A. STATISTICAL TABLES AND PROCEDURES

A.1 Normal Distribution

Table A.1 Cumulative Normal Distribution Function Φ(z)

z	0.00	0.01	0.02	0.03	0.04	0.05	0.06	0.07	0.08	0.09
0.00	0.5000	0.5040	0.5080	0.5120	0.5160	0.5199	0.5239	0.5279	0.5319	0.5359
0.10	0.5398	0.5438	0.5478	0.5517	0.5557	0.5596	0.5636	0.5674	0.5714	0.5753
0.20	0.5793	0.5832	0.5871	0.5910	0.5948	0.5987	0.6026	0.6064	0.6103	0.6141
0.30	0.6179	0.6217	0.6255	0.6293	0.6331	0.6368	0.6406	0.6443	0.6480	0.6517
0.40	0.6554	0.6591	0.6628	0.6664	0.6700	0.6736	0.6772	0.6808	0.6844	0.6879
0.50	0.6915	0.6950	0.6985	0.7019	0.7054	0.7088	0.7123	0.7157	0.7190	0.7224
0.60	0.7257	0.7291	0.7324	0.7357	0.7389	0.7422	0.7454	0.7486	0.7517	0.7549
0.70	0.7580	0.7611	0.7642	0.7673	0.7704	0.7734	0.7764	0.7794	0.7823	0.7852
0.80	0.7881	0.7910	0.7939	0.7967	0.7995	0.8023	0.8051	0.8078	0.8106	0.8133
0.90	0.8159	0.8186	0.8212	0.8238	0.8264	0.8289	0.6315	0.8340	0.8365	0.8389
1.00	0.8413	0.8438	0.8461	0.8485	0.8508	0.8531	0.8554	0.8577	0.8599	0.8621
1.10	0.8643	0.8665	0.8686	0.8708	0.8729	0.8749	0.8770	0.8790	0.8810	0.8830
1.20	0.8849	0.8869	0.8888	0.8907	0.8925	0.8944	0.8962	0.8980	0.8997	0.9015
1.30	0.9032	0.9049	0.9066	0.9082	0.9099	0.9115	0.9131	0.9147	0.9162	0.9177
1.40	0.9192	0.9207	0.9222	0.9236	0.9251	0.9265	0.9279	0.9292	0.9306	0.9319
1.50	0.9332	0.9345	0.9357	0.9370	0.9382	0.9394	0.9406	0.9418	0.9429	0.9441
1.60	0.9452	0.9463	0.9474	0.9484	0.9495	0.9505	0.9515	0.9525	0.9535	0.9545
1.70	0.9554	0.9564	0.9573	0.9582	0.9591	0.9599	0.9608	0.9616	0.9625	0.9633
1.80	0.9641	0.9649	0.9656	0.9664	0.9671	0.9678	0.9686	0.9693	0.9699	0.9706
1.90	0.9713	0.9719	0.9726	0.9732	0.9738	0.9744	0.9750	0.9756	0.9761	0.9767
2.00	0.9772	0.9778	0.9783	0.9788	0.9793	0.9798	0.9803	0.9808	0.9812	0.9817
2.10	0.9821	0.9826	0.9830	0.9834	0.9838	0.9842	0.9846	0.9850	0.9854	0.9857
2.20	0.9861	0.9864	0.9868	0.9871	0.9875	0.9878	0.9881	0.9884	0.9887	0.9890
2.30	0.9893	0.9896	0.9898	0.9901	0.9904	0.9906	0.9909	0.9911	0.9913	0.9916
2.40	0.9918	0.9920	0.9922	0.9925	0.9927	0.9929	0.9931	0.9932	0.9934	0.9936
2.50	0.9938	0.9940	0.9941	0.9943	0.9945	0.9946	0.9948	0.9949	0.9951	0.9952
2.60	0.9953	0.9955	0.9956	0.9957	0.9959	0.9960	0.9961	0.9962	0.9963	0.9964
2.70	0.9965	0.9966	0.9967	0.9968	0.9969	0.9970	0.9971	0.9972	0.9973	0.9974
2.80	0.9974	0.9975	0.9976	0.9977	0.9977	0.9978	0.9979	0.9979	0.9980	0.9981
2.90	0.9981	0.9982	0.9982	0.9983	0.9984	0.9984	0.9985	0.9985	0.9986	0.9986
3.00	0.9987	0.9987	0.9987	0.9988	0.9988	0.9989	0.9989	0.9989	0.9990	0.9990
3.10	0.9990	0.9991	0.9991	0.9991	0.9992	0.9992	0.9992	0.9992	0.9993	0.9993
3.20	0.9993	0.9993	0.9994	0.9994	0.9994	0.9994	0.9994	0.9995	0.9995	0.9995
3.30	0.9995	0.9995	0.9995	0.9996	0.9996	0.9996	0.9996	0.9996	0.9996	0.9997
3.40	0.9997	0.9997	0.9997	0.9997	0.9997	0.9997	0.9997	0.9997	0.9997	0.9998

Negative values of z can be obtained from the relationship $\Phi(-z) = 1 - \Phi(z)$

A.2 Sample Sizes for Statistical Tests

Table A.2a Sample Sizes for Sign Test

(Number of measurements to be performed in each survey unit)

	(α,β) or (β,α)														
	0.01	**0.01**	**0.01**	**0.01**	**0.01**	**0.025**	**0.025**	**0.025**	**0.025**	**0.05**	**0.05**	**0.05**	**0.1**	**0.1**	**0.25**
Δ/σ	**0.01**	**0.025**	**0.05**	**0.1**	**0.25**	**0.025**	**0.05**	**0.1**	**0.25**	**0.05**	**0.1**	**0.25**	**0.1**	**0.25**	**0.25**
0.1	4,095	3,476	2,984	2,463	1,704	2,907	2,459	1,989	1,313	2,048	1,620	1,018	1,244	725	345
0.2	1,035	879	754	623	431	735	622	503	333	518	410	258	315	184	88
0.3	468	398	341	282	195	333	281	227	150	234	185	117	143	83	40
0.4	270	230	197	162	113	192	162	131	87	136	107	68	82	48	23
0.5	178	152	130	107	75	126	107	87	58	89	71	45	54	33	16
0.6	129	110	94	77	54	92	77	63	42	65	52	33	40	23	11
0.7	99	83	72	59	41	70	59	48	33	50	40	26	30	18	9
0.8	80	68	58	48	34	57	48	39	26	40	32	21	24	15	8
0.9	66	57	48	40	28	47	40	33	22	34	27	17	21	12	6
1.0	57	48	41	34	24	40	34	28	18	29	23	15	18	11	5
1.1	50	42	36	30	21	35	30	24	17	26	21	14	16	10	5
1.2	45	38	33	27	20	32	27	22	15	23	18	12	15	9	5
1.3	41	35	30	26	17	29	24	21	14	21	17	11	14	8	4
1.4	38	33	28	23	16	27	23	18	12	20	16	10	12	8	4
1.5	35	30	27	22	15	26	22	17	12	18	15	10	11	8	4
1.6	34	29	24	21	15	24	21	17	11	17	14	9	11	6	4
1.7	33	28	24	20	14	23	20	16	11	17	14	9	10	6	4
1.8	32	27	23	20	14	22	20	16	11	16	12	9	10	6	4
1.9	30	26	22	18	14	22	18	15	10	16	12	9	10	6	4
2.0	29	26	22	18	12	21	18	15	10	15	12	8	10	6	3
2.5	28	23	21	17	12	20	17	14	10	15	11	8	9	5	3
3.0	27	23	20	17	12	20	17	14	9	14	11	8	9	5	3

Table A.2b Sample Sizes for Wilcoxon Rank Sum Test

(Number of measurements to be performed on the reference material and for each survey unit)

	(α,β) or (β,α)														
	0.01	**0.01**	**0.01**	**0.01**	**0.01**	**0.025**	**0.025**	**0.025**	**0.025**	**0.05**	**0.05**	**0.05**	**0.1**	**0.1**	**0.25**
Δ/σ	**0.01**	**0.025**	**0.05**	**0.1**	**0.25**	**0.025**	**0.05**	**0.1**	**0.25**	**0.05**	**0.1**	**0.25**	**0.1**	**0.25**	**0.25**
0.1	5,452	4,627	3,972	3,278	2,268	3,870	3,273	2,646	1,748	2,726	2,157	1,355	1,655	964	459
0.2	1,370	1,163	998	824	570	973	823	665	440	685	542	341	416	243	116
0.3	614	521	448	370	256	436	369	298	197	307	243	153	187	109	52
0.4	350	297	255	211	146	248	210	170	112	175	139	87	106	62	30
0.5	227	193	166	137	95	162	137	111	73	114	90	57	69	41	20
0.6	161	137	117	97	67	114	97	78	52	81	64	40	49	29	14
0.7	121	103	88	73	51	86	73	59	39	61	48	30	37	22	11
0.8	95	81	69	57	40	68	57	46	31	48	38	24	29	17	8
0.9	77	66	56	47	32	55	46	38	25	39	31	20	24	14	7
1.0	64	55	47	39	27	46	39	32	21	32	26	16	20	12	6
1.1	55	47	40	33	23	39	33	27	18	28	22	14	17	10	5
1.2	48	41	35	29	20	34	29	24	16	24	19	12	15	9	4
1.3	43	36	31	26	18	30	26	21	14	22	17	11	13	8	4
1.4	38	32	28	23	16	27	23	19	13	19	15	10	12	7	4
1.5	35	30	25	21	15	25	21	17	11	18	14	9	11	7	3
1.6	32	27	23	19	14	23	19	16	11	16	13	8	10	6	3
1.7	30	25	22	18	13	21	18	15	10	15	12	8	9	6	3
1.8	28	24	20	17	12	20	17	14	9	14	11	7	9	5	3
1.9	26	22	19	16	11	19	16	13	9	13	11	7	8	5	3
2.0	25	21	18	15	11	18	15	12	8	13	10	7	8	5	3
2.25	22	19	16	14	10	16	14	11	8	11	9	6	7	4	2
2.5	21	18	15	13	9	15	13	10	7	11	9	6	7	4	2
2.75	20	17	15	12	9	14	12	10	7	10	8	5	6	4	2
3.0	19	16	14	12	8	14	12	10	6	10	8	5	6	4	2
3.5	18	16	13	11	8	13	11	9	6	9	8	5	6	4	2
4.0	18	15	13	11	8	13	11	9	6	9	7	5	6	4	2

A.3 Critical Values for the Sign Test

Table A.3 Critical Values for the Sign Test Statistic S+

	Alpha								
N	**0.005**	**0.01**	**0.025**	**0.05**	**0.1**	**0.2**	**0.3**	**0.4**	**0.5**
4	4	4	4	4	3	3	3	2	2
5	5	5	5	4	4	3	3	3	2
6	6	6	5	5	5	4	4	3	3
7	7	6	6	6	5	5	4	4	3
8	7	7	7	6	6	5	5	4	4
9	8	8	7	7	6	6	5	5	4
10	9	9	8	8	7	6	6	5	5
11	10	9	9	8	8	7	6	6	5
12	10	10	9	9	8	7	7	6	6
13	11	11	10	9	9	8	7	7	6
14	12	11	11	10	9	9	8	7	7
15	12	12	11	11	10	9	9	8	7
16	13	13	12	11	11	10	9	9	8
17	14	13	12	12	11	10	10	9	8
18	14	14	13	12	12	11	10	10	9
19	15	14	14	13	12	11	11	10	9
20	16	15	14	14	13	12	11	11	10
21	16	16	15	14	13	12	12	11	10
22	17	16	16	15	14	13	12	12	11
23	18	17	16	15	15	14	13	12	11
24	18	18	17	16	15	14	13	13	12
25	19	18	17	17	16	15	14	13	12
26	19	19	18	17	16	15	14	14	13
27	20	19	19	18	17	16	15	14	13
28	21	20	19	18	17	16	15	15	14
29	21	21	20	19	18	17	16	15	14
30	22	21	20	19	19	17	16	16	15

Table A.3 Critical Values for the Sign Test Statistic S+ (continued)

	Alpha								
N	**0.005**	**0.01**	**0.025**	**0.05**	**0.1**	**0.2**	**0.3**	**0.4**	**0.5**
31	23	22	21	20	19	18	17	16	15
32	23	23	22	21	20	18	17	17	16
33	24	23	22	21	20	19	18	17	16
34	24	24	23	22	21	19	19	18	17
35	25	24	23	22	21	20	19	18	17
36	26	25	24	23	22	21	20	19	18
37	26	26	24	23	22	21	20	19	18
38	27	26	25	24	23	22	21	20	19
39	27	27	26	25	23	22	21	20	19
40	28	27	26	25	24	23	22	21	20
41	29	28	27	26	25	23	22	21	20
42	29	28	27	26	25	24	23	22	21
43	30	29	28	27	26	24	23	22	21
44	30	30	28	27	26	25	24	23	22
45	31	30	29	28	27	25	24	23	22
46	32	31	30	29	27	26	25	24	23
47	32	31	30	29	28	26	25	24	23
48	33	32	31	30	28	27	26	25	24
49	33	33	31	30	29	27	26	25	24
50	34	33	32	31	30	28	27	26	25

A.4 Critical Values for the WRS Test

The parameter “m” is the number of reference area samples and the parameter “n” is the number of survey unit samples. When using this table under Scenario A, m is the number of reference area samples and n is the number of survey unit samples. When using this table for Scenario B, the roles of m and n in this table are reversed.

Table A.4 Critical Values for the WRS Test

		n																		
m	**α**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**	**14**	**15**	**16**	**17**	**18**	**19**	**20**
2	**0.001**	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43
	0.005	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	40	42
	0.01	7	9	11	13	15	17	19	21	23	25	27	28	30	32	34	36	38	39	41
	0.025	7	9	11	13	15	17	18	20	22	23	25	27	29	31	33	34	36	38	40
	0.05	7	9	11	12	14	16	17	19	21	23	24	26	27	29	31	33	34	36	38
	0.1	7	8	10	11	13	15	16	18	19	21	22	24	26	27	29	30	32	33	35
3	**0.001**	12	15	18	21	24	27	30	33	36	39	42	45	48	51	54	56	59	62	65
	0.005	12	15	18	21	24	27	30	32	35	38	40	43	46	48	51	54	57	59	62
	0.01	12	15	18	21	24	26	29	31	34	37	39	42	45	47	50	52	55	58	60
	0.025	12	15	18	20	22	25	27	30	32	35	37	40	42	45	47	50	52	55	57
	0.05	12	14	17	19	21	24	26	28	31	33	36	38	40	43	45	47	50	52	54
	0.1	11	13	16	18	20	22	24	27	29	31	33	35	37	40	42	44	46	48	50
4	**0.001**	18	22	26	30	34	38	42	46	49	53	57	60	64	68	71	75	78	82	86
	0.005	18	22	26	30	33	37	40	44	47	51	54	58	61	64	68	71	75	78	81
	0.01	18	22	26	29	32	36	39	42	46	49	52	56	59	62	66	69	72	76	79
	0.025	18	22	25	28	31	34	37	41	44	47	50	53	56	59	62	66	69	72	75
	0.05	18	21	24	27	30	33	36	39	42	45	48	51	54	57	59	62	65	68	71
	0.1	17	20	22	25	28	31	34	36	39	42	45	48	50	53	56	59	61	64	67
5	**0.001**	25	30	35	40	45	50	54	58	63	67	72	76	81	85	89	94	98	102	107
	0.005	25	30	35	39	43	48	52	56	60	64	68	72	77	81	85	89	93	97	101
	0.01	25	30	34	38	42	46	50	54	58	62	66	70	74	78	82	86	90	94	98
	0.025	25	29	33	37	41	44	48	52	56	60	63	67	71	75	79	82	86	90	94
	0.05	24	28	32	35	39	43	46	50	53	57	61	64	68	71	75	79	82	86	89
	0.1	23	27	30	34	37	41	44	47	51	54	57	61	64	67	71	74	77	81	84
6	**0.001**	33	39	45	51	57	63	67	72	77	82	88	93	98	103	108	113	118	123	128
	0.005	33	39	44	49	54	59	64	69	74	79	83	88	93	98	103	107	112	117	122
	0.01	33	39	43	48	53	58	62	67	72	77	81	86	91	95	100	104	109	114	118
	0.025	33	37	42	47	51	56	60	64	69	73	78	82	87	91	95	100	104	109	113
	0.05	32	36	41	45	49	54	58	62	66	70	75	79	83	87	91	96	100	104	108
	0.1	31	35	39	43	47	51	55	59	63	67	71	75	79	83	87	91	94	98	102

Table A.4 Critical Values for the WRS Test (continued)

		n																		
m	**α**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**	**14**	**15**	**16**	**17**	**18**	**19**	**20**
7	**0.001**	42	49	56	63	69	75	81	87	92	98	104	110	116	122	128	133	139	145	151
	0.005	42	49	55	61	66	72	77	83	88	94	99	105	110	116	121	127	132	138	143
	0.01	42	48	54	59	65	70	76	81	86	92	97	102	108	113	118	123	129	134	139
	0.025	42	47	52	57	63	68	73	78	83	88	93	98	103	108	113	118	123	128	133
	0.05	41	46	51	56	61	65	70	75	80	85	90	94	99	104	109	113	118	123	128
	0.1	40	44	49	54	58	63	67	72	76	81	85	90	94	99	103	108	112	117	121
8	**0.001**	52	60	68	75	82	89	95	102	109	115	122	128	135	141	148	154	161	167	174
	0.005	52	60	66	73	79	85	92	98	104	110	116	122	129	135	141	147	153	159	165
	0.01	52	59	65	71	77	84	90	96	102	108	114	120	125	131	137	143	149	155	161
	0.025	51	57	63	69	75	81	86	92	98	104	109	115	121	126	132	137	143	149	154
	0.05	50	56	62	67	73	78	84	89	95	100	105	111	116	122	127	132	138	143	148
	0.1	49	54	60	65	70	75	80	85	91	96	101	106	111	116	121	126	131	136	141
9	**0.001**	63	72	81	88	96	104	111	118	126	133	140	147	155	162	169	176	183	190	198
	0.005	63	71	79	86	93	100	107	114	121	127	134	141	148	155	161	168	175	182	188
	0.01	63	70	77	84	91	98	105	111	118	125	131	138	144	151	157	164	170	177	184
	0.025	62	69	76	82	88	95	101	108	114	120	126	133	139	145	151	158	164	170	176
	0.05	61	67	74	80	86	92	98	104	110	116	122	128	134	140	146	152	158	164	170
	0.1	60	66	71	77	83	89	94	100	106	112	117	123	129	134	140	145	151	157	162
10	**0.001**	75	85	94	103	111	119	128	136	144	152	160	167	175	183	191	199	207	215	222
	0.005	75	84	92	100	108	115	123	131	138	146	153	160	168	175	183	190	197	205	212
	0.01	75	83	91	98	106	113	121	128	135	142	150	157	164	171	178	186	193	200	207
	0.025	74	81	89	96	103	110	117	124	131	138	145	151	158	165	172	179	186	192	199
	0.05	73	80	87	93	100	107	114	120	127	133	140	147	153	160	166	173	179	186	192
	0.1	71	78	84	91	97	103	110	116	122	128	135	141	147	153	160	166	172	178	184
11	**0.001**	88	99	109	118	127	136	145	154	163	171	180	188	197	206	214	223	231	240	248
	0.005	88	98	107	115	124	132	140	148	157	165	173	181	189	197	205	213	221	229	237
	0.01	88	97	105	113	122	130	138	146	153	161	169	177	185	193	200	208	216	224	232
	0.025	87	95	103	111	118	126	134	141	149	156	164	171	179	186	194	201	208	216	223
	0.05	86	93	101	108	115	123	130	137	144	152	159	166	173	180	187	195	202	209	216
	0.1	84	91	98	105	112	119	126	133	139	146	153	160	167	173	180	187	194	201	207
12	**0.001**	102	114	125	135	145	154	164	173	183	192	202	210	220	230	238	247	256	266	275
	0.005	102	112	122	131	140	149	158	167	176	185	194	202	211	220	228	237	246	254	263
	0.01	102	111	120	129	138	147	156	164	173	181	190	198	207	215	223	232	240	249	257
	0.025	100	109	118	126	135	143	151	159	168	176	184	192	200	208	216	224	232	240	248
	0.05	99	108	116	124	132	140	147	155	165	171	179	186	194	202	209	217	225	233	240
	0.1	97	105	113	120	128	135	143	150	158	165	172	180	187	194	202	209	216	224	231
13	**0.001**	117	130	141	152	163	173	183	193	203	213	223	233	243	253	263	273	282	292	302
	0.005	117	128	139	148	158	168	177	187	196	206	215	225	234	243	253	262	271	280	290
	0.01	116	127	137	146	156	165	174	184	193	202	211	220	229	238	247	256	265	274	283
	0.025	115	125	134	143	152	161	170	179	187	196	205	214	222	231	239	248	257	265	274
	0.05	114	123	132	140	149	157	166	174	183	191	199	208	216	224	233	241	249	257	266
	0.1	112	120	129	137	145	153	161	169	177	185	193	201	209	217	224	232	240	248	256

Table A.4 Critical Values for the WRS Test (continued)

		n																		
m	**α**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**	**13**	**14**	**15**	**16**	**17**	**18**	**19**	**20**
14	**0.001**	133	147	159	171	182	193	204	215	225	236	247	257	268	278	289	299	310	320	330
	0.005	133	145	156	167	177	187	198	208	218	228	238	248	258	268	278	288	298	307	317
	0.01	132	144	154	164	175	185	194	204	214	224	234	243	253	263	272	282	291	301	311
	0.025	131	141	151	161	171	180	190	199	208	218	227	236	245	255	264	273	282	292	301
	0.05	129	139	149	158	167	176	185	194	203	212	221	230	239	248	257	265	274	283	292
	0.1	128	136	145	154	163	171	180	189	197	206	214	223	231	240	248	257	265	273	282
15	**0.001**	150	165	178	190	202	212	225	237	248	260	271	282	293	304	316	327	338	349	360
	0.005	150	162	174	186	197	208	219	230	240	251	262	272	283	293	304	314	325	335	346
	0.01	149	161	172	183	194	205	215	226	236	247	257	267	278	288	298	308	319	329	339
	0.025	148	159	169	180	190	200	210	220	230	240	250	260	270	280	289	299	309	319	329
	0.05	146	157	167	176	186	196	206	215	225	234	244	253	263	272	282	291	301	310	319
	0.1	144	154	163	172	182	191	200	209	218	227	236	246	255	264	273	282	291	300	309
16	**0.001**	168	184	197	210	223	236	248	260	272	284	296	308	320	332	343	355	367	379	390
	0.005	168	181	194	206	218	229	241	252	264	275	286	298	309	320	331	342	353	365	376
	0.01	167	180	192	203	215	226	237	248	259	270	281	292	303	314	325	336	347	357	368
	0.025	166	177	188	200	210	221	232	242	253	264	274	284	295	305	316	326	337	347	357
	0.05	164	175	185	196	206	217	227	237	247	257	267	278	288	298	308	318	328	338	348
	0.1	162	172	182	192	202	211	221	231	241	250	260	269	279	289	298	308	317	327	336
17	**0.001**	187	203	218	232	245	258	271	284	297	310	322	335	347	360	372	384	397	409	422
	0.005	187	201	214	227	239	252	264	276	288	300	312	324	336	347	359	371	383	394	406
	0.01	186	199	212	224	236	248	260	272	284	295	307	318	330	341	353	364	376	387	399
	0.025	184	197	209	220	232	243	254	266	277	288	299	310	321	332	343	354	365	376	387
	0.05	183	194	205	217	228	238	249	260	271	282	292	303	313	324	335	345	356	366	377
	0.1	180	191	202	212	223	233	243	253	264	274	284	294	305	315	325	335	345	355	365
18	**0.001**	207	224	239	254	268	282	296	309	323	336	349	362	376	389	402	415	428	441	454
	0.005	207	222	236	249	262	275	288	301	313	326	339	351	364	376	388	401	413	425	438
	0.01	206	220	233	246	259	272	284	296	309	321	333	345	357	370	382	394	406	418	430
	0.025	204	217	230	242	254	266	278	290	302	313	325	337	348	360	372	383	395	406	418
	0.05	202	215	226	238	250	261	273	284	295	307	318	329	340	352	363	374	385	396	407
	0.1	200	211	222	233	244	255	266	277	288	299	309	320	331	342	352	363	374	384	395
19	**0.001**	228	246	262	277	292	307	321	335	350	364	377	391	405	419	433	446	460	473	487
	0.005	227	243	258	272	286	300	313	327	340	353	366	379	392	405	419	431	444	457	470
	0.01	226	242	256	269	283	296	309	322	335	348	361	373	386	399	411	424	437	449	462
	0.025	225	239	252	265	278	290	303	315	327	340	352	364	377	389	401	413	425	437	450
	0.05	223	236	248	261	273	285	297	309	321	333	345	356	368	380	392	403	415	427	439
	0.1	220	232	244	256	267	279	290	302	313	325	336	347	358	370	381	392	403	415	426
20	**0.001**	250	269	286	302	317	333	348	363	377	392	407	421	435	450	464	479	493	507	521
	0.005	249	266	281	296	311	325	339	353	367	381	395	409	422	436	450	463	477	490	504
	0.01	248	264	279	293	307	321	335	349	362	376	389	402	416	429	442	456	469	482	495
	0.025	247	261	275	289	302	315	329	341	354	367	380	393	406	419	431	444	457	470	482
	0.05	245	258	271	284	297	310	322	335	347	360	372	385	397	409	422	434	446	459	471
	0.1	242	254	267	279	291	303	315	327	339	351	363	375	387	399	410	422	434	446	458

Reject the null hypothesis if the test statistic (W_r) is greater than the table (critical) value.

For n or m greater than 20 with few or no ties, the table (critical) value can be calculated from:

$$Critical\,Value = \frac{m(n+m+1)}{2} + z\sqrt{\frac{nm(n+m+1)}{12}} \tag{A-1}$$

If there are ties, the critical value can be calculated from:

$$Critical\,Value = \frac{m(n+m+1)}{2} + z\sqrt{\frac{nm}{12}\left[(n+m+1) - \sum_{j=1}^{g}\frac{t_j(t_j^2-1)}{(n+m)(n+m+1)}\right]} \tag{A-2}$$

Where:

g = the number of groups of tied measurements.
t_j = the number of tied measurements in the jth group.
z = the (1-α) percentile of a standard normal distribution (see list below).

α	z
0.001	3.090
0.005	2.575
0.01	2.326
0.025	1.960
0.05	1.645
0.1	1.282

Other values for z can be obtained from Table A.1.

A.5 Critical Values for the Quantile Test

Table A.5a Values of *r* and *k* for the Quantile Test When α Is Approximately 0.01

	Number of Survey Unit Measurements, *n*																			
m	**5**	**10**	**15**	**20**	**25**	**30**	**35**	**40**	**45**	**50**	**55**	**60**	**65**	**70**	**75**	**80**	**85**	**90**	**95**	**100**
5	***r,k***		11,11	13,13	16,16	19,19	22,22	25,25	28,28											***r,k***
	α		0.008	0.015	0.014	0.013	0.013	0.013	0.012											α
10		6,6	7,7	9,9	11,11	13,13	14,14	16,16	18,18	19,19	21,21	23,23	25,25	26,26	28,28	30,30				
		0.005	0.013	0.012	0.011	0.01	0.014	0.013	0.012	0.015	0.014	0.013	0.012	0.015	0.014	0.013				
15	3,3	7,6	6,6	7,7	8,8	10,10	11,11	12,12	13,13	15,15	16,16	17,17	18,18	19,19	21,21	22,22	23,23	24,24	26,26	27,27
	0.009	0.007	0.008	0.011	0.014	0.009	0.011	0.013	0.014	0.011	0.012	0.013	0.014	0.015	0.012	0.013	0.014	0.015	0.013	0.013
20	6,4	4,4	5,5	6,6	7,7	8,8	9,9	10,10	11,11	12,12	13,13	14,14	15,15	16,16	17,17	18,18	19,19	19,19	20,20	21,21
	0.005	0.008	0.009	0.01	0.011	0.011	0.011	0.011	0.011	0.011	0.011	0.012	0.012	0.012	0.012	0.012	0.012	0.015	0.015	0.015
25	4,3	7,5	4,4	5,5	6,6	7,7	8,8	9,9	9,9	10,10	11,11	12,12	12,12	13,13	14,14	15,15	16,16	16,16	17,17	18,18
	0.009	0.012	0.015	0.013	0.011	0.01	0.009	0.009	0.014	0.012	0.011	0.011	0.015	0.014	0.013	0.012	0.011	0.014	0.014	0.013
30	4,3	3,3	4,4	5,5	6,6	6,6	7,7	8,8	8,8	9,9	10,10	10,10	11,11	1211	12,12	13,13	14,14	14,14	15,15	15,15
	0.006	0.012	0.009	0.007	0.006	0.012	0.01	0.008	0.013	0.011	0.009	0.013	0.011	0.014	0.013	0.012	0.011	0.014	0.012	0.015
35	2,2	3,3	4,4	4,4	5,5	6,6	6,6	7,7	7,7	8,8	9,9	9,9	10,10	10,10	11,11	11,11	12,12	13,13	13,13	14,14
	0.013	0.008	0.006	0.014	0.01	0.007	0.012	0.009	0.014	0.011	0.009	0.013	0.01	0.014	0.011	0.015	0.012	0.011	0.013	0.012
40	2,2	3,3	7,5	4,4	5,5	5,5	6,6	6,6	7,7	7,7	8,8	8,8	9,9	9,9	10,10	10,10	11,11	11,11	12,12	12,12
	0.01	0.006	0.013	0.01	0.006	0.012	0.008	0.013	0.009	0.013	0.01	0.014	0.011	0.014	0.011	0.014	0.012	0.014	0.012	0.014
45	2,2	6,4	3,3	4,4	4,4	5,5	5,5	6,6	6,6	7,7	7,7	8,8	8,8	9,9	9,9	10,10	10,10	10,10	11,11	11,11
	0.008	0.008	0.013	0.007	0.014	0.008	0.014	0.009	0.013	0.009	0.013	0.009	0.012	0.009	0.012	0.009	0.012	0.015	0.012	0.014
50		4,3	3,3	4,4	4,4	5,5	5,5	5,5	6,6	6,6	7,7	7,7	8,8	8,8	8,8	9,9	9,9	10,10	10,10	10,10
		0.013	0.01	0.005	0.01	0.006	0.01	0.015	0.009	0.013	0.009	0.012	0.009	0.011	0.014	0.011	0.013	0.01	0.012	0.015
55		4,3	3,3	7,5	4,4	4,4	5,5	5,5	6,6	6,6	6,6	7,7	7,7	8,8	8,8	8,8	9,9	9,9	9,9	10,10
		0.01	0.008	0.013	0.008	0.014	0.007	0.011	0.007	0.01	0.014	0.009	0.012	0.008	0.01	0.013	0.009	0.012	0.014	0.011
60		4,3	3,3	3,3	4,4	4,4	5,5	5,5	5,5	6,6	6,6	6,6	7,7	7,7	7,7	8,8	8,8	8,8	9,9	9,9
		0.008	0.007	0.014	0.006	0.011	0.006	0.009	0.013	0.007	0.01	0.014	0.009	0.011	0.014	0.01	0.012	0.015	0.01	0.013
65		4,3	3,3	3,3	6,5	4,4	4,4	5,5	5,5	5,5	6,6	6,6	6,6	7,7	7,7	7,7	8,8	8,8	8,8	9,9
		0.007	0.006	0.012	0.006	0.009	0.013	0.007	0.01	0.014	0.008	0.011	0.014	0.009	0.011	0.014	0.009	0.011	0.014	0.01
70		2,2	6,4	3,3	7,5	4,4	4,4	5,5	5,5	5,5	5,5	6,6	6,6	6,6	7,7	7,7	7,7	8,8	8,8	8,8
		0.014	0.008	0.01	0.013	0.007	0.011	0.005	0.008	0.011	0.015	0.008	0.011	0.014	0.009	0.011	0.013	0.009	0.011	0.013
75		2,2	4,3	3,3	3,3	4,4	4,4	4,4	5,5	5,5	5,5	6,6	6,6	6,6	6,6	7,7	7,7	7,7	8,8.	8,8
		0.013	0.014	0.008	0.014	0.006	0.009	0.013	0.006	0.009	0.012	0.007	0.009	0.011	0.014	0.009	0.011	0.013	0.008	0.01
80		2,2	4,3	3,3	3,3	6,5	4,4	4,4	5,5	5,5	5,5	5,5	6,6	6,6	6,6	6,6	7,7	7,7	7,7	7,7
		0.011	0.012	0.007	0.012	0.006	0.008	0.011	0.005	0.007	0.01	0.013	0.007	0.009	0.012	0.014	0.009	0.01	0.013	0.015
85		2,2	4,3	3,3	3,3	7,5	4,4	4,4	4,4	5,5	5,5	5,5	5,5	6,6	6,6	6,6	6,6	7,7	7,7	7,7
		0.01	0.01	0.006	0.011	0.013	0.006	0.009	0.013	0.006	0.008	0.011	0.014	0.008	0.01	0.012	0.014	0.008	0.01	0.012
90			4,3	3,3	3,3	3,3	4,4	4,4	4,4	5,5	5,5	5,5	5,5	5,5	6,6	6,6	6,6	6,6	7,7	7,7
			0.009	0.005	0.009	0.014	0.005	0.008	0.011	0.005	0.007	0.009	0.012	0.015	0.008	0.01	0.012	0.014	0.008	0.019
95			4,3	6,4	3,3	3,3	6,5	4,4	4,4	4,4	5,5	5,5	5,5	5,5	6,6	6,6	6,6	6,6	6,6	7,7
			0.008	0.008	0.008	0.013	0.005	0.007	0.01	0.013	0.006	0.008	0.01	0.013	0.007	0.008	0.01	0.012	0.014	0.008
100	***r,k***		4,3	4,3	3,3	3,3	7,5	4,4	4,4	4,4	4,4	5,5	5,5	5,5	5,5	6,6	6,6	6,6	6,6	6,6
	α		0.007	0.014	0.007	0.011	0.013	0.006	0.008	0.011	0.015	0.007	0.009	0.011	0.013	0.007	0.008	0.01	0.012	0.014

Table A.5b Values of *r* and *k* for the Quantile Test When α Is Approximately 0.025

	Number of Survey Unit Measurements, *n*																			
m	**5**	**10**	**15**	**20**	**25**	**30**	**35**	**40**	**45**	**50**	**55**	**60**	**65**	**70**	**75**	**80**	**85**	**90**	**95**	**100**
5	***r,k*** α		9,9 0.03	12,12 0.024	15,l5 0.021	17,17 0.026	20,20 0.024	22,22 0.028	25,25 0.025											***r,k*** ***α***
10		7,6 0.029	6,6 0.028	8,8 0.022	9,9 0.029	11,11 0.024	12,12 0.029	14,14 0.025	17,17 0.025	18,18 0.029	20,20 0.026	21,21 0.029	23,23 0.026	24,24 0.029	26,26 0.026	27,27 0.029				
15	11,5 0.03	6,5 0.023	5,5 0.021	6,6 0.024	7,7 0.026	8,8 0.027	9,9 0.028	10,10 0.029	11,11 0.03	13,13 0.022	15,15 0.023	14,14 0.023	16,16 0.024	17,17 0.025	18,18 0.025	19,19 0.026	21,21 0.021	21,21 0.027	22,22 0.027	23,23 0.027
20	8,4 0.023	3,3 0.03	4,4 0.026	5,5 0.024	6,6 0.022	7,7 0.02	12,11 0.021	13,12 0.024	9,9 0.028	10,10 0.026	11,11 0.024	12,12 0.023	13,13 0.022	13,13 0.029	14,14 0.027	15,15 0.026	16 16 0.025	17,17 0.024	17,17 0.029	18,18 0.028
25	2,2 0.023	8,5 0.027	6,5 0.021	7,6 6.023	5,5 0.025	6,6 0.02	10,9 0.026	7,7 0.027	8,8 0.023	13,12 0.027	9,9 0.027	10,10 0.024	11,11 0.022	11,1l 0.028	12,12 0.025	13,13 0.823	13,13 0.628	14,14 0.025	15,15 0.023	15,15 0.028
30	6,3 0.026	6,4 0.026	9,6 0.026	4,4 0.021	7,6 0.029	5,5 0.026	9,8 0.024	6,6 0.029	7,7 0.023	12,11 0.021	8,8 0.025	9,9 0.021	9,9 0.027	10,10 0.023	10,10 0.029	11,11 0.025	11,11 0.03	12,12 0.026	13,13 0.023	13,13 0.027
35	7,3 0.03	4,3 0.03	3,3 0.023	6,5 0.02	4,4 0.026	10,8 0.022	5,5 0.027	9,8 0.024	6,6 0.027	7,7 0.02	7,7 0.027	8,8 0.021	8,8 0.027	9,9 0.022	9,9 0.027	10,10 0.022	10,10 0.027	11,11 0.022	11,11 0.027	12,12 0.023
40	3,2 0.029	4,3 0.022	8,5 0.028	11,7 0.025	6,5 0.028	4,4 0.03	10,8 0.026	5,5 0.027	9,8 0.023	6,6 0.026	10,9 0.028	7,7 0.024	12,11 0.02	8,8 0.023	8,8 0.029	9,9 0.022	9,9 0.027	10,10 0.021	10,10 0.026	11,11 0.021
45	3,2 0.023	8,4 0.029	6,4 0.036	3,3 0.026	8,6 0.021	4,4 0.023	7,6 0.025	5,5 0.02	5,5 0.028	9,8 0.023	6,6 0.024	10,9 0.026	7,7 0.022	7,7 0.027	8,8 0.02	8,8 0.025	8,8 0.03	9,9 0.023	9,9 0.027	10,10 0.02l
50		2,2 0.025	6,4 0.022	3,3 0.021	11,7 0.077	6,5 6.026	4,4 0.026	7,6 0.028	5,5 0.021	5,5 0.028	9,8 0.022	6,6 0.023	6,6 0.029	7,7 0.02	7,7 0.025	12,11 0.02	8,8 0.022	8,8 0.026	13,12 0.027	9,9 0.023
55		2,2 0.022	4,3 0.029	8,5 0.028	3,3 0.028	8,6 0.021	4,4 0.02	4,4 0.029	10,8 0.021	5,5 0.022	5,5 0.028	9,8 0.022	6,6 0.092	6,6 0.028	10,9 0.029	7,7 0.023	7,7 0.027	12,11 0.023	8,8 0.023	8,8 0.027
60		14,5 0.022	4,3 0.024	8,5 0.021	3,3 0.023	11,7 0.029	6,5 0.024	4,4 0.023	7,6 0.023	10,8 0.024	5,5 0.023	5,5 0.029	9,8 0.022	6,6 0.022	6,6 0.027	10,9 0.027	7,7 0.021	7,7 0.025	7,7 0.03	8,8 0.021
65		6,3 0.028	7,4 0.021	6,4 0.025	10,6 0.025	3,3 0.029	8,6 0.021	6,5 0.029	4,4 0.026	7,6 0.026	10,8 0.026	5,5 0.023	5,5 0.029	9,8 0.022	6,6 0.021	6,6 0.026	10,9 0.026	7,7 0.020	7,7 0.024	7,7 0.028
70		6,3 0.024	2,2 0.029	6,4 0.021	8,5 0.028	3,3 0.025	13,8 0.026	6,5 0.023	4,4 0.022	4,4 0.028	7,6 0.028	10,8 0.027	5,5 0.024	5,5 0.029	9,8 0.022	6,6 0.021	6,6 0.025	6,6 0.029	10,9 0.03	7,7 0.022
75		11,4 0.022	2,2 0.026	4,3 0.028	8,5 0.022	3,3 0.022	9,6 0.028	8,6 0.021	6,5 0.027	4,4 0.024	7,6 0.023	7,6 0.03	10,8 0.029	5,5 0.024	5,5 0.029	9,8 0.021	6,6 0.021	6,6 0.024	6,6 0.028	10,9 0.028
80		7,3 0.028	2,2 0.024	4,3 0.024	6,4 0.028	10,6 0.024	3,3 0.027	13,8 0.027	6,5 0.023	4,4 0.02	4,4 0.026	7,6 0.024	10,8 0.023	5,5 0.07	5,5 0.025	5,5 0.029	9,8 0.021	6,6 0.02	6,6 0.024	6,6 0.027
85		3,2 0.029	2,2 0.021	4,3 0.021	6,4 0.023	8,5 0.028	3,3 0.023	9,6 0.03	8,6 0.02	6,5 0.026	4,4 0.022	4,4 0.028	7,6 0.026	10,8 0.024	5,5 0.021	5,5 0.025	5,5 0.029	9,8 0.021	6,6 0.02	6,6 0.023
90			5,3 0.02	11,5 0.027	9,5 0.023	8,5 0.023	3,3 0.021	3,3 0.028	13,8 0.028	6,5 0.022	6,5 0.029	4,4 0.024	4,4 0.029	7,6 0.028	10,8 0.026	5,5 0.022	5,5 0.025	5,5 0.03	9,8 0.021	9,8 0.025
95			10,4 0.029	2,2 0.029	4,3 0.028	6,4 0.029	10,6 0.023	3,3 0.025	11,7 0.026	8,6 0.02	6,5 0.025	4,4 0.021	4,4 0.026	7,6 0.024	7,6 0.029	10,8 0.027	5,5 0.022	5,5 0.026	5,5 0.03	9,8 0.021
100	***r,k*** α		6,3 0.029	2,2 0.027	4,3 0.025	6,4 0.025	8,5 0.028	3,3 0.022	3,3 0.029	13,8 0.028	6,5 0.022	6,5 0.028	4,4 0.023	4,4 0.027	7,6 0.025	10,8 0.022	10,8 0.028	5,5 0.022	5,5 0.026	5,5 0.03

Table A.5c Values of *r* and *k* for the Quantile Test When α Is Approximately 0.05

m	Number of Survey Unit Measurements, n																		
	5	**10**	**15**	**20**	**25**	**30**	**35**	**40**	**45**	**50**	**55**	**60**	**65**	**70**	**75**	**80**	**85**	**90**	**95**	**100**
5	**r,k** α		8,8 0.051	10,10 0.057	13 13 0.043	15 15 0.048	17,17 0.051	19,19 0.054	21,21 0.056											**r,k** **α**
10		4,4 0.043	5,5 0.057	14,12 0.045	8,8 0.046	9,9 0.052	10,10 0.058	12,12 0.046	13,13 0.05	14,14 0.054	15,15 0.057	17,17 0.049	18,18 0.052	19,19 0.055	20,20 0.057	21,21 0.059	23,23 0.053			
15	2,2 0.053	3,3 0.052	4,4 0.05	5,5 0.048	6,6 0.046	7,7 0.045	8,8 0.052	9,9 0.043	9,9 0.06	10,10 0.057	11,11 0.055	12,12 0.054	13,13 0.052	14,14 0.051	15,15 0.05	16,16 0.049	16,16 0.058	17,17 0.057	18,18 0.056	19,19 0.055
20	9,4 0.04	8,5 0.056	6,5 0.04	4,4 0.053	5,5 0.043	9,8 0.052	6,6 0.056	7,7 48	8,8 0.043	8,8 0.057	9,9 0.051	10,10 0.046	10,10 0.057	11,11 0.052	12,12 0.048	12,12 0.057	13,13 0.053	14,14 0.049	14,14 0.057	15,15 0.054
25	6,3 0.041	6,4 0.043	3,3 0.046	6,5 0.052	4,4 0.055	5,5 0.041	5,5 0.059	6,6 0.046	11,10 0.042	7,7 0.05	8,8 0.042	8,8 0.053	9,9 0.045	9,9 0.055	10,10 0.048	11,11 0.042	11,11 0.05	11,11 0.058	12,12 0.052	12,12 0.06
30	3,2 0.047	2,2 0.058	10,6 0.052	3,3 0.058	11,8 0.045	4,4 0.056	8,7 0.044	5,5 0.054	6,6 0.04	6,6 0.053	7,7 0.041	7,7 0.052	8,8 0.042	8,8 0.051	9,9 0.042	9,9 0.05	9,9 0.059	10,10 0.049	10,10 0.057	11,11 0.049
35	8,3 0.046	2,2 0.045	6,4 0.058	3,3 0.043	6,5 0.041	4,4 0.04	4,4 0.057	8,7 0.043	5,5 0.051	9,8 0.052	6,6 0.047	6,6 0.057	7,7 0.043	7,7 0.053	8,8 0.041	8,8 0.049	8,8 0.057	9,9 0.046	9,9 0.053	10,10 0.044
40	4,2 0.055	5,3 0.048	4,3 0.057	10,6 0.059	3,3 0.053	6,5 0.048	4,4 0.043	4,4 0.058	8,7 0.042	5,5 0.048	9,8 0.047	6,6 0.042	6,6 0.051	11,10 0.042	7,7 0.045	7,7 0.053	8,8 0.041	8,8 0.048	8,8 0.055	9,9 0.043
45	4,2 0.045	9,4 0.047	2,2 0.059	8,5 0.052	3,3 0.042	8,6 0.041	6,5 0.054	4,4 0.045	4,4 0.058	8,7 0.041	5,5 0.046	5,5 0.057	9,8 0.056	6,6 0.047	6,6 0.055	11,10 0.046	7,7 0.047	7,7 0.054	8,8 0.041	8,8 0.047
50		6,3 0.051	2,2 0.05	6,4 0.051	12,7 0.05	3,3 0.049	8,6 0.049	6,5 0.059	4,4 0.047	4,4 0.059	8,7 0.041	5,5 0.045	5,5 0.054	9,8 0.051	6,6 0.043	6,6 0.05	6,6 0.058	7,7 0.041	7,7 0.048	7,7 0.054
55		3,2 0.059	2,2 0.043	4,3 0.056	8,5 0.058	3,3 0.041	5,4 0.041	6,5 0.046	9,7 0.042	4,4 0.048	4,4 0.059	8,7 0.04	5,5 0.043	5,5 0.052	9,8 0.048	6,6 0.04	6,6 0.047	6,6 0.054	11,10 0.043	7,7 0.043
60		3,2 0.052	5,3 0.052	4,3 0.046	6,4 0.059	3,3 0.035	3,3 0.047	8,6 0.043	6,5 51	9,7 0.046	4,4 0.049	4,4 0.059	13,10 0.052	5,5 0.042	5,5 0.05	5,5 0.058	9,8 0.054	6,6 0.044	6,6 0.05	6,6 0.056
65		.3,2 0.045	5,3 0.043	2,2 0.053	6,4 0.048	10,6 0.05	3,3 0.04	3,3 0.052	6,5 0.041	6,5 0.055	4,4 0.042	4,4 0.05	4,4 0.06	13,10 0.052	5,5 0.041	5,5 0.048	5,5 0.055	9,8 0.051	6,6 0.041	6,6 0.047
70		8,3 0.057	9,4 0.048	2,2 0.047	4,3 0.055	8,5 0.05	5,4 0.041	3,3 0.046	3,3 0.057	6,5 0.045	6,5 0.058	4,4 0.043	4,4 0.051	4,4 0.06	13,10 0.051	5,5 0.041	5,5 0.047	5,5 0.054	9,8 0.048	9,8 0.057
75		8,3 0.049	6,3 0.056	2,2 0.043	4,3 0.047	6,4 0.054	10,6 0.053	3,3 0.04	3,3 0.051	8,6 0.044	6,5 0.049	9,7 0.041	4,4 0.044	4,4 0.052	5,5 0.06	13,10 0.051	8,7 0.047	5,5 0.046	5,5 0.052	5,5 0.058
80		4,2 0.059	6,3 0.048	5,3 0.053	2,2 0.055	6,4 0.046	8,5 0.055	5,4 0.041	3,3 0.045	3,3 0.055	6,5 0.041	6,5 0.052	9,7 0.043	4,4 0.045	4,4 0.053	7,6 0.058	13,10 0.051	8 7 0.046	5,5 0.045	5,5 0.051
85		4,2 0.054	3,2 0.058	5,3 0.047	2,2 0.05	4,3 0.054	4,3 0.048	10,6 0.056	5,4 0.049	3,3 0.049	3,3 0.059	6,5 0.044	6,5 0.055	9,7 0.046	4,4 0.046	4,4 0.053	7,6 0.059	10,8 0.06	8,7 0.045	5,5 0.044
90			3,2 0.053	5,3 0.041	2,2 0.046	6,4 0.059	6,4 0.051	8,5 0.058	5,4 0.042	3,3 0.044	3,3 0.053	8,6 0.045	6,5 0.047	6,5 0.058	4,4 0.04l	4,4 0.047	4,4 0.054	7,6 0.059	10,8 0.06	8,7 0.041
95			3,2 0.048	9,4 0.048	2,2 0.042	2,2 0.056	4,3 0.059	8,5 0.05	10,6 0.058	5,4 0.048	3,3 0.048	3,3 0.056	6,5 0.041	6,5 0.05	9,7 0.040	4,4 0.042	4,4 0.048	4,4 0.054	7,6 0.59	10,8 0.059
100	**r,k** α		3,2 0.044	6,3 0.057	5,3 0.054	2,2 0.052	4,3 0.053	6,4 0.056	10,6 0.049	5,4 0.043	3,3 0.043	3,3 0.051	3,3 0.059	6,5 0.044	6,5 0.053	9,7 0.042	4,4 0.043	4,4 0.049	4,4 0.055	7,6 0.059

Table A.5d Values of *r* and *k* for the Quantile Test When α Is Approximately 0.10

	Number of Survey Unit Measurements, *n*																			
m	**5**	**10**	**15**	**20**	**25**	**30**	**35**	**40**	**45**	**50**	**55**	**60**	**65**	**70**	**75**	**80**	**85**	**90**	**95**	**100**
5	***r,k*** α		7,7 0.083	8,8 0.116	10,10 0.109	12,12 0.104	14,14 0.1	15,15 0.117	17,17 0.112											***r,k*** ***α***
10		3,3 0.105	4,4 0.108	5,5 0.109	6,6 0.109	7,7 0.109	8,8 0.109	9,9 0.109	10,10 0.109	11,11 0.109	12,12 0.109	13,13 0.109	14,14 0.109	15,15 0.109	16,16 0.109	1712 0.109	18,18 0.109			
15	9,4 0.098	10,6 0.106	3,3 0.112	4,4 0.093	5,5 0.081	5,5 0.117	6,6 0.102	7,7 0.092	7,7 0.118	8,8 0.106	9,9 0.098	9,9 0.118	10,10 0.109	11,11 0.101	11,11 0.118	12,12 0.11	13,13 0.104	13,13 0.118	14,14 0.111	15,15 0.106
20	3,2 0.091	2,2 0.103	5,4 0.093	3,3 0.115	4,4 0.085	4,4 0.119	5,5 0.093	10,9 0.084	6,6 0.099	7,7 0.083	7,7 0.102	8,8 0.088	8,8 0.105	9,9 0.092	9,9 0.107	10,10 0.095	10,11 0.108	11,11 0.098	11,11 0.11	12,12 0.1
25	4,2 0.119	7,4 0.084	8,5 0.112	3,3 0.08	3,3 0.117	4,4 0.08	4,4 0.107	8,7 0.108	5,5 0.101	10,9 0.088	6,6 0.096	6,6 0.114	7,7 0.093	7,7 0.108	8,8 0.091	8,8 0.104	8,8 0.117	9,9 0.1	9,9 0.112	10,10 0.098
30	4,2. 0.089	5,3 0.089	2,2 0.106	14,8 0.111	3,3 0.088	3,3 0.119	9,7 0.116	4,4 0.1	8,7 0.093	5,5 0.088	5,5 0.106	6,6 0.08	6,6 0.095	6,6 0.11	7,7 0.087	7,7 0.1	7,7 0.113	8,8 0.092	8,8 0.103	8,8 0.115
35	5,2 0.109	3,2 0.119	2,2 0.086	6,4 0.12	5,4 0.091	3,3 0.093	3,3 0.12	9,7 0.112	4,4 0.094	4,4 0.114	8,7 0.107	5,5 0.094	5,5 0.11	6,6 0.081	6,6 0.094	6,6 0.107	6,6 0.12	7,7 0.094	7,7 0.105	7,7 0.116
40	5,2 0.087	3,2 0.098	5,3 0.119	2,2 0.107	12,7 0.109	5,4 0.102	3,3 0.097	6,5 0.100	9,7 0.109	4,4 0.09	4,4 0.107	8,7 0.097	5,5 0.086	5,5 0.099	5,5 0.112	6,6 0.082	6,6 0.093	6,6 0.104	6,6 0.116	7,7 0.089
45	6,2 0.103	3,2 0.082	5,3 0.094	2,2 0.091	6,4 0.115	7,5 0.086	5,4 0.112	3,3 0.1	6,5 0.101	9,7 0.107	4,4 0.087	4,4 0.102	4,4 0.117	8,7 0.107	5,5 0.091	5,5 0.103	5,5 0.115	6,6 0.083	6,6 0.093	6,6 0.103
50		7,3 0.083	9,4 0.115	7,4 0.097	2,2 0.108	10,6 0.112	5,4 0.09	3,3 0.084	3,3 0.103	6,5 0.102	9,7 0.105	4,4 0.084	4,4 0.098	4,4 0.112	8,7 0.099	5,5 0.084	5,5 0.95	5,5 0.105	5,5 0.116	6,6 0.083
55		4,2 0.109	3,2 0.114	5,3 0.114	2,2 0.095	6,4 0.112	14,8 0.111	5,4 0.098	3,3 0.088	3,3 0.104	6,5 0.103	9,7 0.104	4,4 0.082	4,4 0.095	4,4 0.107	4,4 0.12	8,7 0.107	5,5 0.088	5,5 0.098	5,5 0.108
60		4,2 0.095	3,2 0.1	5,3 0.097	2,2 0.084	2,2 0.109	8,5 0.119	5,4 0.082	5,4 0.105	3,3 0.091	3,3 0.106	6,5 0.103	9,7 0.102	4,4 0.081	4,4 0.092	4,4 0.103	4,4 0.115	8,7 0.1	5,5 0.083	5,5 0.092
65		4,2 0.084	3,2 0.089	5,3 0.082	7,4 0.090	2,2 0.097	6,4 0.11	12,7 0.113	5,4 0.089	5,4 0.111	3,3 0.093	3,3 0.108	6,5 0.104	9,7 0.101	7,6 0.084	4,4 0.09	4,4 0.1	4,4 0.11	8,7 0.094	8,7 0.107
70		5,2 0.115	7,3 0.101	9,4 0.106	5,3 0.112	2,2 0.088	2,2 0.109	8,5 0.114	7,5 0.081	5,4 0.096	3,3 0.083	3,3 0.096	3,3 0.109	6,5 0.104	9,7 0.191	7,6 0.082	4,4 0.088	4,4 0.097	4,4 0.107	4,4 0.117
75		5,2 103	7,3 0.088	3,2 0.111	5,3 0.098	7,4 0.101	2,2 0.099	2,2 0.119	10,6 0.117	5,4 0.083	5,4 0.102	3,3 0.085	3,3 0.098	3,3 0.11	6,5 0.105	9,7 0.1	7,6 0.081	4,4 0.086	4,4 0.095	4,4 0.104
80		5,2 0.093	4,2 0.116	3,2 0.101	5,3 0.086	7,4 0.086	2,2 0.09!	2,2 0.109	8,5 0.111	14,8 0.11	5,4 0.089	5,4 0.107	3,3 0.088	3,3 0.099	3,3 0.111	6,5 0.105	6,5 0.12	9,7 0.116	4,4 0.084	4,4 0.093
85		5,2 0.084	4,2 0.106	3,2 0.092	9,4 117	5,3 0.111	2,2 0.083	2,2 0.101	2,2 0.118	10,6 0.112	7,5 0.084	5,4 0.094	5,4 0.111	3,3 0.09	3,3 0.101	3,3 0.112	6,5 0.105	6,5 0.119	9,7 0.114	4,4 0.083
90			4,2 0.097	3,2 0.085	3,2 0.119	5,3 0.099	7,4 0.095	2,2 0.093	2,2 0.109	8,5 0.108	12,7 0.114	5,4 0.083	5,4 0.099	3,3 0.082	3,3 0.092	3,3 0.102	3,3 0.113	6,5 0.105	6,5 0.119	9,7 0.113
95			4,2 0.089	7,3 100	3,2 0.11	5,3 0.089	7,4 0.084	2,2 0.086	2,2 0.102	2,2 0.117	10,6 0.08	14,8 0.117	5,4 0.088	5,4 0.103	3,3 0.084	3,3 0.094	3,3 0.103	3,3 0.113	6,5 0.106	6,5 0.118
100	***r,k*** ***α***		4,2 0.082	7,3 0.09	3,2 0.102	5,3 0.08	5,3 0.109	2,2 0.08	2,2 0.095	2,2 0.11	6,4 0.118	12,7 0.109	7,5 0.086	5,4 0.093	5,4 0.08	3,3 0.086	3,3 0.095	3,3 0.104	3,3 0.114	6,5 0.106

Table A.5 contains values of the parameters *r* and *k* needed for the Quantile test calculated by Gilbert and Simpson (Gilbert 1992) for certain combinations of *m* (the number of measurements in the reference area) and *n* (the number of measurements in the survey unit). The value of α listed is that obtained from simulation studies.

B. SOURCES OF BACKGROUND RADIOACTIVITY

B.1 Introduction

Background radioactivity can complicate the disposition decision for M&E. Background radioactivity may be the result of environmental radioactivity, inherent radioactivity, instrument noise, or some combination of the three. Special consideration is given to issues associated with technologically enhanced naturally occurring radioactive materials (TENORM) and orphan sources as contributors to background. The planning team should consider these potential sources of background activity and determine what effect, if any, they may have on the design of the disposition survey.

Information on background radioactivity can be obtained from many sources, including:

- The Nuclear Regulatory Commission (NRC) provides information concerning background radioactivity in *Background as a Residual Radioactivity Criterion for Decommissioning* NUREG-1501 (NRC 1994).
- The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has published a report on *Sources and Effects of Ionizing Radiation* (UNSCEAR 2000) and provides a searchable version of the report on the World Wide Web at www.unscear.org.
- The National Council on Radiation Protection and Measurements (NCRP) has published reports on *Exposure of the Population in the United States and Canada from Natural Background Radiation*, NCRP Report No. 94 (NCRP 1988a) and *Radiation Exposure of the U.S. Population from Consumer Products and Miscellaneous Sources*, NCRP Report No. 95 (NCRP 1988b).

B.2 Environmental Radioactivity

Environmental radioactivity is radioactivity from the environment where the M&E is located. There are three sources contributing to environmental radioactivity; terrestrial (Section B.2.1), manmade (Section B.2.2), and cosmic and cosmogenic (Section B.2.3). Although background radiation is present everywhere, the component radionuclide concentrations and distributions are not constant. Certain materials have higher concentrations of background radiation, and varying environmental and physical conditions can result in accumulations of background radiation. Information on environmental radioactivity is usually available from historic measurements identified during the IA.

If high levels of environmental radioactivity interfere with the disposition decision (e.g., action level less than environmental background, variability in environmental radioactivity determines level of survey effort), the planning team may consider moving the M&E being investigated to a location with less environmental radioactivity (see Sections 3.3.1.3 and 5.3). If the level of environmental radioactivity is unknown, it may be necessary to collect data during a preliminary survey (see Section 2.3) to provide this information.

B.2.1 Terrestrial Radioactivity

The naturally occurring forms of radioactive elements incorporated into the Earth during its formation that is still present are referred to as “terrestrial radionuclides.” The most significant terrestrial radionuclides include the uranium and thorium decay series, potassium-40 and rubidium-87. Virtually all materials found in nature contain some concentration of terrestrial radionuclides. Table B.1 lists average and typical ranges of concentrations of terrestrial radionuclides. Although the ranges in the table are typical, larger variations exist in certain areas (e.g., Colorado).

Bulk materials containing elevated concentrations of terrestrial radionuclides as well as equipment used to handle or process these materials should be identified during the IA even if these materials and equipment were not impacted by site activities.

Radon is an element that occurs as a gas in nature. Isotopes of radon are members of both the uranium and thorium natural decay series. These radon isotopes decay to produce additional radioactive isotopes, which are collectively called radon progeny.

Table B.1 Typical Average Concentration Ranges of Terrestrial Radionuclides

Material	Radium-226 (Bq/kg)[a]	Uranium-238 (Bq/kg)[a]	Thorium-232 (Bq/kg)[a]	Potassium-40 (Bq/kg)[a]
Soil, U.S.	40 (8-160)[b]	35 (4-140)[b]	35 (4-130)[b]	370 (100-700)[b]
Phosphate Fertilizer	200[c] - 100,000[d]	200-1,500[b]	20[b]	--
Concrete	(19-89)[e]	(19-89)[f]	(15-120)[f]	(260-1,100)[f]
Concrete Block	(41-780)[e]	(41-780)[f]	(37-81)[f]	(290-1,100)[f]
Brick	(4-180)[e]	(4-180)[f]	(1-140)[f]	(7-1,200)[f]
Coal Tar	(100-300)[e]	(100-300)[b]	--	--
Fly Ash-Bottom Ash	200[e]	200[b]	200[b]	--
Coal, U.S.	--	18 (1-540)[g]	21 (2-320)[g]	52 (1-710)[g]
Tile	--	(550-810)[h]	650[h]	--
Porcelain, Glazed	--	(180-37,000)[h, i]		--
Ceramic, Glazed[b]	(79-1,200)[h, i]			

[a] To convert Bq/kg to pCi/g, multiply by 0.027.

[b] UNSCEAR, Sources and Effects of Ionizing Radiation (UNSCEAR 2000).

[c] *Evaluation of EPA's Guidelines for Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM)* (EPA 2000).

[d] *Evaluation of Guidelines for Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM),* Committee on Evaluation of EPA Guidelines for Exposure to Naturally Occurring Radioactive Materials Board on Radiation Effects Research Commission on Life Sciences National Research Council, National Academy Press, p. 72 (NAS 1999).

[e] ^{226}Ra is assumed to be in secular equilibrium with ^{238}U.

[f] Eicholz G.G., Clarke F.J., and Kahn, B., *Radiation Exposure From Building Materials*, in "Natural Radiation Environment III," U.S. Department of Energy CONF-780422 (Eicholz 1980).

[g] Beck H.L., Gogolak C.V., Miller K.M., and Lowder W.M., *Perturbations on the Natural Radiation Environment Due to the Utilization of Coal as an Energy Source*, in "Natural Radiation Environment III," U.S. Department of Energy CONF-780422 (Beck 1980).

[h] Hobbs T.G., *Radioactivity Measurements on Glazed Ceramic Surfaces*, J. Res. Natl. Inst. Stand. Technol. **105**, 275-283 (Hobbs 2000).

[i] Values reported as total radioactivity without identification of specific radionuclides.

Radon emissions vary significantly over time based on a wide variety of factors. For example, relatively small changes in the relative pressure between the source material and the atmosphere (indoor or outdoor) can result in large changes in radon concentrations in the air. Soil moisture content also has an affect on the radon emanation rate.

Radon progeny tend to become fixed to solid particles in the air. These particles can become attached to surfaces as a result of electrostatic charge or gravitational settling. Air flow through ventilation ducts can produce an electrostatic charge that will attract these particles. A decrease in atmospheric pressure often precedes a rainstorm, which increases the radon emanation rate. Immediately prior to an electrical storm, an electrostatic charge can build up on equipment resulting in elevated radiation levels from radon progeny. Rainfall acts to scavenge these particles from the air, potentially resulting in elevated dose rates and surface activities during and immediately following rainfall.

^{210}Pb is a decay product of ^{222}Rn and ^{238}U. The 22-year half-life provides opportunities for buildup ^{210}Pb and progeny in sediments and low-lying areas. As mentioned previously, rain acts to scavenge radon progeny from the air. Areas where rain collects and concentrates can result in elevated levels of ^{210}Pb and progeny over time. In addition, lead is easily oxidized and can become fixed to surfaces through corrosion processes. Rust or oxide films on equipment can be indicators of locations with a potential for elevated background radioactivity.

B.2.2 Man-Made Radioactive Materials

Nuclear weapons testing and nuclear power reactors have produced large quantities of radionuclides through the fissioning of uranium and other heavy elements and the activation of various elements. Examples of man-made radionuclides that could be in the environment are ^{137}Cs, ^{90}Sr, and various isotopes of plutonium.

Prior to the 1963 Limited Test Ban Treaty, fallout from atmospheric nuclear tests distributed large quantities of man-made radionuclides around the globe. Following the 1963 treaty most nuclear weapons tests were conducted underground, although China and France continued atmospheric testing of nuclear weapons into the late 1970s. In 1996 a Comprehensive Test Ban Treaty was negotiated with the help of the United Nations. The Comprehensive Test Ban Treaty has not been ratified by China or the United States and was broken by Pakistan and India in 1998. However, worldwide fallout concentrations have been declining since the mid 1960s.

In 1964 a Department of Defense weather satellite containing a radiation source failed to achieve orbit. The Space Nuclear Auxiliary Power (SNAP) 9-A Radioisotopic Thermoelectric Generator (RTG) burned up on re-entry and dispersed the nuclear inventory (primarily plutonium-238) into the atmosphere. Incidents involving Soviet satellites with radioisotopes or nuclear reactors occurred in 1969, 1973, 1978, and 1983. In April 1986 there was a non-nuclear steam explosion and fire at the number four reactor at the nuclear power plant in Chernobyl in north-central Ukraine. Large quantities of radioactive material were released into the environment as a result of the catastrophe. The radionuclides from these incidents have been inhomogenously deposited around the world.

Isolated pockets with elevated concentrations of man-made radionuclides can still be found. For example, ventilation systems that were installed prior to 1963 collected fallout radionuclides. If these systems are still in use and the ducts have not been thoroughly cleaned, there is a potential for elevated background radiation. Another potential source of elevated background radiation from man-made radionuclides is wood ash. Trees filter and store some airborne pollutants, including ^{137}Cs from fallout. When the wood is burned the ^{137}Cs is concentrated in the wood ash. Materials or equipment associated with the ash could have elevated levels of background radiation.

B.2.3 Cosmic Radiation and Cosmogenic Radionuclides

Cosmic radiation consists of highly energetic particles that are believed to originate from phenomena such as solar flares and supernova explosions. The Earth’s atmosphere serves as a shield for these particles, although on rare occasions a solar flare is strong enough to produce a significant radiation dose in the lower reaches of the atmosphere.

Cosmic radiation is also responsible for the production of radioactive elements called cosmogenic radionuclides. These radionuclides are produced from collisions between the highly energetic cosmic radiation with stable elements in the atmosphere. Cosmogenic radionuclides include ^{3}H, ^{7}Be, ^{14}C, and ^{22}Na. Background concentrations of cosmic radiation and cosmogenic radionuclides generally do not impact disposition surveys.

B.3 Inherent Radioactivity

Inherent radioactivity, or intrinsic radioactivity, is radioactivity that is an integral part of the M&E being investigated. For example, concrete is made from materials that contain terrestrial radionuclides and is inherently radioactive. Some equipment is constructed from radioactive components, such as electron tubes or night vision goggles containing thorium components. Information on inherent radioactivity is usually obtained from process knowledge or historical measurements identified during the IA. Manufacturers of equipment that incorporates radioactive components can usually provide the radionuclide and the activity incorporated into the equipment. Information on radionuclides and activity levels for other types of equipment or bulk materials that are inherently radioactive is usually more generic. Table B.1 lists ranges of terrestrial radionuclide concentrations in some common materials (e.g., concrete, soil, brick). The wide range of radionuclide concentrations observed in these materials prevents establishing any general rules of thumb, so it is usually necessary to obtain project-specific information. For release scenarios, it is strongly recommended that all M&E be surveyed before it enters a controlled area. This provides project-specific information on inherent radioactivity and minimizes complications when designing the disposition survey. For interdiction scenarios, it is important to understand the types of M&E being investigated and the potential for inherent radioactivity. It may be necessary to establish an administrative action level that defines the upper end of acceptable inherent radioactivity for different types of M&E (see Section 3.2).

B.4 Instrument Background

Instrument background is a combination of radioactivity in the constituent materials of the detector, ancillary equipment, and shielding, and electronic noise contributing to the instrument response. Instruments designed to measure low levels of radioactivity are generally constructed from materials with very low levels of inherent radioactivity to minimize instrument background. The electronics in radiation instruments are also designed to minimize the signal-to-noise ratio, also reducing instrument background. Instrument background becomes the primary contributor to background only for radionuclide-specific measurements for radionuclides not contributing to environmental or inherent background (e.g., ^{60}Co in bulk soil measured by gamma spectroscopy). Note that radiation from M&E can interact with instrument shielding to produce secondary effects that may contribute to instrument background (e.g., Compton backscatter, generation of secondary photons and characteristic X-rays, photoelectric absorption). Additional information on instrument background is available in Chapter 20 of *Radiation Detection and Measurement* (Knoll 1999).

B.5 Technologically Enhanced Naturally Occurring Radioactive Material

Technologically Enhanced Naturally Occurring Radioactive Material (TENORM) is any naturally occurring material not subject to regulation under the Atomic Energy Act whose radionuclide concentrations or potential for human exposure have been increased above levels encountered in the natural state by human activities (NAS 1999). Some industrial processes involving natural resources concentrate naturally occurring radionuclides, producing TENORM.

Much TENORM contains only trace amounts of radioactivity and is part of our everyday landscape. Some TENORM, however, contains very high concentrations of radionuclides. The majority of radionuclides in TENORM are found in the uranium and thorium natural decay series. Potassium-40 is also associated with TENORM. Radium and radon are typically measured as indicators for TENORM in the environment. TENORM is found in many industrial waste streams (e.g., scrap metal, sludges, slags, fluids) and is being discovered in industries traditionally not thought of as being affected by radiation. Examples of products and processes affected by TENORM include:

- Uranium overburden and mine spoils
- Phosphate industrial wastes
- Phosphate fertilizers and potash
- Coal ash, slag, cinders
- Oil and gas production scale and sludge
- Sludge and other waste materials from treatment of drinking water and waste water
- Metal mining and processing waste
- Geothermal energy production waste
- Paper and pulp
- Scrap metal recycling
- Slag from industrial processes (metal and non-metal)
- Abrasive mineral sands
- Cement production

Radon and radon progeny are concerns when dealing with TENORM. Radon-222 is a decay product of ^{238}U. The 3.8-day half-life means that ^{222}Rn is capable of migrating through several decimeters of soil or building materials and reaching the atmosphere before it decays. The radioactive progeny of unsupported ^{222}Rn have short half-lives (e.g., 27 minutes for ^{214}Pb) and usually decay to background levels within a few hours. ^{220}Rn, which has a 55-second half-life, is a decay product of ^{232}Th. The short half-life limits the mobility of ^{220}Rn since it decays before it can migrate to the atmosphere. However, ^{232}Th activity that is located on or very near the surface can produce significant quantities of ^{220}Rn in the air. The radioactive progeny of unsupported ^{220}Rn can result in elevated levels of surface radioactivity for materials and equipment used or stored in these areas. The 10.6-hour half-life of ^{212}Pb means that this surface radioactivity could take a week or longer to decay to background levels.

B.6 Orphan Sources

Radiation sources are found in certain types of specialized industrial devices, such as those used for measuring the moisture content of soil and for measuring density or thickness of materials. Usually, a small quantity of the radioactive material is sealed in a metal casing and enclosed in a housing that prevents the escape of radiation. These sources present no health risk from radioactivity as long as the sources remain sealed, the housing remains intact, and the devices are handled and used properly.

If equipment containing a sealed source is disposed of improperly or sent for recycling as scrap metal, the sealed source may be 'lost' and end up in a metal recycling facility or in the possession of someone who is not licensed to handle the source. Specially licensed sources bear identifying markings that can be used to trace these sources to their original owners. However, some sources do not have these markings or the markings become obliterated. In these cases, the sources are referred to as 'orphan sources' because no known owner can be identified. They are one of the most frequently encountered sources of radioactivity in shipments received by scrap metal facilities.

Scrap yards and disposal sites attempt to detect orphan sources and other contaminated metals by screening incoming materials with sensitive radiation detectors before they can enter the processing stream and cause contamination. Housings that make the sources safe also make detection difficult. Further, if the source is buried in a load of steel, the steel acts as further shielding and thus these sources may elude detection. Consequently, there is always a potential for sources to become mixed within and impact scrap metal.

C. EXAMPLES OF COMMON RADIONUCLIDES

Table C.1 Examples of Common Radionuclides at Selected Types of Facilities

Facility Type	Common Radionuclides
Accelerator/Cyclotron	^{22}Na Activation products (e.g., ^{60}Co)
Aircraft Manufacturing and Maintenance Facility	^{3}H (dials and gauges) Magnesium-thorium alloys Nickel-thorium alloys ^{137}Pm (lighted dials and gauges) ^{226}Ra and progeny (radium dials) Depleted uranium
Cement Production Facility	Thorium series radionuclides Uranium series radionuclides
Ceramic Manufacturing Facility	Thorium series radionuclides Uranium series radionuclides
Fertilizer Plant	^{40}K Uranium series radionuclides
Fuel Fabrication Facility	^{99}Tc (reprocessing only) Enriched uranium Transuranics (e.g., ^{237}Np, ^{239}Pu) (reprocessing only)
Gaseous Diffusion Plant	^{99}Tc Enriched uranium Transuranics (e.g., ^{237}Np, ^{239}Pu)

Table C.1 Examples of Common Radionuclides at Selected Types of Facilities (continued)

Facility Type	Common Radionuclides
Medical Imaging and Therapy Facility	^{60}Co ^{90}Sr ^{99m}Tc ^{131}I ^{137}Cs ^{192}Ir ^{201}Tl ^{226}Ra and progeny Depleted uranium collimators
Metal Foundry	^{40}K ^{60}Co ^{137}Cs Thorium series radionuclides Uranium series radionuclides
Munitions and Armament Manufacturing and Testing Facility	^{3}H (fire control devices) ^{226}Ra and progeny Depleted uranium
Nuclear Medicine Laboratory or Pharmaceutical Laboratory	^{99m}Tc ^{131}I ^{137}Cs ^{192}Ir ^{201}Tl ^{226}Ra and progeny

Table C.1 Examples of Common Radionuclides at Selected Types of Facilities (continued)

Facility Type	Common Radionuclides
Nuclear Power Reactor	Activation products (e.g., ^{55}Fe, ^{60}Co, ^{63}Ni) Fission products (e.g., ^{90}Sr, ^{137}Cs) Transuranics (e.g., ^{237}Np, ^{239}Pu)
Oil and Gas	^{226}Ra and progeny
Optical Glass Facility	Thorium series radionuclides Uranium series radionuclides
Paint and Pigment Manufacturing Facility	Thorium series radionuclides Uranium series radionuclides
Paper and Pulp Facility	Thorium series radionuclides Uranium series radionuclides
Radium Dial Painting	^{226}Ra and progeny
Rare Earth Facility	^{40}K Thorium series radionuclides Uranium series radionuclides
R&D Facility with Broad Scope License	^{3}H ^{14}C

Table C.1 Examples of Common Radionuclides at Selected Types of Facilities (continued)

Facility Type	Common Radionuclides
Research Laboratory	^{3}H ^{14}C ^{22}Na ^{24}Na ^{32}P ^{57}Co ^{63}Ni ^{123}I ^{125}I
Scrap Metal Recycling Facility	^{60}Co ^{90}Sr ^{137}Cs ^{226}Ra and progeny Thorium series radionuclides Uranium series radionuclides
Sealed Source Facility	^{60}Co ^{90}Sr ^{137}Cs ^{241}Am
Transuranic Facility	^{241}Am $^{238, 239, 240, 241}$Pu

Table C.1 Examples of Common Radionuclides at Selected Types of Facilities (continued)

Facility Type	Common Radionuclides
Uranium Mill	^{238}U ^{230}Th ^{226}Ra and progeny Thorium series radionuclides Uranium series radionuclides
Waste Water Treatment Facility	Thorium series radionuclides Uranium series radionuclides
Widely Distributed General Commerce	^{3}H (exit signs) ^{40}K (naturally-occurring) ^{57}Co (lead paint analyzer) ^{60}Co (radiography source) ^{63}Ni (chemical agent detectors) ^{109}Cd (lead paint analyzer) ^{137}Cs (soil moisture density gauge, liquid level gauge) ^{192}Ir (radiography source) ^{226}Ra (watch dials) ^{241}Am (AmBe soil moisture density gauge, smoke detectors) Orphan sources

D. INSTRUMENTATION AND MEASUREMENT TECHNIQUES

D.1 Introduction

This appendix provides information on various field and laboratory equipment used to measure radiation levels and radioactive material concentrations. The descriptions provide information pertaining to the general types of available radiation detectors and the ways in which those detectors are utilized for various circumstances. Similar information may be referenced from MARSSIM Appendix H, Description of Field Survey and Laboratory Analysis Equipment (MARSSIM 2002), and NUREG-1761 Appendix B, Advanced/Specialized Information (NRC 2002). The information in this appendix is specifically designed to assist the user in selecting the appropriate radiological instrumentation and measurement technique during the implementation phase of the Data Life Cycle (Chapter 5).

The following topics will be discussed for each instrumentation and measurement technique combination:

- **Instruments** – a description of the equipment and the typical detection instrumentation it employs
- **Temporal Issues** – a synopsis of time constraints that may be encountered through use of the measurement technique
- **Spatial Issues** – limitations associated with the size and portability of the instrumentation as well as general difficulties that may arise pertaining to source-to-detector geometry
- **Radiation Types** – applicability of the measurement technique for different types of ionizing radiation
- **Range** – the associated energy ranges for the applicable types of ionizing radiation
- **Scale** – typical sizes for the M&E applicable to the measurement technique

- **Ruggedness –** a summary of the durability of the instrumentation (note that this is frequently limited by the detector employed by the instrumentation; e.g., an instrument utilizing a plastic scintillator is inherently more durable than an instrument utilizing a sodium iodide crystal); suitable temperature ranges for proper operation of the instrumentation and measurement technique have been provided where applicable

D.2 General Detection Instrumentation

This section summarizes the most common detector types used for the detection of ionizing radiation in the field. This will discuss many of the detector types incorporated into the measurement methods that are described in later sections of this chapter.

D.2.1 Gas-Filled Detectors

Gas-filled detectors are the most commonly-used radiation detectors and include gas-ionization chamber detectors, gas-flow proportional detectors, and Geiger-Muller (GM) detectors. These detectors can be designed to detect alpha, beta, photon, and neutron radiation. They generally consist of a wire passing through the center of a gas-filled chamber with metal walls, which can be penetrated by photons and high-energy beta particles. Some chambers are fitted with mylar windows to allow penetration by alpha and low-energy beta radiation. A voltage source is connected to the detector with the positive terminal connected to the wire and the negative terminal connected to the chamber casing to generate an electric field, with the wire serving as the anode, and the chamber casing serving as the cathode. Radiation ionizes the gas as it enters the chamber, creating free electrons and positively-charged ions. The number of electrons and positively-charged ions created is related to the properties of the incident radiation type (alpha particles produce many ion pairs in a short distance, beta particles produce fewer ion pairs due to their smaller size, and photons produce relatively few ion pairs as they are uncharged and interact with the gas significantly less than alpha and beta radiation). The anode attracts the free electrons while the cathode attracts the positively charged ions. The reactions between these ions and free electrons with either the anode or cathode produce disruptions in the electric field. The voltage applied to the chamber can be separated into different voltage ranges that distinguish the types of gas-filled detectors described below. The different types of gas-filled detectors are described in ascending order of applied voltage.

D.2.1.1 Ionization Chamber Detectors

Ionization chamber detectors consist of a gas-filled chamber operated at the lowest voltage range of all gas-filled detectors.[1] Ionization detectors utilize enough voltage to provide the ions with sufficient velocity to reach the anode or cathode. The signal pulse heights produced in ionization chamber detectors is small and can be discerned by the external circuit to differentiate between different types of radiation. These detectors provide true measurement data of energy deposited proportional to the charge produced in air, unlike gas-flow proportional and GM detectors which are detection devices. These detectors are generally designed to collect cumulative beta and photon radiation without amplification and many have a beta shield to help distinguish between these radiation types. These properties make ionization detectors excellent choices for measuring exposure rates from photon emission radiation in roentgens. These detectors can be deployed for an established period of time to collect data in a passive manner for disposition surveys. Ionization chamber detectors may assist in collecting measurements in inaccessible areas due to their availability in small sizes.

Another form of the ionization chamber detector is the pressurized ion chamber (PIC). As with other ionization chamber detectors, the PIC may be applied for M&E disposition surveys when a exposure-based action level is used. The added benefit of using PICs is that they can provide more accurate dose measurements because they compensate for the various levels of photon energies as opposed to other exposure rate meters (e.g., micro-rem meter), which are calibrated to a ^{137}Cs source. PICs can be used to cross-calibrate other exposure rate detectors applicable for surveying M&E, allowing the user to

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[1] At voltages below the ionization chamber voltage range, ions will recombine before they can reach either the cathode or anode and do not produce a discernable disruption to the electric field.

compensate for different energy levels and reduce or eliminate the uncertainty of underestimating or overestimating the exposure rate measurements.

D.2.1.2 Gas-Flow Proportional Detectors

The voltage applied in gas-flow proportional detectors is the next range higher than ionization chamber detectors, and is sufficient to create ions with enough kinetic energy to create new ion pairs, called secondary ions. The quantity of secondary ions increases proportionally with the applied voltage, in what is known as the gas amplification factor. The signal pulse heights produced can be discerned by the external circuit to differentiate between different types of radiation. Gas-flow proportional detectors are generally used to detect alpha and beta radiation. Systems also detect photon radiation, but the detection efficiency for photon emissions is considerably lower than the relative efficiencies for alpha and beta activity. Physical probe areas for these types of detectors vary in size from approximately 100 cm^2 up to 600 cm^2. The detector cavity in these instruments is filled with P-10 gas which is an argon-methane mixture (90% argon and 10% methane). Ionizing radiation enters this gas-filled cavity through an aluminized mylar window. Additional mylar shielding may be used to block alpha radiation; a lower voltage setting may be used to detect pure alpha activity (NRC 1998b).

D.2.1.3 Geiger-Mueller Detectors

GM detectors operate in the voltage range above the proportional range and the limited proportional range.[2] This range is characterized by extensive gas amplification that results in what is referred to as an "avalanche" of ion and electron production. This mass production of electrons spreads throughout the entire chamber, which precludes the ability to distinguish between different kinds of radiation because all of the signals produced are the same size. GM detectors are most commonly used for the detection of beta activity, though they may also detect both alpha and photon radiation. GM detectors have relatively short response and dead times and are sensitive enough to broad

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[2] The limited proportional range produces secondary ion pairs but does not produce reactions helpful for radiation detection, because the gas amplification factor is no longer constant.

detectable energy ranges for alpha, beta, photon, and neutron emissions (though they cannot distinguish which type of radiation produces input signals) to allow them to be used for surveying M&E with minimal process knowledge.[3]

GM detectors are commonly divided into three classes: "pancake", "end-window", and "side-wall" detectors. GM pancake detectors (commonly referred to as "friskers") have wide diameter, thin mica windows (approximately 15 cm^2 window area) that are large enough to allow them to be used to survey many types of M&E. Although GM pancake detectors are referenced beta and gamma detectors, the user should consider that their beta detection efficiency far exceeds their gamma detection efficiency. The end-window detector uses a smaller, thin mica window and is designed to allow beta and most alpha particles to enter the detector unimpeded for concurrent alpha and beta detection. The side-wall detector is designed to discriminate between beta and gamma radiation, and features a door that can be slid or rotated closed to shield the detector from beta emissions for the sole detection of photons. These detectors require calibration to detect for beta and gamma radiation separately. Energy-compensated GM detectors may also be cross-calibrated for assessment of exposure rates.

D.2.2 Scintillation Detectors

Scintillation detectors (sometimes referred to as "scintillators") consist of scintillation media that emits a light "output" called a scintillation pulse when it interacts with ionizing radiation. Scintillators emit low-energy photons (usually in the visible light range) when struck by high-energy charged particles; interactions with external photons cause scintillators to emit charged particles internally, which in turn interact with the crystal to emit low-energy photons. In either case, the visible light emitted (i.e., the low-energy photons) are converted into electrical signals by photomultiplier tubes and recorded by a digital readout device. The amount of light emitted is generally

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[3] GM detectors may be designed and calibrated to detect alpha, beta, photon, and neutron radiation, though they are much better-suited for the detection of charged particles (i.e., alpha and beta particles) than neutral particles (i.e., photons and neutrons).

proportional to the amount of energy deposited, allowing for energy discrimination and quantification of source radionuclides in some applications.

D.2.2.1 Zinc Sulfide Scintillation Detectors

Zinc sulfide detector crystals are only available as a polycrystalline powder that are arranged in a thin layer of silver-activated zinc sulfide (ZnS(Ag)) as a coating or suspended within a layer of plastic scintillation material. The use of these thin layers makes them inherently-dispositioned for the detection of high linear energy transfer (LET) radiation (radiation associated with alpha particles or other heavy ions). These detectors use an aluminized mylar window to prevent ambient light from activating the photomultiplier tube (Knoll 1999). The light pulses produced by the scintillation crystals are amplified by a photomultiplier tube, converted to electrical signals, and counted on a digital scaler/ratemeter. Low LET radiations (particularly beta emissions) are detected at much lower detection efficiencies than alpha emissions and pulse characteristics may be used to discriminate beta detections from alpha detections.

D.2.2.2 Sodium Iodide Scintillation Detectors

Sodium iodide detectors are well-suited for detection of photon radiation. Energy-compensated sodium iodide detectors may also be cross-calibrated for assessment of exposure rates. Unlike ZnS(Ag), sodium iodide crystals can be grown relatively large and machined into varying shapes and sizes. Sodium iodide crystals are activated with trace amounts of thallium (hence the abbreviation NaI(Tl)), the key ingredient to the crystal's excellent light yield (Knoll, 1999). These instruments most often have upper- and lower-energy discriminator circuits and when used correctly as a single-channel analyzer, can provide information on the photon energy and identify the source radionuclides. Sodium iodide detectors can be used with handheld instruments or large stationary radiation monitors.

D.2.2.3 Cesium Iodide Scintillation Detectors

Cesium iodide detectors are generally similar to sodium iodide detectors. Like NaI(Tl), cesium iodide may be activated with thallium (CsI(Tl)) or sodium (CsI(Na)). Cesium iodide is more resistant to shock and vibration damage than NaI, and when cut into thin sheets it features malleable properties allowing it to be bent into various shapes. CsI(Tl) has variable decay times for various exciting particles, allowing it to help differentiate between different types of ionizing radiation. A disadvantage of CsI scintillation detectors is due to the fact that the scintillation emission wavelengths for CsI are longer than those produced by sodium iodide crystals; since almost all photomultiplier tubes are designed for NaI, there are optical incompatibilities that result in decreased intrinsic efficiencies for CsI detectors. Additionally, CsI scintillation detectors feature relatively long response and decay times for luminescent states in response to ionizing radiation (Knoll 1999).

D.2.2.4 Plastic Scintillation Detectors

Plastic scintillators are composed of organic scintillation material that is dissolved in a solvent and subsequently hardened into a solid plastic. Modifications to the material and specific packaging allow plastic scintillators to be used for detecting alpha, beta, photon, or neutron radiation. While plastic scintillators lack the energy resolution of sodium iodide and some other gamma scintillation detector types, their relatively low cost and ease of manufacturing into almost any desired shape and size enables them to offer versatile solutions to atypical radiation detection needs (Knoll 1999).

D.2.3 Solid State Detectors

Solid state detection is based on ionization reactions within detector crystals composed of an electron-rich (n-type or electron conductor) sector and an electron-deficient (p-type or hole conductor) sector. Reverse-bias voltage is applied to the detector crystal; forming a central region absent of free charge (this is termed the depleted region). When a particle enters this region, it interacts with the crystal structure to form hole-electron pairs. These holes and electrons are swept out of the depletion region to the positive and negative electrodes by the electric field, and the magnitude of the resultant pulse in the external circuit is directly proportional to the energy lost by the ionizing radiation in the depleted region.

Solid state detection systems typically employ silicon or germanium crystals[4] and utilize semiconductor technology (i.e., a substance whose electrical conductivity falls between that of a metal and that of an insulator, and whose conductivity increases with decreasing temperature and with the presence of impurities). Semiconductor detectors are cooled to extreme temperatures to utilize the crystal material's insulating properties to prevent thermal generation of noise. The use of semiconductor technology can achieve energy resolutions, spatial resolutions, and signal-to-noise ratios superior to those of scintillation detection systems.

D.3 Counting Electronics

Instrumentation requires a device to accumulate and record the input signals from the detector over a fixed period of time. These devices are usually electronic, and utilize scalers or rate-meters to display results representing the number of interaction events (between the detector and radionuclide emissions) within a period of time (e.g., counts per minute). A scaler represents the total number of interactions within a fixed period of time, while a rate-meter provides information that varies based on a short-term average of the rate of interactions.

Scalers represent the simpler of these two counting approaches, because they record a single count each time an input signal is received from the detector. Scaling circuits are typically designed with scalers to allow the input signals to be cut by factors of 10, 100, or 1,000 to allow the input signals to be counted directly by electromechanical registers when counting areas with elevated radioactivity. Scalers are generally used when taking in situ measurements and are used to determine average activities.

Contemporary rate-meters utilize analog-to-digital converters to sample the pulse amplitude of the input signal received from the detector and convert it to a series of digital values. These digital values may then be manipulated using digital filters (or shapers) to average or "smooth" the data displayed. The counting-averaging technique

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[4] Solid state detection systems may also utilize crystals composed of sodium iodide, cesium iodide, or cadmium zinc telluride in non-semiconductor applications.

used by rate-meters may be more helpful than scalers in identifying elevated activity. When using scalers in performing scanning surveys to locate areas of elevated activity, small areas of elevated activity may appear as very quick "blips" that are difficult to discern, while rate-meters continue to display heightened count rates once the detector has moved past the elevated activity, and display "ramped up" count rates immediately preceding the elevated activity as well. Rate-meters have the inherent limitation in that the use of their counting electronics varies the signals displayed by the meter since they represent a short-term average of the event rate. It is conceivable that very small areas of elevated activity (e.g., particle) might have their true activity concentrations "diluted" by the averaging of rate-meter counting electronics.

D.4 Hand-Held Instruments

This section discusses hand-held instruments, which may be used for in situ measurements or scanning surveys.

D.4.1 Instruments

In situ measurements with hand-held instruments are typically conducted using the detector types described in Section D.2. These typically are composed of a detection probe (utilizing a single detector) and an electronic instrument to provide power to the detector and to interpret data from the detector to provide a measurement display.

The most common types of hand-held detector probes are GM detectors, ZnS(Ag) alpha/beta scintillation detectors, and NaI(Tl) photon scintillation detectors. There are instances of gas-flow proportional detectors as hand-held instruments, though these are not as common since these detectors operate using a continuous flow of P-10 gas, and the accessories associated with the gas (e.g., compressed gas cylinders, gauges, tubing) make them less portable for use in the field.

D.4.2 Temporal Issues

Hand-held instruments generally have short, simple equipment set-ups requiring minimal time, often less than ten minutes. In situ measurement count times typically range from 30 seconds to two minutes. Longer count times may be utilized to increase resolution and provide lower minimum detectable limits. Typical scanning speeds are approximately 2.5 centimeters per second. Slower scanning speeds will aid in providing lower minimum detectable concentrations.

D.4.3 Spatial Issues

Detectors of hand-held instruments are typically small and portable, having little trouble fitting into and measuring most M&E. Spatial limitations are usually based on the physical size of the probe itself. The user must be wary of curved or irregular surfaces of M&E being surveyed. Detector probes generally have flat faces and incongruities between the face of the detector and the M&E being surveyed have an associated uncertainty. ZnS scintillation and gas-flow proportional detectors are known to have variations in efficiency of up to 10% across the face of the detector. Therefore, the calibration source used should have an area at least the size of the active probe area.

D.4.4 Radiation Types

Assortments of hand-held instruments are available for the detection of alpha, beta, photon, and neutron radiations. Table D.1 illustrates the potential applications for the most common types of hand-held instruments.

D.4.5 Range

The ranges of detectable energy using hand-held instruments are dependent upon the type of instrument selected and type of radiation. Some typical detectable energy ranges for common hand-held instruments are listed above in Table D.1. More detailed information pertaining to the ranges of detectable energy using hand-held instruments are available in the European Commission for Nuclear Safety and the Environment Report 17624 (EC 1998).

Table D.1 Potential Applications for Common Hand-Held Instruments

	Alpha	Beta	Photon	Neutron	Detectable Energy Range	
					Low End Boundary	High End Boundary
Ionization chamber detectors	NA	FAIR	GOOD	NA	40-60 keV	1.3-3 MeV
Gas-flow proportional detectors	GOOD	GOOD	POOR	POOR	5-50 keV	8-9 MeV
Geiger-Muller detectors	FAIR	GOOD	POOR	POOR	30-60 keV	1-2 MeV
ZnS(Ag) scintillation detectors	GOOD	POOR	NA	NA	30-50 keV	8-9 MeV
NaI(Tl) scintillation detectors	NA	POOR	GOOD	NA	40-60 keV	1.3-3 MeV
NaI(Tl) scintillation detectors (thin detector, thin window)	NA	FAIR	GOOD	NA	10 keV	60-200 keV
CsI(Tl) scintillation detectors	NA	POOR	GOOD	NA	40-60 keV	1.3-3 MeV
Plastic scintillation detectors	NA	FAIR	GOOD	NA	40-60 keV	1.3-3 MeV
BF_3 proportional detectors[5]	NA	NA	NA	GOOD	0.025 eV	100 MeV
^{3}He proportional detectors[5]	NA	NA	POOR	GOOD	0.025 eV	100 MeV

Notes:

GOOD	The instrument is well-suited for detecting this type of radiation
FAIR	The instrument can adequately detect this type of radiation
POOR	The instrument may be poorly-suited for detecting this type of radiation
NA	The instrument cannot detect this type of radiation

D.4.6 Scale

There is no definitive limit to the size of an object to be surveyed using hand-held instruments. Hand-held instruments may generally be used to survey M&E of any size; constraints are only placed by the practical sizing of M&E related to the sensitive area of the probe. Limitations may also be derived from the physical size of the detector probes

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[5] The use of moderators enables the detection of high-energy fast neutrons. Either BF_3 or ^{3}He gas proportional detectors may be used for the detection of fast neutrons, but ^{3}He are much more efficient in performing this function. BF_3 detectors discriminate against gamma radiation more effectively than ^{3}He detectors.

used for surveying. The largest hand-held detector probes feature effective detection surface areas of approximately 175 to 200 cm^2. Detection probes larger than this may be of limited use with hand-held instruments.

D.4.7 Ruggedness

All varieties of hand-held instruments discussed here are typically calibrated for use in temperatures with lower ranges from -30 ° to -20 °C and upper ranges from 50 ° to 60 °C. The durability of a hand-held instrument depends largely upon the detection media (crystals, such as sodium iodide and germanium crystals are fragile and vulnerable to mechanical and thermal shock) and the presence of a mylar (or similar material) window:

- **Ionization chamber detectors –** ionization chamber detectors are susceptible to physical damage and may provide inaccurate data (including false positives) if exposed to mechanical shock.
- **Gas-flow proportional detectors –** detection gas used with gas-flow proportional detectors may leak from seals such that these detectors are usually operated in the continuous gas flow mode; the use of flow meter gauges to continuously monitor the gas flow rate is recommended along with frequent quality control checks to ensure the detector still meets the required sensitivity; gas-flow proportional detectors may also use fragile mylar windows to contain the detection gases, which renders the detectors vulnerable to puncturing and mechanical shock.
- **Geiger-Muller detectors –** GM tubes typically use fragile mylar windows to contain the detection gases; the presence of a mylar window renders the detector vulnerable to puncturing and mechanical shock.
- **ZnS(Ag) scintillation detectors –** zinc sulfide is utilized as thin-layer polycrystalline powder in detectors and are noted for being vulnerable to mechanical shock; zinc sulfide detectors may use fragile mylar windows, in which case the detector is vulnerable to puncturing and mechanical shock.
- **NaI(Tl) scintillation detectors –** sodium iodide crystals are relatively fragile and can be damaged through mechanical shock; sodium iodide is also highly

hydroscopic such that the crystals must remain environmentally sealed within the detector housing.

- **Plastic Scintillation Detectors** – plastic scintillators are typically robust and resistant to damage from mechanical and thermal shock.

D.5 Volumetric Counters (Drum, Box, Barrel, Four Pi Counters)

The term Box Counter is a generic description for a radiation measurement system that typically involves large area, four pi (4π) radiation detectors and includes the following industry nomenclature: tool counters, active waste monitors, surface activity measurement systems, and bag/barrel/drum monitors. Box counting systems are most frequently used for conducting in situ surveys of M&E that is utilized in radiologically-controlled areas. These devices are best-suited for performing gross activity screening measurements on Class 2 and Class 3 M&E (NRC 2002). Typical items to be surveyed using box counters are hand tools, small pieces of debris, bags of trash, and waste barrels. Larger variations of box counting systems can count objects up to a few cubic meters in size.

D.5.1 Instruments

Box counting systems typically consist of a counting chamber, an array of detectors configured to provide a 4π counting geometry, and microprocessor-controlled electronics that allow programming of system parameters and data-logging. Systems typically survey materials for photon radiation and usually incorporate a shielded counting chamber and scintillation detectors (plastic scintillators or sodium iodide scintillation detectors). These systems most commonly utilize four or six detectors, which are situated on the top, bottom, and sides of the shielded counting chamber (Figure D.1). Some systems monitor M&E for beta activity, using a basic design similar to photon radiation detection systems, but utilizing gas-flow proportional counters. In rare cases, neutron detection has been used for criticality controls and counter-proliferation screening.

Figure D.1 Example Volumetric Counter (Thermo 2005)

Box counting systems for alpha activity feature a substantial departure in design from beta/gamma detection systems. Alpha activity systems do not require heavy shielding to filter out ambient sources of radiation. These devices utilize air filters to remove dust and particulates from air introduced into counting chambers that incorporate airtight seals. Filtered air introduced into the counting chamber interacts with any surface alpha activity associated with the M&E.

Each alpha interaction with a surrounding air molecule produces an ion pair. These ion pairs are produced in proportion to the alpha activity per unit path length. This air (i.e., the ion pairs in the air) is then counted using an ion detector for quantification of the specific activity. The specific activity of the air in the counting chamber provides a total surface activity quantification for the M&E (BIL 2005).

D.5.2 Temporal Issues

Typically, box counting systems require approximately one to 100 seconds to conduct a measurement (Thermo 2005). The count times are dependent on a number of factors to include required measurement sensitivity and background count rates with accompanying subtraction algorithms. The count times for box counting are typically considered relatively short for most disposition surveys.

D.5.3 Spatial Issues

Since box counters typically average activity over the volume or mass of the M&E, the spatial distribution of radioactivity may be a significant limitation on the use of this measurement technique. The design of box counting systems is not suited to the identification of localized elevated areas, and therefore may not be the ideal choice when the disposition criteria is not based on average or total activity.

Some systems incorporate a turntable inside the counting chamber to improve measurement of difficult-to-measure areas or for heterogeneously distributed radioactivity. When practical, performing counts on objects in two different orientations (i.e., by rotating the M&E 90 or 180 degrees and performing a subsequent count) will yield more thorough and defensible data.

Proper use of box counters includes segregating the M&E to be surveyed and promoting accurate measurements through uniform placement of items to be surveyed in the counting chamber. For example, a single wrench placed on its side in a box counter has different geometric implications from a tool of similar size standing up inside the counting chamber. Counting jigs for sources and M&E to be surveyed are frequently employed to facilitate consistent, ideal counting positions between the M&E and the counting chamber detector array.

D.5.4 Radiation Types

Box counting systems are intrinsically best-suited for the detection of moderate- to high-energy photon radiation. As described in Section D.5.1, specific systems may be designed for the detection of low-energy photon, beta, alpha, and in some cases neutron radiation. For proper calibration and utilization of box counters, it is often necessary to establish the radiation types and anticipated energy ranges prior to measurement.

D.5.5 Range

Photon radiation can typically be measured within a detectable energy range of 40 to 60 keV up to 1.3 to 3 MeV. For example, typical box counters positioned at radiological control area exit points are configured to alarm at a set point of 5,000 dpm total activity. The precise count time is adjusted automatically by setting the predetermined count rate to limit the error. Measurement times will range from 5 to 45 seconds in order to complete counts of this kind, depending on current background conditions (Thermo 2005). Lower detection capabilities are achievable by increasing count times or incorporating background reduction methodologies.

D.5.6 Scale

Size limitations pertaining to the M&E to be surveyed are inherently linked to the physical size of the counting chamber. Smaller box counting systems have a counting chamber of less than 0.028 cubic meters (approximately one cubic foot) and are often used for tools and other frequently-used small items. The maximum size of box counters is typically driven by the logistics of managing the M&E to be measured, and this volume is commonly limited to a 55-gallon waste drum. Some box counting systems allow counts to be performed on oversized items protruding from the counting chamber with the door open.

D.5.7 Ruggedness

Many volumetric counter models feature stainless steel construction with plastic scintillation detectors and window-less designs, which translates to a rugged instrument that is resistant to mechanical shock.

D.6 Conveyorized Survey Monitoring Systems

Conveyorized survey monitoring systems automate the routine scanning of M&E. Conveyorized survey monitoring systems have been designed to measure materials such as soil, clothing (laundry monitors), copper chop (small pieces of copper), rubble, and debris. Systems range from small monitoring systems comprised of a single belt that passes materials through a detector array, to elaborate multi-belt systems capable of measuring and segregating material while removing extraneously-large items. The latter type comprises systems that are known as segmented gate systems. These automated scanning systems segregate materials by activity by directing material that exceeds an established activity level onto a separate conveyor. Simpler conveyorized survey monitoring systems typically feature an alarm/shut-down feature that halts the conveyor motor and allows for manual removal of materials that have exceeded the established activity level.

D.6.1 Instruments

A typical conveyorized survey monitoring system consists of a motorized conveyor belt that passes materials through an array of detectors, supporting measurement electronics, and an automated data-logging system (Figure D.2). Systems typically survey materials for photon radiation and usually incorporate scintillation detectors (plastic scintillators or sodium iodide scintillation detectors) or high-purity germanium detectors. Scintillation detector arrays are often chosen for gross gamma activity screening. Conveyorized survey monitoring systems designed to detect radionuclide mixtures with a high degree of process knowledge work best using plastic scintillators, while systems categorizing material mixtures where the radionuclide concentrations are variable are better-suited to the use of sodium iodide scintillation detectors. Conveyorized survey monitoring systems designed for material mixtures where the radionuclide concentrations are unknown may be suitable for more expensive and maintenance-intensive high-purity germanium detector arrays, which will allow for quantitative measurement of complex photon energy spectra. An alternative method for screening materials for different photon energy regions of interest is to incorporate sodium iodide detector arrays with crystals of varying thickness to target multiple photon energies. Systems may also be fitted with gas flow proportional counters for the detection of alpha and beta emissions. Laundry conveyorized survey monitoring systems are typically designed for the detection of alpha and beta radiation, as the nature of clothes allows the survey media to be compressed, allowing the detector arrays to be close to or in contact with the survey media.

Figure D.2 Example Conveyorized Survey Monitoring System (Laurus 2001)

D.6.2 Temporal Issues

Typically, conveyorized survey monitoring systems require approximately one to six seconds to count a given field of detection (Novelec 2001a). Systems are designed to provide belt speeds ranging from 0.75 meters up to 10 meters (2.5 to 33 feet) per minute to accommodate the necessary response time for detection instrumentation (Thermo 2006; Eberline 2004). This yields processing times of 15 to 45 metric tons (16 to 50 tons) of material per hour for soil or construction demolition-type material conveyorized survey monitoring systems (NRC 2002).

D.6.3 Spatial Issues

The M&E that are typically surveyed by conveyorized survey monitoring systems may contain difficult-to-measure areas. Most systems employ the detector arrays in a staggered, off-set configuration, which allows the sensitive areas of the detectors to overlap with respect to the direction of movement. This off-set configuration helps to eliminate blind spots (i.e., locations where activity may be present but cannot be detected because the radiation cannot reach the detectors). Some systems are designed specifically for materials of relatively small particles of uniform size (e.g., soil), while others have been designed to accommodate heterogeneous materials like rubble and debris.

The data logging system accepts the signal pulses from the detector systems and stores the pulse data in counting scalers. The recorded values are continuously compared with pre-set alarm values corresponding to the selected action level(s). The detectors incorporate integral amplifiers which are routed to a PC containing multi-channel scaler hardware. The multi-channel scaler hardware allows data to be collected in a series of short, discrete scaler channels known as “time bins”. The count time for each time bin is selected as a function of the speed of the conveyor belt. The time bin length is frequently set up to be half the length of “dwell time,” which is the time the material aliquot to be surveyed spends within the detection field (Miller 2000).

The approach cited in the paragraph above ensures that activity present within the survey unit will be in full view of the detector for one complete time bin. Data collection is optimized by performing the measurement when the activity is concentrated (i.e., within an area of elevated activity) as well as when the activity is approximately homogenously-distributed within a given material aliquot.

D.6.4 Radiation Types

Conveyorized survey monitoring systems are generally best-suited for the detection of photon radiation. Specific systems may be tailored for the detection of beta emissions of moderate energy and even alpha radiation by employing gas flow proportional counter detector arrays.

D.6.5 Range

Photon radiation can typically be measured with a detectable energy range from 50 keV up to 2 MeV. Conveyorized survey monitoring systems equipped to measure alpha and beta emissions can typically measure from 100 keV up to 6 MeV.

D.6.6 Scale

Most conveyorized survey monitoring systems are designed for soils or laundry, both of which are compressible media. Applicable sample/material heights range from 2 cm to 12.5 cm (Life Safety 2005).

D.6.7 Ruggedness

Conveyorized survey monitoring systems have typical operating ranges from -20° C to 50° C. Conveyorized survey monitoring systems are often constructed from steel and with plastic scintillation detectors and windowless designs, which makes them generally resistant to damage from extraneous pieces of debris during scanning. Mechanical shock is not a typical concern for conveyorized survey monitoring systems because there is little need for moving these systems. For this reason conveyorized survey monitoring systems are seldom transported from one location to another.

D.7 In Situ Gamma Spectroscopy

In situ gamma spectroscopy (ISGS) systems combine the peak resolution capabilities of laboratory methods with instrumentation that is portable and rugged enough to be used in field conditions. These solid state systems can perform quantitative, multi-channel analysis of gamma-emitting isotopes in both solid and liquid media over areas as large as 100 m^2, enabling spectrographic analysis of M&E that assists the user in identifying constituent radionuclides and differentiating them from background radiation. ISGS system measurements can also provide thorough coverage within broad survey areas, minimizing the risk of failing to detect isolated areas of elevated radioactivity that could potentially be missed when collecting discrete samples.

D.7.1 Instruments

ISGS systems consist of a semiconductor detector, a cryostat, a multi-channel analyzer (MCA) electronics package that provides amplification and analysis of the energy pulse heights, and a computer system for data collection and analysis. Semiconductor detection systems typically employ a cryostat and a Dewar filled with liquid nitrogen (-196 °C). The cryostat transmits the cold temperature of the liquid nitrogen to the detector crystal, creating the extreme cold environment necessary for correct operation of the high-resolution semiconductor diode. ISGS systems may have electronic coolers as well.

ISGS systems use detectors referred to as N- and P-type detectors. N-type detectors contain small amounts of elements with five electrons in their outer electron shell (e.g., phosphorus, arsenic) within the germanium crystal (the inclusion of these elements within the germanium crystal is called “doping”). These result in free, unbonded electrons in the crystalline structure, providing a small negative current. P-type detectors utilize elements with less than four electrons in their outer electron shell (e.g., lithium, boron, gallium) are also used in doping to create electron holes, providing a small positive current. Use of these two varieties of doped germanium crystals provide different detection properties described below in Section D.7.5.

D.7.2 Temporal Issues

Setup for ISGS semiconductor systems may require one full day. The systems often require one hour to set up physically, six to eight hours for the semiconductor to reach the appropriate temperature operating range after the addition of liquid nitrogen, and quality control measurements may require another hour.[6] Count times using ISGS semiconductor systems tend to be longer than those associated with simpler detector systems for conducting static measurements, though this may be off-set by enlarging the field-of-view. A measurement time of several minutes is common, depending on the intensity of the targeted gamma energies and the presence of attenuating materials.

Count times can be shortened by reducing the distance between the area being surveyed and the detector to improve the gamma incidence efficiency or by using a larger detector. Each option will ultimately help the detection system see more gamma radiation in a shorter time. Yet either approach creates greater uncertainty associated with the source-to-detector geometry. A slight placement error (e.g., a 0.5 cm placement error) will result in significantly higher quantification error at a distance of one centimeter than at a distance of 10 centimeters. Additionally, this technique for decreasing count times promotes an effect called cascade summing, a phenomena affecting detection of gamma radiation from radionuclides that emit multiple gamma photons in a single decay event (e.g., ^{60}Co, which yields gamma particles of 1.17 and 1.33 MeV). If both incident gammas deposit their energy in a relatively short period of time (i.e., when compared to

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[6] It is important not to move the apparatus prematurely, as failure to allow the ISGS system to cool and equilibrate to its proper operating temperatures as may cause damage to the semiconductor detector.

the detector response time and/or the resolving time for the associated electronics), limitations of the detection system may prevent these individual photons from being distinguished (Knoll 1999).

D.7.3 Spatial Issues

ISGS semiconductor systems require calibration for their intended use. While ISGS semiconductor systems can be calibrated using traditional prepared radioactive sources, some ISGS systems have software that enables the user to calculate efficiencies by entering parameters such as elemental composition, density, stand-off distance, and physical dimensions. Supplied geometry templates assist in generating calibration curves that can be applied to multiple collected spectra. The high resolution of these systems coupled with advanced electronic controls for system parameters allows them to overcome issues related to source-to-detector geometry and produce quantitative concentrations of multiple radionuclides in a variety of media (e.g., soil, water, air filters). Because ISGS systems integrate all radioactivity within their field-of-view, lead shielding and collimation may be required to "focus" the field-of-view on a specified target for some applications.

D.7.4 Radiation Types

ISGS systems can accurately identify and quantify only photon-emitting radionuclides.

D.7.5 Range

ISGS systems can identify and quantify low-energy gamma emitters (50 keV with P-type detectors, 10 keV with N-type detectors) and high-energy gamma emitters (ISGS systems can be configured to detect gamma emissions upwards of 2.0 MeV). Specially-designed germanium detectors that exhibit very little deterioration in resolution as a function of count rate use N-type detectors or planar crystals with a very thin beryllium window for the measurement of photons in the energy range 5 to 80 keV.

D.7.6 Scale

These systems therefore offer functional quantitative abilities to analyze small objects (e.g., samples) for radionuclides. They can also effectively detect radioactivity over areas as large as 100 m^2 or more (Canberra 2005a). With the use of an appropriate dewar, the detector may be used in a vertical orientation to determine gamma isotope concentrations in the ground surface and shallow subsurface.

D.7.7 Ruggedness

ISGS semiconductor systems are fragile, because the extremely low temperatures utilized by the cryostat render portions of the system brittle and susceptible to damage if not handled with care. Some ISGS systems are constructed of more rugged materials and their durability is comparable to most hand-held instruments.

D.8 Hand-Held Radionuclide Identifiers

Hand-held radionuclide identifiers represent a relatively new addition to the radiation detection market, merging the portability of hand-held instruments with some of the analytical capabilities of ISGS systems. Hand-held radionuclide identifiers also feature data logging and storage capabilities (including user-definable radionuclide libraries) and the ability to transfer data to external devices. These devices are most commonly used for nuclear non-proliferation, where immediate isotope identification is more critical than low-activity detection sensitivity. Design parameters for hand-held radionuclide identifiers required by ANSI N42.34 (ANSI 2003) are user-friendly controls and intuitive menu structuring for routine modes of operation, enabling users without health physics backgrounds (e.g., emergency response personnel) to complete basic exposure rate or radionuclide identification surveys. These units also feature restricted “expert” survey modes of operation to collect activity concentration data for more advanced applications, including disposition surveys.

D.8.1 Instruments

Hand-held radionuclide identifiers consist of two general types: integrated systems and modular systems. The integrated systems have the detector and electronics contained in a single package; modular systems separate the detector from the electronics. These spectrometers employ small scintillators, typically NaI(Tl) or CsI(Tl), or room temperature solid semiconductors, such as cadmium zinc telluride (CZT), linked to multi-channel analyzers and internal radionuclide libraries to enable gamma-emitting radionuclide identification.

D.8.2 Temporal Issues

Hand-held radionuclide identifiers require minimal time to set up.[7] Depending upon the conditions in which data is being collected (i.e., climatic, environmental, the presence of sources of radiological interference), it may require seconds to several minutes for the unit to stabilize the input signals from the field of radiation and properly identify the radionuclides.

D.8.3 Spatial Issues

Detectors of hand-held radionuclide identifiers are typically small and portable. Spatial limitations are usually based on the physical size of the probe itself, and whether the probe is coupled internally within the casing or externally via an extension cord.

D.8.4 Radiation Types

Hand-held radionuclide identifiers are most commonly used for the detection of photon radiation, although many devices have capabilities for detecting neutron and beta emissions (the detection of neutron radiation requires a different probe from the photon radiation probe).

D.8.5 Range

Photon radiation can typically be measured within a detectable energy range of 10 to 30 keV up to 2.5 to 3 MeV. Neutron radiation can typically be measured within a detectable energy range of 0.02 eV up to 100 MeV.

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[7] The use of multi-point calibrations may add an estimated one to two hours to the time required for instrument set up.

D.8.6 Scale

There is no definitive limit to the size of an object to be surveyed using hand-held radionuclide identifiers. Hand-held radionuclide identifiers may generally be used to survey M&E of any size; constraints are only placed by the practical sizing of M&E related to the sensitive area of the probe.

D.8.7 Ruggedness

All varieties of hand-held radionuclide identifiers discussed here are typically calibrated for use in temperatures from -20 °C to 50 °C and feature seals or gaskets to prevent water ingress from rain, condensing moisture, or high humidity. Most hand-held radionuclide identifiers have a limited resistance to shock, though the durability of an instrument depends largely upon the detection media (e.g., NaI(Tl) crystals are fragile and vulnerable to mechanical and thermal shock).

D.9 Portal Monitors

Portal monitors screen access points to controlled areas, and are designed for detecting radioactivity above background. These systems are used for interdiction-type surveys, and generally do not provide radionuclide identification. Portal monitors are primarily designed to monitor activity on vehicles.

Historically, portal monitors have been used to detect radioactive materials at entrance points to scrap metal facilities and solid waste landfills, and radiological control area exit points within nuclear facilities to screen for the inadvertent disposal of radionuclides. The proximity of other items to be surveyed containing high concentrations of activity may influence the variability of the instrument background, because portal monitors survey activity by detecting small variations in ambient radiation (NRC 2002).

D.9.1 Instruments

Portal monitors can easily be arranged in various geometries that maximize their efficiencies. Most national and international standards, for example ANSI 42.35 (ANSI 2004) require both gamma- and neutron-detecting capabilities, but gamma-only versions are available. Portal monitors typically use large-area polyvinyl toluene scintillators (a form of plastic scintillators) to detect photon radiation and ^{3}He proportional tubes to detect neutrons.[8] Individual detectors may be cylindrical or flat. The detectors are usually arranged to form a detection field between two detectors, and items to be surveyed pass through the detection field (i.e., between the detectors) as shown in Figure D.3.

The system usually consists of one or more detector array(s), an occupancy sensor, a control box, and a monitoring PC. The control box and monitoring PC store and analyze alarm and occupancy data, store and analyze all gamma and neutron survey data, and may even send data through an integrated internet connection. The monitoring PC also manages software that operates multiple arrangements of detector arrays as well as third party instruments. For example, security cameras can take high-resolution images of objects that exceed a radiation screening level (Novelec 2001b).

D.9.2 Temporal Issues

Count or integration times are very short, typically just a few seconds (NRC 2002). Set-up time in the field is variable, since temporary systems may require two hours to one half-day to set up, while permanent systems may require one week to install. For vehicular portal monitor systems, objects may typically pass through the field of detection at speeds of 8 to 9.5 kilometers per hour (Canberra 2005b). Most systems use speed correction algorithms to minimize the effects of variations in dwell time (i.e., the time a given area to be surveyed spends within the detection field).

D.9.3 Spatial Issues

There are a large number of factors that affect portal monitor performance. The isotopic content of a radioactive material can determine the ease of detection. For example, high-enriched uranium (HEU) is easier to detect in a gamma portal than low-enriched uranium (LEU) or natural uranium because of the larger gamma emission rate from ^{235}U.

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[8] Neutron detectors use materials that detect thermal neutrons, which may be fast neutrons that are thermalized for detection through the use of moderators.

Figure D.3 Example Portal Monitor (Canberra 2005b)

The chemical composition of a material is also important; background levels of radioactivity must also be considered. Neutron portals are an effective method for detecting plutonium in areas with large gamma backgrounds. The surface area and size of the detectors and distance between the detectors all affect the geometry and response of the system. In a large area system set-up, the closer together the detector arrays are, the better the geometric efficiencies are going to be. Finally, for each system there is a maximum passage speed through the portal that gives a counting time necessary to meet the required detection sensitivity.

D.9.4 Radiation Types

Portal monitors typically detect gamma radiation and can also be equipped to detect neutron radiation. Gamma portals often use integrated metal detectors to provide an indication of suspicious metal containers that could be used to shield radioactive materials. If the gamma radiation is not shielded adequately, the detector's alarm will sound. Portal monitors can detect radioactive material even if it is shielded with a material with a high atomic number, like lead.

D.9.5 Range

Photon radiation can typically be measured within a detectable energy range of 60 keV up to 2.6 MeV. Neutron radiation can typically be measured within a detectable energy range of 0.025 eV up to 100 MeV. Required detection sensitivities for gamma and neutron sources are described in ANSI 42.35, Table 3 (ANSI 2004). Portal monitors provide gross counts and cannot compute quantitative measurements (e.g., activity per unit mass).

D.9.6 Scale

Most systems are designed to monitor items ranging in size from bicycles and other small vehicles to tractor trailers, railroad cars, and even passenger airplanes (Canberra 2005b). The width of the detection field (i.e., space between the detector arrays) can usually be modified.

D.9.7 Ruggedness

Portal monitors have typical operating ranges from -20 ° to 55 °C, and some systems may be functional in temperatures as low as -40 °C according to ANSI 42.35 (ANSI 2004). Portal monitors are usually designed with weatherproofing to withstand prolonged outdoor use and exposure to the elements.

D.10 Sample with Laboratory Analysis

Laboratory analysis allows for more controlled conditions and more complex, less rugged instruments to provide lower detections limits and greater delineation between radionuclides than any measurement method that may be utilized in a field setting. For this reason, laboratory analyses are often applied as quality assurance measures to validate sample data collected using field equipment.

D.10.1 Instruments

This section provides a brief overview of instruments used for radiological analyses in a laboratory setting. For additional detail on these instruments, please refer to the accompanying section references in MARLAP.

D.10.1.1 Instruments for the Detection of Alpha Radiation

- Alpha Spectroscopy with Multi-Channel Analyzer

This system consists of an alpha detector housed in an evacuated counting chamber, a bias supply, amplifier, analog-to-digital converter, multi-channel analyzer, and computer. Samples are placed at a fixed distance from the solid state partially-implanted silica for analysis, and the multi-channel analyzer yields an energy spectrum that can be used to both identify and quantify the radionuclides. The overall properties of the instrumentation allow for excellent peak resolution, although this technique often requires a complex chemical separation to obtain the best results.

- Gas-Flow Proportional Counter

The system consists of a gas-flow detector, supporting electronics, and an optional guard detector for reducing the background count rate. A thin window can be placed between the gas-flow detector and sample to protect the detector from contamination, or the sample can be placed directly into the detector. This system does not typically provide data useful for identifying radionuclides unless it is preceded by nuclide-specific chemical separations.

- Liquid Scintillation Spectrometry

Typically, samples will be subjected to chemical separations and the resulting materials placed in a vial with a scintillation cocktail. When the alpha particle energy is absorbed by the cocktail, light pulses are emitted, which are detected by photomultiplier tubes. One pulse of light is emitted for each alpha particle absorbed. The intensity of light emitted is related to the energy of the alpha. This system can provide data useful for identifying radionuclides if the system is coupled to a multi-channel analyzer.

- Low-Resolution Alpha Spectrometry

The system consists of a small sample chamber, mechanical pump, two-inch diameter silicon detector, multi-channel analyzer, readout module, and a computer. Unlike alpha spectroscopy with multi-channel analyzer, this method allows the technician to load samples for analysis without drying since the presence of moisture generally has negligible effects on the results. This method is therefore estimated to substantially reduce the time for analysis. However, the low resolution may limit the ability to identify individual radionuclides in a sample containing multiple radionuclides and thus may limit the applicability of this method (Meyer 1995).

- Alpha Scintillation Detector

This system is used primarily for the quantification of ^{226}Ra by the emanation and detection of ^{222}Rn gas. The system consists of a bubbler system with gas transfer apparatus, a vacuum flask lined with scintillating material called a Lucas Cell,[9] a photomultiplier tube, bias supply, and a scaler to record the count data.

D.10.1.2 Instruments for the Detection of Beta Radiation

- Gas-Flow Proportional Counter

The system consists of a gas-flow detector, supporting electronics, and an optional guard detector for reducing the background count rate. A thin window can be placed between the gas-flow detector and sample to protect the detector from non-fixed activity, or the sample can be placed directly into the detector. This technique does not provide data useful for identifying individual radionuclides unless it is preceded by nuclide-specific chemical separations.

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[9] One end of a Lucas cell is covered with a transparent window for coupling to a photomultiplier tube and the remaining inside walls are coated with zinc sulfide.

- Liquid Scintillation Spectrometry

Typically, samples will be subjected to chemical separations and the resulting materials placed in a vial with a scintillation cocktail. When the beta particle energy is absorbed by the cocktail, light pulses are emitted, which are detected by photomultiplier tubes. One pulse of light is emitted for each beta particle absorbed. The intensity of light emitted is related to the energy of the beta. This system can provide data useful for identifying radionuclides if the system is coupled to a multi-channel analyzer. This system must be allowed to darken (i.e., equilibrate to a dark environment) prior to measurement.

D.10.1.3 Instruments for the Detection of Gamma or X-Radiation

- High-Purity Germanium Detector with Multi-Channel Analyzer

This system consists of a germanium detector connected to a cryostat (either mechanical or a dewar of liquid nitrogen), high voltage power supply, spectroscopy grade amplifier, analog to digital converter, and a multi-channel analyzer. This system has high resolution for peak energies and is capable of identifying and quantifying individual gamma peaks in complex spectra. It is particularly useful when a sample may contain multiple gamma-emitting radionuclides and it is necessary to both identify and quantify all nuclides present.

- Sodium Iodide Detector with Multi-Channel Analyzer

This system consists of a sodium iodide detector, a high voltage power supply, an amplifier, an analog to digital converter, and a multi-channel analyzer. This system has relatively poor energy resolution and is not effective for identifying and quantifying individual gamma peaks in complex spectra. It is most useful when only a small number of gamma-emitting nuclides are present or when a gross-gamma measurement is adequate.

D.10.2 Temporal Issues

Laboratory analysis is usually controlled by the turnaround time involved in preparing and accurately measuring the collected samples. The sample matrix impacts the preparation time, since soils and bulk chemicals typically require more extensive preparation than liquids or smears. Table D.2 describes the typical preparation and counting times associated with the various analytical instruments and methods described in Section D.10.1. Additional issues that may result in extended time for sample preparation and analysis are described in MARLAP.

Table D.2 Typical Preparation and Counting Times

	Typical Preparation Time	Typical Counting Time
Alpha Spectroscopy with Multi-Channel Analyzer	1 to 7 days	100 to 1,000 minutes
Gas-Flow Proportional Counter	hours to days	10 to 1,000 minutes
Liquid Scintillation Spectrometer	Minutes,[10] hours to 2 days[11]	>60 to 300 minutes
Low-Resolution Alpha Spectroscopy	minutes (DOE, 1995)	10 to 1,000 minutes
High-Purity Germanium (HPGe) Detector with Multi-Channel Analyzer	minutes to 1 day	10 to 1,000 minutes
Sodium Iodide (NaI) Detector with Multi-Channel Analyzer	minutes to 1 day	1 to 1,000 minutes
Alpha Scintillation Detector	1 to 4 days; 4 to 28 days[12]	10 to 200 minutes

D.10.3 Spatial Issues

This section addresses issues related to detector-M&E geometry and provides information on the range of impacts resulting from dissenting geometries between the calibration source and the measured sample. Other topics may include detector dimensions and problems positioning instruments.

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[10] Minimal preparation times are possible if the sample does not require concentration prior to being added to the liquid scintillation cocktail vial.

[11] Longer preparation times are necessary for speciation of low-energy beta emitters.

[12] Longer count times represent the necessary time for in-growth of ^{222}Rn for ^{226}Ra analyses.

D.10.3.1 Alpha Spectroscopy with Multi-Channel Analyzer

Sample geometry (lateral positioning on a detector shelf) in some detectors may be a small source of additional uncertainty. Uncertainty in the preparation of the actual calibration standards as well as the applicability of the calibration standards to the sample analysis should also be considered.

D.10.3.2 Gas-Flow Proportional Counter

Even deposition of sample material on the planchette is critical to the analytical process. In some analyses, ringed planchettes may aid in the even deposition of sample material. An uneven deposition may result in an incorrect mass-attenuation correction as well as introducing a position-dependent bias to the analysis. The latter situation arises from the fact that gas-flow proportional counters are not radially-symmetric, so rotation of an unevenly deposited sample by 45 degrees may drastically change the instrument response.

D.10.3.3 Liquid Scintillation Spectrometer

For gross counting, samples (e.g., smears and filters) can be placed directly into a liquid scintillation counter (LSC) vial with liquid scintillation cocktail, and counted with no preparation. There are samples with more complicated matrices that require chemical separation prior to being placed and counted in LSC vials. Calibration sources are also kept and counted in these vials, so the geometry of the source and the sample compared to the detector are generally similar.

D.10.3.4 Low-Resolution Alpha Spectroscopy

Sample geometry (lateral positioning on a detector shelf) in some detectors may be a small source of additional uncertainty. Uncertainty in the preparation of the actual calibration standards as well as the applicability of the calibration standards to the sample analysis should be considered.

D.10.3.5 High-Purity Germanium Detector with Multi-Channel Analyzer

Geometry considerations are most important for spectroscopic gamma analyses. Sample positioning on the detector may significantly affect the analytical results, depending on the size and shape of the germanium crystal. Moreover, the instrument is calibrated with a source that should be the same physical size, shape, and weight as the samples to be analyzed.[13] Discrepancies between the volume or density of the sample and the source introduce additional uncertainty to the analytical results.

Sample homogeneity is a critical factor in gamma spectroscopy analyses, particularly with relatively large samples. For example, sediment settling during the course of analysis of a turbid aqueous sample will result in a high bias from any activity contained in the solid fraction. Likewise, the positioning of areas containing elevated activity in a solid sample will create a bias in the overall sample activity (the activity will be disproportionately high if the particle is located at the bottom of the sample, and the activity will be disproportionately low if it is located at the top of the sample).

D.10.3.6 Sodium Iodide Detector with Multi-Channel Analyzer

The spatial considerations for NaI detectors are the same as those listed above for high-purity germanium detectors.

D.10.3.7 Alpha Scintillation Detectors

Accurate sample analysis depends heavily on the complete dissolution of the ^{226}Ra or other radionuclides of interest in the bubbler solution. Adequate sample preparation will help ensure that spatial issues do not influence results, as the apparatus itself minimizes any other potential geometry-related sources of error or uncertainty.

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[13] Some software packages allow a single calibration geometry to be modeled to assimilate the properties of other geometries.

D.10.4 Radiation Types

Table D.3 describes the types of radiation that each laboratory instrument and method can measure.

Table D.3 Radiation Applications for Laboratory Instruments and Methods

	Alpha	Beta	Photon	Neutron	Differentiate Radiation Types	Identify Specific Radionuclides
Alpha Spectrometry with a Multi-Channel Analyzer	GOOD	NA	NA	NA	NA	GOOD
Gas-Flow Proportional Counter	GOOD	GOOD	POOR	NA	FAIR	POOR
Liquid Scintillation Spectrometer	POOR	GOOD[14]	POOR	NA	FAIR	FAIR
Low-Resolution Alpha Spectroscopy	GOOD	NA	NA	NA	NA	FAIR[15]
High-Purity Germanium Detector with Multi-Channel Analyzer	NA	NA	GOOD	NA	NA	GOOD
Sodium Iodide Detector with Multi-Channel Analyzer	NA	NA	GOOD	NA	NA	FAIR
Alpha Scintillation Detector	GOOD	NA	NA	NA	NA	FAIR

Notes:

GOOD	The instrumentation and measurement technique is well-suited for this application
FAIR	The instrumentation and measurement technique can adequately perform this application
POOR	The instrumentation and measurement technique may be poorly-suited for this application
NA	The instrumentation and measurement technique cannot perform this application

D.10.5 Range

All of the instrumentation discussed here has physical limitations as to the amount of activity that can be analyzed. This limitation arises primarily from the ability of the detector to recover after an ionizing event, and the speed with which the component

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[14] This system is designed for the detection of low-energy beta particles.

[15] The low resolution may limit the ability to identify individual radionuclides in a sample containing multiple radionuclides.

electronics can process the data. Typically, a count rate on the order of 10^6 counts per second taxes the physical limitations of most detectors. Other practical considerations, (such as the potential to impact the detector with non-fixed activity) often override the physical limitations of the counting system.

There are energy range limitations as well. For example: window proportional counters are poor choices for very low energy beta emitters; some gamma spectrometers have poor efficiencies at low energies; and some systems are not calibrated for high-energy gammas. Table D.4 describes the energy range that each instrument and method can be used to determine, and the maximum activity per sample that the method can be used to count.[16]

Table D.4 Typical Energy Ranges and Maximum Activities

	Energy Range	Maximum Activity
Alpha Spectrometry with Multi-Channel Analyzer	3 to 8 MeV	<10 Bq (<270 pCi)
Gas-Flow Proportional Counter	3 to 8 MeV (α) 100 to 2,000 keV (β)	35 Bq (946 pCi)
Liquid Scintillation Spectrometer	>3 Mev 15 to 2,500 keV (β); >1.5 MeV (β)[17]	100,000 Bq (2.7 μCi)
Low-Resolution Alpha Spectrometry	3 to 8 MeV (α)	<10 Bq (<270 pCi)
High-Purity Germanium (HPGe) Detector with Multi-Channel Analyzer	50 to >2,000 keV (P-type detector); 5 to 80 keV (N-type detector)	370 Bq (10,000 pCi)
Sodium Iodide (NaI) Detector with Multi-Channel Analyzer	>80 to 2,000 keV	370 Bq (10,000 pCi)
Alpha Scintillation Detector	All α emission energies	<10 Bq (<270 pCi)

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[16] David Burns, Paragon Analytics, Inc., private communication with Nick Berliner, Cabrera Services, Inc., March 2005.

[17] Very high-energy beta emitters may be counted using liquid scintillation equipment without liquid scintillation cocktails by the use of the Cerenkov light pulse emitted as high energy charged particles move through water or similar substances.

D.10.6 Scale

There is no minimum sample size required for a given analysis. Smaller sample sizes will necessarily result in elevated detection limits. Minimum sample sizes (e.g., 0.1 gram) may be specified in order to ensure that the sample is reasonably representative given the degree of homogenization achieved in the laboratory. Typical liquid and solid sample sizes are noted in Table D.5.

Table D.5 Typical Liquid and Solid Sample Sizes

	Typical Liquid Sample Size	Typical Solid Sample Size
Alpha Spectrometry with Multi-Channel Analyzer	1 liter	2 grams; 50 grams[18]
Gas-Flow Proportional Counter	1 liter	2 grams
Liquid Scintillation Spectrometer	<10 milliliters; 1 liter[19]	<0.5 grams; 500 grams
Low-Resolution Alpha Spectrometry	1 liter	2 grams; 50 grams[17]
High-Purity Germanium (HPGe) Detector with Multi-Channel Analyzer	4 liters	1 kilogram
Sodium Iodide (NaI) Detector with Multi-Channel Analyzer	4 liters	1 kilogram
Alpha Scintillation Detector	1 liter	2 grams

D.10.7 Ruggedness

Ruggedness does not hold relevance to laboratory analyses, because they are performed in a controlled environment that precludes the instrumentation from being exposed to conditions requiring durability.

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[18] The use of sample digestion processes allows the processing of larger sample masses.

[19] Direct depositing of sample material into the scintillation cocktail limits the sample size to the smaller samples sizes noted; prepared analyses may use substantially larger sample quantities as noted (this applies to both liquid and solid sample matrices).

E. DISPOSITION CRITERIA

E.1 Department of Energy

Disposition criteria specified by DOE regulations and orders are found in the Code of Federal Regulations, Title 10 (especially 10 CFR 835, Occupational Radiation Protection) and in applicable DOE Orders (especially DOE Order 5400.5, Radiation Protection of the Public and the Environment). The DOE regulations and orders govern the conduct of DOE employees and contractors in the operation of DOE facilities and in the disposition of real property (e.g., buildings and land) and non-real property ("personal property" such as materials, equipment, materials in containers, clothing, etc.). The DOE Order requirements are applicable to DOE activities only and are enforceable as contractual provisions in most DOE contracts and DOE rules are enforceable under 10 CFR Part 820. The following list of DOE requirements is not exhaustive. In addition, a listing of some non-mandatory guidance documents is also provided.

E.1.1 10 CFR 835 (non-exhaustive excerpts)

E.1.1.1 § 835.405 Receipt of Packages Containing Radioactive Material

(a) If packages containing quantities of radioactive material in excess of a Type A quantity (as defined at 10 CFR 71.4) are expected to be received from radioactive material transportation, arrangements shall be made to either:

(1) Take possession of the package when the carrier offers it for delivery; or

(2) Receive notification as soon as practicable after arrival of the package at the carrier's terminal and to take possession of the package expeditiously after receiving such notification.

(b) Upon receipt from radioactive material transportation, external surfaces of packages known to contain radioactive material shall be monitored if the package:

(1) Is labeled with a Radioactive White I, Yellow II, or Yellow III label (as specified at 49 CFR 172.403 and 172.436–440); or

(2) Has been transported as low specific activity material (as defined at 10 CFR 71.4) on an exclusive use vehicle (as defined at 10 CFR 71.4); or

(3) Has evidence of degradation, such as packages that are crushed, wet, or damaged.

(c) The monitoring required by paragraph (b) of this section shall include:

(1) Measurements of removable contamination levels, unless the package contains only special form (as defined at 10 CFR 71.4) or gaseous radioactive material; and

(2) Measurements of the radiation levels, unless the package contains less than a Type A quantity (as defined at 10 CFR 71.4) of radioactive material.

(d) The monitoring required by paragraph (b) of this section shall be completed as soon as practicable following receipt of the package, but not later than 8 hours after the beginning of the working day following receipt of the package.

E.1.1.2 § 835.605 Labeling items and containers

Except as provided at § 835.606, each item or container of radioactive material shall bear a durable, clearly visible label bearing the standard radiation warning trefoil and the words “Caution, Radioactive Material” or “Danger, Radioactive Material.” The label shall also provide sufficient information to permit individuals handling, using, or working in the vicinity of the items or containers to take precautions to avoid or control exposures.

E.1.1.3 § 835.606 Exceptions to labeling requirements

(a) Items and containers may be excepted from the radioactive material labeling requirements of § 835.605 when:

(1) Used, handled, or stored in areas posted and controlled in accordance with this subpart and sufficient information is provided to permit individuals to take precautions to avoid or control exposures; or

(2) The quantity of radioactive material is less than one tenth of the values specified in appendix E of this part; or

(3) Packaged, labeled, and marked in accordance with the regulations of the Department of Transportation or DOE Orders governing radioactive material transportation; or

(4) Inaccessible, or accessible only to individuals authorized to handle or use them, or to work in the vicinity; or

(5) Installed in manufacturing, process, or other equipment, such as reactor components, piping, and tanks; or

(6) The radioactive material consists solely of nuclear weapons or their components.

(b) Radioactive material labels applied to sealed radioactive sources may be excepted from the color specifications of § 835.601(a).

E.1.1.4 § 835.1101 Control of material and equipment

(a) Except as provided in paragraphs (b) and (c) of this section, material and equipment in contamination areas, high contamination areas, and airborne radioactivity areas shall not be released to a controlled area if:

(1) Removable surface contamination levels on accessible surfaces exceed the removable surface contamination values specified in appendix D of this part; or

(2) Prior use suggests that the removable surface contamination levels on inaccessible surfaces are likely to exceed the removable surface contamination values specified in appendix D of this part.

(b) Material and equipment exceeding the removable surface contamination values specified in appendix D of this part may be conditionally released for movement on-site from one radiological area for immediate placement in another radiological area only if appropriate monitoring is performed and appropriate controls for the movement are established and exercised.

(c) Material and equipment with fixed contamination levels that exceed the total contamination values specified in appendix D of this part may be released for use in controlled areas outside of radiological areas only under the following conditions:

(1) Removable surface contamination levels are below the removable surface contamination values specified in appendix D of this part; and (2) The material or equipment is routinely monitored and clearly marked or labeled to alert personnel of the contaminated status.

E.1.1.5 § 835.1102 Control of areas

(a) Appropriate controls shall be maintained and verified which prevent the inadvertent transfer of removable contamination to locations outside of radiological areas under normal operating conditions.

(b) Any area in which contamination levels exceed the values specified in appendix D of this part shall be controlled in a manner commensurate with the physical and chemical characteristics of the contaminant, the radionuclides present, and the fixed and removable surface contamination levels.

(c) Areas accessible to individuals where the measured total surface contamination levels exceed, but the removable surface contamination levels are less than, corresponding surface contamination values specified in appendix D of this part, shall be controlled as follows when located outside of radiological areas:

(1) The area shall be routinely monitored to ensure the removable surface contamination level remains below the removable surface contamination values specified in appendix D of this part; and

(2) The area shall be conspicuously marked to warn individuals of the contaminated status.

(d) Individuals exiting contamination, high contamination, or airborne radioactivity areas shall be monitored, as appropriate, for the presence of surface contamination.

(e) Protective clothing shall be required for entry to areas in which removable contamination exists at levels exceeding the removable surface contamination values specified in appendix D of this part.

E.1.2 Appendix D to Part 835 – Surface Contamination Values

The data presented in appendix D are to be used in identifying the need for posting of contamination and high contamination areas in accordance with § 835.603(e) and (f) and identifying the need for surface contamination monitoring and control in accordance with §§ 835.1101 and 835.1102.

Table E.1 Surface Contamination Values[1] in dpm/100 cm^2 as Reported in Appendix D to Part 835

Radionuclide	Removable[2,4]	Total (Fixed+ Removable)[2,3]
U-nat, U-235, U-238, and associated decay products	1,000[7]	5,000[7]
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	20	500
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above[5]	1,000	5,000
Tritium and tritiated compounds[6]	10,000	N/A

[1] The values in this appendix, with the exception noted in footnote 5, apply to radioactive contamination deposited on, but not incorporated into the interior or matrix of, the contaminated item. Where surface contamination by both alpha-and beta-gamma-emitting nuclides exists, the limits established for alpha-and beta-gamma-emitting nuclides apply independently.

[2] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[3] The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the value specified. For purposes of averaging, any square meter of surface shall be considered to be above the surface contamination value if: (1) From measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds three times the applicable value.

[4] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by swiping the area with dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note - The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area shall be based on the actual area and the entire surface shall be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

[5] This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

[6] Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply.

[7] (alpha)

E.1.3 DOE Order 5400.5 (non-exhaustive excerpts) from Chapter II

5. Release of Property Having Residual Radioactive Material

(a) Release of Real Property. Release of real property (land and structures) shall be in accordance with the guidelines and requirements for residual radioactive material presented in Chapter IV. These guidelines and requirements apply to both DOE-owned facilities and to private properties that are being prepared by DOE for release. Real properties owned by DOE that are being sold to the public are subject to the requirements of Section 120(h) of the Comprehensive Environmental Response Compensation and Liability Act (CERCLA), as amended, concerning hazardous substances, and to any other applicable Federal, State, and local requirements. The requirements of 40 CFR Part 192 are applicable to properties remediated by DOE under Title I of the Uranium Mill Tailings Radiation Control Act (UMTRA).

(b) Release of Personal Property. Personal property, which potentially could be contaminated, may be released for unrestricted use if the results of a survey with appropriate instruments indicate that the property is less than the contamination limits presented in Figure IV-1.

(c) Release of Materials and Equipment.

(1) Surface Contamination Levels. Prior to being released, property shall be surveyed to determine whether both removable and total surface contamination (Including contamination present on and under any coating) are in compliance with the levels given in Figure IV-1 and that the contamination has been subjected to the ALARA process.

(2) Potential for Contamination. Property shall be considered to be potentially contaminated if it has been used or stored in radiation areas that could contain unconfined radioactive material or that are exposed to beams of particles capable of causing activation (neutrons, protons, etc.).

(3) Surveys. Surfaces of potentially contaminated property shall be surveyed using instruments and techniques appropriate for detecting the limits stated in Figure IV-1.

(4) Inaccessible Areas. Where potentially contaminated surfaces are not accessible for measurement (as in some pipes, drains, and ductwork), such property may be released after case-by-case evaluation and documentation based on both the history of its use and available

measurements demonstrate that the unsurveyable surfaces are likely to be within the limits given in Figure IV-1.

(5) Records. The records of released property shall include:

(a) A description or identification of the property;

(b) The date of the last radiation survey;

(c) The identity of the organization and the individual who performed the monitoring operation;

(d) The type and identification number of monitoring instruments;

(e) The results of the monitoring operation; and

(f) The identity of the recipient of the released material.

(6) Volume Contamination. No guidance is currently available for release of material that has been contaminated in depth, such as activated material or smelted contaminated metals (e.g., radioactivity per unit volume or per unit mass). Such materials may be released if criteria and survey techniques are approved by EH-1.

E.1.4 DOE Guidance and Similar Documents

The following discussion summarizes DOE policy, practice, and guidance for the disposition of personal property, including materials and equipment from several DOE guidance documents.

“Application of DOE 5400.5 requirements for release and control of property containing residual radioactive material,” a guidance memorandum dated November 17, 1995. This guidance memorandum explains the procedures through which authorized limits can be approved for the disposition of waste materials to sanitary waste landfills. It also discusses the disposition criteria for certain radionuclides. Finally, it delegates some responsibilities for the approval of release of volumetrically contaminated materials to DOE field office managers when specified conditions are met.

Table E.2 Figure IV-1, from DOE Order 5400.5, as Supplemented in November, 1995 Memorandum: Surface Activity Guidelines – Allowable Total Residual Surface Activity ($dpm/100cm^2$)[1]

Radionuclides[2]	Average[3,4]	Maximum[4,5]	Removable[4,6]
Group 1 - Transuranics, I-125, I-129, Ac-227, Ra -226, Ra-228, Th-228, Th-230, Pa-231	100	300	20
Group 2 - Th-natural, Sr-90, I-126, I-131, I-133, Ra-223, Ra-224, U-232, Th-232	1,000	3,000	200
Group 3 - U-natural, U-235, U-238, and associated decay products, alpha emitters	5,000	15,000	1,000
Group 4 - Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above[7]	5,000	15,000	1,000
Tritium (applicable to surface and subsurface)[8]	N/A	N/A	10,000

[1] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[2] Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

[3] Measurements of average contamination should not be averaged over an area of more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

[4] The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

[5] The maximum contamination level applies to an area of not more than 100 cm^2.

[6] The amount of removable material per 100 cm^2 of surface area should be determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wiping with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

[7] This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

[8] Property recently exposed or decontaminated, [sic] should have measurements (smears) at regular time intervals to ensure that there is not a build-up of contamination over time. Because tritium typically penetrates material it contacts, the surface guidelines in group 4 are not applicable to tritium. The Department has reviewed the analysis conducted by the DOE Tritium Surface Contamination Limits Committee ("Recommended Tritium Surface Contamination Release Guides," February 1991), and has assessed potential doses associated with the release of property containing residual tritium. The Department recommends the use of the stated guideline as an interim value for removable tritium. Measurements demonstrating compliance of the removable fraction of tritium on surfaces with this guideline are acceptable to ensure that non-removable fractions and residual tritium in mass will not cause exposures that exceed DOE dose limits and constraints.

"Control and Release of Property with Residual Radioactive Material for use with DOE Order 5400.5, Radiation Protection of the Public and the Environment," DOE G 441.1-XX, a draft guidance document approved for interim use and issued on May 1, 2002. This guidance document contains detailed discussions of the disposition approaches for real and personal property, as well as summaries of DOE's policies regarding the disposition or release of property.

"Cross-Cut Guidance on Environmental Requirements for DOE Real Property Transfers (Update)," DOE/EH-413/97-12, originally issued October, 1997, revised March, 2005. This guidance document contains a summary of various environmental requirements for the release or transfer of real property.

"Managing the Release of Surplus and Scrap Materials," January 19, 2001, from DOE Secretary Richardson to all DOE elements. This memorandum provides direction as well as guidance regarding the release of property from DOE radiological control. It also restricts the release of metal from radiological areas for recycle until certain steps are taken by DOE.

E.2 International Organizations

In general, each country establishes its own disposition criteria for materials and equipment. These national disposition criteria may be consistent with guidance promulgated by multi-national organizations, such as the International Atomic Energy Agency (IAEA) or the European Commission (EC). One example of widely-accepted regulations is the "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material SAFETY GUIDE No. TS-G-1.1 (ST-2)." The references listed below provide the detailed information on guidance from the IAEA and the EC. URLs are provided for internet access of this information. Disposition criteria from specific nations should be obtained from those nations.

E.2.1 International Atomic Energy Agency (IAEA)

Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material SAFETY GUIDE No. TS-G-1.1 (ST-2): http://www-pub.iaea.org/MTCD/publications/PDF/Pub1109_scr.pdf.

Planning and Preparing for Emergency Response to Transport Accidents Involving Radioactive Material, SAFETY GUIDE No. TS-G-1.2 (ST-3) http://www-pub.iaea.org/MTCD/publications/PDF/Pub1119_scr.pdf.

Application of the Concepts of Exclusion, Exemption and Clearance SAFETY GUIDE No. RS-G-1.7: http://www-pub.iaea.org/MTCD/publications/PDF/Pub1202_web.pdf.

E.2.2 European Commission

The publication list for radiation protection may be found on the EC website at: http://europa.eu.int/comm/energy/nuclear/radioprotection/publication_en.htm. Contact information for most of the authorities in the European Union may be found in Annex 3, in the last pages of publication 139, "A review of consumer products containing radioactive substances in the European Union," which can be found at: http://europa.eu.int/comm/energy/nuclear/radioprotection/publication/doc/139_en.pdf.

Radiation protection publications pertaining to disposition criteria for materials and equipment include:

134: Evaluation of the application of the concepts of exemption and clearance for practices according to title III of Council Directive 96/29/Euratom of 13 May 1996 in EU Member States, Volume 1, Main Report: http://europa.eu.int/comm/energy/nuclear/radioprotection/publication/doc/134_en.pdf.

122: Practical Use of the Concepts of Clearance and Exemption Part I: Guidance on General Clearance Levels for Practices: http://europa.eu.int/comm/energy/nuclear/radioprotection/publication/doc/122_part1_en.pdf.

122: Practical Use of the Concepts of Clearance and Exemption Part II: Application of the Concepts of Exemption and Clearance to Natural Radiation Sources: http://europa.eu.int/comm/energy/nuclear/radioprotection/publication/doc/122_part2_en.pdf.

114: Definition of Clearance Levels for the Release of Radioactively Contaminated Buildings and Building Rubble: http://europa.eu.int/comm/energy/nuclear/radioprotection/publication/doc/114_en.pdf.

European legislation related to the transport of radioactive materials (database): http://europa.eu.int/comm/energy/nuclear/transport/legislation_en.htm.

E.3 Nuclear Regulatory Commission

Disposition criteria specified by NRC regulations are found in the Code of Federal Regulations, Title 10 (10 CFR). NRC regulations in 10 CFR are structured in Parts, which apply to respective areas of applicability. For example, 10 CFR Part 20 addresses "Standards for Protection against Radiation." The regulatory citations below indicate the specific Part by the number to the left of the decimal point, for example, §20.2003 is in 10 CFR Part 20, and 2003 indicates a specific portion. In this appendix only the radiological component of those criteria pertaining to quantitative measurement attributes are listed; there are almost always additional regulatory requirements. "Disposition criteria" refers to the quantitative radiological portion of the complete criteria. In some circumstances, disposition criteria are not addressed in the regulations, and these cases are handled by existing policy and practices. A list of NRC disposition criteria, which is not necessarily exhaustive, follows.

E.3.1 § 20.2003 Disposal by release into sanitary sewerage.

(2) The quantity of licensed or other radioactive material that the licensee releases into the sewer in 1 month divided by the average monthly volume of water released into the sewer by the licensee does not exceed the concentration listed in table 3 of appendix B to part 20; and

(4) The total quantity of licensed and other radioactive material that the licensee releases into the sanitary sewerage system in a year does not exceed 5 curies (185 GBq) of hydrogen-3, 1 curie (37 GBq) of carbon-14, and 1 curie (37 GBq) of all other radioactive materials combined.

E.3.2 § 20.2005 Disposal of specific wastes.

(a) A licensee may dispose of the following licensed material as if it were not radioactive

(1) 0.05 microcurie (1.85 kBq), or less, of hydrogen-3 or carbon-14 per gram of medium used for liquid scintillation counting; and

(2) 0.05 microcurie (1.85 kBq), or less, of hydrogen-3 or carbon-14 per gram of animal tissue, averaged over the weight of the entire animal.

E.3.3 § 35.92 Decay-in-storage.

(a) A licensee may hold byproduct material with a physical half-life of less than 120 days for decay-in-storage before disposal without regard to its radioactivity if it--

(1) Monitors byproduct material at the surface before disposal and determines that its radioactivity cannot be distinguished from the background radiation level with an appropriate radiation detection survey meter set on its most sensitive scale and with no interposed shielding

E.3.4 § 35.315 Safety precautions.

(4) Either monitor material and items removed from the patient's or the human research subject's room to determine that their radioactivity cannot be distinguished from the natural background radiation level with a radiation detection survey instrument set on its most sensitive scale and with no interposed shielding, or handle the material and items as radioactive waste.

E.3.5 § 36.57 Radiation surveys.

(e) Before releasing resins for unrestricted use, they must be monitored before release in an area with a background level less than 0.5 microsievert (0.05 millirem) per hour. The resins may be released only if the survey does not detect radiation levels above background radiation levels. The survey meter used must be capable of detecting radiation levels of 0.5 microsievert (0.05 millirem) per hour.

E.3.6 Appendix A to Part 40--Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content

(6) The design requirements in this criterion for longevity and control of radon releases apply to any portion of a licensed and/or disposal site unless such portion contains a concentration of radium in land, averaged over areas of 100 square meters, which, as a result of byproduct material, does not exceed the background level by more than: (i) 5 picocuries per gram (pCi/g) of radium-226, or, in the case of thorium byproduct material, radium-228, averaged over the first 15 centimeters (cm) below the surface, and (ii) 15 pCi/g of radium-226, or, in the case of thorium byproduct material, radium-228, averaged over 15-cm thick layers more than 15 cm below the surface.

E.3.7 § 71.4 Definitions.

The following terms are as defined here for the purpose of this part. To ensure compatibility with international transportation standards, all limits in this part are given in terms of dual units: The International System of Units (SI) followed or preceded by U.S. standard or customary units. The U.S. customary units are not exact equivalents but are rounded to a convenient value, providing a functionally equivalent unit. For the purpose of this part, either unit may be used.

A_1 means the maximum activity of special form radioactive material permitted in a Type A package. This value is either listed in Appendix A, Table A-1, of this part, or may be derived in accordance with the procedures prescribed in Appendix A of this part.

A_2 means the maximum activity of radioactive material, other than special form material, LSA, and SCO material, permitted in a Type A package. This value is either listed in Appendix A, Table A-1, of this part, or may be derived in accordance with the procedures prescribed in Appendix A of this part.

Low Specific Activity (LSA) material means radioactive material with limited specific activity which is nonfissile or is excepted under §71.15, and which satisfies the descriptions and limits set forth below. Shielding materials surrounding the LSA material may not be considered in determining the estimated average specific activity of the package contents. LSA material must be in one of three groups:

(1) LSA--I.

(i) Uranium and thorium ores, concentrates of uranium and thorium ores, and other ores containing naturally occurring radioactive radionuclides which are not intended to be processed for the use of these radionuclides;

(ii) Solid unirradiated natural uranium or depleted uranium or natural thorium or their solid or liquid compounds or mixtures;

(iii) Radioactive material for which the A_2 value is unlimited; or

(iv) Other radioactive material in which the activity is distributed throughout and the estimated average specific activity does not exceed 30 times the value for exempt material activity concentration determined in accordance with Appendix A.

(2) LSA--II.

(i) Water with tritium concentration up to 0.8 TBq/liter (20.0 Ci/liter); or

(ii) Other material in which the activity is distributed throughout and the average specific activity does not exceed $10^{-4}A_2$/g for solids and gases, and $10^{-5}A_2$/g for liquids.

(3) LSA--III. Solids (e.g., consolidated wastes, activated materials), excluding powders, that satisfy the requirements of § 71.77, in which:

(i) The radioactive material is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.);

(ii) The radioactive material is relatively insoluble, or it is intrinsically contained in a relatively insoluble material, so that even under loss of packaging, the loss of radioactive material per package by leaching, when placed in water for 7 days, would not exceed 0.1 A_2; and

(iii) The estimated average specific activity of the solid does not exceed $2 \times 10^{-3}A_2$/g.

Low toxicity alpha emitters means natural uranium, depleted uranium, natural thorium; uranium-235, uranium-238, thorium-232, thorium-228 or thorium-230 when contained in ores or physical or chemical concentrates or tailings; or alpha emitters with a half-life of less than 10 days.

Surface Contaminated Object (SCO) means a solid object that is not itself classed as radioactive material, but which has radioactive material distributed on any of its surfaces. SCO must be in one of two groups with surface activity not exceeding the following limit:

(1) SCO-I: A solid object on which:

(i) The nonfixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 Bq/cm^2 (10^4 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 0.4 Bq/cm^2 (10^{-5} microcurie/cm^2) for all other alpha emitters;

(ii) The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 × 10^4 Bq/cm^2 (1.0 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 4 × 10^3 Bq/cm^2 (0.1 microcurie/cm^2) for all other alpha emitters; and

(iii) The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 4 × 10^4 Bq/cm^2 (1 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters, or 4 × 10^3 Bq/cm^2 (0.1 microcurie/cm^2) for all other alpha emitters.

(2) SCO-II: A solid object on which the limits for SCO-I are exceeded and on which:

(i) The nonfixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 400 Bq/cm^2 (10^2 microcurie/cm^2) for beta and gamma and low toxicity alpha emitters or 40 Bq/cm^2 (10^3 microcurie/cm^2) for all other alpha emitters;

(ii) The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8 × 10^5 Bq/cm2 (20 microcuries/cm^2) for beta and gamma and low toxicity alpha emitters, or 8 × 10^4 Bq/cm^2 (2 microcuries/cm^2) for all other alpha emitters; and

(iii) The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed 8 × 10^5 Bq/cm^2 (20 microcuries/cm^2) for beta and gamma and low toxicity alpha emitters, or 8 × 10^4 Bq/cm^2 (2 microcuries/cm^2) for all other alpha emitters.

E.3.8 § 71.14 Exemption for low-level materials.

(a) A licensee is exempt from all the requirements of this part with respect to shipment or carriage of the following low-level materials:

(1) Natural material and ores containing naturally occurring radionuclides that are not intended to be processed for use of these radionuclides, provided the activity concentration of the material does not exceed 10 times the values specified in Appendix A, Table A-2, of this part.

(2) Materials for which the activity concentration is not greater than the activity concentration values specified in Appendix A, Table A-2 of this part, or for which the consignment activity is not greater than the limit for an exempt consignment found in Appendix A, Table A-2, of this part.

(b) A licensee is exempt from all the requirements of this part, other than §§ 71.5 and 71.88, with respect to shipment or carriage of the following packages, provided the packages do not contain any fissile material, or the material is exempt from classification as fissile material under § 71.15:

(1) A package that contains no more than a Type A quantity of radioactive material;

(2) A package transported within the United States that contains no more than 0.74 TBq (20 Ci) of special form plutonium-244; or

(3) The package contains only LSA or SCO radioactive material, provided--

(i) That the LSA or SCO material has an external radiation dose of less than or equal to 10 mSv/h (1 rem/h), at a distance of 3 m from the unshielded material; or

(ii) That the package contains only LSA-I or SCO-I material.

E.3.9 § 110.22 General license for the export of source material.

(3) Th-227, Th-228, U-230, and U-232 when contained in a device, or a source for use in a device, in quantities of less than 100 millicuries of alpha activity (3.12 micrograms Th-227, 122 micrograms Th-228, 3.7 micrograms U-230, 4.7 milligrams U-232) per device or source.

E.3.10 § 110.23 General license for the export of byproduct material.

(2) Actinium-225 and -227, americium-241 and -242m, californium-248, -249, -250, -251, -252, -253, and -254, curium-240, -241, -242, -243, -244, -245, -246 and -247, einsteinium-252, -253, -254 and -255, fermium-257, gadolinium-148, mendelevium-258, neptunium-235 and -237,

polonium-210, and radium-223 must be contained in a device, or a source for use in a device, in quantities of less than 100 millicurie of alpha activity (see Sec. 110.2 for specific activity) per device or source, unless the export is to a country listed in Sec. 110.30. Exports of americium and neptunium are subject to the reporting requirements listed in paragraph (b) of this section.

(3) For americium-241, exports must not exceed one curie (308 milligrams) per shipment or 100 curies (30.8 grams) per year to any country listed in Sec. 110.29, and must be contained in industrial process control equipment or petroleum exploration equipment in quantities not to exceed 20 curies (6.16 grams) per device or 200 curies (61.6 grams) per year to any one country.

(5) For polonium-210, the material must be contained in static eliminators and may not exceed 100 curies (22 grams) per individual shipment.

(6) For tritium in any dispersed form, except for recovery or recycle purposes (e.g., luminescent light sources and paint, accelerator targets, calibration standards, labeled compounds), exports must not exceed the quantity of 10 curies (1.03 milligrams) or less per item, not to exceed 1,000 curies (103 milligrams) per shipment or 10,000 curies (1.03 grams) per year to any one country. Exports of tritium to the countries listed in Sec. 110.30 must not exceed the quantity of 40 curies (4.12 milligrams) or less per item, not to exceed 1,000 curies (103 milligrams) per shipment or 10,000 curies (1.03 grams) per year to any one country, and exports of tritium in luminescent safety devices installed in aircraft must not exceed a quantity of 40 curies (4.12 milligrams) or less per light source.

E.3.11 Policies and Practices

Disposition criteria for the release of materials and equipment that are not specified in NRC regulations are determined by the current policies and practices. NRC’s current approaches for making decisions on disposition of solid materials is different for materials licensees, i.e., industrial, research, and medical facilities, and for reactors, which include power, test, and research reactors. These are summarized in Table E-3, and discussed in more detail below.

For non-reactor licensees—materials licensees—licensee requests for release of solid material will continue to be evaluated using the nuclide concentration tables in Regulatory Guide 1.86 and its equivalent, Fuel Cycle Policy and Guidance Directive FC 83-23. Many materials licensees obtain approval, as a license condition, to routinely use these guidelines. For residual radioactivity within the volume of solid materials (for example, within a concrete or soil matrix), non-reactor licensee requests for release of solid material may continue to be approved under a disposal request (10 CFR 20.2002); a license termination plan; decommissioning plan review; or other specific license amendment. In verifying that the dose from such release is maintained ALARA and below the limits of our regulations in 10 Part 20, approval of a release is possible. The disposition of materials with volumetrically-distributed radioactivity from materials licensees is considered on a case-by-case basis with a reference of an annual individual dose criterion of a "few mrem per year (a few 0.01 mSv/a)."

Non-reactor licensees, that is, materials licensees, and reactor licensees have essentially the same detection level criteria for surface activity. But for materials licensees, radioactivity below these detection level criteria is allowed—detectable radioactivity is not allowed at any level for reactor licensees.

For reactor licensees, licensees may release of solid material using the "no detectable" policy of NRC's Inspection and Enforcement Circular 81-07 and Information Notices 85-92 and 88-22. For reactors, the policy is that released material can have no detectable licensed radioactivity. The levels of detection are specified by each reactor licensee's procedures and are frequently consistent with a now discontinued Regulatory Guide issued in 1974. In practice, these detection levels for radioactivity on surfaces are: 5/6 Bq /cm^2 (5000 dpm/100 cm^2) total β-γ and 1/6 Bq /cm^2 (1000 dpm/100 cm^2) removable β-γ. Non-detection at these levels of delectability wasconsidered to result in potential doses to an individual significantly less than 5 mrem/yr (<<0.05 mSv/a) from any non-detectable radioactivity that could remain on surfaces.

Detection levels for α-emitting radioactivity are specified as 1/60 Bq /cm^2 (100 dpm/100 cm^2) total and 1/300 Bq /cm^2 (20 dpm/100 cm^2) for removable α-emitting radioactivity. For volumetric radioactivity from reactors, the detection levels are from guidance written in the late 1970's and specifies β-γ concentrations in the general range of 3-4 Bq/kg (81-108 pCi/kg).

Table E.3 Summary of NRC Disposition Criteria from Current Practices for the Release of Materials and Equipment

	Surficial Radioactivity	**Volumetric Radioactivity**
Reactor Licenses	**β-γ**: Non-detectable [MDC 5/6 Bq/cm^2; 1/6 Bq/cm^2 removable]	**β-γ**: Non-detectable [MDC in General range of ≈ 3-4 Bq/kg]
	α: Non-detectable [MDC 1/60 Bq/cm^2; 1/300 Bq/cm^2 removable]	**α:** Non-detectable [MDC not indicated]
Materials Licenses	**β-γ:** 5/6 Bq/cm^2; 1/6 Bq/cm^2 removable[1]	**β-γ:** Case-by-case [Reference to a few 0.01 mSv in a year]
	α: 1/60 Bq/cm^2; 1/300 Bq/cm^2 removable[2]	**α:** Case-by-case [Reference to a few 0.01 mSv in a year]

[1] Except Sr-90, I-126, I-131, and I-133, where 1/6 Bq/cm^2 and 1/30 Bq/ cm^2 removable applies; and except I-125, and I-129 where 1/60 Bq/cm^2 and 1/300 Bq/ cm^2 removable applies.

[2] Except natural U, U-235, U-238, and associated decay products where 5/6 Bq/cm^2 and 1/6 Bq/cm^2 removable applies; and except transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, and Ac-227, where 1/60 Bq/cm^2 and 1/300 Bq/cm^2 removable applies.

E.3.12 Issues Related to International Trade

With regard to issues relating to international trade of solid materials released from facilities, NRC’s regulations contain requirements for export and import of material and could be considered in handling materials that meet established international clearance criteria and, at the same time, do not meet the guidelines for NRC licensees. Among other things, these regulations require that “the proposed import does not constitute an unreasonable risk to the public health and safety.”

F. SCAN MDCS FOR SECTION 7.3

The methodology used to determine the scan MDC is based on NUREG-1507 (NRC 1998b). An overview of the approach to determine scan MDCs follows:

- Calculate the fluence rate relative to the exposure rate (FRER) for the range of energies of interest (Section F.1),
- Calculate the probability of interaction (P) between the radiation of interest and the detector (Section F.2),
- Calculate the relative detector response (RDR) for each of the energies of interest (Section F.3),
- Determine the relationship between the detector's net count rate to net exposure rate in counts per minute per microRoentgen per hour, (cpm per μR/h, Section F.4),
- Determine the relationship between the detector response and the radionuclide concentration (Section F.5),
- Obtain the minimum detectable count rate (MDCR) for the ideal observer, for a given level of performance, by postulating detector background and a scan rate or observation interval (Section F.6), and
- Relate the MDCR for the ideal observer to a radionuclide concentration (in Bq/kg) to calculate the scan MDC (Section F.7).

F.1 Calculate the Relative Fluence Rate to Exposure Rate (FRER)

For particular gamma energies, the relationship of NaI scintillation detector count rate and exposure rate may be determined analytically (in cpm per μR/h). The approach is to determine the gamma fluence rate necessary to yield a fixed exposure rate (μR/h) as a function of gamma energy. The fluence rate, following NUREG-1507 (NRC 1998b), is directly proportional to the exposure rate and inversely proportional to the incident photon energy and mass energy absorption coefficient. That is,

$$\textit{Fluence Rate}(\mathrm{FRER}) \propto \dot{X}\,\frac{1}{E_\gamma}\,\frac{1}{(\mu_{en}/\rho)_{air}} \qquad \textbf{(F-1)}$$

Where:

$\dot{X}$ = the exposure rate (set equal to 1 μR/hr for these calculations)
E_γ = energy of the gamma photon of concern (keV)
$(\mu_{en}/\rho)_{air}$ = mass energy absorption coefficient in air at the gamma photon energy of concern (cm^2/g)

The mass energy absorption coefficients in air are presented in Table F-1 (natural uranium) and Table F-2 (natural thorium) along with the calculated fluence rates (up to a constant of proportionality, since only the ratios of these values are used in subsequent calculations). Note that while the mass energy absorption coefficients in air, $(\mu_{en}/\rho)_{air}$, are tabulated values (NIST 1996), the selected energies are determined by the calculation of the detector response based on radionuclide concentration (see Section F.5).

F.2 Calculate the Probability of Interaction

Assuming that the primary gamma interaction producing the detector response occurs through the end of the detector (i.e., through the beryllium window of the detector, as opposed to the sides), the probability of interaction (P) for a gamma may be calculated using Equation F-2:

$$\mathrm{P} = 1 - e^{-(\mu/\rho)_{\mathrm{NaI}}(x)(\rho_{\mathrm{NaI}})} = 1 - e^{-(0.117\ \mathrm{cm}^2/\mathrm{g})(0.16\ \mathrm{cm})(3.67\ \mathrm{g/cm}^3)} = 0.066 \text{ at } 400\ \mathrm{keV} \qquad \textbf{(F-2)}$$

Where:

P = probability of interaction (unitless)
$(\mu/\rho)_{NaI}$ = mass attenuation coefficient of FIDLER NaI crystal at the energy of interest (e.g., 0.117 cm^2/g at 400 keV)
x = thickness of the thin edge of the FIDLER NaI crystal (0.16 cm)
ρ = density of the NaI crystal (3.67 g/cm^3)

The mass attenuation coefficients for the NaI crystal and the calculated probabilities for each of the energies of interest are presented in Table F.1 (natural uranium) and Table F.2 (natural thorium). The mass attenuation coefficients for NaI were calculated using the XCOM program (NIST 1998).

Table F.1 Calculation of Detector Response to Natural Uranium

Energy (keV)	**$(\mu_{en}/\rho)_{air}$ (cm^2/g)**	**FRER (Section F.1)**	**$(\mu/\rho)_{NaI}$ cm^2/g**	**P (Section F.2)**	**RDR (Section F.3)**	**cpm per μR/h (Section F.4)**
15	1.334	0.04998	47.4	1.000	0.04998	28,374
20	0.5389	0.09278	21.8	1.000	0.09278	52,678
30	0.1537	0.2169	7.36	0.9867	0.2140	121,498
40	0.06833	0.3659	18.8	1.000	0.3659	207,725
50	0.04098	0.4880	10.5	0.9979	0.4870	276,511
60	0.03041	0.5481	6.45	0.9773	0.5356	304,123
80	0.02407	0.5193	3.00	0.8282	0.4301	244,204
100	0.02325	0.4301	1.67	0.6249	0.2688	152,606
150	0.02496	0.2671	0.611	0.3015	0.08052	45,717
200	0.02672	0.1871	0.328	0.1752	0.03278	18,613
300	0.02872	0.1161	0.166	0.09288	0.01078	6,120
400	0.02949	0.08477	0.117	0.06640	0.005629	3,196
500	0.02966	0.06743	0.0950	0.05426	0.003659	2,077
600	0.02953	0.05644	0.0822	0.04712	0.002660	1,510
662	0.02931	0.05154	0.0766	0.04398	0.002267	1,287
800	0.02882	0.04337	0.0675	0.03886	0.001685	957
1,000	0.02789	0.03586	0.0588	0.03394	0.001217	691
1,500	0.02547	0.02617	0.0470	0.02722	0.0007125	405
2,000	0.02345	0.02132	0.0415	0.02407	0.0005133	291

Table F.2 Calculation of Detector Response for Natural Thorium

Energy (keV)	$(\mu_{en}/\rho)_{air}$ (cm^2/g)	FRER (Section F.1)	$(\mu/\rho)_{NaI}$ cm^2/g	P (Section F.2)	RDR (Section F.3)	cpm per μR/h (Section F.4)
40	0.06833	0.3659	18.8	1.000	0.3659	207,725
60	0.03041	0.5481	6.45	0.9773	0.5356	304,123
80	0.02407	0.5193	3.00	0.8282	0.4301	244,204
100	0.02325	0.4301	1.67	0.6249	0.2688	152,606
150	0.02496	0.2671	0.611	0.3015	0.08052	45,717
200	0.02672	0.1871	0.328	0.1752	0.03278	18,613
300	0.02872	0.1161	0.166	0.09288	0.01078	6,120
400	0.02949	0.08477	0.117	0.06640	0.005629	3,196
500	0.02966	0.06743	0.0950	0.05426	0.003659	2,077
600	0.02953	0.05644	0.0822	0.04712	0.002660	1,510
662	0.02931	0.05154	0.0766	0.04398	0.002267	1,287
800	0.02882	0.04337	0.0675	0.03886	0.001685	957
1,000	0.02789	0.03586	0.0588	0.03394	0.001217	691
1,500	0.02547	0.02617	0.0470	0.02722	0.0007125	405
2,000	0.02343	0.02134	0.0415	0.02407	0.0005137	292
3,000	0.02057	0.01620	0.0368	0.02138	0.0003464	197

F.3 Calculate the Relative Detector Response

The relative detector response (RDR) for each of the energies of interest is determined by multiplying the FRER by P. The results are presented in Table F.1 (natural uranium) and Table F.2 (natural thorium).

F.4 Relationship Between Detector Response and Exposure Rate

Using the same methodology described in Sections F.1 through F.3, FRER, P, and RDR are calculated at the cesium-137 (^{137}Cs) energy of 662 keV and are presented in Table F.1 and Table F.2. The manufacturer of the FIDLER NaI detector provides an estimated response of the crystal in a known radiation field, which is 1,287 cpm per μR/h at the ^{137}Cs energy of 662 keV. The response at 662 keV can be used to determine the response at all other energies of interest using Equation F-3:

$$\frac{\text{cpm}}{\mu\text{R/h}}_{E_i} = \left(\frac{1{,}287\ \text{cpm}}{\mu\text{R/h}}\right) \times \frac{\text{RDR}_{E_i}}{\text{RDR}_{^{137}\text{Cs}}} \tag{F-3}$$

Where:

E_i = energy of the photon of interest (keV),

$\frac{\text{cpm}}{\mu\text{R/h}}_{E_i}$ = response of the detector for energies of interest, Table F.1 and Table F.2,

RDR_{E_i} = RDR at the energy of interest, Table F.1 and Table F.2, and

$\text{RDR}_{^{137}\text{Cs}}$ = RDR for ^{137}Cs, Table F.1 and Table F.2.

The responses in cpm per μR/h for each of the decay energies of interest are presented in Table F.1 and Table F.2.

F.5 Relationship Between Detector Response and Radionuclide Concentration

The minimum detectable exposure rate is used to determine the MDC by modeling a specific impacted area. The relationship between the detector response (in cpm) and the radionuclide concentration (in Bq/kg) uses a computer gamma dose modeling code to model the presence of a normalized 1 Bq/kg total activity source term for natural uranium and natural thorium. The following assumptions from NUREG-1507 (NRC 1998b) were used to generate the computer gamma dose modeling runs:

- Impacted media is concrete,
- Density of concrete is 2.3 g/cm^3,

- Activity is uniformly distributed into a layer of crushed concrete 15 cm thick,
- Measurement points are 10 cm above the concrete surface,
- Areas of elevated activity are circular with an area of 0.25 m^2 and a radius of 28 cm,
- 0.051 cm beryllium shield simulates the window of the FIDLER detector, and
- Normalized 1 Bq/kg source term decayed for 50 years to allow ingrowth of decay progeny.

The weighted cpm per μR/h response (weighted instrument sensitivity [WS_i]) for each decay energy is calculated by multiplying the μR/h at 1 Bq/kg (exposure rate with buildup, R_i) by the cpm per μR/h and dividing by the total μR/h (at 1 Bq/kg) for all decay energies of interest (equation F-4):

$$WS_i = \frac{R_i \times (\text{cpm per } \mu\text{R / h})}{R_T} \qquad \textbf{(F-4)}$$

Where:

WS_i = weighted instrument sensitivity (cpm per μR/h), and
R_i = exposure rate with buildup (μR/h)
R_T = Total exposure rate with buildup (μR/h)

Calculate the percent of FIDLER response for each of the decay energies of interest by dividing WS_i by the total weighted cpm per μR/h and multiplying by 100 percent (equation F-5):

$$\text{Percent of FIDLER response} = \frac{WS_i \times 100\%}{W_T} \qquad \textbf{(F-5)}$$

Where:

W_T = Total WS_i weighted instrument sensitivity (cpm per μR/h).

The exposure rates for each of the decay energies of interest are presented in Table F.3 (assuming natural uranium for the source term) and Table F.4 (assuming natural thorium for the source term).

Table F.3 Detector Response to Natural Uranium

Energy keV	**R_i (μR/h) (Section F.5)**	**cpm per μR/h (Section F.4)**	**WS_i (cpm per μR/h) (Section F.5)**	**Percent of FIDLER Response (Section F.5)**
15	4.473×10^{-10}	28,374	0	0.00%
20	3.597×10^{-12}	52,678	0	0.00%
30	2.623×10^{-07}	121,498	226	0.504%
40	1.299×10^{-10}	207,725	0	0.00%
50	1.052×10^{-07}	276,511	206	0.460%
60	5.065×10^{-06}	304,123	10903	24.3%
80	1.518×10^{-06}	244,204	2625	5.86%
100	2.309×10^{-05}	152,606	24938	55.7%
150	5.138×10^{-06}	45,717	1663	3.71%
200	2.881×10^{-05}	18,613	3796	8.48%
300	2.237×10^{-07}	6,120	10	0.0216%
400	2.434×10^{-07}	3,196	6	0.0123%
500	4.208×10^{-07}	2,077	6	0.0138%
600	2.048×10^{-06}	1,510	22	0.0489%
800	1.478×10^{-05}	957	100	0.224%
1,000	5.759×10^{-05}	691	282	0.629%
1,500	1.695×10^{-06}	405	5	0.0108%
2,000	2.841×10^{-07}	291	1	0.00131%
Total	1.413×10^{-04}		44,923	100%

Table F.4 Detector Response to Natural Thorium

Energy keV	**R_i (μR/h) (Section F.5)**	**cpm per μR/h (Section F.4)**	**WS_i (cpm per μR/h) (Section F.5)**	**Percent of FIDLER Response (Section F.5)**
40	1.299×10^{-06}	207,725	10	0.266%
60	1.816×10^{-06}	304,123	21	0.544%
80	1.989×10^{-04}	244,204	1855	47.8%
100	5.027×10^{-05}	152,606	293	7.55%
150	5.862×10^{-05}	45,717	102	2.64%
200	1.135×10^{-03}	18,613	807	20.8%
300	8.922×10^{-04}	6,120	209	5.37%
400	1.105×10^{-04}	3,196	13	0.348%
500	8.146×10^{-04}	2,077	65	1.67%
600	2.218×10^{-03}	1,510	128	3.30%
800	2.892×10^{-03}	957	106	2.72%
1,000	6.443×10^{-03}	691	170	4.38%
1,500	2.062×10^{-03}	405	32	0.821%
2,000	5.822×10^{-05}	292	1	0.0167%
3,000	9.249×10^{-03}	197	69	1.79%
Total	2.619×10^{-02}		3881	100%

F.6 Calculation of Scan Minimum Detectable Count Rates

In the computer gamma dose modeling, an impacted area with a radius of 28 cm or approximately 0.25 m was assumed. Using a scan speed of 0.25 meters per second (m/s) provides an observation interval of one second.

A typical background exposure rate is 10 μR/h. Using a conversion factor based upon field measurements of 1,287 cpm per μR/h for ^{137}Cs (see Section F.4) results in an estimated background count rate of 12,870 cpm. Converting this value from cpm to counts per second (cps) using Equation F-6 results in a background of 214.5 cps.

$$b(\text{cpm}) \times \frac{1\,\text{min}}{60\,\text{sec}} \times i(\text{sec}) = \frac{1{,}287\ \text{cpm}}{1\ \mu\text{R/h}} \times 10\ \mu\text{R/h} \times \frac{1\,\text{min}}{60\,\text{sec}} \times 1\,\text{sec} = 214.5\ \text{cps} \qquad \textbf{(F-6)}$$

Where:

b = background count rate (12,870 cpm)
i = the observation interval length (one second)

The MDCR is calculated using the methodology in NUREG-1507 (NRC 1998b) shown in Equations F-7 and F-8:

$$s_i = d'\sqrt{b_i} = 1.38 \times \sqrt{214.5} = 20.21\ \text{counts} \qquad \textbf{(F-7)}$$

$$s_{i,\ surveyor} = \frac{d'\sqrt{b_i}}{\sqrt{p}} = \frac{1.38 \times \sqrt{214.5}}{\sqrt{0.5}} = 28.58\ \text{counts}$$

$$\text{MDCR} = s_i \times \left(60/i\right) = 20.21 \times \left(60/1\right) = 1{,}212\ \text{cpm} \qquad \textbf{(F-8)}$$

$$\text{MDCR}_{surveyor} = s_{i,\ surveyor} \times \left(60/i\right) = 28.58 \times \left(60/1\right) = 1{,}715\ \text{cpm}$$

Where:

b_i = the average number of counts in the background interval (214.5 counts)
i = the observation interval length (one second)
p = efficiency of a less than ideal surveyor, range of 0.5 to 0.75 from NUREG-1507 (NRC 1998b); a value 0.5 was chosen as a conservative value

d' = detectability index from Table 6.1 of NUREG-1507 (NRC 1998b); a value of 1.38 was selected, which represents a true positive detection rate of 95% and a false positive detection rate of 60%
s_i = minimum detectable number of net source counts in the observation interval (counts)
$s_{i,surveyor}$ = minimum detectable number of net source counts in the observation interval by a less than ideal surveyor
MDCR = minimum detectable count rate (cpm)
$\text{MDCR}_{surveyor}$ = MDCR by a less than ideal surveyor (cpm)

F.7 Calculate the Scan Minimum Detectable Concentration

The scan minimum detectable concentration (MDC) can be calculated from the minimum detectable exposure rate (MDER). The MDER can be calculated using the previously calculated total weighted instrument sensitivities (WS_i), in cpm per μR/h, for natural uranium and natural thorium as shown in equations F-9 and F-10:

$$\text{MDER} = \frac{\text{MDCR}_{surveyor}}{W_T} \qquad \textbf{(F-9)}$$

$$\text{Scan MDC} = C \times \frac{\text{MDER}}{R_T} \qquad \textbf{(F-10)}$$

Where:

MDER = MDER for the “ith” source term, by a less than ideal surveyor, (µR/h)
$\text{MDCR}_{surveyor}$ = MDCR rate by a less than ideal surveyor (cpm), from Section F.5
W_T = Total weighted instrument sensitivity (cpm per µR/h, Table F.3 and Table F.4)
R_T = Total exposure rate with buildup (µR/h, Table F.3 and Table F.4)
C = concentration of source term (set at 1 Bq/kg in Section F.5)
Scan MDC = minimum detectable concentration (Bq/kg)

The Scan MDCs for the FIDLER were calculated using Equations F-9 and F-10, and the instrument response information from Table F.3 (assuming natural uranium as the source term) and Table F.4 (assuming natural thorium as the source term). The scan MDCs for natural uranium and natural thorium using a FIDLER are listed in Table F.5.

Table F.5 Scan MDCs for FIDLER

Source Term	MDCR$_{surveyor}$ (cpm) Section F.6	W_T (cpm per μR/h) Section F.5	MDER (μR/h) Section F.7	R_T (μR/h) Section F.5	C (Bq/kg) Section F.5	Scan MDC (Bq/kg) Section F.7
Natural Uranium	1,715	44,786	0.03829	1.413×10^{-04}	1	**271 ≈ 300**
Natural Thorium	1,715	3,881	0.4419	2.619×10^{-02}	1	**16.9 ≈ 20**

The scan MDCs of approximately 300 Bq/kg for uranium and 20 Bq/kg for thorium are both less than their respective NUREG-1640-based activity action levels of 38,000 and 330 Bq/kg, respectively.

APPENDIX G ESTABLISHING MQOS FOR MEASUREMENT UNCERTAINTY, MDCs AND MQCs

G.1 Establishing MQOs

This section provides the rationale and guidance for establishing project-specific MQOs for controlling σ_M. This control is achieved by establishing a desired maximum measurement method uncertainty at the upper boundary of the gray region. This control also will assist in both the measurement method selection process and in the evaluation of measurement data. Approaches applicable to several situations are detailed below.

Table G.1 Notation for Section G.1

Symbol	***Definition***	***Formula or reference***	***Type***
α	Probability of a Type I decision error		Chosen during DQO process
β	The probability of a Type II decision error		Chosen during DQO process
Δ	Width of the gray region	(UBGR-LBGR)	Chosen during DQO process
φ_{MR}	Required relative method uncertainty above the UBGR	u_{MR} / UBGR	Chosen during DQO process
S_{C}.	The critical value of the net instrument signal (e.g., net count)	Calculation of S_{C} requires the choice of a significance level for the test. The significance level is a specified upper bound for the probability, α, of a Type I error. The significance level is usually chosen to be 0.05.	If a measured value exceeds the critical value, a decision is made that radiation or radioactivity has been detected
σ	The total standard deviation of the data	$(\sigma_S^2 + \sigma_M^2)^{½}$	Theoretical population parameter
σ_S	Standard deviation of the concentration in the sampled population		Theoretical population parameter
σ_M	Standard deviation of the measurement method		Theoretical population parameter
u_{MR}	Required method uncertainty at and below the UBGR	Upper bound to the value of σ_M	Chosen during DQO process
$u_c^2(y)$	Combined variance of y	Uncertainty propagation	
$u_{\mathrm{c}}(y)$	Combined standard uncertainty of y.	Uncertainty propagation	
$z_{1-\alpha}$ $(z_{1-\beta})$	1-α (or 1-β) quantile of a standard normal distribution function	Table of Standard normal distribution.	Theoretical

G.1.1 Developing a Requirement for Measurement Method Uncertainty For MARSSIM-Type surveys

When, as in MARSSIM-Type surveys, a decision is to be made about the mean of a sampled population, generally the average of a set of measurements on a survey unit is compared to the disposition criterion.

The total variance of the data, σ^2, is the sum of two components

$$\sigma^2 = \sigma_M^2 + \sigma_S^2 \tag{G-1}$$

Where:

σ_M^2 = measurement method variance (M = “measurement”), and

σ_S^2 = variance of the radionuclide concentration or activity concentration in the sampled population (S = “sampling”).

The spatial and temporal distribution of the concentration, the extent of the survey unit, the physical sizes of the measured material, and the choice of measurement locations may affect the sampling standard deviation, σ_S. The measurement standard deviation, σ_M , is affected by the measurement methods. The value of σ_M is estimated in MARSAME by the combined standard uncertainty of a measured value for a measurement of material whose concentration equals the hypothesized population mean concentration. The calculation of measurement uncertainties is covered in Section 5.6.

Four cases are considered below where target values for σ_M can be suggested depending on what is known about σ_S. Cases 1 and 2 treat the desired overall objective of keeping $\Delta/\sigma \approx 3$ or higher. When this is not possible, Cases 3 and 4 treat the less desirable alternative of attempting to prevent Δ/σ from going lower than 1.

Case 1: σ_S is known relative to Δ / 3

Generally, it is easier to control σ_M than σ_S. If σ_S is known (approximately), a target value for σ_M can be determined.

Case 1a: $\sigma_S \leq \Delta / 3$

If $\sigma_S \leq \Delta / 3$, then a value of σ_M no greater than $\sqrt{(\Delta^2 / 9) - \sigma_S^2}$ ensures that $\sigma \leq \Delta / 3$, because we have $\sigma^2 = \sigma_M^2 + \sigma_S^2 \leq (\Delta^2 / 9 - \sigma_S^2) + \sigma_S^2 = \Delta^2 / 9$, as desired.

Case 1b: $\sigma_S > \Delta / 3$

If $\sigma_S > \Delta / 3$, the requirement that the total σ be less than $\Delta/3$ cannot be met regardless of σ_M. In this case, it is sufficient to make σ_M negligible in comparison to σ_S. Generally, σ_M can be considered negligible in comparison to σ_S if it is no greater than $\sigma_S/3$.

Case 2: σ_S is not known relative to $\Delta / 3$

Often one needs a method for choosing σ_M in the absence of specific information about σ_S. Since it is desirable to have $\sigma \leq \Delta / 3$, this condition is adopted as a primary requirement. Assume for the moment that σ_S is large. Then σ_M should be made negligible by comparison. As mentioned above, σ_M can be considered negligible if it is no greater than $\sigma_S/3$. When this condition is met, further reduction of σ_M has little effect on σ and therefore is usually not cost-effective. So, the inequality $\sigma_M \leq \sigma_S/3$ is adopted as a secondary requirement.

Starting with the definition $\sigma^2 = \sigma_M^2 + \sigma_S^2$ and substituting the secondary requirement $\sigma_M \leq \sigma_S/3$ we get $\sigma^2 \geq \sigma_M^2 + 9\sigma_M^2 = 10\sigma_M^2$, thus

$$\sigma_M \leq \frac{\sigma}{\sqrt{10}} \tag{G-2}$$

Substituting the primary requirement that $\Delta/\sigma \geq 3$ (i.e., $\sigma \leq \Delta / 3$) we get $\sigma_M \leq \frac{\sigma}{\sqrt{10}} \leq \frac{\Delta / 3}{\sqrt{10}}$, thus

$$\sigma_M \leq \frac{\Delta}{3\sqrt{10}} \tag{G-3}$$

Or approximately

$$\sigma_M \leq \frac{\Delta}{10} \tag{G-4}$$

The required upper bound for the standard deviation σ_M will be denoted by σ_{MR}. MARSAME recommends the equation

$$\sigma_{MR} = \frac{\Delta}{10} \tag{G-5}$$

by default as a requirement when σ_S is unknown and a decision is to be made about the mean of a sampled population.

This upper bound was derived from the assumption that σ_S was large, but it also ensures that the primary requirement $\sigma \leq \Delta / 3$ (i.e., $\Delta / \sigma \geq 3$) will be met if σ_S is small. When the measurement standard deviation σ_M is less than σ_{MR}, the primary requirement will be met unless the sampling variance, σ_S^2, is so large that σ_M^2 is negligible by comparison, in which case little benefit can be obtained from further reduction of σ_M.

It may be that the primary requirement that Δ/σ be at least 3 is not achievable. Suppose that the primary requirement is relaxed to achieving Δ/σ at least 1 (i.e., $\sigma \leq \Delta$). This leads to consideration of:

Case 3: σ_S is known relative to Δ

As in Case 1, it is generally easier to control σ_M than σ_S. If σ_S is known (approximately), a target value for σ_M can be determined.

Case 3a: $\sigma_S \leq \Delta$

If $\sigma_S \leq \Delta$, then a value of σ_M no greater than $\sqrt{\Delta^2 - \sigma_S^2}$ ensures that $\sigma \leq \Delta$, because we have $\sigma^2 = \sigma_M^2 + \sigma_S^2 \leq (\Delta^2 - \sigma_S^2) + \sigma_S^2 = \Delta^2$ as desired.

Case 3b: $\sigma_S > \Delta$

If $\sigma_S > \Delta$, the requirement that the total σ be less than Δ cannot be met regardless of σ_M. In this case, it is sufficient to make σ_M negligible in comparison to σ_S. Generally, σ_M can be considered negligible if it is no greater than $\sigma_S/3$.

Case 4: σ_S is not known relative to Δ

Suppose $\sigma \le \Delta$ is adopted as the primary requirement. As in Case 2, if σ_S is large then σ_M should be made negligible by comparison. As mentioned above, σ_M can be considered negligible if it is no greater than $\sigma_S/3$. When this condition is met, further reduction of σ_M has little effect on σ and therefore is usually not cost-effective. So, the inequality $\sigma_M \le \sigma_S/3$ is adopted as a secondary requirement.

Starting with the definition $\sigma^2 = \sigma_M^2 + \sigma_S^2$ and substituting the secondary requirement $\sigma_M \le \sigma_S / 3$ we get $\sigma^2 \ge \sigma_M^2 + 9\sigma_M^2 = 10\sigma_M^2$, thus

$$\sigma_M \le \frac{\sigma}{\sqrt{10}}$$

Substituting the primary requirement that $\Delta/\sigma \ge 1$ (i.e., $\sigma \le \Delta$) we get $\sigma_M \le \frac{\sigma}{\sqrt{10}} \le \frac{\Delta}{\sqrt{10}}$, thus

$$\sigma_M \le \frac{\Delta}{\sqrt{10}} \approx \frac{\Delta}{3}$$

G.1.2 Developing a Requirement for Measurement Method Uncertainty When Decisions Are to Be Made About Individual Items

When decisions are to be made about individual items, the total variance of the data equals the measurement variance, σ_M^2, and the data distribution in most instances should be approximately normal. The decision in this case may be made by comparing the measured concentration, x, plus or minus a multiple of its combined standard uncertainty, to the action level. The combined standard uncertainty, $u_c(x)$, is assumed to be an estimate of the true standard deviation of the measurement process as applied to the item being measured; so, the multiplier of $u_c(x)$ equals $z_{1-\alpha}$, the $(1 - \alpha)$-quantile of the standard normal distribution (see MARLAP appendix C).

Alternatively, if AL = 0, so that any detectable amount of radioactivity is of concern, the decision may involve comparing the net instrument signal (e.g., count rate) to the critical value of the concentration, S_C, as defined in Section 5.7.1.

Two cases are considered below where target values for σ_M can be suggested depending on what is known about the width of the gray region and the desired Type I and Type II decision error rates. Case 5 is for Scenario A, and Case 6 is for Scenario B.

Case 5: Suppose the null hypothesis is $X \geq \text{AL}$ (see Scenario A in Chapter 4), so that the action level is the upper bound of the gray region. Given the measurement variance σ_M^2, only a measured result that is less than (UBGR – $z_{1-\alpha}\sigma_M$) will be judged to be clearly less than the action level. Then the desired power of the test $1 - \beta$ is achieved at the lower bound of the gray region only if the $\text{LBGR} \leq \text{UBGR} - z_{1-\alpha}\sigma_M - z_{1-\beta}\sigma_M$. Algebraic manipulation transforms this requirement to

$$\sigma_M \leq \frac{\text{UBGR - LBGR}}{z_{1-\alpha} + z_{1-\beta}} = \frac{\Delta}{z_{1-\alpha} + z_{1-\beta}} \qquad \textbf{(G-6)}$$

Case 6: Suppose the null hypothesis is $X \leq \text{AL}$ (see Scenario B in Chapter 4), so that the action level is the lower bound of the gray region. In this case, only a measured result that is greater than LBGR + $z_{1-\alpha}\sigma_M$ will be judged to be clearly greater than the action level. The desired power of the test $1 - \beta$ is achieved at the upper bound of the gray region only if the $\text{UBGR} \geq \text{LBGR} + z_{1-\alpha}\sigma_M + z_{1-\beta}\sigma_M$. Algebraic manipulation transforms this requirement to:

$$\sigma_M \leq \frac{\text{UBGR - LBGR}}{z_{1-\alpha} + z_{1-\beta}} = \frac{\Delta}{z_{1-\alpha} + z_{1-\beta}}$$

So, in either Scenario A or Scenario B, the requirement remains that:

$$\sigma_M \leq \frac{\Delta}{z_{1-\alpha} + z_{1-\beta}} \qquad \textbf{(G-7)}$$

Therefore, MARSAME uses the equation:

$$u_{MR} = \sigma_{MR} = \frac{\Delta}{z_{1-\alpha} + z_{1-\beta}} \qquad \textbf{(G-8)}$$

as an MQO for method uncertainty when decisions are to be made about individual items or locations and not about population parameters.

If both α and β are at least 0.05, one may use the value $u_{MR} = 0.3\Delta$.

The recommended value of u_{MR} is based on the assumption that any known bias in the measurement process has been corrected and that any remaining bias is well less than a third of the method uncertainty.

G.2 Uncertainty Calculation

Table G.2 Notation for Section G.2

Symbol	***Definition***	***Formula or reference***	***Type***
a	Half-width of a bounded probability distribution	Type B evaluation of uncertainty	Estimated
c_i	Sensitivity coefficient	$\partial f / \partial x_i$, the partial derivative of f with respect to x_i	Evaluated at the measured values $x_1, x_2, \ldots, x_N$
$f(x_1, x_2, \ldots, x_N)$	The calculated value of the output quantity from measurable input quantities for a particular measurement	$y = f(x_1, x_2, \ldots, x_N)$	Experimental
$f(X_1, X_2, \ldots, X_N)$	Model equation expressing the mathematical relationship, between the measurand, Y and the input quantities X_i.	$Y = f(X_1, X_2, \ldots, X_N)$	Theoretical
k	Coverage factor for expanded uncertainty	Numerical factor used as a multiplier of the combined standard uncertainty in order to obtain an expanded uncertainty	Chosen during DQO process
p	Coverage probability for expanded uncertainty	Probability that the interval surrounding the result of a measurement determined by the expanded uncertainty will contain the value of the measurand	Chosen during DQO process
$r(x_i,x_j)$	Correlation coefficient for two input estimates, x_i and x_j,	$u(x_i,x_j) / (u(x_i)\, u(x_j))$	Experimental
$s(x_i)$	Sample standard deviation of the input estimate x_i	$s(x_i) = \sqrt{\frac{1}{(n-1)} \sum_{k=1}^{n} (x_{i,k} - \overline{x}_i)^2}$	Experimental
$u(x_i)$	Type B standard uncertainty of the input estimate x_i		Estimated
$u_i(y)$	Component of the combined standard uncertainty $u_c(y)$ generated by the standard uncertainty of the input estimate x_i, $u(x_i)$	$u_i(y) = c_i\, u(x_i)$	Estimated
$u_c(y)$	Combined standard uncertainty of y.	Uncertainty propagation	
$u_c^2(y)$	Combined variance of y	Uncertainty propagation	

Table G.2 Notation for Section G.2 (continued)

Symbol	***Definition***	***Formula or reference***	***Type***
U	Expanded uncertainty	"Defining an interval about the result of a measurement that may be expected to encompass a large fraction of values that could reasonably be attributed to the measurand" (GUM)	
$u(x_i,x_j)$	Covariance of two input estimates, x_i and x_j,		Experimental
$u_c(y)/y$	Relative combined standard uncertainty of the output quantity for a particular measurement		Experimental
$u(x_i)/x_i$	Relative standard uncertainty of a nonzero input estimate x_i for a particular measurement		Experimental
$w_1, w_2, \ldots, w_n$	input quantities appearing in the numerator of $y = f(x_1, x_2, \ldots, x_N)$	See $z_1, z_2, \ldots, z_m$ below	
$X_1, X_2, \ldots, X_N$	Measurable input quantities		Theoretical
$x_1, x_2, \ldots, x_N$	Estimates of the measurable input quantities for a particular measurement		Experimental
Y	The output quantity or measurand		Theoretical
y	Estimate of the output quantity for a particular measurement		Experimental
$z_1, z_2, \ldots, z_m$	input quantities appearing in the denominator of $y = f(x_1, x_2, \ldots, x_N)$	$N=n+m$	

G.2.1 Procedures for Evaluating Uncertainty

The usual eight steps for evaluating and reporting the uncertainty of a measurement are summarized in the following subsections (adapted from Chapter 8 of the GUM):

G.2.1.1 Identify the Measurand, Y, and all the Input Quantities, X_i, for the Mathematical Model

Include all quantities whose variability or uncertainty could have a potentially significant effect on the result. Express the mathematical relationship, $Y = f(X_1, X_2, \ldots, X_N)$, between the measurand and the input quantities.

The procedure for assessing the uncertainty of a measurement begins with listing all significant sources of uncertainty in the measurement process. A good place to begin is with the input quantities' mathematical model $Y = f(X_1, X_2, \ldots, X_N)$. When an effect in the measurement process that is not explicitly represented by an input quantity has been identified and quantified, an additional quantity should be included in the mathematical measurement model to correct for it. The quantity, called a correction (additive with a nominal value of zero) or correction factor (multiplicative with a nominal value of one), will have an uncertainty that should also be evaluated and propagated. Each uncertainty that is potentially significant should be evaluated quantitatively.

G.2.1.2 Determine an Estimate, x_i, of the Value of Each Input Quantity, X_i

This involves simply determining for the particular measurement at hand, the specific value, x_i, that should be substituted for the input quantity X_i in the mathematical relationship, $Y = f(X_1, X_2, \ldots, X_N)$.

G.2.1.3 Evaluate the Standard Uncertainty, $u(x_i)$, for Each Input Estimate, x_i, Using a Type A Method, a Type B Method, or a Combination of Both

Methods for evaluating standard uncertainties are classified as either "Type A" or "Type B" (NIST, 1994). Both types of uncertainty need to be taken into consideration. A Type A evaluation of an uncertainty uses a series of measurements to estimate the standard deviation empirically. Any other method of evaluating an uncertainty is a Type B method. A Type B evaluation of standard uncertainty is usually based on scientific judgment using all the relevant information available, which may include:

- Previous measurement data,
- Experience with, or general knowledge of, the behavior and property of relevant materials and instruments,
- Manufacturer's specifications,
- Data provided in calibration and other reports, and
- Uncertainties assigned to reference data taken from handbooks.

The Type A standard uncertainty of the input estimate x_i is defined to be the experimental standard deviation of the mean:

$$u(x_i) = \sqrt{\frac{1}{n(n-1)}\sum_{k=1}^{n}(x_{i,k} - \overline{x_i})^2} = s(x_i)/\sqrt{n} \qquad \textbf{(G-9)}$$

Example 1: Type A uncertainty calculation using equation G-9**:**

Ten independent one-minute measurements of the counts from a check source X_i were made with a digital survey meter, yielding the values: 12,148, 12,067, 12,207, 12,232, 12,284, 12,129, 11,862, 11,955, 12,044, and 12,150.

The estimated value x_i is the arithmetic mean of the values $X_{i,k}$.

$$x_i = X_i \frac{1}{n}\sum_{k=1}^{n} x_{i,k} = \frac{121078}{10} = 12107.8$$

The standard uncertainty of x_i is

$$u(x_i) = \sqrt{\frac{1}{n(n-1)}\sum_{k=1}^{n}(x_{i,k} - \overline{x_i})^2} = \sqrt{\frac{1}{10(10-1)}\sum_{k=1}^{10}(x_{i,k} - 12107.8)^2}$$

$$= \sqrt{16628.84} = 128.95$$

There are other Type A methods, but all are based on repeated measurements.

Any evaluation of standard uncertainty that is not a Type A evaluation is a Type B evaluation. Sometimes a Type B evaluation of uncertainty involves making a best guess based on all available information and professional judgment. Despite the reluctance to make this kind of evaluation, it is almost always better to make an informed guess about an uncertainty component than to ignore it completely.

There are many ways to perform Type B evaluations of standard uncertainty. One example of a Type B method is the estimation of counting uncertainty using the square root of the observed counts. If the observed count is N, when the Poisson approximation is used, the standard uncertainty of N may be evaluated as $u(N) = \sqrt{N}$. For example, the standard uncertainty of the first value in Example 1, 12,148, could be estimated as $\sqrt{12148} = 110.218$. When N may be very small or even zero, the equation $u(N) = \sqrt{N+1}$ may be preferable.

Another Type B evaluation of an uncertainty $u(x)$ consists of estimating an upper bound, a, for the magnitude of the error of x based on professional judgment and the best available information. If nothing else is known about the distribution of the measured result, then after a is estimated, the standard uncertainty may be calculated using the equation

$$u(x) = \frac{a}{\sqrt{3}}, \qquad \textbf{(G-10)}$$

which is the standard deviation of a random variable uniformly distributed over the interval (x - a, x + a). The variable a is called the half-width of the interval. Suppose in Example 1, all that was given was the observed range of the data from an analog survey meter dial, i.e., from 11,862 to 12,284, a difference of 422. If it was assumed that the data came from a uniform distribution across this range, then the average is (11,862+12,284)/2 = 12,073, and an estimate of the standard uncertainty would be $u(x) = \frac{211}{\sqrt{3}} = 121.821$.

Given the same information on the range, if values near the middle of the range were considered more likely than those near the endpoints, a triangular distribution may be more appropriate. The mean would be the same as above, 12,073. However the standard uncertainty then be calculated using the equation

$$u(x) = \frac{a}{\sqrt{6}} = \frac{211}{\sqrt{6}} = 86.14 \qquad \textbf{(G-11)}$$

which is the standard deviation of a random variable with a triangular distribution over the interval (x - a, x + a).

When the estimate of an input quantity is taken from an external source, such as a book or a calibration certificate, the stated standard uncertainty can be used.

G.2.1.4 Evaluate the Covariances, *u*(x_i,x_j), for all Pairs of Input Estimates with Potentially Significant Correlations

A Type A evaluation of the covariance of the input estimates x_i = and x_j = is

$$u(x_i, x_j) = \frac{1}{n(n-1)} \sum_{k=1}^{n} (x_{i,k} - \overline{x_i})(x_{j,k} - \overline{x_j}) \qquad \textbf{(G-12)}$$

An evaluation of variances and covariances of quantities determined by the method of least squares may also be a Type A evaluation. Evaluation of the covariance of two input estimates, x_i and x_j, whose uncertainties are evaluated by Type B methods may require expert judgment. In such cases it may be simpler to estimate the correlation coefficient, $r(x_i,x_j) = [u(x_i,x_j) / u(x_i) \cdot u(x_j)]$, first and then multiply it by the standard uncertainties, $u(x_i)$ and $u(x_j)$ to obtain the covariance, $u(x_i,x_j)$.

A covariance calculation is demonstrated in Example 2 in Section G.2.2.

G.2.1.5 Calculate the Estimate, *y*, of the Measurand from the Relationship $y = f(x_1,x_2,\ldots,x_N)$

This involves simply substituting, for the particular measurement at hand, the specific values of x_i for the input quantity X_i into the mathematical relationship, $Y = f(X_1,X_2,\ldots,X_N)$, and calculating the result $y = f(x_1,x_2,\ldots,x_N)$.

G.2.1.6 Determine the Combined Standard Uncertainty, $u_c(y)$, of the Estimate, *y*

The combined standard uncertainty of *y* is obtained using the following formula:

$$u_c^2(y) = \sum_{i=1}^{N} \left(\frac{\partial f}{\partial x_i} \right)^2 u^2(x_i) + 2 \sum_{i=1}^{N-1} \sum_{j=i+1}^{N} \frac{\partial f}{\partial x_i} \frac{\partial f}{\partial x_j} u(x_i, x_j) \qquad \textbf{(G-13)}$$

Here $u^2(x_i)$ denotes the estimated variance of x_i, or the square of its standard uncertainty; $u(x_i,x_j)$ denotes the estimated covariance of x_i and x_j; $\partial f / \partial x_i$ (or $\partial y / \partial x_i$) denotes the partial derivative of *f* with respect to x_i evaluated at the measured values $x_1,x_2,\ldots,x_N$; and $u_c^2(y)$ denotes the combined variance of *y*, whose positive square root, $u_c(y)$, is the combined standard uncertainty of *y*. The partial derivatives, $\partial f / \partial x_i$, are called sensitivity coefficients, usually denoted c_i. The sensitivity coefficient measures how much *f* changes when x_i changes. Equation G-13 is called the "law of propagation of uncertainty" in the GUM (ISO 1995).

If the input estimates $x_1, x_2, \ldots, x_N$ are uncorrelated, the uncertainty propagation formula reduces to

$$u_c^2(y) = \sum_{i=1}^{N} \left(\frac{\partial f}{\partial x_i} \right)^2 u^2(x_i) \qquad \textbf{(G-14)}$$

Suppose the values $x_1, x_2, \ldots, x_N$ are composed of two groups $w_1, w_2, \ldots, w_n$ and $z_1, z_2, \ldots, z_m$ *with* $N = n + m$. If the w's and the z's are uncorrelated and nonzero, the combined standard uncertainty of $y = \dfrac{w_1 w_2 \ldots w_n}{z_1 z_2 \ldots z_m}$ may be calculated from the formula:

$$u_c^2(y) = y^2 \left(\frac{u^2(w_1)}{w_1^2} + \frac{u^2(w_2)}{w_2^2} + \ldots + \frac{u^2(w_n)}{w_n^2} + \frac{u^2(z_1)}{z_1^2} + \frac{u^2(z_2)}{z_2^2} + \ldots + \frac{u^2(z_m)}{z_m^2} \right) \qquad \textbf{(G-15)}$$

The symbols $z_1, z_2, \ldots, z_m$ have been introduced simply to differentiate those values appearing in the denominator of the model equation from the $w_1, w_2, \ldots, w_n$ appearing in the numerator.

If $y = \dfrac{f(w_1, w_2, \ldots, w_n)}{z_1 z_2 \ldots z_m}$, where *f* is some specified function of $w_1, w_2, \ldots, w_n$, all the z_i are nonzero, and all the input estimates are uncorrelated. Then:

$$u_c^2(y) = \frac{u_c^2(f(w_1, w_2, \ldots, w_n))}{z_1 z_2 \ldots z_m} + y^2 \left(\frac{u^2(z_1)}{z_1^2} + \frac{u^2(z_2)}{z_2^2} + \ldots + \frac{u^2(z_m)}{z_m^2} \right) \qquad \textbf{(G-16)}$$

An alternative to uncertainty propagation is the use of computerized Monte Carlo methods to propagate not the uncertainties of input estimates but their distributions. Given assumed distributions for the input estimates, the method provides an approximate distribution for the output estimate, from which the combined standard uncertainty or an uncertainty interval may be derived.

G.2.1.7 Optionally Multiply $u_c(y)$ by a Coverage Factor k to Obtain the Expanded Uncertainty U such that the Interval $[y - U, y + U]$ can be Expected to Contain the Value of the Measurand with a Specified Probability

The specified probability, p, is called the level of confidence or the coverage probability and is generally only an approximation of the true probability of coverage. When the distribution of the measured result is approximately normal, the coverage factor is often chosen to be $k = 2$ for a coverage probability of approximately 95%. An expanded uncertainty calculated with $k = 2$ or 3 is sometimes informally called a "two-sigma" or "three-sigma" uncertainty, respectively. The GUM recommends the use of coverage factors in the 2 to 3 range when the combined standard uncertainty represents a good estimate of the true standard deviation. Attachment 19D of MARLAP describes a more general procedure for calculating the coverage factor that gives a desired coverage probability p when there is substantial uncertainty in the value of $u_c(y)$.

G.2.1.8 Report the Result as $y \pm U$ with the Unit of Measurement

At a minimum, state the coverage factor used to compute U and the estimated coverage probability. Alternatively, report the result, y, and its combined standard uncertainty, $u_c(y)$, with the unit of measurement.

The number of significant figures that should be reported for the result of a measurement depends on the uncertainty of the result. A common convention, recommended by MARLAP, is to round the uncertainty (standard uncertainty or expanded uncertainty) to two significant figures and to report both the measured value and the uncertainty to the same number of decimal places. Only final results should be rounded in this manner. Intermediate results in a series of calculation steps should be carried through all steps with additional figures to prevent unnecessary round-off errors. Additional figures are also recommended when the data are stored electronically. Rounding should be performed only when the result is reported.

All results, whether positive, negative, or zero, should be reported as obtained, together with their uncertainties.

A measured value y of a quantity Y that is known to be positive may be so far below zero that it indicates a possible blunder, procedural failure, or other quality control problem. Usually, if

$y + 3u_c(y) < 0$, the result may be invalid. For example, if $y = -10$ and $u_c(y) = 1$, this would imply that Y is negative with high probability, which is known to be impossible. However, if $y = -1$ and $u_c(y) = 1$, the expanded uncertainty covers positive values with reasonable probability. The accuracy of the uncertainty estimate $u_c(y)$ must be considered in evaluating such results, especially in cases where only few counts are observed during the measurement and counting uncertainty is the dominant component of $u_c(y)$. (See MARLAP Chapter 18 and Attachment 19D).

G.2.2 Examples of Some Parameters that Contribute to Uncertainty

The sources of uncertainty described in the following sections, drawn from MARLAP Section 19.5, should be considered.

G.2.2.1 Instrument Background

Single-channel background measurements are usually assumed to follow the Poisson model, in which the uncertainty in the number of counts obtained, N, is given by $\sqrt{N}$. There may be effects that increase the variance beyond what the model predicts. For example, cosmic radiation and other natural sources of instrument background may vary between measurements, the instrument may become contaminated, or the instrument may simply be unstable. Generally, the variance of the observed background is somewhat greater than the Poisson counting variance, although for certain types of instruments, the Poisson model may overestimate the background variance (Currie et al., 1998). If the background does not closely follow the Poisson model, its variance should be estimated by repeated measurements.

The “instrument background,” or “instrument blank,” is usually measured under the same conditions that will be encountered in the field. Ambient background sources should be minimized, and kept constant during the measurements of M&E. Periodic checks should be made to ensure that the instrument has not picked up additional radioactivity from the M&E during the measurements. If the background drifts or varies nonrandomly over time (i.e., is nonstationary), it is important to minimize the consequences of the drift by performing frequent background measurements.

If repeated measurements demonstrate that the background level is stable, then the average, $\bar{x}$, the results of *n* similar measurements performed over a period of time may give the best estimate of the background. In this case, if all measurements have the same duration, the experimental standard deviation of the mean, $s(\bar{x})$, is also a good estimate of the measurement uncertainty.

Given the Poisson assumption, the best estimate of the uncertainty is still the Poisson estimate, which equals the square root of the summed counts, divided by the number of measurements, $\sqrt{n\bar{x}}/n = \sqrt{\bar{x}/n}$ but the experimental standard deviation may be used when the Poisson assumption is invalid. It is always wise to compare the value of $s(\bar{x})$ to the value of the Poisson uncertainty when possible to identify any discrepancies.

G.2.2.2 Counting Efficiency

The counting efficiency for a measurement of radioactivity (usually defined as the detection probability for a particle or photon of interest emitted by the source) may depend on many factors, including source geometry, placement, composition, density, activity, radiation type and energy and other instrument-specific factors. The estimated efficiency is sometimes calculated explicitly as a function of such variables (in gamma-ray spectroscopy, for example). In other cases a single measured value is used (e.g., alpha-particle spectrometry). If an efficiency function is used, the uncertainties of the input estimates, including those for both calibration parameters and sample-specific quantities, must be propagated to obtain the combined standard uncertainty of the estimated efficiency. Calibration parameters tend to be correlated; so, estimated covariances must also be included. If a single value is used instead of a function, the standard uncertainty of the value is determined when the value is measured. An example of the calculation of the uncertainty in counting efficiency is given in Example 2.

Example 2; A radiation counter is calibrated, taking steps to ensure that the geometry of the source position, orientation of the source, pressure, temperature, relative humidity, and other factors that could contribute to uncertainty are controlled, as described below:

The standard source is counted 15 times on the instrument for 300 s.

The radionuclide is long-lived; so, no decay corrections are needed. The uncertainties of the count times are assumed to be negligible.

Within the range of linearity of the instrument, the mathematical model for the calibration is:

$$\varepsilon = \frac{1}{n}\sum_{i=1}^{n}\frac{(N_{S,i}/t_S)-(N_B/t_B)}{a_s} \qquad \textbf{(G-17)}$$

Where:

ε is the counting efficiency,
n is the number times the source is counted (15),
$N_{S,i}$ is the gross count observed during the i^{th} measurement of the source,
t_S is the source count time (300 s),
N_B is the observed background count (87),
t_B is the background count time (6,000 s),
a_S is the activity of the standard source (150.0 Bq). The standard uncertainty of the source, 2.0 Bq, was given by the certificate for the source.

The combined standard uncertainty of ε can be evaluated using Equation G-13. For the purpose of uncertainty evaluation, it is convenient to rewrite the model as:

$$\varepsilon = \frac{\overline{R}}{a_s}$$

Where:

$$\overline{R} = \frac{1}{n}\sum_{i=1}^{n} R_i \quad \text{and} \quad R_i = (N_{S,i}/t_S)-(N_B/t_B)\,, \qquad i = 1,2,\ldots,n$$

The values R_i and their average, $\overline{R}$, are estimates of the count rate produced by the standard, while $\overline{R}/a_\mathrm{S}$ is an estimate of the count rate produced by 1 Bq of activity. The standard uncertainty of $\overline{R}$ can be evaluated experimentally from the 15 repeated measurements: $u^2(\overline{R}) = s^2(\overline{R}) = \frac{1}{n(n-1)}\sum_{i=1}^{n}(R_i-\overline{R})^2$. Since only one background measurement was made, the input estimates R_i are correlated with each other. The uncertainty of N_B, $u(N_B)=\sqrt{87}$, using a Type B evaluation based on an assumption of a Poisson distribution for the number of background counts.

The covariance between R_i and R_j, for $i \neq j$, may be estimated as

$$u(R_i, R_j) = \frac{\partial R_i}{\partial N_B}\frac{\partial R_j}{\partial N_B}u^2(N_B) = \frac{-1}{t_B}\frac{-1}{t_B}u^2(N_B) = \frac{u^2(N_B)}{t_B^2} = \frac{\sqrt{87}^2}{6000^2} \cong 2\times10^{-6}$$

However, the correlation is negligible here because the uncertainty of the background count, N_B, is much smaller than the uncertainty of each source count, $N_{\mathrm{S},i}$. So, the correlation of the input estimates R_i will be approximated as zero (i.e., treated as if they were uncorrelated), and the correlation terms dropped from Equation G-13. This means the evaluation used to calculate the combined standard uncertainty of ε can proceed using equation G-14:

$$u_c^2(y) = \sum_{i=1}^{N}\left(\frac{\partial f}{\partial x_i}\right)^2 u^2(x_i)\text{, so since } \varepsilon = \frac{\bar{R}}{a_s},$$

$$u_c^2(\varepsilon) = \left(\frac{\partial(\frac{\bar{R}}{a_s})}{\partial \bar{R}}\right)^2 u^2(\bar{R}) + \left(\frac{\partial(\frac{\bar{R}}{a_s})}{\partial a_s}\right)^2 u^2(a_s) = \left(\frac{1}{a_s}\right)^2 u^2(\bar{R}) + \left(\frac{-\bar{R}}{a_s^2}\right)^2 u^2(a_s)$$

$$= \left(\frac{u^2(\bar{R})}{a_s^2}\right) + \varepsilon^2\left(\frac{u^2(a_s)}{a_s^2}\right)\text{. Therefore, } u_c(\varepsilon) = \sqrt{\frac{u^2(\bar{R})}{a_S^2} + \varepsilon^2\,\frac{u^2(a_S)}{a_S^2}}$$

Assume the following data were obtained for the 15 separate counts of the calibration source.

Count Number, *i*	Gross count, $N_{\mathrm{S},i}$	R_i (s^{-1})
1	18,375	61.236
2	18,644	62.132
3	18,954	63.166
4	19,249	64.149
5	19,011	63.356
6	18,936	63.106
7	18,537	61.776
8	18,733	62.429
9	18,812	62.692
10	18,546	61.806

11	18,810	62.686
12	19,273	64.229
13	18,893	62.962
14	18,803	62.662
15	18,280	60.919
	Average, $\overline{R}$ (s^{-1})	62.6202
	Experimental standard deviation, $s(R_i)$ (s^{-1})	0.9483
	Experimental standard deviation of the mean, $s\left(\overline{R}\right)$ (s^{-1})	0.2449

Then the estimated counting efficiency is:

$$\varepsilon = \frac{\overline{R}}{a_s} = \frac{62.6202\ \text{s}^{-1}}{150.0\ \text{Bq}} = 0.4176$$

And the combined standard uncertainty of ε is given by

$$u_c(\varepsilon) = \sqrt{\frac{(0.2449\ \text{s}^{-1})^2}{(150.0\ \text{Bq})^2} + 0.4176^2 \times \frac{(2.0\ \text{Bq})^2}{(150.0\ \text{Bq})^2}} = 0.005802$$

Which may be rounded to 0.0058.

The true counting efficiency may vary because of variations in geometry, position and other influence quantities not explicitly included in the model. These sources of uncertainty may not be controlled as they were in the above example. If this is the case, the standard uncertainty of ε should include not only the standard uncertainty of the estimated mean, as calculated in the example, but also another component of uncertainty due to variations of the true efficiency during subsequent measurements. The additional component may be written as $\varepsilon\phi$, where ϕ is the coefficient of variation (i.e., the standard deviation divided by the mean) of the true efficiency. Then the total uncertainty of ε is obtained by squaring the original uncertainty estimate, adding $\varepsilon^2\phi^2$, and taking the square root of the sum.

$$u_c(\varepsilon) = \sqrt{\frac{u^2(\overline{R})}{a_S^2} + \varepsilon^2\left(\frac{u^2(a_S)}{a_S^2} + \phi^2\right)} \qquad \textbf{(G-18)}$$

In the example above, the experimental variance of the count rates, R_i, may be used to estimate ϕ. Section 18B.2 of Attachment 18B of MARLAP describes an approach for estimating such "excess" variance in a series of measurements.

Variations in counting efficiency due to source placement should be reduced as much as possible through the use of positioning devices that ensure a source with a given geometry is always placed in the same location relative to the detector. If such devices are not used, variations in source position may significantly increase the measurement uncertainty.

Calibrating an instrument under conditions different from the conditions under which M&E sources are counted may lead to large uncertainties in the activity measurements. Source geometry in particular tends to be an important factor for many types of radiation counters. If correction factors are used, their uncertainties should be evaluated and propagated, as mentioned in section G.2.1.1.

G.2.2.3 Digital Displays and Rounding

If a measuring device has a digital display with readability[1] δ, the standard uncertainty of a measured value is at least $\delta / 2\sqrt{3}$, which is the variance of a random variable uniformly distributed over the interval $(x - \delta/2, x + \delta/2)$. Note that this is the same result as given by equation G-10 with $a = \delta/2$. This uncertainty component exists even if the instrument is completely stable.

A similar Type B method may be used to evaluate the standard uncertainty due to computer round-off error. When a value x is rounded to the nearest multiple of 10^n, where n is an integer, the component of uncertainty generated by round-off error is $10^n\ /(2\sqrt{3})$. This component of uncertainty should be kept small in comparison to the total uncertainty of x by performing

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[1] Readability is the smallest difference that can still be read on a display. For instruments with an analog indicating device, the readability is equal to the smallest fraction of a scale interval that can still be estimated with reasonable reliability or which can be determined by an auxiliary device. For instruments with a numeric indicator (digital display), the readability is equal to one digital step.

rounding properly and printing with an adequate number of figures. In a long calculation involving mixed operations, carry as many digits as possible through the entire set of calculations and then round the final result appropriately as described in MARLAP Section 19.3.7 (MARLAP 2004).

Example 3: The readability of a digital survey doserate meter is 1 nGy/h. Therefore, the minimum standard uncertainty of a measured absorbed dose rate is $1/2\sqrt{3}$ = 0.29 nGy/h.

Example 4: Suppose the results for R_i in Example 2 had been rounded to the nearest whole number before the analysis. Then the average would be computed as 62.6 instead of 62.6202 and the standard deviation would be computed as 0.9103 instead of 0.9483. This demonstrates the effect that rounding intermediate results can have on subsequent calculations. If this rounding to the nearest positive integer had already occurred prior to receiving the data, and the original data were no longer available, a correction for it could be made when estimating the combined standard uncertainty of R_i. The component of uncertainty generated by round-off error is $1/(2\sqrt{3})$:

$$u(R_i) = \sqrt{0.9103^2 + \left(\frac{1}{2\sqrt{3}}\right)^2} = 0.9549.$$

G.2.3 Example Uncertainty Calculation

To illustrate how the uncertainty calculations are performed in practice, the following example is given based on that of Lewis et al. (Lewis 2005). The calculation will be that of the combined standard uncertainty in the calibration of a surface contamination monitor.

G.2.3.1 Model Equation and Sensitivity Coefficients

Surface contamination monitors are calibrated in terms of their response to known rates of radioactive emissions. In practice this is achieved by using large-area, planar sources that have a defined area and whose emission rates have been determined in a traceable manner. The calibration is usually determined in terms of response per emission rate per unit area. In this example, the source is positioned with its active face parallel to and at a distance of 3 mm from the face of the detector. The monitor detector area (50 cm^2) is smaller than the area of the calibration source, which is a 10 cm × 10 cm layer of ^{14}C on a thick aluminum substrate. The monitor has an analog display and has a means to set the detector voltage.

The efficiency, ε, is defined by:

$$\varepsilon = \frac{(M - B) \times f_V \times f_d \times f_u \times f_{bs}}{\left(E \middle/ A \right)} \tag{G-19}$$

Where:

M	observed monitor reading, s^{-1}
B	background reading, s^{-1}
E	emission rate of the calibration source, s^{-1}
A	area of the active portion of the calibration source, cm^2
f_V	plateau voltage factor,
f_d	source-detector separation factor,
f_u	source uniformity factor,
f_{bs}	backscatter factor.

The sensitivity coefficients of Equation G-19 are given by:

$$\frac{\partial \varepsilon}{\partial M} = (A/E) \times f_V \times f_d \times f_u \times f_{bs} = \frac{\varepsilon}{(M - B)} \tag{G-20}$$

$$\frac{\partial \varepsilon}{\partial B} = -(A/E) \times f_V \times f_d \times f_u \times f_{bs} = \frac{-\varepsilon}{(M - B)} \tag{G-21}$$

$$\frac{\partial \varepsilon}{\partial E} = -(M - B)(A/E^2) \times f_V \times f_d \times f_u \times f_{bs} = \frac{-\varepsilon}{E} \tag{G-22}$$

$$\frac{\partial \varepsilon}{\partial A} = (M - B)(1/E) \times f_V \times f_d \times f_u \times f_{bs} = \frac{\varepsilon}{A} \tag{G-23}$$

$$\frac{\partial \varepsilon}{\partial f_V} = (M - B)(A/E) \times f_d \times f_u \times f_{bs} = \frac{\varepsilon}{f_V} \qquad \textbf{(G-24)}$$

$$\frac{\partial \varepsilon}{\partial f_d} = (M - B)(A/E) \times f_V \times f_u \times f_{bs} = \frac{\varepsilon}{f_d} \qquad \textbf{(G-25)}$$

$$\frac{\partial \varepsilon}{\partial f_u} = (M - B)(A/E) \times f_V \times f_d \times f_{bs} = \frac{\varepsilon}{f_u} \qquad \textbf{(G-26)}$$

$$\frac{\partial \varepsilon}{\partial f_{bs}} = (M - B)(A/E) \times f_V \times f_d \times f_u = \frac{\varepsilon}{f_{bs}} \qquad \textbf{(G-27)}$$

Under normal conditions, the factors f_V, f_d, f_u and f_{bs} are each assumed to have a value of one. If the uncertainties are to be calculated in relative terms, the uncertainty equation becomes (see Equation G-16):

$$\left(\frac{\sigma_C}{\varepsilon}\right)^2 = \left(\frac{M}{M-B}\right)^2\left(\frac{\sigma_M}{M}\right)^2 + \left(\frac{B}{M-B}\right)^2\left(\frac{\sigma_B}{B}\right)^2 + \left(\frac{\sigma_E}{E}\right)^2 + \left(\frac{\sigma_A}{A}\right)^2 + \left(\frac{\sigma_{f_V}}{f_V}\right)^2 + \left(\frac{\sigma_{f_d}}{f_d}\right)^2 + \left(\frac{\sigma_{f_u}}{f_u}\right)^2 + \left(\frac{\sigma_{f_{bs}}}{f_{bs}}\right)^2 \qquad \textbf{(G-28)}$$

If the relative uncertainties are all expressed as percentages, $\left(\frac{\sigma_{x_i}}{x_i}\right)$, where x_i is an input quantity, then the combined standard uncertainty will be a percentage. The relative sensitivity coefficients, c_i, are the terms multiplying each relative uncertainty term $\left(\frac{\sigma_{x_i}}{x_i}\right)$ in Equation G-28. This approach produces relative sensitivity coefficients of unity for the last 6 terms.

G.2.3.2 Uncertainty Components

Monitor reading of source, *M* (Type A)

Several techniques can be used to determine the mean observed monitor reading, *M*, and its uncertainty. Assume a snap-shot technique is used whereby six successive, but randomly timed, readings are recorded, giving 350, 400, 400, 325, 350, 350 s^{-1}. The mean and standard deviation of the mean becomes 362.5 ± 12.5 s^{-1}. This equates to a percentage uncertainty in *M* of 3.45% and the relative sensitivity coefficient from Equation G-28, $\frac{M}{(M-B)}$, is 362.5/(362.5 – 32.5), which is equal to 1.10. The distribution is assumed to be normal.

Monitor reading of background, *B* (Type B)

In this case, an eye-averaging technique was used whereby the highest and lowest count rates were recorded over a given period of time. These count rates were 40 and 25 s^{-1} respectively, giving a mean value of 32.5 s^{-1}. This value is assumed to have a rectangular distribution with a half-width of 7.5 s^{-1}, and an uncertainty of $7.5/\sqrt{3} = 4.330$, equating to a percentage uncertainty of $4.330/32.5 = 0.1332$ or 13%. The relative sensitivity coefficient from Equation G-28, $\frac{B}{(M-B)}$, is 32.5/(362.5 - 32.5), which gives a value of 0.098.

Emission rate of calibration source, *E* (Type B)

The emission rate of the source and its uncertainty were provided on the calibration certificate by the laboratory that calibrated the source using a windowless proportional counter. The statement on the certificate was:

"The measured value of the emission rate is $E = 2{,}732 \pm 13\ \mathrm{s}^{-1}$

The reported uncertainty is based on a standard uncertainty multiplied by a coverage factor of *k* = 2, which provides a level of confidence of approximately 95%. The standard uncertainty on *E* is therefore 13/2 = 6.5 s^{-1} or 0.24%. Unless the certificate provides information to the contrary, it is assumed that the uncertainty has a normal distribution.

Source area, *A* (Type B)

In the absence of an uncertainty statement by the manufacturer, the only information available is the product drawing that shows the active area dimensions to be 10 cm × 10 cm. On the assumption that the outer bounds of the length, *L*, and the width, *W*, are 9.9 and 10.1 cm, the uncertainty of the linear dimensions may be taken to be a rectangular distribution with a half-width of 0.1 cm.

L = 10 and $u(L) = 0.1/\sqrt{3} = 0.0577$. W = 10 and $u(W) = 0.1/\sqrt{3} = 0.0577$. Since $A = LW$, we get $u^2(A) = u^2(LW) = L^2u^2(W) + W^2u^2(L) = 2(10)^2(0.0577)^2 = 0.665858$, therefore $u(A) = 0.816\ \mathrm{cm}^2$ or 0.816%.

Plateau voltage factor, f_V (Type B)

This applies only to those instruments where voltage adjustments are possible. If the setting is not checked and/or adjusted between calibrations, then this has no effect. Changing the plateau voltage without performing a recalibration is not recommended. If, however, the user is allowed to do this, the setting may not be returned to exactly that used during the calibration. In this particular example, the slope of the response curve in this region is taken to be 10% / 50 v. It is assumed that an operator is more likely to set the voltage nearer to the optimum than the extremes and that ± 50 v represents the range at the 100% confidence level. Accordingly, a triangular distribution is assumed with a half-width of 50 v, equating to an uncertainty for the voltage of $50/\sqrt{6} = 20.4124$ and an uncertainty for the voltage factor of 20.4124(10%)/ 50 = 4.0825%.

Source-detector separation factor, f_d (Type B)

This effect arises from the uncertainty in mounting the calibration source exactly 3 mm from the detector face. Experimental evidence has shown that, for the particular ^{14}C source at 3 mm source-detector separation, the change in response was 2.6% / mm. It is assumed that the deviation from the nominal 3 mm separation is no greater than 1 mm but that all values are equally probable between 2 and 4 mm, a rectangular distribution. The uncertainty in the separation is thus $1/\sqrt{3} = 0.5774$. The uncertainty of the separation factor is thus 0.5774 mm × 2.6% / mm, equal to 1.5011%.

Non-uniformity of calibration source, f_u (Type B)

Large area sources may have a non-uniform activity distribution across their surfaces. For the ^{14}C source, the uniformity is assumed to be better than ± 10%. This is based on comparing 10 cm^2 sections of the source. For a typical monitor with a detector area of 50 cm^2 and a calibration source area of 100 cm^2, a worst-case condition could be that the area under the detector has an activity per unit area that is 10% greater than the mean value for the whole source. (The outer area correspondingly will be 10% less than mean value.) Assuming a rectangular distribution, this represents an uncertainty of $10/\sqrt{3} = 5.774\%$ for the source non-uniformity factor.

Backscatter factor, f_{bs} (Type B)

Variations in backscatter effects arise from factors such as the nature of the surface on which the calibration source is resting and the proximity to scattering surfaces such as walls. This effect can be quite marked for photon emitters, but for ^{14}C on aluminum substrates the effect is negligible.

G.2.3.3 Uncertainty Budget

An important part of the uncertainty analysis is to determine which factors are contributing the most to the overall uncertainty.

Table G.3: Uncertainty Budget for the Efficiency Example

Source of uncertainty	Type	Probability distribution	*Relative Sensitivity Coeffient,*c_i	$u_i(x_i)$ (%)	$u_i(y)= c_i\ u_i(x_i)$ (%)	$(u_i(y))^2$	$(u_i(y))^2/Total$
Standard deviation of mean of *M*	A	Normal	1.10	3.45	3.80	14.44	0.21
Standard deviation of mean of *B*	B	Rectangular	0.098	13.32	1.31	1.72	0.02
Standard uncertainty of calibration source emission rate, *E*	B	Normal	1.0	0.24	0.24	0.06	0.00
Half -width of source length, L and width W on the area A	B	Product of 2 independent rectangular	1.0	0.816	0.816	0.666	0.01
Half -width of voltage factor, f_V	B	Triangular	1.0	4.08	4.08	16.65	0.24
Half -width of source-detector separation factor, f_d	B	Rectangular	1.0	1.50	1.50	2.25	0.03
Half-width of calibration source non-uniformity factor, f_u	B	Rectangular	1.0	5.77	5.77	33.29	0.48
Uncertainty of backscatter factor, f_{bs}	B	n.a.	1.0	0.0	0.0	0.00	0.00
Combined standard uncertainty		Normal	- - -	- - -	8.31 = $\sqrt{69.07}$	Total= 69.07	0.99
Expanded uncertainty (*k*=2)		Normal	- - -	- - -	2·8.31= 16.6	- - -	

The relative sensitivity coefficients, $c_{i,}$ are the terms multiplying each relative uncertainty term $\left(\frac{\sigma_{x_i}}{x_i}\right)$ in Equation G-28. To do this, each component of uncertainty $u_i(y)=c_i\ u_i(x_i)$ is squared to give its component of variance $(u_i(y))^2$. These are totaled to get the total variance, in this case 69.07. Finally, the ratio of each component of variance to the total is computed.

Examining the last column of the uncertainty budget table (Table G.3) shows that the major source of uncertainty is due to source non-uniformity (48%) followed by the voltage factor (24%) and the reading of the source (21%). Thus, to decrease the overall uncertainty, attention should be paid to those factors first.

G.2.3.4 Reported Result

Using the formula above, the calibration factor in terms of emission rate becomes:

$$\varepsilon = \frac{(M-B)\times f_V \times f_d \times f_u \times f_{bs}}{\left(E/A\right)} = \frac{(362.5-32.5)\times 1\times 1\times 1\times 1}{\left(2732/100\right)} = 12.1\,(\text{counts}\times\text{s}^{-1})/(\text{s}^{-1}\times\text{cm}^{-2})$$

The combined standard uncertainty is (12.1)(.0831) = 1.0056. The reported expanded uncertainty will be 2.0, based on a standard uncertainty of 1.0 multiplied by a coverage factor of k = 2, which provides a level of confidence of approximately 95%.

G.3 Calculation of the Minimum Detectable Concentration

Table G.4 Notation for Section G.3

Symbol	***Definition***	***Formula or reference***	***Type***
ε	efficiency		
F	calibration function	$X = F(Y)$	
F^{-1}	evaluation function	$Y = F^{-1}(X)$, closely related to the *mathematical model* $Y = f(X_1, X_2, \ldots, X_N)$	
S_C.	Critical net signal	Net signal is calculated from the gross signal by subtracting the estimated blank value and any interferences	
S_D	Mean value of the net signal that gives a specified probability, 1−β, of yielding an observed signal greater than its critical value S_C.		
X	observable response variable,measureable signal		
x_C.	The critical value of the response variable	Calculation of y_C requires the choice of a significance level for the test. The significance level is a specified upper bound for the probability, α, of a Type I error. The significance level is usually chosen to be 0.05.	If a measured value exceeds the critical value, a decision is made that radiation or radioactivity has been detected
Y	state variable, measurand		
y_C	Critical value of the concentration	$y_C = F^{-1}(x_C)$.	
$y_D = \frac{S_D}{\varepsilon}$	Minimum detectable concentration (MDC)	$y_D = \frac{S_D}{\varepsilon}$	

G.3.1 Critical Value

In the terminology of ISO 11843-1 (1997), the measured concentration is the state variable, denoted by Y, which represents the state of the material being analyzed. The state variable usually cannot be observed directly, but it is related to an observable response variable, denoted by X, through a calibration function F, the mathematical relationship being written as $X = F(Y)$. The response variable X is most often an instrument signal, such as the number of counts observed. The inverse, $Y = F^{-1}(X)$ of the calibration function is sometimes called the evaluation function. The evaluation function, which gives the value of the net concentration in terms of the response variable, is closely related to the mathematical model $Y = f(X_1, X_2, \ldots, X_N)$ described in Section G.2.1.1.

In a Scenario B detection decision, either the null or alternative hypothesis is chosen on the basis of the observed value of the response variable, X. The value of X must exceed a certain threshold value to justify rejection of the null hypothesis and acceptance of the alternative hypothesis. This threshold is called the critical value of the response variable and is denoted by x_C.

The calculation of x_C requires the choice of a significance level for the test. The significance level is a specified upper bound for the probability, α, of a Type I error. The significance level is usually chosen to be 0.05. This means that when there is no radiation or radioactivity present (above background), there should be at most a 5% probability of incorrectly deciding that it is present.

The critical value of the concentration, y_C, is defined as the value obtained by applying the evaluation function, F^{-1}, to the critical value of the response variable, x_C. Thus, $y_C = F^{-1}(x_C)$. When x is the gross instrument signal, this formula typically involves subtraction of the background signal and division by the counting efficiency, and possibly other factors.

A detection decision can be made by comparing the observed gross instrument signal to its critical value, x_C, as indicated above. However, it has become standard practice to make the decision by comparing the net instrument signal to its critical value, S_C. The net signal is calculated from the gross signal by subtracting the estimated blank value and any interferences.[2] The critical net signal, S_C, is calculated from the critical gross signal, x_C, by subtracting the same correction terms; so, in principle, either approach should lead to the same detection decision.

Since the term “critical value” alone is ambiguous, one should specify the variable to which the term refers. For example, one may discuss the critical (value of the) radionuclide concentration, the critical (value of the) net signal, or the critical (value of the) gross signal. In this document, the signal is usually a count, and the critical value generally refers to the net count.

The response variable is typically an instrument signal, whose mean value generally is positive even when there is radioactivity present (i.e., above background). The gross signal must be

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[2] Interference is the presence of other radiation or radioactivity that hinder the ability to analyze for the radiation or radioactivity of interest.

corrected by subtracting an estimate of the signal produced under those conditions. See Section G.2.2.1 (Instrument Background).

G.3.2 Minimum Detectable Concentration

The minimum detectable concentration (MDC) is the minimum concentration of radiation or radioactivity that must be present in a sample to give a specified power, $1 - \beta$. It may also be defined as:

- The minimum radiation or radioactivity concentration that must be present to give a specified probability, $1 - \beta$, of detecting the radiation or radioactivity; or
- The minimum radiation or radioactivity concentration that must be present to give a specified probability, $1 - \beta$, of measuring a response greater than the critical value, leading one to conclude correctly that there is radiation or radioactivity present.

The *power* of any hypothesis test is defined as the probability that the test will reject the null hypothesis when it is false, i.e., the correct decision. Therefore, if the probability of a Type II error is denoted by β, the power is $1 - \beta$. In the context of radiation or radioactivity detection, the power of the test is the probability of correctly detecting the radiation or radioactivity (concluding that the radiation or radioactivity is present), which happens whenever the response variable exceeds its critical value. The power depends on the concentration of the radiation or radioactivity and other conditions of measurement; so, one often speaks of the "power function" or "power curve." Note that the power of a test for radiation or radioactivity detection generally is an increasing function of the radiation or radioactivity concentration − i.e., the greater the radiation or radioactivity concentration the higher the probability of detecting it.

In the context of MDC calculations, the value of β that appears in the definition, like α, is usually chosen to be 0.05 or is assumed to be 0.05 by default if no value is specified. The minimum detectable concentration is denoted in mathematical expressions by y_D. The MDC is usually obtained from the minimum detectable value of the net instrument signal, S_D. S_D, is defined as the mean value of the net signal that gives a specified probability, $1 - \beta$, of yielding an observed signal greater than its critical value S_C. The relationship between the critical net signal, S_C, and the minimum detectable net signal, S_D, is shown in Figure 5.2 in Section 5.7.2.

The term MDC must be carefully and precisely defined to prevent confusion. The MDC is by definition an estimate of the true concentration of the radiation or radioactivity required to give a specified high probability that the measured response will be greater than the critical value.

The common practice of comparing a measured concentration to the MDC, instead of to the S_C, to make a detection decision is incorrect. If this procedure were used, then there would be only a a 50% chance of deciding that radioactivity was present when the concentration was actually at the MDC. This is in direct contradiction to the definition of MDC. See MARLAP Appendix B, Attachment B1 for a further discussion of this issue.

Since the MDC is calculated from measured values of input quantities such as the counting efficiency and background level, the MDC estimate has a combined standard uncertainty, which in principle can be obtained by uncertainty propagation. To avoid confusion, it may be useful to remember that a detection decision is usually made by comparing the instrument response to the critical value, and that the critical value generally does not even have the units of radiation or radioactivity concentration.

G.3.3 Calculation of the Critical Value

If the net signal is a count, then in many circumstances the uncertainty in the count can be estimated by a Type B evaluation using the fact that for a Poisson distribution with mean N_B, the variance is also N_B. Thus the uncertainty in the background count is estimated as $\sqrt{N_B}$.

Hence, the critical value is often an expression involving $\sqrt{N_B}$.

The most commonly used approach for calculating the critical net signal, $S_{C,}$ is given by the following equation.[3]

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[3] This expression for the critical net count depends for its validity on the assumption of Poisson counting statistics. If the variance of the blank signal is affected by interferences, or background instability, then Equation 20.7 of MARLAP may be more appropriate.

$$S_C = z_{1-\alpha}\sqrt{N_B \frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)} \qquad \textbf{(G-29)}$$

Where:

N_B is the background count,

t_S is the count time for the sample,

t_B is the count time for the background, and

$z_{1-\alpha}$ is the $(1-\alpha)$-quantile of the standard normal distribution.

Example 5: A 6,000-second background measurement is performed on a proportional counter and 108 beta counts are observed. A sample is to be counted for 3,000 s. Estimate the critical value of the net count when $\alpha = 0.05$.

$$S_C = z_{1-\alpha}\sqrt{N_B \frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)}$$

$$S_C = 1.645\sqrt{108\times\left(\frac{3{,}000\text{ s}}{6{,}000\text{ s}}\right)\left(1+\frac{3{,}000\text{ s}}{6{,}000\text{ s}}\right)} = 14.8\text{ net counts}$$

If $\alpha = 0.05$ and $t_B = t_S$, equation G-29 leads to the well-known expression $2.33\sqrt{N_B}$ for the critical net count (Currie, 1968).

When the background count is high (e.g., 100 or more) Equation G-29 works well, but at lower background levels it can produce a high rate of Type I errors. Since this is a Scenario B hypothesis test, this means that too often a decision will be made that there is radiation or radioactivity present when it actually is not.

When the mean background counts are low and $t_B \neq t_S$, another approximation formula for S_C appears to out-perform all of the other approximations reviewed in MARLAP, namely the Stapleton Approximation:

$$S_C = d \times \left(\frac{t_S}{t_B} - 1 \right) + \frac{z_{1-\alpha}^2}{4} \times \left(1 + \frac{t_S}{t_B} \right) + z_{1-\alpha} \sqrt{\left(N_B + d \right) \frac{t_S}{t_B} \left(1 + \frac{t_S}{t_B} \right)} \qquad \textbf{(G-30)}$$

When $\alpha = 0.05$, setting the parameter $d = 0.4$ yields the best results. When, in addition, $t_B = t_S$, the Stapleton approximation gives the equation

$$S_C = 1.35 + 2.33 \sqrt{N_B + 0.4} \qquad \textbf{(G-31)}$$

G.3.4 Calculation of the Minimum Detectable Value of the Net Instrument Signal

The traditional method for calculating the MDC involves two steps: first calculating the minimum detectable value of the net instrument signal and then converting the result to a concentration using the mathematical measurement model.

The minimum detectable value of the net instrument signal, denoted by S_D, is defined as the mean value of the net signal that gives a specified probability, $1 - \beta$, of yielding an observed signal greater than its critical value S_C.

The MDC may be estimated by calculating the minimum detectable value of the net instrument signal, S_D, and converting the result to a concentration.

Counting data rarely, if ever, follow the Poisson model exactly, but the model can be used to calculate S_D if the variance of the background signal is approximately Poisson and a conservative value of the efficiency constant, ε, is used to convert S_D to y_D. The equation below shows how to calculate S_D using the Poisson model.

$$S_D = S_C + \frac{z_{1-\beta}^2}{2} + z_{1-\beta} \sqrt{\frac{z_{1-\beta}^2}{4} + S_C + R_B t_S \left(1 + \frac{t_S}{t_B} \right)} \qquad \textbf{(G-33)}$$

Where:

S_C is the critical value,

R_B is the mean count rate of the blank, $R_B = \frac{N_B}{t_B}$,

N_B is the background count,

t_S is the count time for the test source,

t_B is the count time for the background, and

$z_{1-\beta}$ is the $(1-\beta)$-quantile of the standard normal distribution.

When Equation G-29 is appropriate for the critical net count, and $\alpha = \beta$, this expression for S_D simplifies to $z_{1-\beta}^2 + 2S_C$. If in addition, $\alpha = \beta = 0.05$ and $t_B = t_S$ then

$$S_D = 2.71 + 2S_C = 2.71 + 2(2.33\sqrt{N_B}) = 2.71 + 4.66\sqrt{N_B}$$

Example 6 A 6,000-second background measurement on a proportional counter produces 108 beta counts and a source is to be counted for 3,000 s. Assume the background measurement gives the available estimate of the true mean background count rate, R_B and use the value 0.05 for Type I and Type II error probabilities. From Section,G.3.3 Example 5, the critical net count, S_C, equals 14.8, so $S_D = z_{1-\beta}^2 + 2S_C = 1.645^2 + 2\,(14.8) = 32.3$ net counts.

When the Stapleton approximation (Equation G-30) is used for S_C, the minimum detectable net count S_D may be calculated using the equation G-33, but when the Poisson model is assumed, a better estimate is given by the equation:

$$S_D = \frac{(z_{1-\alpha} + z_{1-\beta})^2}{4}\left(1 + \frac{t_S}{t_B}\right) + (z_{1-\alpha} + z_{1-\beta})\sqrt{R_B t_S\left(1 + \frac{t_S}{t_B}\right)}$$

G-34)

This equation is the same as that recommended by ISO 11929-1 (ISO 2000) in a slightly different form.

When $\alpha = \beta = 0.05$ and $t_\mathrm{B} = t_\mathrm{S}$, the preceding equation becomes:

$$S_D = 5.41 + 4.65\sqrt{R_B t_S}$$

(G-35)

Consult MARLAP Chapter 20 for a discussion of the calculation of S_D and y_D when both Poisson counting statistics and other sources of variance are considered.

G.3.5 Calculation of the Minimum Detectable Concentration

The MDC is often used to compare different measurement procedures against specified requirements. The calculation of the nominal MDC is complicated by the fact that some input quantities in the mathematical model, such as interferences, counting efficiency, and instrument background may vary significantly from measurement to measurement. Because of these variable quantities, determining the value of the radiation or radioactivity concentration that corresponds to the minimum detectable value of the net instrument signal, S_D, may be difficult in practice. One common approach to this problem is to make conservative choices for the values of the variable quantities, which tend to increase the value of the MDC.

The mean net signal, S, is usually directly proportional to Y, the true radiation or radioactivity concentration present. Hence, there is a efficiency constant, ε, such that $S = \varepsilon Y$. The constant ε is typically the mean value of the product of factors such as the source count time, decay-correction factor, and counting efficiency. Therefore, the value of the minimum detectable concentration, y_D, is

$$y_D = \frac{S_D}{\varepsilon} \qquad \textbf{(G-36)}$$

The preceding equation is only true if all sources of variability are accounted for when determining the distribution of the net signal, $\hat{S}$. Note that ensuring the MDC is not underestimated also requires that the value of ε not be overestimated.

Using any of the equations in Section G.3.4 to calculate S_D is only appropriate if a conservative value of the efficiency constant, ε, is used when converting S_D to the MDC.

Example 7: Consider a scenario where t_B = 6,000 s, t_S = 3,000 s, and $R_\mathrm{B} \approx 0.018\ \mathrm{s}^{-1}$. Let the measurement model be $Y = \dfrac{N_S - (N_B t_S / t_B)}{t_S \varepsilon}$

Where:

Y is the activity of the radionuclide in the sample and

ε is the counting efficiency (counts per second)/(Bq/cm^2)

Assume the source count time, t_S, has negligible variability, the counting efficiency has mean 0.42 and a 10% relative combined standard uncertainty, and from Example 6, S_D = 32.3 net counts.

The mean minimum detectable concentration is $y_D = \frac{S_D}{t_S \varepsilon} = \frac{32.3}{(3000)(0.42)} = 0.0256 \text{ Bq/cm}^2$.

Adjusting for the 10% variability in the counting efficiency, the uncertainty is (0.10) × (0.42) = 0.042. Assuming that the efficiency is normally distributed, the lower 5th percentile for ε is (0.42) - (1.645)(0.042) = 0.35, where -1.645 is the 5th percentile of a standard normal distribution.. Therefore a conservative estimate of the efficiency constant is $\varepsilon = 0.35$ and a conservative estimate of the minimum detectable concentration is:

$$y_D = \frac{S_D}{t_S \varepsilon} = \frac{32.3}{(3000)(0.35)} = 0.0308 \text{ Bq/cm}^2.$$

An alternative procedure could be to recognize that because of the uncertainties in the input estimates entered into the measurement model to convert from S_D to Y, that the MDC is actually a random variable. Then the methods for propagation of uncertainty given in Section G.2 can be applied. Using the same assumptions as above we would find that $y_D = 0.0256 \pm 0.0051$ with 95% confidence based on a coverage factor of 2. Therefore the 95% upper confidence level for y_D would be 0.0307 Bq.

More conservative (higher) estimates of the MDC may be obtained by following NRC recommendations (NRC 1984), in which formulas for the MDC include estimated bounds for relative systematic error in the background determination (Δ_B) and the sensitivity (Δ_A). The critical net count S_C is increased by $\Delta_B N_B \frac{t_S}{t_B}$, and the minimum detectable net count S_D is increased by $2\Delta_B\ N_B \frac{t_S}{t_B}$. Next, the MDC is calculated by dividing S_D by the efficiency and multiplying the result by $1+\Delta_A$. The conservative approach presented in NRC 1984 treats random errors and systematic errors differently to ensure that the MDC for a measurement process is unlikely to be consistently underestimated, which is an important consideration if it is required by regulation or contract to achieve a specified MDC.

G.4 Calculation of the Minimum Quantifiable Concentration

Table G.5 Notation for Section G.4

Symbol	***Definition***	***Formula or reference***	***Type***
k_Q	Multiple of the standard deviation defining y_Q, usually chosen to be 10.	$k_Q = \frac{\sqrt{\sigma^2(y \mid Y = y_Q)}}{y_Q}$	Chosen during DQO process
$\sigma^2(y \mid Y = y_Q)$	The variance of the estimator y given the true concentration *Y* equals y_Q.		Theoretical
y_Q	Minimum quantifiable concentration (MQC)	The concentration at which the measurement process gives results with a specified relative standard deviation $1/k_Q$, where k_Q is usually chosen to be 10.	Theoretical

Calculation of the MQC requires that one be able to estimate the standard deviation for the result of a hypothetical measurement performed on a sample with a specified radionuclide concentration. The MQC is defined symbolically as the value y_Q that satisfies the relation:

$$y_Q = k_Q \sqrt{\sigma^2(y \mid Y = y_Q)} \qquad \textbf{(G-37)}$$

Where the specified relative standard deviation of y_Q is $1/k_Q$ (usually chosen to be 10% so that $k_Q = 10$). $\sigma^2(y \mid Y = y_Q)$ is the variance of the estimator y given the true concentration *Y* equals y_Q. If the function $\sigma^2(y \mid Y = y_Q)$ has a simple form, it may be possible to solve the above equation for y_Q using only algebraic manipulation. Otherwise, fixed-point iteration, or other more general approaches, may be used, as discussed in MARLAP Section 20.4.3.

When Poisson counting statistics are assumed, and the mathematical model for the radionuclide concentration is $Y = S/\varepsilon$, where *S* is the net count, S/t_S is the net count rate and ε is the efficiency of the measurement, the above equation may be solved for y_Q to obtain:

$$y_Q = \frac{k_Q^2}{2t_S\varepsilon(1-k_Q^2\phi_{\hat{\varepsilon}}^2)}\left(1+\sqrt{1+\frac{4(1-k_Q^2\phi_{\hat{\varepsilon}}^2)}{k_Q^2}\left(R_B t_S\left(1+\frac{t_S}{t_B}\right)+R_I t_S+\sigma^2(\widehat{R}_I)t_S^2\right)}\right) \qquad \textbf{(G-38)}$$

Where:

t_S is the count time for the source, s,

t_B is the count time for the background, s,

R_B is the mean background count rate, s^{-1},

R_I is the mean interference count rate, s^{-1},

$\sigma(\hat{R}_I)$ is the standard deviation of the measured interference count rate, s^{-1}, and

$\phi_{\hat{\varepsilon}}^2$ is the relative variance of the measured efficiency, $\hat{\varepsilon}$.

If the efficiency ε may vary, then a conservative value, such as the 0.05-quantile $\varepsilon_{0.05}$, should be substituted for ε in the formula. Note that $\phi_{\hat{\varepsilon}}^2$ denotes only the relative variance of $\hat{\varepsilon}$ due to subsampling and measurement error – it does not include any variance of the efficiency ε itself (see discussion in Section G.2).

Note that equation G-38 defines the MQC only if $1 - k_Q^2 \phi_{\hat{\varepsilon}}^2 > 0$. If $1 - k_Q^2 \phi_{\hat{\varepsilon}}^2 \leq 0$, the MQC is infinite, because there is no concentration at which the relative standard deviation of y fails to exceed $1 / k_Q$. In particular, if the relative standard deviation of the measured efficiency $\hat{\varepsilon}$ exceeds $1 / k_Q$, then $1 - k_Q^2 \phi_{\hat{\varepsilon}}^2 < 0$ and the MQC is infinite.

If there are no interferences, equation G-37 simplifies to:

$$y_Q = \frac{k_Q^2}{2 t_S \varepsilon (1 - k_Q^2 \phi_{\hat{\varepsilon}}^2)} \left(1 + \sqrt{1 + \frac{4(1 - k_Q^2 \phi_{\hat{\varepsilon}}^2)}{k_Q^2} \left(R_B t_S \left(1 + \frac{t_S}{t_B} \right) \right)} \right) \qquad \textbf{(G-39)}$$

Example 8: Consider the scenario of Example 5, where t_B = 6,000 s, t_S = 3,000 s, and $R_B \approx 0.018\ s^{-1}$. Suppose the measurement model is $Y = \dfrac{N_S - (N_B t_S / t_B)}{t_S \varepsilon}$

Where:

Y is the specific activity of the radionuclide in the sample and

ε the counting efficiency $\text{(counts per second)/(Bq/cm}^2\text{)}$.

Assume:

The source count time, t_S, has negligible variability,

the counting efficiency has mean 0.42 and a 5% relative combined standard uncertainty, and

S_D = 32.3 net counts. S_D / t_S =32.3/3000 is the net count rate.

The counting efficiency ε = 0.42

The mean minimum detectable concentration is $y_D = \dfrac{S_D}{t_S\varepsilon} = \dfrac{32.3}{(3000)(0.42)} = 0.0256\ \text{Bq/cm}^2$.

Also assume:

$k_Q = 10$

$\phi_{\hat{\varepsilon}} = 0.05$

$\phi_{\hat{\varepsilon}}^2 = 0.05^2$

$1 - k_Q^2\phi_{\hat{\varepsilon}}^2 = 1 - 100 \times (0.05^2) = 0.75$, and

there are no interferences so that equation G-38 can be used.

Note that if the counting efficiency has mean 0.42 and a 10% relative standard uncertainty as in Example 11, then $1 - k_Q^2\phi_{\hat{\varepsilon}}^2 = 1 - 100 \times (0.10^2) = 0$ and the MQC would be infinite. Therefore it was necessary to change the procedure for evaluating the efficiency in this example so that the relative combined standard uncertainty could be reduced. In this example it is assumed to be 5%.

The MQC can be calculated as:

$$y_Q = \frac{k_Q^2}{2t_S\varepsilon(1 - k_Q^2\phi_{\hat{\varepsilon}}^2)}\left(1 + \sqrt{1 + \frac{4(1 - k_Q^2\phi_{\hat{\varepsilon}}^2)}{k_Q^2}\left(R_B t_S\left(1 + \frac{t_S}{t_B}\right) + 0\right)}\right)$$

$$y_Q = \frac{100}{2\,(3000)(0.42)(0.75)}\left(1 + \sqrt{1 + \frac{4\,(0.75)}{100}\left((0.018\ \text{s}^{-1})(3000\ \text{s})\left(1 + \frac{(3000\ \text{s})}{(6000\ \text{s})}\right) + 0\right)}\right)$$

$= 0.151 \text{ Bq/cm}^2$

As a check, y_Q can be calculated in a different way. If y_Q is the MQC and $k_Q = 10$, then the relative combined standard uncertainty of a measurement of concentration y_Q is 10%. The procedure described in Section 5.6 can be used to predict the combined standard uncertainty of a measurement made on a hypothetical sample whose concentration is exactly $y_Q = 0.151 \text{ Bq/cm}^2$.

The measurement model is $Y = \dfrac{N_S - (N_B t_S / t_B)}{t_S \varepsilon}$.

Recall from Section G.2.1.6 that if $y = \dfrac{f(x_1, x_2, \ldots, x_n)}{z_1 z_2 \ldots z_m}$, where *f* is some specified function of $x_1, x_2, \ldots, x_n$, all the z_i are nonzero, and all the input estimates are uncorrelated that the combined standard uncertainty may be calculated using Equation G-16:

$$u_c^2(y) = \frac{u_c^2(f(x_1, x_2, \ldots, x_n))}{z_1 z_2 \ldots z_m} + y^2 \left(\frac{u^2(z_1)}{z_1^2} + \frac{u^2(z_2)}{z_2^2} + \ldots + \frac{u^2(z_m)}{z_m^2} \right)$$

Substituting

$y = Y$

$f(x_1, x_2, \ldots, x_n) = f(N_S, N_B, t_S, t_B) = N_S - (N_B t_S / t_B) / t_S$

$z_1 = \varepsilon$, and

$$u_c^2(N_S - (N_B t_S / t_B)/t_S) = u_c^2(N_S / t_S) + u_c^2((N_B t_S / t_B)/t_S) = \frac{u_c^2(N_S) + (t_S / t_B)^2 u_c^2(N_B)}{t_S^2} =$$

$$\frac{\sqrt{N_S}^2 + \sqrt{N_B}^2 (t_S^2 / t_B^2)}{t_S^2} = \frac{N_S + N_B (t_S^2 / t_B^2)}{t_S^2}$$

Results in:

$$u_c^2(Y) = \frac{N_S + (N_B t_S^2 / t_B^2)}{t_S^2 \varepsilon^2} + Y^2 \left(\frac{u^2(\varepsilon)}{\varepsilon^2} \right) \text{ or}$$

$$u_c(Y) = \sqrt{\frac{N_S + (N_B t_S^2 / t_B^2)}{t_S^2 \varepsilon^2} + Y^2 \left(\frac{u^2(\varepsilon)}{\varepsilon^2} \right)}$$

Inserting the values

$Y = y_Q = 0.151$ Bq/cm^2

$t_B = 6{,}000$ s

$t_S = 3{,}000$ s

$\varepsilon = 0.42$ (counts per second)/(Bq/cm^2)

$N_B = R_B t_B = (0.018\,\text{s}^{-1})(3{,}000\,\text{s}) = 108$ and

$N_S = x_Q t_S \varepsilon + R_B t_B = (0.151\text{ Bq})(3000\text{ s})(0.42) + (0.018\text{ s}^{-1})(3{,}000\text{ s}) = 244.26$

yields

$$u_c(Y) = \sqrt{\frac{244.26 + (108)(3{,}000)^2/(6{,}000)^2}{(3000)^2(0.42)^2} + (0.151)^2(0.05^2)} = 0.0151\text{ Bq/cm}^2$$

Thus, the uncertainty at y_Q = 0.151 is 0.0151 and the relative uncertainty is 0.1, so y_Q is verified to be the MQC.

As in example 7, we adjust for the (now) 5% relative combined standard uncertainty in the counting efficiency. The uncertainty is (0.05) × (0.42) = 0.02142. Assuming that the efficiency is normally distributed, the lower 5th percentile is (0.42) - (1.645)(0.021) = 0.385. Therefore a conservative estimate of the efficiency is ε = 0.385 and a conservative estimate of the minimum detectable concentration is: $y_Q = \frac{(0.151)(0.42)}{0.385} = 0.165\text{ Bq/cm}^2$.

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GLOSSARY

Accessible Area is an area that can be easily reached or obtained. In many cases an area must be physically accessible to perform a measurement. However, radioactivity may be measurable even if an area is not physically accessible. See in this glossary *measurable radioactivity*.

Action Level is the numerical value that causes a decision maker to choose one of the alternative actions. In the context of MARSAME, the numerical value is the radionuclide concentration or level of radioactivity corresponding to the disposition criterion, and the alternative actions are determined by the selection of a disposition option.

Alternative Action is the choice between two mutually exclusive possibilities. See in this glossary *decision rule*.

Ambient Radiation is radiation that is currently present in the surrounding area. Ambient radiation may include natural background, instrument background, intrinsic radiation from surrounding materials, intrinsic radiation from the item(s) being measured, contamination, or radiation from nearby machines (e.g., x-ray machines when operating) depending on the local conditions. Ambient radiation changes with location.

Background Radiation (as defined in Nuclear Regulatory Commission regulations) is radiation from cosmic sources; naturally occurring radioactive material including radon (except as a decay product of source or special nuclear material); and global fallout as it exists in the environment from the testing of nuclear explosive devices or from past nuclear accidents such as Chernobyl that contribute to background radiation and are not under the control of the licensee. "Background radiation" does not include radiation from source, byproduct or special nuclear materials regulated by the Nuclear Regulatory Commission (10 CFR 20.1003).

Biased Measurements are measurements performed at locations selected using professional judgment based on unusual appearance, location relative to known contamination areas, high potential for residual radioactivity, and general supplemental information. Biased measurements are not included in the statistical evaluation of survey unit data because they violate the assumption of randomly selected, independent measurements. Instead, biased measurement results are individually compared to the action levels. Biased measurements are also called judgment measurements (MARSSIM 2002).

Calibration Function is the function that relates the net instrument signal to activity (e.g., relates counts to disintegrations or radiations).

Categorization is the act of determining whether M&E are impacted or non-impacted. This is a departure from MARSSIM where this decision was included in the definition of classification.

Class 1 M&E are impacted M&E that have, or had, the following: (1) highest potential for, or known, radionuclide concentration(s) or radioactivity above the action level(s); (2) highest potential for small areas of elevated radionuclide concentration(s) or radioactivity; and (3) insufficient evidence to support reclassification as Class 2 or Class 3. Such potential may be based on historical information and process knowledge, while known radionuclide concentration(s) or radioactivity may be based on preliminary surveys. See in this glossary *Class 2*, *Class 3*, *classification*, and *impacted*.

Class 2 M&E are impacted M&E that have, or had, the following: (1) low potential for radionuclide concentration(s) or radioactivity above the action level(s); and (2) little or no potential for small areas of elevated radionuclide concentration(s) or radioactivity. Such potential may be based on historical information, process knowledge, and preliminary surveys. See in this glossary *Class 1*, *Class 3*, *classification*, and *impacted*.

Class 3 M&E are impacted M&E that have, or had, the following: (1) little or no potential for radionuclide concentrations(s) or radioactivity above background; and (2) insufficient evidence to support categorization as non-impacted. See in this glossary *Class 1*, *Class 2*, *classification*, *impacted*, and *non-impacted*.

Classification is the act or result of separating impacted M&E or survey units into one of three designated classes: Class 1, Class 2, or Class 3. Classification is the process of determining the appropriate level of survey effort based on estimates of activity levels and comparison to action levels, where the activity estimates are provided by historical information, process knowledge, and preliminary surveys. See in this glossary *Class 1*, *Class 2*, *Class 3*, and *impacted*.

Clearance is the removal of radiological regulatory controls from materials and equipment. Clearance is a subset of release. See in this glossary *release*.

Combined Standard Uncertainty is the standard uncertainty of an output estimate calculated by combining the standard uncertainties of the input estimates. The combined standard uncertainty of y is denoted by $u_c(y)$. See also in this glossary *expanded uncertainty*, *input estimate*, *measurement method uncertainty*, *output estimate*, and *standard uncertainty*.

Combined Variance is the square of the combined standard uncertainty. The combined variance of y is denoted by $[u_c(y)]^2$. See in this glossary *combined standard uncertainty*.

Concentration is activity per unit volume (e.g., Bq/kg or pCi/g) or activity per unit area (e.g., Bq/m^2 or dpm/100 cm^2).

Conceptual Model is a description of a component or area to be surveyed and the associated radionuclides or radioactivity expected to be present. The initial conceptual model is based on the results of the initial assessment. Additional data is used to update the conceptual model throughout the development, implementation, and assessment of the disposition survey. See in this glossary *initial assessment*.

Coverage Factor (k) is the value multiplied by the combined standard uncertainty $u_c(y)$ to give the expanded uncertainty, U. See in this glossary *combined standard uncertainty* and *expanded uncertainty*.

Coverage Probability is the approximate probability that the reported uncertainty interval will contain the value of the measurand. See in this glossary *level of confidence* and *measurand*.

Critical Value in the context of radiation detection is the minimum measured value (e.g., of the instrument signal or the radionuclide concentration) required to give a specified probability that a positive (nonzero) amount of radioactivity is present in the material being measured. The critical value is the same as the critical level or decision level in publications by Currie (Currie 1968 and NRC 1984).

Critical Value in the context of statistical testing is the value, which, if exceeded by the test statistic, results in rejection of the null hypothesis. See in this glossary *null hypothesis*.

Data Life Cycle is the process of planning the survey, implementing the survey plan, and assessing the survey results prior to making a decision (MARSSIM 2002).

Data Quality Objectives (DQOs) are qualitative and quantitative statements derived from the DQO process that clarify study technical and quality objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions (MARSSIM 2002).

Data Quality Objectives Process is a systematic strategic planning tool based on the scientific method that identifies and defines the type, quality, and quantity of data needed to satisfy a specific use (MARSSIM 2002). See also in this glossary *data quality objectives*.

Data Quality Assessment (DQA) is a scientific and statistical evaluation that determines whether data are the right type, quality and quantity to support their intended use (EPA 2006b).

Decision Rule is a statement that describes a logical basis for choosing among alternative actions (MARSSIM 2002). A theoretical decision rule is developed early in the planning process assuming ideal data are available to support a disposition decision (see Chapter 3). An operational decision rule is developed based on the measurements that will be performed as part of the final disposition survey (see Chapter 4).

Detection Capability is a generic term describing the capability of a measurement process to distinguish small amounts of radioactivity from zero. It may be expressed in terms of the minimum detectable concentration. See in this glossary *minimum detectable concentration*.

Difficult-to-Measure Radioactivity is radioactivity that is not measurable until the M&E to be surveyed is prepared. Preparation of M&E may be relatively simple (e.g., cleaning) or more complicated (e.g., disassembly or complete destruction). Given sufficient resources, all radioactivity can be made measurable; however, it is recognized that increased survey costs can outweigh the benefit of some dispositions.

Discrimination Limit is the level of radioactivity selected by the members of the planning team that can be reliably distinguished from the action level. The lower bound of the gray region (LBGR) for Scenario A and the upper bound of the gray region (UBGR) for Scenario B are examples of discrimination limits.

Disposition is the future use, fate, or final location for something.

Disposition Decision is the selection between alternative actions to determine acceptable future use. In statistical decision making, when the null hypothesis is rejected based on the survey data the decision maker is left with the alternative hypothesis. A failure to reject the null hypothesis is not the same as demonstrating the null hypothesis is true. See in this glossary *null hypothesis*.

Disposition Survey is a radiological survey designed to collect information to support a disposition decision.

Distinguishable from Background means that the detectable concentration of a radionuclide is statistically different from the background concentration of that radionuclide in the vicinity of the site or, in the case of structures, in similar materials using adequate measurement technology, survey and statistical techniques (10 CFR 20.1003).

Energy Resolution is the quantifiable ability of a measurement method to distinguish between radiations with different energies.

Environmental Radioactivity is radioactivity from the environment where the M&E are located. Environmental radioactivity includes background radiation as well as inherent radioactivity and radioactivity from nearby sources.

Evaluation Function is a mathematical expression that allows the user to compare options and draw a conclusion or calculate a result.

Expanded Uncertainty is the product, U, of the combined standard uncertainty of a measured value y and a coverage factor, k, chosen so that the interval from $y - U$ to $y + U$ has a desired high probability of containing the value of the measurand. See in this glossary *combined standard uncertainty*, *coverage factor*, and *measurand*.

Fluence is the number of photons or particles passing through a cross-sectional area. The international standard (SI) unit for fluence is m^{-2}.

Frequency Plot is a chart plotting the number of data points against their measured values.

Graded Approach is the process of basing the level of application of managerial controls applied to an item or work according to the intended use of the results and the degree of confidence needed in the quality of the results. See in this glossary *data quality objectives process*.

Gray Region is the range of radionuclide concentrations or quantities between the discrimination limit and the action level. See in this glossary *action level, discrimination limit, lower bound of the gray region*, and *upper bound of the gray region*.

Hard Data are quantitative data used to directly determine levels of radioactivity associated with measurement results.

Impacted is a term applied to M&E that are not classified as non-impacted. M&E with a reasonable potential to contain radionuclide concentration(s) or radioactivity above background are considered impacted (10 CFR 50.2). See in this glossary *background radiation* and *non-impacted*.

Inherent Radioactivity is radioactivity resulting from radionuclides that are an essential constituent of the material being measured (e.g., ^{40}K in fertilizer containing potassium).

Initial Assessment (IA) is an investigation to collect existing information describing materials and equipment and is similar to the Historical Site Assessment (HSA) described in MARSSIM.

Input Quantity is any of the quantities in a mathematical measurement model whose values are measured and used to calculate the value of another quantity, called the output variable.

Instrument Efficiency is the ratio between the instrument net reading and the surface emission rate of a source under given geometrical conditions (ISO 1988). For a given instrument, the instrument efficiency depends on the energy of the radiations emitted by the source. See in this glossary *source efficiency* and *total efficiency*.

Interdiction is the authoritative refusal to approve or assent to an action.

Interdiction Survey is the collection of data to support an interdiction decision regarding M&E. In general, interdiction surveys are used to accept or refuse to accept control of M&E that is potentially radioactive. In some cases an interdiction survey may result in the impoundment of radioactive M&E that represent an unacceptable risk to human health or the environment.

Interference is the presence of other radiation or radioactivity that hinders the ability to analyze for the radiation or radioactivity of interest.

Intrinsic Radioactivity is radioactivity resulting from radionuclides that are an essential constituent of the material being measured (e.g., ^{40}K in fertilizer containing potassium).

Level of Confidence (*p*) is the approximate probability that the reported uncertainty interval will contain the value of the measurand. See in this glossary *coverage probability* and *measurand*.

Lower Bound of the Gray Region (LBGR) is the radionuclide concentration or level of radioactivity that corresponds with the lowest value from the range where decision errors are not controlled for statistical hypothesis testing. For Scenario A the LBGR corresponds to the discrimination limit. For Scenario B the LBGR corresponds to the action level. See in this glossary *action level*, *discrimination limit*, *gray region*, *Scenario A*, and *Scenario B*.

Mathematical Model is the general characterization of a process, object, or concept in terms of mathematics, which enables the relatively simple manipulation of variables to be accomplished in order to determine how the process, object, or concept would behave in different situations.

Materials and Equipment (M&E) are items considered for disposition that include metals, concrete, dispersible bulk materials, tools, equipment, piping, conduit, furniture, solids, liquids, and gases in containers, etc. M&E are considered non-real property distinguishable from buildings and land, which are considered real property. See in this glossary *disposition*.

Measurand is a particular quantity subject to measurement (ISO 1996).

Measurement Method Uncertainty is the parameter, associated with the result of a measurement that characterizes the dispersion of the values that could reasonably be attributed to the measurand (ISO 1996).

Measurement Quality Objectives (MQOs) are a statement of a performance objective or requirement for a particular method performance characteristic (MARLAP 2004).

Measurable Radioactivity is radioactivity that can be quantified using known or predicted relationships developed from historical information, process knowledge or preliminary measurements as long as the relationships are developed, verified, and validated as specified in the data quality objectives (DQOs) and measurement quality objectives (MQOs).

Median is the middle value of the data set when the number of data points is odd, or the average of the two middle values when the number of data points is even.

Minimum Detectable Activity (MDA) is the minimum detectable value of activity for a measurement. See in this glossary *minimum detectable value*.

Minimum Detectable Concentration (MDC) is the minimum detectable value of the radionuclide or radioactivity concentration for a measurement. See in this glossary *minimum detectable value*.

Minimum Detectable Value is an estimate of the smallest true value of the measurand that ensures a specified high probability, $1 - \beta$, of detection. This definition presupposes that an appropriate detection criterion has been specified (e.g., critical value). See in this glossary *measurand* and *critical value*.

Minimum Quantifiable Concentration (MQC) is the smallest concentration or quantity of radioactivity the measurement method will indicate within a specified relative standard deviation.

Non-impacted is a term applied to M&E where there is no reasonable potential to contain radionuclide concentration(s) or radioactivity above background (10 CFR 50.2). See in this glossary *background radioactivity* and *impacted*.

Null Hypothesis, or baseline condition, is a tentative assumption about the true, but unknown, radionuclide concentration or level of radioactivity that can be retained or rejected based on the available evidence. When hypothesis testing is applied to disposition decisions, the data are used to select between a presumed baseline condition (the null hypothesis) and an alternate condition (the alternative hypothesis). The null hypothesis is retained until evidence demonstrates with a previously specified probability that the baseline condition is false.

Output Quantity is the quantity in a mathematical measurement model whose value is calculated from the measured values of other quantities in the model. See in this glossary *input quantity*.

Planning Team is the group of people who perform the DQO process. Members include the decision maker (senior manager), site manager, representatives of other data users, senior program and technical staff, someone with statistical expertise, and a quality assurance and quality control advisor (such as a QA manager) (EPA 2000a).

Posting Plot is a map of the survey unit with the data values entered at the measurement locations. This type of plot potentially reveals heterogeneities in the data, especially possible patches of elevated contamination.

Preliminary Survey is any survey performed prior to the disposition survey in MARSAME, and is generally performed to provide information required to support the design of the final survey. See also in this glossary *disposition survey*.

Process Knowledge is information concerning the characteristics, history of prior use, and inherent radioactivity of the materials and equipment being considered for release. Process knowledge is obtained through a review of the operations conducted in facilities or areas where materials and equipment may have been located and the processes where the materials and equipment were involved.

Radioactive Materials consist of any material, equipment or system component determined or suspected to contain radionuclides in excess of inherent radioactivity. Radioactive material includes activated material, sealed and unsealed sources, and substances that emit radiation. See in this glossary *inherent radioactivity*.

Radiological Controls are any means, method or activity (including engineered or administrative) designed to protect personnel or the environment from exposure to a radiological risk.

Radionuclides or Radiations of Concern are radionuclides or radiations that are present at a concentration or activity that poses an unacceptable risk to human health or the environment. In MARSAME, the term radionuclides or radiations of concern is used to describe the radionuclides or radiations that are actually measured during the disposition survey. See also in this glossary *radionuclides or radiations of potential concern* and *disposition survey*.

Radionuclides or Radiations of Potential Concern are radionuclides or radiations that are identified during the initial assessment as potentially being associated with the M&E being investigated. See also in this glossary *initial assessment*.

Ratemeter is an instrument that indicates the counting rate of an electronic counter. In the context of radiological measurements, a ratemeter displays the counting rate from a radiation detector. The averaging time for calculating the rate is determined by the time constant of the meter. See in this glossary *scaler*.

Recycle is beneficial reuse of constituent materials incorporated within the M&E. A hammer that is melted down as scrap metal so the component metals can be reused is an example of recycle.

Reference Material is material of similar physical, radiological, and chemical characteristics as the M&E considered for disposition. Reference material provides information on the level of radioactivity that would be present if the M&E being investigated had not been radiologically impacted. See in this glossary *impacted*.

Relative Standard Uncertainty is the ratio of the standard uncertainty of a measured result to the result itself. The relative standard uncertainty of x may be denoted by $u_r(x)$. See in this glossary *standard uncertainty*.

Release is a reduction in the level of radiological control, or a transfer of control to another party. Examples of release include clearance (i.e., unrestricted release of materials and equipment to the public sector), recycle, reuse, disposal as waste, or transfer of control of radioactive M&E from one authorized user to another. See also in this glossary *reuse*, *recycle*, *restricted release,* and *clearance*.

Release Survey is a type of disposition survey designed to collect information to support a release decision. See also in this glossary *disposition survey* and *release*.

Restricted Release is a reduction in the level of radiological control, or transfer of control to another party, where restrictions are placed on how the released items will be used or transferred. Maintaining a tool crib in a radiologically controlled area restricts reuse of those tools to that radiologically controlled area, and tools returned to the tool crib represent a restricted release of those tools.

Reuse is the continued use of M&E for their original purpose(s). An example of reuse is a hammer that continues to be used as a hammer.

Ruggedness is the relative stability of a measurement technique's performance when small variations in method parameter values are made.

Scaler is an electronic counter that displays the aggregate of a number of signals, which usually occur too rapidly to be recorded individually. In the context of radiological measurements, a scaler records the number of counts from a radiation detector over a specified time interval. See in this glossary *ratemeter*.

Scenario A uses a null hypothesis that assumes the level of radioactivity associated with the M&E exceeds the action level. Scenario A is sometimes referred to as "presumed not to comply" or "presumed not clean."

Scenario B uses a null hypothesis that assumes the level of radioactivity associated with the M&E is less than or equal to the action level. Scenario B is sometimes referred to as “indistinguishable from background” or “presumed clean.”

Secular Equilibrium is the condition in which the precursor radionuclide in a decay series has a longer half-life than any subsequent members of the series. Secular equilibrium is achieved when the activities for all members of the decay series are equal to the activity of the precursor radionuclide.

Segregation is the process of separating or isolating from a main body or group. In the context of disposition surveys, segregation is based on the physical and radiological attributes of the M&E being investigated and is used to help control measurement method uncertainty.

Sensitivity Coefficient for an input estimate, x_i, used to calculate an output estimate, $y=f(x_1, x_2, \ldots, x_N)$, is the value of the partial derivative, $\partial f/\partial x_i$, evaluated at $i=x_1, x_2, \ldots, x_N$. The sensitivity coefficient represents the ratio of the change in y to a small change in x_i.

Sentinel Measurement is a biased measurement performed at a key location to provide information specific to the objectives of the Initial Assessment (IA).

Significance Level is, in the context of a hypothesis test, a specified upper limit for the probability of a Type I decision error.

Sign Test is a non-parametric statistical test used to evaluate disposition survey results if the radionuclide being measured is not present in background, or is present at such a small fraction of the action level as to be considered insignificant.

Smear is a non-quantitative test for the presence of removable radioactive materials in which the suspected surface or area is wiped with a filter paper or other substance, which is then tested for the presence of radioactivity. The surface area tested may be related to the release criterion. Smear is also referred to as a smear test, swipe, or wipe.

Soft data are qualitative and/or quantitative data that do not directly determine levels of radioactivity. Soft data provide information that is used to infer or deduce knowledge concerning the levels of radioactivity in materials and equipment.

Source Efficiency is the ratio between the number of particles of a given type above a given energy emerging from the front face of a source or its window per unit time and the number of particles of the same type created or released within the source (for a thin source) or its saturation layer thickness (for a thick source) per unit time (ISO 1988). See also in this glossary *instrument efficiency* and *total efficiency*.

Specific Activity is the radioactivity per unit mass for a specified radionuclide.

Specificity is the ability of the measurement method to measure the radionuclide of concern in the presence of interferences.

Spectrometry is a measurement across a range of energies. The measurement of alpha particles by energy is called alpha spectrometry.

Spectroscopy is the measurement and analysis of electromagnetic spectra produced as the result of the emission or absorption of energy by various substances. The measurement of gamma-ray emissions from a substance is called gamma spectroscopy.

Standard Operating Procedure (SOP) is a written document that details the method for an operation, analysis, or action with thoroughly prescribed techniques and steps, and that is officially approved as the method for performing certain routine or repetitive tasks (MARSSIM 2002).

Standard Uncertainty is the uncertainty of a measured value expressed as an estimated standard deviation, often called a "1-sigma" (1σ) uncertainty (MARLAP 2004). The standard uncertainty of a value x is denoted by $u(x)$.

Standardized Initial Assessment is a set of instructions or questions that are used to perform the initial assessment, usually documented in a standard operating procedure. See also in this glossary *initial assessment* and *standard operating procedure*.

Structures are buildings or other objects constructed from several parts.

Surficial Radioactive Material is radioactive material distributed on any of the surfaces of a solid object. Surficial radioactive material may be removable by non-destructive means (such as casual contact, wiping, brushing, or washing) or fixed.

Surrogate Measurement is a measurement where one radionuclide is quantified and used to demonstrate compliance with the release criterion for additional radionuclide(s) based on known or accepted relationships between the measured radionuclide and unmeasured radionuclides.

Survey Unit for M&E is the specific lot, amount, or piece of equipment on which measurements are made to support a disposition decision concerning the same specific lot, amount, or piece of equipment.

Total Efficiency is the product of the instrument efficiency and the source efficiency. See in this glossary *instrument efficiency* and *source efficiency*.

Type I Decision Error occurs when the null hypothesis is rejected when it is actually true. The Type I decision error rate, or significance level, is represented by α. See in this glossary *null hypothesis* and *significance level*.

Type II Decision Error occurs when the null hypothesis is not rejected when it is actually false. The Type II decision error rate is denoted by β. See in this glossary *null hypothesis*.

Unrestricted Release is the removal of radiological regulatory controls from materials and equipment. Clearance is a subset of release. See in this glossary *release* and *clearance*.

Upper Bound of the Gray Region (UBGR) is the radionuclide concentration or level of radioactivity that corresponds with the highest value from the range where decision errors are not controlled for statistical hypothesis testing. For Scenario A the UBGR corresponds to the action level. For Scenario B the UBGR corresponds to the discrimination limit. See in this glossary *action level*, *discrimination limit*, *gray region*, *Scenario A*, and *Scenario B*.

Volumetric Radioactive Material is radioactive material that is distributed throughout or within the materials or equipment being measured, as opposed to a surficial distribution. Volumetric radioactive material may be homogeneously (e.g., uniformly activated metal) or heterogeneously (e.g., activated reinforced concrete) distributed throughout the M&E.

Wilcoxon Rank Sum (WRS) Test is a non-parametric statistical tests used to evaluate disposition survey results if the radionuclide being measured is present in background by comparing the results to measurements performed using an appropriately chosen reference material.

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The Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME) is a supplement to the Multi-Agency Radiation Survey and Site Assessment Manual (MARSSIM). MARSAME provides information on planning, conducting, evaluating, and documenting radiological measurements and decisions on the disposition of materials and equipment based on action levels for release or interdiction. MARSAME is a multi-agency consensus document that was developed collaboratively by four Federal agencies having authority and control over radioactive materials: Department of Defense (DOD), Department of Energy (DOE), Environmental Protection Agency (EPA), and Nuclear Regulatory Commission (NRC). MARSAME's objective is to describe consistent and technically defensible approaches for radiological measurements of materials and equipment, while at the same time encouraging an effective use of resources.

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