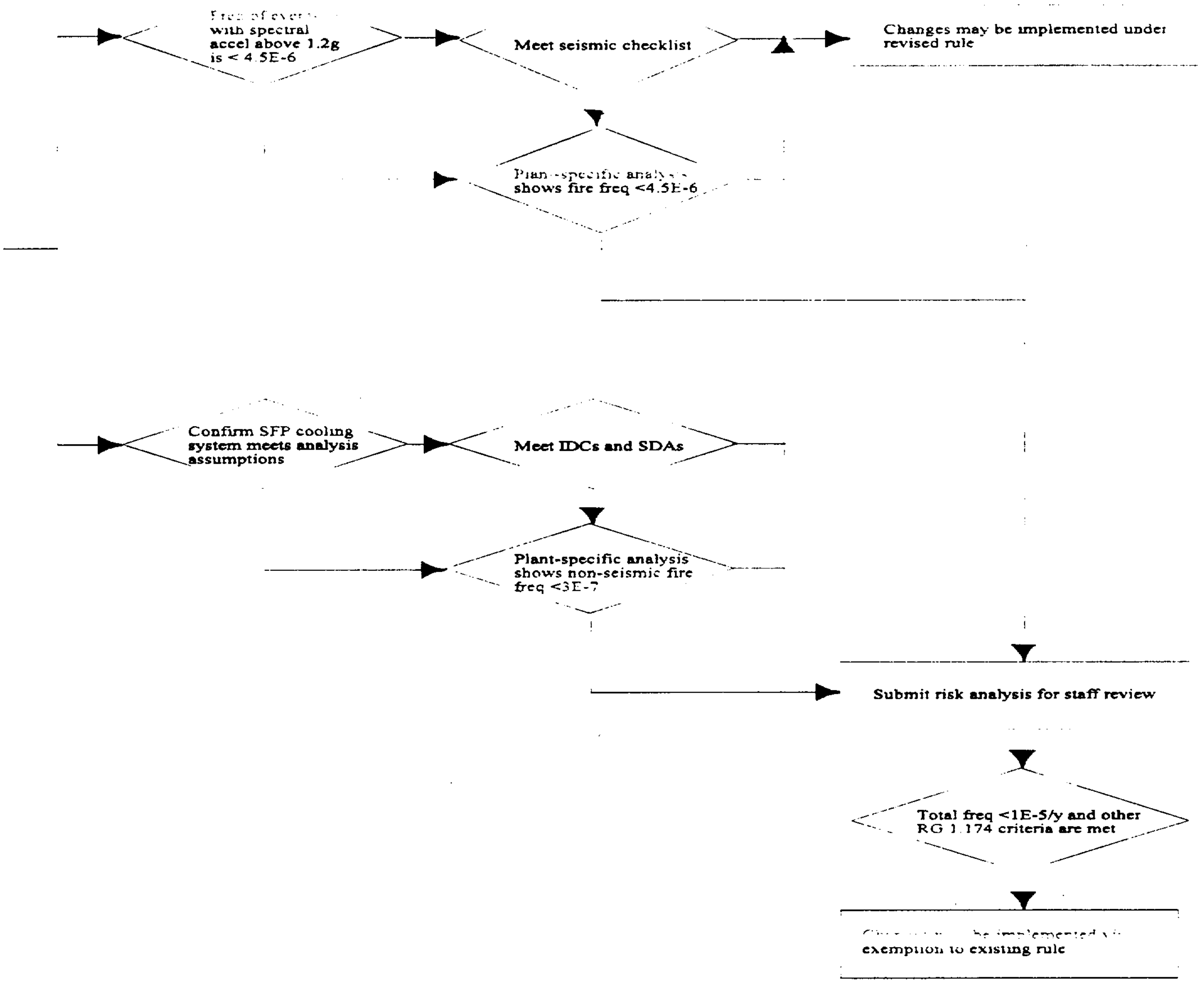
Supporting Data for Report

F/2



with spectral accel above 1.2g is < 4.5E-6
Meet seismic checklist
Changes may be implemented under revised rule
shows fire freq <4.5E-6
Confirm SFP cooling system meets analysis assumptions
Meet IDCs and SDAs
Plant-specific analysis shows non-seismic fire freq <3E-7
Submit risk analysis for staff review
Total freq <1E-5/y and other RG 1.174 criteria are met
exemption to existing rule

Air Cooled Heatup Times

PWR W 17x17

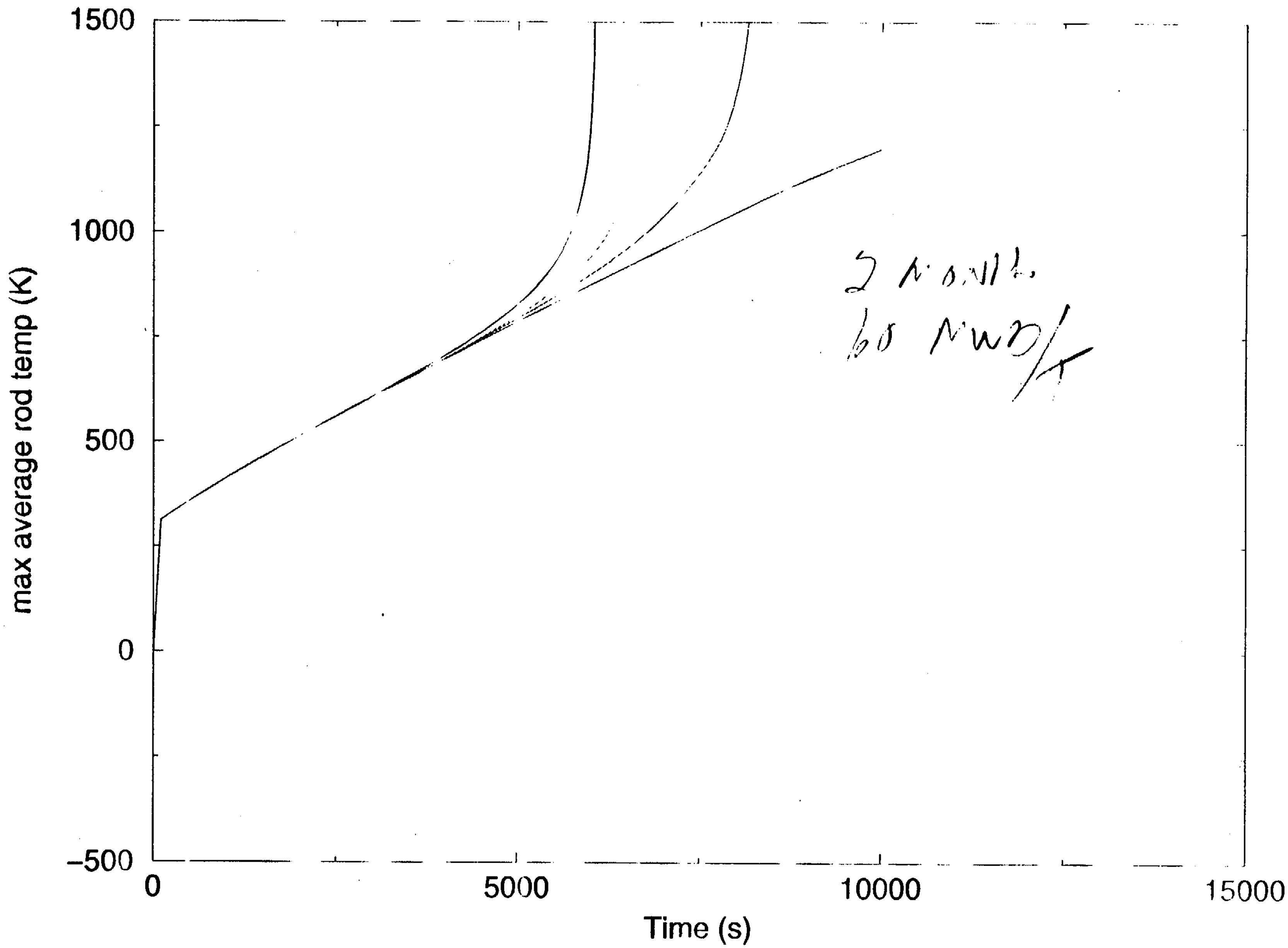
Heatup Time oxidation model				Decay Time	Burnup	Building Flow
nur1h	nur1l	ait1	noox			
1.6	2.0	1.8	2.3	2 months	60	full
4.1	5.3	4.6	6.7	1 year	60	full
4.1	5.3	4.6	6.6	1 year	60	half
4.9	6.5	5.6	8.5	1 year	50	full
7.7	11.1	9.0	17.5	2 years	60	full
T < 800 C				5 years	60	full
27.3	T<800	38.6	T<800	5 years	60	half

Air Cooled Heatup Times

PWR W 17x17

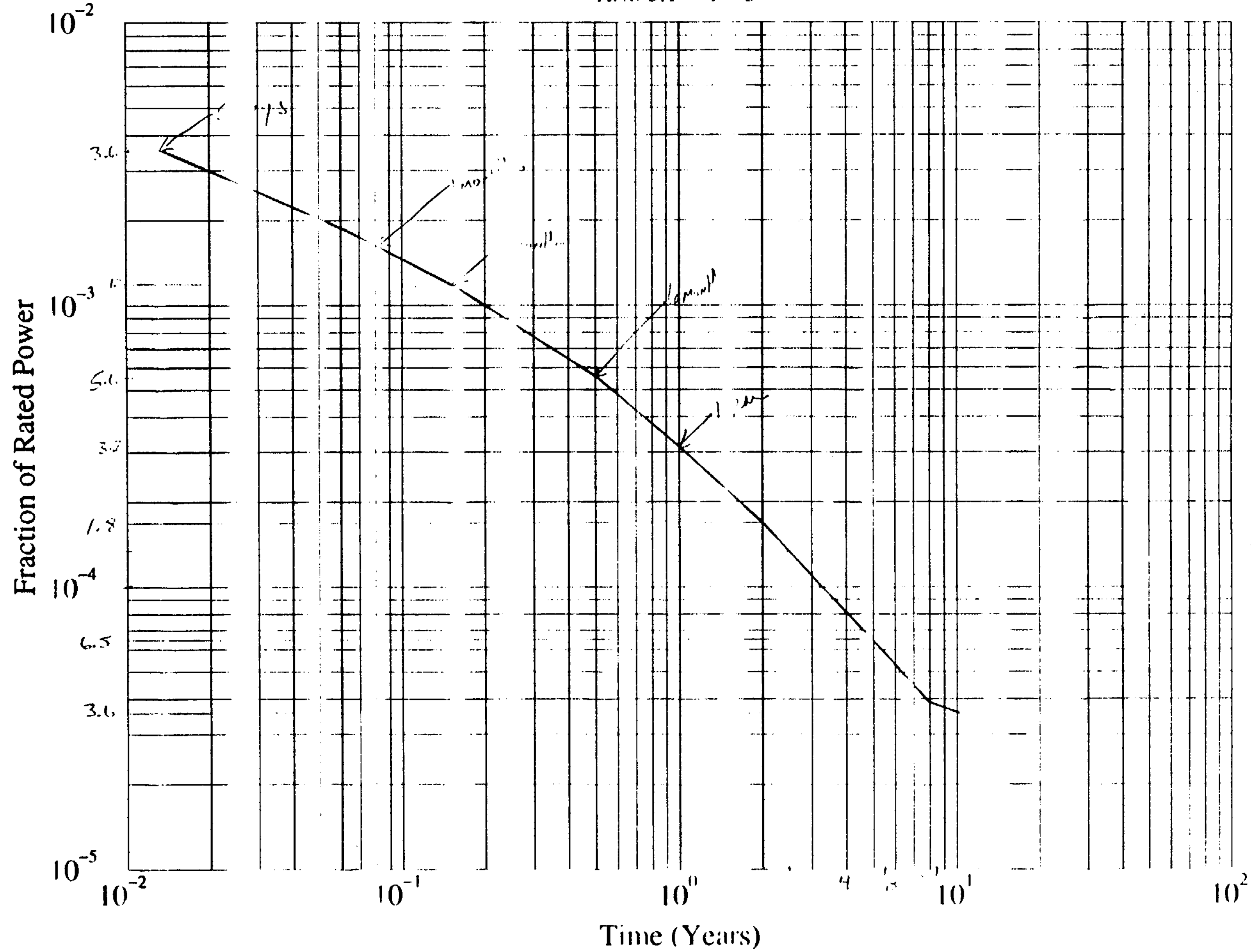
Heatup Time oxidation model nur1h	nur1l	ait1	noox	Decay Time	Burnup	Building Flow
1.6	2.0	1.8	2.3	2 months	60	full
4.1	5.3	4.6	6.7	1 year	60	full
4.1	5.3	4.6	6.6	1 year	60	half
4.9	6.5	5.6	8.5	1 year	50	full
7.7	11.1	9.0	17.5	2 years	60	full
T < 800 C				5 years	60	full
27.3	T<800	38.6	T<800	5 years	60	half

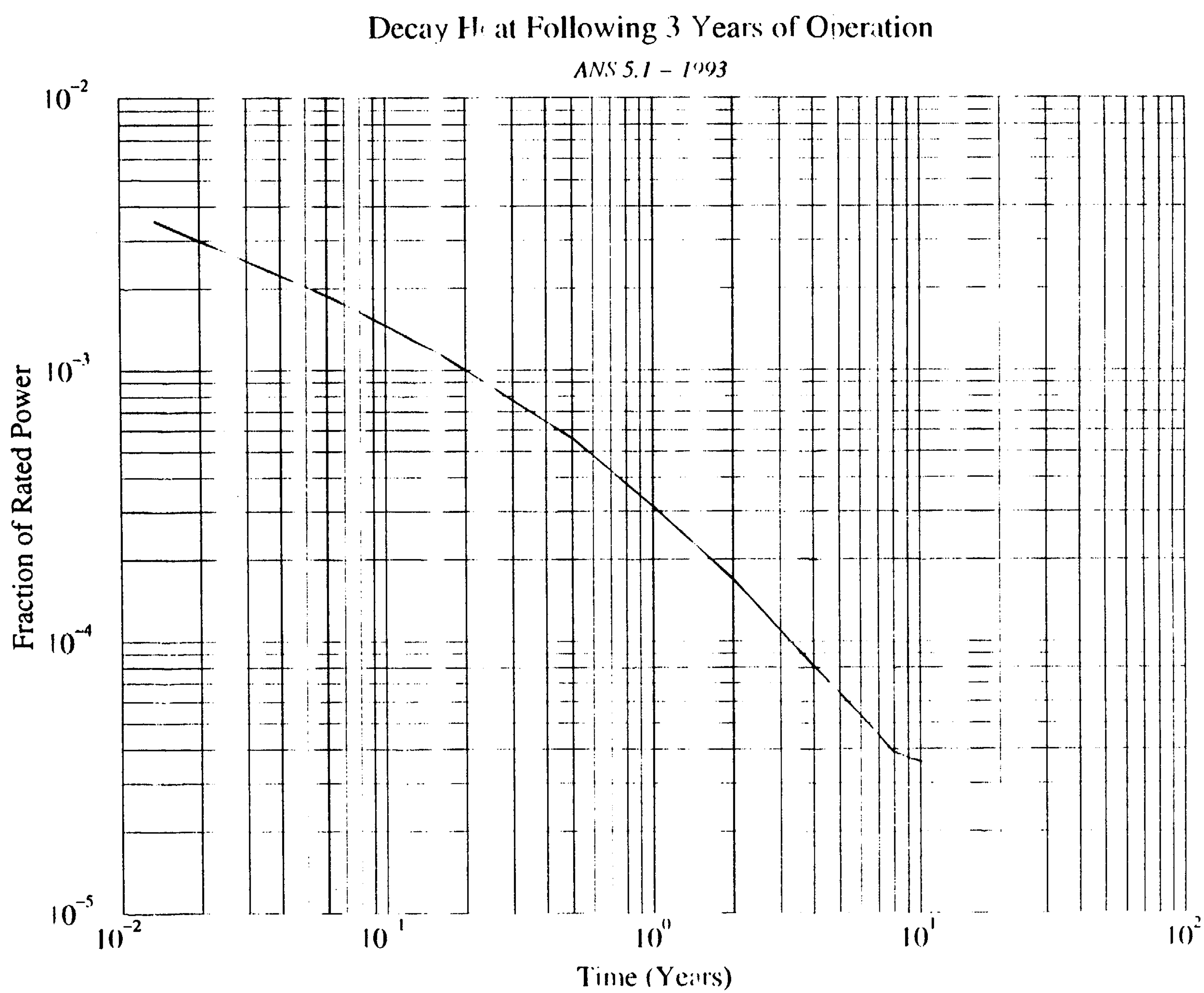
for BWR 7.0 hrs


1500
1000
500
0
−500
0
5000
10000
15000
max average rod temp (K)
Time (s)
2 months
60 MWD/t

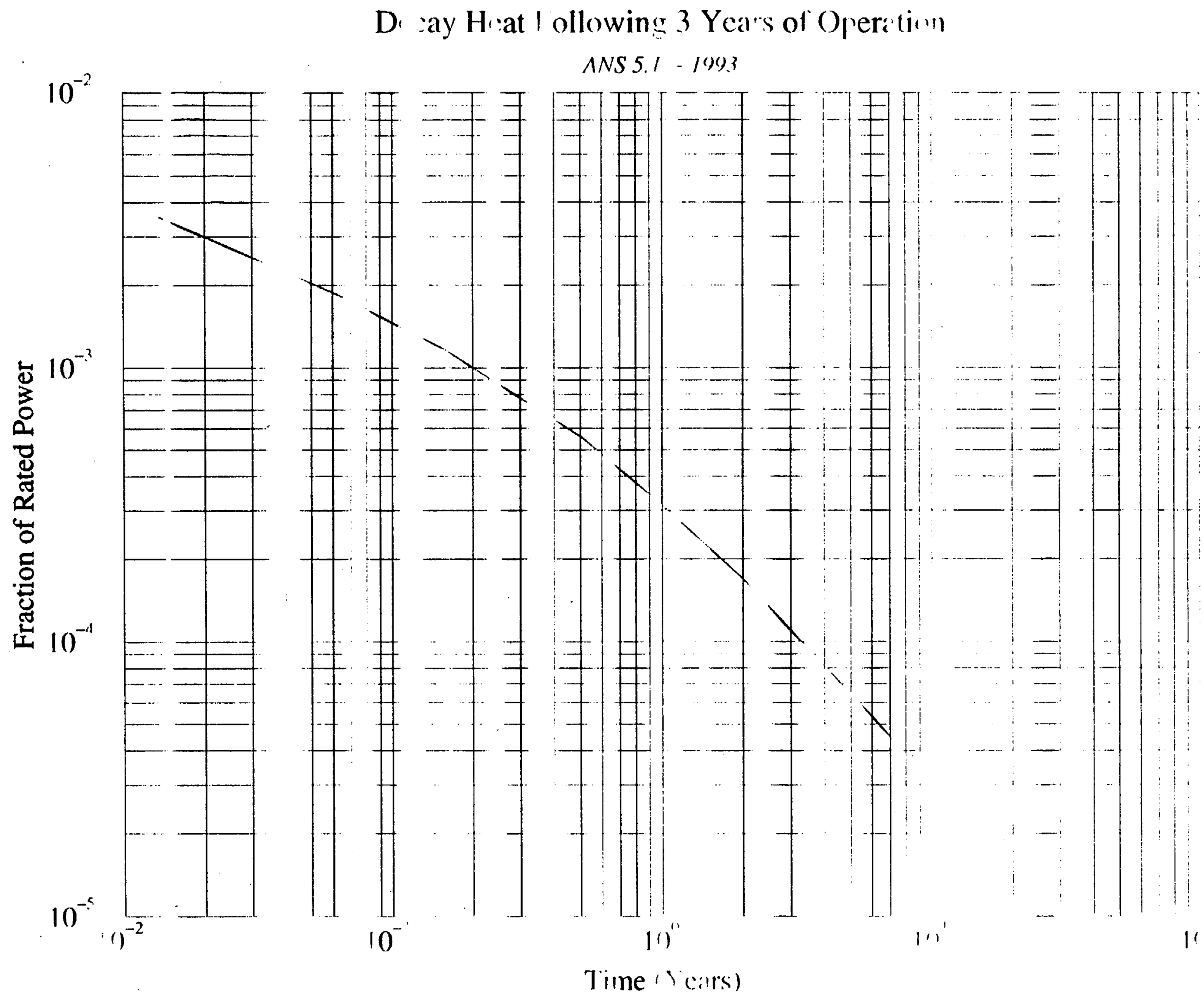
Decay Heat Following 3 Years of Operation

ANS 5.1 – 1993



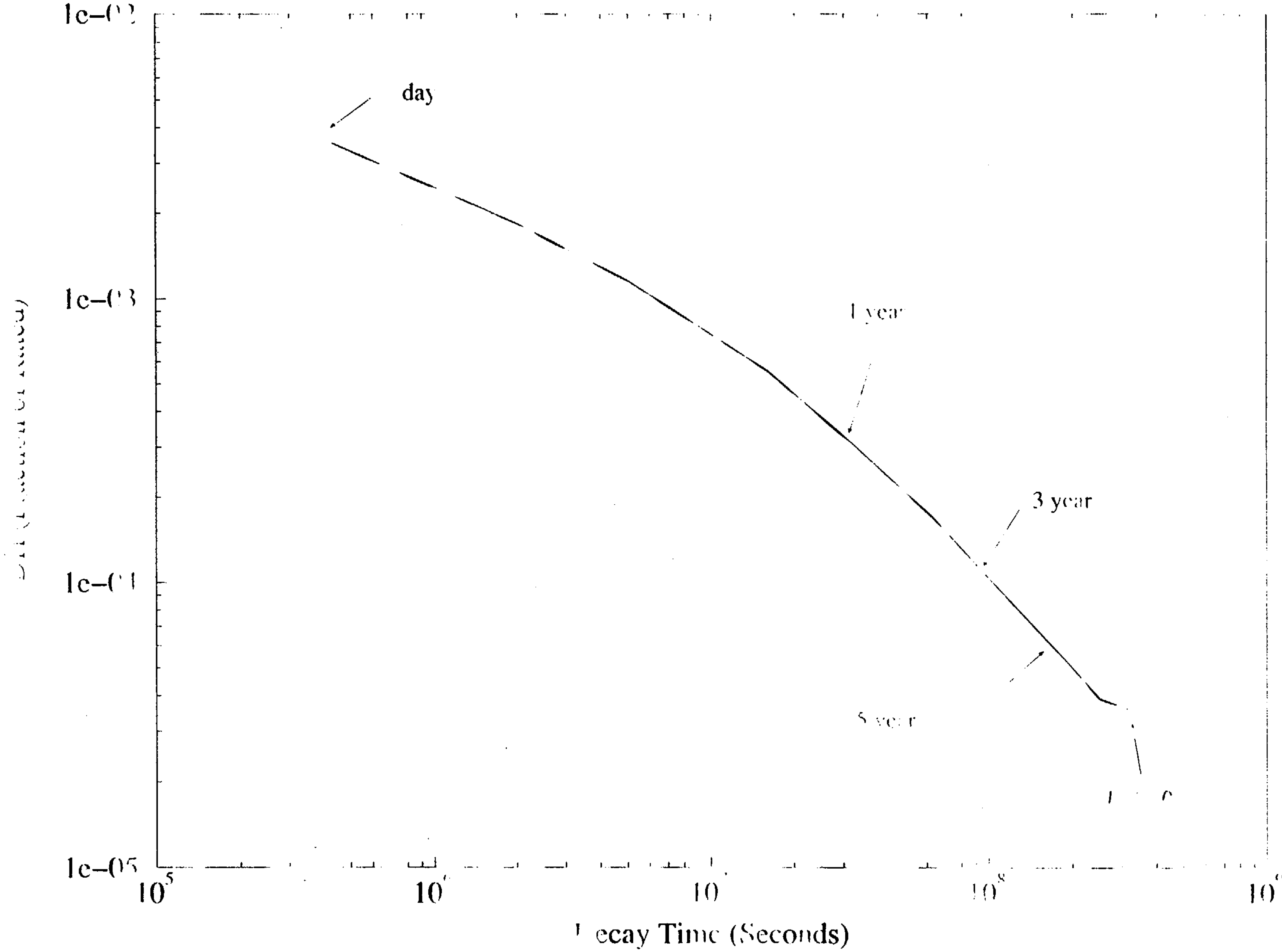


Decay Heat Following 3 Years of Operation
ANS 5.1 – 1993
Fraction of Rated Power
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
Time (Years)


Decay Heat Following 3 Years of Operation
ANS 5.1 - 1993
Fraction of Rated Power
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
Time (Years)

Decay Heat Following 3 Years of Operation

ANS 5.1 – 1993



Time to reach 800°C

Air Cooling (EP)

Case	Decay Time	BU	Oxidation Model	Bldg Air	
AC-1	1 yr	60	A	A	
AC-2	1 yr	60	B	A	
AC-3	"	60	A	B	
AC-4	1 yr	60	B	B	
AC-5	1 yr	50	A	A	?
AC-6	2 MO	60	A	A	
AC-7	2 yr	60	A	A	
AC-8	5 yr	60	A	A	
AC-9					
AC-10					
AC-11					
A[illegible]					

Boildown Case

BD-1	2 MO	60	N/A	N/A	[Human Error Sensitivity]

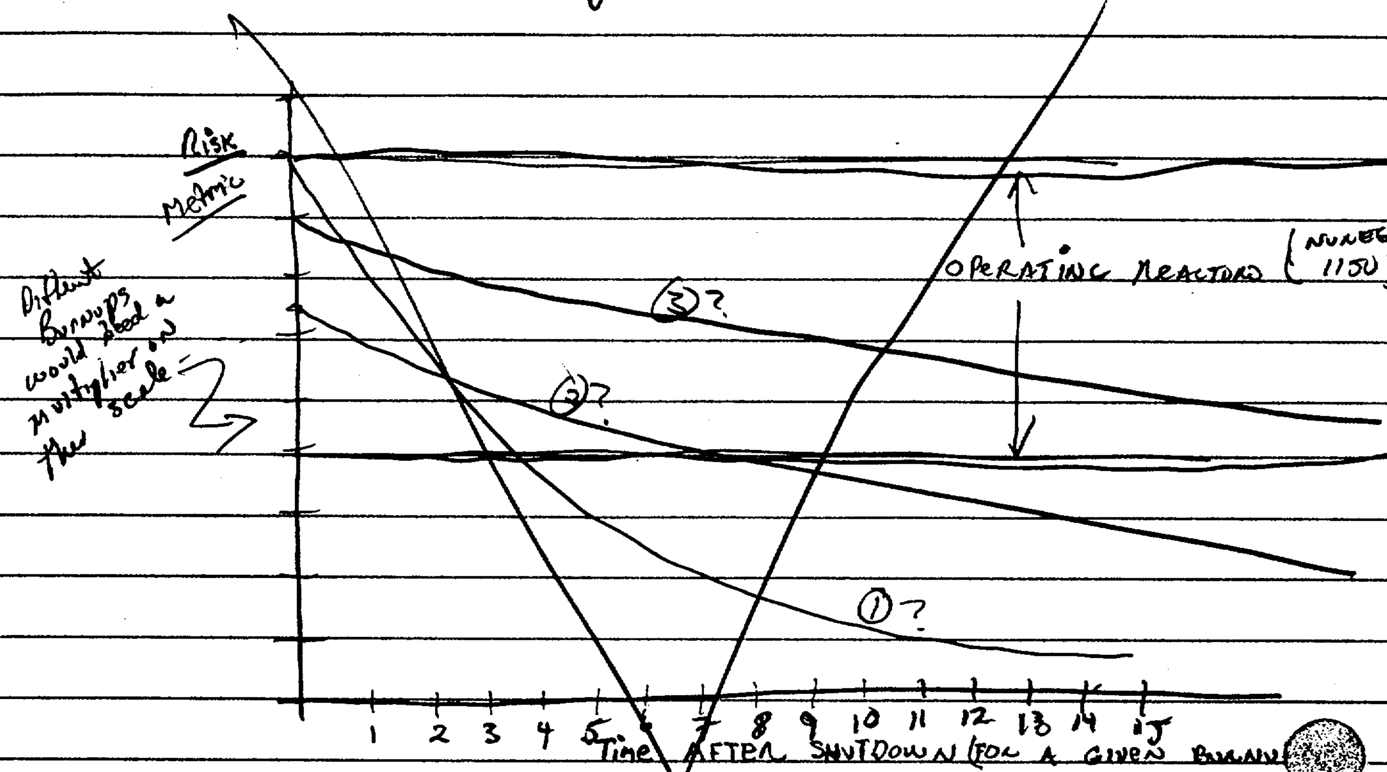
STEAM Cooling

SC-1	1 YR	60	A^{STEAM}	N/A	[Show Importance of ADIABATIC ASSUMPTION]

Critical DECAY Time (Insurance)

CDT-1	AIR	60	A	A	
-2	AIR	60	B	A	
-3	AIR	60	B	B	
-4	AIR	60	A	B	
-5	AIR	50	A	A	} ?
-6	AIR	70	A	A	}
-7	STEAM	60	A^{STEAM}	A	}
-8					
-9					
-10					

Indemnity Decision Plot

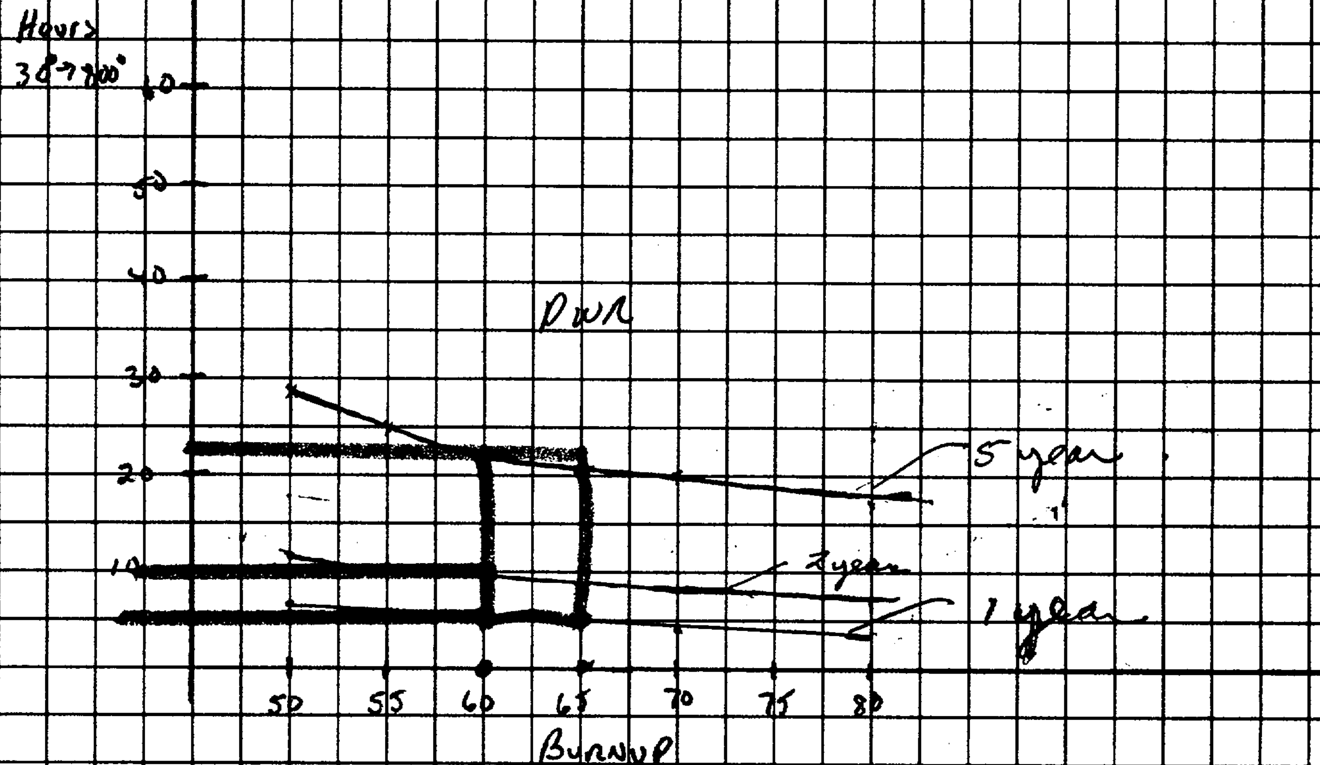


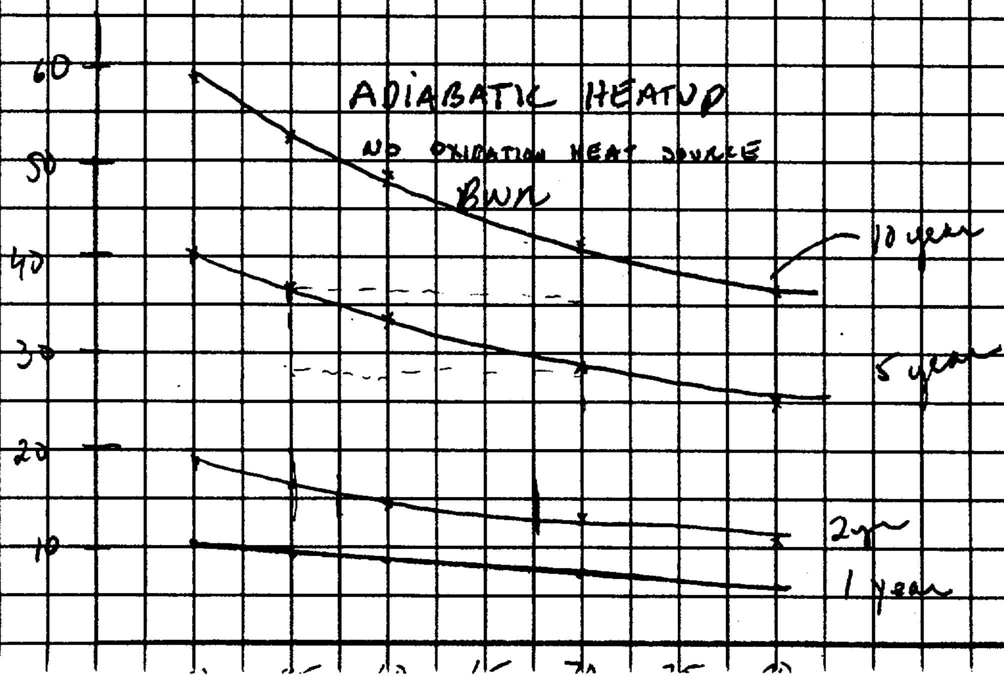
Risk metric could be: person rem/year
curies/year released
cancers/year

Key Question: Is operating reactor insurance sufficient to cover claims from a major accident?
If not, then lower risk may not be sufficient basis for decision. If the damages could still ~~be as~~ exceed the limits of liability maybe insurance should be maintained.

18 MONTH cycle

Batch	AGE	Relative Decay Heat
1a	1	1.0
1b	1	1.0
1c	1	1.0
2	2.5	~.33
3	4.0	
4	5.5 y	0.2
5	7.0	
6	8.5	
7	10.0	0.1





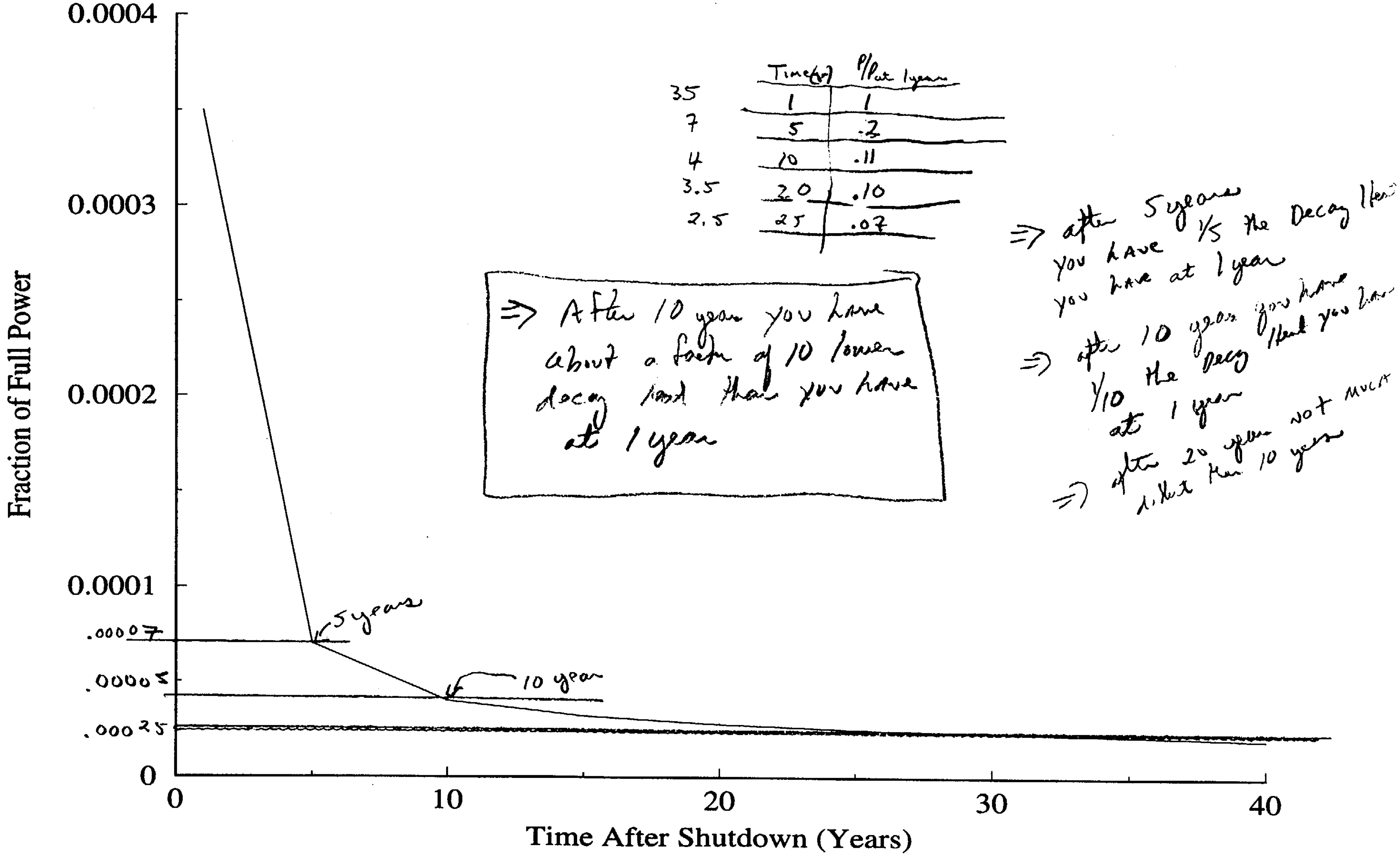
For Expected burnup (55→70)

BWR

Years	Time
1	8-10 h
2	13-18
5	30-35
10	43-53

Decay Heat Following Shutdown for TMI

Estimated Values



8-21-00

The times calculated are in hours for an adiabatic heatup from 30 C to 800 C with no oxidation heat source. Adiabatic Heatups are based on a Peaking Factor of 1.1 for PWRs and 1.2 for BWRs. The decay heats at these burnup values are interpolations or extrapolations of the decay heat from NUREG/CR-5625. The thermal mass of the BWR fuel is modeled as 9x9 fuel assemblies and the associated fuel rack structure. The mass per assembly is 170 kg UO2, 97.5 kg Zirconium and 42.4 kg stainless steel. The thermal mass PWR fuel is modeled 17x17 fuel assemblies and the associated rack structure. The mass per assembly is kg UO2, 101 kg Zirconium, and 68.6 kg stainless steel. Temperature dependent values of the specific heat are used for steel, zircaloy, and UO2.

Adiabatic Heatup Time at 1 Year

Burnup	PWR	BWR
50	6.1	10.1
55	5.6	9.2
60	5.2	8.5
70	4.4	7.2
80	3.8	6.4

Adiabatic Heatup Time at 2 Years

Burnup	PWR	BWR
50	11.2	17.9
55	10.2	16.1
60	9.4	14.9
70	8.0	12.8
80	7.1	11.1

Adiabatic Heatup Time at 5 Years

Burnup	PWR	BWR
50	28.0	40.0
55	25.4	36.4
60	23.3	33.4
70	19.9	28.5
80	17.4	25.0

Adiabatic Heatup Time at 10 Years

Burnup	PWR	BWR
50	42.8	58.0
55	38.9	52.9
60	35.6	48.4
70	30.5	41.5
80	26.7	36.2

8-21-00

Decay Time in Years for a 10 Hour Adiabatic Heatup Time

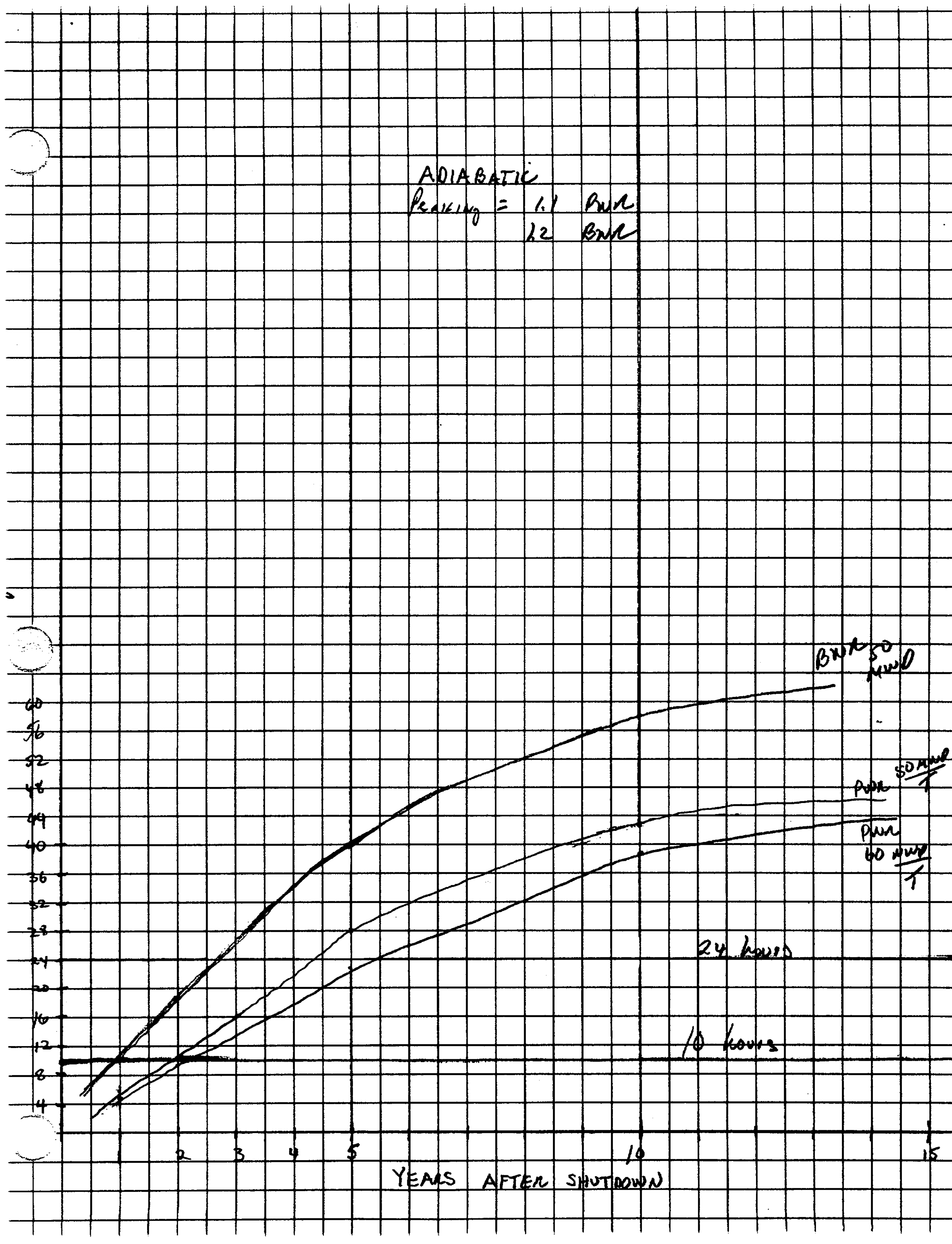
Burnup	PWR	BWR
50	1.8	1.0
55	2.0	1.2
60	2.2	1.3
70	2.6	1.6
80	2.9	1.9

Decay Time in Years for a 24 Hour Adiabatic Heatup Time

Burnup	PWR	BWR
50	4.3	2.8
55	4.8	3.2
60	5.1	3.5
70	6.6	4.1
80	8.4	4.9

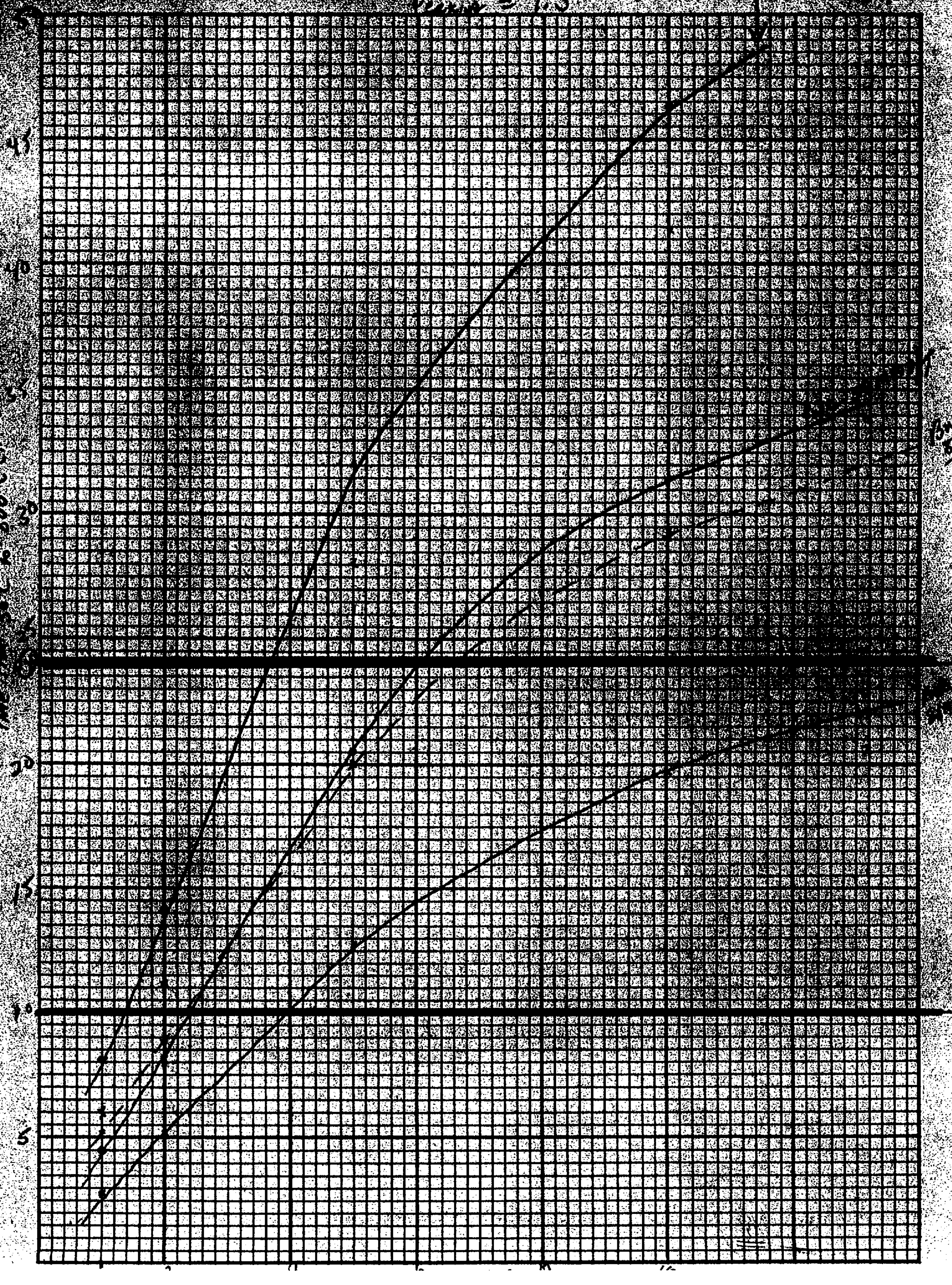
Spent Fuel Pool Heatup and Boiloff Time in hours to 3 feet Above Active Fuel. Fuel Burnup is 62.5 Gwd/MTU with a 2 year cycle time. The decay heat at this value of burnup is an extrapolation of the decay heat from NUREG/CR-5625. The BWR pool holds 4200 9x9 fuel assemblies. The pool surface are is 105.7 square meters. The PWR pool holds 965 17x17 fuel assemblies. The pool surface are is 61.3 square meters. The pools have a water depth of 11.54 meters and are assumed to be at an initial temperature of 30 C. An estimated volume fraction of 0.5 of water in the racks and assemblies was used. Errors in this value can impact the heatup time portion of the heatup and boiloff calculation. The specific heat of water was assumed to be constant at 4200 J/kg for the heatup calculation. Temperature dependent properties were used for steel, zircaloy, and UO2. The enthalpy change due to vaporization used in the boiloff calculation is 2257 KJ/kg.

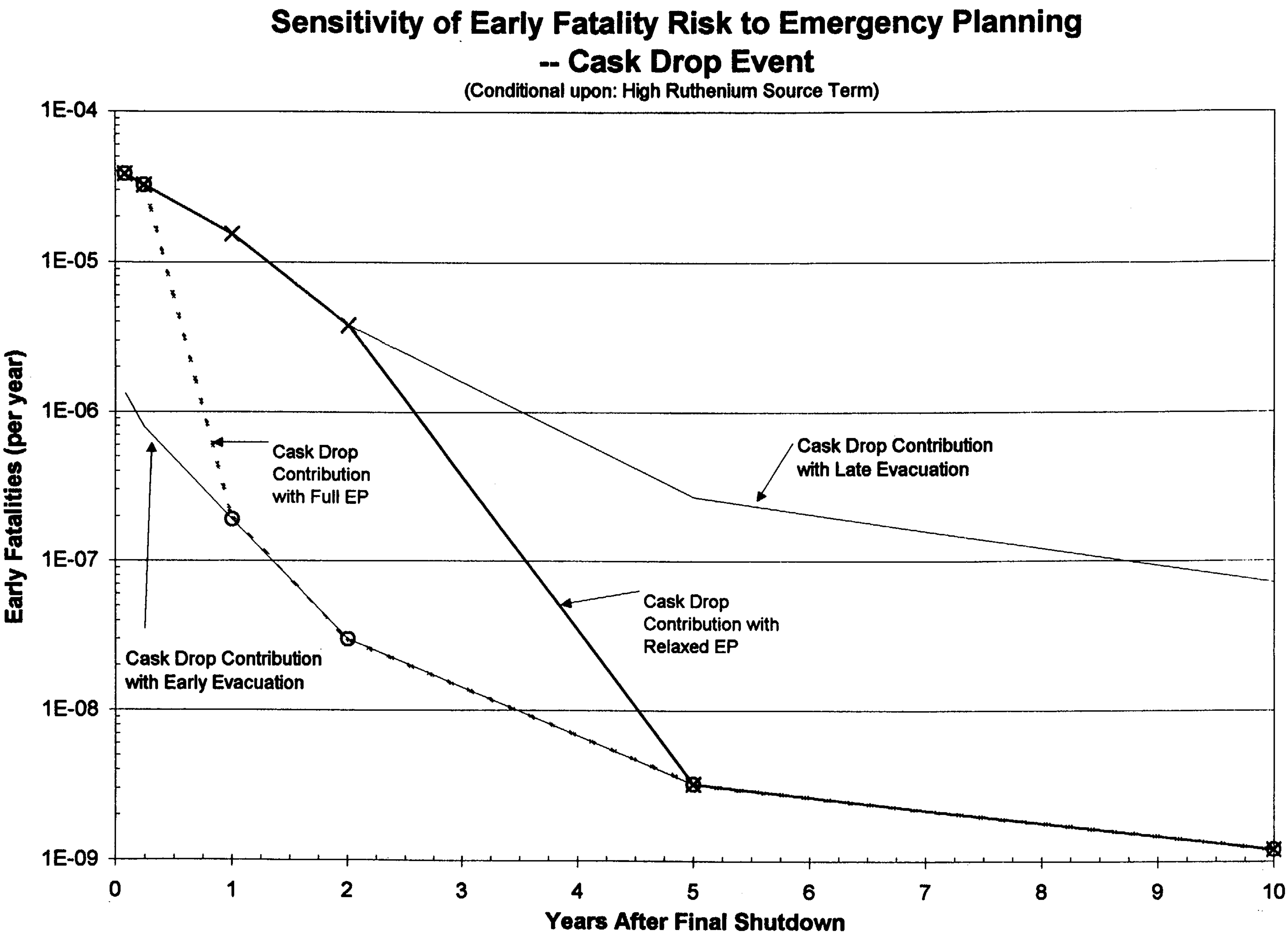
Decay Time	PWR	BWR
1 year	195	253
2 year	272	337
5 year	400	459
10 year	476	532



ADIABATIC
Peaking = 1.1 PWR
1.2 BWR
BWR 50 MWD
PWR 50 MWD/T
PWR 60 MWD/T
24 hours
10 hours
60
56
52
48
44
40
36
32
28
24
20
16
12
8
4
1
2
3
4
5
10
15
YEARS AFTER SHUTDOWN

ADIABATIC HEATUP





Sensitivity of Early Fatality Risk to Emergency Planning
-- Cask Drop Event
(Conditional upon: High Ruthenium Source Term)
1E-04
1E-05
1E-06
1E-07
1E-08
1E-09
Early Fatalities (per year)
0
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown
Cask Drop Contribution with Full EP
Cask Drop Contribution with Late Evacuation
Cask Drop Contribution with Relaxed EP
Cask Drop Contribution with Early Evacuation

Figure 3.7-5

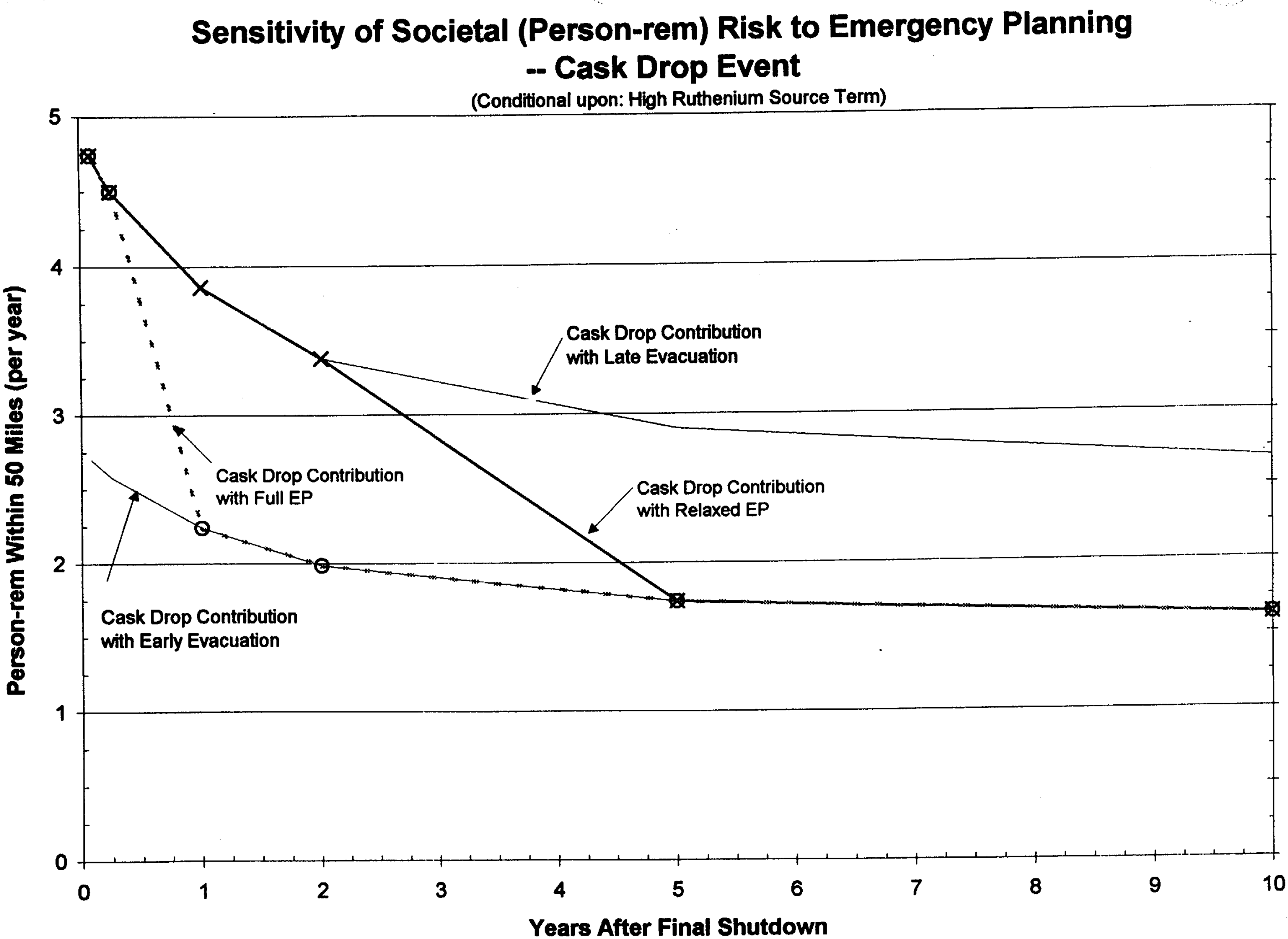
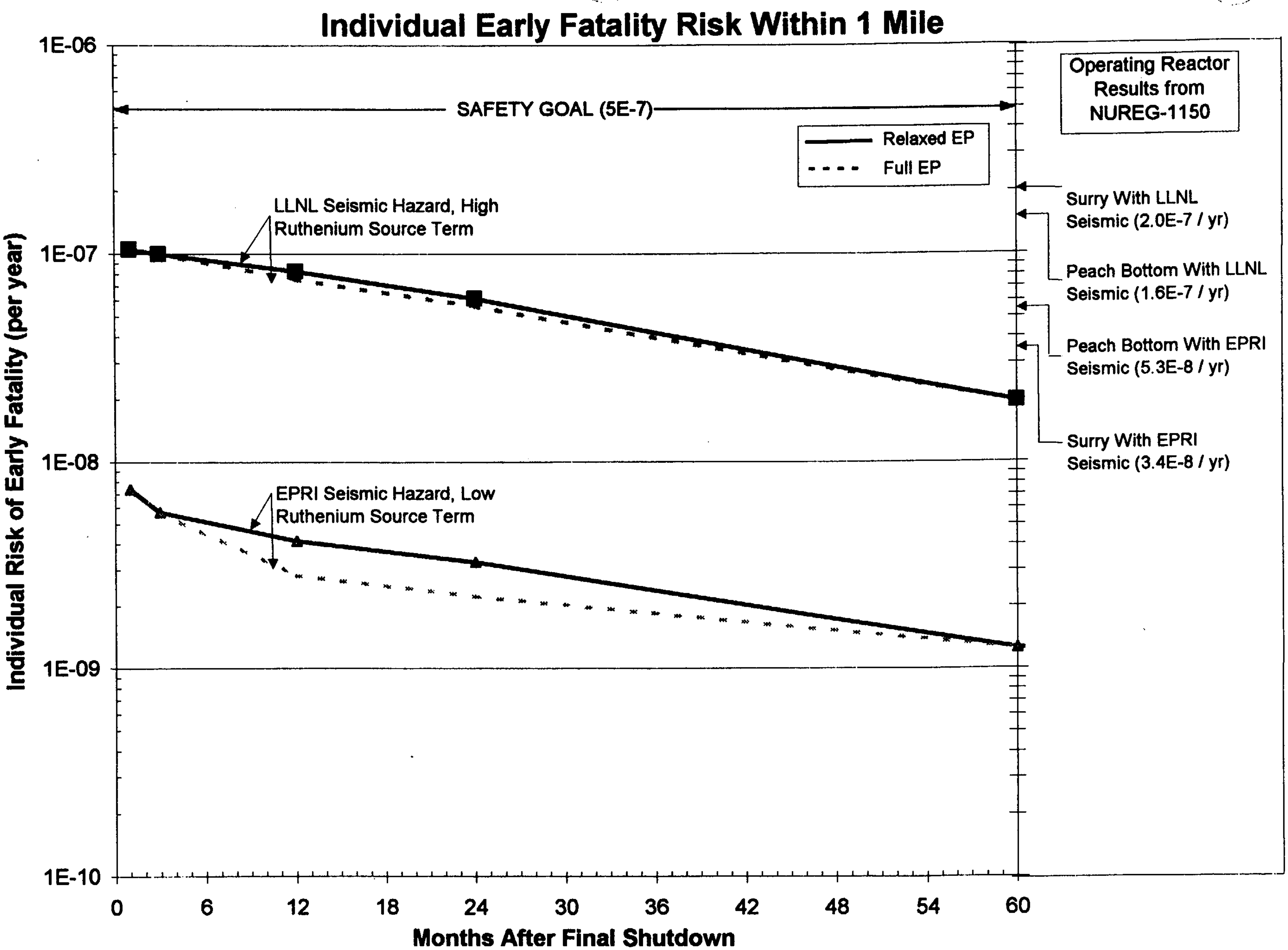

Sensitivity of Societal (Person-rem) Risk to Emergency Planning
-- Cask Drop Event
(Conditional upon: High Ruthenium Source Term)
Person-rem Within 50 Miles (per year)
5
4
3
2
1
0
Cask Drop Contribution
with Late Evacuation
Cask Drop Contribution
with Full EP
Cask Drop Contribution
with Relaxed EP
Cask Drop Contribution
with Early Evacuation
0
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown

Figure 3.7-6



Individual Early Fatality Risk Within 1 Mile
1E-06
1E-07
1E-08
1E-09
1E-10
Individual Risk of Early Fatality (per year)
SAFETY GOAL (5E-7)
Relaxed EP
Full EP
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Operating Reactor Results from NUREG-1150
Surry With LLNL Seismic (2.0E-7 / yr)
Peach Bottom With LLNL Seismic (1.6E-7 / yr)
Peach Bottom With EPRI Seismic (5.3E-8 / yr)
Surry With EPRI Seismic (3.4E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure ES-1

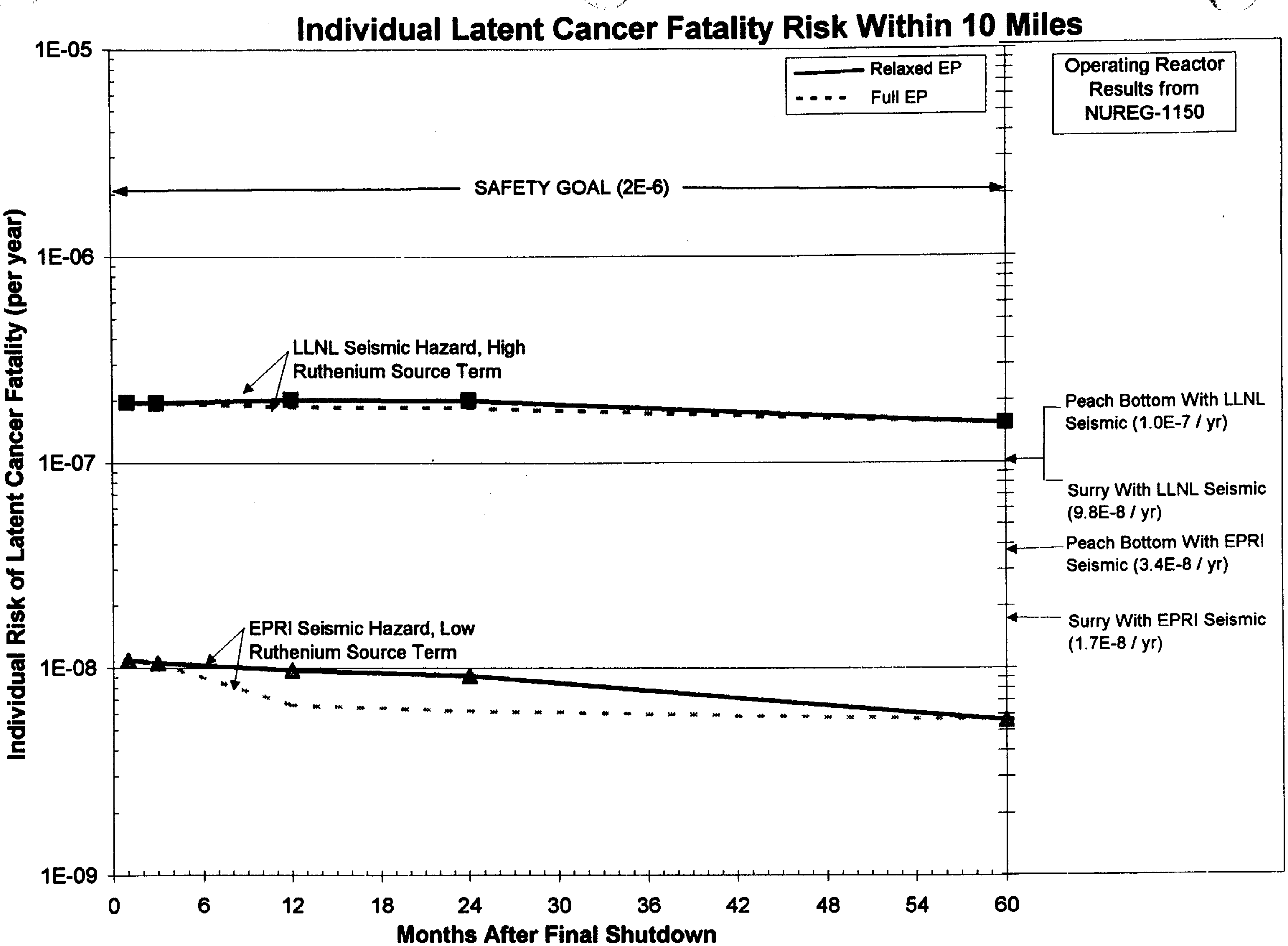
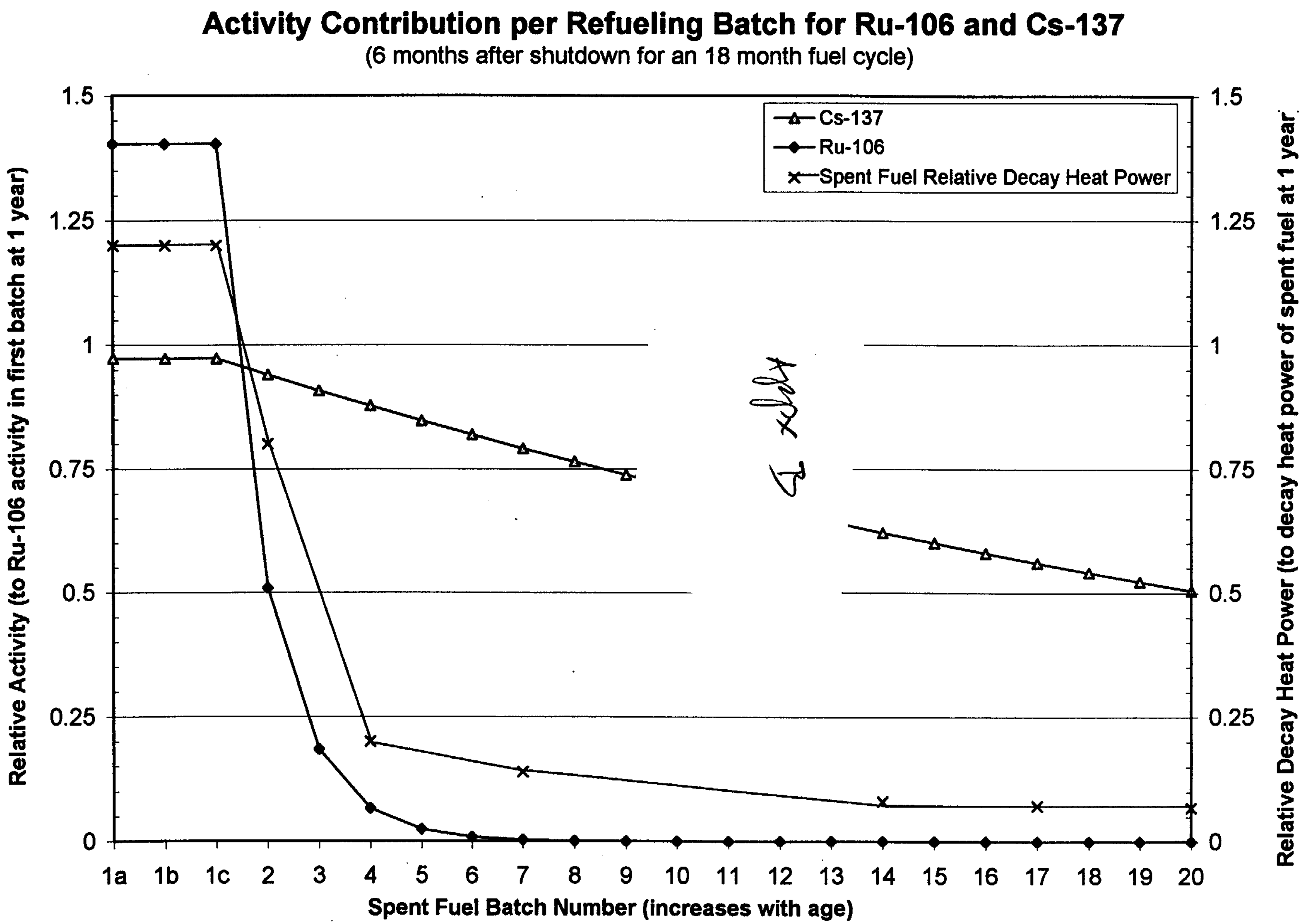

Individual Latent Cancer Fatality Risk Within 10 Miles
Individual Risk of Latent Cancer Fatality (per year)
1E-05
1E-06
1E-07
1E-08
1E-09
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
SAFETY GOAL (2E-6)
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Peach Bottom With LLNL Seismic (1.0E-7 / yr)
Surry With LLNL Seismic (9.8E-8 / yr)
Peach Bottom With EPRI Seismic (3.4E-8 / yr)
Surry With EPRI Seismic (1.7E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure ES-2

9-21-00

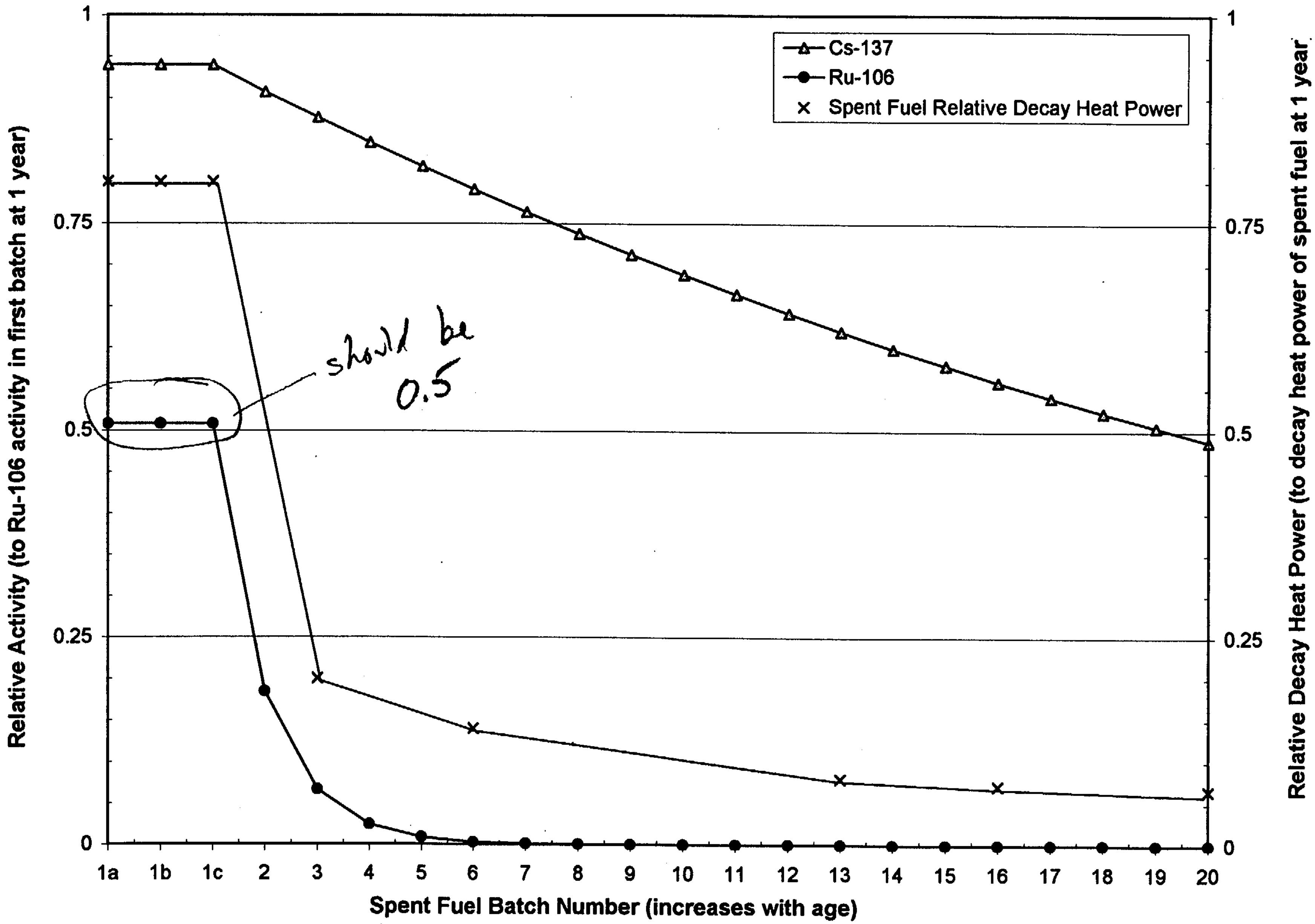


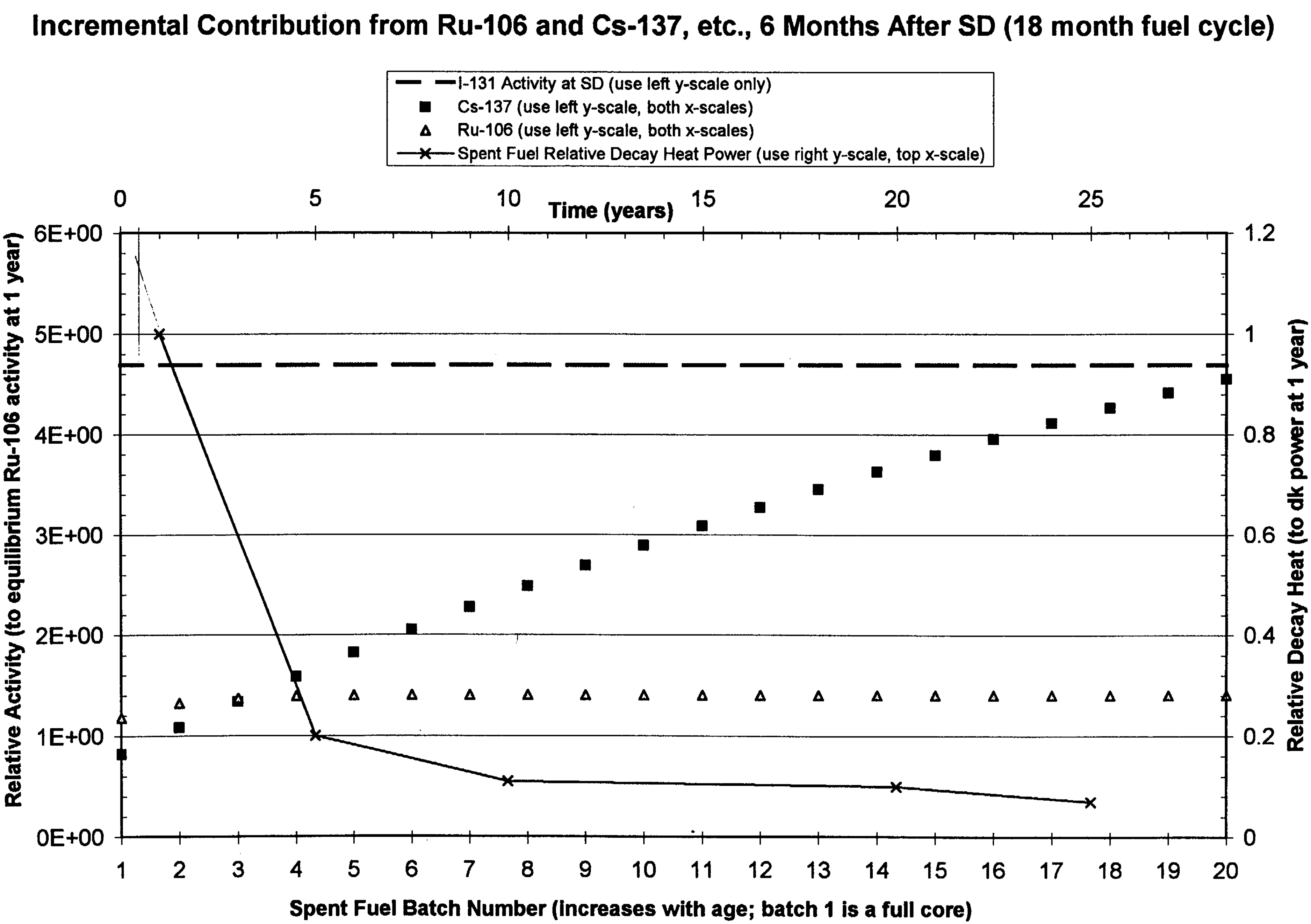
Activity Contribution per Refueling Batch for Ru-106 and Cs-137
(6 months after shutdown for an 18 month fuel cycle)
Cs-137
Ru-106
Spent Fuel Relative Decay Heat Power
Relative Activity (to Ru-106 activity in first batch at 1 year)
Relative Decay Heat Power (to decay heat power of spent fuel at 1 year
Spent Fuel Batch Number (increases with age)
1.5
1.25
1
0.75
0.5
0.25
0
1a
1b
1c
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20

9-21-00

Activity Contribution per Refueling Batch for Ru-106 and Cs-137

(2 years after shutdown for an 18 month fuel cycle)





Incremental Contribution from Ru-106 and Cs-137, etc., 6 Months After SD (18 month fuel cycle)
I-131 Activity at SD (use left y-scale only)
Cs-137 (use left y-scale, both x-scales)
Ru-106 (use left y-scale, both x-scales)
Spent Fuel Relative Decay Heat Power (use right y-scale, top x-scale)
Time (years)
0
5
10
15
20
25
Relative Activity (to equilibrium Ru-106 activity at 1 year)
6E+00
5E+00
4E+00
3E+00
2E+00
1E+00
0E+00
Relative Decay Heat (to dk power at 1 year)
1.2
1
0.8
0.6
0.4
0.2
0
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20
Spent Fuel Batch Number (increases with age; batch 1 is a full core)

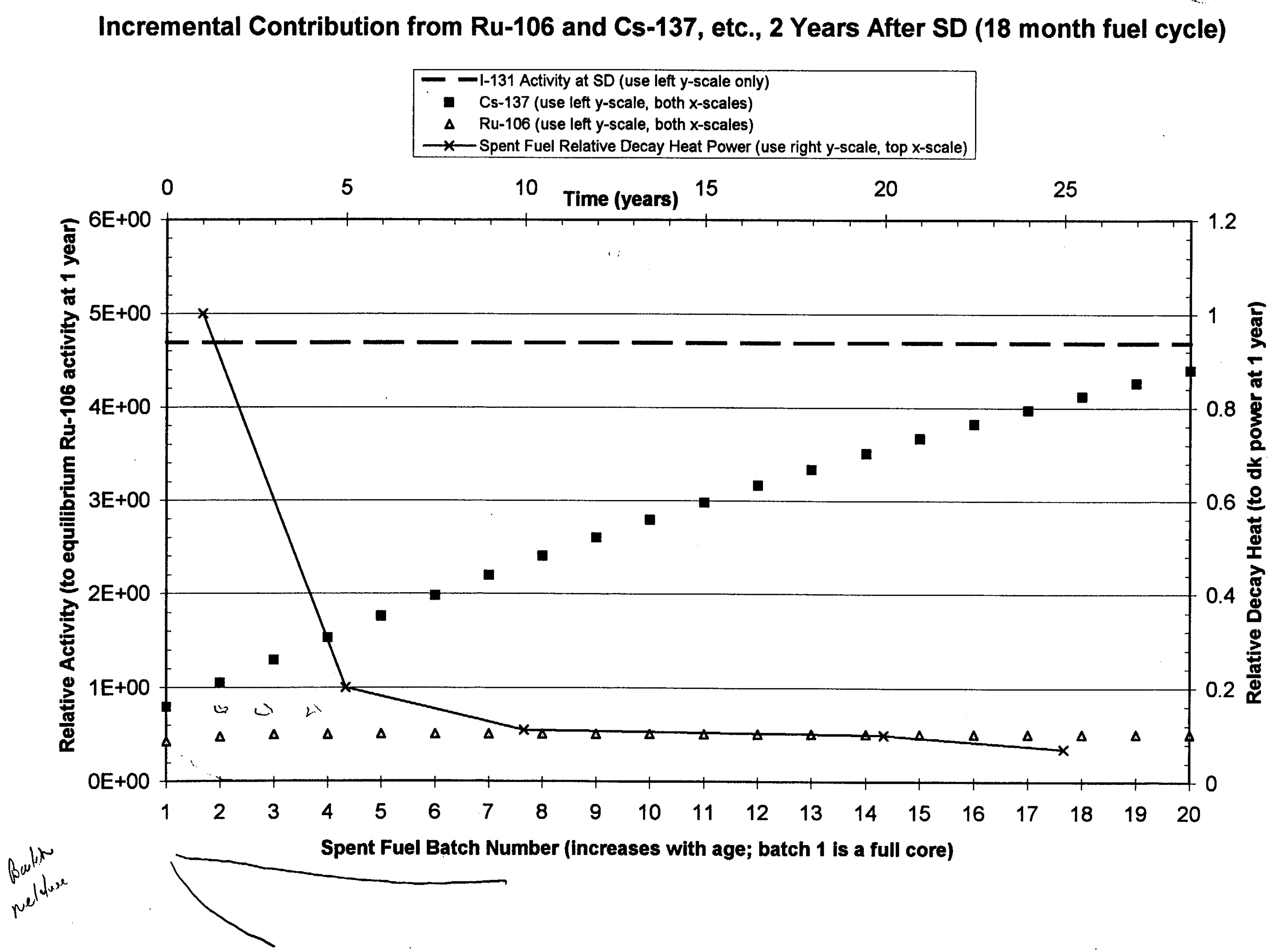

Incremental Contribution from Ru-106 and Cs-137, etc., 2 Years After SD (18 month fuel cycle)
I-131 Activity at SD (use left y-scale only)
Cs-137 (use left y-scale, both x-scales)
Ru-106 (use left y-scale, both x-scales)
Spent Fuel Relative Decay Heat Power (use right y-scale, top x-scale)
Time (years)
0
5
10
15
20
25
6E+00
5E+00
4E+00
3E+00
2E+00
1E+00
0E+00
1.2
1
0.8
0.6
0.4
0.2
0
Relative Activity (to equilibrium Ru-106 activity at 1 year)
Relative Decay Heat (to dk power at 1 year)
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
Spent Fuel Batch Number (increases with age; batch 1 is a full core)

Table C-1

Fission product inventories (Ci/MWe)

A/A₀

Fission product	Release Fraction	t 1/2	Shutdown Inventory (Ci/MWe)	40 days	6 months	1 Yr	3 Yr
Kr-85		10.72Y	560	1			
~~Kr-85m~~		~~4.48H~~	24,000	–			
~~Kr-87~~		~~76.3M~~	47,000	–			
~~Kr-88~~		~~2.84H~~	68,000	–			
Sr-89		50.6 D	94,000	.578)	.085	–	
Sr-90		28.6Y	3,700	1			
~~Sr-91~~		~~9.5H~~	110,000				
Y-91		58.5 D	120,000	.623	.12	–	
~~Mo-99~~		~~66H~~	~~160,000~~				
Ru-103		39.4 D	110,000	.494	.042	–	
Ru-106		368.2 D	25,000	.93	.713	.50	
Te-129m		33.6 D	5,300	.47	.02	–	
Te-131m		30H	13,000	–			
Te-132		78H	120,000	–			
Sb-127		3.85D	6,100	–			
~~Sb-129~~		~~4.4H~~	33,000	–			
I-131		8.0D	85,000	.03	–		
I-132		2.3H	120,000	–			
I-133		20.8H	170,000	–			
I-134		52.6M	190,000	–			
I-135		6.61 H	150,000	–			
Xe-131m		11.84 D	1,000	.1	–		
Xe-133		5.2 D	170,000	–			
Xe-133m		2.19 D	6,000	–			
Xe-135		9.1 H	34,000	–			
~~Xe-138~~		~~14.3M~~	170,000	–			
Cs-134		2.0 Y	7,500	.96	.84	.71	
Cs-136		13.16 D	3,000	–			
Cs-137		30.2 Y	4,700	1.0	.99	.98	
Ba-140		12.8D	160,000	.11			
La-140		40.22H	160,000	–			
Ce-144		284 D	85,000	.9	.64	.41	
Np-239		2.36D	1.64E+6	–	–	–	

why is CE-144 not important? →

Source: WASH-1400

For end of cycle core, only the fission products with half lives greater than 1/2 hour.

September 5, 2000

1. Below is a table containing the dose conversion factors you requested. MACCS calculates the early fatality risk as a combination of the dose to the lungs and red marrow.

2. Iodine is important for reactor accidents, because of its high inventory in the core and its high thyroid dose conversion factor. Table 4.1 of NUREG/CR-4982 shows the following inventories (in Curies) for an equilibrium core for Millstone 1:

I-131	4.74E7
Ru-106	2.48E7
Cs-137	5.84E6

3. One of your health physicists (e.g., Steve LaVie) might be able to provide further insight into the importance of iodine.

Dose Conversion Factors for I-131, Ru-106, and Cs-137*

	organ	cloud-shine (Sv sec/ Bq m^3)	ground-shine (Sv sec/ Bq m^2)	inhalation/ acute (Sv/Bq)	inhalation/ chronic (Sv/Bq)	ingestion (Sv/Bq)
I-131	**lungs**	1.41E-14	2.97E-16	**4.54E-10**	6.57E-10	1.02E-10
	red marrow	1.45E-14	3.06E-16	3.52E-11	6.26E-11	9.44E-11
Ru-106	**lungs**	7.90E-15	1.58E-16	**2.09E-08**	1.04E-06	1.44E-09
	red marrow	8.05E-15	1.61E-16	8.74E-11	1.77E-09	1.48E-09
Cs-137	**lungs**	2.18E-14	4.35E-16	**8.29E-10**	8.80E-09	1.27E-08
	red marrow	2.22E-14	4.41E-16	5.63E-10	8.30E-09	1.32E-08
Ratio of Ru-106 to Cs-137	lungs	.4	.4	**25**	118	.1
	red marrow	.4	.4	.2	.2	.1

*The dose conversion factors are from the MACCS input file DOSDATA.INP.

Acute Lung

$\frac{\text{Ru-106}}{\text{I-131}} = \frac{2.1E-08}{4.5E-10} = 47$

$\frac{1.04E-06}{6.57E-10}$

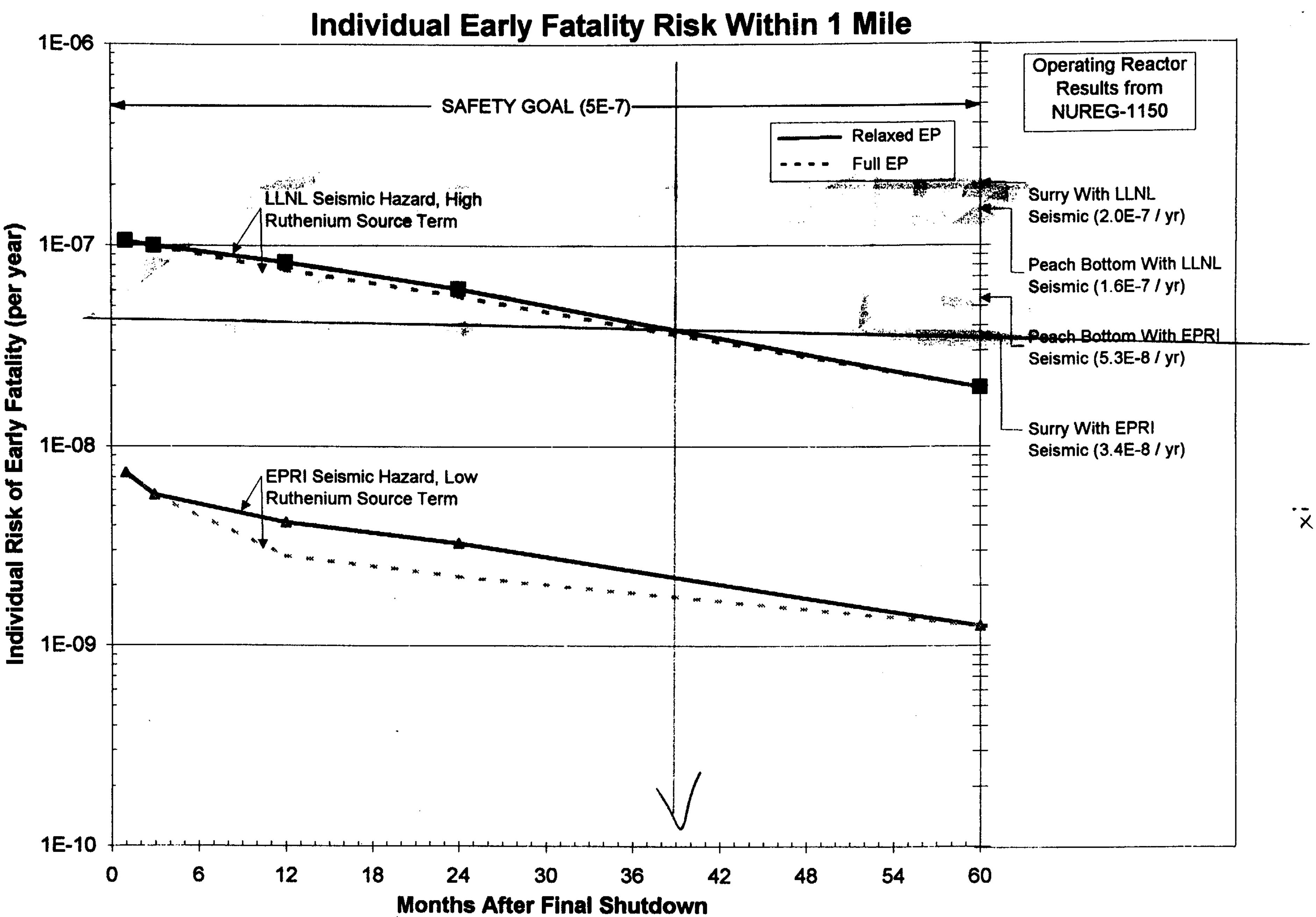
$\frac{10.4E-7}{6.57E-10}$

$\frac{17.7E-10}{6.3E-11}$

5 = 12.5
1 = 3.7

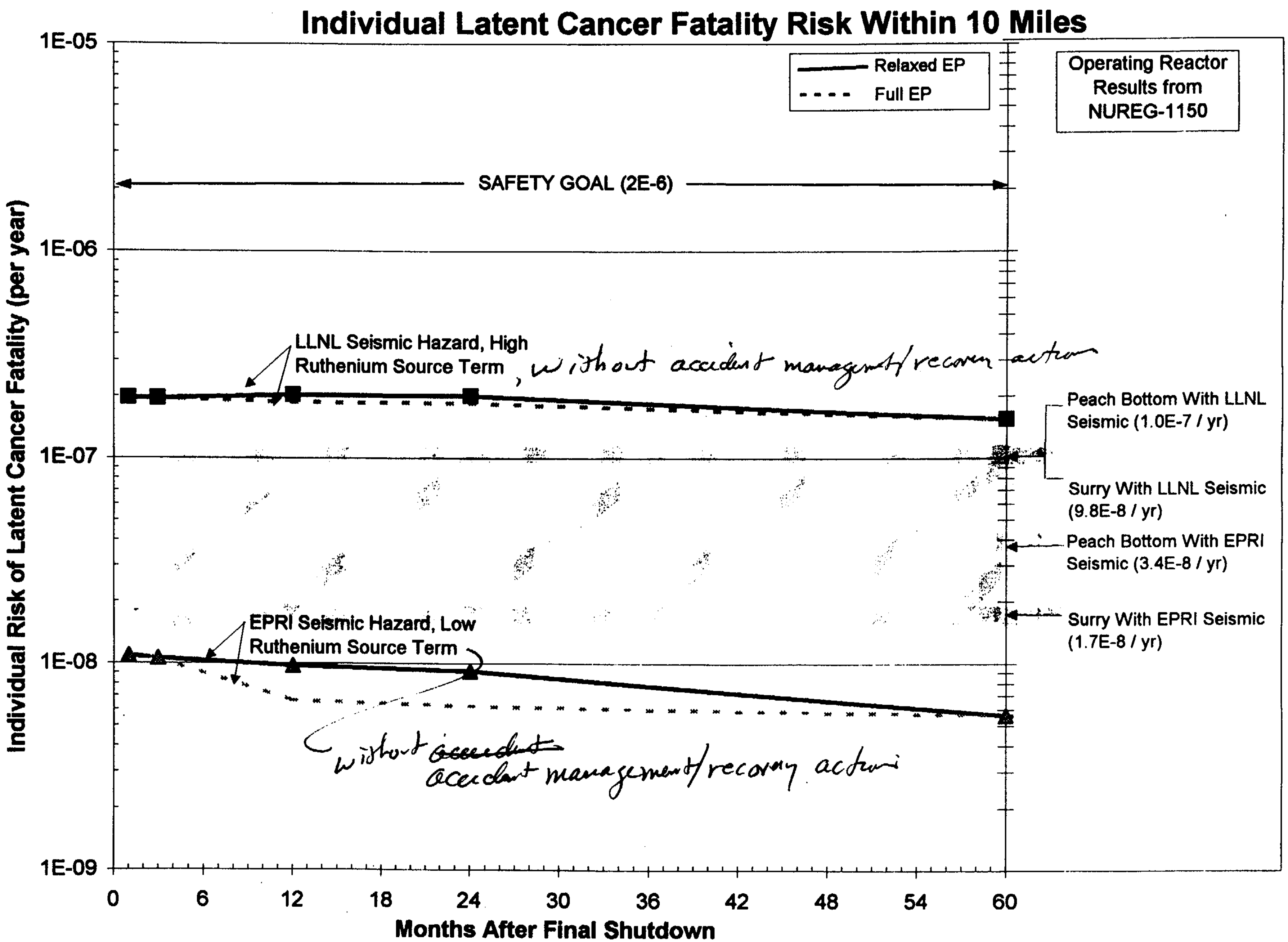
NUREG - 1150 (1987)

	Early fatality/yr	Latent fatality/yr
Seq	3.5×10^{-6} to 3.1×10^{-4}	6.3×10^{-2} to 5.1×10^{-1}
Zion	3.5×10^{-6} to 8.3×10^{-4}	3.0×10^{-3} to 1.2×10^{-1}
Surry	3.7×10^{-8} to 1.4×10^{-5}	8.0×10^{-4} to 3.0×10^{-2}
PB	8.5×10^{-9} to 8.1×10^{-6}	6.5×10^{-4} to 1.1×10^{-1}
GG	1.4×10^{-8} to 6.5×10^{-6}	1.4×10^{-3} to 5.2×10^{-2}



Individual Early Fatality Risk Within 1 Mile
1E-06
1E-07
1E-08
1E-09
1E-10
Individual Risk of Early Fatality (per year)
SAFETY GOAL (5E-7)
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Surry With LLNL Seismic (2.0E-7 / yr)
Peach Bottom With LLNL Seismic (1.6E-7 / yr)
Peach Bottom With EPRI Seismic (5.3E-8 / yr)
Surry With EPRI Seismic (3.4E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

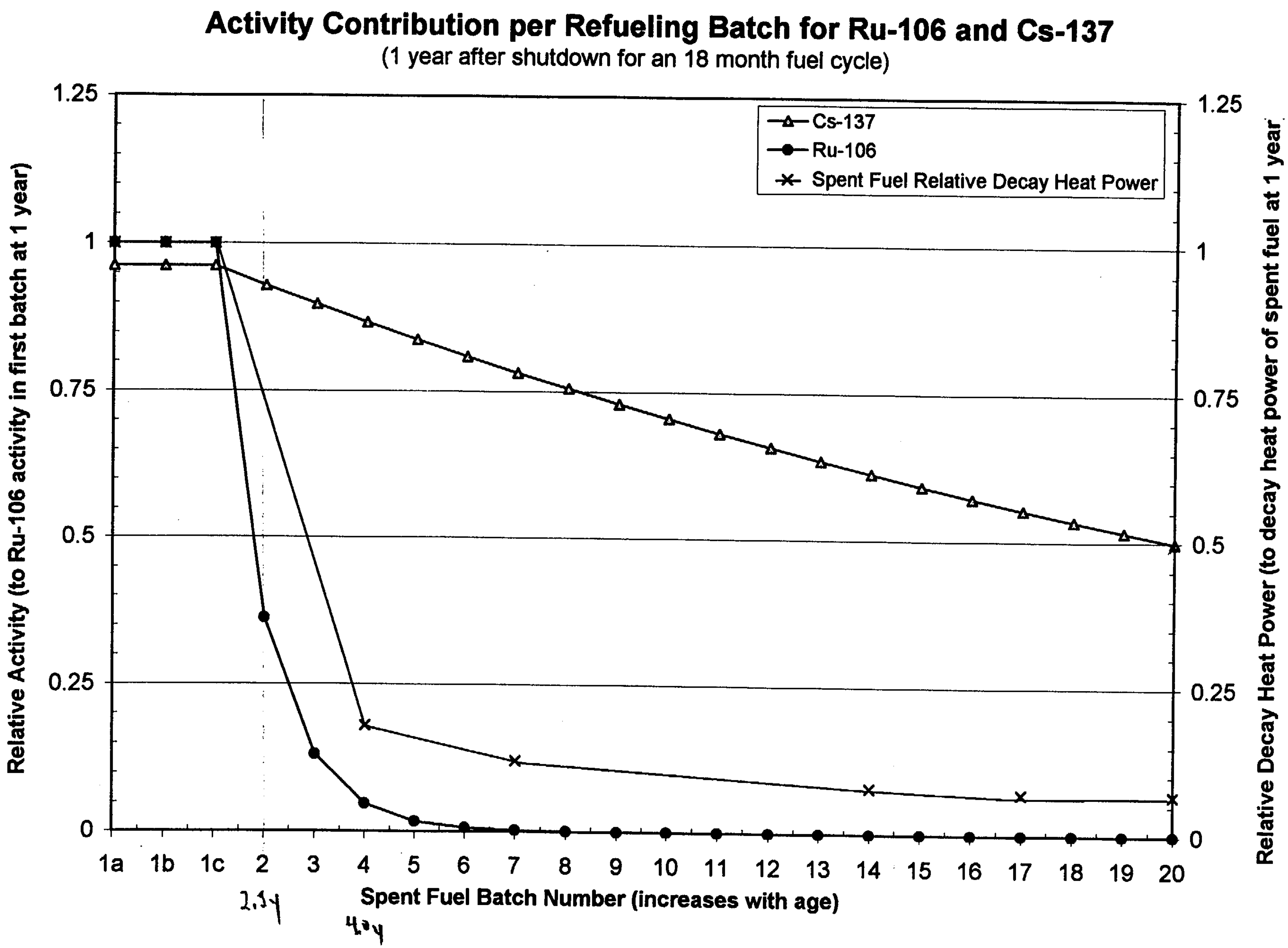
Figure ES-1



Individual Latent Cancer Fatality Risk Within 10 Miles
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
SAFETY GOAL (2E-6)
LLNL Seismic Hazard, High Ruthenium Source Term
Peach Bottom With LLNL Seismic (1.0E-7 / yr)
Surry With LLNL Seismic (9.8E-8 / yr)
Peach Bottom With EPRI Seismic (3.4E-8 / yr)
Surry With EPRI Seismic (1.7E-8 / yr)
EPRI Seismic Hazard, Low Ruthenium Source Term
Individual Risk of Latent Cancer Fatality (per year)
1E-05
1E-06
1E-07
1E-08
1E-09
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure ES-2

9-21-00


Activity Contribution per Refueling Batch for Ru-106 and Cs-137
(1 year after shutdown for an 18 month fuel cycle)
Cs-137
Ru-106
Spent Fuel Relative Decay Heat Power
Relative Activity (to Ru-106 activity in first batch at 1 year)
Relative Decay Heat Power (to decay heat power of spent fuel at 1 year)
Spent Fuel Batch Number (increases with age)
1.25
1
0.75
0.5
0.25
0
1a
1b
1c
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
2.5 y
4.0 y

consequences.xls

SPENT FUEL POOL ACCIDENT CONSEQUENCES

09/19/2000

Event Freq ~ 1×10^{-6} per year

	60 Mwd +Oxidation+full aircooling					
	Early Fatalities		Time Delay		LC at 100 miles	
	Evacuation Model		PWR	BWR	Evacuation Model	
Decay Time	Early	Late			Early	Late
30 days	[redacted]					
30 days	7	192	1	3	15400	21100
60 days	[redacted]					
60 days	4	162	2	5	14300	20000
1 year	[redacted]					
1 year	1	77	5	7	11500	17400
2 years	[redacted]					
2 years	0.1	19	9	17	9480	15400
5 years	[redacted]					
5 years	0.02	1	33?	inf	7620	12600
10 years	[redacted]					
10 years	0.01	-		inf	6490	11400
	Source Term:					
	[redacted]					
	Red = Ruthenium Rich Upper Bound					