	2.338	25.74	0.1849	0.1770
Cumulative Yields Infinite	2 h	1.712	0.4859	4.463	5.230-02	5.730-02
	30 d	17.47	4.541	26.70	0.1943	0.2030
Cumulative Yields 100 d	2 h	0.1373	0.1999	4.463	1.567-02	5.718-02
	30 d	1.309	1.632	26.70	0.1195	0.2015

*Using peak element, 0.44342 MW, in 10 MW HEU generic reactor with a 1.0%/d leak rate and the release of 100% of Noble gases, 25% of halogens, and 1% of all other to containment.

remarkably well with the 100 FPD ORIGEN results, while the infinite irradiation time yield results substantially overestimate the bone dose. The differences in the cumulative yield cases are largely due to the difference in the ^{90}Sr inventory. At infinite irradiation times the ^{90}Sr contribution is ~84% of the total bone dose, while for a 100 day irradiation only ~7% of the bone dose is due to ^{90}Sr buildup. The infinite irradiation cumulative yield method can provide a conservative estimate for the dose. The yield table data do not include the buildup of actinides. The importance of plutonium buildup in the LEU fuel is considered in another section.

Site Specific Data and Sensitivity of Dose Estimates

The cumulative yield method provides a convenient way of presenting the dose per unit power. The results may be generalized to any power level. The dose levels correspond to a single element at the highest power again with assumed release fractions of 100% for the noble gases, 25% for the halogens, and 1% for all others. The levels may be scaled to fractional or multiple element cases. The term site boundary has been purposely used in the previous comparisons so that the results may be interpreted either as doses at the exclusion area boundary or at the outer boundary of the low population zone for the reactor using USNRC definitions. The sensitivity of these dose data to changes in distance from the source, leakage rate, and other accident or site specific conditions will now be considered.

Although there are no site specific meteorological data for the generic reactor, the changes in dose that can be expected by using site specific data can be indicated by some examples. The dose received is directly proportional to the χ/Q values for the specific atmospheric conditions. Thus, the comparison can be limited to χ/Q values. The dispersion models from Regulatory Guide 1.145[7] will be used for this comparison with a 0-2 hour time interval and a distance from the source of 500 m. Data for both a ground-level release and release from an effective stack height of 50 m are presented in Table 6. The non-site specific data are also shown for comparison. For a ground-level release under Pasquill class F conditions, the diffusion factor value is simply reduced to 1/4 if the wind speed is really 4 m/s at the site rather than 1 m/s, as assumed for the non-site specific case. This value is reduced by a full order of magnitude with class C conditions and a wind speed of 8 m/s. With releases from a stack of 50 m, this factor can be further reduced. A range of χ/Q values are shown for Pasquill classes A-F and a wind speed of 8 m/s. The values increase from unstable to more stable atmospheric conditions until class D, after which the values decrease. The peak location for the class F curve is well beyond 500 m. The class C curve has a peak near the 500 m location. These data show the same trends suggested by the curves in Regulatory Guide 1.3.[5] The corresponding doses that would be determined with site specific data can be substantially reduced compared to the non-site specific data.

The variation in dose/MW to the thyroid and whole body with leakage rate from the containment and distance from the source of release is illustrated in Fig. 1. These data are for a ground-level release and for non-site specific conditions. The doses vary strongly with both the leakage rate and distance from the source.

Table 6. χ/Q Data with Site Specific Conditions.

	χ/Q, s/m^3
Ground-level Release	
Non-site Specific	
Pasquill Class F, 1 m/s	2.1 - 03*
With Wake Effect Correction	7.0 - 04
Site Specific	
Class F (stable), 4 m/s	1.8 - 04
Class C (slightly unstable), 8 m/s	1.8 - 05
Release from 50 m Stack	
Non-site Specific	
With Fumigation	4.0 - 04
Without Fumigation	5.9 - 05
Site Specific	
Class A, 8 m/s	2.6 - 06
Class B	6.0 - 06
Class C	7.5 - 06
Class D	1.3 - 06
Class E	1.1 - 07
Class F	~0.0

*Value used in previous calculations at 500 m.

DISCLAIMER

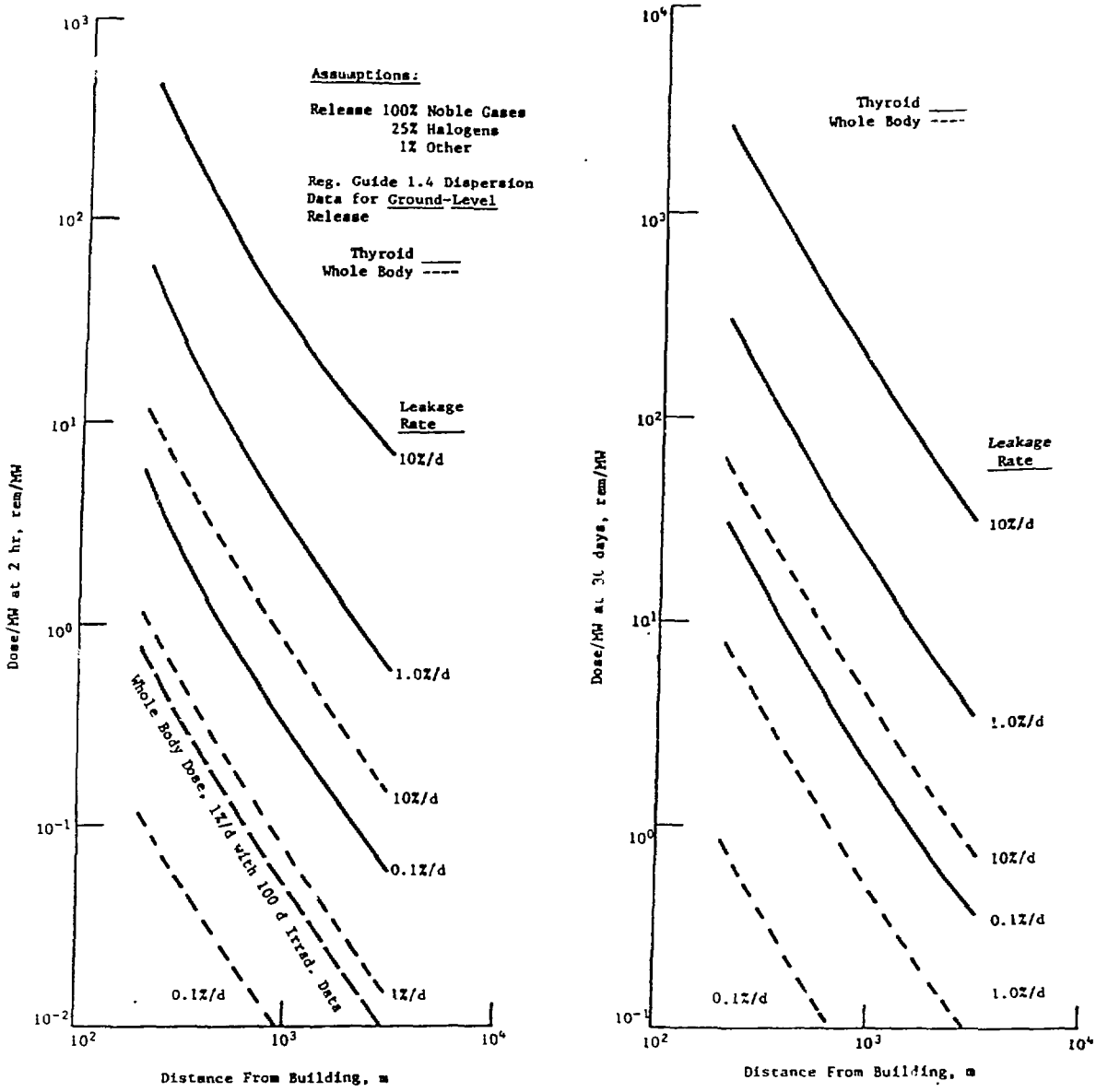

Fig. 1. Dose/MW Variation With Leakage Rate and Location

The influence of building wake effects at ground level and of various stack heights are illustrated in Fig. 2. The thyroid dose/MW is shown in Fig. 2 for ground-level releases with and without the wake correction factor and for releases at effective stack heights of 50, 100, and 150 m. The wake correction factor near the building reduces the dose to 1/3. The stack releases assume full fumigation conditions for the first one-half hour after release. The effective stack heights are also assumed to be more than two and one half times the height of adjacent solid structures. The above ground-level release results in a further reduction in the dose. The maximum dose after 30 days does not occur at the nearest distance from the source but at some distance from the release point. The location of the peak dose is determined by the χ/Q for each stack height.

LEU Fuel and Doses with Plutonium Buildup

With HEU fuel the content of ^{238}U is small and the production of plutonium and other actinides is negligible. The LEU fuel will have a content of ~80% ^{238}U and a greater potential for the buildup of plutonium with burnup. While the LEU fueled reactors can be expected to show an increase in dose due to this plutonium buildup, the consequences may still be insignificant.

In order to compare results with HEU and LEU fuel, the 10 MW generic reactor with LEU fuel is used again with the same release fractions, non-site specific conditions, and site boundary assumptions as before. Table 7 shows the buildup of ^{239}Pu and ^{241}Pu in grams as predicted by the ORIGEN code and the corresponding doses to bone, lung, and thyroid at the 500 m site boundary after two hours and after 30 days for irradiations times up to 500 FPD (66 atom percent burnup in ^{235}U). The 100 FPD data is identical to that for the LEU case shown in Table 2. In that comparison the bone dose for the LEU core is slightly higher than the HEU core. The bone and lung doses increase with increased irradiation time and burnup, while the thyroid shows a peak for 200 FPD and decreases slightly at higher burnups. At 500 FPD and 66% burnup, the fuel is well beyond the normal burnup limits for the fuel, and the bone dose is still substantially below that for the thyroid.

It is also interesting to look at the changes in importance of the major contributors with burnup as shown in Fig. 3. After 100 FPD, Ce-144 at 23% dominates with Pu-241 contributing less than 2% of the dose. Ce-144 continues to have the highest percent through 400 FPD with Pu-241 steadily increasing, and at 500 FPD the Pu-241 becomes the largest contributor at 23%. The Pu-238 contribution shows a similar behavior, while both Pu-239 and Pu-240 reach peak values and begin to decrease with burnup.

In an effort to put the effects of plutonium on dose in perspective, it is useful to compare the peak doses by organ to some reference limits:

	Dose, rem		
Organ	Peak	Reference	Margin
Thyroid	27.1	300	11.1
Bone	6.15	150	24.4
Lung	2.80	75	26.8

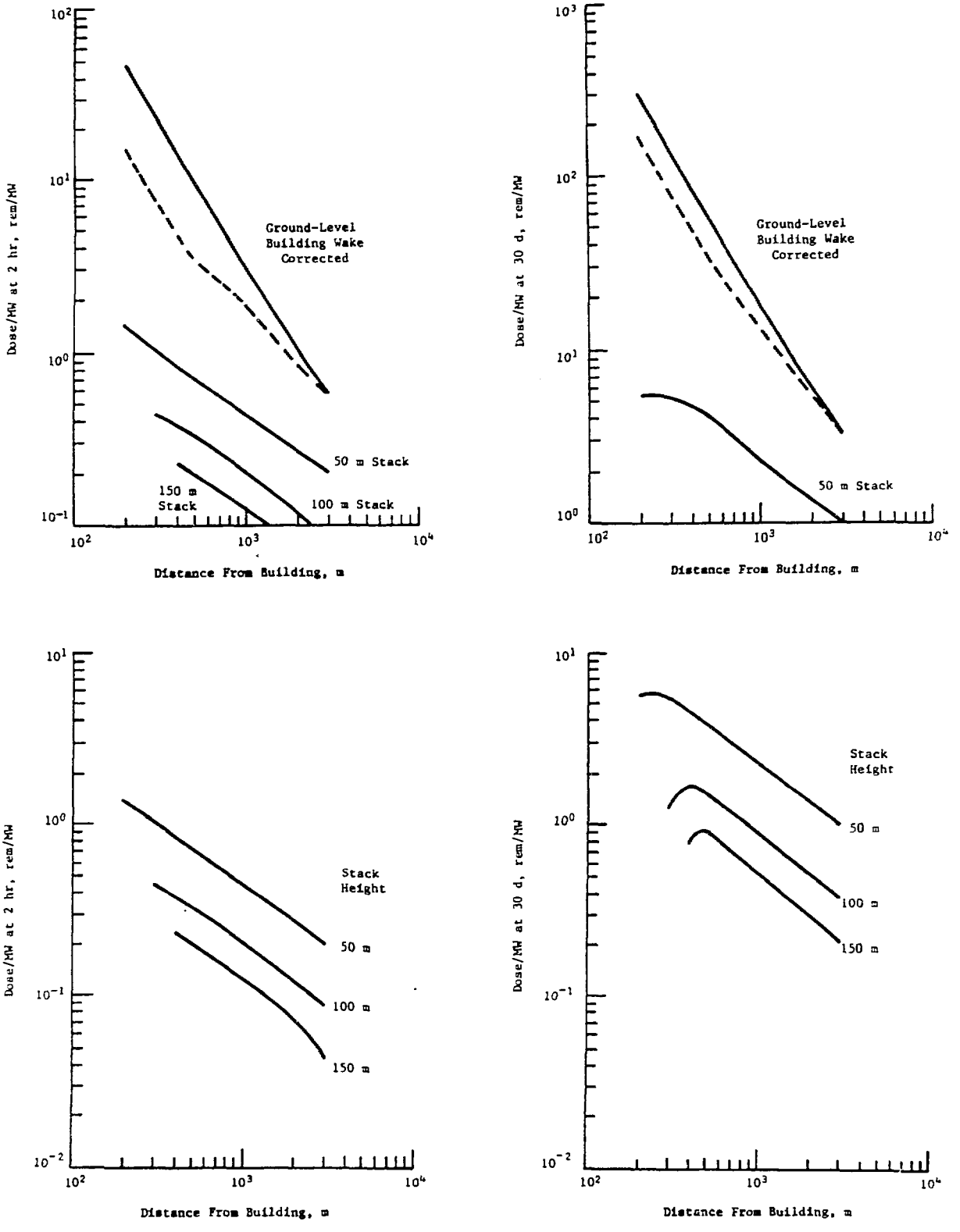

Fig. 2. Thyroid Dose/MW Variation with Stack Height

Table 7. 10 MW LEU Generic Reactor - Plutonium Buildup and Dose

	390 g ^{235}U LEU Peak Power (0.4511 MW) Element							
Irrad. Time, d	Atom % Burnup ^{235}U	Mass, g				Dose, rem at 2 h (30 d)		
		Pu-238	Pu-239	Pu-240	Pu-241	Bone	Lung	Thyroid
100	14	-	3.61	0.18	0.02	0.150 (1.42)	0.203 (1.63)	4.52 (27.0)
200	28	0.01	6.71	0.66	0.14	0.233 (2.25)	0.250 (2.10)	4.53 (27.1)
300	41	0.04	9.19	1.36	0.46	0.319 (3.14)	0.278 (2.37)	4.48 (26.8)
400	54	0.10	11.0	2.20	1.03	0.440 (4.37)	0.300 (2.60)	4.48 (26.8)
500	66	0.23	12.3	3.802	1.84	0.613 (6.15)	0.319 (2.80)	4.43 (26.7)

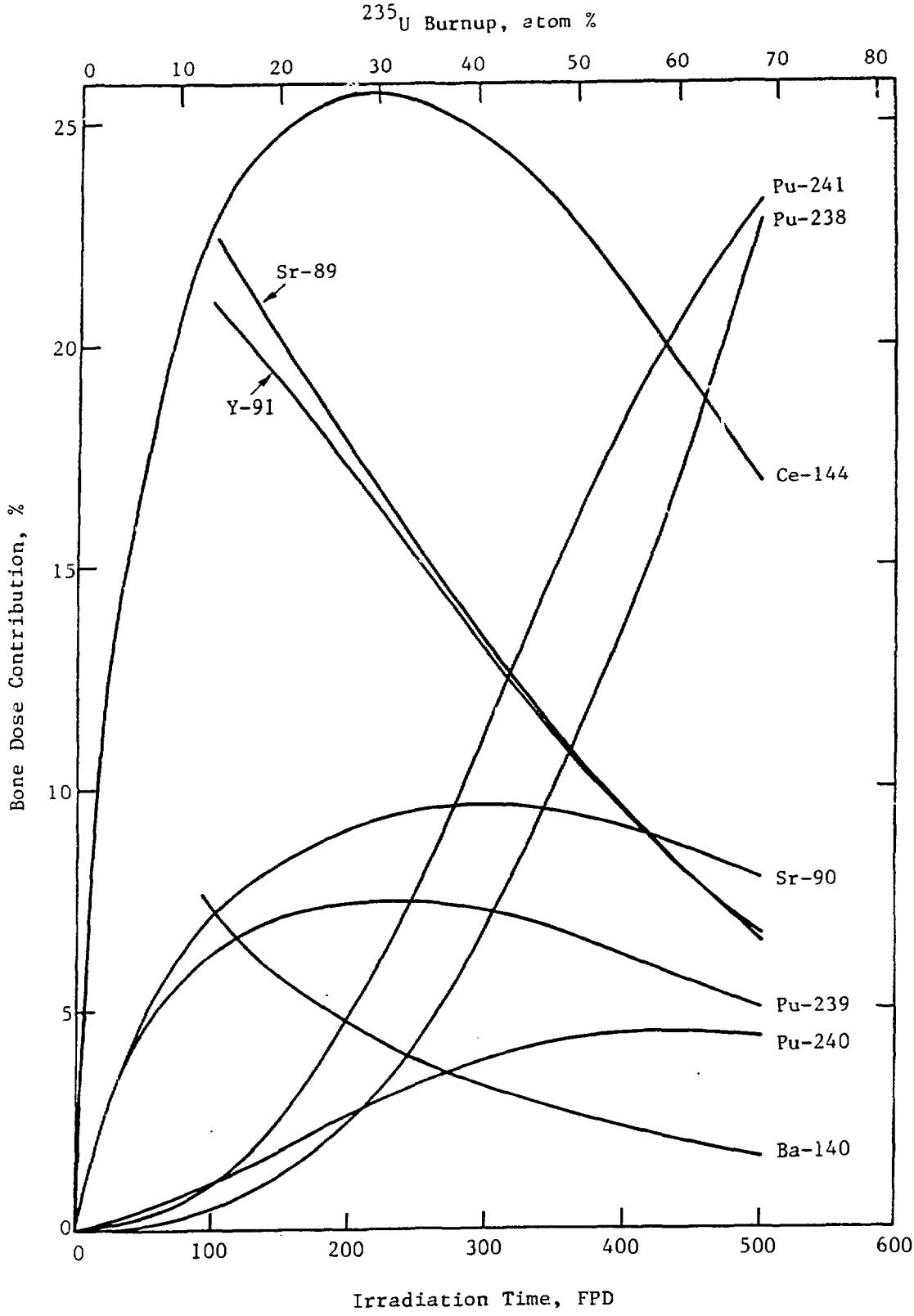

Fig. 3. Variation in Isotopic Contribution to Bone Dose With Irradiation Time for LEU Fuel

The thyroid margin is more than a factor of two lower than either the bone or lung margins even with an unrealistically high burnup. The thyroid dose is still the limiting factor in the LEU core. However, the thyroid dose may not be the limiting dose under all operating conditions and experiment inventories.

SUMMARY

Models and methods with varying degrees of detail and conservatism have been considered. If the safety requirements for a reactor can be satisfied by using a simple but conservative model, there is no need for more detailed models or methods. At the same time it should be recognized that more detailed evaluations can substantially reduce the dose estimates for a given accident.

A detailed inventory of fission products and actinides in the reactor fuel may be useful. The operating history (duty factor, fuel cycle, etc.) can substantially affect the radiological consequences. Accounting for the at power and down-time history can significantly reduce the source term for the release.

In some applications fission product yields can be used to estimate the reactor inventory to sufficient accuracy. Only a limited number of isotopes are important contributors to the dose, and most can be neglected. Assuming an infinite irradiation time for the fuel provides a conservative estimate. This method does not normally account for the production of actinides, but this component is not essential to the more limiting thyroid dose.

The assumed leakage rate from the containment and the assumed local meteorological conditions play a strong role in estimating the radiological consequences of an accident. Credits for elevated releases and site specific wind speed and atmospheric conditions can result in order of magnitude reductions in the dose estimates.

No attempt has been made to assess the effectiveness of any filtering system, washdown spray system, or other engineered safety features that may be included in some reactor designs. Such systems may provide additional safety margins. These must be assessed individually. Also, no attempt has been made to alter the release fractions assumed through out this section. However, there is growing evidence that the iodine release fraction could be substantially reduced. The model used here does not account for the finite transit time from the point of release to the receptor site, for the finite passage time of the radioactive cloud past the receptor, or for any evacuation procedures that might limit the exposure time. The full evaluation of the radiological consequences must be tailored to the characteristics and requirements of each individual reactor.

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