4. Recent experience at the DOE Savannah River site suggests frequencies of dissolver seal failure as much as 1,000 times higher.

5. Recent experience at the DOE Savannah River Site suggests frequencies of fire in low level waste and fuel assembly drop as much as 100 times higher.

6. The iodine-129 part of Table C.81 is suspect. I-129 has a half-life of 17 million years and, correspondingly, specific activity of 1.8E-4 Ci/g. I-129 emits a 150 kev beta and, 9% of the time, a 40 kev gamma, both significantly lower energies than the corresponding values for I-131. The biological half-life of I-129 in the thyroid is 120 days. The dose conversion factor for I-129 would be approximately 0.5 rem/micro-Ci administered to the thyroid. The values given in the table for I-129 releases and the corresponding thyroid doses seem inconsistent with each other and with the properties of I-129 given above. The thyroid is relatively radio-resistant and thyroid cancer relatively treatable; the mortality risk factor for the thyroid is 5.0E-6/person-rem (i.e., one fatality per 2.0E+5 person-rem exposure to the thyroid).

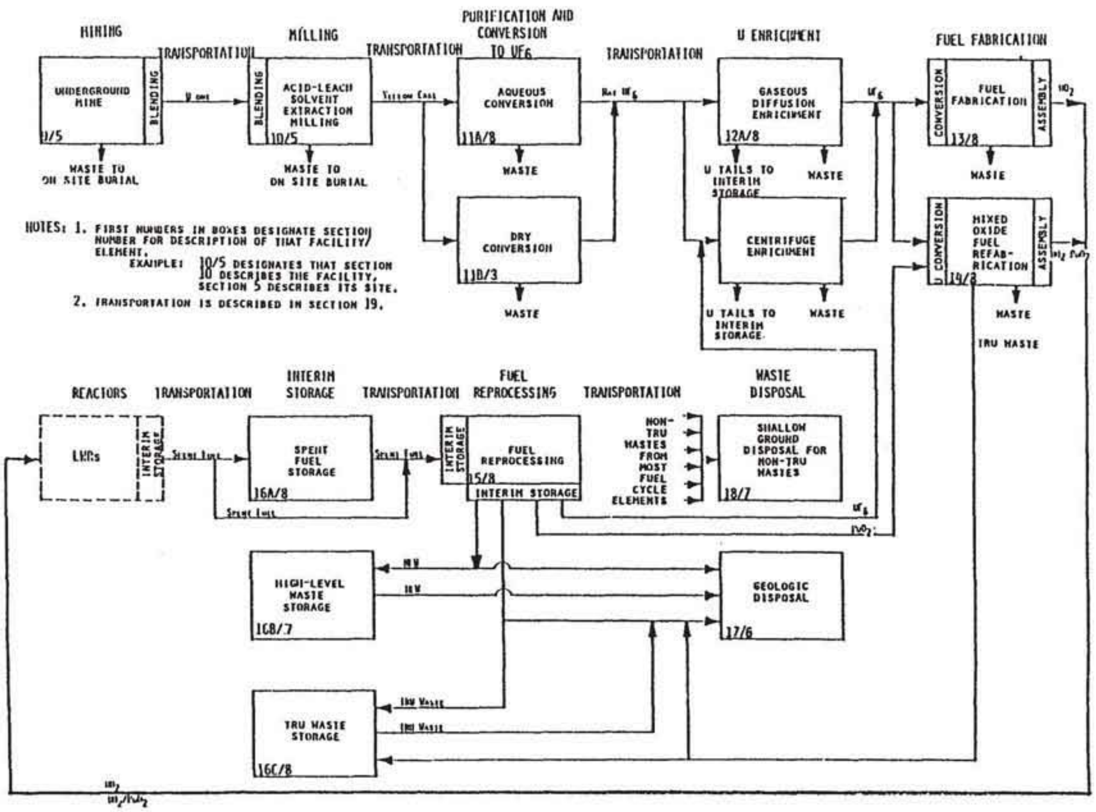

Figure C.1 Uranium process flow among fuel cycle facilities

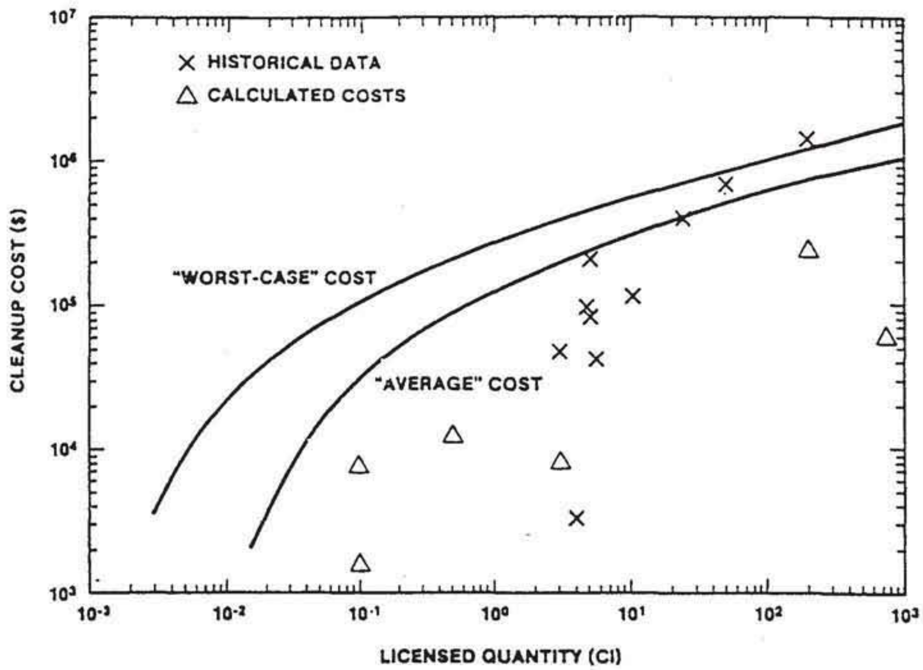

Figure C.2 Cleanup cost as a function of licensed radionuclide quantity for non-reactor nuclear material licensees (Ostmeyer and Skinner 1987, Figure 4.3)

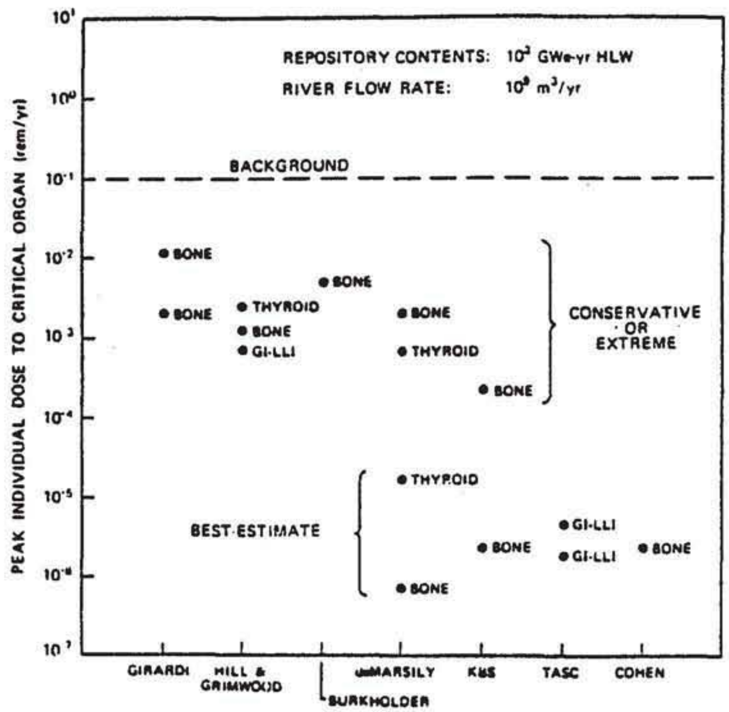

Figure C.3 Normalized peak individual doses for reviewed studies of geologic waste disposal postclosure period (TASC 1979)

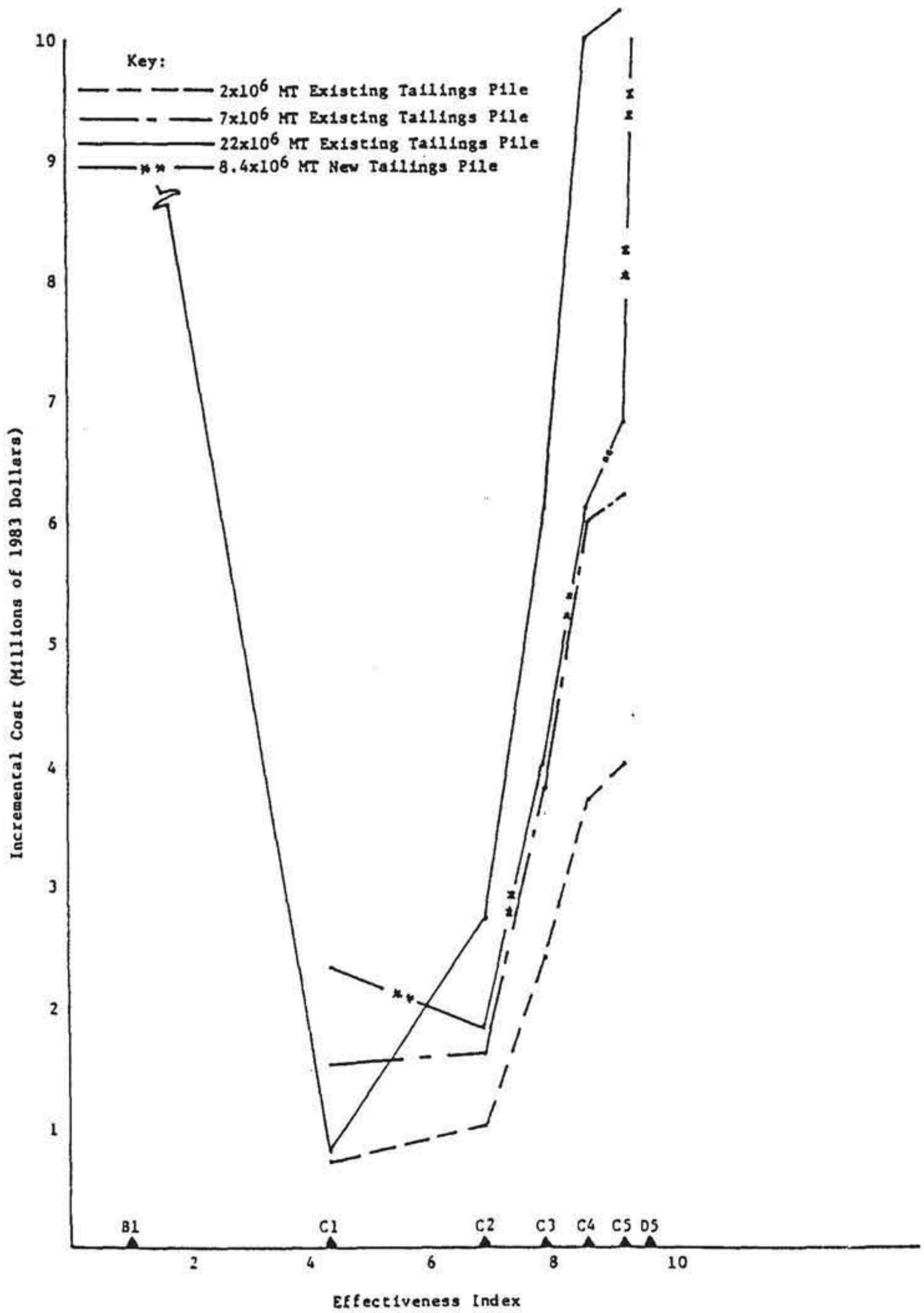

Figure C.4 Incremental cost of alternative control methods for uranium mill tailings (EPA 1983, Figure 4.6)

Table C.S.1 Summary description of representative uranium fuel cycle facilities (Schneider et al. 1982, Table 2.2)

	Fuel Cycle Element						
			Conversion		Enrichment		
Item	Mining (Section 9)	Milling (Section 10)	Aqueous (Section 11.A)	Dry (Section 11.B)	Gaseous Diffusion (Section 12.A)	Gas Centrifuge (Section 12.B)	Fuel Fabrication (Section 13)
Facility Based On	Ambrosia Lake	Highland	Sequoyah	Metropolis	Stand-alone, combination of 3 US plants	Conceptual stand-alone	Westinghouse/ Columbia, SC
Major Process	Underground room-and-pillar, cutting, blasting	Acid-leach, solvent extn., precipitation	Solvent extraction hydrofluorination, fluorination	Hydrofluorination, fluorination, fractional distillation	Gaseous diffusion, cold trapping, waste recovery	Gas centrifuge, cold trapping, waste recovery	ADU process, calcination, compaction, sintering, waste recovery
Capacity							
Feed/Mg/yr	Ore Vein/varies	Ore/6.6E5	Yellowcake/1.2E4	Yellowcake/7400	UF_6/1.3E4	UF_6/1.3E4	UF_6/2100
Product/Mg/yr[a]	Ore/1.3E6	Yellowcake/930	UF_6/9100	UF_6/6800	UF_6/1400	UF_6/1400	Fuel assemblies/1460
GWd_e Equivalent/yr[b]	3300	1600	15,400	11,500	15,500	15,500	16,000
Operating hr/d and d/yr	16/312	24/365	24/365	24/300	24/365	24/365	24/350
Total Staff	1160	92	155	NA	1400	2150	1850
Contact Operations	~All; most is not direct contact	~All; most is not direct contact	~All; most is not direct contact	~All; most is not direct contact	~All maintenance	All maintenance	Receiving, rod and element assemblage, maintenance
Remote Operations	None	None	None	None	Most operations	Most operations	Chemical processing, scrap recovery (not shielding)
Alternative Concepts	Open-pit, in-situ (Solution)	Alkaline leach, ion exchange	None	None	U Laser, UF_6 Laser, U plasma ion	U Laser, UF_6 Laser, U plasma ion	Fluidized bed, powder front-end

Table C.S.1 (Continued)

Item	MOX Fuel Refabrication (Section 14)	Fuel Reprocessing (Section 15)	Waste Storage: Spent Fuel (Section 16.A)	Waste Storage: High-Level Waste (Section 16.B)	Waste Storage: TRU Waste (Section 16.C)	Geologic Waste Disposal (Section 17)	Shallow Land Waste Disposal (Section 18)	Transportation (Section 19)
	Fuel Cycle Element							
Facility	Conceptual Westinghouse Recycle Fuels Plant	Barnwell with conceptual additions	Conceptual, standalone, water basin	Conceptual, standalone, dry-well	Conceptual, standalone, vault and outside pad	Conceptual NWTS disposal repository in salt formation	Conceptual standalone	State-of-the-art; specific to each material
Major Process	Powder blending, compaction, sintering, waste recovery	PUREX, UF_6 and Pu conversion, HLW vitrification	Wet unloading and storage, ion exchange, heat exchange	Wet unloading, encapsulation, dry-well storage	Solids handling (shielded and unshielded), above grade storage	Solids handling, underground blasting, machine excavation	Burial in below-grade trenches	Truck and rail transport cross-country
Capacity								
Feed/Mg/yr	UO_2; PuO_2/436; 18	Spent fuel/1500	Spent fuel/500 HM	Solidified HLW/320	TRU-waste/50,000	Spent fuel, HLW TRU waste/3900 HM equiv.	LLW, ILW/50,000 m^3	Individual shipping capacity/container for each material
Product/Mg/yr[a]	MOX assemblies/ 400 HM	U/1410; Pu/15	NAp	NAp	NAp	NAp	NAp	
GWd_e Equivalent/yr[b]	4400	15,500	5500	15,500	27,600	43,000	29,000	--
Operating hr/d and d/yr	24/350	24/300	24/365	24/365	20/300	24/365	8/250	Varies
Total Staff	260	500	~50	~100	28	259	70	1-2/shipment
Contact Operations	~All; most is not direct contact	Receiving, some maintenance	Receiving, maintenance	Receiving, maintenance	All CH-TRU ~1/2 RH-TRU	Receiving, ~All CH-TRU ~1/2 RH-TRU	~All; most is not direct contact	Direct contact with containers
Remote Operations	Pellet preparation, scrap recovery	Most operations	Fuel unloading and handling, waste-treatment	Most operations	~1/2 RH-TRU	~1/2 RH-TRU ~All spent fuel, HLW	None	Remote unloading for most materials
Alternative Concepts	Co-precipitation, remote maintenance	Many variations of PUREX, Others	Dry well, cask, tunnel rack, vault consolidation	Dry well, cask, tunnel rack, vault	Below-grade, mine storage, berms	Basalt, granite, tuff; self-shielded packages	Onsite processing, various burial variations	Variations of hardware for most containers

NA = not available
NAp = not applicable
(a) As U and/or Pu except from mining.
(b) Based on product rate to fuel fabrication and 11,000 MWd_e/MgHM.

Table C.1 Frequency of contamination incidents for non-reactor nuclear material licensees (Ostmeyer and Skinner 1987, Table 3.1)

	Application/use class	Number of Incidents[a]	Number of Licenses	Frequency (incidents licensed-activity-yr)
I)	Research/teaching & Diagnostic/therapeutic	7	5100	0.00023
II)	Measurement/calibration & irradiation	6	5715	0.00018
III)	Manufacture/distribution	8	510	0.0026
IV)	Service organizations/ waste processing/storage	0	49	---
V)	Source and Special Nuclear Material Fuel cycle	6	72	0.014

(a) For a six year reporting period.

Table C.2 Incident cleanup cost by material quantity class for non-reactor nuclear material licensees (Ostmeyer and Skinner 1987, Table 4.1)

Licensed Material Quantity	Incident Cleanup Cost ($) LQR Case	Average
10 mCi - 0.1 Ci	70,000	15,000
0.1 Ci - 1.0 Ci	200,000	75,000
1.0 Ci - 10 Ci	450,000	230,000
10 Ci - 100 Ci	800,000	500,000
100 Ci - 1000 Ci	1,500,000	900,000

Table C.3 Economic risk as a function of material application/use and licensed curie quantity for non-reactor nuclear material licensees (Ostmeyer and Skinner 1987, Table 5.1)

Application/Use Class	Economic Risk ($/licensed activity/yr) by Licensed Quantity[a]				
	0.01 Ci- 0.1 Ci	0.1 Ci- 1.0 Ci	1.0 Ci- 10 Ci	10 Ci- 100 Ci	100 Ci- 1000 Ci
I) Research/Teaching/ Experimentation and Diagnostic/Therapeutic	4	29	50	120	200
II) Measurement/Calibration Irradiation	3	20	40	90	160
III) Manufacture/Distribution	40	230	520	1,300	2,300

(a) Risk is given by the product of incident frequency and average incident cost.

Table C.4 Summary of economic risk at a reference uranium mill (Philbin et al. 1990, Table 4.1)

Incident Scenario	Consequence Description	Cleanup Cost [uncertainty]	Frequency per year [uncertainty]	Economic Risk (per year) [uncertainty]
Minor facility releases	Hundreds of g to tens of kg U released. Confined to small areas in plant.	$1100 [$900-$1,400]	0.0077 [0.0048-0.014]	$8 [$5 - $15]
Solvent Extraction Fire	Up to several kg U released. Cleanup limited to process area.	$370,000 [$300,000-$460,000]	0.0031 [0.0014-0.0082]	$1100 [$460-$2900]
Fire/Explosion in Yellocake Dryer	Up to several Kg U released. Cleanup limited to process area.	$500,000 [$400,000-$630,000]	0.0031 [0.0014-0.0082]	$1600 [$620-$3900]
Major Facility Fire	Cleanup of main process area and downwind facility area (22.5° sector).	$1.5M [$1.2M-$1.9M]	0.00020 [0.00013-0.00040]	$300 [$160-$550]
Retention Pond Failure with Slurry Release	8 x 10^6 lbs solids released. Stabilize pond and spill areas and clean up spill.	$2.5M [$2M-$3.1M]	0.023 [0.017-0.033]	$58,000 [$39,000-$86,000]
Slurry Release from Distribution Pipe	2.2 x 10^5 lbs solids released on site. Stabilize spill area. Clean up spill area.	$69,000 [$55,000-$86,000]	0.0062 [0.0037-0.012]	$430 [$230-$800]
Tornado	Thousands of kg U released - Clean up buildings and downwind site area (45° sector).	$3M [$2.4M-$3.8M]	0.000080 [0.000025-0.00025]	$240 [$70-$780]
Transportation	Entire load of ore spilled or 1/3 yellowcake drums spill. Area cleanup	$300,000 [$225,000-$375,000]	0.0031 [0.0014-0.0082]	$930 [$370-$2300]
			TOTAL FACILITY ECONOMIC RISK	$63,000 [$43,000-$91,000]

Table C.5 Summary of economic risk at a reference uranium hexafluoride conversion plant (Philbin et al. 1990, Table 4.2)

Incident Scenario	Consequence Description	Cleanup Cost [uncertainty]	Frequency per year [uncertainty]	Economic Risk (per year) [uncertainty]
Minor facility release	Release of hundreds of grams to tens of kg U. Cleanup limited to immediate area of the release.	$1,100 [$900-$1,400]	0.13 [0.081-0.22]	$140 [$80-$250]
Uranyl Nitrate Evaporator Explosion	Release of several kg of U. Cleanup of process building.	$730,000 [$580,000-$910,000]	0.00032 [0.00010-0.0010]	$230 [$70-$750]
Hydrogen explosion during reduction	Release of several kg of U. Cleanup of process area.	$730,000 [$580,000-$910,000]	0.0070 [0.0010-0.050]	$5,100 [$710-$37,000]
Solvent extraction fire	Several hundred kg U released - Clean up solvent extraction building.	$81,000 [$65,000-$100,000]	0.00040 [0.00013-0.0013]	$30 [$10-$100]
Release from UF_6 cylinder	Release of up to 2500 kg of U. Clean up immediate area.	$1.2M [$0.96M-$1.5M]	0.021 [0.011-0.081]	$25,000 [$9,100-$70,000]
Distillation Valve Rupture	Release of tens of kg of U. Clean up immediate area.	$130,000 [$100,000-$160,000]	0.050 [0.016-0.16]	$6,500 [$2,000-$21,000]
Waste Pond Release	7 x 10^5 lbs solids released. Stabilize pond and spill area and clean up spill.	$230,000 [$180,000-$290,000]	0.056 [0.029-0.22]	$13,000 [$4,600-$36,000]
Transportation	Small rupture of UF_6 cylinder. Hundred of kg of U released. Cleanup of area.	$400,000 [$320,000-$500,000]	0.0031 [0.0014-0.0082]	$1,200 [$500-$3,100]
Tornado	Thousands of kg U dispersed. Cleanup of 45° sector of downwind site area.	$1.9M [$1.5M-$2.4M]	0.0023 [0.00074-0.0074]	$4,400 [$1,400-$14,000]
			TOTAL FACILITY ECONOMIC RISK	$56,000 [$20,000-$109,000]

Table C.6 Summary of economic risk at a reference uranium fuel fabrication facility (Philbin et al. 1990, Table 4.3)

Incident Scenario	Consequence Description	Cleanup Cost [uncertainty]	Frequency per year [uncertainty]	Economic Risk (per year) [uncertainty]
Minor Facility Release	Release of hundreds of gms to tens of kg U. Confined to small areas in plant.	$3,500 [$2,800 - $4,400]	0.21 [0.15 - 0.32]	$740 [$470-$1,100]
Large Spills due to accidents or natural phenomena	800m³ waste solution, 24 Ci solids, 40000 m² surface contaminated.	$1.0M [$0.80M-$1.3M]	0.024 [0.015 - 0.044]	$24,000 [$13,000-$43,000]
Transportation accident	Trailer overturns; No contamination outside trailer.	$10,000 [$7,500 - 13,000]	0.0028 [0.0026 - 0.0030]	$28 [$22-$35]
Explosion	Rotary Kiln. Batch of 100 kg U, 1kg released to environment (outside), 1/3 of main building contaminated.	$3.9M [$3.1M - $4.9M]	0.01 [0.002 - 0.05]	$39,000 [$7,700-$200,000]
Major Fire	Decontamination of entire main building is required.	11M [$8.8M - $14M]	0.00021 [0.00012 - 0.00051]	$2,300 [$1,100-$4,900]
Criticality	10^{19} fissions; 8 hr duration. 1/3 of main building contaminated.	$3.9M [$2.9M - $4.9M]	0.0033 [0.00050 - 0.011]	$13,000 [$2,700-$61,000]
Major UF_6 Release	Rupture of one or two cylinders. Thousands of kg of U released. Major site contamination, 6 acres. Offsite cleanup is not expected.	$1.2M [$0.96M - $1.5M]	0.021 [0.011 - 0.081]	$25,000 [$9,100-$70,000]
			TOTAL FACILITY ECONOMIC RISK	$104,000 [$43,000-$250,000]

Table C.7 Summary of economic risk at a reference byproduct material manufacture/distribution facility (Philbin et al. 1990, Table 4.4)

Incident Scenario	Consequence Description	Cleanup Cost [uncertainty]	Frequency per year [uncertainty]	Economic Risk (per year) [uncertainty]
Minor Facility Releases	Small decontamination incident limited to the immediate area of the release.	$6500 [$5,200 - $8,100]	0.0022 [0.0015 - 0.0033]	$14 [$9 - $22]
Iodine-125 Spill Outside a Filtered Enclosure	Millicurie spill of NaI-125 an unfiltered area of laboratory. Laboratory decontamination required. No offsite cleanup required.	$30,000 [$24,000 - $38,000]	0.0022 [0.0015 - 0.0033]	$66 [$42 - $100]
Fire in a Fume Hood	Small fire involving molybdenum-99 generators in fume hood. Laboratory decontamination required. No off-site cleanup required.	$44,000 [$35,000- $55,000]	0.00059 [0.00034 - 0.0013]	$26 [$13 - $53]
Major Fire in an Iodine Laboratory	Fire in iodine-125 process-laboratory. Four curies volatilized and dispersed into two laboratories. 0.4 curies released to environment.	$290,000 [$230,000-$360,000]	0.00059 [0.00034 - 0.0013]	$170 [$84 - $350]
Waste Warehouse Fire (single drum)	Single waste drum fire. Several millicuries volatilized. Entire warehouse decontamination required.	$300,000 [$240,000-$380,000]	0.0081 [0.0074 - 0.0088]	$2,400 [$1,900 - $3,100]
Waste Warehouse Fire (multiple drums)	10% of waste inventory released in fire. Offsite decontamination required.	$1.1M [$0.9M - $1.4M]	0.0081 [0.0074 - 0.0088]	$8,900 [$7,000 - $11,000]
Tornado	Building 200 or 250 severely damaged or Bldg. 32 destroyed. 1% of in-process material released. 75% of waste inventory released.	$2M [$1.6M - $2.5M]	0.000030 [0.000009-0.00009]	$60 [$19 - $190]
Earthquake	Several buildings severely damaged. 1% of in-process material released.	$1.3M [$1.0M - $1.6M]	0.0040 [0.0010 - 0.020]	$5,200 [$1,100 - $24,000]
			TOTAL FACILITY ECONOMIC RISK	$17,000 [$8,600 - $31,000]

Table C.8 Summary of economic risk at a reference waste warehouse (Philbin et al. 1990, Table 4.5)

Incident Scenario	Consequence Description	Cleanup Cost [uncertainty]	Frequency per year [uncertainty]	Economic Risk (per year) [uncertainty]
Minor Facility Releases	Failure of one BLSV waste drum. Local decontamination.	$4000 [$3,200 - $5,000]	0.0041 [0.0022-0.016]	$16 [$6 - $45]
Waste Compactor Fire	Fire involving one drum of DAW waste. Local area decontamination.	$62,000 [$50,000-$78,000]	0.0081 [0.0074-0.0088]	$500 [$400 - $640]
Waste Drum Fire (single drum)	Fire consumes one BLSV waste drum. Entire warehouse decontamination required. No offsite cleanup required.	$410,000 [$330,000-$510,000]	0.0081 [0.0074-0.0088]	$3,300 [$2,600 - $4,200]
Transportation Accident	Highway accident (without fire -- 0.2 curies released, with fire -- 1 curie released) into two laboratories. 0.4 curies released to environment.	$40,000 [$32,000 - $50,000] $53,000 [$42,000 - $66,000]	0.0011 [0.00035-0.0035] 0.00024 [0.000076-0.00076]	$44 [$14 - $140] $13 [$4 - $41]
Facility Fire	Fire consumes ten percent of radiological inventory. Offsite decontamination required.	$1.2M [$0.9 M - $1.5M]	0.0081 [0.0074 - 0.0088]	$9,700 [$7,700-$12,000]
Tornado	Building destroyed. Seventy-five percent of waste inventory released.	$1.5M [$1.2M - $1.9M]	0.00020 [0.00006 - 0.0006]	$300 [$93 - $970]
			TOTAL FACILITY ECONOMIC RISK	$14,000 [$11,000-$16,000]

BLSV = bulk liquids and scintillation vials DAW = dry radioactive waste

Table C.9 Estimated 70-year population and worker exposures for repository construction (Daling et al. 1990, Table 4.2)

Geologic Medium	Worker Exposures (person-rem)	Maximum Individual Exposures (rem)	80-km Population Exposures (person-rem)
Salt	1.8E-1	2.8E-8	6.8E-3
Granite	5.0E+3	4.1E-4	1.0E+2
Basalt	6.2E+3	5.9E-5	1.5E+1
Shale	1.9E+3	1.5E-4	3.8E+1

Table C.10 Radiation exposure from normal construction and operation for repository preclosure period (Daling et al. 1990, Table 4.13)

Exposure Category	Estimated 50-yr Dose Commitment
Construction	
Maximally Exposed Individual	
-Annual	0.044 mrem
-50-yr	0.42 mrem
80-km Population	
-50-yr	2.0E+4 person-mrem
Operation	
Maximally Exposed Individual	
-Annual	0.17 mrem
-50-yr	5.6 mrem
80-km Population	
-50-yr	3.9E+5 man-mrem

Table C.11 Total radiological worker fatalities from construction and emplacement periods of three alternative Repository Sites (Daling et al. 1990, Table 4.20)

	Radiological Fatalities[a]			
Geologic Medium	Underground Construction	Underground Operations	Waste Handling Operations	Total
Salt	1.4E-2	4.4E-2	1.5E00	1.6E00
Tuff	7.7E-1	4.0E00	1.0E00	5.8E00
Basalt	1.6E00	5.4E00	1.9E00	8.9E00

(a) Based on 5-year construction and 26-year emplacement operations period.

Table C.12 Occupational dose during normal operation and from a shaft drop accident for repository preclosure period (Daling et al. 1990, Table 4.5)

Scenario	Number of Persons Involved	Average Annual Dose (rem/yr)	Total Dose (person-rem/yr)
Reference Case			
- Normal Operation	1,000	0.9	902
- Accident	300	1.5	454
Case 1			
- Normal Operation	1,068	1.2	1,295
Accident	352	1.6	569
Case 2			
- Normal Operation	1,045	1.1	1,188
- Accident	332	1.6	532
Case 3			
- Normal Operation	1,985	1.2	2,301
- Accident	603	1.6	978

Table C.13 Public dose during normal operation and from a shaft drop accident for repository preclosure period (Daling et al. 1990, Table 4.6)

Whole-body Dose Scenario	Public Dose (person-rem/yr)
Reference Case	
- Normal Operation	1.5E-5
- Accident	6.5E-2
Case 1	
- Normal Operation	5.0E-6
- Accident	5.6E-2
Case 2	
- Normal Operation	7.7E-6
- Accident	5.6E-2
Case 3	
- Normal Operation	1.1E-5
- Accident	5.6E-2

- Case 1. Simple encapsulation and disposal of spent fuel after storage at an away-from reactor storage facility (AFR) for 9 years.
- Case 2. Encapsulation of fuel, end fittings, and secondary wastes after chopping the fuel bundle and removal of volatile materials.
- Case 3. Encapsulation of fuel, end fittings, and secondary wastes after chopping, removal of volatile materials, calcination, and vitrification.

Table C.14 Summary of repository accident releases, frequencies, consequences, and risk values for repository preclosure period, operations phase (Daling et al. 1990, Table 4.11)

Accident Description	Release Quantity (Ci)	Frequency (per yr)	Consequences[a] (person-rem)	Risk Value (person-rem/yr)
Fuel truck crash into HLW area	H-3; 3 Cs-134; 300 Cs-137; 70	2.0E-6	2.0E+3	4.0E-3
Fuel truck crash into cladding waste area	FP[b]; 400 Actinides; 0.1	2.0E-6	2.0E00	4.0E-6
Fuel truck crash into NHLW area	Actinides; 100	2.0E-6	4.0E+1	8.0E-5
Aircrash into receiving area	H-3; 3 Cs-134; 300 Cs-137; 70 FP; 400 Actinides; 100	1.0E-7	4.0E+3	4.0E-4
Elevator drop	H-3; 4E-3 FP; 1E-2 Actinides; 4E-3	4.0E-8	5.0E-2	2.0E-9
Non-HLW pallet drop	Actinides; 0.02	5.0E-2	8.0E-1	4.0E-4
Final filter failure	Actinides; 0.2	3.0E-3	2.0E00	6.0E-3
	Total Preclosure Risk			1.0E-2

(a) Population doses are 50-year whole-body dose commitments.
(b) FP = Various fission products.

Table C.15 Radiation exposure from accidents for repository preclosure period, operations phase (Daling et al. 1990, Table 4.14)

Accident	Maximally Exposed Individual (mrem)	Population 50-yr Dose Commitment (person-mrem)
Spent Fuel Drop	4.68E+1	2.99E+3
Commercial HLW Drop	2.74E00	1.75E+2
Spent Fuel Handling	3.98E-2	1.29E+3
Remote TRU Drop	3.10E-3	1.98E-1
Contact TRU Puncture	2.07E-9	6.70E-5

TRU = transuranic HLW = high level waste NHLW = non-HLW

Table C.16 Occupational dose during repository operation (Daling et al. 1990, Table 4.15)

Activity	Number of Workers	Collective Dose (Person-rem/yr)
Receiving	35	44.8
Handling and Packaging	16	6.9
Surface Storage to Emplacement Horizon	14	6.0
Emplacement		
Vertical	18	12.4
Horizontal	7	8.7

Table C.17 Summary of annual occupational exposures for spent fuel and HLW operation at a tuff repository (Daling et al. 1990, Table 4.16)

Operation	Total Number of Workers	Total Annual Dose (person-rem/yr)
Receiving	35	44.6
Handling and Packaging	22	12.3
Transfer to Underground Facilities		
Shaft Access	9	3.35
Ramp Access	7	2.68
Emplacement in Boreholes		
Vertical	18	12.4
Horizontal	7	9.59
Retrieval from Boreholes		
Vertical	22	12.6
Horizontal	6	8.86
Return to Surface (Ramp)	5	2.68
Handling, Packaging, Shipping	17	20.48
Totals[a]		
Shaft Access/Vert. Empl.		72.68
Shaft Access/Horiz. Empl.		69.84
Ramp Access/Vert. Empl.		71.98
Ramp Access/Horiz. Empl.		69.17

(a) Totals do not include retrieval and loadout operations.

Table C.18 Estimated 50-year whole-body dose commitment to the public, maximally exposed individual workers from accidents for repository preclosure period, operations phase (Daling et al. 1990, Table 4.17)

Accident Scenario	Maximally Exposed Individual Dose (rem)	80 km Population Dose (person-rem)	Worker (person-rem)
Natural Phenomena			
Flood	2.8E-11	1.2E-9	5.0E-10
Earthquake	2.4E-4	3.1E-3	0.37
Tornado	2.4E-4	3.1E-3	0.37
Man-made Events			
Aircraft Impact	6.8E-2	110	5.5
Nuclear Test	2.4E-4	3.1E-3	0.37
Operational Accidents			
Fuel Assembly Drop	5.3E-6	8.0E-5	8.1E-3
Loading Dock Fire			
Spent Fuel	2.1E-2	6.8E-3	8.9E-3 - 3.5[a]
Commercial HLW	3.6E-3	9.2E-4	1.5E-3 - 0.6[a]
Waste Handling Ramp Fire	1.8E-7	3.6E-7	3.8E-8 - 64 [b]
Emplacement Drift Fire	1.8E-7	3.6E-7	3.8E-8 - 180[b]

(a) The first value represents the estimated dose to workers at the site surface and subsurface facilities; the second value is for the worker exposures at the loading dock.
(b) The first value is for the doses to workers in the surface facilities; the second value is for underground waste emplacement workers.

Table C.19 Preliminary risk estimates for postulated accidents at a repository in tuff for operations phase (Daling et al. 1990, Table 4.18)

Accident Scenario	Estimated Frequency (events/yr)	50-yr Dose Commitment (person-rem)	Population Risk (person-rem/yr)
Natural Phenomena			
Flood	1.0E-2	1.2E-9	1.2E-11
Earthquake	<1.3E-3	3.1E-3	<4.0E-6
Tornado	<9.1E-11	3.1E-3	<2.8E-13
Man-made Events			
Aircraft Impact	<2.0E-10	1.1E+2	<2.2E-8
Nuclear Test	<1.0E-3	3.1E-3	<3.1E-6
Operational accidents			
Fuel Assembly Drop	1.0E-1	8.0E-5	8.0E-6
Loading Dock Fire			
Spent Fuel	<1.0E-7	6.8E-3	<6.8E-10
Commercial HLW	<1.0E-7	9.2E-4	<9.2E-11
Waste Handling Ramp Fire	<1.0E-7	4.8E-7	<4.8E-14
Emplacement Drift Fire	<1.0E-7	4.8E-7	<4.8E-14
Total			1.5E-5

Table C.20 Frequencies and consequences of accident scenarios projected to result in offsite doses greater than 0.05 rem for repository preclosure period, operations phase (Daling et al. 1990, Table 4.23)

Accident Scenario Description	Frequency, per year	Consequence mrem
Internally Initiated Events		
Crane drops shipping cask, cask breached	5E-6	340
Crane drops fuel assembly in hot cell, HVAC fails	1E-8	170
Crane drops open consolidated fuel container, HVAC fails	1E-9	1100
Container dropped in storage vault, filtration system fails to activate	3E-8	230
Externally Initiated Events (all caused by earthquake)		
Crane fails, falls on or drops cask in receiving area	5E-8	340
Train falls on cask	5E-8	290
Structural object falls on fuel in cask unloading cell	5E-7	110
Crane fails, falls on or drops fuel in cask unloading cell	1E-6	110
Structural object falls on fuel in consolidation cell	5E-7	110
Crane fails, falls on or drops fuel in consolidation cell	1E-6	110
Structural object falls on fuel in packaging cell	5E-7	330
Crane fails, falls on or drops fuel in packaging cell, HVAC fails	1E-6	1100
Structural object falls on fuel in transfer tunnel	5E-7	200

HVAC = heating, ventilation, air conditioning

Table C.21 Occupational dose during normal operation and from accidents during decommissioning and retrieval phases of a repository (Daling et al. 1990, Table 4.7)

Scenario	Annual Dose (person-rem/yr)	
	Decommissioning	Retrieval[a]
Reference Case		
- Normal Operation	6	163
- Accident	5	89
Case 1		
- Normal Operation	23	588
- Accident	16	254
Case 2		
- Normal Operation	22	487
- Accident	15	215
Case 3		
- Normal Operation	40	1,116
- Accident	28	491

(a) Represents sum of doses from waste removal, offgas recovery and release, and mining and drilling activities.

- Case 1. Simple encapsulation and disposal of spent fuel after storage at an away-from reactor storage facility (AFR) for 9 years.
- Case 2. Encapsulation of fuel, end fittings, and secondary wastes after chopping the fuel bundle and removal of volatile materials.
- Case 3. Encapsulation of fuel, end fittings, and secondary wastes after chopping, removal of volatile materials, calcination, and vitrification.

Table C.22 Comparison of normalized public accident risk values from various studies for repository preclosure period (Daling et al. 1990, Table 4.27)

Document	Risk (person-rem/MTU)	Comment
GEIS	8.4E-9	One accident
Bechtel (1979)	1.1E-10	One accident
Waite et al. (1986)	1.7E-8	Five accidents
Jackson et al. (1984)	5.7E-9	Ten accidents
Erdmann et al (1979)	1.8E-6	Seven accidents
Pepping et al. (1981)	6.3E-10	One accident

Table C.23 1985 Revised EPA estimates of 10,000-year health effects for 100,000-MTHM repositories in basalt, bedded salt, tuff, and granite (Daling et al. 1990, Table 4.29)

Scenario	Basalt	Bedded Salt[(a)]	Tuff	Granite
Undisturbed	97	0	0	184
Drilling (misses canister)	2.30	3.16	0	0.92
Drilling (hits canister)	1.73	3.41	0.44	0.44
Faulting	24.4	0	3.00	8.49
Total Health Effects	125	6.57	3.44	194

(a) Palo Duro Basin

Table C.24 70-year cumulative maximally exposed individual and regional population doses for the two peak dose periods for a tuff repository (Daling et al. 1990, Table 4.35)

Organ	Accumulated Dose at the 27,000-Year Peak	Accumulated Dose at the 250,000-Year Peak
Total Body	0.2	0.2
Bone	0.6	3.0
Thyroid	2.0	2.0
Gastro-intestinal	4.0	2.0

Lifetime Population Doses from the Drinking Water Scenario for Two Future Times (person-rem)

Organ	Accumulated Dose at 27,000 Years	Accumulated Dose at 250,000 Years
Total Body	2.0	200
Bone	4.0	4,000
Thyroid	600	600
Gastro-intestinal	200	400

Table C.25 Peak conditional cancer risks due to ingestion for the 100,000-year postclosure period for a 90,000-MTU spent fuel repository in bedded salt (Daling et al. 1990, Table 4.38)

Scenario (Number) And Description	Zone 1: Area From Repository to River 40 km Away, Plus 6 km Along River	Zone 2: Area Bounded by a 40-km Stretch of River and 2 km Along Both Sides
(1) Borehole(s) with Lower Aquifer Wells	8.0E-2	8.0E-7
(2) U-Tube with Upper Aquifer Wells	2.0E-1	4.0E-6
(3) Dissolution Cavity with Wells	3.0E-1	7.0E-6
(4) Borehole(s)	1.0E-6	1.0E-6
(5) U-Tube	2.0E-6	1.0E-6
(6) Borehole(s) intersecting a Canister	3.0E-6	2.0E-6

Table C.26 Radiation exposures from routine operations at the MRS facility (Daling et al. 1990, Table 4.42)

Pathway and Location in the Body	50-Year Dose Commitment from Annual Release	
	Maximally Exposed Individual (rem)	Population (person-rem)
Total Body	2.4×10^{-4}	2×10^{1}
Bone	3.0×10^{-6}	1×10^{-1}
Lungs	2.4×10^{-4}	2×10^{1}
Thyroid	1.3×10^{-3}	1×10^{2}

Table C.27 Radiological impacts of potential MRS facility accidents for sealed storage cask at the Clinch River Site for operations phase (Daling et al. 1990, Table 4.43)

Accident	Location in the body	50-Year Dose Commitment to the Public: Maximally Exposed Individual (rem)	50-Year Dose Commitment to the Public: Population (person-rem)
Fuel Assembly Drop	Total Body	4.4×10^{-3}	3×10^{-2}
	Bone	1.4×10^{-4}	7×10^{-3}
	Lungs	4.6×10^{-3}	3×10^{-2}
	Thyroid	2.9×10^{-2}	2×10^{-1}
Shipping Cask Drop	Total Body	9.1×10^{-4}	6×10^{-3}
	Bone	3.0×10^{-5}	1×10^{-3}
	Lungs	9.6×10^{-4}	6×10^{-3}
	Thyroid	6.0×10^{-3}	3×10^{-2}
Storage Cask Drop	Total Body	8.9×10^{-4}	6×10^{-3}
	Bone	2.9×10^{-5}	1×10^{-3}
	Lungs	9.3×10^{-4}	6×10^{-3}
	Thyroid	5.9×10^{-3}	3×10^{-2}

Table C.28 Occupational dose from MRS facility operations (Daling et al. 1990, Table 4.44)

Operation	Unit Occupational (person-rem/1,000 MTU)
Receipt and Unloading	58
Consolidation	6
Loading Consolidated Fuel Rods	9
Maintenance/Monitoring	2
Emplacement and Retrieval	20
Total	95

Table C.29 Summary of occupational doses from MRS facility operations (Daling et al. 1990, Table 4.49)

Operation	(person-rem/yr)
Receipt, Inspection, Unloading	148.0
Transfer to Storage Casks	6.2
Emplacement in Storage Area	7.2
Surveillance in Storage Area	5.3
Retrieval from Storage Area	7.1
Transfer to Process Cells	4.0
Shipment to Repository	140.9
Total	318.7

Table C.30 Occupational dose estimates for selected MRS operations (Daling et al. 1990, Table 4.50)

Operation	Occupational Dose (person-mrem/1,000MTU)
Consolidate and package fuel	3.6
Consolidate and package non-fuel components	1.1
Receiving and unloading - Truck	135
- Rail	25

Table C.31 Summary of MRS drywell risk analysis for operations phase (Daling et al. 1990, Tables 4.45 and 4.46)

	Frequency Per Year	Release Category	Latent Cancer Fatalities	Risk
Transporter collision during emplacement				
- no fire	1.7E-8	III	3.4E-5	5.8E-11
- fire	6.1E-7	IV	1.9E-3	1.2E-9
Transporter collision during retrieval				
- no pin failure; no fire	8.9E-3	II	5.9E-7	5.3E-9
- pin failure; no fire	2.8E-2	III	3.8E-5	1.1E-6
- no pin failure; fire	1.4E-4	IV	2.6E-6	3.6E-12
- pin failure; fire	1.4E-4	IV	2.6E-4	3.6E-8
Transporter motion with canister partially in place				
- emplacement	8.6E-2	V	1.8E-2	1.5E-2
- retrieval; no pin failure	8.9E-3	II	5.9E-7	5.3E-9
- retrieval; pin failure	1.4E-1	V	1.6E-3	2.2E-4
Canister drop - emplacement	1.7E-8	I	3.9E-6	6.6E-14
Canister drop - retrieval	1.1E-2	I	9.9E-7	1.1E-8
Plane crash; no fire	4.0E-10	V	2.6E-1	1.0E-10
Plane crash; fire	7.4E-9	VI	1.3E+0	9.6E-9
Earthquake; no pin failure	4.8E-9	II	6.1E-2	2.9E-10
Earthquake; pin failure	4.3E-8	II	3.3E+0	1.4E-7
Total				1.7E-3

Release Category	Release Type (Generic Event)	Assumed Damage Per Canister Involved In Event	Fraction Release of Radionuclides to Environment	
I	Filtered gap release (canister impact in the interface areas)	Gap inventory from 10% pins released through filters	Gases:[a] I:	3.0E-2 3.0E-4
II	Limited gap release (canister leak)	Gap inventory from 1% pins (assumed to develop leaks while in storage) released via leaks and exit channels	Gases: I:	3.0E-3 5.0E-4
III	Unlimited gap release (canister impact in storage areas)	Complete gap inventory from 10% pins	Gases: I:	3.0E-2 3.0E-2
IV	Elevated temperature release (temporary loss of cooling)	Complete inventory of gases and I and 1% of volatiles released via leaks and exit channels	Gases: I: Cs, Ru:	1.0E+0 1.7E-1 1.0E-4
V	Exposed fuel release (severe canister impact)	10% of fuel exposed releasing gap inventory, volatiles, and particulates. Remainder releases gap inventory via leaks and exit channels	Gases: I: Cs, Ru: Particles:	3.0E-1 6.0E-1 1.0E-3 1.5E-6
VI	Exposed heated-fuel release (severe canister impact with fire)	As in V, with increased releases	Gases: I: Cs, Ru: Particles:	1.0E+0 2.0E-1 5.1E-3 3.0E-6

(a) Gases include C-14, H-3, and Kr-85.

Table C.32 Summary of results of MRS operations phase (Daling et al. 1990, Table 4.48)

Accident Scenario	Frequency (events/yr)	Number of Assemblies	Release Category	Consequence (LCF)	Risk (LCF/yr)
Fuel Assembly Drop During Loading	1E-1	1	1	4E-5	4E-6
Drop of Transport Cask During Loading					
Cask	4E-3	10	1	4E-4	2E-6
Drywell	7E-2	10	1	4E-4	3E-5
Venting of Cask During Transport					
Cask	2E-3	24	2	1E-1	2E-4
Drywell	3E-2	1	2	4E-3	1E-4
Collision During Transport					
Cask	2E-4	24	3	1E-1	2E-5
Drywell	2E-5	1	3	4E-3	8E-8
Collision with Fire During Transport					
Cask	2E-6	24	5	5E-1	1E-6
Drywell	2E-7	1	5	2E-2	4E-9
Canister Drop During Emplacement					
Drywell	1E-6	1	3	4E-3	4E-9
Canister Shear During Emplacement					
Drywell	2E-6	1	3	4E-3	8E-9
Cask Drop During Emplacement					
Cask	1E-5	24	3	1E-1	1E-6
Tornado Missile Penetration					
Cask	6E-6	10	3	4E-2	2E-7
Drywell	1E-4	10	3	4E-2	4E-6
Plane Crash Topples Cask with Fire					
Cask	6E-9	24	5	5E-1	3E-9
Plane Crash Plus Fire					
Cask	9E-9	24	5	5E-1	4E-9
Drywell	2E-7	1	5	2E-2	4E-9
	2E-8	10	5	2E-1	4E-9
Earthquake					
Cask	4E-6	24	3	1E-1	4E-7
	4E-8	2400	3	1E+1	4E-7
Drywell	8E-6	1	3	4E-3	3E-8
	8E-7	10	3	4E-2	3E-8
	2E-8	2400	3	2.4	5E-8
	Total Risk:	Cask		2.3E-4	
		Drywell		1.4E-4	

LCF = latent cancer fatality

Table C.33 Projected maximum individual exposures from normal spent fuel transport by truck cask[a] (Daling et al. 1990, Table 4.61)

(Service or Activity)	Distance to Center of Cask	Exposure Time	Maximum Dose Rate and Total Dose
Caravan			
Passengers in vehicles traveling in adjacent lanes in the same direction as cask vehicle	10 m	30 min	40 µrem/min 1 mrem
Traffic Obstruction			
Passengers in stopped vehicles in lanes adjacent to the cask vehicle which have stopped due to traffic obstruction	5 m	30 min	100 µrem/min 3 mrem
Residents and Pedestrians			
Slow transit (due to traffic control devices through area with pedestrians)	6 m	6 min	70 µrem/min 0.4 mrem
Truck stop for driver's rest. Exposures to residents and passers-by.	40 m	8 hours (assumes overnight)	6 µrem/min 3 mrem
Slow transit through area with residents (homes, businesses, etc.)	15 m	6 min	20 µrem/min 0.1 mrem
Truck Servicing			
Refueling (100 gallon capacity)	7 m (at tank)		60 µrem/min
- 1 nozzle from 1 pump		40 min	2 mrem
- 2 nozzles from 1 pump		20 min	1 mrem
Load inspection/enforcement	3 m (near personnel barrier)	12 min	160 µrem/min 2 mrem
Tire change or repair to cask trailer	5 m (inside tire nearest cask)	50 min	100 µrem/min 5 mrem
State weight scales	5 m	2 min	80 µrem/min 0.2 mrem

(a) These exposures should not be multiplied by the expected number of shipments to a repository in an attempt to calculate total exposures to an individual; the same person would probably not be exposed for every shipment, nor would these maximum exposure circumstances necessarily arise during every shipment.

Table C.34 Projected maximum individual exposures from normal spent fuel transport by rail cask[a] (Daling et al. 1990, Table 4.62)

(Service or Activity)	Distance to Center of Cask	Exposure Time	Maximum Dose Rate and Total Dose
Caravan			
Passengers in rail cars or highway vehicles traveling in same direction and vicinity as cask vehicle	20 m	10 min	30 µrem/min 0.3 mrem
Traffic Obstruction			
Exposures to persons in vicinity of stopped/slowed cask vehicle due to rail traffic obstruction	6 m	25 min	100 µrem/min 2 mrem
Residents and Pedestrians			
Slow transit (through station or due to traffic control devices) through area with pedestrians	8 m	10 min	70 µrem/min 0.7 mrem
Slow transit through area with residents (homes, businesses, etc.)	20 m	10 min	30 µrem/min 0.3 mrem
Train stop for crew's personal needs (food, crew change, first aid, etc.)	50 m	2 hours	5 µrem/min 0.6 mrem
Train Servicing			
Engine refueling, car changes, train maintenance, etc.	10 m 6 mrem	2 hours	50 µrem/min
Cask inspection/enforcement by train, state or federal officials	3 m	10 min	200 mrem 2 mrem
Cask car coupler inspection/maintenance	9 m	20 min	70 µrem/min 1 mrem
Axle, wheel or brake inspection/lubrication/maintenance on cask car	7 m	30 min	90 µrem/min 3 mrem

(a) These exposures should not be multiplied by the expected number of shipments to a repository in an attempt to calculate total exposures to an individual; the same person would probably not be exposed for every shipment, nor would these maximum exposure circumstances necessarily arise during every shipment.

Table C.35 Summary of results from the NRC for spent fuel shipments (Daling et al. 1990, Table 4.54)

Year	Mode	Shipments Per Year	Normal Population Dose, (person-rem/yr)	Accident Risk, Latent Cancer (fatalities/yr)
1975	Truck	254	93.80	0.047
	Rail	17	7.78	0.021
1985	Truck	1,530	565.0	0.29
	Rail	652	298.0	0.8

Table C.36 Maximum individual radiation dose estimates for rail cask accidents during spent fuel transportation (Daling et al. 1990, Table 4.63)

	Dose (mrem)[a]		
Accident Class	Inhalation	Plume Gamma	Ground Gamma
Impact	179	10.7	12.3
Impact and Burst	6,130	71.1	90.9
Impact, Burst and Oxidation	8,950	547	707

(a) The maximally exposed individual dose occurs about 70 meters downwind of the release point and assumes that the individual remains at this location for the duration of the passage of the plume of nuclides that are released.

Table C.37 50-year population dose estimates for spent fuel rail cask accidents with no cleanup of deposited nuclides[a] (Daling et al. 1990, Table 4.64)

	Urban Area (3,860 people/km^2)				Rural Area (6 people/km^2)			
Accident Class	Inhalation	Plume Gamma	Ground Gamma	Total	Inhalation	Plume Gamma	Ground Gamma	Total
Impact								
Dose (person-rem)	3.09	0.33	936	939	0.005	0.0005	1.45	1.45
Latent Health Effects[b]				0.19				0.00029
Impact and Burst								
Dose (person-rem)	106	2.23	13,400	13,500	0.16	0.0034	20.8	21
Latent Health Effects[b]				2.7				0.0042
Impact, Burst and Oxidation								
Dose (person-rem)	154	17.2	112,000	112,000	0.24	0.27	174	174
LHE[b]				22				

(a) The ground gamma dose is what would be received if each member of the population stayed at the same location for 50 years. The inhalation dose is a 50-year dose commitment from inhalation of the passing plume. Doses are for the population within 80 kilometers of the release point. It is assumed that there is no cleanup of deposited nuclides and that no other measures are used to reduce radiation exposures.

(b) Based on 1 person-rem = 2.0E^{-4} LHEs. An LHE is defined here as an early cancer death by an exposed person or a serious genetic health problem in the two generations after those exposed. About half of the LHEs are expected to be cancers and the rest genetic health problems.

LHE = latent health effect

Table C.38 Population radiation exposure from water ingestion for severe but credible spent fuel rail cask accidents (Daling et al. 1990, Table 4.65)

Accident Class	Total Release from Rail Cask (Ci)[a]	Population Dose Effects from Water Ingestion
Impact	8.07	182 person-rem 0.036 LHE[b]
Impact and Burst	153	6870 person-rem 1.4 LHE[b]
Impact, Burst	1379	63,000 person-rem 12.6 LHE[b]

(a) The noble gas Kr-85 is omitted because of its negligible uptake by a surface water body.
(b) LHE estimates are based upon 1 person-rem = 2.0E-4 LHE.

Table C.39 Summary of spent fuel truck and rail transportation risks (Daling et al. 1990, Table 4.58)

Model/Fuel Age	Annual Quantity Shipped, (MTU/yr)	Average Shipping Distance, (km)	Number of (shipments/yr)	Probability of One or More (LHE/yr)
Truck				
180-day	380	690	885	2.2E-5
4-yr	380	690	885	3.6E-6
Rail				
180-day	1,474	912	471	5.5E-5
4-yr	1,474	912	471	8.3E-7

Table C.40 Summary of the routine transportation risks for the waste management system without an MRS facility (Daling et al. 1990, Table 4.59)

Mode	Repository Location		
	Deaf Smith	Yucca Mt.	Hanford
100% Truck from origin SF to Repository			
Radiological[(a)]	6.2	9.2	10
Nonradiological[(b)]	18	29	31
HLW to Repository			
Radiological	1.7	2.1	2.1
Nonradiological	6.2	7.4	7.4
100% Rail from origin SF to Repository			
Radiological	0.18	0.24	0.25
Nonradiological	1.0	1.6	1.6
HLW to Repository			
Radiological	0.063	0.079	0.074
Nonradiological	0.64	0.84	0.79
TOTALS			
Truck from origin			
Radiological	7.9	11	12
Nonradiological	24	36	38
Rail from origin			
Radiological	0.24	0.32	0.32
Nonradiological	1.6	2.4	2.4

(a) Radiological health effects include lethal cancer fatalities and genetic effects in all generations.
(b) Nonradiological fatalities.

SF = spent fuel

Table C.41 Summary of the routine transportation risks for the waste management system with an MRS facility (Daling et al. 1990, Table 4.60)

	Repository Location		
Mode	Deaf Smith	Yucca Mt.	Hanford
100% Truck from origin SF to MRS			
Radiological[a]	3.6	3.6	3.6
Nonradiological[b]	9.1	9.1	9.1
HLW to Repository by Truck			
Radiological	1.7	2.1	2.1
Nonradiological	6.2	7.4	7.4
100% Rail from origin SF to MRS			
Radiological	0.14	0.14	0.14
Nonradiological	0.92	0.92	0.92
HLW to Repository by Rail			
Radiological	0.063	0.079	0.074
Nonradiological	0.64	0.84	0.79
150T Rail from MRS			
Radiological	0.035	0.054	0.042
Nonradiological	3.8	1.0	6.1
TOTALS			
Truck from origin, 150T Rail from MRS			
Radiological	5.3	5.8	5.7
Nonradiological	19	18[c]	23
Rail from origin, 150T Rail from MRS			
Radiological	0.24	0.27	0.26
Nonradiological	5.3	12	7.8

(a) Radiological health effects include lethal cancer fatalities and genetic effects in all generations.
(b) Nonradiological fatalities
(c) An error was found in the source document. The value in this table is believed to be correct.

Table C.42 Aggregated public risks for the preclosure phases of the waste management system without an MRS Facility[a] (Daling et al. 1990, Table 5.11)

System Element Operating Phase	Radiological Risks[b] (LHE/yr) Accidents	Radiological Risks[b] (LHE/yr) Routine Operations	Nonradiological Risks Accidents (fatalities/yr)	Nonradiological Risks Routine (health effects/yr)
Repository Preclosure				
Construction	N/A	1E-5	(c)	Negligible
Operations	6E-9	9E-4	(c)	Negligible
Decommissioning	Information Not Available	2E-11	(c)	Negligible
Transportation System[d]				
Operations	1E-3	9E-2	3E-1	1E-2
Total Aggregated Risks (For Facility Operating Phases Only)	1E-3	9E-2	3E-1	1E-2

(a) Risks for the facility operations phase are annual risks for a fully functioning waste management system operating at a 3,000 MTU/yr throughput rate. Risks for other facility phases are levelized annual risks prorated over the number of years required for the specific phase.

(b) Health effects include latent cancer fatalities plus first and second generation genetic effects.

(c) There are not expected to be site-related public nonradiological fatalities. Traffic-related public fatalities are included with traffic-related worker fatalities in Table 5.12.

(d) Shipping modes are as follows: spent fuel, 30% truck and 70% rail; HLW, 100% rail.

Table C.43 Aggregated occupational risks for the preclosure phases of the waste management system without an MRS facility[a] (Daling et al. 1990, Table 5.12)

System Element Operating Phase	Radiological Risks[b] (LHE/yr) Accidents	Routine Operations	Nonradiological Risks Accidents (fatalities/yr)	Operations (health effects/yr)
Repository Preclosure Construction	N/A	1E-1	2E+0	No Significant Impact
Operations	6E-5	2E-2	3E+0	No Significant Impact
Decommissioning	Information Not Available	3E-2	8E-1	No Significant Impact
Transportation System[c] Operations	Included With Public Risks	2E-2	8E-2	Information Not Available
Total Aggregated Risks (For Facility Operating Phases Only)[c]	6E-5	4E-2	3E+0	Information Not Available

(a) Risks for the facility operations phase are annual risks for a fully functioning waste management system operating at a 3,000 MTU/yr throughput rate. Risks for other facility phases are levelized annual risks prorated over the number of years required for the specific phase.
(b) Health effects include latent cancer fatalities plus first and second generation genetic effects.
(c) Shipping modes are as follows: spent fuel, 30% truck and 70% rail; HLW, 100% rail.

Table C.44 Aggregated public risks for the preclosure phases of the waste management system with an MRS facility[a] (Daling et al. 1990, Table 5.13)

System Element Operating Phase	Radiological Risks[b] (LHE/yr) Accidents	Radiological Risks[b] (LHE/yr) Routine Operations	Nonradiological Risks Accidents (fatalities/yr)	Nonradiological Risks Routine (health effects/yr)
Repository Preclosure				
Construction	N/A	1E-5	(c)	Negligible
Operations	6E-9	8E-7	(c)	Negligible
Decommissioning	Information Not Available	2E-11	(c)	Negligible
MRS Facility				
Construction	No Radioactive Materials Onsite		(c)	No Significant Impacts
Operations	8E-7	5E-3		
Decommissioning	Not Evaluated	2E-11		
Transportation System Operations[d]	2E-3	3E-2	4E-1	8E-3
Total Aggregated Risks (For Facility Operating Phases Only)[c]	2E-3	4E-2	4E-1	8E-3

(a) Risks for the facility operations phase are annual risks for a fully functioning waste management system operating at a 3,000 MTU/yr throughput rate. Risks for other facility phases are levelized annual risks prorated over the number of years required for the specific phase.
(b) Health effects include latent cancer fatalities plus first and second generation genetic effects.
(c) There are not expected to be site-related public nonradiological fatalities. Traffic-related public fatalities are included with traffic-related worker fatalities in Table 5.14.
(d) Shipping modes are as follows: spent fuel from reactors to MRS, 30% truck and 70% rail; HLW, 100% rail; all wastes from MRS facility to repository, 100% rail.

Table C.45 Aggregated occupational risks for the preclosure phases of the waste management system with an MRS facility[a] (Daling et al. 1990, Table 5.14)

System Element Operating Phase	Radiological Risks[b] (LHE/yr) Accidents	Routine Operations	Nonradiological Risks Accidents (fatalities/yr)	Routine (health effects/yr)
Repository Preclosure				
Construction	N/A	1E-1	2E+0	No Significant Impacts
Operations	5E-5	2E-2	2E+0	No Significant Impacts
Decommissioning	Information Not Available	3E-2	7E-1	No Significant Impacts
MRS Facility				
Construction	No Radioactive Materials Onsite		2E+0	No Significant Impacts
Operations	1E-4	6E-2	2E+0	No Significant Impacts
Decommissioning	3E-3	5E-3	1E-1	No Significant Impacts
Transportation System[c]	Included With Public Risks	8E-3	4E-2	Information Not Available
Total Aggregated Risks (For Facility Operating Phases Only)[c]	2E-4	9E-2	4E+0	Information Not Available

(a) Risks for the facility operations phase are annual risks for a fully functioning waste management system operating at a 3,000 MTU/yr throughput rate. Risks for other facility phases are levelized annual risks prorated over the number of years required for the specific phase.
(b) Health effects include latent cancer fatalities plus first and second generation genetic effects.
(c) Shipping modes are as follows: spent fuel from reactors to MRS, 30% truck and 70% rail; HLW, 100% rail; all wastes from the MRS to the repository, 100% rail.

Table C.46 Total preclosure life-cycle risk[a] estimates for the waste management system[b] (Daling et al. 1990, Table 5.15)

Population Group	Radiological Risks (LHE) Accidents	Routine	Nonradiological Fatalities[c]
Public Risks	0.04	2	10
Occupational Risks	0.004	3	100

(a) Sum of risks during construction, operation, and decommissioning phases of the waste management system.
(b) Average life-cycle risks with respect to system configurations with and without an MRS facility.
(c) Sum of nonradiological accident and routine risks.

Table C.47 Summary of annual and total life-cycle risk estimates for the waste management system[a] (Daling et al. 1990, Table S.2)

Risk Category	Operating Phase[b,c] Annual Risks	Total Life-[c,d] Cycle Risks
Public Risks		
- Radiological Accidents[e]	0.001	0.04
- Radiological Routine[e]	0.06	2
- Nonradiological[f]	0.4	10
- Postclosure Radiological[g]	0.001	--Not calculated--
Occupational Risks		
- Radiological Accidents[e]	0.0001	0.004
- Radiological Routine[e]	0.06	3
- Nonradiological[f]	0.4	100
Risk Perspective		
- Natural Background Radiation[h]	60	2000

(a) Average for waste management system configurations with and without an MRS facility.
(b) Annual risks from facility operating phases only. Does not include construction, decommissioning, and repository retrieval risks.
(c) Based on 30% truck/70% rail shipments from reactors, 100% rail from the MRS facility (where applicable), and 100% rail shipments from high-level waste (HLW) generators.
(d) Risks associated with spent fuel storage at reactor and other commercial sites are not included on the total life-cycle risk estimates.
(e) Annual radiological risks are given in units of latent health effects per year (LHE/yr); total life-cycle risks are given in units of LHEs.
(f) Annual nonradiological risks are given in units of fatalities/yr; total life-cycle nonradiological risks are given in units of fatalities.
(g) Peak annual radiological health effects from routine releases and selected disruptive events.
(h) Based on the estimated latent health effects from the population dose from natural background radiation within 80 km of the repository and MRS sites and within 0.5 km of a highway or railroad.

Table C.48 Accident frequencies and population doses for milling in the nuclear fuel cycle (Cohen and Dance 1975)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Fire in solvent extraction circuit	4E-4 to 3E-3	1.0E-1
Release of tailings slurry from tailings pond	4E-2	1.9E-1
Release of tailings slurry from tailings distribution pipeline	1E-2	8.3E-3

A key assumption is that 1% of the solvent extraction inventory is dispersed during a fire. Study limitations include the small number of accident

Table C.49 Accident frequencies and population doses for conversion in the nuclear fuel cycle (Cohen and Dance 1975)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Uranyl nitrate evaporator explosion	1E-4 to 1E-3	4.0
Hydrogen explosion in reduction	1E-3 to 5E-2	4.0
Fire in solvent extraction operation	4E-4	3.9E-1
Release from a hot UF_6 cylinder	3E-2	4.3E-1
Valve rupture in distillation step	5E-2	1.6E-1
Release of raffinate from waste retention pond	2E-2	3.1E-1

Table C.50 Accident frequencies and population doses for enrichment in the nuclear fuel cycle (Cohen and Dance 1975)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Catastrophic fire	4E-4 to 3E-2	4.9
Release from a hot UF_6 cylinder	4E-1	7.5E-1
Leaks or failure of valves and piping	1.8	7.7E-3
Criticality	8E-5	1.2E-2

Table C.51 Accident frequencies and population doses for fuel fabrication in the nuclear fuel cycle (Cohen and Dance 1975)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Hydrogen explosion in reduction furnace	2E-3 to 5E-2	7.4E-5 to 7.4E-2
Major facility fire	2E-4	7.4E-2 to 7.4E1
Fire in a roughing filter	1E-2	1.8E-5 to 1.8E-2
Release from a hot UF_6 cylinder	3E-2	7.8E-3 to 7.8
Failure of valves and piping	4E-3	2.2E-3 to 2.2
Criticality	8E-4	1.1
Waste Retention Pond Failure	2E-3 to 2E-2	3.5E-2

Table C.52 MOX fuel refabrication radiological accident risk

Study	Expected Population Dose (person-rem/GW_e-year)	Dominant Risk Contributor
Cohen and Dance (1975)	1.2E-2 to 1.9E-2 (total body)	Disolver fire in scrap recovery combined with HEPA failure.
Erdman et al. (1979)	4.0E-2 (total body)	Greater than design basis earthquake.
Fullwood and Jackson (1980)	4.0E-7 (total body)	Criticality in wet scrap.

Table C.53 Accident frequencies and population doses for MOX fuel refabrication in the nuclear fuel cycle (Cohen and Dance 1975)[3]

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Explosion in oxidation-reduction scrap furnace		
Normal HEPA filtration	2E-3 to 5E-2	3.1E-2
HEPA filter failure	2E-6 to 5E-5	3.1E3
Major facility fire		
Normal HEPA filtration	2E-4	1.6
HEPA filter failure	2E-7	1.4E5
Fire in waste compaction glove box		
Normal HEPA filtration	1E-2	3.1E-3
HEPA filter failure	1E-5	3.1E2
Ion-exchange resin fire		
Normal HEPA filtration	1E-4 to 1E-1	9.2E-3
HEPA filter failure	1E-7 to 1E-4	9.2E2
Dissolver fire in scrap recovery		
Normal HEPA filtration	1E-2	1.6E-1
HEPA filter failure	1E-5	1.6E4
Glove failure		
Normal HEPA filtration	1	1.3E-5
HEPA filter failure	1E-3	1.3
Severe glove box damage		
Normal HEPA filtration	1E-2	6.1E-2
HEPA filter failure	1E-5	6.1E3
Criticality		
Normal HEPA filtration	3E-5 to 8E-3	3.8E-1
HEPA filter failure	3E-8 to 8E-6	4.2E2

HEPA = high efficiency particulate air

Table C.54 Accident frequencies and population doses for MOX fuel refabrication in the nuclear fuel cycle (Erdmann et al. 1979)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Greater than design basis earthquake	5E-6	1E5
Aircraft crash	3E-7	3E4
Hydrogen explosion in ROR reactor	1E-3	5E-9
Hydrogen explosion in sintering furnace	1E-3	2E-7
Ion exchange resin fire	5E-4	2E-9
Dissolver explosion wet scrap recovery	5E-3	2E-6
Loaded final filter failure	2E-4	3E-1
Criticality	6E-5	5

Table C.55 Accident frequencies and population doses for MOX fuel refabrication in the nuclear fuel cycle (Fullwood and Jackson 1980)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Aircraft crash	1.5E-9	5E2
Hydrogen explosion in ROR	5E-3	1.1E-11
Hydrogen explosion in sintering	5E-3	4E-10
Hydrogen explosion in wet scrap	3E-4	1.1E-11
Criticality in wet scrap	6E-5	2
Powder shipping container spill	3E-5	1.1E-11
Exothermic reactions in powder storage	1.5E-6	1E-10

Table C.56 Fuel reprocessing radiological accident risk

Study	Expected Population Dose (person-rem/GW_e-year)	Dominant Risk Contributor
Cohen and Dance (1975)	2.8E-3 to 6.3E-3 (total body)	Fuel assembly rupture combined with HEPA failure.
Erdman et al. (1979)	2.0E-4 (total body)	Krypton cylinder failure; explosion in HLW calciner.
Fullwood and Jackson (1980)	7.0E-5 (total body)	Krypton cylinder failure.

ROR = reduction-oxidation reactor

Table C.57 Accident frequencies and population doses for reprocessing in the nuclear fuel cycle (Cohen and Dance 1975)[4]

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Explosion in HAW concentration		
Normal HEPA	1E-5	4.3E2
Failed HEPA	1E-8	9.5E3
Explosion in LAW concentration		
Normal HEPA	1E-4	2.8E1
Failed HEPA	1E-7	4.8E1
Explosion in HAW feed tank		
Normal HEPA	1E-5	1.6E3
Failed HEPA	1E-7	1.7E3
Explosion in waste calciner		
Normal HEPA	1E-6	4.3E3
Failed HEPA	1E-9	1.3E4
Explosion in iodine absorber	2E-4	4.8
Solvent fire in codecon cycle		
Normal HEPA	1E-6 to 1E-4	2.3E1
Failed HEPA	1E-9 to 1E-7	5.6E1
Solvent fire in Pu extraction cycle		
Normal HEPA	1E-6 to 1E-4	3.1E-4
Failed HEPA	1E-11 to 1E-9	5.2E2
Ion exchange resin fire		
Normal HEPA	1E-4 to 1E-1	3.6E-1
Failed HEPA	1E-9 to 1E-6	1.8E3
Fuel assembly rupture in fuel receiving and storage		
Normal HEPA	1E-2 to 1E-1	1.3E-2
Failed HEPA	1E-5 to 1E-4	1.3E3
Dissolver seal failure		
Normal HEPA	1E-5	2.3E-2
Failed HEPA	1E-8	2.3E3
Release from hot UF_6 cylinder	5E-2	1.5
Criticality		
Normal HEPA	3E-5 to 8E-3	3.0E-2
Failed HEPA	3E-8 to 8E-6	3.5E-2

HAW = high activity waste LAW = low activity waste

Table C.58 Accident frequencies and population doses for reprocessing in the nuclear fuel cycle (Erdmann et al. 1979)[5]

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Loss of fuel storage pool water	3E-6	50
Ion exchange bed fire and explosion	5E-4	2E-1
Criticality	6E-5	5
Hydrogen explosion in HAF tank	7E-5	7E-2
Fire in low level waste	1E-2	1E-1
Fuel assembly drop	2E-3	1E-1
Explosion in high-level waste calciner combined with HEPA filter failure	5E-10	6E6
Krypton cylinder rupture	1E-4	50

HAF = high aqueous feed

Table C.59 Accident frequencies and population doses for reprocessing in the nuclear fuel cycle (Fullwood and Jackson 1980)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
H_2 fire an explosion in HAF tank combined with one HEPA filter failed	3E-6	9E-4
Solvent fire in the H_2 concentration combined with one HEPA filter failed	2E-6	7E-4
Red oil explosion in HLW concentration combined with one HEPA filter failed	4E-8	8E-3
Explosion in the HLW calciner combined with one HEPA filter failed	2E-7	2E-1
Red oil explosion in the fuel product concentration combined with one HEPA failed	4E-8	6E-4
Explosion in fuel product deitrator combined with one HEPA failed	4E-9	1.2E-2
Criticality in a process cell	2E-5	2
Failure of Krypton storage cylinder	1.3E-4	4E1
Hydrogen explosion in uranium reduction combined with one HEPA filter failed	9E-6	1.4E-4
Fuel assembly drop	1.2E-3	5E-2
Hydrogen explosion in fuel product denitrator fuuel tank combined wiith one HEPA filter failed	3E-6	1.2E-2

Table C.60 Accident frequencies and population doses for reprocessing in the nuclear fuel cycle (Cooperstein et al.)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
HAW concentration explosion	1E-5	57
Codecontamination solvent fire	1E-6	2.6
LAW concentrator explosion	1E-4	3.2
HAF tank explosion	1E-5	4.9E2
Waste calciner explosion	1E-6	5.1E2
Fuel receiving and storage accident	1E-2	2.0E-3

Table C.61 Accident frequencies and population doses for spent fuel storage in the nuclear fuel cycle (Karn-Bransle-Sakerhat 1977)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Fuel transfer basket is dropped		
PWR	1E-4	2
BWR	2.5E-4	1.8
Fuel assemblies dropped		
PWR	9E-4	7E-1
BWR	6E-3	3E-1

Table C.62 Accident frequencies and population doses for solidified HLW storage in the nuclear fuel cycle (Smith and Kastenberg 1976)

Accident	Frequency (per plant year)	Population Dose for Reference Plant (person-rem total body)
Major rupture of a waste canister dropped during handling. Vent system effective	1.0E-4	7.2
Major rupture of a waste canister with an independent failure of one HEPA filter	1.0E-6	7.2E3
0.1-1 ton meteor impact in storage area	4.1E-9	1.0E5
10-100 ton meteor impact in storage area	2.0E-10	5.1E6
0.1-1 ton meteor impact in receiving area	4.8E-10	3.1E5
1-10 ton meteor impact in receiving area	1.25E-11	2.6E7

Table C.63 Preclosure geologic waste disposal radiological accident risk

Study	Expected Population Dose (person-rem/GW_e-year)	Dominant Risk Contributor
USDOE (1979)	Spent Fuel 2.1E-9 (whole body)	Waste Package dropped down shaft
	Glass HLSW 9.6E-12 (whole body)	
Erdman et al. (1979)	Glass HLSW 4.0E-5 (whole body)	Final Filter Failure

Table C.64 Transportation radiological accident risk[a]

Study	Plutonium Oxide	Spent Fuel	High Level Waste
Cohen and Dance (1975)	1.2E-3 to 1.7E-2 (total body)	3.5E-3 to 1.6 (total body)	
Erdman et al. (1979)	1.0E-3 (total body)	3.0E-5 (total body)	3.0E-3 (total body)
Fullwood and Jackson (1980)		3.0E-5 (total body)	1.0E-5 (total body)
USDOE (1979)*		5.0E-5 (total body)	1.1E-7 (total body)
USNRC (1977)*		1.4E-1 (total body)	
Berman et al. (1978)*			9.4E-3 (total body)
USAEC (1972); USNRC* (1975); USNRC (1976)		8.3E-3 (total body)	
Hodge and Jarrett* (1974)		1.2E-2 (total body)	5.1E-4 (total body)
USNRC (1976)*		2.3E-6 (total body)	5.4E-7 (total body)

(a) Measured in person-rem/GWe-year

Table C.65 Accident frequencies and population doses for transportation of spent fuel by rail and PuO_2 by truck in the nuclear fuel cycle (Cohen and Dance 1975)

Accident	Frequency (per shipment)	Population Dose for Generic Shipment (person-rem total body)
Spent Fuel		
Leakage of coolant from spent fuel cask	3E-4	5.8E-4
Release from a collision involving spent fuel	2E-8 to 9E-6	1.9E4
Release from a collision involving spent fuel followed by release of fuel from the cask	2E-10 to 9E-8	2.7E4
Plutonium Oxide		
Improperly closed plutonium oxide container	4E-4 to 1E-3	1.1
Release from a collision involving plutonium oxide	2E-9 to 3E-6	1.4E3
Criticality of plutonium oxide	2E-11 to 3E-8	2.5E4

Table C.66 Accident frequencies and population doses for transportation in the nuclear fuel cycle (Erdmann et al. 1979)

Accident	Frequency (per shipment)	Population Dose for Generic Shipment (person-rem total body)
Spent Fuel by Rail		
Loss of gases from inner cavity from rail accident	9E-6	1E-6
Loss of confinement and 50% fuel damage	4E-7	1E-1
Loss of confinement, 50% fuel damage, extensive fire	2E-9	2E3
Spent Fuel by Truck		
Loss of gas from inner cavity from truck accident	2E-5	5E-9
Loss of confinement and 50% fuel damage	2E-7	1E2
Loss of confinement, 50% fuel damage, extensive fire	2E-9	6E2
Plutonium Oxide by Truck		
Truck accident 1E-6 release fraction	1E-6	2
Truck accident 1E-4 release fraction	4E-11	2E1
Truck accident 1E-2 release fraction	6E-8	2E4
High-Level Waste by Rail		
Release to atmosphere and one canister breakage from rail accident	1E-5	7E2
Release to atmosphere and significant overheating	6E-8	6E3

Table C.67 Accident frequencies and population doses for rail transportation in the nuclear fuel cycle (Fullwood and Jackson 1980)

Accident	Frequency (per shipment)	Population Dose for Generic Shipment (person-rem total body)
Spent Fuel		
Loss of neutron shielding from a rail accident	2E-5	8E-7
Exposure of the inner spent fuel containing cavity	9E-6	1.7E-6
Exposure of the inner spent fuel containing cavity and 50% fuel damage	4E-7	0.5
Exposure of spent fuel with severe damage and fire	3E-9	1.7E3
High Level Waste		
Loss of neutron shielding from a rail accident	2E-8	5E-5
Release and extensive canister damage	3E-10	30
Release, extensive canister damage and fire	3E-12	3E3

Table C.68 Accident frequencies and population doses for rail transportation in the nuclear fuel cycle (PSE 1981)

Accident	Frequency (per year)	Population Dose for Generic Shipment (person-rem total body)
25-40 m fall	2E-6	2.8E-1
9-25 m fall	2E-5	2.8E-1
50-80 km/hr collision	2E-5	2.8E-1
80-100 km/hr collision	3E-4	2.8E-1
Collision and fire 1000°C >1 hr	8E-5	1.7E2
Collision and fire 800°C > 2 hr	2E-5	1.7E2
Fire 1000°C >1 hr	1E-4	2.0E-1
Fire 800°C >2 hr	2E-5	2.0E-1
Collision and closure errors	1E-4	1.1

Table C.69 Accident frequencies and population doses for rail transportation in the nuclear fuel cycle (Elder 1981)

Accident	Frequency (per shipment)	Population Dose for Generic Shipment (person-rem total body)
Rail accident and impact fails cask seal, causes loss of coolant and fuel fails	6.4E-6	6.8E2
Side impact fails pressure relief valve causing loss of coolant and fuel fails	1.2E-6	1.9E3
End impact fails pressure relief valve causing loss of coolant and fuel fails	6.4E-6	1.9E3
Side impact fails cask seal causing loss of coolant and fuel fails	1.2E-6	6.8E2

Table C.70 Normalized risk results for nuclear fuel cycle

Fuel Cycle Element	Expected Population Dose (Total Body person-rem/GWe-year) Original	Normalized	Reference
Milling	1.0E-3	2.7E-4	(Cohen and Dance 1975)
Conversion	5.6E-3	1.2E-2	(Cohen and Dance 1975)
Enrichment	3.7E-3	1.2E-2	(Cohen and Dance 1975)
Fuel Fabrication	1.0E-2	5.0E-3	(Cohen and Dance 1975)
MOX Fuel Refabrication	1.9E-2	1.2E-1	(Cohen and Dance 1975)
	4.0E-2	3.6E-2	(Erdmann et al. 1979)
	4.0E-7	3.3E-5	(Fullwood and Jackson 1980)
Fuel Reprocessing	---	3.1E-2	(Wood and Becar 1979)
	6.3E-3	3.2E-3	(Cohen and Dance 1975)
	---	5.6E-4	(PSE 1981)
	2.0E-4	2.2E-4	(Erdmann et al. 1979)
	---	1.5E-4	(Cooperstein et al. 1979)
	7.0E-5	5.4E-5	(Fullwood and Jackson 1980)
Spent Fuel Storage	---	1.8E-1	(PSE 1981)
	---	3.1E-2	(Wood and Becar 1979)
	1.7E-6	3.7E-5	(USDOE 1979)
	2.0E-5	2.7E-5	(Erdmann et al. 1979)
	8.9E-5	5.7E-6	(KBS 1977)
Solidified High Level Waste	2.3E-4	2.3E-4	(Smith and Kastenberg 1976)
Geologic Waste	4.0E-5	4.0E-5	(Erdmann et al. 1979)
Disposal (preclosure)	2.1E-9	2.1E-9	(USDOE 1979)
Transportation			
Plutonium Oxide	1.7E-2	6.6E-2	(Cohen and Dance 1975)
	1.0E-3	1.3E-3	(Erdmann et al. 1979)
Spent Fuel	---	1.6E-1	(Elder 1981)
	1.4E-1	1.6E-1	(USNRC 1977)
	1.6	7.8E-2	(Cohen and Dance 1975)
	1.2E-2	1.3E-2	(Hodge and Jarrett 1974)
	8.3E-3	9.3E-3	(USAEC 1972)
	---	7.1E-4	(PSE 1981)
	5.0E-5	5.6E-5	(USDOE 1979)
	3.0E-5	8.4E-6	(Erdmann et al. 1979)
	3.0E-5	8.4E-6	(Fullwood and Jackson 1980)
	2.3E-6	2.6E-6	(USNRC 1976)
High Level Waste	9.4E-3	4.2E-2	(Berman et al. 1978)
	5.1E-4	2.3E-3	(Hodge and Jarrett 1974)
	3.0E-3	8.4E-4	(Erdmann et al. 1979)
	1.0E-5	2.8E-6	(Fullwood and Jackson 1980)
	5.4E-7	2.4E-6	(USNRC 1976)

Table C.71 Capital equipment costs for fuel pellet fabrication (Mishima et al. 1983, Table 1)

Equipment/Procedure	Description	Manufacturer	Cost
2 Glove boxes	Inside floor dimensions: 5' 3" x 4' 11" 16 glove ports Box wall: 0.25" lead sandwiched between stainless steel sheets sheets 0.125" Windows: Leaded glass Gloves: Lead-loaded neoprene, 0.040" thick	Molitar Englewood, Colorado	$ 52,000
2 Balances	Cat. #3330-04 Load cell with remote controls and readouts. Dual range: To 3 kg, 0.1 g sensitivity; to 300 g, 0.01 g sensitivity	Scientech Boulder, Colorado	$ 4,100
Dry Granulator	ERWEKA Granulator Drive AR 400 Granulator TG 2/S	Chemical and Pharmaceutical Co., Inc. 225 Broadway, New York	$ 3,600
Blender	"Turbula:[a] Type T2C	Chemical and Pharmaceutical Co., Inc. 225 Broadway, New York	$ 3,000
Press	30 Ton Hydraulic, double acting Reservoir and pumps remote (outside glove box) All controls outside glove box	Western Sintering Richland, Washington	$110,000
Glove box installation	$10,000/box Engineering and Crafts: 425 h at $47/h		$ 20,000
Equipment installation	Press: 200 h at $46/h Other: 120 h at $46/h		$ 14,720
TOTAL			$207,420

[a] Registered trademark of Willy A. Bachofer, Manufacturer, Basil, Switzerland

Table C.72 Capital equipment costs for powder reconstitution during fuel fabrication (Mishima et al. 1983, Table 2)

Equipment/Procedure	Description	Manufacturer	Cost
2 Glove boxes	Inside floor dimensions: 5' 3" x 4' 11" 16 Glove ports Box wall: 0.25" lead sandwiched between stainless steel sheets 0.125" Windows: Leaded glass Gloves: Lead-loaded neoprene, 0.040" thick	Molitar Englewood, Colorado	$52,000
Balance	Cat. #3330-04 Load cell with remote controls and and readouts. Dual range: To 3 kg, 0.1 g sensitivity; to 300 g, 0.01 g sensitivity	Scientech Boulder, Colorado	$ 2,100
Dry Granulator	ERWEKA Granulator Drive AR 400 Grnaulator TG 2/S	Chemical Pharmaceutical Co., Inc. 225 Broadway, New York	$ 3,600
Furnace	Model #51442 Control model #59344 (remote) 4800 watts Exterior dimensions: 20" W x 20" H x 24.5" L	Lindberg Watertown, Wisconsin	$ 1,950
Mill rack and mills	Rack Model #764AV: 30 1/4" x 12 3/4" x 15 3/4" H 3 Mills: Rubber-lined steel size 1 Stainless steel balls, 0.5", 100 lbs	E. T. Horn La Mirada, California	$ 2,310
Glove box installation	$10,000/box Engineering and Crafts: 425 h at $47/hr		$20,000
Equipment installation	160 hr at $46/h		$ 7,360
TOTAL			$89,320

Table C.73 Start-up operation costs for fuel fabrication (Mishima et al. 1983, Table 3)

Process	Personnel	Job Description	Cost
Pellet fabrication	Engineer	120 h at $65/h	$16,400
		Prepare detailed operating procedures in conjunction with an operator. Supervise equipment shakedown.	
	Operator	120 h at $50/h	
		Operate equipment start-up and shakedown	
	---	Preparation of criticality specification: 40 h at $65/h	
	---	Radiation monitoring: Included in labor contract	
Powder reconstitution	Engineer	120 h at $65/hr	$16,400
		Prepare detailed operating procedures in conjunction with an operator. Supervise equipment shakedown.	
	Operator	120 h at $50/h	
		Operate equipment start-up and shakedown	

Table C.74 Process operation costs for fuel fabrication (Mishima et al. 1983, Table 4)

Process		
Pellet Fabrication	Estimate assumes 3 snifts/day processing a 100-kg minimum lot of PuO_2 powder.	
	Two operators/shift at $50/h/operator	
	Maximum 20 kg powder processed/day	
	Labor cost/kg	$120.00
	Radiation monitoring: Included in labor overhead.	
	Supplies/kg: Does not include items required for shipping as powder. Includes such items as stainless steel cylinders, neoprene lead-loaded gloves for replacement, organics.	1.50
	Only utilities: Electricity/kg	0.80 kWh
	Total pellet fabrication price/kg	$122.00
Powder Reconstitution	One operator/shift for 4 h at $50/hr	
	10 kg pellets processed to powder in 4 shifts	16 h labor
	Labor cost/kg	$ 80.00
	Radiation monitoring: Included in labor overhead.	
	Supplies/kg	$ 0.75
	Only utilities: Electricity/kg	12.0 kWh
	Total powder reconstitution price/kg	$ 81.00

Table C.75 Summary of dose equivalent estimates for fabricating PuO_2 powder to unfired pellets during fuel Fabrication (Mishima et al. 1983, Table 9)

	Total Dose Equivalent for Three-Person Crew Processing 100 kg of PuO_2 (man-rem)	
	Average of Light Water Reactor Plutonium Produced in 1985	Low-Exposure Plutonium
Contact or hand exposure (gamma only)	67.0	18.0
Whole body dose equivalent including room background		
Average	0.95	0.14
Range based on variations in room background	(0.87 to 1.1)	(0.11 to 0.15)

Table C.76 Summary of dose equivalent estimates for reconstituting unfired PuO_2 pellets back to powder during fuel fabrication (Mishima et al. 1983, Table 10)

	Total Dose Equivalent for Two-Person Crew Processing 100 kg of PuO_2 (man-rem)	
	Average of Light Water Reactor Plutonium Produced in 1985	Low-Exposure Plutonium
Contact or hand exposure (gamma only)	64.0	17.0
Whole-body dose equivalent including room background		
Average	0.19	0.038
Range based on variations in room background	(0.14 to .26)	(.03 to .06)

Table C.77 Accident source terms and doses from uranium mill accidents (McGuire 1988, Table 3)

	Tornado		Tailing Pond Release		Fire in Solvent Extraction Circuit		Failure of the Air Cleaning System Serving the Yellowcake Drying Area	
Reference	Release	Dose	Release	Dose	Release	Dose	Release	Dose
GEIS	11,400 kg U total < 11,400 kg U respirable	< 2.2 x 10^{-7} rem to lungs at 500m	1400 tons solid 14,000,000 gal. liquids	Small. Cleanup assumed	< 13 kg U < 0.65 kg thorium[a]	< 1.36 rem[a] to bone at 500 m	11 kg insoluble U oxides over 8 hours	86 mrem to lung at 2000 m
Sand Rock DES	4550 kg U total < 4550 kg U respirable	< 1.1 x 10^{-7} rem at 4000m (max. dose)	Same as GEIS	-	< 1.1 kg U	10^{-7} rem to bone at 8000 m (nearest residence)	12 kg insoluble U oxides over 8 hours	10^{-2} rem to lung at 8000 m (nearest residence)
This Report	-	-	-	-	1.3 kg U	0.01 to 0.1 rem EDE	-	-

References

GEIS: "Final Generic Environmental Impact Statement on Uranium Milling," NUREG-0706, Volume 1, pp 7-1 to 7-20, September, 1980.
Sand Rock DES: "Draft Environmental Statement Related to the Operation of Sand Rocks Mill Project," NUREG-0889, pages 5-1 to 5-12, March, 1982.

[a]The dose value from GEIS is in error The solvent extraction was assumed to contain 5% as much Th-230 as uranium by weight. The value should have been 5% by activity This error causes the dose to be overestimated by a factor of about 50,000 times

Table C.78 Offsite doses calculated for fuel fabrication plants (McGuire 1988, Table 9)

		Criticality		UF_6-low enrich.		UF_6-high enrich.
Analysis	Key Assumptions	Effective DE	Thyroid DE	Effective DE	Bone DE	Effective DE
NUREG-1140	Building size: 250 m^2 Wind: F, 1 m/sec Release height: ground	0.5 to 2.6 rems at 100 m	1.1 to 8.2 rems at 100 m (child's thyroid)		-	0.2 to 1.5 rem at 100 m
Combustion Engineering	Building size: 0 Wind: F, 1 m/sec Release height. stack	0.27 rem at 800 m	1.7 rems at 800 m	0.05 rem at 800 m	0.82 rem at 800 m	-
Exxon	Building size: 0 Wind: F, 1m/sec Release height: ground	0.009 rem at 2000 m	4.5 rems at 2000 m	0.11 rem at 2000 m	1.7 rems at 2000 m	-
NFS, Erwin	Building size: 0 Wind: G, 0.5 m/sec Release height: same level as residence	-	5 rems at 1000 m	-	-	1 rem at 1000 m

DE = dose equivalent EDE = effective DE

Table C.79 Dose commitments from plutonium fuel fabrication facility accidents (McGuire 1988)

Type of accident	Dose commitment (rem)
Criticality	0.36 (thyroid)
Fire	0.02 (bone)
Explosion	0.02 (bone)

Table C.80 Maximum offsite individual dose commitments (Rem) from spent fuel reprocessing facility accidents (McGuire 1988)

"Maximum Offsite Individual Dose Commitment (rem)	
Accident	PWR MOX Fuel
Criticality	0.056 (thyroid)
Waste Concentrator Explosion	0.0069 (bone)
Pu Evaporator Explosion	0.019 (bone)
Fire	0.0135 (bone)

Table C.81 Calculated releases and doses from spent fuel storage accidents (McGuire 1988, Table 10)[(6)]

Reference	Accident	Kr-85 Release	Skin Dose	Effective Dose Equivalent	I-129 Release	Thyroid Dose
Storage in pools: Generic Environmental Impact Statement, NUREG-0575	Tornado driven missile followed by calm	19,000 Ci	0.06 rem at 275 m	Not calculated	0.00006 Ci	0.03 rem at 275 m
Storage in pools: GE-Morris SER, NUREG-0709	Drop of a fuel storage basket	6,000 Ci	Not calculated	0.016 rem at 150 m	0.00008 Ci	0.0004 rem at 150 m
Dry cask, drywell, or dry vault storage: NUREG-1140	Removal of cask lid with all fuel elements ruptured	8,000 Ci	Not calculated	0.003 rem at 100 m	0.004 Ci	0.005 to 0.04 rem within 100 m (child)

Table C.82 Maximum possession limits, release fractions, and doses due to a major facility fire for radiopharmaceutical manufacturing (McGuire 1988, Table 14)

Radioactive material	Maximum licensed possession limit (Ci)	Licensee	Release fraction	Effective dose equivalent, rem**
H-3	150,000	NEN*	0.5	0.1 to 10.
C-14	500	NEN-Boston	0.01***	0 to 0.01
P-32	500	NEN	0.5	0.04 to 4.
S-35	1,000	NEN	0.5	0.01 to 1.
Ca-45	50	NEN	0.01	0 to 0.003
Cr-51	100	NEN	0.01	0
Fe-55	200	NEN	0.01	0 to 0.005
Ni-63	1,000	NEN	0.01	0.001 to 0.06
Se-75	100	NEN	0.01	0 to 0.008
Kr-85	10,000	NEN	1.0	0 to 0.002
Rb-86	50	NEN	0.01	0 to 0.003
Sr-90	500	NEN	0.01	0.05 to 5.
Mo-99	2,000	NEN/Squibb	0.01	0.001 to 0.08
Ru-103	25	NEN	0.01	0 to 0.002
Sn-113	100	NEN	0.01	0 to 0.01
I-125	100	NEN/Mallinckrodt	0.5	0.3 to 30. (child's thyroid)
I-131	500	Mallinckrodt	0.5	5 to 500. (child's thyroid)
Xe-133	1,000	NEN	1.0	0 to 0.001
Cs-134	25	NEN	0.01	0 to 0.01
Cs-137	500	NEN	0.01	0.002 to 0.2
Ce-141	50	NEN	0.01	0 to 0.004
Yb-169	50	NEN	0.01	0 to 0.004
Tm-170	25	NEN	0.01	0 to 0.006
Au-198	200	NEN	0.01	0 to 0.008

*NEN = New England Nuclear, North Billerica, Mass.

**zero in the dose column indicates a dose of less than one millirem.

***Non-carbon dioxide release fraction.

Table C.83 Maximum possession limits, release fractions, and doses due to a major facility fire for a radiopharmacy (McGuire 1988, Table 15)

Radioactive material	Maximum licensed possession limit (Ci)	Chemical forms	Release fraction	Dose equivalent, rem
H-3	0.05 Ci	In vitro test kits	0.5	0
C-14	0.05	In vitro test kits	0.01[a]	0
Cr-51	0.15	Labeled serum, sodium chromate	0.01	0
Co-58	0.15	Cyanocobalamin (vitamin B12)	0.001	0
Fe-59	0.15	Chloride, citrate, sulfate	0.01	0
Se-75	0.1	Labeled compound	0.01	0
Sr-90	0.5	Nitrate, chloride	0.01	0 to 0.006
Mo-99/Tc-99m	75.	Mo-99/Tc-99m generators (liquid)	0.01	0 to 0.004
I-125	0.15	Na I, fibrogen, diagnostic kits	0.5	0.001 to 0.1 (child's thyroid
I-131	0.75	Na I, labeled organic compounds	0.5	0.007 to 0.7 (child's thyroid)
Xe-133	1.	Gas or saline	1.0	0

Note: sealed sources are not included.
Reference: Sutter report.

[a]Non-carbon dioxide release fraction.

Table C.84 Maximum possession limits, release fractions, and doses due to a major facility fire for sealed source manufacturing (McGuire 1988, Table 16)

Radioactive material	Maximum licensed posession limit (Ci)	Form	Licensee	Release fraction	Effective dose equivalen rems
H-3	100,000 Ci	volatile	Safety Light	0.5	0.06 to 6
C-14	50		Amersham	0.01[a]	0 to 0.00
Co-60	20,000	75% metallic pellets 25% sealed sources	Automation Ind.	0.0001	0.004 to 0.4
Kr-85	1,500	noble gas	3M	1.0	0
Sr-90	3,000	1000 Ci in solution in 0.1 liter of 0.1 N HCl also, sealed sources	3M	0.01	0.3 to 33
Sb-124	50		Monsanto	0.01	0 to 0.01
I-125	100	5 Ci in KOH liquid 5 Ci on resin beads	3M	0.5	0.7 to 70. (child's thyroid)
Cs-137	10,000		Tech/Ops	0.01	0.03 to 3.
Pm-147	3,500	800 Ci in solution in 0.1 liter of 0.1 N HCl also, sealed sources	3M	0.01	0.008 to 0.
Yb-169	100	5 Ci liquid Yb chelate	3M	0.5	0.004 to 0.4
Tm-170	5,000		Tech/Ops	0.01	0.01 to 1.
Ta-182	200	metallic or carbide	Tech/Ops	0.01	0 to 0.001
Ta-183	2,000	metallic or carbide	Tech/Ops	0.01	0 to 0.001
Ir-192	50,000	solid metal or sealed source	Tech/Ops	0.0001	0.001 to 0.1
Tl-204	50		Monsanto	0.01	0 to 0.001
Bi-210	200	metal slugs	3M	0.001	0 to 0.03
Po-210	4,000	up to 1500 Ci in 40 liters of 2M HNO_3; up to 2500 Ci in waste primarily as microspheres	3M	0.01 0.001	1. to 100. (per 1500 Ci) 0.2 to 20. (per 2500 Ci)
Np-237	0.1		Monsanto	0.001	0 to 0.04
Pu-238, 236, 239, 240, 241, 242	199 g	250 Ci as unsealed powder oxide	Monsanto	0.001	0.75 to 75. (per 250 Ci)
Am-241	6,000	250 Ci as unsealed powder oxide; remainer as sealed sources	Monsanto	0.001	1.2 to 120. (per 250 Ci)
Cm-242	600		Monsanto	0.001	0.1 to 10.
Cm-243	10		Monsanto	0.001	0.03 to 3.0
Cm-244	600		Monsanto	0.001	1.5 to 150.
Cf-252	10 mg	solid pellet	Monsanto	0.001	0.006 to 0.6

[a]Non-carbon dioxide release fraction.

Table C.85 Maximum possession limits, release fractions, and doses due to a major facility fire for university research laboratories (McGuire 1988, Table 17)

Radioactive material	Maximum licensed possession limit (Ci)	Release fraction	Effective dose equivalent, rems
H-3	3000	0.5	0.002 to 0.2
C-14	10	0.01*	0
P-32	5	0.5	0 to 0.04
S-35	5	0.5	0 to 0.01
Ni-63	1	0.01	0
Sr-90	0.5	0.01	0 to 0.005
Mo-99/Tc-99m	10	0.01	0
I-125	8	0.5	0.06 to 5.5 (child's thyroid)
I-131	1	0.5	0.01 to 1. (child's thyroid)
Xe-133	10	1.	0
Po-210	10	0.01	0.009 to 0.9
Am-241	0.5	0.001	0.003 to 0.3
Cm-244	1	0.001	0.003 to 0.3
Cf-252	0.1	0.001	0 to 0.01

*Non-carbon dioxide release fraction.

Table C.86 Waste warehousing airborne releases and doses due to a major facility fire (McGuire 1988, Table 18)

Radioactive material	Quantity present (Ci)	Release fraction	Effective dose equivalent, rem
H-3	6200	0.5	0.004 to 0.4
C-14	160	0.01*	0 to 0.004
P-32	160	0.5	0.01 to 1.
S-35	120	0.5	0.002 to 0.2
Cr-51	60	0.01	0
I-125	280	0.5	4 to 400. (child's thyroid)
I-131	20	0.5	0.4 to 40. (child's thyroid)

*Non-carbon dioxide release fraction.

Table C.87 Alternative disposal standards for uranium mill tailings (EPA 1983, Table S.1)

Longevity Requirement	Radon Control after Disposal (pCi/m^2s)				
	No Radon Requirement	60	20	6	2
No Controls	A				
Active control for 100 years	B1	B2	B3		
Passive control for 1000 years	C1	C2	C3	C4	C5
Passive control for 1000 years, with improved radon control during operations for new piles		D2	D3	D4	D5

Table C.88 Alternative standards and control methods for existing uranium mill tailings piles (EPA 1983, Table 4.2)

		Control Method Characteristics					
Alternative Standard	Control Method Designation	Earth Cover Thickness (m)	Slope	Rock on Slopes	.5m Pebbly Soil on Top	Maintenance	Landscaping
A	-						
B1	B1-E	0.5	3:1			100 years	X
B2	B2-E	1.5	3:1			100 years	X
B3	B3-E	2.4	3:1			100 years	X
C1	C1-E	0.5	5:1	X	X		
C2	C2-E	1.5	5:1	X	X		
C3	C3-E	2.4	5:1	X	X		
C4	C4-E	3.4	5:1	X	X		
C5	C5-E	4.3	5:1	X	X		
D2	Same as C2						
D3	Same as C3						
D4	Same as C4						
D5	Same as C5						

Table C.89 Alternative standards and control methods for new uranium mill tailings piles (EPA 1983, Table 4.3)

		Control Method Characteristics							
Alternative Standard	Control Method Designation	Earth Cover Thickness (m)	Slope	Rock on Slopes	.5m Pebbly Soil on Top	Maintenance	Put Below Grade	Liner	Landscaping
A	A-N	Construction of initial embankments only							
B1	B1-N	.5	3:1			100 years		X	X
B2	B2-N	1.5	3:1			100 years		X	X
B3	B3-N	2.4	3:1			100 years		X	X
C1	C1-N	.5	5:1	X	X			X	
C2	C2-N	1.5	5:1	X	X			X	
C3	C3-N	2.4	5:1	X	X			X	
C4	C4-N	3.4	5:1	X	X			X	
C5	C5-N	4.3	5:1	X	X			X	
D2	D2-N	1.5					X	X	X
D3	D3-N	2.4					X	X	X
D4	D4-N	3.4					X	X	X
D5	D5-N	4.3					X	X	X

Table C.90 Summary of values for alternative disposal standards for uranium mill tailings (EPA 1983, Table S.2)

Alternative Standards	Stabilization: Chance of Misuse	Stabilization: Tailings Erosion Avoided (years)	Radon Control: Maximum Risk[a] of Lung Cancer (% reduction)	Radon Control: Deaths Avoided[b] First 100 years	Radon Control: Deaths Avoided 1,000 years	Radon Control: Deaths Avoided Total	Water Protection Longevity (years)
A	Very likely	0	2 in 10^2(0)	0	0	0	0
B1	Likely	Hundred	1 in 10^2(50)	300	1200	1200	100
B2	Less Likely	Hundreds	4 in 10^3(80)	480	1800	1800	100
B3	Less Likely	Hundreds	1 in 10^3(95)	570	2100	2100	100
C1	Likely	Hundred	1 in 10^2(50)	300	3000	Thousands	100
C2	Less Likely	Thousands	4 in 10^3(80)	480	4800	Many 1000's	100's
C3	Unlikely	Thousands	1 in 10^3(95)	570	5700	Tens of 1000's	1000
C4	Very Unlikely	Many thousands	3 in 10^4(98.5)	590	5900	Tens of 1000's	>1000
C5	Very Unlikely	Many thousands	1 in 10^4(99.5)	600	6000	Tens of 1000's	>1000
D2	Unlikely	Thousands	4 in 10^3(80)	480	4800	Many 1000's	1000
D3	Unlikely	Many thousands	1 in 10^3(95)	570	5700	Tens of 1000's	1000
D4	Very unlikely	Many thousands	3 in 10^4(98.5)	590	5900	Tens of 1000's	>1000
D5	Very unlikely	Many thousands	1 in 10^4(99.5)	600	6000	Tens of 1000's	>1000

(a) Lifetime risk of fatal cancer to an individual assumed to be living 600 meters from the center of a model tailings pile. The estimates of benefits assume no credit for engineering factors required to provide "reasonable assurance" of design compliance for the specified radon control level and period of longevity.

(b) These estimates pertain to the control of 26 existing piles and 9 projected new pile equivalents. Of the approximately 600 deaths which are estimated to occur in the first 100 years under no control conditions, about 500 are the result of the existing tailings and 100 are due to future tailings.

Table C.91 Cost-effectiveness of control methods for uranium mill tailings (EPA 1983, Table 4.8)

Control Method	Effectiveness Index	Total Cost (10^6 1983 $)	Average Cost	Incremental Cost
2 million MT Existing Pile				
A	0	0	---	---
B1	1.0	4.2	Eliminated from consideration	
B2	1.8	6.9	Eliminated from consideration	
B3	3.1	9.2	Eliminated from consideration	
C1	4.3	3.2	.7	.7
C2	6.9	5.9	.9	1.0
C3	7.9	8.3	1.1	2.4
C4	8.6	10.9	1.3	3.7
C5	9.2	13.3	1.4	4.0
7 million MT Existing Pile				
A	0	0	---	---
B1	1.0	6.4	Eliminated from consideration	
B2	1.8	10.4	Eliminated from consideration	
B3	3.1	14.0	Eliminated from consideration	
C1	4.3	6.3	1.5	1.5
C2	6.9	10.5	1.5	1.6
C3	7.9	14.3	1.8	3.8
C4	8.6	18.5	2.2	6.0
C5	9.2	22.2	2.4	6.2
22 million MT Existing Pile				
A	0	0	---	---
B1	1.0	10.8	10.8	10.8
B2	1.8	17.3	Eliminated from consideration	
B3	3.1	23.0	Eliminated from consideration	
C1	4.3	13.6	3.2	0.8
C2	6.9	20.6	3.0	2.7
C3	7.9	26.8	3.4	6.2
C4	8.6	33.8	3.9	10.0
C5	9.2	40.0	4.3	10.3
8.4 million MT New Pile				
A	0.0	1.3	---	---
B1	1.0	11.4	Eliminated from consideration	
B2	1.8	15.0	Eliminated from consideration	
B3	3.1	19.0	Eliminated from consideration	
C1	4.3	11.4	2.7	2.3
C2	6.9	16.0	2.3	1.8
D2	7.5	32.3	Eliminated from consideration	
C3	7.9	20.0	2.5	4.0
D3	8.3	35.5	Eliminated from consideration	
C4	8.6	24.3	2.8	6.1
D4	9.0	39.5	Eliminated from consideration	
C5	9.2	28.4	3.1	6.8
D5	9.6	43.1	4.5	36.8

Table C.92 Summary of costs in millions of 1983 dollars for alternative disposal standards for uranium mill tailings (EPA 1983, Table S.3)

Alternative Standard	Assumed Control Method	Cover Thickness (meters)	Industry Costs, Undiscounted			Present Worth Costs (10% discount rate)
			Existing Tailings	Future Tailings	Total	
A	No control	-	0	4	4	1
B1	Above-grade,	0.5	155	84-474	239-629	141-319
B2	3:1 slope,	1.5	253	98-549	351-802	219-424
B3	irrigation and maintenance for 100 years	2.4	338	114-632	452-970	288-524
C1	Above-grade,	0.5	152	124-474	276-626	157-316
C2	5:1 slope,	1.5	253	145-570	398-823	240-433
C3	rock cover on	2.4	343	165-653	508-996	314-537
C4	slopes, 0.5 m	3.4	443	186-744	629-1187	397-651
C5	of pebbly soil on top of pile	4.3	532	215-829	747-1361	474-755
D2	Same as C for	1.5	253	184-837	437-1090	249-546
D3	existing piles	2.4	343	201-906	544-1249	323-644
D4	and staged	3.4	443	221-989	664-1432	406-755
D5	disposal below-grade for new piles	4.3	532	252-1065	784-1597	483-855

Table C.93 Estimated risks from spent fuel pool fires (Jo et al. 1989, Table 3.1)

Event	Probability	
	PWR Plant	BWR Plant
Structural Failure of Pool Resulting from Seismic Events	1.8E-6/Ry*	6.7E-6/Ry
Probability of a Cask Drop Caused by Human Error	3.1E-4/Ry	3.1E-4/Ry
Reduction in Failure Rate for Cask Drop Implementing Generic Issue A-36	1.0E-3	1.0E-3
Conditional Probability of Pool Structural Failure Given a Cask Drop	1.0	1.0
Conditional Probability of a Clad Fire Given a Pool Structural Failure**	1.0	0.25
Frequency of Spent Fuel Pool Fire from Seismic Initiator	1.8E-6/Ry	1.68E-6/Ry
Frequency of Spent Fuel Pool Fire from a Cask Drop Initiator	3.1E-7/Ry	7.75E-8/Ry

*Ry = Reactor year.
**NUREG/CR-4982, p. 75.

Table C.94 Offsite consequence calculations for spent fuel pool fires (Jo et al. 1989, Table 3.2)

Case	Characterization	Source Term*	Population	Public Health Dose (person-rem)	Offsite Property Damage ($1983)
1	Average Case	Last fuel discharged 90 days after discharge	340 persons/mile^{2}	7.97×10^{6}	3.41×10^{9}
2	Worst Case	Entire pool inventory 30 days after discharge	Zion population (roughly 860 persons/mile^{2})	2.56×10^{7}	2.62×10^{10}

*From NUREG/CR-4982.

Table C.95 Onsite property damage costs in dollars per spent fuel pool accident (Jo et al. 1989, Table 3.3)

Item	Best Estimate	Worst Case
Cleanup and Decontamination	1.65E8	1.65E8
Repair	7.2E7	7.2E7
Replacement Power	8.67E8	1.66E9
Total Number of Operating Years Remaining	29.8 years	29.8 years
Number of Years Plant is Out of Service	5 years	7 years
Expected Dollar Loss	8.24E9	1.29E10

Table C.96 Incremental storage costs in 1983 dollars associated with limited low-density racking in the primary spent fuel pool (Jo et al. 1989, Table 3.6)

STORAGE OPTION	PER UNIT			ALL PLANTS		
	0%*	5%	10%	0%*	5%	10%
POOL	2.17+7	1.67+7	1.28+7	2.34+9	1.80+9	1.38+9
DRYWELL	9.13+6	8.24+6	6.85+6	9.86+8	8.90+8	7.40+8
VAULT	2.07+7	1.67+7	1.28+7	2.24+9	1.80+9	1.38+9
CASK	1.20+7	1.22+7	1.05+7	1.30+9	1.32+9	1.13+9
SILO	1.56+7	1.22+7	9.35+6	1.68+9	1.32+9	1.01+9

*Zero % discount rate corresponds to the case where additional storage capacity is built now.

Notes: 1. These costs include the cost of in-pool reracking and the incremental costs associated with the change in additional storage requirements resulting from the decrease in primary pool capacity.
2. Assuming the extra storage capacity is built when required, two discount rates are applied.

Table C.97 Summary of Parameters affecting attributes for the spent fuel pool inventory reduction option (Jo et al. 1989, Table 3.8)

Attributes	Factors Affecting Attributes	Description	Quantification	References
Public Health Dose Reduction	A. Pool Failure Probability	Seismic Structural Failure High - PWR - BWR Low Failure due to Cask Drop High - PWR - BWR Low Others	 1.8×10^{-6} /Ry 1.68×10^{-6} ≈ 0 3.1×10^{-7} /Ry 7.75×10^{-8} ≈ 0 ≈ 0	Table 3.1 Ref. 2 Ref. 2
	B. Number of Pools Involved	PWR BWR	69 39	DOE/RL-87-11
	C. Average Remaining Life-Time of Plant	PWR BWR	29.8 27.9	DOE/RL-87-11
	D. Radioactive Inventory Release	Worst Case Best Estimate	Total Inventory 30 days After Discharge Last Fuel Discharge 90 Days After Discharge	NUREG/CR-4982
	E. Meteorology		Zion	
	F. Population	Worst Case U.S. Average	Zion (860 people/sq. mi.) 340 people/sq. mi.	
	G. Risk Reduction	80% Sequence Frequency Reduction	80%	NUREG/CR-4982
Reduction of Occupational Exposure --Accidental			Considered to be insignificant compared to Public Health Impact	
Reduction of Occupational Exposure --Routine			No significant change expected	
Offsite Property Damage	A, B, C, D, E, F, G Ecomony Discount Rate	Same as those of Public Health	 Zion 10%	
Onsite Property Damage	Decontamination, Refurbishment and Replaceement Power Time. Discount Rate		5 years 10%	NUREG/CR-3568 EPRI NP-3380
Reg. Efficiency	Unaffected			
Improvement in Knowledge	Unaffected			
Industry Implementation and Operation	Additional Storage Option and Reracking Cost. Discount Rate	High (Pool Option) Low (Drywell Option)	 10%	DOE/RL-87-11 EPRI NP-3365
NRC Development /Implementation/ Operation	Unaffected			

Table C.98 Summary of industry-wide value-impact analysis of the spent fuel pool inventory reduction option[(a)] (Jo et al. 1989, Table 3.9)

Attributes	Dose Reduction (Person-Rem) Best Estimate	Dose Reduction (Person-Rem) High Estimate[(b)]	Evaluation ($1983) Best Estimate	Evaluation ($1983) High Estimate[(b)]
Public Health	4.00×10^4	1.28×10^5	4.00×10^7	1.28×10^8
Occupational Exposure				
/Accidental	≈ 0	≈ 0	≈ 0	≈ 0
/Routine	≈ 0	≈ 0	≈ 0	≈ 0
Offsite Property			1.42×10^6	2.22×10^6
Onsite Property			5.54×10^6	4.25×10^7
Regulatory Efficiency			Unaffected	
Improvement in Knowledge			Unaffected	
Industry Implementation and Operation			-1.38×10^9	-1.13×10^9
NRC Development, Implementation and Operation			Unaffected	
Net Benefit ($)			-1.33×10^9 (c)	-9.57×10^8
Benefit ($)/Cost ($) Ratio			0.035(c)	0.15
Ratio of Public Dose Reduction per Million Dollars Cost (Person-rem/$\$10^6$)			29.0(c)	113.0
Cost of Implementation per Averted Person-rem ($/Person-rem)			3.45×10^4(c)	8.83×10^4

(a) Based upon a U.S. pool population of 108.
(b) High estimate is based on the 'Worst Case' source term release and Zion site population (see Table 3.2).
(c) Based on 1988 dollars, the Best Estimate Net Benefit, Benefit/Cost Ratio, Public Dose Reduction per Million Dollars Cost and Cost per Averted Person-rem would be -1.47×10^9 Dollars, 0.032, 26.4 Person-rem and 3.79×10^4 Dollar/Person-rem, respectively. Cost escalation during 1983-1988 was assumed to be 9.8% (Reference 17).

Table C.99 Failure frequency for generic spent fuel pool cooling and makeup systems (Jo et al. 1989, Table 4.1)

		Failure Rates Per Demand					
		Cooling System		Makeup System			Total Failure Frequency Per System Year
System Type	Description	Train 1**	Train 2	Train 1	Train 2	Fire System	
A.	Minimum SRP Requirement	0.1	0.05	0.015	0.05	--	3.8×10^{-6}
B.	Minimum SRP Requirement With Credit for Fire System	0.1	0.05	0.015	0.05	0.05	1.9×10^{-7}
C.	Old Existing Plant with Both Cooling Pumps Required 30% of Timett	0.1	0.3	0.015	0.05	--	2.2×10^{-5}
D.	Old Existing Plant With Credit for Fire System	0.1	0.3	0.015	0.05	0.05	1.1×10^{-6}

*Reference 1.
**Units of failure per system year.

SRP = Standard Review Plan

Table C.100 Value-impact for generic improvements to the spent fuel pool cooling system* (Jo et al. 1989, Table 4.2)

System	Description	Improvement	Improvement Cost (1983$)	Expected Averted Cost (1983$)	Benefit/ Cost Ratio
A.	Minimum SRP	1. Additional pump	50,000	None	0.0
		2. Additional train	1.0E6	545 to 6640	<<0.01
B.	Minimum SRP Requirement With Credit for Fire System	1. Additional pump	50,000	None	0.0
		2. Additional train	1.0E6	27 to 330	0.0
C.	Old Existing Plant With Both Cooling Pumps Required 30% of Time	1. Additional pump	50,000	2500 to 30,400	.05 to 0.61
		2. Additional train	1.0E6	3160 to 38,550	.003 to 0.04
D.	Old Existing Plant With Credit for Fire System	1. Additional pump	50,000	125 to 1500	.0025 to 0.03
		2. Additional train	1.0E6	159 to 1940	<.002

*Quantification reflects a single spent fuel pool.

System A - Minimum cooling and makeup system required by the SRP:[13] One full capacity cooling train with redundant active components (i.e., redundant valves and pumps). One Category I makeup system and one backup pump or system (not required to be Category I) which can be aligned to a Category I water supply.

System B - Minimum cooling and makeup system with credit for makeup from fire system (Note that some plants may identify the fire system as the backup in System A).

System C - Typical older system comparable to current SRP requirements: One cooling train with backup active components (but backup components are required to supplement cooling about 30% of time[11]); One safety grade makeup train and one non-safety grade makeup system.

System D - Typical older system (System C) with third makeup train available (e.g., fire system).

Table C.101 Offsite property damage and health costs per spent fuel pool accident* (Jo et al. 1989, Table 5.1)

Case	Characterization	Source Term	Population	Use of Spray System	Radiological Dose (person-rem)	Property Damage Costs $
1	Average Case	Last fuel discharged 90 days after discharge	340 persons/ sq. mile	No	7.97E6	3.41E9
2	Average Case	Last fuel discharged 90 days after discharge	340 persons/ sq. mile	Yes	1.25E6	6.16E7
3	Worst Case	Entire pool density 30 days after discharge	Zion Population (roughly 860 persons/sq. mile)	No	2.56E7	2.62E10
4	Worst Case	Entire pool density 30 days after discharge	Zion Population (roughly 860 persons/sq. mile)	Yes	6.78E6	4.48E8

*MACCS Calculations.

Table C.102 Summary of industry-wide value-impact analysis of the spent fuel pool post-accident spray system[a] (Jo et al. 1989, Table 5.2)

Attributes	Total Dose Reduction (Person-rem) Best Estimate[b]	High Estimate[b]	Total Monetary Risk Reduction ($1983) Best Estimate[b]	High Estimate[b]
Public Health	4.20E4	1.18E5	4.20E7	1.18E8
Occupational Exposure	~ 0	~ 0	~ 0	~ 0
Offsite Property			6.77E6	5.20E7
Onsite Property			~ 0	~ 0
Industry Implementation and Operation			-1.08E8	-1.08E8
Net Benefit ($)			-5.92E7[c]	6.2E7
Benefit ($)/Cost ($) Ratio			0.45[c]	1.57
Ratio of Public Dose Reduction per Million Dollars Cost (Person-rem/$$10^6$)			3.89E2[c]	1.09E3
Cost of Implementation per Averted Person-rem ($/Person-rem)			2.57E3[c]	9.15E2

(a) Population of 108 spent fuel pools.
(b) See Table 3.2 for source terms and demographic assumptions.
(c) Based on 1988 dollars, Best Estimate Net Benefit, Benefit/Cost Ratio, Public Dose Reduction per Million Dollar Cost and Cost per Averted Person-rem would be -6.92E7 dollars, 0.42, 354 Person-rem and 2.82E3 dollars/person-rem,respectively. Cost escalation during 1983-1988 was assumed to be 9.8% (Reference 17).

Table C.103 Facility descriptors for accident analysis (Ayer et al. 1988, Table 2.1)

Descriptor
Accident Compartment
Wall material
Ceiling material
Floor material
Thickness of wall
Thickness of ceiling
Thickness of floor
Length of room
Width of room
Height of room
Volume of room
Vessels in Accident Compartment
Type of vessel (pressurized, unpressurized)
Construction material
Height of vessel
Exposed width
Elevation of vessel
Weight of empty vessel (or wall thickness and density)
Failure pressure
Ventilation System
Schematic
Elevation of inlet duct to compartment
Filter type
Filter efficiency
Blower performance curve
Duct height
Duct equivalent diameter
Duct heat transfer area
Duct floor area
Duct length
Duct X-sectional flow area
Duct Wall properties
Outside emissivity
Outside absorptivity
Density
Thermal conductivity
Specific heat
Thickness
Volume of rooms, cells, plenums
Alternate Flow Paths
Time of generation
Elevation of path
Size of opening (equivalent area circular diameter)
Pressure on other side

Table C.104 Fuel manufacturing process descriptors (Ayer et al. 1988, Table 3.6)

Descriptor
Radioactive Material Inventories
Form
Containment
Location
Quantity
Properties
Radioactivity
Radioactive Material in Containers
Volume of Powder
Moisture Content of Powder
Volume of Air in Closed Containers
Mass of Liquid
Volume of Liquid
Hazardous Material Inventories
Location
Quantity
Surface Area
Material Type
Energy
Process Parameters
Initial Temperatures Compartment
Radioactive Powders in Closed Containers
Radioactive Liquids in Closed Containers
Radioactive Liquids in Open Containers
Outside of Vessels
Duct Wall
Initial Pressures in
Inlet Duct
Compartment
Exit Duct

Table C.105 Fuel reprocessing process descriptors (Ayer et al. 1988, Table 3.8)

Descriptor
Radioactive Material Inventories
Form
Location
Containment
Quantity
Properties
Radioactivity
Radioactivity
Containment
Radioactive Material in Containers
Volume of Powder
Moisture Content of Powder
Volume of Air in Closed Containers
Mass of Liquid
Volume of Liquid
Hazardous Material Inventories
Energy
Location
Quantity
Surface Area
Material Type
Process Parameters
Initial Temperatures Compartment
Radioactive Powders in Closed Containers
Radioactive Liquids in Closed Containers
Radioactive Liquids in Open Containers
Outside of Vessels
Duct Wall
Solvent Stream
Initial Pressures in
Inlet Duct
Compartment
Exit Duct
Solvent Stream

Table C.106 Waste storage/solidification process descriptors (Ayer et al. 1988, Table 3.10)

Descriptor
Radioactive Material Inventories
Form
Containment
Location
Quantity
Properties
Radioactivity
Radionuclide Volatility
Radioactive Material in Containers
Volume of Powder
Moisture Content of Powder
Volume of Air in Closed
Mass of Liquid
Volume of Liquid Containers
Hazardous Material Inventories
Location
Quantity
Surface Area
Material Type
Energy
Process Parameters
Initial Temperatures Compartment
Radioactive Powders in Closed Containers
Radioactive Liquids in Closed Containers
Radioactive Liquids in Open Containers
Outside of Vessels
Glass Surface
Duct Wall
Initial Pressures in
Inlet Duct
Compartment
Exit Duct

Table C.107 Spent fuel storage process descriptors (Ayer et al. 1988, Table 3.11)

Descriptor
Radioactive Material Inventories
Form
Containment
Location
Quantity
Properties
Radioactivity
Radioactive Material in Containers
Volume of Air in Closed Containers
Mass of Liquid
Volume of Liquid
Hazardous Material Inventories
Location
Quantity
Surface Area
Material Type
Energy
Process Parameters
Initial Temperatures
Compartment
Radioactive Powders in Closed Containers
Radioactive Liquids in Closed Containers
Radioactive Liquids in Open Containers
Outside of Vessels
Duct Wall
Initial Pressures in
Inlet Duct
Compartment
Exit Duct

Table C.108 Behavior mechanisms for airborne particles (Ayer et al. 1988, Table 4.1)

Mechanism	Description	Influencing Elements
Diffusion	Movement of particles due to random gas molecular collisions and microscopic eddies in air	Particle size Temperature
Settling	Effect of gravity upon airborne particles	Particle size Turbulence Induced gas flow
Coagulation	The adherence of a particle to another upon collision to produce a particle of larger size and, for solids, less dense	Number of particles Eddy velocity Particle size
Condensation	Particle Generation (condensation of vapors upon condensate nuclei), or particle growth (condensation of vapors on existing particles)	Type of vapor Local temperature Particle size
Agglomeration	Same as coagulation (for colloids) and coalescence (for liquids)	Number of particles Eddy velocity Particle size
Scavenging	The removal of airborne particles by materials falling through a fluid volume	Particle size
Diffusiophoresis	Movement of particles caused by concentration gradients in the gas phase	Vapor condensation rate
Thermophoresis	Movement of particles down a temperature gradient	Temperature gradient

Table C.109 Unscaled and scaled total accident risks to the public for non-reactor fuel cycle facilities

Fuel Cycle Element	Total Accident Risk (person-rem/yr) Unscaled	Scaled (1/GWe)[a]	Table
Uranium Milling	--	2.7E-4	C.70
UF_6 Conversion	--	0.012	C.70
Enrichment	--	0.012	C.70
Fuel Fabrication	--	0.0050	C.70
MOX Fuel Refabrication	--	0.12	C.70
		0.036	C.70
		3.3E-5	C.70
Fuel Reprocessing	--	0.031	C.70
		0.0032	C.70
		5.6E-4	C.70
		2.2E-4	C.70
		1.5E-4	C.70
		5.4E-5	C.70
Spent Fuel Storage	--	0.18	C.70
		0.031	C.70
		3.7E-5	C.70
		2.7E-5	C.70
		5.7E-6	C.70
Cask Storage	1.2[b]	--	C.32
Drywell Storage	8.5[b]	--	C.31
	0.7[b]	--	C.32
Operations Phase	0.004[b]	--	C.44
HLW Storage	--	2.3E-4	C.70
Geologic Waste Disposal			
Total Preclosure	--	4.0E-5	C.70
Operations Phase	0.010	--	C.14
	1.5E-5	--	C.19
Without MRS	3E-5[b]	--	C.42
With MRS	3E-5[b]	--	C.44
Total Postclosure	--	5.0E-11[c]	--
Transportation			
Without MRS	5[b]	--	C.42
With MRS	10[b]	--	C.44

Table C.109 (Continued)

Fuel Cycle Element	Total Accident Risk (person-rem/yr) Unscaled	Scaled (1/GWe)[a]	Table
Plutonium Oxide			
Truck	--	0.0013	C.70
Rail	--	0.066	C.70
Spent Fuel			
Truck			
in 1975	240[b]	--	C.35
in 1985	1500[b]	--	C.35
Rail	--	0.16	C.70
	--	0.16	C.70
	--	0.078	C.70
	--	0.013	C.70
	--	0.0093	C.70
	--	7.1E-4	C.70
	--	5.6E-5	C.70
	--	8.4E-6	C.70
	--	8.4E-6	C.70
	--	2.6E-6	C.70
in 1975	110[b]	--	C.35
in 1985	4000[b]	--	C.35
HLW			
Rail	--	0.042	C.70
	--	0.0023	C.70
	--	8.4E-4	C.70
	--	2.8E-6	C.70
	--	2.4E-6	C.70

(a) Measured in terms of the annual requirements of a 1,000-MWe (1-GWe) LWR

(b) Converted to person-rem/yr using 5,000 person-rem/health effect

(c) From Erdmann et al. (1979), see Section C.6.

Table C.110 Preliminary occupational risk estimates for postulated accidents at a repository in tuff for preclosure operations phase of geologic waste disposal (see Tables C.18 and C.19) (Daling et al. 1990)

Accident Scenario	Frequency (1/yr)	Worker Dose (person-rem)	Worker Risk (person-rem/yr)
Natural Phenomena			
Flood	0.010	5.0E-10	5.0E-12
Earthquake	< 0.0013	0.37	< 4.8E-4
Tornado	< 9.1E-11	0.37	< 3.4E-11
Man-made Events			
Aircraft Impact	< 2.0E-10	5.5	< 1.1E-9
Nuclear Test	< 0.0010	0.37	< 3.7E-4
Operational Accidents			
Fuel Assembly Drop	0.10	0.0081	8.1E-4
Loading Dock Fire			
Spent Fuel	< 1.0E-7	3.5	< 3.5E-7
HLW	< 1.0E-7	0.6	< 6.0E-8
Waste Handling Ramp Fire	< 1.0E-7	64	< 6.4E-6
Emplacement Drift Fire	< 1.0E-7	180	< 1.8E-5
Total			.0017

Appendix D

Safety Goal Policy Statement and Backfit Rule

Appendix D

Safety Goal Policy Statement and Backfit Rule

D.1 Safety Goals for the Operations of Nuclear Power Plants (51 FR 30028; August 21, 1986)

SUMMARY: This policy statement focuses on the risks to the public from nuclear power plant operation. Its objective is to establish goals that broadly define an acceptable level of radiological risk. In developing the policy statement, the NRC sponsored two public workshops during 1981, obtained public comments and held four public meetings during 1982, conducted a 2-year evaluation during 1983 to 1985, and received the views of its Advisory Committee on Reactor Safeguards.

The Commission has established two qualitative safety goals which are supported by two quantitative objectives. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks. The Committee wants to make clear that no death attributable to nuclear power plant operation will ever be "acceptable" in the sense that the Committee would regard it as a routine or permissible event. The Committee is discussing acceptable risks, not acceptable deaths.

- The *qualitative safety* goals are as follows:

  - Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
  - Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

- The following *quantitative objectives* are to be used in determining achievement of the above safety goals:

  - The risk to an average individual in the vicinity of a nuclear power plant of prompt facilities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
  - The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

EFFECTIVE DATE: August 4, 1986.

SUPPLEMENTARY INFORMATION: The following presents the Commission's Final Policy Statement on Safety Goals for the Operation of Nuclear Power Plants:

I. Introduction

A. Purpose and Scope

In its response to the recommendations of the President's Commission on the Accident at three Mile Island, the Nuclear Regulatory Commission (NRC) stated that it was "prepared to move forward with an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs in the NRC safety decisions." This policy statement is the result.

Current regulatory practices are believed to ensure that the basic statutory requirement, adequate protection of the public, is met. Nevertheless, current practices could be improved to provide a better means for testing the adequacy of and need for current and proposed regulatory requirements. The Commission believes that such improvement could lead to a more coherent and consistent regulation of nuclear power plants, a more predictable regulatory process, a public understanding of the regulatory criteria that the NRC applies, and public confidence in the safety of operating plants. This statement of NRC safety policy expresses the Commission's views on the level of risks to public health and safety that the industry should strive for in its nuclear power plant.

This policy statement focuses on the risks to the public from nuclear power plant operation. These are the risks from release of radioactive materials from the reactor to the environment from normal operations as well as from accidents. The Commission will refer to these risks as the risks of nuclear power plant operation. The risks from the nuclear fuel cycle are not included in the safety goals.

These fuel cycle risks have been considered in their own right and determined to be quite small. They will continue to receive careful consideration. The possible effects of sabotage or diversion of nuclear material are also not presently included in the safety goals. At present there is no basis on which to provide a measure of risk on these matters. It is the Commission's intention that everything that is needed will be done to keep these types of risks at their present very low level; and it is the Commission's expectation that efforts on this point will continue to be successful. With these exceptions, it is the Commission's intent that the risks from all the various initiating mechanisms be taken into account to the best of the capability of current evaluation techniques.

In the evaluation of nuclear power plant operation, the staff considers several types of releases. Current NRC practice addresses the risks to the public resulting from operating nuclear power plants. Before a nuclear power plant is licensed to operate, NRC prepares an environmental impact assessment which includes an evaluation of the radiological impacts of routine operation of the plant and accidents on the population in the region around the plant site. The assessment undergoes public comment and may be extensively probed in adjudicatory hearings. For all plants licensed to operate, NRC has found that there will be no measurable radiological impact on any member of the public from routine operation of the plant. (Reference: NRC staff calculation of radiological impact on humans contained in Final Environmental Statements for specific nuclear power plants: e.g., NUREG-0779, NUREG-0812, and NUREG-0854.)

The objective of the Commission's policy statement is to establish goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation. While this policy statement includes the risks of normal operation, as well as accidents, the Commission believes that because of compliance with Federal Radiation Council (FRC) guidance, (40 CFR Part 190), and NRC's regulations (10 CFR Part 20 and Appendix I to Part 50), the risks from routine emissions are small compared to the safety goals. Therefore, the Commission believes that these risks need not be routinely analyzed on a case-by-case basis in order to demonstrate conformance with the safety goals.

B. Development of this Statement of Safety Policy

In developing the policy statement, the Commission solicited and benefited from the information and suggestions provided by workshop discussions. NRC-sponsored workshops were held in Palo Alto, California, on April 1-3, 1981 and in Harpers Ferry, West Virginia, on July 23-24, 1981. The first workshop addressed general issues involved in developing safety goals. The second workshop focused on a discussion paper which presented proposed safety goals. Both workshops featured discussions among knowledgeable persons drawn from industry, public interest groups, universities, and elsewhere, who represented a broad range of perspectives and disciplines.

The NRC Office of Policy Evaluation submitted to the Commission for its consideration a Discussion Paper on Safety Goals for Nuclear Power Plants in November 1981 and a revised safety goal report in July 1982.

The Commission also took into consideration the comments and suggestions received from the public in response to the proposed Policy Statement on "Safety Goals for Nuclear Power Plants," published on February 17, 1982 (47 FR 7023). Following public comment, a revised Policy Statement was issued on march 14, 1983 (48 FR 10772) and a 2-year evaluation period began.

The Commission used the staff report and its recommendations that resulted from the 2-year evaluation of safety goals in developing this final Policy Statement. Additionally, the Commission had benefit of further comments from its Advisory Committee on Reactor Safeguards (ACRS) and by senior NRC management.

Based on the results of this information, the Commission has determined that the qualitative safety goals will remain unchanged from its March 1983 revised policy statement and the Commission adopts these as its safety goals for the operation of nuclear power plants.

II. Qualitative Safety Goals

The Commission has decided to adopt qualitative safety goals that are supported by quantitative health effects objectives for use in the regulatory decisionmaking process. The Commission's first quantitative safety goal is that risk from nuclear power plant operation should not be a significant contributor to a person's risk to accidental death or injury. The intent is to require such a level of safety that individuals living or working near nuclear power plants should be able to go about their daily lives without special concern by virtue of their proximity to these plants. Thus, the Commission's first safety goal is -

Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

Even though protection of individual members of the public inherently provides substantial societal protection, the Commission also decided that a limit should be placed on the societal risks posed by nuclear power plant operation. The Commission also believes that the risks of nuclear power plant operation should be comparable to or less than the risks from other viable means of generating the same quantity of electrical energy. Thus, the Commission's second safety goal is -

Societal risk to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The broad spectrum of expert opinion on the risks posed by electrical generation by coal and the absence of authoritative data make it impractical to calibrate nuclear safety goals by comparing them with coal risks based on what we know today. However, the Commission has established the quantitative health effects objectives in such a way that nuclear risks are not a significant addition to other societal risks.

Severe core damage accidents can lead to more serious accidents with the potential for life-threatening offsite release of radiation, for evacuation of members of the public, and for contamination of public property. Apart from their health and safety consequences, severe core damage accidents can erode public confidence in the safety of nuclear power and can lead to further instability and unpredictability for the industry. In order to avoid these adverse consequences, the Commission intends to continue to pursue a regulatory program that has as its objective providing reasonable assurance, while giving appropriate consideration to the uncertainties involved, that a severe core damage accident will not occur at a U.S. nuclear power plant.

III. Quantitative Objectives Used to Gauge Achievement of The Safety Goals

A. General Considerations

The quantitative health effects objectives establish NRC guidance for public protection which nuclear plant designers and operators should strive to achieve. A key element in formulating a qualitative safety goal whose achievement is measured by quantitative health effects objectives is to understand both the strengths and limitations of the techniques by which one judges whether the qualitative safety goal has been met.

A major step forward in the development and refinement of accident risk quantification was taken in the Reactor Safety Study (WASH-1400) completed in 1975. The objective of the Study was "to try to reach some meaningful conclusions about the risk of nuclear accidents." The Study did not directly address the question of what level of risk from nuclear accidents was acceptable.

Since the completion of the Reactor Safety Study, further progress in developing probabilistic risk assessment and in accumulating relevant data has led to a recognition that it is feasible to begin to use quantitative safety objectives for limited purposes. However, because of the sizable uncertainties still present in the methods and the gaps in the data base--essential elements needed to gauge whether the objectives have been achieved--the quantitative objectives should be viewed as aiming points or numerical benchmarks of performance. In particular, because of the present limitations in the state of the art of quantitatively estimating risks, the quantitative health effects objectives are not a substitute for existing regulations.

The Commission recognizes the importance of mitigating the consequences of a core-melt accident and continues to emphasize features such as containment, siting in less populated areas, and emergency planning as integral parts of the defense-in-depth concept associated with its accident prevention and mitigation philosophy.

B. Quantitative Risk Objectives

The Commission wants to make clear at the beginning of this section that no death attributable to nuclear power plant operation will ever be "acceptable" in the sense that the Commission would regard it as a routine or permissible event. We are discussing acceptable risks, not acceptable deaths. In any fatal accident, a course of conduct posing an acceptable risk at one moment results in an unacceptable death moments later. This is true whether one speaks of driving, swimming, flying, or generating electricity from coal. Each of these activities poses a calculable risk to society and to individuals. Some of those who accept the risk (or are part of a society that accepts risk) do not survive it. We intend that no such accidents will occur, but the possibility cannot be entirely eliminated. Furthermore, individual and societal risks from nuclear power plants are generally estimated to be considerably less than the risk that society is now exposed to from each of the other activities mentioned above.

C. Health Effects--Prompt and Latent Cancer Mortality Risks

The Commission has decided to adopt the following two health effects as the quantitative objectives concerning mortality risks to be used in determining achievement of the qualitative safety goals -

The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

The risk to the population the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The Commission believes that this ratio of 0.1 percent appropriately reflects both of the qualitative goals--to provide that individuals and society bear no significant additional risk. However, this does not necessarily mean that an additional risk that exceeds 0.1 percent would by itself constitute a significant additional risk. The 0.1 percent ratio to other risks is low enough to support an expectation that people living or working near nuclear power plants would have no special concern due to the plant's proximity.

The average individual in the vicinity of the plant is defined as the average individual biologically (in terms of age and other risk factors) and locationally who resides within a mile from the plant site boundary. This means that the average individual is found by accumulating the estimated individual risks and dividing by the number of individuals residing in the vicinity of the plant.

In applying the objective for individual risk of prompt fatality, the Commission has defined the vicinity as the area within one (1) mile of the nuclear power plant site boundary, since calculations of the consequences of major reactor accidents suggest that individuals within a mile of the plant site boundary would generally be subject to the greatest risk of prompt death attributable to radiological causes. If there are no individuals residing within a mile of the plant boundary, an individual should, for evaluation purposes, be assumed to reside one (1) mile from the site boundary.

In applying the objective for cancer fatalities as a population guideline for individuals in the area near the plant, the Commission has defined the population generally considered subject to significant risk as the population within ten (10) miles of the plant site. The bulk of significant exposures of the population to radiation would be concentrated within this distance, and thus this is the appropriate population for comparison with cancer fatality risks from all other causes. This objective would ensure that the estimated increase in the risk of delayed cancer fatalities from all potential radiation releases at a typical plant would be no more than a small fraction of the year-to-year normal variation in the expected cancer deaths from nonnuclear causes. Moreover, the prompt fatality objective for protecting individuals generally provides even greater protection to the population as a whole. That is, if the quantitative objective for prompt fatality is met for individuals in the immediate vicinity of the plant, the estimated risk of delayed cancer fatality to persons within ten (10) miles of the plant and beyond would generally be much lower than the quantitative objective for cancer fatality. Thus, compliance with the prompt fatality objective applied to individuals close to the plant would generally mean that the aggregate estimated societal risk would be a number of times lower than it would be if compliance with just the objective applied to the population as a whole were involved. The distance foe averaging the cancer fatality risk was taken as 50 miles in the 1983 policy statement. The change to ten (10) miles could be viewed to provide additional protection to individuals in the vicinity of the plant, although analyses indicate that this objective for cancer fatality will not be the controlling one. It also provides more representative societal protection, since the risk to the people beyond ten (10) miles will be less than the risk to the people within ten (10) miles.

IV. Treatment of Uncertainties

The Commission is aware that uncertainties are not caused by use of quantitative methodology in decisionmaking but are merely highlighted through use of the quantification process. Confidence in the use of probabilistic and risk assessment techniques has steadily improved since the time these were used in the Reactor Safety Study. In fact, through use of quantitative techniques, important uncertainties have been and continue to be brought into better focus and may even be reduced compared to those that would remain with sole reliance on deterministic decisionmaking. To the extent practicable, the Commission intends to ensure that the quantitative techniques used for regulatory decisionmaking take into account the potential uncertainties that exist so that an estimate can be made on the confidence level to be ascribed to the quantitative results.

The Commission has adopted the use of mean estimates for purposes of implementing the quantitative objectives of this safety goal policy (i.e., the mortality risk objectives). Use of the mean estimates comports with the customary practices for cost-benefit analyses and it is the correct usage for purposes of the mortality risk comparisons. Use of mean estimated does not however resolve the need to quantify (to the extent reasonable) and understand those important uncertainties involved in the reactor accident risk predictions. A number of uncertainties (e.g., thermal-hydraulic assumptions and the phenomenology of core-melt progression, fission product release and transport, and containment loads and performance) arise because of a direct lack of severe accident experience or knowledge of accident phenomenology along with data related to probability distributions.

In such a situation, it is necessary that proper attention be given not only to the range of uncertainty surrounding probabilistic estimates, but also to the phenomenology that most influences the uncertainties. For this reason, sensitivity studies should be performed to determine those uncertainties most important to the probabilistic estimate. The results of sensitivity of studies should be displayed showing, for example, the range of variation together with the underlying science or engineering assumptions that dominate this variation. Depending on the decision needs, the probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgements can be made by the decisionmaker about the degree of confidence to be given to these estimates and assumptions. This is a key part of the process of determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety.

V. Guidelines for Regulatory Implementation

The Commission approves use of the qualitative safety goals, including use of the quantitative health effects objectives in the regulatory decisionmaking process. The Commission recognizes that the safety goal can provide a useful tool by which the adequacy of regulations or regulatory decisions regarding changes to the regulations can be judged. Likewise, the safety goals could be of benefit in the much more difficult task of assessing whether existing plants, designed, constructed and operated to comply with past and current regulations, conform adequately with the intent of the safety goal policy.

However, in order to do this, the staff will require specific guidelines to use as a basis for determining whether a level of safety ascribed to a plant is consistent with the safety goal policy. As a separate matter, the Commission intends to review and approve guidance to the staff regarding such determinations. It is currently envisioned that this guidance would address matters such as plant performance guidelines, indicators for operational performance, and guidelines for conduct of cost-benefit analyses. This guidance would be derived from additional studies conducted by the staff and resulting in recommendations to the Commission. The guidance would be based on the following general performance guideline which is proposed by the commission for further staff examination -

Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and to mitigate their consequences. Siting in less populated areas is emphasized. Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population.

These safety goals and these implementation guidelines are not meant as a substitute for NRC's regulations and do not relieve nuclear power plant permittees and licensees from complying with regulations. Nor are the safety goals and these implementation guidelines in and of themselves meant to serve as a sole basis for licensing decisions. However, if pursuant to these guidelines, information is developed that is applicable to a particular licensing decision, it may be considered as one factor in the licensing decision.

The additional views of Commissioner Asselstine and the separate views of Commissioner Bernthal are attached.

Dated at Washington, D.C., this 30th day of July 1986.

For the Nuclear Regulatory Commission. **Lando W. Zech, Jr.**, *Chairman*.

Additional Views by Commissioner Asselstine on the Safety Goals Policy Statement

The commercial nuclear power industry started rather slowly and cautiously in the early 1960's. By the late 1960's and early 1970's, the growth of the industry reached a feverish pace. New orders were coming in for regulatory review on almost a weekly basis. The result was the designs of the plants outpaced operational experience and the development of safety standards. As experience was gained in operational characteristics and in safety reviews, safety standards were developed or modified with a general trend toward stricter requirements. Thus, in the early 1970's, the industry demanded to know "how safe is safe enough." In this Safety Goal Policy Statement, the Commission is reaching a first attempt at answering the question. Much credit should go to Chairman Palladino's efforts over the past five (5) years to develop this policy statement. I approve this policy statement but believe it needs to go further. There are four additional aspects which should have been addressed by the policy statement.

Containment Performance

First, I believe the Commission should have developed a policy on the relative emphasis to be given to accident prevention and accident mitigation. Such guidance is necessary to ensure that the principle of defense-in-depth is maintained. The Commission's Advisory Committee on Reactor Safeguards has repeatedly urged the Commission to do so. As a step in that direction, I offered for Commission consideration the following containment performance criterion:

In order to assure a proper balance between accident prevention and accident mitigation, the mean frequency of containment failure in the event of a severe core damage accident should be less than 1 in 100 severe core damage accidents.

Since the Chernobyl accident, the nuclear industry has been trying to distance itself from the Chernobyl accident on the basis of the expected performance of the containments around the U.S. power reactors. Unfortunately, the industry and the Commission are unwilling to commit to a level of performance for the containments.

The argument has been made that we do not know how to develop containment performance criteria (accident mitigation) because core meltdown phenomena and containment response thereto are very complex and involve substantial uncertainties. On the other hand, to measure how close a plant comes to the quantitative guidelines contained in this policy statement and to perform analyses required by the Commission's backfit rule, one must perform just those kinds of analyses. I find these positions inconsistent.

The other argument against a containment performance criterion is that such a standard would overspecify the safety goal. However, a containment performance objective is an element of ensuring that the principle of defense-in-depth is maintained. Since we cannot rule out core meltdown accidents in the foreseeable future, given the current level of safety, I believe it unwise not to establish an expectation on the performance of the final barrier to a substantial release of radioactive materials to the environment, given a core meltdown.

General Performance Guideline

While I have previously supported an objective of reducing the risks to an as low as reasonably achievable level, the general performance guideline articulated in this policy (i.e., "...the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation") is a suitable compromise. I believe it is an objective that is consistent with the recommendations of the Commission's chief safety officer and our Director of Research, and past urgings of the Advisory Committee on Reactor Safeguards. Unfortunately, the Commission stopped short of adopting this guideline as a performance objective in the policy statement, but I am encouraged that the Commission is willing at least to examine the possibility of adopting it. Achieving such a standard coupled with the containment performance objective given above would go a long way toward ensuring that the operating reactors successfully complete their useful lives and that the nuclear option remains a viable component of the nation's energy mix.

In addition to preferring adoption of this standard now, I also believe the Commission needs to define a "large release" of radioactive materials. I would have defined it as "a release that would result in a whole body dose of 5 rem to an individual located at the site boundary." This would be consistent with the EPA's emergency planning Protective Action Guidelines and with the level proposed by the NRC staff for defining an Extraordinary Nuclear Occurrence under the Price-Anderson Act. In adopting such a definition, the Commission would be saying that its objective is to ensure that there is no more than a 1 in 1,000,000 chance per year that the public would have been to be evacuated from the vicinity of a nuclear reactor and that the waiver of defenses provisions of the Price-Anderson Act would be invoked. I believe this to be an appropriate objective in ensuring that there is no undue risk to the public health and safety associated with nuclear power.

Cost-Benefit Analyses

I believe it is long overdue for the Commission to decide the appropriate way to conduct cost-benefit analyses. The Commission's own regulations require these analyses, which play a substantial role in the decisionmaking on whether to improve safety. Yet, the commission continues to postpone addressing this fundamental issue.

Future Reactors

In my view, this safety goal policy statement has been developed with a steady eye on the apparent level of safety already achieved by most of operating reactors. That level has been arrived at by a piecemeal approach to designing, constructing and upgrading of the plants over the years as experience was gained with the plants and as the results of required research became available. Given the performance of the current generation of plants. I believe a safety goal for these plants is not good enough for the future. This policy statement should have had a separate goal that would require substantially better plants for the next generation. To argue that the level of safety achieved by plant designs that are over 10 years old is good enough for the next generation is to have little faith in the ingenuity of engineers and in the potential for nuclear technology. I would have required the next generation of plants to be substantially safer than the currently operating plants.

Separate Views of Commissioner Bernthal on Safety Goals Policy

I do not disapprove of what has been said in this policy statement, but too much remains unsaid. The public is understandably desirous of reassurance since Chernobyl: the NRC staff needs clear guidance to carry out its responsibilities to assure public health and safety; the nuclear industry needs to plan for the future. All want and deserve to see clear, unambiguous, practical safety objectives that provide the Commission's answer to the question, "How safe is safe enough?" at U.S. nuclear power plants. The question remains unanswered.

It is unrealistic for the Commission to expect that society, for the foreseeable future, will judge nuclear power by the same standard as it does all other risks. The issue today is not so much calculated risk; the issue is public acceptance and, consistent with the intent of Congress, preservation of the nuclear option.

In these early decades of nuclear power, TMI-style incidents must be rendered so rare that we would expect to recount such an event only to our grandchildren. For today's population of reactors, that implies a probability for severe core damage of 10^{-4} per reactor year; for the longer term, it implies something better. I see this as a straightforward policy conclusion that every newspaper editor in the country understands only too well. If the Commission fails to set (and realize) this objective, then the nuclear option will cease to credible before the end of the century. In other words, if TMI-style events were to occur with 10-15 year regularity, public acceptance of nuclear power would almost certainly fail.

And while the Commission's primary charge is to protect public health and safety, it is also the clear intent of Congress that the Commission, if possible, regulate in a way that preserves rather than jeopardizes the nuclear option. So, for example, if the Commission were to find 100 percent confidence in some impervious containment design, but ignored what was inside the containment, the primary mandate would be satisfied, but in all likelihood, the second would not. Consistent with the Commission's long-standing defense-in-depth philosophy, both core-melt and containment performance criteria should therefore be clearly stated parts of the Commission's safety goals.

In short, this pudding lacks a theme. Meaningful assurance to the public; substantive guidance to the NRC staff; the regulatory path to the future for the industry--all these should be provided by plainly stating that, consistent with the Commission's "defense-in-depth" philosophy:

(1) Severe core-damage accidents should not be expected, on average, to occur in the U.S. more than once in 100 years:

(2) Containment performance at nuclear power plants should be such that severe accidents with substantial offsite damages are not expected, on average, to occur in the U.S. more than one in 1,000 years:

(3) The goal for offsite consequences should be expected to be met after conservative considerations of the uncertainties associated with the estimated frequency of severe core-damage and the estimated mitigation thereof by containment.[a]

The term "substantial offsite damages" would correspond to the Commission's legal definition of "extraordinary nuclear occurrence." "Conservative consideration of associated uncertainties" should offer at least 90 percent confidence (typical good engineering judgment, I would hope) that the offsite release goal is met.

The broad core-melt and offsite-release goals should be met "for the average power plant"; i.e., for the aggregate of U.S. power plants. The decision to fix or not to fix a specific plant would then depend on achieving "the goal for offsite consequences." As a practical matter, this offsite societal risk objective would (and should) be significantly dependent on site-specific population density.

The absence of such explicit population density considerations in the Commission's 0.1 percent goals for offsite consequences deserves careful thought. Is it reasonable that Zion and Palo Verde, for example, be assigned the same theoretical "standard person" risk, even though they pose considerably different risks for the U.S. population as a whole? As they stand, these 0.1 percent goals do not explicitly include population density considerations; a power plant could be located in Central Park and still meet the Commission's quantitative offsite release standard.

I believe the Commission's standards should preserve the important principle that the site-specific population density be quantitatively considered in formulating the Commission's societal risk objective; e.g., by requiring that for the *entire* U.S. population, the risk of fatal injury as a consequence of the U.S. nuclear power plant operations should not exceed some appropriate specified fraction of the sum of the expected risk of fatality form all other hazards to which members of the U.S. population are generally exposed.

I am further concerned by the arbitrary nature of the 0.1 percent incremental "societal" health risk standard adopted by the Commission, a concept grounded in a purely subjective assessment of what the public might accept. The Commission should seriously consider a more rational standard, tied statistically to the average variations in natural exposure to radiation from all other sources.

Finally, as noted in its introductory comments, the Commission long ago committed to "move forward with an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs in NRC safety decisions." While this policy statement may not be very "explicit", as discussed above, it contains nothing at all on the subject of "'safety-cost' tradeoffs in NRC safety decisions." For example, *is* $1,000 per person-rem an appropriate cost-benefit standard for NRC regulatory action? While I have long argued that such fundamental decisions are more rightly the responsibility of Congress, the NRC staff continues to use its ad-hoc judgment in lieu of either the Commission or the Congress speaking to the issue.

In summary, while the Commission has produced a document which is not in conflict with my broad philosophy in such matters, I doubt that the public expected a philosophical dissertation, however erudite. It is a tribute to Chairman Palladino's efforts that the Commission has come this far. But the task remains unfinished.

(a) Interestingly enough the Commission has adopted proposed goals similar to the above core-melt and containment performance objectives-without clearly saying so. Taken together, the Commission's: (1) 0.1 percent offsite prompt fatality goals: (2) proposed 10^{-4} per-reactor-year "large offsite release" criterion: (3) commitment "to provide reasonable assurance...that a severe core-damage accident will not occur at a U.S. nuclear power plant" though they may be ill-defined, can be read to be more stringent than the plainly stated criteria suggested above.

D.2 Backfit Rule (10 CFR 50.109)

(a)(1) Backfitting is defined as the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position after:

(i) The date of issuance of the construction permit for the facility for facilities having construction permits issued after October 21, 1985; or

(ii) Six months before the date of docketing of the operating license application for the facility for facilities having construction permits issued before October 21, 1985; or

(iii) The date of issuance of the operating license for the facility for facilities having operating license; or

(iv) The date of issuance of the design approval under appendix M, N, or O of part 52.

(2) Except as provided in paragraph (a)(4) of this section, the Commission shall require a systematic and documented analysis pursuant to paragraph (c) of this section for backfits which it seeks to impose.

(3) Except as provided in paragraph (a)(4) of this section, the Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (c) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defence and security to be derived from the backfit and that the direct and indirect costs if implementation for that facility are justified in view of this increased protection.

(4) The provisions of paragraphs (a)(2) and (a)(3) of this section are inapplicable and, therefore, backfit analysis is not required and the standards in paragraph (a)(3) of this section do not apply where the Commission or staff, as appropriate, finds and declares, with appropriated documented evaluation for its finding, either:

(i) That a modification is necessary to bring a facility into compliance with license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or

(ii) That regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or

(iii) That the regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate.

(5) The Commission shall always require the backfitting of a facility if it determines that such regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety or the common defense and security.

(6) The document evaluation required by paragraph (a)(4) of this section shall include a statement of the objectives of and reasons for the modification and the basis for invoking the exception. If immediately effective regulatory action is required, then the documented evaluation may follow rather than precede the regulatory action.

(7) If there are two or more ways to achieve compliance with a license or the rules or orders of the Commission, or with written licensee commitments, or there are two or more ways to reach a level of protection which is adequate, then ordinarily the applicant or licensee is free to choose the way which best suits its purposes. However, should it be necessary or appropriate for the Commission to prescribe a specific way to comply with its requirements or to achieve adequate protection, then cost may be a factor in selecting the way, provided that the objective of compliance or adequate protection is met.

(b) Paragraph (a)(3) of the section shall not apply to backfits imposed prior to October 21, 1985.

(c) In reaching the determination required by paragraph (a)(3) of this section, the Commission will consider how the backfit should be scheduling light of other ongoing regulatory activities at the facility and, in addition, will consider information available concerning any of the following factors as may be appropriate and any other information relevant and material to proposed backfit:

(1) Statement of the specific objectives that the proposed backfit is designed to achieve;

(2) General description of the activity that would be required by the licensee or applicant in order to complete the backfit;

(3) Potential change in the risk to the public from accidental off-site release of radioactive material;

(4) Potential impact on radiological exposure of facility employees;

(5) Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay;

(6) The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements;

(7) The estimated resource burden on the NRC associated with the proposed backfit and the availability of such resources;

(8) The potential impact or differences in facility type, design or age on the relevancy and practicality of the proposed backfit;

(9) Whether the proposed backfit is interim or final and, if interim, the justification for imposing the proposed backfit on an interim basis.

(d) No licensing action will be withheld during the pendency of backfit analyses required by the commissions rules.

(e) The Executive Director for Operations shall be responsible for implementation of this section, and all analyses required by this section shall be approved by the Executive Director for Operations or his designee.

[54 FR 20610, June 6, 1988, as amended 54 FR 15398, April 18, 1989]

Appendix E

Index

Appendix E

Index

A

B

C

D

E

G

H

I

L

M

N

O

P

R

S

T

V