Human errors were divided into the following overlapping categories for the sensitivity analysis:

- *Timing* - when the human error occurs relative to the accident initiating event or transient
- *Accident Initiator* - which accident initiating event is related to the human error
- *System* - the system in which the human error occurs
- *Personnel* - which individuals are responsible for the human error
- *Omission/Commission* - whether the human error is one where a needed action is not performed (omission) or one where an improper action is performed (commission)
- *Event Type* - relating the human error to the category assigned in the Oconee risk assessment (EPRI and Duke Power Co. 1984)
- *Location* - where the personnel most responsible for the human error are located
- *Activity* - which type of nuclear power plant activity relates to the human error
- *Dependence* - whether or not the human error results from another human error
- *NRC Program* - which NRC inspection area may detect the occurrence of the human error.

The sensitivity of the three risk parameters mentioned above to changes in HEPs for these various categories are conveniently presented as figures in NUREG/CR-5319. All HEPs within each category were simultaneously varied relative to the base-case value from the Oconee risk assessment. In addition, the effect of simultaneous variation of all HEPs on the three risk parameters was evaluated. The results were compared with those from the first study.

Both these studies provide information which would be useful in human factors issues where categories of HEPs would be affected. For example, plant-wide improvements in maintenance procedures or more stringent testing of reactor operators would be expected to reduce all HEPs falling within the appropriate categories. These two studies provide relative values for the change in selected risk parameters for such simultaneous variation of HEPs. Most human factors issues appear to be of this "global" nature, hence the usefulness of the studies' results.

The NRC (NRC 1983b), with assistance from Pacific Northwest National Laboratory (PNNL) (Andrews et al. 1983), has been systematically prioritizing generic safety issues since 1982, many of which involve human factors for nuclear power plants. Simple methods were initially established to handle human factors issues which fell into the "concrete" and "abstract" categories discussed earlier in this appendix section. The earlier discussion summarizes the approach that was taken in the prioritization assessments. NUREG/CR-2800 and its supplements (Andrews et al. 1983) provide numerous examples of human factors issues analyzed using these simple methods. In 1985, Andrews et al. conducted a study (NUREG/CR-2800, Supplement 3) in which they 1) developed an alternative approach to prioritizing human factors issues and 2) prioritized the elements of the 1983 Human Factors Program Plan (HFPP) developed by the NRC.

The development of the alternative human factors methodology by Andrews et al. (1985) involved investigation of four attributes of human factors analyses: 1) the general guidelines used by the decision-making panel in the initial prioritizations, 2) the impact of using alternate representative plants, 3) human factors modeling related to maintenance and plant availability, and 4) human factors data bases. For the first attribute, decision-making basis was documented in terms of

plant-related guidelines, human error assumptions, independence of human factors issues, and cost guidance. For the second attribute, the differences in core-melt frequency resulting from reducing HEPs for three different representative plants, the Oconee and Calvert Cliffs PWRs and the Grand Gulf BWR, as modeled by their Reactor Safety Study Methodology Application Program (RSSMAP) studies (Kolb et al. 1981; Hatch et al. 1981, 1982) was quantified. For the third attribute, new maintenance and plant availability models were developed and tested. For the fourth attribute, available human factors data bases were examined and found to be only of limited use in prioritization analyses.

Andrews et al. (1985) also prioritized the following six elements of the 1983 HFPP: 1) staffing and qualifications, 2) training, 3) licensing examinations, 4) procedures, 5) man-machine interfaces, and 6) management and organization. Eighteen generic safety issues were divided among the six elements. For each, expert opinion on the effects on HEPs and costs resulting from resolution was solicited through a structured series of questionnaires. The consensus changes in HEPs were transformed into public risk changes via the Oconee and Grand Gulf RSSMAP models. Public risk, industry, and NRC cost estimates for implementing the HFPP as a whole and for implementing each specific element were calculated and used to assign priorities to the six elements.

As in the studies by Samanta et al. (1981, 1989), this study by Andrews et al. (1985) provides information which would be useful to human factors issues where categories of HEPs would be affected. It provides relative values for the change in core-melt frequency and public risk for simultaneous variation of HEPs. In addition, since a comprehensive program for human factors improvements has been examined, estimates of maximum possible reductions in public risk and increases in industry and NRC costs attainable by implementing such a program are available. Individual issues within each element of the HFPP were also examined, with their public risk reductions and industry and NRC cost increases evaluated. Therefore, this information is available for several types of human factors issues.

A.1.2 Methods Documents

In NUREG/CR-2255, Stillwell et al. (1982) reviewed probability assessment and psychological scaling techniques that could be used to estimate human error probabilities in nuclear power plant operations. The techniques rely on expert opinion and can be used where data do not exist or are inadequate. An extensive literature search was performed, and the results are discussed under two categories: 1) subjective probability assessment, and 2) psychological scaling. While this report is primarily a qualitative overview of the various techniques, it provides useful background as to which ones would be appropriate and when, as well as serving as a reference document for additional information.

The first category examined by Stillwell et al. considered seven aspects of subjective probability assessment: 1) use of expert judgment for assessing probabilities, 2) probabilistic assessment techniques, 3) use of multiple experts in assessing probabilities, 4) problems and biases in the assessment of subjective probability, 5) training probability assessors, 6) new methods for resolving inconsistent judgments, and 7) defining and structuring judgments. The second category compared the following five techniques of psychological scaling, with emphasis on their validity and reliability: 1) paired comparisons, 2) ranking, 3) sorting, 4) rating, and 5) fractionation.

In a follow-on report (NUREG/CR-2743), Seaver and Stillwell (1983) described and evaluated the following five procedures for employing expert opinion to estimate HEPs for nuclear power plant operations: 1) paired comparisons, 2) ranking and rating, 3) direct numerical estimation, 4) indirect numerical estimation, and 5) multiattribute utility measurement. The following criteria were used to evaluate these techniques: quality of judgments, difficulty of data collection, empirical support, acceptability, theoretical justification, and data processing. Quantitative guidance on the implementation of these procedures is provided, along with situational constraints (e.g., the number of HEPs to be estimated) which impact the choice of a procedure.

Third in this series of studies was NUREG/CR-3688, in which Comer et al. (1984) examined selected techniques for psychological scaling, first introduced by Stillwell et al. (1982) in NUREG/CR-2255. Two techniques—direct numerical estimation and paired comparison scaling—were evaluated in detail. Comer et al. answered the following 11 questions as a result of their study:

1. Do psychological scaling techniques produce consistent judgments from which to estimate HEPs?

2. Do psychological scaling techniques produce valid HEP estimates?

3. Can the data collected using psychological scaling techniques be generalized?

4. Are the HEP estimates that are generated from psychological scaling techniques suitable for use in probabilistic risk assessments and the human reliability data bank?

5. Can psychological scaling procedures be used by persons who are not experts to generate HEP estimates?

6. Do the experts used in the psychological scaling process have confidence in their ability to make judgments?

7. Is there any difference in the quality of estimates obtained from the two scaling techniques?

8. Is there any difference in the results based on the type of task that is being judged?

9. Do education and experience have any effect of the experts' judgments?

10. How should the paired comparison scale be calibrated into a probability scale?

11. Can reasonable uncertainty bounds be estimated judgmentally?

The HEPs for 35 BWR tasks that were estimated as part of the study are also presented.

These three studies provide guidance on the estimation of HEPs by expert judgment. Although intended for estimating HEPs directly, the techniques presented in these three studies are readily adapted to estimating changes in HEPs by expert judgment, typically what is needed to quantify the value-impact of a human factors issue. Techniques such as these can be used to estimate the changes in individual or families of HEPs. Subsequently, they can be combined with knowledge on the overall effect of more global changes in HEPs on core-melt frequency and public risk as provided by studies such as those of Samanta et al. (1981, 1989) and Andrews et al. (1985).

A.2 Cumulative Accounting of Past and Ongoing Safety Improvements

When performing a regulatory analysis, an analyst should be aware of previous or ongoing safety improvements which already have impacted or bear the potential to impact the status quo for the issue being addressed. Incorporation of such improvements could be accommodated if there existed a "master" risk assessment (or a few "masters") deemed representative of all facilities for which all previous safety improvements have been included and the baseline risk recalculated. Since this currently is not practical, the analyst must resort to a "best effort" approach in accounting for preexisting or concurrent impacts, consistent with NRC policy regarding the treatment of voluntary activities by affected licensees (see NRC Guidelines Section 4.3).

During Step 1 of the regulatory analysis (see Section 4.1), the analyst should make a thorough effort to identify any previous or ongoing safety improvements which may impact the issue under consideration. For example, an analyst addressing proposed improvements in diesel generator performance at power reactors should be aware of any diesel generator improvements already addressed in station blackout (SBO) considerations. To the extent possible, the analyst should modify the risk equation of the plant chosen as representative to reflect the upgraded status quo from these other safety improvements. The analyst can then proceed to assess the difference between this new status quo and the proposed improvements from the issue under consideration. The analyst should also seek out and use (when appropriate) the most recent risk assessments (including IPE and IPEEE reports) affecting the facilities impacted by the issues under consideration (see Table 5.2).

An attempt to accommodate "dependences" between issues was informally tried during the Prioritization of Safety Issues Program (Andrews et al. 1983). Issues of "high" rank were divided into "families" with similar issue resolutions (e.g., diesel generator reliability and SBO were assigned to an electrical family). The issues within each family were examined for all pairwise combinations where Issue A was implemented before Issue B and vice versa. Within these families, few dependent pairs were found and, for those found, the dependent effects were generally small (<10%). A similar approach could be taken, although the analyst may wish to consider greater than pairwise combinations if necessary.

A.3 Use of Industry Risk and Cost Estimates

As a general rule, analysts can use risk and cost data prepared by industry sources provided the analyst can independently attest to the reasonableness of the data.

Table 5.2 in Section 5.6.1 lists nuclear power plant risk/reliability studies (other than IPE and IPEEE reports) for use in regulatory analyses for power reactors. Several studies have been performed by the nuclear industry (i.e., the utilities themselves and/or their contractors). Theoretically, some bias may exist depending upon the source of the study (NRC contractor or industry). Some indication of such bias may be obtained by comparing studies performed for the same plant by different sources. However, one would have to take care not to attribute differences to bias if plant changes, more recent data, or different analytical methods are the reasons for differing results. The issue of bias may often be rendered useless to debate since the analyst may not have a wide choice of representative plants with existing risk/reliability studies. The analyst should always opt for the most representative plant, whether its risk/reliability study was performed by an NRC contractor or industry. The same considerations apply to regulatory analyses for non-reactor facilities, to the extent that representative risk/reliability studies are available (see Sections 5.6.1 and C.2.1.1).

Wider choice may be available to the analyst for cost estimates, and the analyst may be faced with different costs from equally valid sources. A sensitivity analysis may be best in which the analyst uses each set of costs for those attributes most strongly affected. However, should the analyst have reason to believe one set to be more representative than the other, the more representative set should be selected. The analyst may still use the other set in a sensitivity study should it be deemed appropriate.

Appendix B

Supplemental Information For Value-Impact Analyses

Appendix B

Supplemental Information For Value-Impact Analyses

This appendix presents data on the number of operating power reactors and their remaining lifetimes, methods of economic discounting and present value calculation, data on occupational exposure experience at nuclear power plants and some non-reactor facilities, additional cost information, and a description of the calculational method used to generate Table 5.3, "Expected Population Doses," for power reactor plant damage states. These can be used by the analyst to support his evaluation of attributes during the value-impact analysis portion of a regulatory analysis.

B.1 Numbers of Operating Power Reactors and Their Remaining Lifetimes

Table B.1 lists the numbers of operating power reactors and their remaining lifetimes relative to 1993. The lifetimes are based on the years in which the Operating Licenses currently expire, as reported in NUREG-1350, Vol.4 (NRC 1992). Table B.1 lists the plants by vendor and reactor type.

Table B.1 Numbers and lifetimes of operating nuclear power plants

Reactor Supplier	Type	Number of Operating Units (N)	Average Remaining Lifetime (T) (years)[a]
Westinghouse	PWR	52	25.4
General Electric	BWR	37	23.3
Combustion Engineering	PWR	15	23.7
Babcock and Wilcox	PWR	7	21.4
	N	T (years)	
All PWRs	74	24.7	
All BWRs	37	23.3	
All Plants	111	24.2	

(a) Relative to 1993.

B.2 Economic Discounting and Calculation of Present Value

To evaluate the economic consequences of proposed regulatory actions, the costs incurred or saved over a period of years must be summed.

This summation cannot be done directly because an amount of money available today has greater value than the same amount at a future date. There are several reasons for this difference in value:

- the present amount of money can be invested and the total amount increased through accumulated interest
- certain consumption today is superior to contingent consumption in the future
- the option of present or future consumption is superior to future consumption alone.

A method known as "discounting" is used to compare amounts of money expended at different times. The result of discounting is called the "present value," the amount of money that must be invested today to achieve a specified sum in the future. To perform the discounting procedure, the analyst must know three parameters:

- the discount rate
- the time period over which discounting is to be performed
- the amount of money or value that is to be discounted.

B.2.1 Discount Rate

The appropriate discount rate to use is often a controversial issue in the application of value-impact analysis. NRC Guidelines Section 4.3.3 states that the discount rates specified in the most recent version of OMB Circular A-94 are to be used in preparing regulatory analyses. Circular A-94 currently specifies use of a real discount rate (r) of 7% per year (OMB 1992). NRC Guidelines Section 4.3.3 further states that a discount rate of 3% should be used for sensitivity analysis to indicate the sensitivity of the results to the choice of discount rate.

When the time horizon associated with a regulatory action exceeds 100 years, Section 4.3.3 of the Guidelines specifies that the 7% real discount rate should not be used. Instead the net value should be calculated using the 3% real discount rate. In addition, the results should be displayed showing the values and impacts at the time they are incurred with no discounting (see Section 5.7).

OMB Circular A-94 defines the term "discount rate" as the interest rate used in calculating the present value of expected yearly benefits and costs. When a real discount rate is used as specified in Section 4.3 of the Guidelines, yearly benefits and costs should be in real or constant dollars. Circular A-94 defines "real or constant dollar values" as economic units measured in terms of constant purchasing power. A real value is not affected by general price inflation. Real values can be estimated by deflating nominal values with a general price index, generally the GDP deflator as discussed in Section 5.8.

B.2.2 Discrete Discounting

The following formula is used to determine the present value (PV) of an amount (F_t) at the end of a future time period:

$$PV = F_t/(1 + r)^t,$$

where r = the real annual discount rate (as fraction, not percent)
t = the number of years in the future in which the costs occur.

For example, to determine how much $750 to be received 25 years (t) hence is worth today, using a 7% real discount rate (r), the formula yields

$$PV = \$750/(1 + .07)^{25} = (\$750)(0.184)$$
$$= \$138$$

Table B.2 contains values of the discount factor $1/(1 + r)^t$ for discount rates (r) of 3% and 7% and for various values of t, the number of years. To find the present value of a stream of costs and revenues, the analyst should record the costs and revenues occurring in each year. Then, for each year, the net cost is determined by simply adding algebraically the costs and revenues for that year. After this has been done for each year, the net cost in each year is discounted to the present using Table B.2. The sum of these present values is the present value of the entire stream of costs and revenues. A sample use of this formula in value-impact analysis would be in determining the PV of implementation costs for industry and the NRC which occur in the future.

The above formula is used for discounting single amounts backward in time. However, some of the costs encountered in value-impact analysis recur on an annual basis. These include not only industry and NRC operating costs, but also the monetized values of the annual per-facility reductions in routine public and occupational dose due to operation (see Sections 5.7.2 and 5.7,4). Such costs can be discounted by the use of the following annuity formula (only if they are the same amount for each time period):

$$PV = C_A[(1 + r)^t - 1]/r(1 + r)^t$$

where C_A = identical annual costs
r = the real discount rate (as fraction, not percent)
t = the number of years over which the costs recur.

For example, if the increase in annual industry costs is $1,000, due to increased maintenance expenses, with a 7% real discount rate for 20 years, starting at the present time, the present value of these costs is

$$PV = (\$1,000)[(1 + .07)^{20} - 1]/(.07)(1 + .07)^{20}$$
$$= (\$1,000)(10.6) = \$10,600$$

Table B.3 contains values of the annuity discount factor: $[(1 + r)^t - 1]/r(1 + r)^t$, for real discount rates (r) of 3% and 7% and for various values of t, the number of years over which the costs are incurred.

In most cases, operating costs will start to be incurred at some date in the future, after which the real costs will be constant on an annual basis for the remaining life of the facility. To discount the costs in this situation, a combination of the above two methods or formulas is needed. For example, given the same $1,000 annual cost for a 20-year period at a 7% real discount rate, but starting five years in the future, the formula to calculate the PV is

$$PV = (\$1,000)[(1 + r)^{t_2} - 1]/r(1 + r)^{t_1}(1 + r)^{t_2}$$

where r = 7% discount rate (i.e., .07/yr)
t_1 = 5 years
t_2 = 20 years for annuity period.

Therefore, PV = ($1,000)(10.6)(0.713) = $7,560.

Table B.2 Present value of a future dollar (yearly compounding)

Year	3%	7%
1	0.971	0.935
2	0.943	0.873
3	0.915	0.816
4	0.889	0.763
5	0.863	0.713
6	0.838	0.666
7	0.813	0.623
8	0.789	0.582
9	0.766	0.544
10	0.744	0.508
11	0.722	0.475
12	0.701	0.444
13	0.681	0.415
14	0.661	0.388
15	0.642	0.362
16	0.623	0.339
17	0.605	0.317
18	0.587	0.296
19	0.570	0.277
20	0.554	0.258
25	0.478	0.184
30	0.412	0.131
40	0.307	0.0668
50	0.228	0.0339

Table B.3 Present value of annuity of a dollar, received at end of each year (yearly compounding)

Year	3%	7%
1	0.971	0.935
2	1.91	1.81
3	2.83	2.62
4	3.72	3.39
5	4.58	4.10
6	5.42	4.77
7	6.23	5.39
8	7.02	5.97
9	7.79	6.52
10	8.53	7.02
11	9.25	7.50
12	9.95	7.94
13	10.6	8.36
14	11.3	8.75
15	11.9	9.11
16	12.6	9.45
17	13.2	9.76
18	13.8	10.1
19	14.3	10.3
20	14.9	10.6
25	17.4	11.7
30	19.6	12.4
40	23.1	13.3
50	25.7	13.8

Tables B.2 and B.3 contain the appropriate discount factors to be multiplied together. Additional background on discrete discounting can be found in EPRI (1986), DOE (1982), and Wright (1973).

B.2.3 Continuous Discounting

Discrete discounting, as discussed above, deals with costs and revenues that occur at discrete instances over a period of time. For most regulatory analyses, discrete discounting and the present value factors shown in Tables B.2 and B.3 can be used. Technically, discrete discounting does not correctly account for consequences that occur constantly, but the difference is viewed as minimal, and the additional effort is generally not warranted in a standard regulatory analysis.

Continuous discounting should be used in regulatory analyses beyond the standard analysis when costs and revenues occur continuously over a period of time, such as those which must be weighed by an accident frequency over the remaining life of a facility. The accident frequency is a continuous variable, although the real cost of the accident consequences is constant.

The formula for continuous discounting is derived from the discrete discounting formula as follows. Assume that in one period (t), the time will be subdivided into n intervals. The formula for discrete discounting, with a real discount rate of r, is $1/(1 + r/n)^n$. As we subdivide the time period into an infinite number of intervals in the limit, we would abandon discrete intervals altogether and so set the limit as

$$\lim_{n \to \infty} 1/(1 + r/n)^n = \exp(-r)$$

For t periods, instead of one period as above, the formula becomes exp(-rt), where r and t are defined over the same time period.

The monetized values for the reductions in public and occupational dose due to accidents, as well as the avoided onsite and offsite property damage costs, require continuous discounting. To calculate the present value for the public health (accident) and offsite property attributes, when the monetary value or cost C_O can occur with a frequency f, Strip (1982) provides the following formula:

$$\int_{t_i}^{t_f} C_o f \exp(-rt)dt = C_o f[\exp(-rt_i) - \exp(-rt_f)]/r$$

where t_i = time of onset of accident risk
t_f = time of end of accident risk.

For public (accident) risk, the product $C_O f$ is replaced by Z_{PHA} representing the monetary value of avoided risk before discounting (\$/facility-yr [see Section 5.7.1.3]). As an example, assume the monetary value of avoided public risk due to an accident is \$1.0E+4/facility-yr ($C_o f$ = \$1.0E+4). The facility is operational ($t_i = 0$) with a remaining lifetime of 25 years ($t_f = 25$). For an annual discount rate of 7% (r = .07/yr) the present value of avoided risk (monetized) becomes

$$
\begin{aligned}
PV &= (\$1.0E+4/yr)\,[\exp\{-(.07)(0)\} \\
&\quad - \exp\{-(.07)(25)\}]/(.07/yr) \\
&= (\$1.0E+4)(11.8) \\
&= \$1.18E+5/\text{facility}
\end{aligned}
$$

To determine the present value of a reduction in offsite property risk, the frequency (f in the general equation above) is replaced with the frequency reduction (Δf). As an example, let the frequency reduction (Δf) be 1.0E-5/facility-yr and the cost (C_O) be \$1.0E+9. The annual discount rate is 7% (r = .07/yr), and the reduction in accident frequency takes place 5 years in the future ($t_i = 5$) and will remain in place for 20 years ($t_f = 5 + 20 = 25$). The present value of the avoided offsite property damage becomes

$$
\begin{aligned}
PV &= (\$1.0E+9)(1.0E-5/yr)[\exp\{-(.07)(5)\} - \exp\{-(.07)(25)\}]/(.07/yr) \\
&= (\$1.0E+9)(1.0E-5)(7.58) = \$7.58E+4/\text{facility}
\end{aligned}
$$

To calculate present values for the occupational health (accident) and onsite property attributes, the continuous discounting formula must be modified. The modifications account for the fact that 1) some components of severe accident costs are not represented by constant annual charges as noted in Section B.2.2, and 2) the single-event present values must be reintegrated because the accident costs and risks would be spread over a period of time (e.g., over the remaining plant lifetime for replacement power costs and over the estimated 10 years for cleanup and decontamination following a severe accident, for onsite property damage). Sections 5.7.3.3 and 5.7.6.4 address these modifications and provide estimation guidelines for regulatory initiatives that affect accident frequencies in current and future years.

B.3 Occupational Exposure Experience

Two documents contain considerable information related to occupational exposure experience at nuclear power plants and some non-reactor facilities. In the first (NUREG/CR-5035), Beal et al. (1987) state the following concerning generic dose rate data for use in regulatory analyses:

> "...The NRC is generally concerned with the average exposures potentially experienced at all plants within a specific class (i.e., BWRs, PWRs, or PWRs manufactured by a particular vendor), rather than with the exposures at a specific plant. Therefore, it is desirable to have a generic dose-rate data base available to NRC analysts for making radiation exposure estimates."

The dose rates have been classified by Beal et al. (1987) according to the EEDB (United Engineers and Constructors, Inc. 1988b) code-of-accounts for nuclear power plant systems and components. The analyst can estimate the radiation exposure as the product of the estimated labor hours for work on a specific EEDB system/component and the dose rate for that system/component. Tables B.4 and B.5 list occupational dose rates for PWR and BWR systems and components, respectively, by EEDB classification.

Chapter 4 of NUREG/CR-5035 provides illustrative examples of the estimation of occupational radiation exposure for specific tasks at a power plant. Labor-hour estimates are obtained from the EEDB (United Engineers and Constructors, Inc. 1986). Adjustments to account for differences in labor productivity are taken from Riordan (1986). If hardware is to be removed, and/or a learning curve is to be involved, these effects are accounted for using information from Sciacca et al. (1986).

Table B.4 Occupational dose rates by EEDB classification for PWR systems and components (Beal et al. 1987)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
REACTOR EQUIPMENT		
221.122	Reactor Vessel Closure & Attachments	650
221.123	Reactor Vessel Studs, Fasteners, Seals, & Gaskets	140
221.131-2	Reactor Vessel Upper and Lower Internals	800
221.211	Control Rods	---
221.212	Control Rod Drives	1400
221.213	Control Rod Drive Missile Shield	---
221.214	CRDM Seismic Supports	---
MAIN HEAT TRANSFER TRANSPORT SYSTEM		
222.1111	Main Coolant Pumps & Drive	65
222.118	Main Coolant Pumps Instr. & Control	2
222.119	Main Coolant Pumps Foundations/Skids	40
222.12	Reactor Coolant Piping System	270
222.1321	Steam Generators	
	- at manway and inside steam generator	5100
	- manway vicinity and general area	110
222.1431	Pressurizer	95
222.1432	Pressurizer Relief Tank	32
222.148	Pressurizer Instrumentation & Control	15
222.149	Pressurizer Foundation/Skids	---
RESIDUAL HEAT REMOVAL SYSTEM		
223.111	RHR Pumps & Drives	45
223.121	RHR Heat Exchangers	35
223.15,16,17	RHR Piping System	65
223.18	RHR Instrumentation & Control	45
SAFETY INJECTION SYSTEM		
223.311	Safety Injection System Pumps and Drives	8
223.312	Boron Injection Pumps and Drive	---
223.331	Accumulator Tank	6
223.332-3	Boron Injection Tanks	70
223.334	Refueling Water Storage Tank	<1
223.35,36,37	Safety Injection System Piping System	55

Table B.4 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
223.38	Safety Injection System Instr. & Control	5
CONTAINMENT SPRAY SYSTEM		
223.411	Containment Spray Pumps & Motors	15
223.421	Containment Spray Heat Exchanger	---
223.431	Containment Spray Additive Tank	<1
223.45,46,47	Containment Spray Piping System	25
223.48	Containment Spray Instrument. & Control	120
COMBUSTIBLE GAS CONTROL SYSTEM		
223.55,56,57	Combustible Gas Control System Piping	10
223.58	Combustible Gas Control System Instr. & Control	10
223.591	Hydrogen Recombiner	10
LIQUID WASTE SYSTEM		
Primary Equipment Drain System		
224.1111-33	Tanks, Pumps, & Motors	250
224.1141	Equipment Drain Filter	50
224.115,116,117	Equipment Drain Piping	35
Miscellaneous Drain Waste System		
224.1211-32	Tanks, Pumps, & Motors	170
224.1241-3	Waste Filters, Demineralizers, & R/O Units	150
224.125,126,127	Misc. Waste Piping System	75
Detergent Waste System		
224.1311-32	Tanks, Pumps, & Motors	2
224.1241-4	Waste Filters, Demineralizers, & R/O Units	3
224.135,136,137	Detergent Waste Piping System	2
Chemical Waste System		
224.1411-31	Tanks, Pumps, & Motors	60
224.144	Purification & Filter Equipment	---
225.145,146,147	Chemical Waste Piping System	13

Table B.4 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
Steam Generator Blowdown System		
224.1511-3	Tanks, Pumps, & Heat Exchangers	3
224.15141-4	Demineralizers and Filters	4
224.151,1516,1517	S.G.B.D. Piping System	8
224.1518	S.G.B.D. Instrument. & Control	2
Regen. Chemical Waste System		
224.1611-32	Tanks, Pumps, & Motors	---
224.1641-3	Demineralizers, Filters, & Evaporator	100
224.165,166,167	Regen. Waste Piping System	---
224.171	Chemical Feed Package (tks., pumps, piping, etc)	2
224.18	Liquid Waste System Instr. & Control	2
RADIOACTIVE GAS WASTE PROCESSING SYSTEM		
224.2111-32	Radioactive Gas Compressors, Drives, & Decay Tanks	7
224.2141	Recombiner Packages	2
224.2142	Gas Waste Vent Filter	3
224.215,216,217	Radioactive Gas Waste Piping System	2
224.218	Radioactive Gas Waste Instr. & Control	---
SOLID WASTE SYSTEM		
Dry Active Waste Volume Reduction		
224.3111-32	Tanks, Pumps, & Motors	120
224.3141	Filters	2000
Volume Reduction and Solidification System		
224.325,326,327	Solid Waste System Piping	7
224.328	Solid Waste System Instrument. & Control	2
FUEL HANDLING AND STORAGE		
225.111-4	New and Spent Fuel Cranes and Hoists	25
225.131-2	Transfer Systems	210
225.31-2	Reactor Service & Fuel Storage Pool Service Platform	13
225.41	New Fuel Storage Racks	<1
225.42	Spent Fuel Storage Racks	---
225.4311-45	Spent Fuel Pool Cleaning & Purification Equipment	85

Table B.4 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
225.435,436,436	Spent Fuel Pool Clean. & Purif. Piping System	15
225.438	Spent Fuel Pool Clean. & Purif. System Instrument & Control	---
INERT GAS SYSTEM		
226.11	H_2/N_2 Gas Supply System	20
REACTOR MAKEUP WATER SYSTEM		
226.311	Reactor Makeup Water Pumps & Drives	4
226.331	Reactor Makeup Water Tank	120
226.35,36,37	Reactor Makeup Water Piping System	20
226.38	Reactor Makeup Water System Instr. & Control	3
COOLANT TREATMENT & RECYCLE		
226.4111-5	Chemical & Volume Control System Pumps, Motors, & Equipment	13
226.4121-8	CVCS Heat Transfer Equipment	80
226.4121-7	CVCS Tanks and Pressure Vessels	140
226.4141-5	CVCS Purification and Filtration Equipment	1800
226.415,416,417	CVCS Piping System	95
226.418	CVCS Instr. & Control	21
226.4191-2	Foundations & Skids for Boron System Equipment	22
226.4211-33	Boron Recycle System Pumps, Motors, Tanks, & Equip.	100
226.4241-7	Boron Recycle System Purif. & Filter Equipment	38
226.425,426,427	Boron Recycle Piping System	---
226.428	Boron Recycle Instrument. & Control	3
FLUID LEAK DETECTION SYSTEM		
226.6	Fluid Leak Detection System	---

Table B.4 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
AUXILIARY COOLING SYSTEMS		
Nuclear Service Water System		
226.7111-2	Safeguards Cooling Tower Pumps, Equip, & Cooling Tower	---
226.715,716,717	Cooling Tower Piping System	80
226.718	Cooling Tower Instr. & Control	---
Primary Component Cooling Water		
226.7211-31	Prim. Comp. Cooling Water Pumps, Motors & Equip. Tanks	2
226.725,726,727	Prim. Comp. Cool. Water Piping System	25
226.728	Prim. Comp. Cool. Water Instr. & Control	---

CRDM = Control Rod Drive Mechanism
CVCS = Chemical and Volume Control System
EEDB = Energy Economic Data Base
mr = millirem
SGBD = Steam Generator Blowdown
* Average of across-plant "typical" values

Table B.5 Occupational dose rates by EEDB classification for BWR systems and components (Beal et al. 1987)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
REACTOR EQUIPMENT		
221.122-133	Reactor Vessel Closure & Attachments, Studs, Fasteners, Seals, Gaskets, Core Support, and Shroud Assembly	---
221.134	Jet Pump Assemblies	4400
221.135	Fluid Distribution Assemblies	210
221.136	Steam Dryer Assembly	800
221.211	Control Rods	170
221.212	Control Rod Drives	110
MAIN HEAT TRANSFER TRANSPORT SYSTEM		
222.1111	Reactor Recirculation Pumps & Motors	90
222.15,16,17	Recirculation Piping System	240
222.18	Reactor Recirculation Instrument. & Control	200
RESIDUAL HEAT REMOVAL SYSTEM		
223.11	RHR Pumps & Drives	60
223.12	RHR Heat Exchangers	320
223.14	RHR Purification & Filtration Equipment	---
223.15,16,17	RHR Piping System	100
223.18	RHR Instrumentation & Control	80
REACTOR CORE ISOLATION COOLING SYSTEM		
223.21-24	RCIC Pumps, Motors, & Equipment	90
223.25,26,27	RCIC Piping System	100
223.28	RCIC Instrumentation & Control	---
HIGH PRESSURE CORE SPRAY SYSTEM		
223.31-34	HPCS Pumps, Motors, & Strainers	30
223.35,36,37	HPCS Piping System	100
223.38	HPCS Instrumentation & Control	20

Table B.5 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
LOW PRESSURE CORE SPRAY SYSTEM		
223.41-44	LPCS Pumps, Motors, & Strainers	15
223.45,46,47	LPCS Piping System	190
223.48	LPCS Instrumentation & Control	---
COMBUSTIBLE GAS CONTROL SYSTEM		
223.55,56,57	Combustible Gas Control System Piping System	1
223.58	Combust. Gas Control System Instr. & Control	---
223.591	Hydrogen Recombiner	20
STANDBY LIQUID CONTROL SYSTEM		
223.61	Standby Liquid Control System Pump & Motor	1
223.631	SLCS Main Storage Tank	5
223.632	SLCS Test Tank	---
223.65,66,67	SLCS Piping System	55
223.68	SLCS Instrumentation & Control	---
STANDBY GAS TREATMENT SYSTEM		
223.711-722	SGTS Fans, Motors, Heat Transfer & Equipment	---
223.74	SGTS Purification & Filtration Equipment	1
223.75,76,77	SGTS Piping System	---
223.78	SGTS Instrumentation & Control	---
LIQUID WASTE SYSTEM		
High Purity System		
224.111-113	High Purity Collection Tanks, Pumps, Motors, & Equipments	280
224.114	High Purity Waste Filter, Demineralizers	---
224,115,116,117	High Purity Waste Piping System	10

Table B.5 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
Low Purity System		
224.121-123	Low Purity Collection Tanks, Pumps, Motors, & Equipment	190
224.124	Low Purity Waste Evaporators Demineralizers and Filters	---
224.125,126,127	Low Purity Waste Piping System	60
Detergent Waste System		
224.131-133	Detergent Waste Tanks, Pumps, Motors, & Equipment	40
224.134	Detergent Waste Filter, Demineralizers, R/O Unit Package	65
224.135,136,137	Detergent Waste Piping System	2
Chemical Waste System		
224.141-143	Chemical Waste Tanks, Pumps, Motors, & Equipment	40
224.144	Chemical Waste Purification & Filter Equipment	---
224.145,146,147	Chemical Waste Piping System	---
Cleanup Floor Drain Waste System		
224.15	Cleanup Floor Drain Waste Pumps, Motors, & Eq.	---
Chemical Waste Train		
224.16	Regen. Waste Pumps, Motors, Equipment, & Piping	---
224.17	Misc. Radwaste Equipment	---
224.18	Liquid Waste System Instrument & Control	---
RADIOACTIVE GAS WASTE PROCESSING		
224.211-214	Gas Waste Processing System Equipment	---
224.215,216,217	Radioactive Gas Waste Piping System	10
224.218	Radioactive Gas Waste Instrument & Control	---

Table B.5 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
SOLID WASTE SYSTEM		
224.321	Dry Active Waste Volume Reduction Centrifuge, Pumps, Motors, & Equipment	200
224.322-324	Solid Waste System Equipment, Tanks, Purification & Filtration	---
224.325,326,327	Solid Waste System Piping System	250
224.328	Solid Waste System Instruments, & Control	---
FUEL HANDLING AND STORAGE		
225.11	Fuel Handling Equipment, Cranes, & Hoists	20
225.12-14	Fuel Handling Tools, Transfer Systems, & Machines	---
225.2-3	Remote Viewing Equipment, Refueling Platform, Fuel Handling Platform	4
225.41-42	Fuel Storage Equipment & Racks	---
225.431-434	Spent Fuel Pool Cleaning & Purification Pumps Motors, Equipment, Filters, & Demineralizers	400
225.435,436,437	Spent Fuel Pool Clean. & Purif. Piping Systems	40
225.438	Spent Fuel Pool Clean. & Purif. Piping System Instrument & Cont	---
REACTOR WATER CLEANUP SYSTEM		
226.41-42	RWCU System Pumps, Motors, & Heat Exchangers	120
226.43	RWCU Tanks & Pressure Vessels	2
226.44	RWCU Purification & Filter Equipment	80
226.45,46,47	RWCU Piping System	120
226.48	RWCU System Instrument & Control	---
FLUID LEAK DETECTION SYSTEM		
226.6	Fluid Leak Detection System	---
AUXILIARY COOLING SYSTEMS		
226.71	Essential Service Water System	---
226.72	Closed Cooling Water System	---
226.731-732	Plant Chilled Water System Pumps, Motors, & Heat Transfer Equipment	80

Table B.5 (Continued)

EEDB Code-of-Account	Description	Average Dose Rate* (mr/hr)
226.734	Purification & Filtration Equipment	---
226.735,736,737	Plant Chilled Water Piping System	---
226.738	Plant Chilled Water Instrument & Control	---
FEED HEATING SYSTEM		
234.1	Feed Water Heaters	1
234.211	Feed Water Pumps	2
234.25	Feed Water Piping	70
234.26	Feed Water Valves	850
OTHER TURBINE PLANT EQUIPMENT		
235.115	Main Vapor System Piping	50
235.116	Main Vapor System Valves	260
235.117	Main Vapor System Misc. Piping	2
235.118	Main Vapor System Instrument & Control	100
235.21	Main Steam/Reheat Vents & Drains	16
235.35	T.B. Closed Cooling Water System Piping	20
235.4	Demin. Water Makeup System	1
235.631	Neutralization System Tank	1

HPCS = High Pressure Core Spray
LPCS = Low Pressure Core Spray
mr = millirem
RCIC = Reactor Core Isolation Cooling
RWCU = Reactor Water Cleanup
SGTS = Standby Gas Treatment System
SLCS = Standby Liquid Control System
TB = Turbine Building
* Average of across-plant "typical" values

The NRC maintains occupational exposure data in the Radiation Exposure Information and Reporting System (REIRS). The following six categories of licensees have reported occupational exposure data:

1. power reactors (LWRs)
2. industrial radiographers
3. fuel processors, fabricators, and reprocessors
4. manufacturers and distributors of byproduct material
5. independent spent fuel storage installations
6. facilities for land disposal of low level waste.

Annual reports for 1993 were received from 360 NRC licensees, of which 114 were operators of power reactors. Raddatz and Hagemeyer (1995) have compiled and processed the 1993 and previous years' data in the second document related to occupational exposure experience of NRC-licensed facilities. No data from Agreement State licensees are included in the report.

Data limitations are discussed in Chapter 2 of Raddutz and Hagemeyer (1995), prior to the presentation of the processed results. Annual exposure data are given for the six facility classes listed above. Annual occupational exposure data for 1991-1993 are tabulated in Tables B.6 to B.8 for industrial radiographers, manufacturers and distributors of byproduct material, and fuel fabricators. For low level waste disposers and independent spent fuel storers, the annual number of workers with measurable doses and the collective and average doses for 1991-1993 are shown in Table B.9. For power reactors, the annual occupational exposure data from 1973 through 1993 are presented for BWRs, PWRs, and LWRs in Tables B.10 to B.12, respectively.

Chapter 4 of Raddatz and Hagemeyer (1995) examines occupational exposure data at LWRs in more detail. Included are annual whole body dose distributions; plant rankings by the collective dose per reactor; and the average, median, and extreme values of the collective dose per reactor. Table B.13 lists the numbers of employees and collective and average doses for 1993 as a function of occupation and personnel type for LWRs.

B.4 Calculational Method for Table 5.3, "Expected Population Doses for Power Reactor Release Categories"

The information in this section is from the letter report, "MACCS Economic Consequence Tables for Regulatory Applications" (Young 1995) prepared for the NRC. It provides an overview of the calculations and assumptions used in the preparation of Table 5.3. Young's results represent mean results conditional on the occurrence of each release category.

B.4.1 Introduction

The MACCS Version 1.5.11.1 was used to complete the calculations performed for the analysis reported in Young (1995). MACCS was designed to assess the potential off-site dose, health, and economic consequences of postulated nuclear power plant (NPP) accidents. Interdiction criteria specified by the user determine the dose levels at which long-term mitigative actions are implemented.

Table B.6 1991-1993 annual occupational exposure information for industrial radiographers (Raddatz and Hagemeyer 1995)

Year	Type of License	Number of Licenses	Number of Monitored Workers	Workers with Measurable Doses	Collective Dose (person-cSv or person-rem)	Average Measurable Dose (cSv or rems)
1993	Single location	39	673	183	23	0.13
	Multiple locations	137	4,046	2,824	1,603	0.57
	Total	176	4,721	3,007	1,627	0.54
1992	Single location	48	771	182	37	0.20
	Multiple locations	198	5,392	4,082	1,827	0.45
	Total	246	6,703	4,265	1,864	0.44
1991	Single location	56	822	338	44	0.13
	Multiple location	192	5,998	4,311	2,116	0.49
	Total	248	6,820	4,649	2,160	0.46

Table B.7 1991-1993 annual occupational exposure information for byproduct manufacturers and distributors (Raddatz and Hagemeyer 1995)

Year	Type of License	Number of Licenses	Number of Monitored Workers	Workers with Measurable Doses	Collective Dose (person-cSv or person-rem)	Average Measurable Dose (cSv or rem)
1993	M & D-Broad	8	2,455	925	512	0.55
	M & D-Limited	50	2,458	1,329	168	0.13
	Total	58	4,913	2,254	680	0.30
1992	M & D-Broad	11	3,632	1,674	718	0.43
	M & D-Limited	56	1,578	576	72	0.13
	Total	67	5,210	2,250	784	0.35
1991	M & D-Broad	12	3,732	1,443	674	0.47
	M & D-Limited	46	1,198	513	47	0.09
	Total	58	4,930	1,956	721	0.37

Table B.8 1991-1993 annual occupational exposure information for fuel fabricators (Raddatz and Hagemeyer 1995)

Year	Type of License	Number of Licenses	Number of Monitored Workers	Workers with Measurable Doses	Collective Dose (person-rems or person-cSv)	Average Measurable Dose (rems or cSv)
1993	Uranium Fuel Fab	8	9,649	2,611	339	0.13
1992	Uranium Fuel Fab	11	8,439	5,061	545	0.11
1991	Uranium Fuel Fab	11	11,702	3,929	378	0.10

Table B.9 Annual occupational doses for low level waste disposal and spent fuel storage facilities, 1991-1993 [Raddatz and Hagemeyer 1995]

Licensee	Year	Workers with measurable doses	Collective dose (person-cSv)	Average measurable dose (cSv)
Low Level Waste Disposers	1991	147	39	0.27
	1992	82	37	0.45
	1993	76	21	0.27
Independent Spent Fuel Storers	1991	24	4	0.17
	1992	85	11	0.13
	1993	52	14	0.26

Table B.10 Summary of 1973-1993 annual occupational exposure information reported by commercial BWRs (Raddatz and Hagemeyer 1995)

Year	Number of Reactors Included	Annual Collective Doses (person-cSv or person-rem)	No. of Workers With Measurable Doses	Gross Electricity Generated (MW-yr)	Average Dose Per Worker (cSv or rem)	Average Collective Dose Per Reactor (person-cSv or person-rem)	Average No. Personnel With Measurable Doses Per Reactor	Average Collective Dose per MW-yr (person-cSv /MW-yr)	Average Electricity Generated Per Reactor (MW-yr)	Average Maximum Dependable Capacity Net (MWe)
1973	12	4,564	5,340	3,393.9	0.85	380	445	1.34	283	438
1974	14	7,095	8,769	4,060.2	0.81	507	626	1.75	290	485
1975	18	12,611	14,607	5,786.4	0.86	701	812	2.18	321	595
1976	22	12,300	16,604	8,137.9	0.74	559	755	1.51	370	630
1977	23	19,041	21,388	9,102.5	0.89	828	930	2.09	396	637
1978	25	15,273	20,278	11,856.0	0.75	611	811	1.29	474	660
1979	25	18,325	25,245	11,671.0	0.73	733	1,010	1.57	467	660
1980	26	29,530	34,094	10,868.2	0.87	1,136	1,311	2.72	418	663
1981	26	25,471	34,755	10,899.2	0.73	980	1,337	2.34	419	663
1982	26	24,437	32,235	10,614.6	0.76	940	1,240	2.30	408	663
1983	26	27,455	33,473	9,730.1	0.82	1,056	1,287	2.82	374	663
1984	27	27,097	41,105	10,019.2	0.66	1,004	1,522	2.70	371	754
1985	29	20,573	38,237	12,284.0	0.54	709	1,319	1.67	424	775
1986	30	19,349	37,928	12,102.1	0.51	645	1,264	1.60	403	786
1987	32	16,717	41,737	15,109.0	0.40	522	1,304	1.11	472	832
1988	34	17,983	40,305	16,665.4	0.45	529	1,185	1.08	490	845
1989	36	15,549	44,360	17,543.5	0.35	432	1,232	0.89	487	857
1990	37	15,780	41,577	21,336.1	0.38	426	1,124	0.74	577	862
1991	37	12,005	38,492	21,505.8	0.31	324	1,040	0.56	581	860
1992	37	13,309	42,095	20,592.2	0.32	360	1,138	0.65	557	859
1993	37	12,221	38,309	21,995.6	0.31	330	1,062	0.56	594	798

*Includes only those reactors that had been in commercial operation for at lease one full year as of December 31 of each of the indicated years, and all figures are uncorrected for multiple reporting of transient individuals.

Table B.11 Summary of 1973-1993 annual occupational exposure information reported by commercial PWRs (Raddatz and Hagemeyer 1995)

Year	Number of Reactors Included	Annual Collective Doses (person-cSv or person-rem)	No. of Workers With Measurable Doses	Gross Electricity Generated (MW-yr)	Average Dose Per Worker (cSv or rem)	Average Collective Dose Per Reactor (person-cSv or person-rem)	Average No. Personnel With Measurable Doses Per Reactor	Average Collective Dose per MW-yr (person-cSv /MW-yr)	Average Electricity Generated Per Reactor (MW-yr)	Average Maximum Dependable Capacity Net (MWe)
1973	12	9,398	9,440	3,770.2	1.00	783	787	2.49	314	544
1974	19	6,555	9,370	6,530.7	0.70	345	493	1.00	344	591
1975	26	8,268	10,884	11,982.5	0.76	318	419	0.69	461	647
1976	30	13,807	17,588	13,325.0	0.79	460	586	1.04	444	701
1977	34	13,467	20,878	17,345.8	0.65	396	614	0.78	510	688
1978	39	16,528	25,700	19,840.5	0.64	424	659	0.83	509	706
1979	42	21,657	38,828	18,255.0	0.56	516	924	1.19	435	746
1980	42	24,265	46,237	18,289.3	0.52	578	1,101	1.33	435	746
1981	44	28,673	47,351	20,553.7	0.61	652	1,076	1.40	467	752
1982	48	27,753	52,146	22,140.6	0.53	578	1,086	1.25	461	777
1983	49	29,017	52,173	23,195.5	0.56	592	1,065	1.25	473	785
1984	51	28,138	56,994	26,478.4	0.49	552	1,118	1.06	519	809
1985	53	22,469	54,633	29,470.7	0.41	424	1,031	0.76	556	820
1986	60	23,032	62,995	33,593.0	0.37	384	1,050	0.69	560	878
1987	64	23,684	62,597	37,007.3	0.38	370	978	0.64	578	900
1988	68	22,786	62,921	42,929.7	0.36	335	925	0.53	631	885
1989	71	20,381	63,894	44,679.5	0.32	287	900	0.46	629	897
1990	73	20,812	67,081	46,955.6	0.31	285	919	0.44	643	907
1991	74	16,510	60,269	51,942.6	0.27	223	814	0.32	702	913
1992	73	15,985	61,048	53,419.8	0.26	219	836	0.30	732	923
1993	73	14,142	56,588	50,480.6	0.25	194	775	0.28	692	919

*Includes only those reactors that had been in commercial operation for at least one full year as of December 31 of each of the indicated years, and all figures are uncorrected for multiple reporting of transient individuals.

Table B.12 Summary of 1973-1993 annual occupational exposure information reported by commercial LWRs (Raddatz and Hagemeyer 1995)

Year	Number of Reactors Included	Annual Collective Doses (person-cSv or person-rem)	No. of Workers With Measurable Doses	Gross Electricity Generated (MW-yr)	Average Dose Per Worker (cSv or rem)	Average Collective Dose Per Reactor (person-cSv or person-rem)	Average No. Personnel With Measurable Doses Per Reactor	Average Collective Dose per MW-yr (person-cSv /MW-yr)	Average Electricity Generated Per Reactor (MW-yr)	Average Maximum Dependable Capacity Net (MWe)	Percent of Maximum Dependable Capacity Achieved
1973	24	13,962	14,780	7,164.1	0.94	582	616	1.95	299	491	61%
1974	33	13,650	18,139	10,590.9	0.75	414	550	1.29	321	546	59%
1975	44	20,879	25,491	17,768.9	0.82	475	579	1.18	404	626	65%
1976	52	26,107	34,192	21,462.9	0.76	502	658	1.22	413	671	62%
1977	57	32,508	42,266	26,448.3	0.77	570	742	1.23	464	667	70%
1978	64	31,801	45,978	31,696.5	0.69	497	718	1.00	495	688	72%
1979	67	39,982	64,073	29,926.0	0.62	597	956	1.34	447	714	63%
1980	68	53,795	80,331	29,157.5	0.67	791	1,181	1.84	429	714	60%
1981	70	54,144	82,106	31,452.9	0.66	773	1,173	1.72	449	719	63%
1982	74	52,190	84,381	32,755.2	0.62	705	1,140	1.59	443	737	60%
1983	75	56,472	85,646	32,925.6	0.66	753	1,142	1.72	439	743	59%
1984	78	55,235	98,099	36,497.6	0.56	708	1,258	1.51	468	790	59%
1985	82	43,042	92,870	41,754.7	0.46	525	1,133	1.03	509	804	63%
1986	90	42,381	100,923	45,695.1	0.42	471	1,121	0.93	508	847	60%
1987	96	40,401	104,334	52,116.3	0.39	421	1,087	0.78	543	877	62%
1988	102	40,769	103,226	59,595.1	0.39	400	1,012	0.68	584	871	67%
1989	107	35,930	108,254	62,223.0	0.33	336	1,012	0.58	582	883	66%
1990	110	36,592	108,658	68,291.7	0.34	333	988	0.54	621	892	70%
1991	111	28,515	98,761	73,448.4	0.29	257	890	0.39	662	895	74%
1992	110	29,294	103,143	74,012.0	0.28	266	938	0.40	673	901	75%
1993	110	26,363	95,896	72,476.2	0.27	240	872	0.36	659	878	75%

*Includes only those reactors that had been in commercial operation for at least one full year as of December 31 of each of the indicated years, and all figures are uncorrected for multiple reporting of transient individuals.

Table B.13 1993 numbers of employees and collective and average doses by occupation and personnel type at LWRs (Raddatz and Hagemeyer 1995)

WORK AND JOB FUNCTION	STATION EMPLOYEES PERSON-cSv	STATION EMPLOYEES % OF TOTAL	UTILITY EMPLOYEES PERSON-cSv	UTILITY EMPLOYEES % OF TOTAL	CONTRACT WORKERS PERSON-cSv	CONTRACT WORKERS % OF TOTAL	TOTAL PER WORK FUNCTION PERSON-cSv	TOTAL PER WORK FUNCTION % OF TOTAL
BOILING WATER REACTORS								
REACTOR OPS & SURV	1,209	9.9%	81	0.7%	459	3.8%	1,749	14.3%
ROUTINE MAINTENANCE	2,140	17.5%	199	1.6%	3,788	31.1%	6,127	50.2%
IN-SERVICE INSPECTION	107	0.9%	35	0.3%	723	5.9%	865	7.1%
SPECIAL MAINTENANCE	659	5.4%	175	1.4%	1,453	11.9%	2,287	18.8%
WASTE PROCESSING	154	1.3%	9	0.1%	128	1.0%	291	2.4%
REFUELING	241	2.0%	97	0.8%	539	4.4%	877	7.2%
TOTAL	4,510	37.0%	596	4.9%	7,090	58.1%	12,196	100.0%
PRESSURIZED WATER REACTORS								
REACTOR OPS & SURV	747	5.2%	31	0.2%	470	3.3%	1,249	8.6%
ROUTINE MAINTENANCE	1,590	11.0%	608	4.2%	2,873	19.9%	5,072	35.1%
IN-SERVICE INSPECTION	167	1.2%	188	1.3%	1,652	11.4%	2,006	13.9%
SPECIAL MAINTENANCE	592	4.1%	192	1.3%	2,805	19.4%	3,589	24.8%
WASTE PROCESSING	161	1.1%	9	0.1%	207	1.4%	378	2.6%
REFUELING	604	4.2%	254	1.8%	1,305	9.0%	2,163	15.0%
TOTAL	3,862	26.7%	1,282	8.9%	9,312	64.4%	14,457	100.0%
ALL LIGHT WATER REACTORS								
REACTOR OPS & SURV	1,957	7.3%	112	0.4%	929	3.5%	2,997	11.2%
ROUTINE MAINTENANCE	3,730	14.0%	807	3.0%	6,661	25.0%	11,199	42.0%
IN-SERVICE INSPECTION	274	1.0%	222	0.8%	2,375	8.9%	2,871	10.8%
SPECIAL MAINTENANCE	1,251	4.7%	367	1.4%	4,258	16.0%	5,877	22.0%
WASTE PROCESSING	316	1.2%	18	0.1%	335	1.3%	669	2.5%
REFUELING	845	3.2%	351	1.3%	1,844	6.9%	3,040	11.4%
TOTAL	8,373	31.4%	1,878	7.0%	16,402	61.5%	26,653	100.0%

The following scenario assumptions and input data were selected by Young and NRC staff as a basis for the analysis:

1. 80th percentile weather data, as defined in draft NUREG/CR-6295 (Davis et al. 1995) were used as the meteorological input data.

2. The site data for the analysis were chosen to represent an 80th percentile NPP site in terms of the population density surrounding the site.

3. Calculations were performed for each of BWR and PWR source terms defined by Nourbakhsh (1992) as representative of severe LWR accident source terms.

4. NUREG-1150 (NRC 1991) emergency response assumptions were implemented as reported in NUREG/CR-4551 (Sprung et al. 1990).

5. The values assigned to the MACCS food ingestion model input parameters PSCMILK, PSCOTH, and GCMAXR are those values recommended by Mubayi as corrections to the values used in the NUREG-1150 analyses (Mubayi 1994). The PSCMILK and PSCOTH parameters define the levels of ground contamination above which crops are interdicted for accidents occurring during the growing season. GCMAXR defines the levels of ground contamination above which land is restricted from agricultural production.

6. Consequence values represent mean results and consequences within a 50-mile radius of the release.

B.4.2 MACCS Input Parameter Assumptions

NUREG-1150 MACCS input parameter values as provided and discussed in Sprung et al. (1990) were applied in the calculations except for those parameters discussed below. In addition, the values recommended by Mubayi (1994) as corrections to the NUREG-1150 values for MACCS input parameters PSCMILK, PSCOTH, and GCMAXR were used.

Meteorological Data

One year of meteorological data from Charleston, South Carolina was selected from Davis et al. (1995) to represent the conservative case (80th percentile) weather data. Wind roses were defined in the EARLY input file. The peak sector was assigned a 15% frequency, the adjacent sectors a frequency of 11%, and the remaining sectors were assigned a frequency of 4.85%. The wind rose sector containing the maximum population for the site was defined as the peak sector. The definition of the wind roses for the site is consistent with the method used to define the 80th percentile wind rose in Davis (1995).

Site Data

Population and land use, data for the Peach Bottom NPP, as defined by the SECPOP90 software package, was implemented in this analysis (Humphreys 1995). The population data provided by SECPOP90 is based on 1990 data. Peach Bottom is at the 84th percentile in terms of U.S. NPP site population density within 30 miles and the 79th percentile in terms of population density within 20 miles (Young 1994). Peach bottom is located within the state of Pennsylvania.

Source Term

Calculations were performed for all of the source-term release categories defined by Nourbakhsh (1992). The accident progression characteristics of these release categories were extracted from Gregory (1995). The analyst is referred to these two references for a detailed discussion of the derivation and application of these source term release categories.

Protection Actions

The duration of the emergency phase was defined as four rather than seven days as in the NUREG-1150 analysis.

The dose criterion for hot spot and normal relocation during the emergency phase was defined as 0.01 Sv. The values assigned to these variables in NUREG-1150 were 0.5 Sv and 0.25 Sv, respectively.

The remaining emergency response input parameter values implemented in Young's analysis are the same as those applied in the NUREG-1150 Peach Bottom analysis. Ninety nine and one-half percent of the population is assumed to evacuate within 10 miles of the NPP. The evacuating population is assumed to disappear at 20 miles from the NPP. The delay time between the notification of off-site emergency response officials to initiate protective actions (input parameter OALARM) and the beginning of evacuation is assumed to be 1.5 hrs. The population is assumed to evacuate at a speed of 4.8 meters per second. It is assumed that the 0.5% of the population not evacuating was relocated based on 0.01 Sv dose criterion for relocation.

Discounting

The MACCS code economic model is not designed to discount doses incurred in the years following the accident release. Consequently, it was not possible to include discounting in the calculations performed for Young's analysis without completing major modifications to the MACCS code.

"Long-term" doses incurred over the period of time following the first year after the accident were tabulated to assess the portion of the total population dose which could be significantly impacted by the discounting of accident costs. The integration period for the calculation of the population-dose resulting from groundshine and resuspension during the long-term phase is 1E+6 years. The level of contamination modeled in the long-term environment is dependent upon the half-life of the released radionuclides and the weathering terms input by the user. The population dose received from food ingestion is dependent upon the long-term transfer factor for each nuclide and crop of concern. The consequences calculated in Young's analysis are based on 1990 census and statistical data applied for the calculation of population dose and per person. The data indicate that the population dose incurred over the long term comprises between 50% and 93% of the total population dose for 94% of the source-term categories.

Appendix C

Supplemental Information for Non-Reactor Regulatory Analyses

Appendix C

Supplemental Information for Non-Reactor Regulatory Analyses

This appendix provides supplemental information for performing a regulatory analysis for non-reactor facilities, both fuel and non-fuel cycle. The procedure is essentially the same as that described in Chapters 2 through 5. However, the variety of facility types and the relatively non-integrated sets of available information lend difficulty to performing a value-impact analysis in the more straightforward manner as that for power reactors. This appendix represents a compilation of information to aid the preparation of a regulatory analysis applicable to non-reactor facilities. The nature of regulatory analyses for non-reactor facilities will continue to evolve as more analyses are performed and more information becomes available.

As discussed in Section 4.3, the analyst should strive to use quantitative attributes when performing a regulatory analysis for non-reactor licensees. The Commission has determined, for example, that PRA should be used for analyses involving materials licensees when the potential safety consequences warrant its use, sufficient data are available, and the licensees can reasonably be expected to be capable of performing such analyses (NRC 1996c). However, it should be recognized that there are many benefits of improved regulation of non-reactor facilities that do not lend themselves to quantification. For example, increased confidence in the margin of safety may be a nonquantifiable benefit of a particular proposed regulatory requirement. As noted in Section 4.5, nonquantifiable benefits and costs can be significant elements of a regulatory analysis and need to be considered by the analyst and decision maker as appropriate.

The approach taken in this appendix has been to first review the relevant literature in sufficient detail to permit the regulatory analyst to judge the value of each report (see Sections C.3-C.11). Tables and figures containing potentially useful data have been extracted from the reports and included in this appendix. Reviews of non-reactor regulatory analyses that have been performed comprise Sections C.8-C.11.

Based on the review of the literature, guidance on the performance of the value-impact analysis portion of a regulatory analysis has been developed. It is presented at the front of this appendix in the form of composite listings developed from the tables and figures to focus the relevant data for the analyst (see Sections C.1 and C.2). These should be used to direct the analyst's search for information that may be needed in the value-impact analysis. In some cases, the analyst may find values differing by several orders of magnitude, presumably the result of varying assumptions between the source documents. The analyst may wish to consult the references before selecting which value to use, especially since these tables are intended to direct analysts to appropriate sources, rather than to be used prima facie.

To assist the analyst, the tables and figures from which the data have been extracted to form these composites are referenced with the data. These composites are not intended to replace the original tables and figures, or the reports from which these tables and figures have been extracted. The analyst needing more detail should refer to the tables and figures, or the actual reports, directly. The analyst should also be aware that the composite listings combine data from multiple tables and figures, most of which were developed with differing sets of assumptions. Thus, the analyst may wish to use a specific table or figure, rather than a composite listing, when performing the analysis.

Two relatively recent data sources not cited in the Appendix C tables are also potentially available to the analyst. The first data source is the Nuclear Material Events Database (NMED) administered by the NRC Office for Analysis and

Evaluation of Operational Data (AEOD). The NMED contains information from materials, fuel cycle, and nonpower reactor licensees on events such as personnel radiation overexposures, medical misadministrations, losses of radioactive material, and potential criticality events. These data sources can be used to supplement and, when appropriate, supersede the information in the Appendix C tables. The second is the Bulletin 91-01 Event Tracking System administered by NMSS. NRC's Bulletin 91-01 requested reports from fuel cycle licensees relating to 1) loss or substantial degradation of a criticality safety control, and 2) conditions with a possible criticality hazard which have not been analyzed.

The analyst should also be aware of Attachment 3 to the CRGR Charter which provides guidance on the application of the "substantial increase" standard at 10 CFR 50.109(a)(3). Footnote 13 in Revision 6 of the CRGR Charter states that although 10 CFR 50.109 does not directly apply to facilities not licensed under Part 50, "much of the guidance in Attachment 3 is applicable and should be considered by the staff in evaluating qualitative factors that may contribute to the justification of proposed backfitting actions directed to nuclear materials facilities/activities."

C.1 Facility Classes

Review of the literature discussed in Sections C.3-C.11 suggested that non-reactor facilities would most appropriately be divided into two groups: fuel-cycle facilities and non-fuel cycle facilities. This grouping is defined in this section and employed throughout the presentation on attribute quantification in Section C.2.

C.1.1 Fuel Cycle Facilities

A division of fuel cycle facilities was made by Pelto et al. in the unpublished PNNL study from 1983 reviewed in Section C.6. The facilities were classified into the following 13 groups:

1. mining
2. milling
3. conversion
4. enrichment
5. fuel fabrication
6. MOX (mixed oxide) fuel refabrication
7. fuel reprocessing
8. spent fuel storage
9. HLW (high level waste) storage
10. TRU (transuranic) waste storage
11. geologic waste disposal
12. shallow land waste disposal
13. transportation.

Table C.S.1, extracted from Schneider et al. (1982), provides a summary description of each of these 13 groups. It is accompanied by Figure C.1, also extracted from Schneider et al. (1982), which shows the uranium process flow and relationship among the 13 groups.

Potential accidents during uranium mining do not yield much higher releases than incurred during normal operation. Philbin et al. (1990) (see Section C.4), Pelto et al. (see Section C.6), McGuire (1988) (see Section C.8), and the EPA (1983) (see Section C.9) addressed uranium mills. The following tables present data related to uranium milling: C.4,

C.48, C.70, C.77, and C.87-C.92. Figure C.4 also provides information on uranium milling. UF_6 conversion was examined by Philbin et al. (1990), Pelto et al., and McGuire (1988). Tables C.5, C.49, and C.70 present data related to UF_6 conversion.

Enrichment facilities have been addressed by Pelto et al. and McGuire (1988). Tables C.50 and C.70 provide data. Fuel fabrication has been examined by Philbin et al. (1990), Pelto et al., Mishima et al. (1983) (see Section C.7), McGuire (1988), and Ayer et al. (1988) (see Section C.11). Relevant data are presented in the following tables: C.6, C.51, C.70-C.76, C.78-C.79, and C.103-C.104. Pelto et al. and Ayer et al. (1988) have addressed MOX fuel refabrication. Seven tables, C.52-C.55, C.70, and C.103-C.104, contain MOX information. Fuel reprocessing was examined by Pelto et al., McGuire (1988), and Ayer et al. (1988). Tables C.56-C.60, C.70, C.80, C.103, and C.105 provide relevant data. Spent fuel storage was examined by Daling et al. (1990) (see Section C.5, Pelto et al., McGuire (1988), Jo et al. (1989) (see Section C.10), and Ayer et al. (1988). Data are provided in the following tables: C.26-C.32, C.44-C.45, C.61, C.70, C.81, C.93-C.103, and C.107.

Philbin et al. (1990), Pelto et al., and Ayer et al. (1988) addressed HLW storage. The following tables contain relevant information: C.62, C.70, C.103, and C.106. No literature on TRU storage was reviewed. Daling et al. (1990) and Pelto et al. examined geologic waste disposal. Data are presented in the following tables: C.9-C.25, C.42-C.45, C.63, and C.70. Figure C.3 also provides data for geologic waste disposal. No literature on shallow land waste disposal was reviewed. Daling et al. (1990) and Pelto et al. addressed transportation. Tables C.33-C.45 and C.64-C.70 contain relevant information.

C.1.2 Non-Fuel Cycle Facilities

A division of non-fuel cycle facilities is in NUREG/CR-4825 (Ostmeyer and Skinner 1987) (see Section C.3). The facilities were classified into the following four groups based on the application/use of the licensed nuclear material:

- research, teaching, experimental, diagnostic, and therapeutic facilities, including hospitals, universities, medical groups, and physicians
- measurement, calibration, and irradiation facilities, including users of sealed sources
- manufacturing and distribution facilities employing byproduct and source materials, such as radiopharmaceuticals
- service organizations, including waste repackagers, processors, and disposers.

Ostmeyer and Skinner (1987) (see Section C.3) examined all four groups. Relevant data are provided in Tables C.1-C.3 and Figure C.2. Philbin et al. (1990) addressed large manufacturers/distributors of nuclear byproducts (Group 3) and waste warehouses (Group 4). Tables C.7 and C.8 present information. McGuire (1988) examined Groups 1, 3, and 4. Relevant data are provided in Tables C.82-C.84 (Group 1), C.85 (Group 3), and C.86 (Group 4).

C.2 Quantification of Attributes

The procedure to quantify the attributes appropriate to the value-impact analysis portion of a regulatory analysis for non-reactor facilities is discussed in Section 5.7. Based on the information from the literature survey (see Sections C.3-C.11), specific quantitative data are presented in this section for use with the following six attributes when performing the value-impact analysis portion of a non-reactor regulatory analysis:

1. public health (accident)
2. public health (routine)
3. occupational health (accident)
4. occupational health (routine)
5. offsite property
6. onsite property.

Note that the last two attributes are discussed together rather than separately due to the nature of the available information.

C.2.1 Public Health (Accident)

The quantification of public health (accident) involves both frequencies and population doses associated with accident scenarios. Because non-reactor facilities tend to be much simpler in system configuration than power reactors, the number of potential accidents is much smaller, simplifying the scope of the accident analysis. However, accident frequency and population dose data are typically less available than for power reactors. This section extracts relevant frequency and dose data from Sections C.3-C.10. Also included are estimates of the total risk from accidents, as available.

C.2.1.1 Accident Frequencies

The literature review yielded accident frequencies for both fuel and non-fuel cycle, non-reactor facilities. Composite listings have been assembled in this section.

Fuel Cycle Facilities

Accident frequencies have been estimated for ten of the 13 non-reactor fuel cycle facilities listed in Section C.1. Only mining, TRU waste storage, and shallow land waste disposal have been excluded (see Section C.1.1).

For **URANIUM MILLING**, estimated frequencies for eight accident scenarios are in Table C.4, both as best estimates and 80% confidence bounds. Three of these scenario frequencies are also estimated in Table C.48, as follows:

1. solvent extraction fire = 4E-4 to 0.003/facility-yr
2. retention pond failure with slurry release = 0.04/facility-yr
3. slurry release from distribution pipe = 0.01/facility-yr.

Except for the second, these estimates lie at least partially within the uncertainty ranges listed in Table C.4. For the retention pond failure with slurry release, the estimate of 0.04/facility-yr slightly exceeds the upper bound in Table C.4.

For **UF_6 CONVERSION**, estimated frequencies for nine accident scenarios are in Table C.5, both as best estimates and 80% confidence bounds. Six of these scenario frequencies are also estimated in Table C.49, as follows:

1. uranyl nitrate evaporator explosion = 1E-4 to 0.001/facility-yr

2. hydrogen explosion during reduction = 0.001 to 0.05/facility-yr

3. solvent extraction fire = 4E-4/facility-yr

4. release from UF_6 cylinder = 0.03/facility-yr

5. distillation valve rupture = 0.05/facility-yr

6. waste pond release = 0.02/facility-yr.

Except for the last, these estimates lie within the uncertainty ranges listed in Table C.5. For the waste pond release, the estimate of 0.02/facility-yr is slightly below the lower bound in Table C.5.

For **ENRICHMENT**, estimated frequencies for four accident scenarios are in Table C.50. For **FUEL FABRICATION**, estimated frequencies for ten accident scenarios are in Tables C.6 and C.51. Table C.6 lists them as both best estimates and 80% confidence bounds. The estimates for the ten scenarios are as follows [parentheses () denote confidence bounds from Table C.6]:

1. minor facility release = 0.21/facility-yr (0.15 to 0.32) from Table C.6

2. large spills due to accidents or natural phenomena = 0.024/facility-yr (0.015 to 0.044) from Table C.6

3. transportation accident = 0.0028/facility-yr (0.0026 to 0.0030) from Table C.6

4. hydrogen explosion in reduction furnace = 0.01/facility-yr (0.002 to 0.05) from Table C.6 and 0.002 to 0.05/facility-yr from Table C.51

5. major fire = 2.1E-4/facility-yr (1.2E-4 to 5.1E-4) from Table C.6 and 2E-4/facility-yr from Table C.51

6. criticality = 0.0033/facility-yr (5.0E-4 to 0.011) from Table C.6 and 8E-4/facility-yr from Table C.51

7. release from hot UF_6 cylinder = 0.021/facility-yr (0.011 to 0.081) from Table C.6 and 0.03/facility-yr from Table C.51

8. fire in a roughing filter = 0.01/facility-yr from Table C.51

9. failure of valves and piping = 0.004/facility-yr from Table C.51

10. waste retention pond failure = 0.002 to 0.02/facility-yr from Table C.51.

For **MOX FUEL REFABRICATION**, estimated frequencies for 14 accident scenarios are in Tables C.53-C.55. The estimates for these scenarios are listed below. Note that the values listed from Table C.53 are those associated with normal high efficiency particulate air (HEPA) filtration. The corresponding estimates with HEPA filter failure are 1,000 times lower:

1. > design basis earthquake = 5E-6/facility-yr (Table C.54)

2. aircraft crash = 3E-7/facility-yr (C.54) and 1.5E-9/facility-yr (C.55)

3. hydrogen explosion in ROR (reduction-oxidation reactor) = 0.002 to 0.05/facility-yr (C.53), 0.001/facility-yr (C.54), and 0.005/facility-yr (C.55)

4. hydrogen explosion in sintering furnace = 0.001/facility-yr (C.54) and 0.005/facility-yr (C.55)

5. hydrogen explosion in wet scrap = 0.01/facility-yr (C.53), 0.005/facility-yr (C.54), and 3E-4/facility-yr (C.55)

6. ion-exchange resin fire = 1E-4 to 0.1/facility-yr (C.53) and 5E-4/facility-yr (C.54)

7. loaded final filter failure = 2E-4/facility-yr (C.54)

8. criticality = 3E-5 to 0.008/facility-yr (C.53), 6E-5/facility-yr (C.54), and 6E-5/facility-yr (C.55)

9. powder shipping container spill = 3E-5/facility-yr (C.55)

10. exothermic reactions in powder storage = 1.5E-6/facility-yr (C.55)

11. major facility fire = 2E-4/facility-yr (C.53)

12. fire in waste compaction glove box = 0.01/facility-yr (C.53)

13. glove failure = 1/facility-yr (C.53)

14. severe glove box damage = 0.01/facility-yr (C.53).

For **FUEL REPROCESSING**, estimated frequencies for 20 accident scenarios are in Tables C.57-C.60. The estimates for these scenarios are listed below. Note that values from Table C.57 are those associated with normal HEPA filtration. The corresponding estimates with HEPA filter failure are generally 1,000 times lower, except where noted. Also note that values from Table C.59 assume HEPA filter failure, except where noted.

1. loss of fuel storage pool water = 3E-6/facility-yr (Table C.58)

2. ion-exchange resin fire and explosion = 1E-4 to 0.1/facility-yr (C.57, with frequencies 1E+5 times lower with HEPA filter failure) and 5E-4/facility-yr (C.58)

3. criticality = 3E-5 to 0.008/facility-yr (C.57), 6E-5/facility-yr (C.58), and 2E-5/facility-yr (C.59, without HEPA filter consideration)

4. hydrogen explosion in high aqueous feed (HAF) tank = 1E-5/facility-yr (C.57, with frequency 100 times lower with HEPA failure), 7E-5/facility-yr (C.58), 3E-6/facility-yr (C.59), and 1E-5/facility-yr (C.60)

5. fire in low level waste = 0.01/facility-yr (C.58)

6. fuel assembly drop = 0.01 to 0.1/facility-yr (C.57), 0.002/facility-yr (C.58), 0.0012/facility-yr (C.59, without HEPA consideration), and 0.01/facility-yr (C.60)

7. explosion in HLW calciner = 1E-6/facility-yr (C.57), 5E-10/facility-yr (C.58, assuming HEPA filter failure), 2E-7/facility-yr (C.59), and 1E-6/ facility-yr (C.60)

8. krypton cylinder rupture = 1E-4/facility-yr (C.58) and 1.3E-4/facility-yr (C.59, without HEPA consideration)

9. explosion in high activity waste (HAW) concentrator = 1E-5/facility-yr (C.57), 4E-8/facility-yr (C.59), and 1E-5/ facility-yr (C.60)

10. solvent fire in codecontamination cycle = 1E-6 to 1E-4/facility-yr (C.57) and 1E-6/facility-yr (C.60)

11. explosion in low activity waste (LAW) concentrator = 1E-4/facility-yr (C.57) and 1E-4/facility-yr (C.60)

12. explosion in iodine absorber = 2E-4/facility-yr (C.57, without HEPA consideration)

13. solvent fire in plutonium extraction cycle = 1E-6 to 1E-4/facility-yr (C.57, with frequencies 1E+5 times lower with HEPA failure)

14. dissolver seal failure = 1E-5/facility-yr (C.57)

15. release from hot UF_6 cylinder = 0.05/facility-yr (C.57, without HEPA consideration)

16. solvent fire in hydrogen concentrator = 2E-6/facility-yr (C.59)

17. red oil explosion in fuel product concentrator = 4E-8/facility-yr (C.59)

18. explosion in fuel product denitrator = 4E-9/facility-yr (C.59)

19. hydrogen explosion in uranium reduction = 9E-6/facility-yr (C.59)

20. hydrogen explosion in fuel product denitrator fuel tank = 3E-6/facility-yr (C.59).

For **SPENT FUEL STORAGE**, estimated frequencies for 17 accident scenarios are in Tables C.31, C.32, C.61, C.93, C.97, and C.99. Data from Tables C.31, C.32, and C.61 have been combined into 14 accident scenarios whose frequencies are listed below. Note that the values taken from Table C.31 correspond to the drywell storage concept only. Tables C.93 and C.97 present frequencies for two additional scenarios—spent fuel pool fires due to seismic and cask drop initiators. Table C.99 addresses one more scenario, deriving failure frequencies for four different configurations of a spent fuel pool cooling and makeup system:

1. collision during highway transport = 2E-4/facility-yr (Table C.32, without fire, cask storage concept), 2E-5/facility-yr (C.32, without fire, drywell storage concept), 2E-6/facility-yr (C.32, with fire, cask storage), and 2E-7/facility-yr (C.32, with fire, drywell storage)

2. tornado = 6E-6/facility-yr (C.32, cask storage) and 1E-4/facility-yr (C.32, drywell storage)

3. fuel assembly drop = 0.1/facility-yr (C.32), 9E-4/facility-yr (C.61, for PWRs), and 0.006/facility-yr (C.61, for BWRs)

4. transport cask drop = 0.004/facility-yr (C.32, cask storage), 0.07/facility-yr (C.32, drywell storage), 1E-4/facility-yr (C.61, PWRs), and 2.5E-4/facility-yr (C.61, BWRs)

5. cask venting during transport = 0.002/facility-yr (C.32, cask storage) and 0.03/facility-yr (C.32, drywell storage)

6. canister drop during emplacement = 1.7E-8/facility-yr (C.31) and 1E-6/facility-yr (C.32, drywell storage)

7. canister shear during emplacement = 2E-6/facility-yr (C.32, drywell storage)

8. cask drop during emplacement = 1E-5/facility-yr (C.32, cask storage)

9. airplane crash = 4.0E-10/facility-yr (C.31, without fire), 7.4E-9/facility-yr (C.31, with fire), 6E-9/facility-yr (C.32, with fire, cask toppled, cask storage), 9E-9/facility-yr (C.32, with fire, cask storage), 2E-7/facility-yr (C.32, one fuel assembly, with fire, drywell storage), and 2E-8/facility-yr (C.32, 10 assemblies, with fire, drywell storage)

10. earthquake = 4.8E-9/facility-yr (C.31, without fuel pin failure), 4.3E-8/facility-yr (C.31, with pin failure), 4E-6/facility-yr (C.32, 24 assemblies, cask storage), 4E-8/facility-yr (C.32, 2,400 assemblies, cask storage), 8E-6/facility-yr (C.32, one assembly, drywell storage), 8E-7/facility-yr (C.32, 10 assemblies, drywell storage), and 2E-8/facility-yr (C.32, 2,400 assemblies, drywell storage)

11. transporter collision during emplacement = 1.7E-8/facility-yr (C.31, without fire) and 6.1E-7/facility-yr (C.31, with fire)

12. transporter collision during retrieval = 0.0089/facility-yr (C.31, without pin failure or fire), 0.028/facility-yr (C.31, with pin failure, without fire), 1.4E-4/facility-yr (C.31, without pin failure, with fire), and 1.4E-4/facility-yr (C.31, with pin failure and fire)

13. transporter motion with canister partially in place = 0.086/facility-yr (C.31, during emplacement), 0.0089/facility-yr (C.31, during retrieval, without pin failure), and 0.14/facility-yr (C.31, during retrieval, with pin failure)

14. canister drop during retrieval = 0.11/facility-yr (C.31).

For **HLW STORAGE**, estimated frequencies for three accident scenarios are in Table C.62 (after grouping by pairs). For **GEOLOGIC WASTE DISPOSAL**, estimated frequencies for 18 accident scenarios are in Tables C.14, C.19, and C.20. Note that Table C.20 divides earthquake-induced accidents into nine categories, which are listed below as 18a-18i. The estimates for the 18 scenarios are as follows:

1. fuel truck crash into HLW area = 2.0E-6/facility-yr (Table C.14)

2. fuel truck crash into cladding waste area = 2.0E-6/facility-yr (C.14)

3. fuel truck crash into non-HLW (NHLW) area = 2.0E-6/facility-yr (C.14)

4. airplane crash = 1.0E-7/facility-yr (C.14) and <2.0E-10/facility-yr (C.19)

5. elevator drop = 4.0E-8/facility-yr (C.14)

6. fuel assembly drop = 0.1/facility-yr (C.19) and 1.E-8/facility-yr (C.20, drop into hot cell with HVAC failure)

7. NHLW pallet drop = 0.050/facility-yr (C.14)

8. final filter failure = 0.003/facility-yr (C.14)

9. shipping cask drop = 5E-6/facility-yr (C.20, with cask breach)

10. open consolidated fuel container drop = 1E-9/facility-yr (C.20, with HVAC failure)

11. container drop in storage vault = 3E-8/facility-yr (C.20, with failure to activate filtration system)

12. nuclear test = <0.001/facility-yr (C.19)

13. loading dock fire = <1.0E-7/facility-yr (C.19, spent fuel) and <1.0E-7/facility-yr (C.19, HLW)

14. waste handling ramp fire = <1.0E-7/facility-yr (C.19)

15. emplacement drift fire = <1.0E-7/facility-yr (C.19)

16. flood = 0.01/facility-yr (C.19)

17. tornado = <9.1E-11/facility-yr (C.19)

18. earthquake = <0.0013/facility-yr (C.19)
    18a. crane fails, falling on or dropping cask in receiving area = 5E-8/facility-yr (C.20)
    18b. train falls on cask = 5E-8/facility-yr (C.20)
    18c. structural object falls on fuel in cask unloading cell = 5E-7/facility-yr (C.20)
    18d. crane fails, falling on or dropping fuel in cask unloading cell = 1E-6/facility-yr (C.20)
    18e. structural object falls on fuel in consolidation cell = 5E-7/facility-yr (C.20)
    18f. crane fails, falling on or dropping fuel in consolidation cell = 1E-6/facility-yr (C.20)
    18g. structural object falls on fuel in packaging cell = 5E-7/facility-yr (C.20)
    18h. crane fails, falling on or dropping fuel in packaging cell = 1E-6/facility-yr (C.20, with HVAC failure)
    18i. structural object falls on fuel in transfer tunnel = 5E-7/facility-yr (C.20).

For **TRANSPORTATION**, it is convenient to identify three categories based on the material being shipped: spent fuel, plutonium oxide, and HLW. For spent fuel transportation, estimated frequencies for eight accident scenarios are in Tables C.65-C.69. The estimates for the scenarios are as follows:

1. leakage of coolant from spent fuel cask during rail shipment = 3E-4/shipment (Table C.65), 6.4E-6/shipment (C.69, impact fails cask seal, fuel failure), 1.2E-6/shipment (C.69, side impact fails pressure relief valve, fuel failure), 6.4E-6/shipment (C.69, end impact fails pressure relief valve, fuel failure), and 1.2E-6/shipment (C.69, side impact fails cask seal, fuel failure)

2. release from a collision during rail shipment = 2E-8 to 9E-6/shipment (C.65), 9E-6/shipment (C.67), and 1E-4/yr (C.68, with closure errors)

3. release from a collision followed by release of fuel from the cask during rail shipment = 2E-10 to 9E-8/shipment (C.65), 2E-5/yr C.68, for 50-80 km/hr collision), 3E-4/yr (C.68, 80-100 km/hr), 8E-5/yr (C.68, with 1000°C fire for >1 hr), and 2E-5/yr (C.68, 800°C for >2 hr)

4. loss of gases from inner cavity = 9E-6/shipment (C.66, rail shipment) and 2E-5/shipment (C.66, truck)

5. loss of confinement and 50% fuel damage = 4E-7/shipment (C.66, without fire, rail), 2E-9/ shipment (C.66, with fire, rail), 2E-7/shipment (C.66, without fire, truck), 2E-9/shipment (C.66, with fire, truck), 4E-7/shipment (C.67, without fire, rail), and 3E-9/shipment (C.67, with fire, rail)

6. loss of neutron shielding during rail shipment = 2E-5/shipment (C.67)

7. fall during rail shipment = 2E-6/yr (C.68, for 25-40 m fall) and 2E-5/yr (C.69, 9-25 m)

8. fire during rail shipment = 1E-4/yr (C.68, 1000°C for >1 hr) and 2E-5/yr (C.68, 800°C for >2 hr).

For plutonium oxide transportation, estimated frequencies for six accident scenarios are in Tables C.65 (three scenarios for rail shipment) and C.66 (three scenarios for truck shipment). For HLW transportation by rail, estimated frequencies for five accident scenarios are in Tables C.66 and C.67.

Non-Fuel Cycle Facilities

For **RESEARCH, TEACHING, EXPERIMENTAL, DIAGNOSTIC, AND THERAPEUTIC FACILITIES**, Table C.1 contains an estimated overall accident frequency of 2.3E-4/facility-yr. For **MEASUREMENT, CALIBRATION, AND IRRADIATION FACILITIES**, Table C.1 contains an estimated overall accident frequency of 1.8E-4/facility-yr. For **MANUFACTURING AND DISTRIBUTION FACILITIES EMPLOYING BYPRODUCT AND SOURCE MATERIALS**, estimated frequencies for eight accident scenarios are in Table C.7, both as best estimates and 80% confidence bounds. Table C.1 also contains an overall estimate of 0.0026/facility-yr, which is noticeably less than the sum of the eight accident frequencies from Table C.7. For **SERVICE ORGANIZATIONS** (waste warehouses), estimated frequencies for six accident scenarios are in Table C.8, both as best estimates and 80% confidence bounds.

McGuire (1988) estimated the frequency of a major radioactive release for a non-reactor facility to be 1E-4/yr, assumed applicable to either fuel- or non-fuel cycle facilities (see Section C.8).

C.2.1.2 Population Doses from Accidents

Unlike accident frequencies, literature review yielded population doses from accidents only for non-reactor fuel cycle facilities. However, safety analysis reports conducted for various DOE non-fuel cycle facilities (e.g., those at the Savannah River Site) contain population doses from accidents. If available, the analyst could use these for particular facilities.

Fuel Cycle Facilities

Estimated population doses from accidents for 10 of the 13 non-reactor fuel cycle facilities listed in Section C.1 are included in this section. Estimates for mining, TRU waste storage, and shallow land waste disposal are not included. For

URANIUM MILLING, estimated population doses from three accident scenarios are in Table C.48. For **UF_6 CONVERSION**, estimated population doses from six accident scenarios are in Table C.49. For **ENRICHMENT**, estimated population doses from four accident scenarios are in Table C.50. For **FUEL FABRICATION**, estimated population doses from seven accident scenarios are in Table C.51.

For **MOX FUEL REFABRICATION**, estimated population doses from 14 accident scenarios are in Tables C.53-C.55. The estimates for these scenarios are listed below. Note that the values listed from Table C.53 are those associated with normal HEPA filtration. The corresponding estimates with HEPA filter failure are generally 1E+5 times higher, except where noted.

1. > design basis earthquake = 1E+5 person-rem (Table C.54)

2. aircraft crash = 3E+4 person-rem (C.54) and 500 person-rem (C.55)

3. hydrogen explosion in Reduction-Oxidation Reactor (ROR) = 0.031 person-rem (C.53), 5E-9 person-rem (C.54), and 1.1E-11 person-rem (C.55)

4. hydrogen explosion in sintering furnace = 2E-7 person-rem (C.54) and 4E-10 person-rem (C.55)

5. hydrogen explosion in wet scrap = 0.16 person-rem (C.53), 2E-6 person-rem (C.54), and 1.1E-11 person-rem (C.55)

6. ion-exchange resin fire = 0.0092 person-rem (C.53) and 2E-9 person-rem (C.54)

7. loaded final filter failure = 0.3 person-rem (C.54)

8. criticality = 0.38 person-rem (C.53, with dose 1100 times higher with HEPA filter failure), 5 person-rem (C.54), and 2 person-rem (C.55)

9. powder shipping container spill = 1.1E-11 person-rem (C.55)

10. exothermic reactions in powder storage = 1E-10 person-rem (C.55)

11. major facility fire = 1.6 person-rem (C.53, with dose 9E+4 times higher with HEPA failure)

12. fire in waste compaction glove box = 0.0031 person-rem (C.53)

13. glove failure = 1.3E-5 person-rem (C.53)

14. severe glove box damage = 0.061 person-rem (C.53).

For **FUEL REPROCESSING**, estimated population doses from 20 accident scenarios are in Tables C.57-C.60. The estimates for these scenarios are listed below. Note that values from Table C.59 assume HEPA filter failure, except where noted.

1. loss of fuel storage pool water = 50 person-rem (Table C.58)

2. ion-exchange resin fire and explosion = 0.36 person-rem (C.57, with normal HEPA filtration), 1800 person-rem (C.57, with failed HEPA filtration), and 0.2 person-rem (C.58)

3. criticality = 0.030 person-rem (C.57, normal HEPA), 0.035 person-rem (C.57, failed HEPA), 5 person-rem (C.58), and 2 person-rem (C.59, without HEPA filter consideration)

4. hydrogen explosion in HAF tank = 1600 person-rem (C.57, normal HEPA), 1700 person-rem (C.57, failed HEPA), 0.07 person-rem (C.58), 9E-4 person-rem (C.59), and 490 person-rem (C.60)

5. fire in low level waste = 0.1 person-rem (C.58)

6. fuel assembly drop = 0.013 person-rem (C.57, normal HEPA), 1300 person-rem (C.57, failed HEPA), 0.1 person-rem (C.58), 0.05 person-rem (C.59, without HEPA consideration), and 0.0020 person-rem (C.60)

7. explosion in HLW calciner = 4300 person-rem (C.57, normal HEPA), 1.3E+4 person-rem (C.57, failed HEPA), 6E+6 person-rem (C.58, assuming HEPA filter failure), 0.2 person-rem (C.59), and 510 person-rem (C.60)

8. krypton cylinder rupture = 50 person-rem (C.58) and 40 person-rem (C.59, without HEPA consideration)

9. explosion in HAW concentrator = 430 person-rem (C.57, normal HEPA), 9500 person-rem (C.57, failed HEPA), 0.008 person-rem (C.59), and 57 person-rem (C.60)

10. solvent fire in codecontamination cycle = 23 person-rem (C.57, normal HEPA), 56 person-rem (C.57, failed HEPA), and 2.6 person-rem (C.60)

11. explosion in LAW concentrator = 28 person-rem (C.57, normal HEPA), 48 person-rem (C.57, failed HEPA), and 3.2 person-rem (C.60)

12. explosion in iodine absorber = 4.8 person-rem (C.57, without HEPA consideration)

13. solvent fire in plutonium extraction cycle = 3.1E-4 person-rem (C.57, normal HEPA) and 520 person-rem (C.57, failed HEPA)

14. dissolver seal failure = 0.023 person-rem (C.57, normal HEPA) and 2300 person-rem (C.57, failed HEPA)

15. release from hot UF_6 cylinder = 1.5 person-rem (C.57, without HEPA consideration)

16. solvent fire in hydrogen concentrator = 7E-4 person-rem (C.59)

17. red oil explosion in fuel product concentrator = 6E-4 person-rem (C.59)

18. explosion in fuel product denitrator = 0.012 person-rem (C.59)

19. hydrogen explosion in uranium reduction = 1.4E-4 person-rem (C.59)

20. hydrogen explosion in fuel product denitrator fuel tank = 0.012 person-rem (C.59).

For **SPENT FUEL STORAGE**, estimated population doses from 18 accident scenarios are in Tables C.27, C.31, C.32, C.61, C.94, and C.101. Those from Tables C.27, C.31, C.32, and C.61 have been combined into 14 accident scenarios whose population doses are listed below. Note that the values taken from Table C.27 are those for total body population dose. The values taken from Table C.31 correspond to the drywell storage concept only. Also note that Tables C.31 and

C.32 are quantified in terms of latent cancer fatalities (LCFs) rather than person-rems. These can be transformed into person-rems via a typical conversion factor such as 200 health effects (or LCFs) per 1E+6 person-rems, or inversely 5,000 person-rem/health effect.[1] Table C.94 presents population doses for two additional scenarios—spent fuel pool fires due to seismic and cask drop initiators, whose estimated frequencies are in Table C.93—in terms of an "average" and "worst" case. Table C.101 addresses two more scenarios, another "average" and "worst" case, deriving population doses for four pairings of the accident scenarios and selected mitigative options.

1. collision during highway transport = 0.1 LCF (Table C.32, without fire, cask storage concept), 0.004 LCF (C.32, without fire, drywell storage concept), 0.5 LCF (C.32, with fire, cask storage), and 0.02 LCF (C.32, with fire, drywell storage)

2. tornado = 0.04 LCF (C.32, cask storage) and 0.04 LCF (C.32, drywell storage)

3. fuel assembly drop = 0.03 person-rem (C.27), 4E-5 LCF (C.32), 0.7 person-rem (C.61, for PWRs), and 0.3 person-rem (C.61, for BWRs)

4. transport cask drop = 0.006 person-rem (C.27), 4E-4 LCF (C.32, cask storage), 4E-4 LCF (C.32, drywell storage), 2 person-rem (C.61, PWRs), and 1.8 person-rem (C.61, BWRs)

5. cask venting during transport = 0.1 LCF (C.32, cask storage) and 0.004 LCF (C.32, drywell storage)

6. canister drop during emplacement = 3.9E-6 LCF (C.31) and 0.004 LCF (C.32, drywell storage)

7. canister shear during emplacement = 0.004 LCF (C.32, drywell storage)

8. cask drop during emplacement = 0.006 person-rem (C.27) and 0.004 LCF (C.32, cask storage)

9. airplane crash = 0.26 LCF (C.31, without fire), 1.3 LCF (C.31, with fire), 0.5 LCF (C.32, with fire, cask toppled, cask storage), 0.5 LCF (C.32, with fire, cask storage), 0.02 LCF (C.32, one fuel assembly, with fire, drywell storage), and 0.2 LCF (C.32, 10 assemblies, with fire, drywell storage)

10. earthquake = 0.061 LCF (C.31, without fuel pin failure), 3.3 LCF (C.31, with pin failure), 0.1 LCF (C.32, 24 assemblies, cask storage), 10 LCF (C.32, 2400 assemblies, cask storage), 0.004 LCF (C.32, one assembly, drywell storage), 0.04 LCF (C.32, 10 assemblies, drywell storage), and 2.4 LCF (C.32, 2400 assemblies, drywell storage)

11. transporter collision during emplacement = 3.4E-5 LCF (C.31, without fire) and 0.0019 LCF (C.31, with fire)

12. transporter collision during retrieval = 5.9E-7 LCF (C.31, without pin failure or fire), 3.8E-5 LCF (C.31, with pin failure, without fire), 2.6E-6 LCF (C.31, without pin failure, with fire), and 2.6E-4 LCF (C.31, with pin failure and fire)

13. transporter motion with canister partially in place = 0.018 LCF (C.31, during emplacement), 5.9E-7 LCF (C.31, during retrieval, without pin failure), and 0.0016 LCF (C.31, during retrieval, with pin failure)

14. canister drop during retrieval = 9.9E-7 LCF (C.31).

For **HLW STORAGE**, estimated population doses from three accident scenarios (after grouping by pairs) are in Table C.62. For **GEOLOGIC WASTE DISPOSAL**, estimated population doses from 19 accident scenarios are in

Tables C.14, C.15, C.18, and C.19. Note that Table C.15 reports population doses as person-mrems. These are listed as person-rems below. Also note that Tables C.18 and C.19 generally provide the same values (and are referenced as coming from Table C.19), except where noted.

1. fuel truck crash into HLW area = 2000 person-rem (Table C.14)
2. fuel truck crash into cladding waste area = 2.0 person-rem (C.14)
3. fuel truck crash into NHLW area = 40 person-rem (C.14)
4. airplane crash = 4000 person-rem (C.14) and 110 person-rem (C.19)
5. elevator drop = 0.050 person-rem (C.14)
6. fuel assembly drop = 2.99 person-rem (C.15) and 8.0E-5 person-rem (C.19)
7. NHLW pallet drop = 0.80 person-rem (C.14)
8. final filter failure = 2.0 person-rem (C.14)
9. HLW drop = 0.175 person-rem (C.15)
10. spent fuel handling = 1.29 person-rem (C.15)
11. remote TRU drop = 1.98E-4 person-rem (C.15)
12. contract TRU puncture = 6.70E-8 person-rem (C.15)
13. nuclear test = 0.0031 person-rem (C.19)
14. loading dock fire = 0.0068 person-rem (C.19, spent fuel) and 9.2E-4 person-rem (C.19, HLW)
15. waste handling ramp fire = 3.6E-7 person-rem (C.18) and 4.8E-7 person-rem (C.19)
16. emplacement drift fire = 3.6E-7 person-rem (C.18) and 4.8E-7 person-rem (C.19)
17. flood = 1.2E-9 person-rem (C.19)
18. tornado = 0.0031 person-rem (C.19)
19. earthquake = 0.0031 person-rem (C.19).

For **TRANSPORTATION**, it is convenient to identify three categories based on the material being shipped: spent fuel, plutonium oxide, and HLW. For spent fuel transportation, estimated population doses from eight accident scenarios are in Tables C.37, C.38, and C.65-C.69. The estimates for these scenarios are listed below. Note that the values reported from Table C.37 are the totals from inhalation, plume gamma, and ground gamma pathways. The values listed below correspond to those for the urban area given in Table C.37. The corresponding values for the rural area in Table C.37 are 640 times lower. Also note that Table C.38 reports population doses from the water ingestion pathway.

1. leakage of coolant from spent fuel cask during rail shipment = 5.8E-4 person-rem (Table C.65), 680 person-rem (C.69, impact fails cask seal, fuel failure), 1900 person-rem (C.69, side impact fails pressure relief valve, fuel failure), 1900 person-rem (C.69, end impact fails pressure relief valve, fuel failure), and 680 person-rem (C.69, side impact fails cask seal, fuel failure)

2. release from a collision during rail shipment = 939 person-rem (C.37), 182 person-rem (C.38), 1.9E+4 person-rem (C.65), 1.7E-6 person-rem (C.67), and 1.1 person-rem (C.68, with closure errors)

3. release from a collision followed by release of fuel from the cask during rail shipment = 1.35E+4 person-rem (C.37, with fire), 1.12E+5 person-rem (C.37, with fire and fuel oxidation), 6870 person-rem (C.38, fire), 6.3E+4 person-rem (C.38, fire and oxidation), 2.7E+4 person-rem (C.65), 0.28 person-rem (C.68, for 50-80 km/hr collision), 0.28 person-rem (C.68, 80-100 km/hr), 0.20 person-rem (C.68, with 1000°C fire for > 1 hr), and 0.20 person-rem (C.68, 800°C for > 2 hr)

4. loss of gases from inner cavity = 1E-6 person-rem (C.66, rail shipment) and 5E-9 person-rem (C.66, truck)

5. loss of confinement and 50% fuel damage = 0.1 person-rem (C.66, without fire, rail), 2000 person-rem (C.66, with fire, rail), 100 person-rem (C.66, without fire, truck), 600 person-rem (C.66, with fire, truck), 0.5 person-rem (C.67, without fire, rail), and 1700 person-rem (C.67, with fire, rail)

6. loss of neutron shielding during rail shipment = 8E-7 person-rem (C.67)

7. fall during rail shipment = 0.28 person-rem (C.68, for 25 to 40 m fall) and 0.28 person-rem (C.69, 9-25 m)

8. fire during rail shipment = 0.20 person-rem (C.68, 1000°C for > 1 hr) and 0.20 person-rem (C.68, 800°C for > 2 hr).

For plutonium oxide transportation, estimated population doses from six accident scenarios are in Tables C.65 (three scenarios for rail shipment) and C.66 (three scenarios for truck shipment). For HLW transportation by rail, estimated population doses from five accident scenarios are in Tables C.66 and C.67.

McGuire (1988) estimated the population doses from a major radioactive release for a non-reactor facility to be 40 and 800 person-rem for an effective dose equivalent (EDE) of 5 rems at distances of 100 and 1,000 m, respectively. These can be assumed applicable to either fuel- or non-fuel cycle facilities (see Section C.8).

C.2.1.3 Total Accident Risks

Total public risks from all accident scenarios have been estimated for 10 of the 13 non-reactor fuel cycle facilities listed in Section C.1. Many of these estimated risks are in Table C.70 after scaling on a consistent basis for comparison (see Section C.6). Tables C.14, C.19, C.31, C.32, C.35, C.42, and C.44 contain additional estimates. The estimates in these eight tables have been assembled into the following table, modeled after Table C.70. The risks from Tables C.14, C.19, C.31, C.32, C.35, C.42, and C.44 are listed as "unscaled" values, after converting units of health effects or fatalities into person-rems via a conversion factor of 5,000 person-rem/health effect.[2] The "normalized" risks from Table C.70 are listed as "scaled" values in Table C.109.

Estimated public risks from three accident scenarios during the postclosure period of **GEOLOGIC WASTE DISPOSAL** in terms of 10,000-yr health effects for four geologic media are in Table C.23. These can be summed to yield the following total public risks:

- basalt = 28.43 health effects
- bedded salt = 6.57 health effects
- tuff = 3.44 health effects
- granite = 9.85 health effects.

These can be converted into person-rems as mentioned above.

C.2.2 Public Health (Routine)

There is considerably less literature on routine public health risks than on accidental risks for non-reactor applications.

For **SPENT FUEL STORAGE**, estimated routine public risks during the operations and decommissioning phases at a monitored retrievable storage (MRS) facility are in Table C.44 in terms of latent health effects (LHEs) per year. These can be transformed into person-rem/yr via a typical conversion factor such as 5,000 person-rem/health effect.[2] Table C.26 also provides the routine public risk during operations at an MRS facility, 20 person-rem/yr (total body).

For **GEOLOGIC WASTE DISPOSAL**, estimated routine public risks during the construction, operations, and decommissioning phases of the preclosure period at a repository are in Tables C.9, C.10, C.13, C.42, and C.44. Note that the values in Tables C.9 and C.10 are given in terms of the 70- and 50-year dose commitments, respectively. The value from Table C.13 is taken for the "reference" case. Also note that the values in Tables C.42 and C.44 are given in terms of LHE/yr, which can be converted into person-rem/yr as discussed above. Tables C.42 and C.44 address the waste management system without and with an MRS facility, respectively. The routine public risks have been estimated as follows:

1. construction = 0.0068 person-rem (Table C.9, salt medium), 100 person-rem (C.9, granite), 15 person-rem (C.9, basalt), 38 person-rem (C.9, shale), 2.0E+4 person-rem (C.10), 1E-5 LHE/yr (C.42), and 1E-5 LHE/yr (C.44)

2. operations = 3.9E+5 person-rem (C.10), 1.5E-5 person-rem/yr (C.13), 9E-4 LHE/yr (C.42), and 8E-7 LHE/yr (C.44)

3. decommissioning = 2E-11 LHE/yr (C.42) and 2E-11 LHE/yr (C.44).

For the postclosure period of geologic waste disposal, estimated routine public risks are in Tables C.23 and C.24. Table C.23 provides the 10,000-yr health effects for an undisturbed repository in four geologic media. Table C.24 provides 27,000- and 250,000-yr population doses to four body organs resulting from ingestion of drinking water.

For **TRANSPORTATION**, estimated routine public risks are in Tables C.35, C.40-C.42, and C.44. The values in Table C.35 apply exclusively to spent fuel shipment. Tables C.40 and C.41 present values for both spent fuel and HLW shipment by truck and rail to three repository locations for the waste management system without and with an MRS facility, respectively. The risks are given in health effects, which can be converted into person-rems as previously discussed. The values in Tables C.42 and C.44 apply to both spent fuel and HLW shipment, assuming that 30% of the spent fuel is shipped by truck and 70% by rail, while all HLW is shipped by rail. Note that the values in Tables C.42 and C.44 are given in terms of LHE/yr. These can be transformed into person-rem/yr via a typical conversion factor such as 5,000 person-rem/health effect.[2] Tables C.42 and C.44 address the waste management system without and with an MRS facility, respectively. The routine public risks have been estimated as follows:

- spent fuel by truck = 93.80 person-rem/yr (C.35, in 1975) and 565.0 person-rem/yr (C.35, 1985)
- spent fuel by rail = 7.78 person-rem/yr (C.35, 1975) and 298.0 person-rem/yr (C.35, 1985)

- spent fuel and HLW combined = 0.09 LHE/yr (C.42) and 0.03 LHE/yr (C.44).

C.2.3 Occupational Health (Accident)

There is less literature available on occupational compared to public health risks due to accidents. Information is particularly scarce for non-fuel cycle facilities. Information for fuel cycle facilities is discussed below.

Estimated risks to the worker from accidents are shown below for four of the 13 non-reactor fuel cycle facilities listed in Section C.1: MOX fuel refabrication, fuel reprocessing, spent fuel storage, and geologic waste disposal (Fullwood and Jackson 1980).

MOX FUEL REFABRICATION = 7.0E-4 person-rem/GWe-yr

FUEL REPROCESSING = 1.0E-4 person-rem/GWe-yr.

For **SPENT FUEL STORAGE**, estimated occupational risks due to accidents during the operations and decommissioning phases at an MRS facility are in Table C.45. The values are in terms of LHE/yr, which can be transformed into person-rem/yr via a typical conversion factor such as 5,000 person-rem/health effect.[2]

For **GEOLOGIC WASTE DISPOSAL**, occupational risks due to accidents have been estimated for aggregates of scenarios during the operations, decommissioning, and retrieval phases in the preclosure period. The estimates are in Tables C.21 (decommissioning and retrieval), C.43 (operations, without an MRS facility), and C.45 (operations, with an MRS facility). The latter two tables provide values in terms of LHE/yr, which can be transformed into person-rem/yr as mentioned above. Table C.12 presents an occupational risk estimate for a shaft drop accident during the operations phase. The information in Tables C.18 and C.19 provide both frequencies and worker doses for individual accident scenarios during the operations phase of the preclosure period. These can be converted into occupational risk estimates in a manner similar to that employed in Table C.19 for public risk, as shown in Table C.110.

C.2.4 Occupational Health (Routine)

There is limited literature available on routine occupational health risks. Information for non-fuel cycle facilities is particularly scarce. Information for non-reactor fuel cycle facilities is discussed below.

Estimated risks to the worker from routine operations are included below for four of the 13 non-reactor fuel cycle facilities listed in Section C.1: fuel fabrication, spent fuel storage, geologic waste disposal, and transportation. For **FUEL FABRICATION**, estimated occupational doses for fabricating PuO_2 powder into unfired pellets and reconstituting the pellets back to powder are in Tables C.75 and C.76, respectively. Average values and ranges are provided.

For **SPENT FUEL STORAGE**, Table C.45 provides the total routine estimated occupational risks (in LHE/yr) for the operations and decommissioning phases at an MRS facility. These can be transformed into person-rem/yr via a typical conversion factor such as 5,000 person-rem/health effect.[2] Daling et al. (1990) provide estimates for the decommissioning phase at an MRS facility of 120 person-rem for drywell storage and 128 person-rem for cask storage (see Section C.5). Totals for the operations phase at an MRS facility are also provided in Tables C.28 and C.29, and can be calculated from Table C.30. Tables C.28-C.30 also list the routine occupational risks for separate activities during the operations phase. Note that Table C.28 gives these in terms of person-rem/1,000 metric tons of uranium (MTU); C.29 lists them in person-rem/yr; and C.30 lists them in person-mrem/1,000 MTU (converted to person-rem/1,000 MTU below). The composites for the seven activities from Tables C.28-C.30 are as follows:

1. receipt, inspection, and unloading = 58 person-rem/1,000 MTU (Table C.28), 148.0 person-rem/yr (C.29), 0.135 person-rem/1,000 MTU (C.30, from truck), and 0.025 person-rem/1,000 MTU (C.30, rail)

2. consolidation and packaging = 15 person-rem/1,000 MTU (C.28), 6.2 person-rem/yr (C.29), 0.0036 person-rem/1,000 MTU (C.30, for fuel), and 0.0011 person-rem/1,000 MTU (C.30, non-fuel)

3. emplacement in storage area = 20 person-rem/1,000 MTU (C.28, including retrieval from storage area) and 7.2 person-rem/yr (C.29)

4. maintenance/monitoring in storage area = 2 person-rem/1,000 MTU (C.28) and 5.3 person-rem/yr (C.29)

5. retrieval from storage area = 20 person-rem/1,000 MTU (C.28, including emplacement) and 7.1 person-rem/yr (C.29)

6. transfer to process cells = 4.0 person-rem/yr (C.29)

7. shipment to repository = 140.9 person-rem/yr (C.29).

For **GEOLOGIC WASTE DISPOSAL**, total estimated routine occupational risks for the construction, operations, decommissioning, and retrieval phases of the preclosure period are in Tables C.9, C.11, C.12, C.16, C.17, C.21, C.43, and C.45. The estimates from Table C.9 are in terms of the 70-yr dose commitment; Table C.11 reports fatalities over 5-yr construction and 26-yr operations phases; Tables C.43 and C.45 give values in terms of LHE/yr for the waste management system without and with an MRS facility, respectively. Both fatalities and LHE/yr can be transformed into person-rem/yr via a typical conversion factor such as 5,000 person-rem/health effect.[2] The values from Tables C.12 and C.21 are taken for the "reference" case. The routine occupational risks have been estimated as follows:

1. construction = 0.18 person-rem (Table C.9, salt medium), 5,000 person-rem (C.9, granite), 6,200 person-rem (C.9, basalt), 1,900 person-rem (C.9, shale), 0.014 fatality (C.11, salt), 0.77 fatality (C.11, tuff), 1.6 fatalities (C.11, basalt), 0.1 LHE/yr (C.43), and 0.1 LHE/yr (C.45)

2. operations = 1.5 fatalities (C.11, salt), 5.0 fatalities (C.11, tuff), 7.3 fatalities (C.11, basalt), 902 person-rem/yr (C.12), 0.02 LHE/yr (C.43), and 0.02 LHE/yr (C.45)

3. decommissioning = 6 person-rem/yr (C.21), 0.03 LHE/yr (C.43), and 0.03 LHE/yr (C.45)

4. retrieval = 163 person-rem/yr (C.21).

Table C.17 lists the routine occupational risks for separate activities during the operations phase at a tuff repository. Table C.16 does likewise for four of the activities listed in Table C.17. The estimates from Table C.16 are as follows:

1. receiving = 44.8 person-rem/yr

2. handling and packaging = 6.9 person-rem/yr

3. transfer to underground facilities = 6.0 person-rem/yr

4. emplacement in boreholes = 12.4 person-rem/yr for vertical emplacement and 8.7 person-rem/yr for horizontal.

These values agree well with the corresponding ones in Table C.17.

For **TRANSPORTATION**, Tables C.43 and C.45 contain estimated routine occupational risks for the waste management system without and with an MRS facility, respectively. These values are given in LHE/yr which can be converted into person-rem/yr as mentioned above.

C.2.5 Offsite and Onsite Property

The offsite and onsite property attributes are examined together in this section for non-reactor facilities because most of the estimates reported in the literature have grouped the associated costs together as cleanup costs. When such costs are multiplied by the accident frequencies, measures of economic risk from accidents are obtained. Several of the reviewed reports contain economic risk estimates from accidents.

C.2.5.1 Fuel Cycle Facilities

Information is included below on estimated cleanup costs and/or economic risks have been estimated for five of the 13 non-reactor fuel cycle facilities listed in Section C.1: uranium milling, UF_6 conversion, fuel fabrication, spent fuel storage, and transportation. Estimates for **URANIUM MILLING, UF_6 CONVERSION, and FUEL FABRICATION** are provided in Tables C.4-C.6, respectively. Each table provides a best estimate and 80% confidence bounds for the cleanup cost (in 1989 dollars) associated with each accident scenario at the reference facility. Each cost is multiplied by the corresponding estimate for the scenario frequency (also given as a best estimate and 80% confidence bounds) to yield the best estimate and 80% confidence bounds for the economic risk associated with each scenario. These scenario risks are then summed to give the best estimate and 80% confidence bounds for the total economic risk from accidents at the reference facility.

For **SPENT FUEL STORAGE**, Table C.94 contains estimates of the offsite property damage in 1983 dollars for two accident scenarios: spent fuel pool fires due to seismic and cask drop initiators. Frequency estimated are in Table C.93—in terms of an "average" and "worst" case. Table C.95 contains estimates of the onsite property damage in 1983 dollars corresponding to these same two scenarios. Table C.101 contains estimates of offsite property damage in 1983 dollars for four pairings of accident scenarios and selected mitigative options. For **TRANSPORTATION** of spent fuel by rail, ranges of estimated cleanup costs for three accident scenarios in 1984 dollars are in Daling et al. (1990) (see Section C.5).

C.2.5.2 Non-Fuel Cycle Facilities

Estimated cleanup costs (presumably in 1986 dollars) which can be associated with the **FOUR NON-REACTOR NON-FUEL CYCLE FACILITIES** listed in Section C.1 are in Figure C.1 and Table C.2. They are expressed as functions of the licensed material quantity for both an "average" and "worst-case" release (see Section C.3). For all but the service organizations, the average costs are multiplied by the accident frequencies for the corresponding facilities estimated in Table C.1 to yield economic risk as a function of licensed material quantity for each of the remaining three facilities in Table C.3.

Tables C.7 and C.8 contain best estimates and 80% confidence bounds for the cleanup cost (in 1989 dollars) associated with each accident scenario at a **REFERENCE MANUFACTURING AND DISTRIBUTION FACILITY EMPLOYING BYPRODUCT AND SOURCE MATERIALS** and **SERVICE ORGANIZATIONS** (waste warehouses). Each cost is multiplied by the corresponding estimate for the scenario frequency (also given as a best estimate and 80% confidence

bounds) to yield the best estimate and 80% confidence bounds for the economic risk associated with each scenario. These scenario risks are then summed to give the best estimate and 80% confidence bounds for the total economic risk from accidents.

C.3 A Preliminary Evaluation of the Economic Risk for Cleanup of Nuclear Material Licensee Contamination Incidents (NUREG/CR-4825)

In NUREG/CR-4825 (Ostmeyer and Skinner 1987) and a subsequent document (NUREG/CR-5381 [Philbin et al. 1990], see Section C.4), the economic risk of cleanup costs resulting from non-reactor NRC licensee contamination incidents was evaluated. This first study focused only on incidents where the cleanup cost was < $2E+6. Owing to the preliminary nature of this study, little information was assembled on the frequencies, severities, and costs associated with the contamination incidents. The analysis objective was to provide a technical basis upon which to develop a financial coverage schedule for a rulemaking which would require certain nuclear material licensees to demonstrate adequate financial coverage for contamination cleanup. The analysis sought to provide three products:

1. a rational method to classify licensees according to the potential magnitude and frequency of contamination incidents
2. a model to rank the classes of licensees according to potential incident costs
3. estimates of the economic risk for licensees in each class.

Three indices were proposed to classify the licensees:

1. application/use of the licensed material
2. the licensed curie (Ci) activity
3. the nuclear material form.

Each class was further divided as follows:

- Class 1

  I. research, teaching, experimental, diagnostic, and therapeutic facilities, including hospitals, universities, medical groups, and physicians
  II. measurement, calibration, and irradiation facilities, including users of sealed sources
  III. manufacturing and distribution facilities employing byproduct and source materials, such as radiopharmaceuticals
  IV. service organizations, including waste repackagers, processors, and disposers
  V. non-reactor fuel cycle facilities, handling source and special nuclear material facilities, such as uranium or thorium ore processors.

- Class 2

  This class was subdivided into seven categories ranging from facilities licensed to handle quantities $\leq$ 0.01 Ci to ones licensed to handle > 1,000 Ci, with each subclass spanning a factor of 10 in licensed Ci quantity.

- Class 3

  I. licensees handling sealed sources
  II. licensees handling non-encapsulated Group A sources (i.e., sources whose potential release fraction is < 0.1)
  III. licensees handling non-encapsulated Group B sources (i.e., sources whose potential release fraction is ≥ 0.1.).

Frequencies of contamination incidents were determined for the Class-1 licensees using historic data from the NRC's Non-Reactor Event Report (NRER) database (spanning 1980-1986 at the time of the study). These frequencies are tabulated in Table C.1. Costs were developed from 19 historic events and order-of-magnitude estimates for selected groups of licensee incidents. They have been plotted as a function of licensed Ci quantity in Figure C.1 for two cases:

1. a "worst" case, where 100% of the licensed quantity was assumed to be released

2. an "average" case, where only 15% of the licensed quantity was assumed to be released.

Cleanup costs were assigned to five of the seven divisions of Class-2 licensees at the geometric midpoints of each division's range from Figure C.1. These are listed in Table C.2 for both the worst (licensed quantity released [LQR]) and average cases.

The economic risk was defined as the product of the incident frequency (according to Index Class 1) and the cleanup cost (according to Index Class 2). Using the incident frequencies from Table C.1 and the average cleanup costs from Table C.2, the economic risk per Class-1/Class-2 licensee is tabulated in Table C.3. Division IV from Class 1 was excluded due to the lack of available data for frequency estimation. Division V from Class 1 was excluded because the incidents required cleanup costs $\geq$ \$2E+6, which fell outside the study scope.

Also provided in NUREG/CR-4825 were the following:

- a tabulation of the contamination incidents from the NRER database (1980-1986) and the NRC's OMIT and Fuel Cycle databases (pre-1980), in NUREG/CR-4825 Appendix B

- a tabulation of the historic cost data for cleanup, in NUREG/CR-4825 Appendix C

- the development of a simple cost model which estimates cleanup cost from contaminated floor space, in NUREG/CR-4825 Appendix D.

C.4 Economic Risk of Contamination Cleanup Costs Resulting from Large Non-Reactor Nuclear Material Licensee Operations (NUREG/CR-5381)

In NUREG/CR-5381 (Philbin et al. 1990) and (NUREG/CR-4825 [Ostmeyer and Skinner 1987], see Section C.3), the economic risk of cleanup costs resulting from non-reactor NRC licensee contamination incidents was evaluated. This latter study focused only on incidents at large non-reactor licensees where the cleanup cost was $\geq$ \$2E+6. Five categories of non-reactor licensees were identified, with a reference facility chosen for each:

1. uranium mines and mills, represented by the White Mesa Mill in Blanding, Utah, as described in NUREG/CR-5381 Appendix A

2. uranium hexafluoride (UF_6) conversion plants, represented by the Sequoyah Plant in Gore, Oklahoma, as described in NUREG/CR-5381 Appendix B

3. uranium fuel fabrication facilities, represented by the Westinghouse Facility in Columbia, South Carolina, as described in NUREG/CR-5381 Appendix C

4. large manufacturers and/or distributors of nuclear byproducts, represented by the DuPont Facility in North Billerica, Massachusetts, as described in NUREG/CR-5381 Appendix D

5. nuclear waste warehouses, represented by ADCO Services in Tinley Park, Illinois, as described in NUREG/CR-5381 Appendix E.

The approach taken in NUREG/CR-5381 consisted of the following steps:

- describe each reference facility, postulating accident scenarios for each process in terms of the radioactive material releases, incident frequencies, decontamination efforts required, and decontamination costs for property cleanup and waste disposal

- define incidents from historic data and systems analysis, covering the risk-dominant ones (i.e., the range from high frequency-low consequence events to those with low frequencies but high consequences; decontamination models were employed for the latter pair when historic data were unavailable)

- calculate the economic risk in 1989 dollars as the sum of the products of frequency and cost for each incident, including uncertainty analysis. In essence, the economic risk is the expected cost to decontaminate the property in the event of a radioactive release at the facility.

Where available, historic data for actual or similar facilities were used to estimate the incident frequencies and cleanup costs. In lieu of these, historic data from related industries were employed. Mathematical models were developed to estimate frequencies and costs where no historic data were available. For each point estimate, upper and lower bounds were specified for an 80% confidence interval. These were propagated to yield 80% confidence bounds on both the individual scenario economic risk and the total economic risk for the sum of all the scenarios for a facility.

Tables C.4-C.8 list the incident scenarios, consequence descriptions, cleanup costs, annual frequencies, and annual economic risks for each of the reference facilities. The uncertainty bounds are included for the latter three parameters. As part of the reference facility descriptions, the radioactive inventories and curies released per accident are tabulated in Appendices A-E of NUREG/CR-5381. The contamination incidents for all five licensee classes based on NRC's NRER, OMIT, and Fuel Cycle databases are listed in Appendix F to NUREG/CR-5381. The NRER database included incidents from 1980 onward, while the others included only pre-1980 incidents. The OMIT database focused on non-fuel cycle activities, while the Fuel Cycle database addressed non-reactor fuel cycle operations. Note that neither Table C.5 nor Table C.6 includes a major UF_6 release that occurred at the Sequoyah nuclear power plant. Only accidents at uranium hexafluoride conversion plants and fuel fabrication facilities were considered in the development of Tables C.5 and C.6.

C.5 Preliminary Characterization of Risks in the Nuclear Waste Management System Based on Information in the Literature (PNL-6099)

In PNL-6099, Daling et al. (1990) surveyed literature on the following three components of the nuclear waste management system to develop a preliminary characterization of the associated risks:

- the waste repository (in tuff, salt, and basalt media)
- the MRS facility
- the transportation system supporting both of these.

Five risk categories were defined, of which only those associated with radiological exposure are of interest in this appendix:

1. public and occupational risks from radiological release accidents
2. public and occupational risks from radiological exposure during routine operations
3. economic risks resulting from radiological release accidents.

For the repository, both the preclosure (construction, operations, decommissioning, and retrieval phases) and postclosure periods were addressed. For the MRS facility, the construction, operations, and decommissioning phases were examined. For the transportation system, only operations were considered. Construction and decommissioning of transport equipment were not addressed.

For each component of the waste management system, descriptions for reference facilities and processes were developed, primarily based on conceptual designs (see Chapter 3 of PNL-6099). These were used to form composite risk estimates from all the reviews on a consistent basis by scaling to the reference facilities. Daling et al. (1990) first presented relevant data taken from the reviewed documents prior to their combination into composite risk estimates. Finally, these composites, as scaled for the reference facilities, were provided.

The repository preclosure period has been fairly well examined with respect to risk estimation. Tables C.9-C.11 list exposures for the construction phase. The operations phase has been addressed extensively, as indicated by the data presented in Tables C.12-C.20. Limited information was available on the latter two phases of the preclosure period (decommissioning and retrieval). Table C.21 summarizes this information. Data for the repository preclosure period on a normalized basis is compared in Table C.22.

The repository postclosure period also has been examined quite well, although the estimates are usually very uncertain due to the extremely long time scale considered. Table C.23 lists the health effects associated with four accident scenarios for a waste repository in four different geologic media. Table C.24 lists accumulated doses by body organ for a repository in a tuff medium. Conditional cancer risks from ingestion for six different accident scenarios are given in Table C.25.

For the MRS facility, no radiological risks exist during the construction phase. Radiological risks arise during the operations phase. Tables C.26 and C.27 provide 50-year dose commitments during the operations phase under routine and accident conditions, respectively. For the three accident scenarios listed in Table C.27, the following frequencies were assumed: 1) fuel assembly drop - reasonable chance of occurring annually; 2) shipping cask drop - reasonable chance of occurring once during the facility lifetime; and 3) storage cask drop - unlikely to occur, but requiring consideration.

Occupational doses for standard activities during the operations phase are tabulated in Tables C.28-C.30. For drywell storage in the MRS facility, operations phase risks from selected accident scenarios are shown in Table C.31. Operations phase risks due to accidents for both drywell and cask storage concepts are listed in Table C.32. The following radiological risks to the worker from routine operations during the decommissioning phase were estimated: 120 person-rem for drywell storage and 128 person-rem for cask storage.

The radiological risks from transportation have been examined extensively. Dose rates and total doses under normal (non-accident) shipping conditions for spent fuel transport by truck and rail cask are listed in Tables C.33 and C.34. Note that both tables were based on a shipping cask modeled as an infinite line source. Thus, the doses reported are reasonable from 3 m to 15 m but probable overestimates beyond 40 m away. Radiological risks are given in Table C.35. Dose estimates from selected accidents during rail shipment of spent fuel are provided in Tables C.36-C.38. Transportation risks under both normal and accident conditions have been combined for truck and rail shipments of spent fuel in Table C.39. The risks encountered during routine transportation (i.e., non-accident) for a waste management system without and with an MRS facility are listed in Tables C.40 and C.41, respectively, for both spent fuel and HLW shipment. A range of cleanup costs (1984 dollars) were estimated for three accident classes for spent fuel transportation by rail: 1) impact = \$2.0E+5 - \$9.5E+6; 2) impact with burst = \$1.4E+6 - \$7.0E+7; and 3) impact with burst and oxidation = \$1.3E+7 - \$6.2E+8.

The radiological risks from all three components of the waste management system were converted into composite estimates for the reference facilities assuming a throughput of 3,000 MTU/yr, a maximum repository capacity of 70,000 MTU, and a conversion factor of 2.0E-4 LHE per person-rem.[1] Public and occupational risks from the preclosure period of the waste management system without an MRS facility are tabulated in Tables C.42 and C.43, respectively. The corresponding risks for the system with an MRS facility are provided in Tables C.44 and C.45, respectively. Total risks for the preclosure period are given in Table C.46. Table C.47 summarizes the annual and total life-cycle risks for the entire waste management system.

C.6 Preliminary Ranking of Nuclear Fuel Cycle Facilities on the Basis of Radiological Risks from Accidents

In an unpublished PNNL study, Pelto et al. examined the risk to the public and plant worker from radiological accidents at non-reactor nuclear fuel cycle facilities. The study was essentially a literature survey, similar to that of PNL-6099 (Daling et al. 1990 [see Section C.5]), but focusing on all non-reactor fuel cycle facilities, rather than just those associated with nuclear waste management. The 13 categories of non-reactor fuel cycle facilities listed in Section C.1.1 were identified.

Representative non-reactor fuel cycle facilities were selected for each of the 13 categories based on actual facilities or conceptual designs provided by Schneider et al. (1982). These representative descriptions, including site characteristics, were combined with the ALLDOS computer code (Strenge et al. 1980) to scale the consequences of radioactive release on a consistent basis. Radiological risk was measured in whole body person-rem/GWe-year (i.e., in terms of the annual requirements of a 1,000-MWe [1-GWe] LWR) as the 50-year population dose commitments for selected organs, based only on the airborne pathway. Although the source documents reviewed by Pelto et al. were dated prior to 1983, they are felt to provide at least conservative results. Any subsequent refinements to the facilities would have tended to reduce risks based on "lessons learned."

Fullwood and Jackson (1980) estimated the radiological risk to the plant worker, citing the following pair of values: 1) 7.0E-4 person-rem/GWe-year for MOX fuel refabrication, and 2) 1.0E-4 person-rem/GWe-year for fuel reprocessing. The remaining literature addressed public risk as discussed below.

Cohen and Dance (1975) performed a risk analysis for uranium milling, yielding an expected population dose (public risk) of about 0.001 person-rem/GWe-year mainly due to the release of mill tailings slurry. Three accident scenarios were identified, and their frequencies and population doses were estimated as tabulated in Table C.48. Cohen and Dance also performed a risk analysis for the conversion phase of the fuel cycle, obtaining an expected population dose ranging from 7.6E-4 to 0.0056 person-rem/GWe-year mainly due to a hydrogen explosion during the reduction step. Six accident scenarios were identified, and their frequencies and population doses were estimated as provided in Table C.49.

Cohen and Dance (1975) also give risk estimates for enrichment and fuel fabrication. For enrichment, the expected population dose ranged from 0.0025 to 0.0037 person-rem/GWe-year, dominated by release from a hot UF_6 (uranium hexafluoride) cylinder. The frequencies and population doses from the four accident scenarios considered for this phase of the fuel cycle are tabulated in Table C.50. For fuel fabrication, the expected population dose ranged widely from 4.8E-5 to 0.010 person-rem/GWe-year, again dominated by release from a hot UF_6 cylinder. Seven accident scenarios were identified and quantified as shown in Table C.51.

Cohen and Dance (1975), Erdmann et al. (1979), and Fullwood and Jackson (1980) addressed the public risk associated with MOX fuel refabrication. The ranges of expected population dose are listed along with the dominant risk contributors in Table C.52. Tables C.53-C.55 present the seven or eight accident scenarios considered for this phase of the fuel cycle, along with the associated frequencies and population doses. The relatively low risk and population doses estimated by Fullwood and Jackson (1980) indicated that results were sensitive to modeling assumptions. The same set of studies also examined the public risk associated with the fuel reprocessing phase of the fuel cycle, yielding the ranges of expected population dose and dominant risk contributors given in Table C.56. Eight to 12 accident scenarios were identified and quantified for this phase; these are listed and quantified in Tables C.57-C.59. Six accident scenarios from a study by Cooperstein et al. are presented in Table C.60, although a public risk estimate was not generated in the report.

Karn-Bransle-Sakerhat (1977), the DOE (1979), and Erdmann et al. (1979) addressed the spent fuel storage phase of the nuclear fuel cycle, estimating expected population doses ranging from 1.7E-6 to 8.9E-5 person-rem/GWe-year, dominated by either a fuel basket or fuel assembly drop accident. The frequency and population dose for the fuel assembly drop accident in Erdmann et al. (1979) were taken from their analysis for the fuel reprocessing phase (see Table C.58). Karn-Bransle-Sakerhat (1977) identified and quantified fuel transfer basket and fuel assembly drop accidents, as indicated in Table C.61. The public risk from HLW storage accidents was examined by Smith and Kastenberg (1976), who reported an expected population dose of 2.3E-4 person-rem/GWe-year mainly due to a major rupture of a waste canister combined with the independent failure of one HEPA filter. Six accident scenarios were identified, and their frequencies and population doses were estimated as tabulated in Table C.62.

Geologic waste disposal has been the subject of several risk studies. Two of the studies, DOE (1979) and Erdmann et al. (1979), were reviewed by Pelto et al. The expected population doses varied widely between these two studies for the preclosure period of geologic disposal, as indicated in Table C.63. The Analytic Sciences Corporation (TASC 1979) reviewed the peak individual dose (rem/year) to the critical organ during the postclosure period as determined from other studies. Figure C.2 summarizes these results. Erdmann et al. (1979) estimated an expected population dose of 5.0E-11 person-rem/GWe-year for the postclosure period.

Risks associated with the transportation phase of the nuclear fuel cycle have been investigated by Cohen and Dance (1975), Erdmann et al. (1979), Fullwood and Jackson (1980), the DOE (1979), the NRC (1975a, 1975b, 1976, 1977), Berman et al. (1978), the U.S. Atomic Energy Commission (AEC 1972), and Hodge and Jarrett (1974). Table C.64 summarizes the expected population doses from accidents during plutonium oxide, spent fuel, and HLW shipment. Table C.65 lists the frequencies and population doses for accident scenarios associated with spent fuel and plutonium oxide transportation, by rail and truck, respectively, as determined by Cohen and Dance (1975). Erdmann et al. (1979) identified accident scenarios for four transportation systems: spent fuel by rail and truck, plutonium oxide by truck, and HLW by rail. The associated frequencies and population doses are tabulated in Table C.66. Fullwood and Jackson (1980) examined rail shipment of spent fuel and HLW, identifying and quantifying the accident scenarios presented in Table C.67. Projekt Sitherkeitsstudien Entsorgung (PSE 1981) and Elder (1981) identified and quantified transportation accident scenarios for rail shipment of spent fuel (Tables C.68 and C.69), although they did not convert these estimates into expected population doses.

Having surveyed available literature and extracted the quantitative information deemed representative of non-reactor fuel cycle risks, Pelto et al. then scaled the risk estimates on a consistent basis for the purpose of comparison. Site-specific

conditions for the representative facilities were input to the ALLDOS computer code to yield the public risks from each nuclear fuel cycle element as summarized in Table C.70. Those elements with comparable risks were grouped together into two categories as follows: 1) conversion, enrichment, MOX fuel refabrication, fuel reprocessing, spent fuel storage, and transportation, with expected population doses from 0.012 to 0.27 person-rem/GWe-year; and 2) milling, fuel fabrication, HLW (solidified) storage, and geologic waste disposal (preclosure period), with expected population doses from 4.0E-5 to 0.0050 person-rem/GWe-year.

C.7 Cost-Benefit Analysis of Unfired PuO_2 Pellets as an Alternative Plutonium Shipping Form (NUREG/CR-3445)

NUREG/CR-3445 (Mishima et al. 1983) is of interest not so much for the value-impact analysis performed (which was fairly preliminary), but for the data presented on industry costs and occupational exposure incurred during the pelletizing and reconstitution processes for PuO_2. Mishima et al. (1983) considered the potential costs of altering the current practice of shipping PuO_2 as a powder to one where it is shipped as unfired pellets. The pellets would then be reconstituted into powder following receipt at the fuel fabrication facility. Direct costs (measured in 1983 dollars) consisted of equipment, labor, redesign of process and transport procedures, supplies, services, and additional transport costs. A facility throughput of 20 kg/day was assumed.

Capital equipment costs for pellet fabrication and powder reconstitution are listed in Tables C.71 and C.72, respectively. Tables C.73 and C.74 present operating costs associated with the startup and process, respectively, for both pellet fabrication and powder reconstitution. Indirect costs (occupational doses) are summarized in Tables C.75 and C.76 for pellet fabrication and powder reconstitution, respectively.

C.8 A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees (NUREG-1140)

In NUREG-1140, McGuire (1988) performed a regulatory analysis covering emergency preparedness for non-reactor nuclear facilities, both fuel and non-fuel cycle. It contained five of the six steps required in a regulatory analysis, omitting only the last (implementation). The regulatory analysis began with the following statement of the problem:

> "Should the NRC impose additional emergency preparedness requirements on certain fuel cycle and other radioactive material licensees for dealing with accidents that might have offsite releases of radioactive material?"

The objective was to answer this question and, if answering yes, determine how to impose the requirements.

The identification and preliminary analysis of alternative approaches to the problem came next. A description of the proposed actions and justification for their need were spelled out. Three alternatives were cited:

1. adopting a regulation containing the proposed requirements
2. imposing the requirements by license condition
3. imposing no new requirements (the status quo, or baseline, case).

As part of the preliminary analysis, McGuire (1988) established the following criterion for deeming an accident significant. A release causing a person outside the plant along the plume centerline to receive an EDE > 1 rem, a thyroid dose > 5 rems, or an intake of soluble uranium > 2 mg would constitute a significant accident. These values were chosen from the lower ends of the dose ranges for which the EPA states that protective actions should be considered. Fifteen classes of licensees were identified, from which those which could have significant accidents were identified for further analysis. Those identified consisted of the following:

- Fuel Cycle Facilities
  - uranium mills
  - UF_6 conversion plants
  - enrichment plants
  - uranium fuel fabrication plants
  - plutonium fuel fabrication plants
  - spent fuel storage facilities
  - spent fuel reprocessing plants
  - nuclear fuels research facilities (special nuclear materials).
- Byproduct Material Facilities (only those handling large enough quantities of unsealed radioactive material so that the need for offsite emergency preparedness should be considered)
  - radiopharmaceutical manufacturers
  - sealed source manufacturers.

For the estimation and evaluation of values and impacts, McGuire (1988) performed the following three steps for each facility class:

1. survey the accident history, including similar facilities in the database
2. quantify the accident source terms, using NRC analyses of several severe accidents possible at non-reactor facilities
3. calculate the offsite dose via a "standard" dose calculation (i.e., assume a release fraction, atmospheric dispersal model, and three exposure pathways [inhalation and cloud- and ground-shine]).

The number of licensees potentially affected consisted of 14 fuel cycle and about 17 byproduct material licensees. Of the three alternatives approaches to the problem identified earlier, the first two would have the same values and impacts, and the third represented the baseline case for comparison. Thus, only one value-impact analysis was performed, with the value measured in terms of public risk reduction.

Two cases were considered for estimating the risk reduction. The first assumed a release occurred with an EDE of 5 rems at a distance of 100 m under the Pasquill Class F atmospheric stability condition and a wind speed of 1 m/s. Under these conditions, the area over which the EDE would exceed 1 rem was estimated to be 0.006 mi^2. For a typical population density of 3000/mi^2 at the facilities, about 20 people would be in the estimated area, with 80% (16) indoors and the remainder (4) outdoors. An outdoor person would receive an average dose of about 3 rems, while one indoors would receive 1/2 of that due to protection from the building. For the base case, this amounted to a total collective dose of about 40 person-rems. The dose savings was assumed to be 1/2 of that, or about 20 person-rems. If 0.0001 cancer death occurred per rem, the number of lives saved would be about 0.002 for the worst meteorology, or about 2E-4 for an overall average meteorology.

To estimate the frequency of a major release, McGuire (1988) used statistics from the insurance industry. A fire loss occurred in unsprinklered commercial and industrial facilities at a rate of about 0.006/yr. Where available, sprinklers failed at a rate of 0.038/demand. Thus, a reasonable estimate of the fire loss rate for a sprinklered facility (typical of radioactive licensees) would be about 0.006/yr x 0.038, or 2E-4/yr. Assuming additional site-specific factors would halve this rate, an estimate of 1E-4/yr was generated for the frequency of a major radioactive release. When multiplied by the consequence estimate of 2E-4 life saved on average, an estimate of 2E-8 life saved per facility per year was obtained as the public risk reduction. In monetary terms, this translated to $0.2/facility-yr, assuming a value of $1E+7/life.

The second case analyzed was essentially equivalent to the first, except that the 5-rem EDE was now assumed at a distance of 1,000 m. This translated into an increase in the area over which the EDE would exceed 1 rem to 0.15 mi^2, encompassing 450 people. Retaining the other assumptions from Case 1, the public risk reduction for Case 2 was estimated at 4E-7 life saved per facility per year, or $4/facility-yr.

Costs to implement the proposed action were based on data from two radiopharmaceutical manufacturers, coupled with the assumption that the licensee would be required to have a 50-page plan containing instructions for what to do in the event of an emergency such as a fire. The initial setup would cost $84,000 ($8,400/yr spread over 10 years) for a small program and $550,000 ($55,000/yr) for a large program. Labor costs were assumed to be included as 1/2 to 2/3 of these costs at a rate of $30/hr. For either program, the annual operating cost would be $18,000. Thus, the industry costs were estimated to be about $26,000/facility-yr for a small program and $73,000/yr for a large one. The NRC cost to review and inspect the plan was estimated to be $4,000/facility-yr, yielding total cost estimates of about $30,000/facility-yr (small program) and $77,000/facility-yr (large program).

For the presentation of results, McGuire utilized a simple table, as follows:

Licensee Size	Cost	Benefit
Small	$30,000/facility-yr	$0.2/facility-yr
Large	$77,000/facility-yr	$4/facility-yr

The expected life savings amounted to 2E-8/facility-yr for small licensees and 4E-7/facility-yr for large ones. Roughly 20-30 small and 2-3 large licensees could be expected to achieve these savings. These results clearly indicated that the potential risk reduction to the public was very small.

The decision rationale for this regulatory analysis was summarized as follows:

> "The cost of this [emergency] preparedness may not be justified in terms of protecting public health and safety. Rather, we would justify it in terms of the intangible benefit of being able to reassure the public that, if an accident happens, local authorities will be notified so they may take appropriate actions."

> "Although emergency preparedness for fuel cycle and other radioactive material licensees cannot be shown to be cost effective, the NRC feels that such preparedness represents a prudent step which should be taken in line with the NRC's philosophy of defense-in-depth, to minimize the adverse effects which could result from a severe accident at one of its facilities."

McGuire (1988) also presented dose tables for various accident releases at selected fuel and non-fuel cycle facilities. Tables C.77-C.81 address selected fuel cycle facilities. Tables C.82-C.86 present doses for non-fuel cycle facilities (i.e., byproduct material facilities).

C.9 Regulatory Impact Analysis of Final Environmental Standards for Uranium Mill Tailings at Active Sites (EPA 520/1-83-010)

In EPA 520/1-83-010 (EPA 1983), the EPA performed a regulatory impact analysis covering uranium mills. Specifically, EPA addressed the disposal of uranium mill tailings at active sites by evaluating the impact of final environmental standards for this disposal. The standards considered were ones which addressed only the disposal of mill tailings; releases during the operations phase of a uranium mill were not included. The study contained the six steps required in a regulatory analysis, following Executive Order 12291 (see Section 1).

The statement of the problem was essentially to investigate final environmental standards for disposal of uranium mill tailings in both the short and long term. Uranium mill tailings pose an environmental hazard through the release of radon, a radioactive gas. Four methods of controlling these releases were identified:

1. discourage misuse (e.g., use of tailings in construction of homes)

2. provide barriers to radon emission

3. prevent the spread of tailings

4. protect the tailings from water intrusion.

The objective was to determine which of many alternative standards proposed to limit emissions from uranium mill tailings would be optimal from a health and cost perspective.

The identification and preliminary analysis of alternative approaches to the problem addressed 13 proposed standards for disposal. These standards were defined according to the ability to control radon release after disposal (in terms of radon release rates) and the length of time for which such control would be required. The spectrum of alternatives is displayed in Table C.87, ranging from a baseline case of no controls (Alternative A) to the most stringent case limiting radon release to 2 pCi/m^2-s using passive control for 1,000 years, with improved radon control during operations for new piles (Alternative D5). Both existing and new tailings piles (at both existing and future facilities) were considered.

As part of the preliminary analysis, the status of licensed conventional U.S. mill sites as of 1/1/83 was ascertained and tabulated in EPA 520/1-83-010 (EPA 1983) Chapter 2. Characteristics of the control methods for both existing and new piles were specified for the 13 alternative standards in Tables C.88 and C.89, respectively.

EPA next proceeded to the estimation and evaluation of values and impacts. The value was quantified in terms of health effects averted through control of radon emissions. This was accomplished in two steps. First, each alternative was characterized in terms of how well it provided for the following three items:

1. stability of the tailings pile

2. control of radon emissions from the pile

3. protection of the pile against water intrusion.

These are summarized in Table C.90. Next, the values were quantified on a comparative basis through the definition of an "effectiveness index" for the four release control methods previously identified. Each alternative was rated in terms of this index using a scale from 1 to 10, considering the factors shown in Table C.90. A weighted average effectiveness was calculated for each alternative.

Costs for disposal of existing and new mill tailings piles were estimated in 1983 dollars for the control method associated with each alternative based on selected model pile sizes (2, 7, and 22 metric tons (MT) for existing piles; 8.4 MT for new piles). The average cost per effectiveness index was calculated for each alternative as the ratio of the model pile disposal costs to the previously estimated effectiveness index. These were then converted to the incremental cost per alternative i as follows:

$$(\text{Disposal Cost}_i - \text{Disposal Cost}_{i-1})/(\text{Effectiveness Index}_i - \text{Effectiveness Index}_{i-1})$$

These calculations are summarized in Table C.91 for both existing (all three sizes) and new tailings piles.

The incremental costs were plotted against the effectiveness indices for the various alternatives for each model pile size (see Figure C.3). The alternatives exhibiting negative or small positive slopes in the plot were the desirable ones. Sensitivity analyses were conducted by varying the weighing factors for the effectiveness index and considering the cost per effectiveness index for 100 rather than 1,000 years.

The analysis of industry cost and economic impact was the next item. Thirty-seven economic impact cases for the 13 alternative standards were identified by considering the following three categories for each of the 12 non-baseline alternatives (i.e., all but Alternative A):

1. existing mill tailings

2. new mill tailings at existing mills

3. new mill tailings at new mills.

For existing tailings, disposal costs were assumed to be incurred from 1983 through 1987. For new tailings, disposal costs were assumed to be incurred from 1983 through 2000. Present worth calculations were performed for three discount rates (0, 5, and 10%). The cost estimates for all 13 alternative standards are summarized in Table C.92.

The presentation of results consisted of the various tables and figures produced during the value-impact analysis, especially the summary Tables C.90 and C.92. The decision rationale for selection of a recommended disposal standard was as follows. The standards were based on current population data, with no "relaxation" for "remote" sites. Passive controls were preferred over institutional ones because of the need to provide long-term protection. The radon emission limit of 20 pCi/m^2-s was selected since both the cost-effectiveness and practicality of providing additional radon control dropped rapidly below this threshold. As a result, Alternative C3 was recommended since it best met these criteria while minimizing economic impact and providing high, although not maximum, values.

The implementation step of the regulatory analysis was briefly addressed when EPA considered the relationship of the proposed standards to the Regulatory Flexibility Act (see Guidelines Section 5.2). An analysis of compliance with this Act was cited as unnecessary because the standards would not significantly impact a substantial number of small entities.

C.10 Value-Impact Analysis of Accident Preventive and Mitigative Options for Spent Fuel Pools (NUREG/CR-5281)

In NUREG/CR-5281, Jo et al. (1989) conducted what essentially amounted to a regulatory analysis of a non-reactor nuclear fuel cycle facility using the 1983 Handbook (Heaberlin et al. 1983) as guidance. It included the six steps required in a regulatory analysis. In the statement of the problem, Jo et al. observed that spent fuel pools at power reactor sites were being required to store more fuel than originally anticipated because of the lack of a waste reprocessing plant or repository. The objective of the analysis was to assess possible preventive and mitigative strategies for spent fuel pool accidents in light of the pools being used to store more spent fuel than originally anticipated.

In the identification and preliminary analysis of alternative approaches to the problem, Jo et al. proposed three main alternatives for spent fuel pool accident prevention and mitigation:

1. reduction of pool inventory
2. improvement of reliability of pool makeup water
3. implementation of one or more "representative" mitigative options.

Under the first alternative (inventory reduction), limited low-density fuel storage would be permitted in the pool. Essentially, fuel discharged from the reactor within the past two years would be stored in a low-density configuration, promoting air cooling of the fuel in the event of a loss of pool water inventory. This alternative would require that a utility replace its current high-density storage racks with low-density ones, increasing the need for added storage capacity. Five options were considered:

1. supplemental wet pool storage
2. drywell storage
3. storage in a vault
4. storage in a cask
5. storage in a silo.

The preliminary analysis consisted of collecting spent fuel and fuel pool data for all U.S. plants through 1986 (presented in NUREG/CR-5281 Chapter 3).

The analysis proceeded to the estimation and evaluation of values and impacts (Alternative 1), using the 1983 Handbook as a guide. Risk-dominant sequences for a spent fuel pool were identified. They consisted of structural failure due to an earthquake and a compromise of structural integrity through impact of a heavy object, such as a storage cask. For this latter accident, the conditional probability of pool structural failure was taken to be one. Public health and offsite property damage were estimated using the MACCS computer code (Chanin et al. 1990), specifying both a best-estimate and worst-case radiological source term. Accidental occupational exposure was assumed to be similar to that from TMI-2 (i.e., < 4580 person-rem). Onsite property damage was assumed to result from loss of pool inventory followed by a zircaloy fire which spread throughout the pool. This resulted in the melting of 1/2 of the fuel cladding and contamination of containment, with a subsequent loss of containment integrity. The accident frequencies, offsite consequences (public health and property damage), and onsite property damage are tabulated in Tables C.93-C.95, respectively. The costs (in 1983 dollars) given in Tables C.94 and C.95 were expanded on a plant-by-plant basis in NUREG/CR-5281 Appendix A, serving as input to the industry cost estimates provided in Table C.96.

The presentation of results (Alternative 1) consisted of two summary tables. The first (Table C.97) listed all parameters affecting the attributes considered in the value-impact analysis, including data references. The second (Table C.98) was the standard value-impact analysis summary table in the 1983 Handbook, including the net value and ratio calculations for both the best-estimate and worst cases. Additional value-impact measures were indicated in the second table (i.e., the ratio of benefits (in dollars) to cost and the cost of implementation per averted person-rem).

Sensitivity studies were performed by varying the following:

- pool failure probability
- discount rate
- monetary conversion factor for health effects
- site economics
- meteorology.

Only the first item (increase in failure probability) could shift the net value to the positive side. Based on the analysis results, the decision rationale for Alternative 1 concluded that it was not justified due to the negative net value and low ratios, indicative of an action whose overall effect is undesirable.

Alternative 2 (improvement of pool makeup water reliability) addressed the problem of interruption of the circulation of pool cooling water. Such interruption could result in a pool temperature rise until boiling would occur. Thermal-hydraulic analyses from FSARs indicated a considerable time lag between loss of circulation and uncovering of fuel assemblies. Therefore, much time would be available to restore normal cooling or implement a standby cooling option.

In the estimation and evaluation of values and impacts (Alternative 2), it was decided to examine four "generic" pool cooling and makeup systems, ranging from the minimum Standard Review Plan (SRP) requirement to crediting three makeup trains, including the fire system. Scoping calculations were performed to estimate failure frequencies. These are quantified in Table C.99. Radiological impacts were found to be negligible. Further quantification was conducted only for averted cost (resulting from replacement power until pool cooling is restored) and industry implementation costs (discounted at 10%), with the costs in 1983 dollars. Table C.100 is essentially the presentation of results (Alternative 2) and indicates very small ratios of averted to implementation cost for each of the four systems. Thus, the decision rationale was that Alternative 2 would not be justified.

Alternative 3 consisted of the following three representative mitigative options for spent fuel pool accidents:

1. M1 = covering fuel debris with solid materials
2. M2 = installing a water spray system above the pool
3. M3 = installing a building ventilation gas treatment system to reduce the airborne concentration of radionuclides prior to their release.

Two representative accident sequences were postulated. The first (A1) consisted of a complete loss of pool water inventory, followed by a zircaloy fire, representing an upper bound in terms of radiological release. The second (A2) consisted of a complete loss of pool water inventory, followed only by cladding failures (i.e., no zircaloy fire). This represented a best estimate in terms of radiological release.

The estimation and evaluation of values and impacts (Alternative 3) considered the six possible pairings of accident and mitigation scenarios (i.e., A1/M1, A1/M2, A1/M3 [dismissed since M3 could not cope with A1], A2/M1, A2/M2 [judged to be the same as A1/M2] and A2/M3). These reduced to four cases, for which a crude value-impact assessment was

performed, similar to what was termed a "first approximation" in Chapter 2 of the 1983 Handbook. Offsite consequences were estimated using MACCS for both a worst case (high population density and worst source term) and an average case (average population density and average source term). Costs (in 1983 dollars) were generated by assuming a Category I storage tank of 200,000-gal capacity and a complete spray system would need to be installed. The calculation results for each of the four cases are presented in Table C.101.

The presentation of results (Alternative 3) consisted of the value-impact summary (Table C.102), which indicated that installation of pool sprays was not cost effective, based on the best-estimate measures provided in the table [net benefit, ratio, ratio of benefits (in dollars) to cost, and cost of implementation per averted person-rem]. The decision rationale (Alternative 3) was the same as that for the other alternatives, namely not to recommend the alternative based on the value-impact results. However, the possibility of implementing Alternative 3 on a plant-by-plant basis was mentioned, since the high-estimate measures indicated marginal cost effectiveness. At plants where the conservative assumptions used in NUREG/CR-5281 might be approached, Alternative 3 might warrant implementation.

C.11 Nuclear Fuel Cycle Facility Accident Analysis Handbook (NUREG-1320)

In NUREG-1320, Ayer et al. (1988) provided methods to determine the release of radioactive material to the atmosphere and within a plant resulting from potential accidents at the following types of nuclear fuel cycle facilities: fuel fabrication, fuel reprocessing, high-level waste storage/solidification, and spent fuel storage. Six types of accidents were addressed: fires, explosions, spills, tornadoes, criticalities, and equipment failures. These were chosen as being the major contributors to the radiological accident risk from the operations of fuel cycle facilities. While NUREG-1320 provided methods for calculating consequences from these accidents, it did not provide methods for determining the accident probabilities.

Ayer et al. assembled accident descriptors for both the facilities and their processes. For simplicity, a representative facility was developed containing common descriptors from each of the four types. These descriptors are shown in Table C.103. For each type of fuel cycle facility, Ayer et al. assembled process accident descriptors, listed in Tables C.104-C.107. These descriptors were based on the following process parameters:

- quantity, chemical, and physical form of radionuclides
- quantity and characteristics of flammable and combustible materials
- radionuclide content of materials with high fissile material content
- characteristics of process equipment providing airborne containment or confinement
- others that could enhance or mitigate airborne release (e.g., pressurized systems).

Source terms for each of the six types of accidents were discussed. Behavioral mechanisms for airborne particles were summarized, as shown in Table C.108. Following these were the detailed descriptions of the calculational methods for estimating the source terms from each type of accident. Both hand and computer calculations were presented. All necessary reference tables and figures for conducting a "standard" analysis were provided, along with additional references for "specialized" assessments.

To illustrate the use of the analytic procedures, Ayer et al. identified four "primary" and seven "secondary" sample problems, as follows:

Primary:

1. Slug Press Fire (MOX Fuel Manufacturing)
2. Solvent Extraction Fire (Fuel Reprocessing)
3. Glove Box Explosion
4. Powder Spill During Tornado

Secondary:

5. Flashing Spray (Fuel Reprocessing)
6. Pressurized Release of Powder
7. Radioactive Powder Spill
8. Liquid Spill of Plutonium Nitrate
9. Aerodynamic Entrainment of Powders from Thick Beds During Tornado
10. Fragmentation of Brittle Solids by Crush Impact During Tornado
11. Inadvertent Criticality in a Fuel Reprocessing System

For each, Ayer et al. conducted a sample source term calculation, showing use of both hand calculations and computer tools. The main computer codes were as follows:

1. TORAC - for analysis of tornado-induced gas dynamics and material transport (Andrae et al. 1985)

2. EXPAC - for analysis of explosion-induced gas dynamics and material transport (Nichols and Gregory 1988)

3. FIRAC - for analysis of fire-induced gas dynamics, thermal, and material transport (Nichols and Gregory 1986)

Although designed mainly for analysis of the ventilation system (the primary airborne release pathway), these codes can be used for other airflow pathways as well. The codes, especially TORAC, can be extended to model accidents associated with criticality, spills, and equipment failure. Limitations involve the gas dynamics models, which are based strictly on lumped-parameter formulations, and the material transport capability, which is very basic and relies on information found in the literature.

For each of the primary sample problems, the authors of NUREG-1320 carried through a complete radioactive airborne release calculation. The results were presented through a series of tables and figures, too numerous to reproduce here.

C.12 Endnotes for Appendix C

1. The 1990 BEIR V report updated the radiation exposure coefficient to 5E-4 fatal cancer/person-rem, or inversely 2,000 person-rem/fatal cancer (National Research Council 1990).

2. For consistency when using Tables C.42-C.47, or values derived from them, the analyst should employ 5,000 person-rem/health effect, the conversion factor assumed by Daling et al. (1990), from whom these tables have been extracted. However, the analyst should be aware that BEIR V updated the radiation exposure coefficient to 5E-4 fatal cancer/ person-rem, or inversely 2,000 person-rem/fatal cancer (National Research Council 1990).

3. Recent experience at the DOE Savannah River Site suggests frequencies of glove failure as much as 10 times higher.