

quantifiable in terms of safety or monetary equivalent. If this is the case, such values and impacts should be treated by other attributes and not included under this attribute. On the other hand, if potential values from the assessments are difficult to identify or are otherwise not easily quantified, then they should be addressed under this attribute.

5.5.14 Regulatory Efficiency

This attribute attempts to measure regulatory and compliance improvements resulting from the proposed action. These may include changes in industry reporting requirements and the NRC's inspection and review efforts. Achieving consistency with international standards groups may also improve regulatory efficiency for both the NRC and the groups. This attribute is qualitative in nature.

In some instances, changes in regulatory efficiency may be quantifiable, in which case the improvements should be accounted for under other attributes, such as NRC implementation or industry operation. Regulatory efficiency actions that are not quantifiable should be addressed under this attribute.

5.5.15 Antitrust Considerations

The NRC has a legislative mandate under the Atomic Energy Act to uphold U.S. antitrust laws. This qualitative attribute is included to account for antitrust considerations for those proposed actions that have the potential to allow violation of the antitrust laws.

If antitrust considerations are involved, and it is determined that antitrust laws could be violated, then the proposed action must be reconsidered and, if necessary, redefined to preclude such violation. If antitrust laws would not be violated, then evaluation of the action may proceed based on other attributes. The decision as to whether antitrust laws could be violated must rely on a criterion of reasonable likelihood, since it is difficult to anticipate the consequences of a regulatory action with absolute certainty.

5.5.16 Safeguards and Security Considerations

The NRC has a legislative mandate to maintain the common defense and security and to protect and safeguard national security information in its regulatory actions. This attribute includes such considerations.

In applying this attribute, it must be determined whether the existing level of safeguards and security is adequate and what effect the proposed action has on achieving an adequate level of safeguards and security. If the effect of the proposed action on safeguards and security is quantifiable, then this effect should be included among the quantitative attributes. Otherwise the contribution of the action will be evaluated in a qualitative way and treated under this attribute.

5.5.17 Environmental Considerations

Section 102(2) of the National Environmental Policy Act (NEPA) requires federal agencies to take various steps to enhance environmental decision-making. NRC's procedures for implementing NEPA are set forth in 10 CFR Part 51. Many of the NRC's regulatory actions are handled through use of a generic or programmatic environmental impact statement (EIS), environmental assessment (EA), or categorical exclusion. If these vehicles are used, no further consideration is required in a regulatory analysis covering the same subject matter as the environmental document, although a summary of the most salient results of the environmental analysis should be included. Otherwise, an evaluation of the action with respect to its impact on the environment is required. Such an evaluation is usually handled separately from the value-impact analysis described in this Handbook. It could be the case that mitigation or other measures resulting from the

environmental review may result in cost increases that should be accounted for in the regulatory analysis. Alternatives examined in an EIS or EA should correspond as closely as possible to the alternatives examined in the corresponding regulatory analysis.

5.5.18 Other Considerations

The above set of attributes is believed to be reasonably comprehensive for most value-impact analyses. It is recognized that any particular analysis may also identify attributes unique to itself. Any such attributes should be appropriately described and factored into the analysis.

5.6 Quantification of Change in Accident Frequency

As expressed in this Handbook, the term "accident" should be viewed generally as an unplanned occurrence which potentially releases radioactive materials, applicable to both power reactor and non-reactor facilities. Discussions in this section assume familiarity with the concepts of risk as related to the nuclear industry, as well as knowledge of event- and fault-tree terminology. The reader unfamiliar with these concepts or in need of review is directed to existing risk assessments or such standard references as the PRA Procedures Guide (NRC 1983a) and the Fault Tree Handbook (Vesely et al. 1981). The NRC formally endorsed the use of PRA methods in nuclear regulatory activities with its issuance of a Final Policy Statement in 1995 (NRC 1995b). The Policy Statement includes four elements, the first of which states that

The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

SECY-95-079 contains a status update of NRC's PRA implementation plan. SECY-95-280 contains a framework for applying PRA in reactor regulation. As noted in Section 3, as this version of the Handbook was being completed a number of NRC staff activities were underway which relate to PRA use in NRC regulatory activities. These include

- completion of the staff's review of licensee-submitted IPEs
- evaluation of these IPEs for potential use in other regulatory activities, documented in a draft report to be published as NUREG-1560 (NRC 1996b)
- development of guidance on the use of PRA in plant-specific requests for license changes, including regulatory guides for use by licensees in preparing applications for changes and standard review plans for use by the NRC staff in reviewing proposed changes.

These activities should result in a more consistent and technically justified application of PRA in NRC's regulatory process. In particular, draft NUREG-1560 contains a detailed and explicit description of acceptable attributes of a quality PRA. The activities, along with staff work planned for FY 1997 to initiate improvements to the economic models now used in NRC's offsite consequence analyses (e.g., the MACCS code), should have a significant impact on the PRA-related portions of this Handbook. Consequently, the discussion in this Handbook on the use of PRA and offsite consequence estimates should be viewed as interim guidance that may be relied upon until the Handbook is updated to accommodate the NRC's new position on these regulatory issues. The staff expect to initiate this update as the preceding PRA guidance nears completion.

Estimates of the change in accident frequency resulting from a proposed NRC action are based on the effects of the action on appropriate parameters in the accident "equation."⁽²⁾ Examples of these parameters might be system or component failure probabilities, including those for the facility's containment structure. The estimation process involves two steps: 1) identification of the parameters affected by a proposed NRC action (see Section 5.6.1); and 2) estimation of the values of these affected parameters before and after the implementation of the action (see Section 5.6.2).

The parameter values are substituted in the accident equation to yield the base- and adjusted-case accident sequence frequencies. The sum of their differences is the reduction in accident frequency due to the proposed NRC action.

The process can be viewed as follows. The frequency for accident sequence ij is⁽³⁾

$$F_{ij} = \sum_k M_{ijk}$$

where M_{ijk} = the frequency of minimal cut set k for accident sequence i initiated by event j .

A minimal cut set represents a unique combination of occurrences at lower levels in a structural hierarchy (e.g., component failures in power reactor systems) which produces an overall occurrence (e.g., reactor core damage) at a higher level. It takes the form of a product of these lower level occurrences. The affected parameters comprise one or more of the multiplicative terms in the minimal cut sets. Thus, the reduction in accident sequence ij 's frequency is

$$\begin{aligned} \Delta F_{ij} &= [F_{ij}^{\text{base}} - F_{ij}^{\text{adjusted}}] \\ &= \sum_k [(M_{ijk})^{\text{base}} - (M_{ijk})^{\text{adjusted}}] \end{aligned}$$

The reduction in accident frequency is the sum of the reductions for each affected accident sequence:

$$\begin{aligned} \Delta F &= \sum_i \sum_j \Delta F_{ij} \\ &= \sum_i \sum_j \sum_k [(M_{ijk})^{\text{base}} - (M_{ijk})^{\text{adjusted}}] \end{aligned}$$

Note that a negative reduction represents an increase in accident frequency from the base to the adjusted case (i.e., an increase resulting from the proposed action).

5.6.1 Identification of Affected Parameters

The level of effort required to identify the parameters affected by implementation of an action depends primarily on the availability of one or more existing power reactor or non-reactor risk/reliability studies which include those parameters. For nuclear power plants, Table 5.2 provides a list of risk studies. The following characteristics are included, as available:

- plant type (BWR/PWR and vendor)
- external events inclusion (yes/no)
- year of commercial operation
- program under which performed (if any)
- level of risk/reliability analysis⁽⁴⁾
- report reference

Table 5.2 Nuclear power plants risk assessments

Plant	Type	Year Commercial	Analysis Level ⁽⁶⁾	External Events?	Program	References
Brunswick-1/2	GE BWRs	1977/75	1	No	Industry Reviewed	April 1988 NUREG/CR-5465 November 1989
Grand Gulf-1	GE BWR	1983	3	No	NUREG-1150	NUREG/CR-4550, V.6, September 1989 Brown et al. 1990
Indian Point-2	W PWR	1974	3	Yes	Industry NRC Report Reviewed Reviewed	PASNY 1982 NUREG/CR-1410 and 1411, August 1980 NUREG/CR-2934, December 1982 NUREG/CR-0850, November 1981
LaSalle County-1	GE BWR	1984	3	Yes	Industry RMIEP, NRC	Call et al. 1985 NUREG/CR-4832, 1992 and 1993
Peach Bottom-2 (Also train level)	GE BWR	1974	3	Yes	NUREG-1150	NUREG/CR-4550, V.4, August 1989 Payne et al. 1990
Sequoyah-1	W PWR	1981	3	No	NUREG-1150	NUREG/CR-4550, V.5, April 1990 Gregory et al. 1990
Surry-1	W PWR	1972	3	Yes	NUREG-1150	NUREG/CR-4550, V3, April 1990 Breeding et al. 1990
Zion-1	W PWR	1973	3	No	NUREG-1150	NUREG/CR-4550, V.7, May 1990 Park et al. 1990
AP-600	W PWR	*			*	Reviewed by NRC 1993
CESAR System 80+	CE PWR	*			*	Reviewed by NRC 1992

* Advanced reactor designs

In addition to the studies shown in Table 5.2, IPE reports covering vulnerabilities to severe accidents and IPEEE reports can serve as additional references. Generic Letter 88-20, issued in November 1988, required all holders of nuclear power plant operating licenses and construction permits to prepare IPE reports. Supplement 4 to General Letter 88-20, issued in July 1991, required these licensees to prepare IPEEE reports. IPE and IPEEE reports are available through the NRC

Public Document Room. The status of the IPE and IPEEE programs is discussed in SECY-96-51 (NRC 1996a) and draft NUREG-1560 (NRC 1996b). NRC staff prepare an evaluation report documenting staff conclusions on each IPE and IPEEE report submitted to NRC (NRC 1996a).

When evaluating generic power reactor issues, where many types of plants may be affected, the five risk assessments performed as part of the NUREG-1150 program (NRC 1991) are particularly useful. One of the primary objectives of that program was to "provide a set of (risk assessment) models and results that can support the ongoing prioritization of potential safety issues and related research" (NRC 1991). As such, these provide a valuable resource for both quantitative and qualitative information on a set of five commercial nuclear power plants of different design.

Several computer codes containing reactor risk assessment information are also available which can be used to support regulatory analyses. Particularly well suited to this type of analysis is the System Analysis and Risk Assessment (SARA) code (Stewart et al. 1989), which contains the dominant accident sequences and cut sets for each of the NUREG-1150 plants. The Integrated Reliability and Risk Analysis System (IRRAS [Russell and Sattison 1988]) is an integrated risk assessment software tool. Using this code, the analyst can create and analyze custom-made fault trees and event trees using a microcomputer.

In addition to these assessments of total plant risk/reliability, some studies focus on specific systems, accident initiators, or accident sequences. For certain actions, such specialized studies may be more appropriate for identifying affected parameters than the various plant-wide assessments.

While risk/reliability assessments have been performed for selected non-reactor facilities, these are generally much less comprehensive than their power reactor counterparts. Available data for accident frequencies at non-reactor facilities have been assembled into composite lists in Section C.2.1.1. They may be used as presented to identify affected parameters in a non-reactor accident equation, or as guides to the more detailed assessments from which they have been extracted.

Additional information sources for non-reactor facility accidents may be found among the numerous Safety Analysis Reports conducted for U.S. Department of Energy (DOE) fuel-cycle facilities. For example, the DOE's Savannah River Site has roughly 30 such reports for fuel fabrication, chemical separation, research laboratories, analytical laboratories, waste handling, irradiated fuel storage, and radioactive material transportation.

At the simplest level, the standard analysis assumes that appropriate risk/reliability studies from which the affected parameters are easily identified are readily available. For example, all currently available reactor risk/reliability studies include accident sequences involving loss of emergency AC power. If the minimal cut sets used in the analytical modeling of these sequences contain parameters appropriate to an action related to loss of emergency AC power, then these risk/reliability studies (supplemented by any new studies published subsequent to this Handbook) would be appropriate for use in the standard analysis. The affected parameters can be readily identified, and the estimation of changes in accident frequency can proceed to the next step (parameter value estimation). Similarly, a major fire accident scenario has been investigated for most non-reactor facilities (see Section C.2.1.1). If a proposed action relates to reducing the fire potential at one or more types of non-reactor facilities, then these risk/reliability studies (supplemented by any new studies published subsequent to this Handbook) would be appropriate for use in the standard analysis. A useful source of data for non-standard events at non-reactor facilities is that maintained at DOE's Savannah River Site (Durant et al. 1988).

At a more detailed level, but still within the scope of a standard analysis, the identification of affected parameters may require more than direct use of existing risk/reliability studies. Existing studies may need to be modified without sacrificing their analytical consistency. The effort may involve performing an expanded or independent analysis of the accident sequences associated with an action, using previous studies only as a guideline, or several existing risk/reliability studies may be combined to form some "composite" study more applicable to a generic action.

Beyond the standard analysis lies the major effort, where identification of affected parameters requires the type of analysis associated with a much greater level of detail and, most likely, a significantly expanded scope. Typical of major efforts are NRC programs related to unresolved power reactor safety issues. Such programs tend to be multi-year tasks conducted by one or more NRC contractors. Clearly, the expected degree of detail and quality of analysis made possible through a major effort to identify affected parameters should be "state-of-the-art," significantly better than could be obtained from the standard effort.

5.6.2 Estimation of Affected Parameter Values

Presumably, the analyst has identified the parameters affected by action implementation. (If not, it is still possible to estimate changes in accident frequencies through expert opinion, discussed as part of the standard analysis.) The next step is to estimate the base- and adjusted-case frequencies/likelihoods of the affected parameters, which are then used to estimate the base- and adjusted-case total accident sequence frequencies. The sum of the differences between the base and adjusted cases is the reduction in accident frequency resulting from the action (a negative reduction is an increase).

At the simplest level, the standard analysis assumes that frequencies/likelihoods for affected parameters are readily available or can be derived easily. The most convenient sources of data are the existing risk/reliability assessments; these provide parameter frequencies/likelihoods in forms appropriate for accident frequency calculations (e.g., frequencies for initiators and unavailabilities or demand failure probabilities for subsequent system/component failures).

For power reactors, NUREG/CR-4639 (Gertman et al. 1988) provides a Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR). Other data sources are available, including the Nuclear Plant Reliability Data System (NPRDS);⁽⁵⁾ and the LERs. These may or may not report data in the forms directly applicable as parameter frequencies/likelihoods. For non-reactor facilities, failure rate data for non-reactor components are available from Dexter and Perkins (1982), Wilkinson et al. (1991), and Blanton and Eide (1993).

The derivation of frequencies/likelihoods from available data should require no more than standard statistical analysis techniques. In addition to statistics textbooks, other sources provide methods for deriving failure rates and probabilities more specifically for use in risk/reliability analyses. McCormick (1981) is a standard reference of this type. If derivation requires more detailed modeling, the analyst should consider the possibility of estimating frequencies/likelihoods through expert opinion. A formalized procedure like the Delphi technique may yield adequate estimates (Dalkey and Helmer 1963; Humphress and Lewis 1982). Also recommended are the "Formal Procedures for Elicitation of Expert Judgment," employed in the NUREG-1150 analyses (NRC 1991) and reviewed in Section 5.4.3.1.

Earlier, it was mentioned that an analyst unable to identify affected parameters for an action can still estimate changes in accident frequency. This removes the need for propagating the effect of change in individual risk parameters through the risk equation to obtain the accident frequency. It involves expert judgment of changes in accident frequency based on the total core-melt frequency of a representative nuclear power plant (although less applicable to the total radioactive release frequency for a non-reactor facility, see below). A formalized procedure like the Delphi method could be used to provide an overall consensus from expert estimates of percent changes in total accident frequency due to action implementation. However, caution is advised, since direct estimation, as compared to more detailed calculations, can result in inaccurate estimates.

Because of the nature of the radioactive material, its multiple locations, and near inconceivability of an accident capable of releasing the total inventory (except, possibly, an "external event"), estimating the frequency of total radioactive release from a non-reactor facility by expert judgment is difficult. It would be more realistic to use the experts to estimate frequencies for individual release locations and initiators.

Expert opinion may also play a prime role in estimating adjusted-case parameter values. Typically, existing data are applied to yield base-case values, leaving only engineering judgment for arriving at adjusted-case values. Consensus can reduce uncertainties, and the magnitudes of parameter values normally encountered in risk/reliability studies can serve as rough guidelines.

At a more detailed level, but still within the scope of a standard analysis, the analyst would be expected to conduct reasonably detailed statistical modeling or extensive data compilation when frequencies/likelihoods for affected parameters are not readily available. While existing risk/reliability assessments may provide some data for use in statistical modeling, the level of detail required would normally be greater than they could provide. Statistical modeling may use stochastic simulation methods and may also involve relatively basic statistical analysis techniques using extensive data.

Beyond the standard analysis lies the major effort, where estimation of affected parameter values requires much greater detail and a significantly expanded scope. When frequencies/likelihoods are unavailable for affected parameters, a major analytical effort is required. The analyst may need to develop specialized statistical models or possibly seek experimental data. On the other hand, data may be so abundant as to require extensive statistical analysis to produce a more workable base. Typically, both detailed statistical modeling and extensive data compilation will be required as part of a major effort. "State-of-the-art" data analysis techniques should be employed.

Estimation of adjusted-case affected parameter values should involve more than just expert opinion for a major effort. Engineering judgment can be incorporated into an overall framework, but this framework should be analytical, not judgmental. If the need for expert opinion proves inevitable, only a rigorous application of the Delphi or other such methods will suffice for a major effort.

5.6.3 Change in Accident Frequency

The change in accident frequency is a key factor for several of the value-impact analysis attributes. Having identified base- and adjusted-case values for the parameters in the plant risk equation affected by the proposed regulatory action, the analyst proceeds to calculate the reduction in accident frequency as the sum of the differences between the base- and adjusted-case values for all affected accident sequences. Section 5.6 presented this calculation in the format of an equation. Reduction in accident frequency is algebraically positive; increase is negative (viewed as a negative reduction).

An error factor⁽⁶⁾ of at least five (typical for a 90% confidence level) on the best estimate of the reduction in total accident frequency may be used to estimate high and low values for the sensitivity calculations in a standard analysis for power reactor facilities. If no better information is available, higher error factors (at least 10) can be assumed for non-reactor standard analyses. If better values are known (e.g., error factors from the specific risk assessment being used), they should be employed. Rigorously derived error factors via computer simulation would be appropriate for a major analysis beyond the standard scope.

NUREG/CR-2800 (Andrews et al. 1983) provides a useful conceptual discussion on the calculation of change in core-melt accident frequency for power reactors, along with detailed examples. Such calculations would be typical of what is expected to be appropriate in the standard value-impact analysis portion of a regulatory analysis.

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for the change in accident frequency.

5.7 Quantification of Attributes

The following sections provide specific guidance in estimating the values of each attribute. However, before looking at specific attributes, there are several generic concepts that need to be explored.

Value and impact estimates are performed relative to a baseline case, which is typically the no-action alternative. In establishing the baseline case, an assumption should be made that all existing NRC and Agreement State requirements and written licensee commitments are already being implemented and that values and impacts associated with these requirements are not part of the incremental estimates prepared for the regulatory analysis. Similarly, the effects of formally proposed concurrent regulatory actions that are viewed as having a high likelihood of implementation need to be incorporated into the baseline before calculating the incremental consequences of the regulatory action under consideration.

The treatment of voluntary incentives on the part of industry also has important implications on the baseline and therefore, the incremental consequences of the proposed action. Section 4.3 of the NRC Guidelines discusses the treatment of voluntary activities by affected licensees when establishing a baseline reference. Basically, analysts should give no credit for voluntary actions in making base case estimates. However, for completeness and sensitivity analysis purposes, the analyst should also display results with credit being given for voluntary actions by licensees.

Section 4.3 of the NRC Guidelines requires the use of best estimates. Often these are evaluated in terms of the mean or "expected value," the product of the probability of some event occurring and the consequences which would occur assuming the event actually happens. Sometimes, measures other than the expected value may be appropriate, such as the median or even a point estimate. However, the expected value is generally preferred.

There are four attributes used in value-impact analysis for which expected value is normally calculated: public health (accident), occupational health (accident), offsite property, and onsite property. All four of these attributes usually rely on estimation of the change in probability of occurrence of an accident as a result of implementation of the proposed action. (Changes in the consequence of the accident [i.e., dose or cost] would also affect these attributes.)

Four attributes involve radiation exposure: 1) public health (accident), 2) public health (routine), 3) occupational health (accident), and 4) occupational health (routine). In quantifying each measure, the analyst should assess the reduction (or risk averted) relative to the existing condition. For accident-related exposures, the measure will be probabilistically weighted (i.e., the potential consequence is multiplied by its probability of occurrence).⁽⁷⁾ The non-accident terms (e.g., routine occupational exposure) are given in terms of annual expected effect. Both types of terms would be integrated over the lifetime of the affected facilities to show the total effect. Each of the attributes involving radiation exposure can be characterized in terms of person-rems, either averted by or resulting from implementation of the proposed action.

The four attributes associated with radiation exposure require a person-rem-to-dollars conversion factor to be expressed monetarily (see Section 5.7.1.2). The remaining quantitative attributes are normally quantified monetarily in a direct manner. When quantified monetarily, attributes should be discounted to present value (see Section B.2 for a discussion of discounting techniques). This operation involves an assumption regarding the remaining lifetime of a facility. If appropriate, the effect of license renewal should be included in the facility lifetime estimate (see Section 4.3 of the Guidelines). The total dollar figures capture both the number of facilities involved (in the case of generic rulemaking) and the economic lifetime of the affected facilities.

Based on OMB's guidance in Circular A-94, Section 4.3.3 of the Guidelines requires that a 7% real (i.e., inflation-adjusted) discount rate be used for a best estimate. For sensitivity analysis, the Guidelines recommend a 3% discount rate. However, for certain regulatory actions involving a timeframe exceeding 100 years (e.g., decommissioning and waste disposal issues), Section 4.3.3 of the Guidelines stipulates the following:

...[T]he regulatory analysis should display results to the decision-maker in two ways. First, on a present worth basis using a 3 percent real rate, and second, by displaying the values and impacts at the time in which they are incurred with no present worth conversion. In this latter case, no calculation of the resulting net value... should be made.

"Qualitative" attributes do not lend themselves to quantification. To the degree to which the considerations associated with these attributes can be quantified, they should be; the quantification should be documented, preferably under one or more of the quantitative attributes. However, if the consideration does not lend itself to any level of quantification, then its treatment should take the form of a qualitative evaluation in which the analyst describes as clearly and concisely as possible the precise effect of the proposed action.

To estimate values for the accident-related attributes in a regulatory analysis, the analyst ideally can draw from detailed risk/reliability assessments or statistically-based analyses. Numerous sources exist for power reactor applications (e.g., see Section 5.6). To a lesser extent, Sections C.3-C.6 and C.10 provide similar data for non-reactor applications. Most regulatory analyses for power reactor facilities are based on detailed risk/reliability assessments or equivalent statistically based analyses.

However, the analyst will sometimes find limited factual data or information sufficiently applicable only for providing a quantitative perspective, possibly requiring extrapolation. These may often involve non-reactor licensees since detailed risk/reliability assessments and/or statistically-based analyses are less available than for power reactor licensees. Two examples illustrate this type of quantitative evaluation.

In 1992, the NRC performed a regulatory analysis for the adoption of a proposed rule (57 FR 56287; November 27, 1992) concerning air gaps to avert radiation exposure resulting from NRC-licensed users of industrial gauges. The NRC found insufficient data to determine the averted radiation exposure. To estimate the reduction in radiation exposure should the rule be adopted, the NRC proceeded as follows. The NRC assumed a source strength of one curie for a device with a large air gap, which produces 1.3 rem/hr at a distance of 20 inches from a Cs-137 source. Assuming half this dose rate would be produced, on average, in the air gap, and that a worker is within the air gap for four hours annually, the NRC estimated the worker would receive 2.6 rem/yr. The NRC estimated that adopting the proposed air-gap rule would be cost-effective if 347 person-rem/yr were saved. At the estimated average savings of 2.6 person-rem/yr for each gauge licensee, incidents involving at least 133 gauges would have to be eliminated. Given the roughly 3,000 gauges currently used by these licensees, the proposed rule would only have to reduce the incident rate by roughly 4%, a value the NRC believed to be easily achievable. As a result, the NRC staff recommended adoption of the air-gap rule.

In 1992, the NRC responded to a petition from General Electric (GE) and Westinghouse for a rulemaking to allow self-guarantee as an additional means for compliance with decommissioning regulations. An NRC contractor estimated the default risks of various types of financial assurance mechanisms, including the proposed self-guarantee. The contractor had to collect data on failure rates both of firms of different sizes and of banks, savings and loans, and other suppliers of financial assurance mechanisms. The contractor estimated a default risk of 0.13% annually for the GE-Westinghouse proposal, with a maximum default risk of only 0.055% annually for third-party guarantors, specifically a small savings and loan issuing a letter of credit. Based on these findings, the NRC initiated a proposed rulemaking which would allow self-guarantee for certain licensees. The final rule was issued December 29, 1993 (58 FR 68726).

Additional examples of this more limited type of quantitative approach to estimation can be found in Sections C.8 and C.9.

5.7.1 Public Health (Accident)

Evaluating the effect on public health from a change in accident frequency due to proposed regulatory actions is a multi-step process. For each affected facility, the analyst first estimates the change in the public health (accident) risk associated with the action and reports this as person-rem avoided exposure. Reduction in public risk is algebraically positive; increase is negative (viewed as a negative reduction). Next the analyst converts person-rem to their monetary equivalent (dollars) and discounts to present value. Finally, the analyst totals the change in public health (accident) as expressed in discounted dollars over all affected facilities.

The steps are as follows:

1. Estimate reduction in accident frequency per facility (see Section 5.6).
2. Estimate reduction in public health (accident) risk per facility (see Section 5.7.1.1).
3. Convert value of public health (accident) risk avoided (person-rem) per facility to monetary equivalent (dollars) via monetary valuation of health effects (see Section 5.7.1.2).

$$Z_{PHA} = RD_{PA}$$

where Z_{PHA} = monetary value of public health (accident) risk avoided per facility-year before discounting (\$/facility-year)

D_{PA} = avoided public dose per facility-year (person-rem/facility-year)

R = monetary equivalent of unit dose (\$/person-rem).

4. Discount to present value per facility (dollars) (see Section 5.7.1.3).
5. Total over all affected facilities (dollars).

$$V_{PHA} = NW_{PHA}$$

where V_{PHA} = discounted monetary value of public health (accident) risk avoided for all affected facilities (\$)

W_{PHA} = monetary value of public health (accident) risk avoided per facility after discounting (\$/facility)

N = number of affected facilities.

If individual facility values rather than generic values are used, the formulations can be replaced with

$$V_{PHA} = \sum_i N_i W_{PHA_i}$$

where i = facility (or group of facilities) index.

5.7.1.1 Estimation of Accident-Related Health Effects

The results of the formulations given in Section 5.6 are reductions in accident frequency. These form the first portion of the public health (accident) risk estimate. For the standard analysis, the analyst would employ data developed in existing risk studies which include offsite effects, if possible. Such studies provide population dose factors that can be applied to accident release categories to yield dose estimates as follows:

$$\text{Avoided Public Dose } [D_{PA}] \text{ (person-rem/facility-yr)} = \sum_{\text{Release Categories}} \left[\text{Reduction in Release Category Frequency (events/facility-yr)} \right] \times \left[\text{Population Dose Factor for Release Category (person-rem/event)} \right]$$

If the risk assessment being used by the analyst to estimate public health (accident) employs its own unique accident release categories with corresponding population dose factors, then these should be used. Otherwise, population dose factors should be based on Table 5.3 (see Appendix B.4 for development of this table). For non-reactor accidents, population dose factors for accident scenarios at selected facilities have been assembled into composite lists in Section C.2.1.2. An error factor of at least five is considered appropriate for use in sensitivity studies.

Table 5.3 Expected population doses for power reactor release categories

Plant Type	Release Category	Accident Progression Characteristics						Population Dose	
		CF Time	PDS	SP Bypass	RB Bypass	CCI	CF Mode	Total (Person-Rem)	% Long Term
PWR	RSUR1	CFatVB	LOSP	Not Applicable		Dry	Rupture	6.15E+6	63
	RSUR2	Late CF					Leak	2.30E+6	88
	RSUR3	No CF					No CF	2.50E+2	67
	RSUR4	Bypass					Bypass	4.29E+6	80
	RZ1	CFatVB	LOCA			Shallow	Rupture	5.77E+6	65
	RZ2	LateCF				Flooded	Leak	1.31E+5	38
	RZ3	No CF				No CF	3.31E+2	67	
	RZ4	Bypass	Bypass			Dry	Bypass	4.80E+6	76
	RSEQ1	CFdurCD	LOSP			Dry	CatRup	1.31E+7	50
	RSEQ2	CFatVB						5.77E+6	56
	RSEQ3	Late CF	LOCA			Flooded	Rupture	1.33E+5	42
	RSEQ4	No CF					No CF	4.06E+2	71
	RSEQ5	Bypass				Bypass	Dry	Bypass	4.94E+6
	BWR	RPB1	CFatVB			LOSP	Early/Late	Sm/None	Dry
RPB2		ATWS		5.32E+6					
RPB3		CFdurCD	None	Large	WWvent	3.26E+6	84		
RPB4		Late CF	Early/Late		DWRup	1.13E+6	92		
RPB5		No CF	LOSP	None	Sm/None	Shallow	No CF	8.27E+3	62
RPB6		CFatVB	Early/Late	Large	Dry	DWMth	1.11E+7		
RLAS1		CFdurCD	Tran	Early/Late	Sm/None	Dry	WWawrup	5.25E+6	80
RLAS2		CFatVB				Shallow	WWaw-lk	3.21E+6	81
RLAS3						DWRup	4.66E+6	82	
RLAS4		CFdurCD				Dry	WWvent	5.92E+6	73
RLAS5		Late CF				Sm/None		Shallow	1.75E+6
RLAS6						Large	Dry	CF-Ped	4.18E+6

Table 5.3 (Continued)

Plant Type	Release Category	Accident Progression Characteristics						Population Dose	
		CF Time	PDS	SP Bypass	RB Bypass	CCI	CF Mode	Total (Person-Rem)	% Long Term
BWR	RLAS7	No CF	Tran	None	Sm/None	Shallow	No CF	3.33E+2	65
	RGG1	CFatVB	STSB	Early/Late	Large	Flooded	Rupture	5.77E+6	75
	RGG2	CFdurCD		None				2.74E+6	90
	RGG3	Late CF		Late Only				2.35E+6	80
	RGG4	CFdurCD		Early/Late				2.70E+6	93
	RGG5	No CF		None				No CCI	No CF

Note: The initials RSUR, RZ, and RSEQ refer to Surry, Zion, and Sequoyah release categories respectively followed by the release category number. The initials RPB, RLAS, and RGG refer to Peach Bottom, LaSalle, and Grand Gulf release categories respectively followed by the release category number.

Key:

- CF Time = Containment failure (CF time)
- CFatVB = CF at vessel breach (VB)
- CFdurCD = CF during core damage (before VB, if it occurs)
- LateCF = CF during core concentration interactions (CCI)
- No CF = no CF
- Bypass = bypass of containment (usually throughout duration of accident)
- PDS = Plant damage state (PDS)
- LOSP = loss of offsite power
- LOCA = loss of coolant accident
- Bypass = bypass of containment (interfacing systems LOCA or steam generator tube rupture)
- ATWS = anticipated transient without scram
- Tran = Transient
- STSB = short-term station blackout
- CCI = Type of molten core concrete interactions (CCI)
- Dry = CCI occurs in a dry cavity
- Shallow = CCI occurs in a wet cavity (nominally 5 ft. of water)
- Flooded = CCI occurs in a deeply flooded cavity (nominally 14 ft. of water)
- No CCI = There is no CCI (the debris bed is coolable with replenishable water or no VB)
- CF Mode = Containment failure mode
- CatRup = Catastrophic rupture failure
- Rupture = Rupture failure of containment
- Bypass = bypass of containment
- Leak = Leak failure of containment
- No CF = no CF
- WWawrup = Rupture above the wetwell water level
- WWawlk = Leak above the wetwell water level
- DWRup = Rupture in the drywell
- WWvent = Venting of the wetwell
- CF-Ped = Rupture in the drywell wall, caused by late failure of the reactor pedestal
- DWMth = Melt-through of the drywell wall by direct contact of the molten core
- SP Bypass = Suppression pool (SP) bypass
- Early/Late = SP is bypassed from the time of VB throughout the accident
- None = SP is never bypassed
- Late Only = SP is only bypassed late in the accident (during CCI)
- RB Bypass = Reactor building (RB) bypass
- Sm/None = Nominal or small leakage from the RB
- Large = Large leakage from the RB or bypass of the RB (for Grand Gulf, all containment failures were assumed to be above the RB)

Should the nature of the issue require that the reduction in accident frequency be expressed as a single number, a single population dose factor, preferably one that has been probabilistically weighted to reflect those for all accident release categories, is generally needed. For this approach, the calculation of avoided public dose becomes:

$$\text{Avoided Public Dose } [D_{PA}] \text{ (person-rem/facility-yr)} = \left[\frac{\text{Reduction in Accident Frequency (events/facility-yr)}}{\text{Population Dose Factor (person-rem/event)}} \right] \times \left[\text{Population Dose Factor (person-rem/event)} \right]$$

Mubayi et al. (1995) have calculated population doses weighted by the frequencies of the accident release categories for the five power reactors analyzed in NUREG-1150 (NRC 1991). These are listed in Table 5.4 based on Version 1.5.11.1 of the MACCS computer code (Chanin et al. 1993). The population doses have been calculated as the sum of those for emergency response and long-term protective action, defined as follows:

- For early consequences, an effective emergency response plan consisted of evacuation of all but 0.5% of the population within a ten-mile radius at a specified speed and delay time following notification of the emergency.
- For long-term relocation and banning of agricultural products, the interdiction criterion was 4 rem to an individual over five years (2 rem in year one, followed by 0.5 rem each successive year).

For regulatory analyses involving nuclear power plants, doses should be estimated over a 50-mile radius from the plant site (see Guidelines Section 4.3.1). Doses for other distances can be considered in sensitivity analyses or special cases, and are available in Mubayi et al. (1995).

It is possible that the proposed action will affect public health (accident) through a mitigation of consequences instead of (or as well as) through a reduction in accident frequency.⁽⁸⁾ Should this be the case, the previous general formulations are replaced with the following:

$$\text{Avoided Public Dose} = \sum_{\text{Release Categories}} \left[\frac{\text{Release Category}}{\text{Frequency}} \times \frac{\text{Category Population}}{\text{Dose Factor}} \right]_{\text{Status Quo}} - \sum_{\text{Release Categories}} \left[\frac{\text{Release Category}}{\text{Frequency}} \times \frac{\text{Category Population}}{\text{Dose Factor}} \right]_{\text{After Action}}$$

or

$$\text{Avoided Public Dose} = \left[\frac{\text{Accident}}{\text{Frequency}} \times \frac{\text{Population Dose}}{\text{Factor}} \right]_{\text{Status Quo}} - \left[\frac{\text{Accident}}{\text{Frequency}} \times \frac{\text{Population Dose}}{\text{Factor}} \right]_{\text{After Action}}$$

Table 5.4 Weighted population dose factors for the five NUREG-1150 power reactors

Reactor	Type	Person-rem Within 50 miles from the Plant
Zion	PWR	1.95E+5
Surry	PWR	1.60E+5
Sequoyah	PWR	2.46E+5
Peach Bottom	BWR	2.00E+6
Grand Gulf	BWR	1.93E+5
Average		1.99E+5

Public risks from non-reactor accidents have been assembled into composite lists in Section C.2.1.3. These represent the products of accident frequencies and population dose factors, whether calculated as release category summations or single frequency and dose numbers.

Beyond the standard analysis lies the major effort. In parallel with the more involved effort to identify and quantify affected parameters in appropriate accident sequences (see Section 5.6) would be an equivalent effort to quantify population dose factors and possibly even specific health effects. Such effort at the "consequence end" of the risk calculation would increase the likelihood of obtaining representative results. Non-representative results can arise through the use of inappropriate or inapplicable dose calculations just as readily as through inappropriate logic models and failure data.

Several computer codes exist for estimation of population dose. Most for reactor applications have been combined under MACCS (Chanin et al. 1990, 1993; Summers et al. 1995a,b). Three codes for non-reactor applications are GENII (Napier et al. 1988), CAP-88 (Beres 1990), and COMPLY (EPA 1989). There have also been recent upgrades to MELCOR itself for modeling severe accidents in light water reactors, including estimation of severe accident source terms and their sensitivities/uncertainties (Summers et al. 1995a,b).

The GENII code package determines individual and population radiation doses on an annual basis, as dose commitments, and as accumulated from acute or chronic radionuclide releases to air or water. It has an additional capability to predict very-long-term doses from waste management operations for periods up to 10,000 years.

The CAP-88 code package is generally required for use at DOE facilities to demonstrate compliance with radionuclide air emission standards where the maximally exposed offsite individual is more than 3 km from the source [40 CFR 61.93(a)]. The code contains modules to estimate dose and risk to individuals and populations from radionuclides released to the air. It comes with a library of radionuclide-specific data and provides the most flexibility of the EPA air compliance codes in terms of ability to input site-specific data. A personal computer version of the CAP-88 code package (Parks 1992) was released in March 1992 under the name CAP88-PC and is also approved for demonstrating compliance at DOE facilities.

The COMPLY code is a screening model intended primarily for use by NRC licensees and federal agencies other than DOE facilities. It is approved for use by DOE facilities where the maximally exposed offsite individual is less than 3 km from the emissions source [40 CFR 61.93(a)]. The code consists of four screening levels, each of which requires increasingly detailed site-specific data to produce a more realistic (and less conservative) dose estimate. COMPLY runs on a personal computer and does not require extensive site-specific data.

5.7.1.2 Monetary Valuation of Accident-Related Health Effects

Section 4.3.3 of the Guidelines states that the conversion factor to be used to establish the monetary value of a unit of radiation exposure is \$2000 per person-rem. This value will be subject to periodic NRC review. The basis for selection of the \$2000 per person-rem value is set out in NUREG-1530 (NRC 1995d). The \$2000 per person-rem value is to be used for routine and accidental emissions for both public and occupational exposure. Unlike past NRC practice, offsite property consequences are to be separately valued and are not part of the \$2000 per person-rem value. Monetary conversion of radiation exposure using the \$2000 per person-rem value is to be performed for the year in which the exposure occurs and then discounted to present value for purposes of evaluating values and impacts.

5.7.1.3 Discounting Monetized Value of Accident-Related Health Effects

The present value for accident-related health effects in their monetized form can be calculated as follows:

$$W_{PHA} = C \times Z_{PHA}$$

where W_{PHA} = monetary value of public health (accident) risk avoided per facility after discounting (\$/facility)
 C = $[\exp(-rt_i) - \exp(-rt_f)]/r$
 t_f = years remaining until end of facility life
 t_i = years before facility begins operating
 r = real discount rate (as fraction, not percent)
 Z_{PHA} = monetary value of public health (accident) risk avoided per facility-year before discounting (\$/facility-year).

If a facility is already operating, t_i will be zero and the equation for C simplifies to

$$C = [1 - \exp(-rt_f)]/r$$

Should public health (accident) risk not be discounted in an analysis, r effectively becomes zero in the preceding equations. In the limit as r approaches zero, $C = t_f - t_i$ (or, $C = t_f$ when $t_i = 0$). This new value of C should be used to evaluate W_{PHA} in the undiscounted case.

The quantity W_{PHA} must be interpreted carefully to avoid misunderstandings. It does not represent the expected reduction in public health (accident) risk due to a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime of the facility. Thus, it reflects the expected annual loss due to a single accident (this is given by the quantity Z_{PHA}); the possibility that such an accident could occur, with some small probability, at any time over the remaining facility life; and the effects of discounting these potential future losses to present value. Since the quantity Z_{PHA} only accounts for the risk of an accident in a representative year, the result is the expected loss over the facility life, discounted to present value.

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for the public health (accident) attribute.

5.7.2 Public Health (Routine)

As with the public health (accident), the evaluation of the effect on public health from a change in routine exposure due to proposed regulatory actions is a multi-step process. Reduction in exposure is algebraically positive; increase is negative (viewed as a negative reduction).

The steps are as follows:

1. Estimate reductions in public health (routine) risk per facility for implementation (D_{PRI}) and operation (D_{PRO}) (see Section 5.7.2.1).
2. Convert each reduction in public health (routine) risk per facility from person-rem to dollars via monetary evaluation of health effects (see Section 5.7.2.2):

$$G_{PRI} = RD_{PRI} \qquad G_{PRO} = RD_{PRO}$$

where G_{PRI} = monetary value of per-facility reduction in routine public dose required to implement the proposed action, before discounting (\$/facility)

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- G_{PRO} = monetary value of annual per-facility reduction in routine public dose to operate following implementation of the proposed action, before discounting (\$/facility-year)
- D_{PRI} = per-facility reduction in routine public dose required to implement the proposed action (person-rem/facility)
- D_{PRO} = annual per-facility reduction in routine public dose to operate following implementation of the proposed action (person-rem/facility-year)
- R = monetary equivalent of unit dose (\$/person-rem).

3. Discount each reduction in public health (routine) risk per facility (dollars) [see Section B.2].
4. Sum the reductions and total over all facilities (dollars):

$$V_{\text{PHR}} = N(H_{\text{PRI}} + H_{\text{PRO}})$$

- where V_{PHR} = discounted monetary value of reduction in public health (routine) risk for all affected facilities (\$)
- H_{PRI} = monetary value of per-facility reduction in routine public dose required to implement the proposed action, after discounting (\$/facility)
- H_{PRO} = monetary value of per-facility reduction in routine public dose to operate following implementation of the proposed action, after discounting (\$/facility)
- N = number of affected facilities.

Note the algebraic signs for D_{PRI} and D_{PRO} . A reduction in exposure is positive; an increase is negative. The dose for implementation (D_{PRI}) would normally be an increase and therefore negative.

If individual facility values rather than generic values are used, the formulations can be replaced with

$$V_{\text{PHR}} = \sum_i N_i (H_{\text{PRI}_i} + H_{\text{PRO}_i})$$

where i = facility (or group of facilities) index.

5.7.2.1 Estimation of Change in Routine Exposure

A proposed NRC action can affect routine public exposures in two ways. It may cause a one-time increase in routine dose due to implementation of the action (e.g., installing a retrofit). It may also cause a change (either increase or decrease) in the recurring routine exposures after the action is implemented.⁽⁹⁾ For the standard analysis, the analyst may attempt to make exposure estimates, or obtain at least a sample of industry or community data for a validation of the estimates developed. Baker (1995) provides estimates of population and individual dose commitments for reported radionuclide releases from commercial power reactors operated during 1991. Tichler et al. (1995) have compiled and reported releases of radioactive materials in airborne and liquid effluents from commercial Light Water Reactors (LWRs) during 1993. Data on solid waste shipments are also included. This report is updated annually. Routine public risks for non-reactor facilities have been assembled into composite lists in Section C.2.2.

5.7.2.2 Monetary Valuation of Routine Exposure

As with public health (accident) (Section 5.7.1.2), monetary valuation for public health (routine) employs the value of \$2,000/person-rem as the best estimate of the monetary conversion factor (R).

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for the public health (routine) attribute.

5.7.3 Occupational Health (Accident)

Evaluating the effect on occupational health from a change in accident frequency due to proposed regulatory actions is a multi-step process. Reduction in occupational risk is algebraically positive; increase is negative (viewed as a negative reduction).

The steps are as follows:

1. Estimate reduction in accident frequency per facility (see Section 5.6).
2. Estimate reduction in occupational health (accident) risk per facility due to the following (see Section 5.7.3.1):
 - "immediate" doses
 - long-term doses
3. Per facility, convert value of occupational health (accident) risk avoided (person-rem) to monetary equivalent (dollars) via monetary evaluation of health effects, due to the following (see Section 5.7.3.2):
 - "immediate" doses $Z_{IO} = RY_{IO}$
 - long-term doses $Z_{LTO} = RY_{LTO}$

where Z_{IO} = monetary value of occupational health (accident) risk avoided per facility-year due to "immediate" doses, before discounting (\$/facility-year)
 Z_{LTO} = monetary value of occupational health (accident) risk avoided per facility-year due to long-term doses, before discounting (\$/facility-year)
 Y_{IO} = avoided occupational "immediate" dose per facility-year (person-rem/facility-year)
 Y_{LTO} = avoided occupational long-term dose per facility-year (person-rem/facility-year)
 R = monetary equivalent of unit dose (\$/person-rem).

4. Discount to present value per facility (dollars) (see Section 5.7.3.3).
5. Total over all affected facilities (dollars) using

$$V_{OHA} = N(W_{IO} + W_{LTO})$$

where V_{OHA} = discounted monetary value of occupational health (accident) risk avoided for all affected facilities
 W_{IO} = monetary value of occupational health (accident) risk avoided per facility due to "immediate" doses, after discounting (\$/facility)
 W_{LTO} = monetary value of occupational health (accident) risk avoided per facility due to long-term doses, after discounting (\$/facility)
 N = number of affected facilities.

Value-Impact

If individual facility values rather than generic values are used, the formulations can be replaced with

$$V_{\text{OHA}} = \sum_i N(W_{\text{IO}_i} + W_{\text{LTO}_i})$$

where i = facility (or group of facilities) index.

5.7.3.1 Estimation of Accident-Related Exposures

There are two types of occupational exposure related to accidents: "immediate" and long-term. The first occurs at the time of the accident and during the immediate management of the emergency. The second is a long-term exposure, presumably at significantly lower individual rates, associated with the cleanup and refurbishment or decommissioning of the damaged facility. The value gained in the avoidance of both types of exposure must be conditioned on the change in frequency of the accident's occurrence (see Section 5.6).⁽¹⁰⁾

"Immediate" Doses

Licensing of nuclear facilities requires the license applicant to consider and attempt to minimize occupational doses. Radiation protection in a reactor control room is required to limit dose to 5-rem whole body under accident conditions (10 CFR 50, Appendix A, Criterion 19). The experience at the Three Mile Island (TMI) Unit 2 nuclear power plant indicated that potential for significant occupational exposures exists for activities outside the control room during a power reactor accident. (However, there was no individual occupational exposure exceeding 5-rem whole body at TMI-2.)

For the standard analysis specifically applied only to power reactor facilities, the analyst may employ the TMI or Chernobyl experience. At TMI, the average occupational exposure related to the incident was approximately 1 rem. A collective dose of 1,000 person-rem could be attributed to the accident. This occurred over a 4-month span, after which time occupational exposure was approaching pre-accident levels. An upper estimate for sensitivity analysis is obtained by assuming that the average individual receives a dose equal to that of the maximum individual dose at TMI. The ratio of maximum to average dose for TMI is 4.2 rem/1 rem; therefore, the upper estimate for the collective dose can be taken as 4,200 person-rem. A lower estimate of zero indicates a case where no increase over the normal occupational dose occurs.

The DOE (1987) summarized results on the collective dose received by the populace surrounding the Chernobyl accident. Average dose equivalents of 3.3 rem/person, 45 rem/person, and 5.3 rem/person were estimated for residents within 3 km, between 3 km and 15 km, and between 15 km and 30 km of Chernobyl, respectively (Mubayi et al. 1995, p. A-5). Although none of these translates readily into an occupational dose as that for TMI, a reasonable, but conservative, assumption would be that the average worker received the average dose for persons closest to the plant (i.e., 3.3 rem/person). For 1,000 workers, an average value of 3,300 person-rem is obtained, about three times that estimated for TMI-2. Given the greater severity of the Chernobyl accident, this seems reasonable. Using TMI's ratio of 4.2/1 for the maximum, an upper bound of 14,000 person-rem results. TMI's average value of 1,000 person-rem would appear to be a reasonable lower bound for Chernobyl.

Given the uncertainties in existing data and variability in severe accident parameters and worker response, the following is suggested as D_{10} (immediate occupational dose) specifically for power reactor accidents:

Best estimate:	3,300 person-rem
High estimate:	14,000 person-rem
Low estimate:	1,000 person-rem

For a major effort beyond the standard analysis, specific calculations to estimate onsite exposures for various accidents could be performed.

Long-Term Doses

After the immediate response to a major power reactor accident, a long process of cleanup and refurbishment or decommissioning will follow. Significant occupational dose will result (individual exposures controlled by normal occupational dose guidelines). The values for the standard analysis specifically applied only to power reactors are based on a study (Murphy and Holter 1982) of decommissioning a reference LWR following postulated accidents. Table 5.5 summarizes the occupational doses estimated by the study and is presented for perspective.

Since this Handbook focuses on avoidance of major large-scale accidents, use of the following long-term doses based on Murphy and Holter (1982) is suggested specifically for power reactor accidents.

D_{LTO} (long-term occupational):

Best estimate: 20,000 person-rem
 High estimate: 30,000 person-rem
 Low estimate: 10,000 person-rem

Table 5.5 Estimated occupational radiation dose from cleanup and decommissioning after a power reactor accident (person-rem or person-cSv)

Activity	Accident Scenario 1 ^(a)	Accident Scenario 2 ^(b)	Accident Scenario 3 ^(c)
Cleanup	670	4,580	12,100
Dismantlement and Decommissioning	<u>1,230</u>	<u>3,060</u>	<u>7,660</u>
Total	1,900	7,640	19,760

- (a) Accident Scenario 1 - a small Loss of Coolant Accident (LOCA) in which Emergency Core Cooling System (ECCS) functions as intended. Some fuel cladding ruptures, but no fuel melts. The containment building is moderately contaminated, but there is minimal physical damage.
- (b) Accident Scenario 2 - a small LOCA in which ECCS is delayed. Fifty percent of the fuel cladding ruptures, and some fuel melts. The containment building is extensively contaminated, but there is minimal physical damage. (This scenario is presumed to simulate the TMI-2 accident.)
- (c) Accident Scenario 3 - a major LOCA in which ECCS is delayed. All fuel cladding ruptures, and there is significant fuel melting and core damage. The containment building is extensively contaminated and physically damaged. The auxiliary building undergoes some contamination.

Avoided Doses

To calculate the avoided accident-related occupational exposures, both the "immediate" and long-term occupational dose are multiplied by the reduction in accident frequency (see Section 5.6) which is postulated as a result of the proposed action.

$$Y_{IO} = \Delta F D_{IO} \quad Y_{LTO} = \Delta F D_{LTO}$$

where ΔF = reduction in accident frequency (events/facility-year)
 Y_{IO} = avoided occupational "immediate" dose per facility-year (person-rem/facility-year)
 D_{IO} = immediate occupational dose
 Y_{LTO} = avoided occupational long-term dose per facility-year (person-rem/facility-year)
 D_{LTO} = long-term occupational dose.

It is possible that the proposed action will mitigate accident-related occupational exposures instead of (or as well as) reducing the accident frequency. In any case, it is the change from current condition to that following implementation of the proposed action that is sought. The formulation above can be replaced with the more explicit formulation below:

$$Y_{IO} = (FD_{IO})_S - (FD_{IO})_A$$
$$Y_{LTO} = (FD_{LTO})_S - (FD_{LTO})_A$$

where F = accident frequency (events/facility-year)
 S = status quo (current conditions)
 A = after implementation of proposed action.

Occupational risks from non-reactor accidents have been assembled into composite lists for selected non-reactor facilities in Section C.2.3. As for the public risks from non-reactor accidents, these also represent the products of accident frequencies and dose factors.

5.7.3.2 Monetary Valuation of Accident-Related Exposures

The analyst should use the \$2000 per person-rem conversion value discussed in Section 5.7.1.2 for the monetary valuation of accident-related exposures.

5.7.3.3 Discounting Monetized Values of Accident-Related Exposures

The present values for "immediate" and long-term accident-related exposures in their monetized forms are calculated in slightly different ways.

"Immediate" Doses

For "immediate" doses, the present value is

$$W_{IO} = C \times Z_{IO}$$

where W_{10} = monetary value of occupational health (accident) risk avoided per facility due to "immediate" doses, after discounting (\$/facility)
 $C = [\exp(-rt_i) - \exp(-rt_f)]/r$
 t_f = years remaining until end of facility life
 t_i = years before facility begins operating
 r = real discount rate (as fraction, not percent)
 Z_{10} = monetary value of occupational health (accident) risk avoided per facility-year due to "immediate" doses, before discounting (\$/facility-year).

If a facility is already operating, t_i will be zero and the equation for C simplifies to

$$C = [1 - \exp(-rt_f)]/r$$

Should occupational health (accident) risk due to "immediate" doses not be discounted in an analysis, r effectively becomes zero in the preceding equations. In the limit as r approaches zero, $C = t_f - t_i$ (or, $C = t_f$ when $t_i = 0$). This new value of C should be used to evaluate W_{10} in the undiscounted case.

The quantity W_{10} must be interpreted carefully to avoid misunderstandings. It does not represent the expected reduction in occupational health (accident) risk due to "immediate" doses as the result of a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime of the facility. Thus, it reflects the expected annual loss due to a single accident (this is given by the quantity Z_{10}); the possibility that such an accident could occur, with some probability, at any time over the remaining facility life; and the effects of discounting these potential future losses to present value. Since the quantity Z_{10} only accounts for the risk of an accident in a representative year, the result is the expected loss over the facility life, discounted to present value.

Long-Term Doses

For long-term doses, the present value is

$$W_{LTO} = \left[\frac{Z_{LTO}}{mr^2} \exp(-rt_i) \right] \left[1 - \exp\{-r(t_f - t_i)\} \right] [1 - \exp(-rm)]$$

where W_{LTO} = monetary value of occupational health (accident) risk avoided per facility due to long-term doses, after discounting (\$/facility)
 m = years over which long-term doses accrue⁽¹⁾
 r = real discount rate (as fraction, not percent)
 t_f = years remaining until end of facility life
 t_i = years before facility begins operating
 Z_{LTO} = monetary value of occupational health (accident) risk avoided per facility-year due to long-term doses, before discounting (\$/facility-year).

If the facility is already operating, t_i will be zero and the equation for W_{LTO} simplifies to

$$W_{LTO} = \left[\frac{Z_{LTO}}{mr^2} \right] [1 - \exp(-rt_f)] [1 - \exp(-rm)]$$

Should occupational health (accident) risk due to long-term doses not be discounted in an analysis, r effectively becomes zero in the preceding equations. In the limit as r approaches zero

$$W_{LTO} = Z_{LTO}(t_f - t_i)$$

or

$$W_{LTO} = Z_{LTO}t_f, \text{ when } t_i = 0$$

The quantity W_{LTO} must be interpreted carefully to avoid misunderstandings. It does not represent the expected reduction in occupational health (accident) risk due to long-term doses as a result of a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime of the facility. Thus, it reflects the expected annual loss due to a single accident (this is given by the quantity Z_{LTO}); the possibility that such an accident could occur, with some probability, at any time over the remaining facility life; and the effects of discounting these potential future losses to present value. Since the quantity Z_{LTO} only accounts for the risk of an accident in a representative year, the result is the expected loss over the facility life, discounted to present value.

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for the occupational health (accident) attribute.

5.7.4 Occupational Health (Routine)

As with occupational health (accident), the evaluation of the effect on occupational health from a change in routine exposure due to proposed regulatory actions is a multi-step process. Reduction in exposure is algebraically positive; increase is negative (viewed as a negative reduction).

The steps are as follows:

1. Estimate reductions in occupational health (routine) risk per facility for implementation (D_{ORI}) and operation (D_{ORO}) (see Section 5.7.4.1)
2. Convert each reduction in occupational health (routine) risk per facility from person-rem to dollars via monetary evaluation of health effects (see Section 5.7.4.2):

$$G_{ORI} = RD_{ORI} \quad G_{ORO} = RD_{ORO}$$

where G_{ORI} = monetary value of per-facility reduction in routine occupational dose to implement the proposed action, before discounting (\$/facility)
 G_{ORO} = monetary value of annual per-facility reduction in routine occupational dose to operate following implementation of the proposed action, before discounting (\$/facility-year)
 D_{ORI} = per-facility reduction in routine occupational dose to implement the proposed action (person-rem/facility)
 D_{ORO} = annual per-facility reduction in routine occupational dose to operate following implementation of the proposed action (person-rem/facility-year)
 R = monetary equivalent of unit dose (\$/person-rem).

3. Discount each reduction in occupational health (routine) risk per facility (dollars) (see Section B.2)⁽⁹⁾

4. Sum the reductions and total over all facilities (dollars):

$$O_{OHR} = N(H_{ORI} + H_{ORO})$$

where V_{OHR} = discounted monetary value of reduction in occupational health (routine) risk for all affected facilities (\$)
 H_{ORI} = monetary value of per-facility reduction in routine occupational dose required to implement the proposed action, after discounting (\$/facility)
 H_{ORO} = monetary value of per-facility reduction in routine occupational dose to operate following implementation of the proposed action, after discounting (\$/facility)
 N = number of affected facilities.

Note the algebraic signs for D_{ORI} and D_{ORO} . A reduction in exposure is positive; an increase is negative. The dose for implementation (D_{ORI}) would normally be an increase and therefore negative.

If individual facility values rather than generic values are used, the formulations can be replaced with

$$V_{OHR} = \sum_i N_i (H_{ORI_i} + H_{ORO_i})$$

where i = facility (or group of facilities) index.

5.7.4.1 Estimation of Change in Routine Exposure

A proposed NRC action can affect routine occupational exposures in two ways. It may cause a one-time increase in routine dose due to implementation of the action (e.g., installing a retrofit). It may also cause a change (either increase or decrease) in the recurring routine exposures after the action is implemented. A new coolant system decontamination technique, for example, may cause a small implementation dose but may result in a decrease in annual exposures from maintenance thereafter.

For the standard analysis, the analyst may attempt to make exposure estimates, or obtain at least a sample of industry or other technical data for a validation of the estimates developed. There are two components in the development of an exposure estimate: estimating the radiation field (rem/hour) and estimating the labor hours required. The product is the exposure (person-rem). In developing operational estimates, the annual frequency of the activity is also required.

General estimates of radiation fields can be obtained from a number of sources. For power reactors, Chapter 12 of the Final Safety Analysis Report (FSAR) for the plant will contain a partitioning of the power plant into estimated radiation zones. Both summary tables and plant layout drawings are usually provided. Some FSARs provide exposure estimates for specific operational activities. The analyst must be cautioned that the FSAR values are calculated, not measured. Actual data from operating facilities, as might be obtained from facility surveys, would have greater accuracy. Generic estimates of dose rates for work on specific Pressurized Water Reactor (PWR) and BWR systems and components are provided by Beal et al. (1987) and included in Section B.3. These are used by Sciacca (1992) in NUREG/CR-4627 along with labor hours and occupational exposure estimates for specific repair and modification activities. Appropriate references are cited. The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) contains a database of default dose rates and ranges for both PWR and BWR systems.

Work in a radiation zone inevitably requires extra labor due to radiation exposure limits and lower worker efficiency. Such inefficiencies arise from restrictive clothing, rubber gloves, breathing through filtered respirators, standing on

ladders or scaffolding, or crawling into inaccessible areas. In addition, paid breaks must be allowed for during a job. Basically, there are five types of adjustment factors identified for work on activated or contaminated systems. LaGuardia et al. (1986) identify the following five time duration multipliers:

1. Height (i.e., work conducted at elevations, e.g., on ladders or scaffolds) = 10-20% of basic time duration
2. Respiratory Protection = 25-50% of basic time duration
3. Radiation Protection = 10-40% of basic time duration
4. Protective Clothing = 30% of adjusted time duration
5. Work Breaks = 8.33% of total adjusted time duration.

Sciacca (1992) provides information from which to estimate relevant labor productivity factors, whose values can vary with the status of the plant and work environment at the time of the action.

Keeping these factors in mind, the analyst can proceed with the estimation of implementation and operational doses. The implementation dose would be

$$D_{ORI} = - F_R \times W_I$$

where D_{ORI} = per-facility reduction in routine occupational dose required to implement the proposed action (person-rem/facility-year)
 F_R = radiation field in area of activity (rem/hour)
 W_I = work force required for implementation (labor-hours/facility).

As mentioned earlier, implementation dose normally involves an increase, hence the negative sign in the equation.

The operational dose is the change from the current level; its formulation is

$$D_{ORO} = (F_R W_O A_F)_S - (F_R W_O A_F)_A$$

where D_{ORO} = annual per-facility reduction in routine occupational dose to operate following implementation of the proposed action (person-rem/facility-year)
 F_R = radiation field in area of activity (rem/hour)
 W_O = work force required for activity (labor-hours/facility-activity)
 A_F = number of activities (e.g., maintenance, tests, inspections) per year (activities/year)
 S = status quo (current conditions)
 A = after implementation of proposed action.

Again, note the algebraic sign for D_{ORO} . As mentioned earlier, an operational dose reduction is positive; an increase is negative.

If the issue does not lend itself to the estimation procedure just presented, the analyst may use the following approximation specifically for reactor facilities. To estimate changes in routine operational dose, the analyst may directly estimate fractional changes for routine doses. The techniques for soliciting expert opinion discussed in Section 5.6.2 could be

employed. The average annual occupational dose for BWRs in 1993 was 330 person-rem/reactor and 0.31 person-rem/worker (see Table B.9). For PWRs, the average was 194 person-rem/reactor and 0.25 person-rem/worker (see Table B.10). The overall average annual occupational dose at LWRs in 1993 was 240 person-rem/reactor and 0.27 person-rem/worker (see Table B.11). Additional data on routine occupational exposure for both power reactors and non-reactor facilities are provided in Section B.3. Also, routine occupational risks for selected non-reactor facilities have been assembled into composite lists in Section C.2.4.

For a major effort beyond the standard analysis, the best source of data to estimate both the implementation and operational exposures would be a thorough survey of health physicists at the affected facilities. This survey could be screened for bias and potential inflated value by a knowledgeable third party.

5.7.4.2 Monetary Valuation of Routine Exposure

The analyst should use the \$2000 per person-rem conversion factor discussed in Section 5.7.1.2 for the monetary valuation of routine exposures.

5.7.4.3 Nonradiological Occupational Impacts

In some cases, it will be possible to identify nonradiological occupational impacts associated with a proposed action. When possible, these should be identified and included in the regulatory analysis. One source of data on the incidence of occupational injuries for various industries is the report *Occupational Injuries and Illnesses in the United States by Industry*, published annually by the Department of Labor's Bureau of Labor Statistics (BLS). Data from this report can be accessed from the BLS Home Page on the Internet (URL: <http://stats.bls.gov:80/datahome.htm>).

Occupational injury data should be converted to a dollar valuation. The value of an injury should include medical costs and the value of lost production (RWG 1996, Section 5). The value of loss production is normally estimated using employee wage rates. Pain and suffering costs attributable to occupational injury can be identified qualitatively, but would not normally be quantified in dollar terms. Potential information sources for occupational injury valuation data are the National Center for Health Statistics (URL: <http://www.cdc.gov/nchswww/nchshome.htm>) and the publication *Accident Facts* published annually by the National Safety Council based in Itaska, Illinois.

5.7.5 Offsite Property

Estimating the effect of the proposed action upon offsite property involves three steps:

1. Estimate reduction in accident frequency (see Section 5.6), incorporating conditional probability of containment/confinement failure, if applicable.
2. Estimate level of property damage.
3. Calculate reduction in risk to offsite property as

$$V_{FP} = N\Delta FD$$

where V_{FP} = monetary value of avoided offsite property damage (\$)
 N = number of affected facilities

- ΔF = reduction in accident frequency (events/facility-year)
 D = present value of property damage occurring with frequency F (\$-year).

It is possible that the proposed action mitigates the consequences of an accident instead of, or as well as, reducing the accident frequency. In that event, the value of the action is

$$V_{FP} = (NFD)_S - (NFD)_A$$

- where F = accident frequency (events/facility-year)
 S = status quo (current conditions)
 A = after implementation of proposed action.

Reduction in offsite property damage costs (i.e., costs savings) is algebraically positive; increase (i.e., cost accruals) is negative (viewed as negative cost savings).

An important tool formerly used by the NRC to estimate power reactor accident consequences is the computer program CRAC2 (Ritchie et al. 1985). More recently, the computer code MACCS (Chanin et al. 1990, 1993; Summers et al. 1995a,b) has been developed to estimate power reactor accident consequences using currently available information. MACCS was employed for the consequence analyses in NUREG-1150 (NRC 1991). The analyst interested in code descriptions for CRAC2 or MACCS is referred to the references cited.

For the standard analysis specifically applied only to power reactor facilities, estimates based on work by Mubayi et al. (1995) can be employed. Mubayi et al. (1995) have developed costs for offsite consequences for the five power reactors analyzed in NUREG-1150 (NRC 1991). These costs have been weighted by the frequencies of the accident release categories for the five plants. The results (in 1990 dollars) are given in Table 5.6. The analysis used Version 1.5.11.1 of the MACCS computer code (Chanin et al. 1993) on a site-specific basis. Offsite costs have been calculated as the sum of those for emergency response and long-term protective action, defined as follows:

- For early consequences, an effective emergency response plan consisted of evacuation of all but 0.5% of the population within a ten-mile radius at a specified speed and delay time following notification of the emergency.

Table 5.6 Weighted costs for offsite property damage for the five NUREG-1150 power reactors

Reactor	Type	Cost (1990 \$) Within 50 Miles from the Plant
Zion	PWR	2.23E+8
Surry	PWR	2.30E+8
Sequoyah	PWR	3.19E+8
Peach Bottom	BWR	2.71E+9
Grand Gulf	BWR	1.87E+8
Average		2.46E+8

- For long-term relocation and banning of agricultural products, the interdiction criterion was 4 rem to an individual over five years (2 rem in year one, followed by 0.5 rem each successive year).

Cost values within 50 miles are to be used in the regulatory analysis. Alternative values reflecting shorter and longer distances from the plant may be used for sensitivity analyses or special cases, and are available in Mubayi et al. (1995).

The present value for offsite property damage can be calculated as

$$D = C \times B$$

where D = present value of offsite property damage (\$-year)

C = $[\exp(-rt_i) - \exp(-rt_f)]/r$

t_f = years remaining until end of facility life

t_i = years before facility begins operating

r = real discount rate (as fraction not percent)

B = undiscounted cost of offsite property damage.

If a facility is already operating, t_i will be zero and the equation for C simplified to

$$C = [1 - \exp(-rt_f)]/r$$

Should offsite property damage not be discounted in an analysis (e.g., when the time frame is sufficiently short to mitigate the need for discounting), r effectively becomes zero in the preceding equations. In the limit as r approaches zero, $C = t_f$ (or, $C = t_i$ when $t_i = 0$). This new value for C should be used to evaluate D in the undiscounted case.

The quantity D must be interpreted carefully to avoid misunderstandings. It does not represent the expected offsite property damage due to a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime of the facility. Thus, it reflects the expected loss due to a single accident (this is given by the quantity B); the possibility that such an accident could occur, with some probability, at any time over the remaining facility life; and the effects of discounting these potential future losses to present value. When the quantity D is multiplied by the annual frequency of an accident, the result is the expected loss over the facility life, discounted to present value.

Costs for offsite property damage from non-reactor accidents have been assembled in Section C.2.5. However, most are given as combined offsite and onsite damage costs and have not been as thoroughly estimated as those by Mubayi et al. (1995) for offsite property damage from power reactor accidents.

At a more detailed level, but still within the scope of a standard analysis, the analyst can identify the affected facilities, then calculate the proper sum effect rather than relying on generic values. The following steps are required:

1. Identify affected facilities.
2. Identify reductions in accident frequency per facility.
3. Calculate value of property damage per facility.

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4. Calculate avoided property damage value per facility.
5. Sum avoided property damage over affected facilities.

In the 1983 Handbook, Heaberlin et al. made extensive use of NUREG/CR-2723 (Strip 1982) for offsite property cost estimation. Strip reported the present value of offsite health and property costs, onsite costs, and replacement power costs for accidents in release categories SST1 through SST3 for 91 U.S. power reactor sites. The offsite property costs were based on CRAC2 results, with 1970 population estimates and state-wide land use. The analyst may find the site-specific emphasis in Strip (1982) helpful in a more detailed value-impact analysis.

For a major effort beyond the standard analysis, it is recommended that the estimates be derived from information more site-specific than that used by Strip (1982). For power reactors, the MACCS code with the most recent data available should be used. This degree of effort would be relatively costly to conduct, both in terms of computer costs and data collection and interpretation costs. However, it would provide the highest degree of reliability.

Burke et al. (1984) examined the offsite economic consequences of severe LWR accidents, developing costs models for the following:

- population evacuation and temporary sheltering, including food, lodging, and transportation
- emergency phase relocation, including food, housing, transportation, and income losses
- intermediate phase relocation, beginning immediately after the emergency phase
- long-term protective actions, including decontamination of land and property and land area interdiction
- health effects, including the two basic approaches (human capital and willingness-to-pay).

Tawil et al. (1991) compared three computer models for estimating offsite property damage from power reactor accidents. Two of the models are the CRAC2 and MACCS codes; the third is the computer code DECON (Tawil et al. 1985). Three accident severity categories—SST1-SST3—are considered for the six Pasquill atmospheric stability categories (A-F). Offsite property damage is calculated for each pairing at cleanup levels from 10 through 200 rems. A study is also performed comparing the effect of modeling offsite damage to radii of 50 and 500 miles. It indicates that the choice of radius is significant only for the SST1 accident category, the differences being quite pronounced.

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for the offsite property attribute.

5.7.6 Onsite Property

Section 4.3.1 of the NRC Guidelines states that onsite property damage cost savings (i.e., averted onsite costs) need to be included in the value-impact analysis. In the net-value formulation it is a positive attribute.

Estimating the effect of the proposed action on onsite property involves three steps:

1. Estimate reduction in accident frequency (see Section 5.6).
2. Estimate onsite property damage.

3. Calculate reduction in risk to onsite property as

$$V_{OP} = N\Delta FU$$

where V_{OP} = monetary value of avoided onsite property damage (\$)
 N = number of affected facilities
 ΔF = reduction in accident frequency (events/facility-year)
 U = present value of property damage occurring with frequency F (\$-year).

Reduction in onsite property damage costs (i.e., costs savings) is algebraically positive; increase (i.e., cost accruals) is negative (viewed as negative cost savings).

For the standard analysis, it is convenient to treat onsite property costs under three categories: 1) cleanup and decontamination, 2) long-term replacement power, and 3) repair and refurbishment. Each of these categories is considered below for power reactors with the focus on large-scale core-melt accidents. Additional categories of costs have been considered by Mubayi et al. (1995) and Burke et al. (1984) as outlined in Section 5.7.6.4, but they were either found to be speculative or contributed small fractions to the costs identified below.

5.7.6.1 Cleanup and Decontamination

Cleanup and decontamination of a nuclear facility, especially a power reactor, following a medium or severe accident can be extremely expensive. For example, Mubayi et al. (1995) report that the total cleanup and decontamination of TMI-2 cost roughly \$750 million (in 1981 dollars). Murphy and Holter (1982) estimated cleanup costs for a reference PWR and BWR for the following three accident scenarios:

- Scenario 1 - a small LOCA in which ECCS functions as intended. Some fuel cladding ruptures, but no fuel melts. The containment building is moderately contaminated, but there is minimal physical damage.
- Scenario 2 - a small LOCA in which ECCS is delayed. Half of the fuel cladding ruptures, and some fuel melts. The containment building is extensively contaminated, but there is minimal physical damage.
- Scenario 3 - a major LOCA in which ECCS is delayed. All fuel cladding ruptures, and there is significant fuel melting and core damaged. The containment building is extensively contaminated and physically damaged. The auxiliary building undergoes some contamination.

In 1981 dollars, Murphy and Holter estimated the following cleanup costs:

Scenario	PWR	BWR
1	\$1.05E+8	\$1.28E+8
2	\$2.24E+8	\$2.28E+8
3	\$4.04E+8	\$4.21E+8

Mubayi et al. (1995) consider the TMI-2 accident to lie between Scenarios 2 and 3, lying closer to Scenario 3 in terms of the contamination and damage to the core. Murphy and Holter's costs were somewhat less than those actually realized at TMI. Mubayi et al. (1995) attribute the difference to three factors:

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1. The start of the TMI cleanup was delayed by 2.5 years due to regulatory and financial requirements. Murphy and Holter assumed no additional delays between the accident and start of the cleanup. Mubayi et al. (1995) consider this somewhat unrealistic.
2. Decontamination at TMI required facilities not included in Murphy and Holter's reference plants (e.g., a hot chemistry laboratory, containment recovery service building, and comment center/temporary personnel access facility).
3. TMI required additional decontamination of the containment building after the reactor was defueled. Murphy and Holter excluded this in their analysis.

When these three factors are considered, the results from Murphy and Holter become reasonably consistent with the actual TMI cleanup costs (\$7.50E+8 in 1981 dollars).

Burke et al. (1984) produced a very rough estimate of \$1.7 billion (in 1982 dollars) for the cleanup and decontamination costs following a severe power reactor accident. An uncertainty range of approximately 50% was assigned, bringing the lower bound reasonably in line with the actual TMI cleanup cost. A study by Konzek and Smith (1990) updated the cleanup costs associated with Murphy and Holter's Scenario 3. Costs ranging from \$1.22E+9 to \$1.44E+9 (in undiscounted 1989 dollars) were estimated, based on real escalation rates of 4% to 8% during the cleanup period. A base cost of \$1.03E+9 was estimated assuming no real escalation during the cleanup period.

After converting the costs to undiscounted 1993 dollars, the cost reported by Mubayi et al. (1995) for TMI is \$1.2E+9, the base estimate from Konzek and Smith (1990) is \$1.2E+9, and the estimate from Burke et al. (1984), which doubled the cost of TMI, is \$2.5E+9. Based on these references, the total onsite cost estimates given in Section 5.7.6.4 are based on \$1.5E+9 (undiscounted) for cleanup and decontamination (C_{CD} in the equations that follow). For sensitivity analysis, lower and upper bounds of \$1.0E+9 and \$2.0E+9 are recommended for evaluating severe accident effects.

Assuming the \$1.5E+9 estimate is spread evenly over a 10-year period for cleanup (i.e., constant annual cost of $C_{CD}/m = \$1.5E+8$ in the equation below, with $C_{CD} = \$1.5E+9$ and $m = 10$ years), and applying a 7% real discount rate, the cost translates into a net present value of \$1.1E+9 for a single event. This quantity is derived from the following equation (see Section B.2.3):

$$PV_{CD} = [C_{CD} / mr] [1 - \exp(-rm)]$$

where PV_{CD} = net present value of cleanup and decontamination costs for single event (\$)
 C_{CD} = total undiscounted cost for single accident in constant year dollars (\$)
 m = years required to return site to pre-accident state
 r = real discount rate (as fraction, not percent).

Before proceeding, this present value must be decreased by the cleanup and decontamination costs associated with normal reactor end-of-life. The Yankee Atomic Electric Co. (NRC 1995c), Sacramento Municipal Utility District (NRC 1994), and Portland General Electric Co. (1995) provided the following estimates to the NRC for decommissioning their Yankee Rowe, Rancho Seco, and Trojan nuclear power plants, respectively: \$3.41E+8 (1991 dollars), \$2.80E+8 (1991 dollars), and \$4.15E+8 (1993 dollars). These suggest a value of approximately \$0.4E+9 (1993 dollars) for "normal" cleanup and decommissioning. The analyst can also consult Bierschbach (1995) for estimating PWR decommissioning costs and Bierschbach (1996) for estimating BWR decommissioning costs.

When spread evenly over the same 10-year period at a 7% real discount rate, this translates into a net present value of \$0.3E+9. However, since this value would "normally" be applied at reactor end-of-life (i.e., 24 years later, using the

estimate from Table B.1), the net present value (at the same 7% real discount rate) is reduced to \$0.06E+9. Since this amounts to only 5% of the net present value for cleanup and decontamination following a severe accident (\$1.1E+9), it can be generally ignored.

The total onsite cost estimates shown in Section 5.7.6.4 integrate this net present value over the average number of remaining service years (24 years) using the following equation:

$$U_{CD} = [PV_{CD} / r] [1 - \exp(-rt_f)]$$

where U_{CD} = net present value of cleanup and decontamination over life of facility (\$-year)
 t_f = years remaining until end of facility life.

The integrated cost is \$1.3E+10 over the life of a power reactor. This cost must be multiplied by the accident frequency (F, expressed in events per facility-year), and the number of reactors, to determine the expected value of cleanup and decontamination costs. To determine averted costs, the reduction in accident frequency ΔF is applied as outlined in Section 5.7.6.

For comparison, these costs can also be estimated for less severe accidents as defined by Murphy and Holter's Scenarios 1 and 2. The estimates shown in the following table were obtained by using \$1.1E+9 (1993 dollars) as a base value for Scenario-3 PV_{CD} costs, and applying the same relative fractions as shown in Murphy and Holter's (1982) results for Scenario-1 and 2 costs. The results from Murphy and Holter were not used directly because of the factors cited by Mubayi et al. (1995) in comparisons of those estimates with actual cleanup and decontamination costs at TMI.

Scenario	PV_{CD}	U_{CD}
1	\$3.1E+8	\$3.7E+9
2	\$6.0E+8	\$7.1E+9
3	\$1.1E+9	\$1.3E+10

The issue of license renewal has only moderate implications for the integrated cost estimates (U_{CD}). With longer operating lifetimes, the reactors are at risk for more years, and the costs would be expected to increase accordingly. However, because the additional costs are discounted to present worth terms, the effect is not substantial. For example, an additional life extension of 20 years would only increase the value of U_{CD} for a Scenario-3 accident 15% from \$1.3E+10 to \$1.5E+10.

5.7.6.2 Long-Term Replacement Power

Replaced power for short-term reactor outages is discussed in Section 5.7.7.1. Following a severe power reactor accident (replacement power need be considered only for electrical generating facilities), replacement power costs must be considered for the remaining reactor lifetime.⁽¹²⁾

Argonne National Laboratory (ANL) has developed estimates for long-term replacement power costs based on simulations of production costs and capacity expansion for representative pools of utility systems (VanKuiken et al. 1992). VanKuiken et al. examined replacement energy and capacity costs, including purchased energy and capacity charges required to provide the same level of system reliability as available prior to the loss of a power reactor (VanKuiken et al. 1993). In the event of a permanent shutdown, it was assumed that a reactor would be replaced by one or more alternative generating units, after an appropriate delay for planning and construction.

Capacity expansion and production cost simulations were performed for six representative power reactors over 40-year study periods. The results were used to estimate replacement power costs for each of 112 reactors which, at the time of the study, were expected to be in operation by 1996. Cost estimates for each reactor reflect the remaining lifetimes, reactor sizes, and ranges in short-term replacement energy costs (as encountered in each utility). Averages were determined by summing the individual reactor costs and dividing by the number of reactors evaluated. Characteristics for the "generic" reactor cited in Section 5.7.6.4 reflect an average unit size of 910-MWe and average life remaining of 24 years for reactors currently operating and planned.

Simulation results were first used to estimate the present value costs of single accidents occurring in each year of remaining facility lifetimes (quantity PV_{RP} used in the discussions that follow). Each of these net present values represents a summation of annual replacement power costs incurred from the year of the assumed accident to the final year of service. For example, the average net present value for an event occurring in 1993 is \$1.1E+9. For 1994, the cost is \$1.0E+9, and for 1995, the cost is \$0.9E+9. The decline in costs with each successive year reflects present value considerations and the fact that there are fewer remaining service years requiring replacement power.

The following equation can be used to approximate the average value of PV_{RP} for alternative discount rates.

$$PV_{RP} = [\$1.2E + 8 / r] [1 - \exp(-rt_f)]^2$$

where PV_{RP} = net present value of replacement power for a single event (\$).

The \$1.2E+8 value used in the above equation has no intrinsic meaning. It is treated in the equation similar to an equivalent annual cost, but it is actually a substitute for a string of non-constant replacement power costs that occur over the lifetime of the generic reactor after an event that takes place in 1993. The equation is only presented here for examining the effects of alternate discount rates and remaining reactor lifetimes.

The above equation for PV_{RP} was developed for discount factors in the range of 5%-10%. Unlike the equations for PV_{CD} and U_{CD} , the equation for PV_{RP} diverges from modeled results at lower discount rates. At a discount rate of 3% the recommended value for PV_{RP} is \$1.4E+9, as compared with the equation estimate of \$1.1E+9. For discount rates between 1% and 5% the analyst is urged to make linear interpolations using \$1.6E+9 at 1% and \$1.2E+9 at 5%. At higher discount rates the equation for PV_{RP} provides recommended estimates of \$1.2E+9 at 5% and \$1.0E+9 at 10%.

The results that are applied in Section 5.7.6.4 sum the single-event costs over all years of reactor service. While these summations were calculated directly from simulation results, ANL found that the outcomes could be closely approximated with the equation that follows. The squared term in this equation serves as a proxy for the fact that costs for events in future years decline due to the reduced number of remaining service years for which replacement power is required:

$$U_{RP} = [PV_{RP} / r] [1 - \exp(-rt_f)]^2$$

where U_{RP} = net present value of replacement power over life of facility (\$-year).

Replacement power costs for the generic unit are estimated to be approximately \$10 billion over the life of the facility. An uncertainty range for this average is estimated at approximately 20%. However, the range of estimates for specific power reactors varies directly with unit size, remaining life, and replacement energy costs. For example, costs were estimated to be \$7.5 billion for the 1040-MWe Zion-2 reactor, assuming 16 years of remaining operating life. Zion-2 is in a power pool with approximately average replacement energy costs. In contrast, costs for Big Rock Point were \$120 million due to its smaller size (67-MWe), shorter remaining life (8 years assumed), and average replacement energy costs. At the upper

limit were costs of \$24 billion for the 1090-MWe Nine Mile Point 2 unit, assuming 34 years of service remaining. Nine Mile Point 2 is in a power pool with above average replacement energy costs.

As noted for PV_{RP} , the equation for U_{RP} was developed for discount rates ranging from 5%-10%. For lower discount rates, linear interpolations for U_{RP} are recommended between $\$1.9E+10$ at 1% and $\$1.2E+10$ at 5%. The equation for U_{RP} yields the recommended values of $\$1.2E+10$ at 5% and $\$0.8E+10$ at 10%, based on PV_{RP} values described previously.

As discussed in Section 5.7.6.4, these summed costs must be multiplied by the accident frequency (expressed in events per facility-year) to determine the expected value of replacement power costs for a typical reactor. To determine the value of reductions in the accident frequency due to regulatory actions, the total integrated costs must be multiplied by the reduction in accident frequency ΔF and the number of reactors affected (N).

The issue of license renewal has a much more significant impact on replacement power costs than on cleanup and decontamination costs. Extending the operating life by an additional 20 years would increase the net present value of a single event (PV_{RP}) by about 38%, and would increase the present value of costs integrated over the reactor life (U_{RP}) by about 90% (VanKuiken et al. 1992). Thus, a license renewal period of 20 years would mean the generic reactor would have a remaining life of 44 years, PV_{RP} would be estimated to be $\$1.5E+9$, and U_{RP} would be approximately $\$1.9E+10$ (1993 dollars).

For less severe accidents such as characterized by Scenario-1 events, the analyst is referred to Section 5.7.7.1 which addresses short-term replacement energy costs. Replacement capacity costs, which contribute to severe accident costs, are not incurred for more temporary reactor shutdowns.

5.7.6.3 Repair and Refurbishment

In the event of recoverable accidents (i.e., for Scenario 1, but not Scenarios 2 or 3), the licensee will incur costs to repair/replace damaged components before a facility can be returned to operation (these costs are not included in the total onsite cost estimates for severe accidents as addressed in Section 5.7.6.4). Burke et al. (1984) have estimated typical costs for equipment repair on the order of \$1,000/hr of outage duration, based on data from outages of varying durations at reactors. They suggest an upper bound of roughly 20% of the long-term replacement power costs for a single event. Mubayi et al. (1995) observe that the \$1,000/hr figure corresponds closely to the repair costs following the Browns Ferry fire and also to the TMI-1 steam generator retubing outage costs.

5.7.6.4 Total Onsite Property Damage Costs

Based on the information included in Sections 5.7.6.1 and 5.7.6.2, ANL has estimated the total cost due to onsite property damage following a severe reactor accident for the Zion-2 reactor and a "generic" 910-MWe reactor assumed to have a remaining life of 24 years. Total costs are assumed to consist of cleanup and decontamination costs and replacement power costs (repair and refurbishment costs are not included for severe accidents). The total costs described below correspond to the "risk-based" costs as defined by Mubayi et al. (1995):

"...risk-based cost, the discounted net present value of the risk over the remaining life of the plant, which is proportional to the accident frequency [F]..."

The risk-based costs (quantities U , U_{CD} , and U_{RP} in the equations that follow) must be interpreted carefully to avoid misunderstandings. They do not represent the expected onsite property damage due to a single accident. Rather, they are the present value of a stream of potential losses extending over the remaining lifetime of the facility. Thus, they reflect the expected loss due to a single accident (given by quantities PV_{CD} and PV_{RP}); the possibility that such an accident could

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occur, with some small probability, at any time over the remaining facility life; and the effects of discounting those potential future losses to the present value. When the quantity U is multiplied by the annual accident frequency, the result is the expected loss over the facility life, discounted to the present value.

The estimates for total risk-based costs attributed to regulatory actions that occur in 1993, expressed in 1993 dollars assuming a 7% real annual discount rate, are as follows:

<u>Variable</u>	<u>Cost Component</u>	<u>Zion-2</u>	<u>"Generic" Reactor</u>
U_{RP}	Replacement Power	$\$0.7E+10 \times F$	$\$1.0E+10 \times F$
U_{CD}	Cleanup & Decontamination	$\$1.0E+10 \times F$	$\$1.3E+10 \times F$
U	Total	$\$1.7E+10 \times F$	$\$2.3E+10 \times F$

Alternate values of U may be approximated for different discount rates, years of operation remaining, and estimates for C_{CD} and PV_{RP} . However, for changes in discount rate or final year of operation, the analyst is cautioned to revise the estimates for PV_{RP} using the equation described in Section 5.7.6.2 prior to re-estimating U from the equation that follows. Also, for discount rates lower than 5%, PV_{RP} and U_{RP} should be estimated from interpolation guidelines presented in Section 5.7.6.2 rather than from the equations. The relationship that defines total lifetime costs is

$$U = U_{CD} + U_{RP} \\ = \left[C_{CD} / mr^2 \right] \left[1 - \exp(-rt_f) \right] \left[1 - \exp(-rm) \right] + \left[PV_{RP} / r \right] \left[1 - \exp(-rt_f) \right]^2$$

where U = total net present value of onsite property damage (\$-year).

The procedure outlined in Section 5.7.6 may be used to evaluate averted onsite property damage using these estimates. For illustration, assume that the reduction in severe accident frequency (ΔF) is $1.0E-6$ and the number of reactors affected (N) is 111. The total averted onsite damage costs would be

$$V_{OP} = N \Delta F U = (111) (1.0E-6) (\$2.3E + 10) = \$2.6E + 6$$

The value of this reduction in accident frequency is \$2.6 million (net present value in 1993 dollars).

The $\$2.3E+10$ value used above is an appropriate generic estimate for regulatory requirements that become effective in 1993 and that affect severe accident probabilities in that year. For regulatory actions that affect accident frequencies in future years, the cost estimates must be adjusted to recognize that the number of reactor-years at risk and the number of service years requiring replacement power are reduced. Table 5.7 shows how these factors affect cost estimates for the 10-year period of 1993-2002. The results are expressed as net present values discounted to the year that the rulemaking is assumed to take effect.

To illustrate the use of these estimates, assume a reduction in accident frequency of $1.0E-6$ begins in 1998 and affects all 111 of the remaining reactors. The revised estimate for U would be $\$1.9E+10$ and the total averted onsite damage costs for this reduction in frequency would be

$$V_{OP} = (111) (1.0E-6) (\$1.9E + 10) = \$2.1E + 6 \text{ (1993 dollars)}$$

Table 5.7 Onsite property damage cost estimates (U) for future years (1993 dollars discounted to year of implementation)

	Cleanup and Decontamination (U _{CD})	Replacement Power (U _{RP})	Total (U)
1993	\$1.3E+10	\$1.0E+10	\$2.3E+10
1994	\$1.2E+10	\$9.6E+9	\$2.2E+10
1995	\$1.2E+10	\$9.1E+9	\$2.1E+10
1996	\$1.2E+10	\$8.6E+9	\$2.1E+10
1997	\$1.1E+10	\$8.1E+9	\$1.9E+10
1998	\$1.1E+10	\$7.6E+9	\$1.9E+10
1999	\$1.1E+10	\$7.1E+9	\$1.8E+10
2000	\$1.1E+10	\$6.6E+9	\$1.8E+10
2001	\$1.0E+10	\$6.2E+9	\$1.6E+10
2002	\$1.0E+10	\$5.7E+9	\$1.6E+10

This would indicate that the reduction in accident frequency valued at \$2.6 million beginning in 1993 would be valued at \$2.1 million if introduced in 1998 (1993 dollars adjusted to 1998).

The following linear equation provides approximate cost estimates for implementation later than 10 years in the future. The result represents net present value (1993 dollars) discounted to the year of implementation. The analyst must adjust the 1993 dollars for general inflation if costs are to be expressed in alternate reference-year dollars. (See Section 5.8 for information on adjusting dollar years.)

$$U = \$2.3E + 10 - (\$6.7E + 8) (t_i - 1993)$$

where t_i = year of reduction in accident frequency.

Thus, for regulatory actions that would affect accident probabilities for 86 reactors remaining in service in 2010, the revised estimate for U would be

$$\begin{aligned} U &= \$2.3E + 10 - (\$6.7E + 8) (2010 - 1993) \\ &= \$1.2E + 10 \text{ (1993 dollars adjusted to 2010)} \end{aligned}$$

The total averted onsite damages costs for a reduction in accident frequency of 1.0E-6 would be

$$\begin{aligned} V_{OP} &= (86) (1.0E - 6) (\$1.2E + 10) \\ &= \$1.0E + 6 \text{ (1993 dollars adjusted to 2010)} \end{aligned}$$

This example also illustrates that the number of reactors at risk and the average remaining years of reactor service change in the evaluation of future regulatory initiatives. Because of the distribution of license expiration dates, the average remaining reactor life does not decrease on a one-to-one basis with each successive year in the future.

For 20-year license renewal considerations, the estimates for U discussed above should be increased by approximately 50%. In 1993, U_{CD} would be estimated at $\$1.5E+10$ (versus $\$1.3E+10$ for 40-year license), and U_{RP} would be estimated to be $\$1.9E+10$ (versus $\$1.0E+10$ for 40-year license). This yields a total of $\$3.4E+10$ (1993 dollars) as compared with $\$2.3E+10$ for the 40-year license assumption.

Costs for onsite property damage from non-reactor accidents have been assembled in Section C.2.5. However, most are given as combined offsite and onsite damage costs.

For a major effort beyond the standard analysis, there are two general ways to achieve a greater level of detail: 1) the analysis can be conducted for individual facilities or groups of similar facilities, using site-specific information; 2) the analysis can provide cost information in much greater detail. With regard to the first approach, the most relevant site-specific information includes the cost of long-term replacement power and the value of the facility and equipment at risk, taking into account the remaining useful life of the facility. The analyst is referred to VanKuiken et al. (1992) for further detail on average shutdown costs for different categories of reactors (e.g., by region, reactor supplier, architect engineer, etc.), and guidance for scaling costs for different unit sizes and remaining lifetimes.

With regard to providing greater detail on the cost information, the major cost elements (in addition to replacement power) are likely to include decontamination and other cleanup costs and repair or replacement of plant and equipment that is physically damaged. Other costs relate to transporting and disposing of contaminated materials and equipment, and startup costs. Costs for monitoring the site for radiation and fixing contamination at the site will likely be insignificant relative to the other costs. The analyst is referred to Murphy and Holter (1982), and the follow-up study by Konzek and Smith (1990), for detailed cost estimates to decontaminate a nuclear power reactor following a postulated accident.

Burke et al. (1984) examined the onsite economic consequences of severe LWR accidents, developing cost models for the following:

- replacement power, drawing information mainly from Buehring and Peerenboom (1982) (which has been updated by VanKuiken et al. [1992])
- plant decontamination, including both medium and large consequence events
- plant repair, spanning small to large consequence events
- early decommissioning for medium and large consequence events
- worker health effects and medical care, primarily for medium and large consequence events
- electric utility "business" (i.e., costs resulting from changed risk perceptions in financial markets and the need to replace the income once produced by the operating plant after a power plant is permanently shutdown)
- nuclear power "industry" (i.e., costs resulting from elimination or slowed growth in the U.S. nuclear power industry due to altered policy decisions and risk perceptions following a severe accident)
- onsite litigation (i.e., "legal fees for the time and effort of those individuals involved in the litigation process").

The first three categories of costs have been covered in Sections 5.7.6.1-5.7.6.3. The other categories are covered elsewhere in this Handbook or are considered to be either speculative or small in magnitude relative to replacement power, cleanup and decontamination, and repair costs.

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for the onsite property attribute.

5.7.7 Industry Implementation

This section provides procedures for computing estimates of the industry's incremental costs to implement the proposed action. Estimating incremental costs during the operational phase that follows the implementation phase is discussed in Section 5.7.8. Incremental implementation costs measure the additional costs to industry imposed by the regulation; they are costs that would not have been incurred in the absence of that regulation. Reduction in the net cost (i.e., cost savings) is algebraically positive; increase (i.e., cost accrual) is negative (viewed as negative cost savings). Both NRC and Agreement State licensees should be addressed, as appropriate.

In general, there are three steps that the analyst should follow in order to estimate industry implementation costs:

Step 1 - Estimate the amount and types of plant equipment, materials, and/or labor that will be affected by the proposed action.

Step 2 - Estimate the costs associated with implementation.

Step 3 - If appropriate, discount the implementation costs, then sum (see Section B.2).

In preparing an estimate of industry implementation costs, the analyst should also carefully consider all cost categories that may be affected as a result of implementing the action. Example categories include

- land and land-use rights
- structures
- hydraulic, pneumatic, and electrical equipment
- radioactive waste disposal
- health physics
- monitoring equipment
- personnel construction facilities, equipment, and services
- engineering services
- recordkeeping
- procedural changes

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- license modifications
- staff training/retraining
- administration
- facility shutdown and restart
- replacement power (power reactors only)
- reactor fuel and fuel services (power reactors only)
- items for averting illness or injury (e.g., bottled water or job safety equipment).

Note that transfer payments (see Section 4.3) should not be included.

For the standard analysis, the analyst should use consolidated information to estimate the cost to industry for implementing the action. Sciacca (1992) is a prime source of such information, providing not only cost estimates, but also labor hours, cost rates, and adjustment factors, mainly for reactor facilities. Appropriate references are cited by Sciacca. The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) incorporates much of the information assembled by Sciacca (1992) into a computer database for the analyst's use in estimating industry implementation as well as other costs.

Step 1 - Estimate the amounts and types of plant equipment, materials, and/or labor that will be affected by the proposed action, including not only physical equipment and craft labor, but professional staff labor for design, engineering, quality assurance, and licensing associated with the action. If the action requires work in a radiation zone, the analyst should account for the extra labor required by radiation exposure limits and low worker efficiency due to awkward radiation protection gear and tight quarters (see discussion of labor productivity in Section 5.7.4.1).

When performing a sensitivity analysis, but not for the best estimate, the analyst should include contingencies, such as the most recent greenfield construction project contingency allowances supplied by Robert Snow Means Co., Inc. (1995). They suggest adding contingency allowances of 15% at the conceptual stage, 10% at the schematic stage, and 2% at the preliminary working drawing stage. The FORECAST computer code (Lopez and Sciacca 1996) contains an option to include an allowance for uncertainty and cost variations at the summary cost level. The Electric Power Research Institute (EPRI 1986) offers guidelines for use in estimating the costs for "new and existing power generating technologies." EPRI suggests applying two separate contingency factors, one for "projects" to cover costs resulting from more detailed design, and one for "process" to cover costs associated with uncertainties of implementing a commercial-scale new technology.

Step 2 - Estimate the costs associated with implementation, both direct and indirect. Direct costs include materials, equipment, and labor used for the construction and initial operation of the facility during the implementation phase. Indirect costs include required services. The analyst should identify any significant secondary costs that may arise. One-time component replacement costs and associated labor costs should be accounted for here. For additional information on cost categories, especially for reactor facilities, see Schulte et al. (1978) and United Engineers and Constructors, Inc. (1979; 1988a, b).

Step 3 - If appropriate, discount the costs, then sum. If costs occur at some future time, they should be discounted to yield present values (see Section B.2). If all costs occur in the first year or if present value costs can be directly estimated, discounting is not required. Generally, implementation costs would occur shortly after adoption of the proposed action.

When performing value-impact analyses for non-reactor facilities, the analyst will encounter difficulty in finding consolidated information on industry implementation costs comparable to that for power reactors. Comprehensive data sources such as Sciacca (1992) and the references from which he drew his information are generally unavailable for non-reactor facilities. Some specific information for selected non-reactor facilities is in Sections C.7-C.10. The types of non-reactor facilities (see Section C.1) are quite diverse. Furthermore, within each type, the facility layouts typically lack the limited standardization of the reactor facilities. These combine to leave the analyst pretty much "on his own" in developing industry implementation costs for non-reactor facilities. The analyst should follow the general guidelines given in this Handbook section. Specific data may be best obtained through direct contact with knowledgeable sources for the facility concerned, possibly even the facility personnel themselves.

For a major effort beyond the standard analysis, the analyst should obtain very detailed information, in terms of the cost categories and the costs themselves. The analyst should seek guidance from NRC contractors or industry sources experienced in this area (AE firms, etc.). The incremental costs of the action should be defined at a finer level of detail. The analyst, should refer to the code of accounts in the Energy Economic Data Base (EEDB [United Engineers and Constructors, Inc. 1988b]) or Schulte et al. (1978) to prepare a detailed account of implementation costs.

5.7.7.1 Short-Term Replacement Power

For power reactors, the possibility that implementation of the proposed action may result in the need for short-term replacement power must be addressed. Section 4.3.2 of the Guidelines indicates that replacement power costs are to be incorporated into a regulatory analysis when appropriate. Unlike the long-term costs associated with severe power reactor accidents discussed in Section 5.7.6.2, the replacement power costs associated with industry implementation of a regulatory action would be short-term.

For a "typical" 910-MWe reactor operating at an average capacity factor of 60%-65%, VanKuiken et al. (1992) suggests \$310,000/day (1993 dollars) as an average cost for short-term replacement power. The 60%-65% range in capacity factor is representative of annual averages, accounting for unplanned outage periods and planned outage periods for maintenance and refueling. However, if the timing of a short-term shutdown coincides with a time when a power reactor is expected to be fully operational, then a higher average cost per day is more appropriate. At a capacity factor of 100%, the average cost for the typical reactor is estimated to be \$480,000/day (1993 dollars).

At a more detailed level, VanKuiken et al. (1992) project the seasonal replacement power costs for potential short-term shutdowns of 112 nuclear power plants over the five-year period from 1992 through 1996. These costs are estimated from probabilistic production-cost simulations of pooled utility-system operations. Average daily replacement power costs are presented by season for each of the 112 plants. The 20 U.S. power pools containing these plants are identified along with their following characteristics: total system capacity, annual peak load, annual energy demand, annual load factor, prices for fuels, and mix of generation by fuel type.

The sensitivity of replacement power costs to changes in oil and gas prices is quantified for each power pool. The effects of multiple plant shutdowns are addressed, with the replacement power costs quantified for each pool assuming all plants within the pool are shutdown.

The replacement power cost information compiled in an analogous but earlier study by VanKuiken et al. (1987) has subsequently been incorporated into two cost analysis computer codes. The Replacement Energy Cost Analysis Package (RECAP [VanKuiken et al. 1994]) determines the replacement energy costs associated with short-term shutdowns of nuclear power plants, and can be applied to determine average costs for general categories based on location, unit type (e.g., BWR), constructor, utility, and other differentiating criteria. Plant-specific costs are also available, and can be evaluated for user-specified outage durations and alternative capacity factor assumptions. FORECAST (Lopez and Sciacca 1996), a computer code for regulatory effects cost analysis, provides the user with the capability to estimate replacement power costs in current year dollars. Sciacca (1992) also provides a discussion and data for use in estimating replacement power costs based on this earlier study by VanKuiken et al. (1987).

Imposition of a new regulation often requires that a nuclear power plant be shutdown while the modification takes place. If the requirement is needed to meet adequate protection, the analyst can assume that the required downtime is independent of any scheduled downtime, thereby realizing full replacement power costs. However, the modification often is not needed to meet adequate protection, enabling it to be completed during already scheduled downtime. Only if the time needed to perform the modification exceeds that allotted for the scheduled downtime should any replacement power costs accrue, these being solely due to the excess time.

The most likely scenario permits the modification to be accommodated completely within already scheduled downtime, and this has frequently been the policy adopted by the NRC. As a result, no replacement power costs accrue. While this assumption holds for a modification performed in the absence of others required by new regulations, it tends to underestimate the cost of multiple modifications resulting from the cumulative effect of new NRC requirements. When multiple modifications are performed, as they often are, the originally scheduled downtime may be insufficient to accommodate all of them. Usually, this results from the limited number of available maintenance personnel and space restrictions for nearby component repair or service.

Historic data indicate roughly 15 days per year, or 17% and 25% of the annually scheduled downtime for PWRs and BWRs, respectively, can be attributed to the cumulative impact of new regulatory requirements. Assuming the contribution of each regulatory requirement to the incremental downtime equals the overall percentage increase, one can assign a prorated share to that requirement (i.e., 17% for PWRs, 25% for BWRs, or roughly 20% for LWRs in general). For example, if a regulatory requirement requires one-week of reactor shutdown time, 1.2 days (PWRs), 1.8 days (BWRs), or 1.4 days (LWRs) of additional downtime and, thus, replacement power costs would accrue.

5.7.7.2 Premature Facility Closing

Several nuclear power plants have been voluntarily shut down prior to the expiration of their operating licenses. Normally, a decommissioning cost of approximately $\$0.3E+9$ (1993 dollars) would be associated with an end-of-life shutdown (see Section 5.7.6.1). However, if a proposed regulatory requirement is expected to result in a premature shutdown, this cost is shifted to an earlier time with an associated net increase in its present value. For example, if a plant with an estimated t years of remaining life is prematurely closed, the net increase in present value, for a real discount rate of r , becomes $(\$0.3E+9) [1 - 1/(1+r)^t]$.

Thus, a plant closed 20 years early will incur an additional cost of $\$0.2E+8$ for a 7% real discount rate.

5.7.8 Industry Operation

This section provides procedures for estimating industry's incremental costs during the operating phase (i.e., after implementation) of the proposed action. The incremental costs measure the additional costs to industry imposed by the proposed action; they are costs that would not have been incurred in the absence of the action. Reduction in the net cost

(i.e., cost savings) is algebraically positive; increase (i.e., cost accrual) is negative (viewed as negative cost savings). Both NRC and Agreement State licensees should be addressed, as appropriate.

In general, there are three steps that the analyst should follow in order to estimate industry operation costs:

Step 1 - Estimate the amount and types of plant equipment, materials, and/or labor that will be affected by the proposed action.

Step 2 - Estimate the associated costs.

Step 3 - Discount the costs over the remaining lifetimes of the affected facilities, then sum (see Section B.2).

Costs incurred for operating and maintaining facilities may include, but are not limited to, the following:

- maintenance of land and land-use rights
- maintenance of structures
- operation and maintenance of hydraulic, pneumatic, and electrical equipment
- scheduled radioactive waste disposal and health physics surveys
- scheduled updates of records and procedures
- scheduled inspection and test of equipment
- scheduled recertification/retraining of facility personnel
- associated recurring administrative costs
- scheduled analytical updates.

The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows user input for industry (licensee) operation costs.

For the standard analysis, the analyst should proceed as follows:

Step 1 - Estimate the amount and types of plant equipment, materials, and/or labor that will be affected by the proposed regulation, including professional staff time associated with reporting requirements and compliance activities. Possible impacts on a facility's capacity factor should be considered. The analyst may consult with engineering and costing experts, as needed. The analyst could seek guidance from NRC contractors, architect engineering firms, or utilities.

Step 2 - Estimate the associated operation and maintenance costs. The analyst should consider direct and indirect effects of the action; for example, the action could have an impact on plant labor, which, in turn, could affect administrative costs.

Step 3 - Discount the total costs over the remaining lifetime of the affected facilities (see Section B.2).

Much of the discussion on industry implementation costs in Section 5.7.7 for non-reactor facilities applies here for operation costs. Again, the analyst will generally not find consolidated cost information comparable to that for power reactors facilities. As before, Sections C.7-C.10 provide some limited data. However, the analyst may again need to rely on "engineering judgement," although specific data may be available through direct contact with cognizant industry/contractor personnel.

For a major effort beyond the standard analysis, the analyst should seek specific guidance from contractor or industry sources experienced in this area. The user may wish to use contractors who have developed explicit methodologies for estimating operating and maintenance costs. The following references can provide useful information for industry operation costs: Budwani (1969); Carlson et al. (1977); Clark and Chockie (1979); Eisenhauer et al. (1982); EPRI (1986); NUS Corporation (1969); Phung (1978); Roberts et al. (1980); Stevenson (1981); and United Engineers and Constructors, Inc. (1979; 1988a, b).

5.7.9 NRC Implementation

Once a proposed action is defined and the Commission endorses its application, the NRC will incur costs to implement the action. Implementation costs refer to those "front-end" costs necessary to realize the proposed action. All costs associated with pre-decisional activities by the NRC are viewed as "sunk" costs and are excluded from the NRC implementation costs. Reduction in the net cost (i.e., cost savings) is algebraically positive; increase (i.e., cost accrual) is negative (viewed as negative cost savings).

Implementation costs to the NRC may arise from developing procedures, preparing aids, and taking other actions to assist in or assure compliance with the proposed action.⁽¹³⁾ The analyst should determine whether the proposed action will be implemented entirely by the NRC or in cooperation with one or more Agreement States. Implementation costs shared by Agreement States may reduce those of the NRC and are discussed in Section 5.7.11.

NRC implementation costs include only the incremental costs resulting from adoption of the proposed action. Examples of these costs are as follows:

- developing guidelines for interpreting the proposed action and developing enforcement procedures
- preparing handbooks for use by the NRC staff responsible for enforcement and handbooks for use by others responsible for compliance
- supporting and reviewing a licensee's change in technical specifications
- conducting initial plant inspections to validate implementation.

Sciacca (1992) and the FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) assist the analyst in calculating these and "other" implementation costs. Implementation costs may include labor costs and overhead, purchases of equipment, acquisition of materials, and the cost of tasks to be carried out by outside contractors. Equipment and materials that would be eventually replaced during operation should be included under operating costs (see Section 5.7.10) rather than implementation costs.

Three steps are necessary for estimating NRC implementation costs:

Step 1 - Determine what steps the NRC must take to put the proposed action into effect.

Step 2 - Determine the requirements for NRC staff, outside contractors, materials, and equipment.

Step 3 - Estimate the costs of the required resources, discount if appropriate, then sum (see Section B.2).

Implementation is likely to affect a number of NRC branches and offices. For example, the Office of Nuclear Regulatory Research (RES) may develop a regulatory guide, the Office of Nuclear Reactor Regulation (NRR) may review any licensee submissions, and the NRC Regional Offices may inspect against some portion of the guide in operating facilities. In developing estimates for the implementation costs, the analyst is encouraged to contact all of the NRC components likely to be affected by the proposed action.

For the standard analysis, the analyst should identify the major tasks that must be performed to get the proposed rule implemented, major pieces of equipment (if any) that must be acquired, and major costs of materials. Major tasks are then assessed to estimate the approximate level of effort (in professional staff person-hours) necessary to complete them. The number of person-hours for each task is multiplied by the appropriate NRC labor rate and then summed over all of the tasks. In 1996 dollars, the average NRC labor rate (salary and benefits plus allocated agency management and support) is \$67.50/person-hr.⁽¹⁴⁾

Similarly, the costs to complete tasks that would be contracted out also need to be estimated. In order to obtain a reasonably good approximation of contractor costs, the analyst should contact the NRC component that would be responsible for contracting for the tasks. Finally, the costs of major pieces of equipment and quantities of materials are added to the labor and contract costs.

When other data are unavailable, the analyst may assume as an approximation that for a non-controversial amendment to an existing rule or regulation implementation will require the following: a total of one professional NRC staff person-year at a cost of \$122,000/person-year (in 1996 dollars), no additional equipment, and no additional materials. For a new rule or regulation, it is much more difficult to supply a rough but reasonable estimate of the implementation cost, because the level of effort and types and quantities of machinery and materials can vary dramatically. One recourse would be to use as a proxy the implementation costs for a recently adopted regulatory requirement that is similar to the proposed measure. The relative similarity of the two requirements should be judged with respect to the effort required to implement the proposed measure.

For a major effort beyond the standard analysis, a more detailed and complete accounting would be expected. The analyst can request the responsible NRC office to provide available information, such as paper submittals or records of initial inspections.

5.7.10 NRC Operation

After a proposed action is implemented, the NRC is likely to incur operating costs. These are the recurring costs that are necessary to ensure continued compliance. For example, adding a new regulation may require that NRC perform periodic inspections to ensure compliance. The analyst should determine whether operations resulting from the proposed action will be conducted entirely by the NRC or in cooperation with one or more Agreement States. Reduction in the net cost (i.e., cost savings) is algebraically positive; increase (i.e., cost accrual) is negative (viewed as negative cost savings).

There are three steps for estimating NRC operating costs:

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Step 1 - Determine the activities that the NRC must perform after the proposed action is implemented.

Step 2 - Estimate NRC staff labor, contractor support and any special equipment and material required.

Step 3 - Estimate the costs of the required resources, discount (usually over the remaining lifetimes of the affected facilities, as for industry operation costs) to yield present value, then sum (see Section B.2).

In determining the required post-implementation activities, the analyst should carefully examine the proposed action, asking such questions as the following:

- How is compliance with the proposed action to be assured?
- Is periodic review of industry performance required?
- What is an appropriate schedule for such review?
- Does this action affect ongoing NRC programs, and, if so, will it affect the costs of those programs?

Since recurring costs attributable to the proposed action may be incurred by several NRC branches and offices, the analyst is encouraged to contact all of the NRC components likely to be affected. The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows user input for NRC operation costs.

For the standard analysis, the analyst should obtain estimates of the number of full-time equivalent professional NRC staff person-hours that would be required to ensure compliance with the proposed rule. Each person-hour should be costed at \$67.50/person-hr (in 1996 dollars) (refer to endnote 14). Major recurring expenditures for special equipment and materials, and for contractors, should be added. Since operating costs are recurring, they must be discounted as described in Section B.2, usually over the remaining lifetimes of the affected facilities.

A major effort beyond the standard analysis would proceed along the lines described above, except that greater detail would be provided to account for acquisitions of special equipment and materials.

5.7.11 Other Government

This attribute measures costs to the federal government (other than the NRC) and state (including Agreement State) and local governments. The discussion parallels that for NRC implementation and operation in Sections 5.7.9-5.7.10. Reduction in the net cost (i.e., cost savings) is algebraically positive; increase (i.e., cost accrual) is negative (viewed as negative cost savings).

Implementation costs to the federal (non-NRC) government and to state and local governments may arise from developing procedures, preparing aids, supporting license amendments, and taking action to assure compliance with the proposed action. For example, placing roadside evacuation route signs for the possibility of a radioactive release from a nearby power reactor would require expenditures from selected government agencies. As another example, requiring criminal investigation checks for nuclear reactor personnel may require resources of the Federal Bureau of Investigation. When estimating the implementation costs, the analyst should be aware that they may differ between Agreement and non-Agreement States. Such differences should be taken into account in preparing cost estimates.

Three steps are needed to estimate the other government implementation costs:

Step 1 - Determine what steps the other governments must take to put the proposed action into effect.

Step 2 - Determine the requirements for government staff, outside contractors, materials, and equipment.

Step 3 - Estimate the costs of the required resources, discount if appropriate, then sum (see Section B.2).

Implementation is likely to affect a number of government branches and offices. In developing estimates for the implementation costs, the analyst is encouraged to contact all of the government components likely to be affected by the proposed action. The FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996) allows input for other government costs.

For the standard analysis, the analyst should identify the major tasks that must be performed to get the proposed rule implemented, major pieces of equipment (if any) that must be acquired, and major costs of materials. Major tasks are then assessed to estimate the approximate level of effort (in professional staff person-hours) necessary to complete them. The number of person-hours for each task is multiplied by the appropriate labor rate and then summed over all of the tasks.

Similarly, the costs to complete tasks that would be contracted out also need to be estimated. In order to obtain a reasonably good approximation of in-house and contractor costs, the analyst should contact the government agencies that would be responsible for carrying out or contracting for the tasks. Finally, the costs of major pieces of equipment and quantities of materials are added to the labor and contract costs.

After a proposed action is implemented, the federal (non-NRC) government and state and local governments may incur operating costs. These are the recurring costs that are necessary to ensure continued compliance. For example, adding a new regulation may require that other government agencies in addition to the NRC perform periodic inspections to ensure compliance. The analyst should determine whether operations resulting from the proposed action will be conducted entirely by the NRC or in cooperation with one or more other government agencies.

The three steps for estimating the other government operating costs are

Step 1 - Determine the activities that the other governments must perform after the proposed action is implemented.

Step 2 - Estimate government staff labor, contractor support, and any special equipment and material required.

Step 3 - Estimate the costs of the required resources, discount (usually over the remaining lifetimes of the affected facilities, as for NRC operation costs) to yield present value, then sum (see Section B.2).

In determining the required post-implementation activities, the analyst should carefully examine the proposed action, asking such questions as the following:

- Does compliance with the proposed action require non-NRC cooperation?
- Is periodic review of industry performance required beyond that of the NRC?
- What is an appropriate schedule for such review?
- Does this action affect ongoing government programs, and, if so, will it affect the costs of those programs?

Since recurring costs attributable to the proposed action may be incurred by several government branches and offices, the analyst is encouraged to contact all components likely to be affected.

For the standard analysis, the analyst should obtain estimates of the number of full-time equivalent professional staff person-hours that would be required to ensure compliance with the proposed rule. Each person-hour should be costed at the appropriate labor rate (an average NRC labor rate of \$67.50/person-hr [in 1996 dollars] maybe used as a substitute if no more specific value is available [refer to endnote 14]). Major recurring expenditures for special equipment and materials, and for contractors, should be added. Since operating costs are recurring, they must be discounted as described in Section B.2, usually over the remaining lifetimes of the affected facilities.

A major effort beyond the standard analysis would proceed along the lines described above, except that a more detailed and complete accounting would be expected. The analyst could request the responsible government agencies to provide available information.

5.7.12 General Public

This attribute measures costs incurred by members of the general public, other than additional taxes, as a result of implementation of a proposed action. Taxes are viewed simply as transfer payments with no real resource commitment from a societal perspective. Reduction in the net cost (i.e., cost savings) is algebraically positive; increase (i.e., cost accrual) is negative (viewed as negative cost savings).

Typically, costs to the general public cover such items as increased cleaning due to dust and construction-related pollutants, property value losses, or inconveniences, such as testing of evacuation sirens. Care must be taken not to double count for general public and other government costs. If a cost could be assigned to either group, it should be assigned where more appropriate, the analyst remembering not to account for it again in the other attribute.

The two steps to estimate costs to the general public are as follows:

Step 1 - Identify the adverse impacts incurred by the general public to implement the proposed action.

Step 2 - Estimate the costs associated with these adverse impacts, discount if appropriate, then sum (see Section B.2).

This attribute is not expected to be one commonly affected by regulatory actions. However, if relevant, the standard analysis would require the analyst to identify the major activities to implement the proposed action that will result in adverse impacts to the general public. Public records or analogous experience from other communities could be used as information sources to estimate the costs to the general public.

5.7.13 Improvements in Knowledge

This attribute relates primarily to proposals for conducting assessments of the safety of licensee activities. At least four major potential benefits are derived from the knowledge produced by such assessments:

- improvements in the materials used in nuclear facilities
- improvement or development of safety procedures and devices
- production of more robust risk assessments and safety evaluations, reducing uncertainty about the relevant processes

- improvement in regulatory policy and regulatory requirements.

To the extent that the effects of regulatory actions can be quantified, they should be treated under the appropriate quantitative attributes. On the other hand, if the effects from the assessments are not easily quantified, the analyst still has the burden of justifying the effort and providing some indication of its effect. If necessary, this justification would be expressed qualitatively under this attribute. An effort should be made to identify the types of values and impacts that are likely to accrue and to whom.

Consider the following statement:

This assessment effort has a reasonable prospect of reducing our uncertainty regarding the likelihood of containment failure resulting from hydrogen burning. Such an accident may be a significant source of risk. The knowledge from the proposed assessments would enable us to assess more accurately the overall accident risk posed by nuclear reactors, and this in turn should benefit the public through better policy decisions.

While this statement describes why the proposed assessment is needed, no information is provided for evaluating the merits of the proposed assessment.

Providing answers to the following questions would help to fill this information gap:

- What are the likely consequences of a hydrogen-burning accident?
- To what extent would the proposed assessment reduce the uncertainty in the likelihood of a hydrogen-burning accident?
- Given our current information, what is the contribution of hydrogen burning to overall accident risk?

The above questions are specific to a particular topic. For the broader problem of providing a value-impact analysis of an assessment proposal, it is recommended that the analyst be responsive to the following list of more general questions:

- What are the objectives?
- If the assessment is successful in meeting its objectives, what will be the social benefits?
- Is there a time constraint on the usefulness of the results?
- Who will benefit from the results, by how much, and when?
- What is the likelihood that the assessment will fail to meet its objectives within the time and budget constraints?
- What will be the social costs (and benefits) if the assessment is not successful, or if the assessment is not undertaken?

5.7.14 Regulatory Efficiency

Regulatory efficiency is an attribute that is frequently difficult to quantify. If it can be quantified, it should be included under one or more of the other quantifiable attributes. If quantification is not practical, regulatory efficiency can be treated in a qualitative manner under this attribute. For example achieving consistency with international standards groups may increase regulatory efficiency for both the NRC and the groups. However, this increase may be difficult to quantify.

If necessary, this justification would be expressed qualitatively under this attribute. An effort should be made to identify the types and recipients of values and impacts likely to accrue. If the proposed NRC action is expected to have major effects on regulatory efficiency, then a proper evaluation of these effects may require a level of effort commensurate with their magnitude. This may mean expending resources to obtain the judgments of experts outside of the NRC if the necessary expertise is not available in-house.

To obtain useful information, the analyst can solicit expert opinion in a number of ways. A general discussion of those methods and others is found in Quade (1975), especially Chapter 12, "When Quantitative Models are Inadequate." One way is to convene the experts in a round-table discussion with the objective of reaching a consensus. This technique has some of the drawbacks of a committee meeting--often the assumptions are not made explicit, and strong-willed (or strong-voiced) individuals often carry undue weight.

Another way of pooling expert opinion in a systematic manner is to use one of the numerous procedures for iterative group decision-making. For example, the Delphi technique (Dalkey and Helmer 1963; Humphress and Lewis 1982) is a procedure that features an anonymous exchange of information or expert opinion. This approach is designed to encourage the modification of earlier answers by each expert so that a group consensus can be achieved. Even if consensus is not achieved, information is produced that allows the analyst to compile statistical estimates of the responses.

Whether the assessment is performed by a panel of experts or by the analyst, the following are questions that might be considered in order to focus on that assessment:

- Does this action conflict with any other NRC/federal/state directives?
- Are there any nuclear facilities for which (or conditions under which) this action might have unexpected or undesirable consequences?
- Do you foresee any major enforcement problems with this action or regulation?
- What sort of adjustments might industry undertake to avoid the regulation's intended effects?
- How will the regulation impact productivity in the nuclear/electric utility industries?
- How will this action affect facility licensing times?
- How will this action affect the regulatory process within the NRC (and/or other regulatory agencies)?

5.7.15 Antitrust Considerations

This qualitative attribute is not expected to be one commonly affected by regulatory actions. However, the NRC does have a legislative mandate in Section 105 of the Atomic Energy Act to uphold the antitrust laws. Thus, this attribute can be relevant for those proposed actions which may potentially violate the antitrust laws. If applicable, antitrust considerations should be explored with the NRC Office of the General Counsel early in the analysis to preclude analyzing an issue clearly in conflict with these laws. If antitrust considerations are involved, and it is determined that antitrust laws would be violated, then the proposed action must be reconsidered and, if necessary, redefined to preclude such violation.

5.7.16 Safeguards and Security Considerations

Safeguards and security considerations include protection of the common defense and security and safeguarding restricted data and national security information. In more practical terms, this means providing adequate physical security and safeguards systems to prevent the diversion of certain types of fissionable and radioactive materials, the perpetration of acts of radiological sabotage, and the theft by unauthorized individuals of restricted data or national security information.

The NRC has a legislative mandate in the Atomic Energy Act to assure the objectives mentioned above. Through its regulations and regulatory guidance, the NRC has established a level of protection deemed to satisfy the legislative mandate. As is the case for adequate protection of the health and safety of the public, this level of protection must be maintained without consideration of cost.

While quantification of safeguards and security changes may be difficult, the analyst should attempt quantification when feasible. If this process is impossible, the analyst may proceed with a qualitative analysis under this attribute. Section 5.7.14, where methods of evaluating expert opinion are discussed, may be helpful.

5.7.17 Environmental Considerations

Section 102 of the National Environmental Policy Act (NEPA) requires federal agencies to consider environmental impacts in the performance of their regulatory missions. NRC's regulations implementing NEPA are in 10 CFR Part 51. Any documentation prepared to satisfy NEPA and Part 51 should be coordinated with any regulatory analysis documentation covering the same or similar subject matter as much as possible.

Environmental impacts can have monetary effects (e.g., environmental degradation, mitigation measures, environmental enhancements), which could render potential alternative actions unacceptable or less desirable than others. Therefore, at a minimum, such effects should be factored into the value-impact analysis, at least to the extent of including a summary of the results of the environmental analysis.

Many of the NRC's regulatory actions are subject to categorical exclusions as set forth in 10 CFR 51.22. In these cases, detailed environmental analyses are not performed, and there will be no environmental consideration to factor into the regulatory analysis. In some cases, a generic or programmatic environmental impact statement (EIS) is prepared. If such is the case, Section 5.3 of the Guidelines allows portions of the EIS to be referenced in lieu of performing certain elements of the regulatory analysis. In the remaining cases, it may be that the regulatory analysis alternative being considered will initiate the requirement for review of environmental effects. For purposes of the regulatory analysis document, the preferred approach to be used in this situation is to perform a preliminary environmental analysis, identifying in general terms anticipated environmental consequences and potential mitigation measures. The results of this preliminary analysis should be quantified under the appropriate quantitative attributes, if possible, or addressed qualitatively under this attribute, if not quantified.

5.7.18 Other Considerations

There may be other considerations associated with a particular proposed action that are not captured in the preceding descriptions. Possible examples might include the way in which the proposed action meets specific requirements of the Commission, EDO, or NRC office director that requested the regulatory analysis; the way in which the proposed action would help achieve NRC policy; or advantages or detriments that the proposed action would have for other NRC programs and actions. If quantifiable, the effect should be included in essentially the same way as in the quantitative attributes. Because such considerations would be expected to be unusual, some additional discussion in the regulatory analysis document should be provided.

Value-Impact

The analyst needs to give thoughtful consideration to the possible effects of the proposed action. Some of the effects may not be immediately obvious. The analyst may wish to consult with other knowledgeable individuals to aid in the identification of all significant effects. These considerations need to be presented clearly to facilitate the reader's understanding of the issues.

When quantification of effects is not feasible, the analyst may still be able to provide some indication of the magnitude to facilitate comparison among alternatives, and comparison with quantifiable attributes. Comparative language (greater than, less than, about equal to) can be very helpful in achieving this objective, as long as the analyst can make the necessary judgements. Consultation with experts or other knowledgeable individuals may be required.

5.8 Summarization of Value-Impact Results

Having completed the value-impact analysis for one or more alternatives of the proposed action, the analyst should summarize the results for each alternative using a summary table such as that shown as Figure 5.1. Such a tabular

Title of Proposed Action / Date

Summary of Problem and Proposed Solution:

Quantitative attribute		Present value estimates (\$)		
		Low ^(a)	Best ^(b)	High ^(c)
Health	Public	Accident		
		Routine		
	Occupational	Accident		
		Routine		
Property	Offsite			
	Onsite			
Industry	Implementation			
	Operation			
NRC	Implementation			
	Operation			
Other Government				
General Public				
NET VALUE (Sum)				

(a) Low estimates correspond to the worst case, i.e., highest costs and lowest benefits, relative to the baseline case.
 (b) Best estimates are normally the expected value, but could be other point estimates such as the mean or median (see Section 4.3 of the Guidelines).
 (c) High estimates correspond to lowest cost estimates and highest benefit estimates.

Comments: Discuss any other attributes considered, compliance with Safety Goal guidance, special considerations, etc.

Figure 5.1 Summary of value-impact results

presentation provides a uniform format for recording the results of the evaluation of all quantitative attributes plus a comments section to discuss other attributes considered, compliance with the Safety Goal guidance, special considerations, etc. It displays the results for the net-value measure, discussed in Section 5.2.

All dollar measures should be present valued and expressed in terms of the same year. This may require conversion of some dollar values from whatever years in which they have been expressed to one common year. Sciacca (1992) describes techniques for these conversions. The Gross Domestic Product (GDP) price deflator can be used to convert historical nominal dollars to dollars of one common year. Financial publications, such as *National Economic Trends* by the Federal Reserve Bank of St. Louis, supply implicit price deflators for the GDP, through the current year. GDP price deflator information from the Federal Reserve Bank of Chicago is also available at the following Internet address: <http://gopher.great-lakes.net:2200/0/partners/ChicagoFed/econind/>.

When recording the low and high estimates for an attribute, the analyst should generally record the lowest and highest estimates if multiple estimates are made. For example, suppose the analyst calculated a best estimate of $-\$5.0\text{E}+5$ for NRC implementation cost (the negative value indicates the cost will be an expense rather than a savings). The analyst then performed two separate sensitivity analyses, obtaining the following sets of low (more negative) and high (less negative) estimates:

	<u>Low Estimate</u>	<u>High Estimate</u>
Sensitivity A	$-\$7.5\text{E}+5$	$-\$2.5\text{E}+5$
Sensitivity B	$-\$1.0\text{E}+6$	$-\$3.0\text{E}+5$

The analyst should record the lowest (most negative) and highest (least negative) estimates in Figure 5.1 (i.e., $-\$1.0\text{E}+6$ and $-\$2.5\text{E}+5$, respectively), even though each comes from a different sensitivity analysis.

The net value is the required value-impact measure (see Section 5.2). Its calculation is the sum of the present value of all the quantitative attributes. Information on computing present value is in Section B.2. A positive net value result indicates an overall cost savings for the proposed action. A negative net value result indicates the opposite. As mentioned in Section 5.2, the net value is an absolute measure, reflecting the magnitude of the proposed action's contribution toward the specified goals. The results of the value-impact assessment can be displayed as a ratio and in tables and/or graphs, in addition to a summary table for additional perspectives.

5.9 Endnotes for Chapter 5

1. Section 4.4 of the Guidelines allows the analyst to display the results of a value-impact analysis as a ratio of values to impacts, all expressed in dollars. The numerator would sum the estimates for all quantifiable attributes classified as values, while the denominator would do likewise for impacts. Section 4.4 of the Guidelines views a value-impact ratio as supplemental to the net value, not as a replacement.
2. The term "equation" is loosely used to indicate anything from a single mathematical expression (e.g., one for a major fire at a non-reactor facility) to a complete computer analysis (e.g., a core damage assessment for a power reactor).
3. The double index notation indicates that an initiating event j can lead to several accident sequences i .

4. Level 1 analyses generally produce a list of core-damage accident sequences, together with the overall core-damage accident frequency as their final product. Level 2 analyses take the Level 1 analyses one step further by evaluating the containment response to the accident sequences and the associated containment release magnitudes. Level 3 analyses take the Level 2 analyses one step further by evaluating the public risk associated with the containment release frequencies and magnitudes. As a result, Level 3 analyses are the preferred tools for evaluating the effect of a proposed action on public risk.
5. Developed by the Southwest Research Institute, San Antonio, Texas.
6. An error factor f is used as follows to estimate upper and lower bounds, presuming a positive value for the best estimate:

Upper Bound = Best Estimate $\times f$
Lower Bound = Best Estimate / f
7. As discussed in Section 5.7.1.1, public health (accident) may be affected through a mitigation of consequences instead of (or as well as) a reduction in accident probability.
8. Andrews et al. (1983) provide a conceptual discussion of assessing the risk for this type of proposed action.
9. The equations included in this Handbook (e.g., Section 5.7.1.3) apply a discounting term to doses associated with both implementation and operational impacts. In practice, the implementation dose may be of such short duration that discounting is not necessary. Its inclusion here is in recognition that, in some cases, implementation may extend over a longer period than one year.
10. NRC has required its contractors to estimate onsite dose rates in the Surry and Grand Gulf risk assessments during low power and shutdown operations (Brown et al. 1992; Jo et al. 1992).
11. Based on ANL estimates, a cleanup period as long as 10 years may be needed following a major power reactor accident (see Section 5.7.6.1). Long-term doses will occur over some portion of this time.
12. Accidents at non-reactor nuclear facilities could also lead to the need for replacement services of the same type provided by the facility where the accident occurred.
13. NRC implementation costs associated with facility closure may be increased if the facility closes prematurely (see Section 5.7.7.2).
14. The \$67.50 hourly rate is derived from June 1996 data and the technique described in Abstract 5.2 of Sciacca (1992).

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Appendix A

Regulatory Analysis Issues



Appendix A

Regulatory Analysis Issues

This appendix addresses three topics of particular interest in connection with the performance of regulatory analyses. Owing to the special nature or extensiveness of these topics, it was judged best to discuss them here rather than in the main body of this Handbook, as has been done with other issues. The topics are human factors issues, cumulative accounting of past and ongoing safety improvements, and use of industry risk and cost estimates.

A.1 Human Factors Issues

Regulatory analyses involving proposed actions related to human factors issues often prove to be difficult to quantify, especially with regard to risk-related attributes. This degree of difficulty varies to the extent that the human factors issue is "concrete" or "abstract." For example, an issue proposing to clarify standard procedures for hardware inspection can be perceived as fairly concrete. Inspection personnel can be expected to perform more efficiently with less likelihood of error during the inspection procedure. This would decrease the likelihood of overlooking a hardware defect. Such an issue can be translated into a reduced unavailability for selected hardware components, several of which most likely appear in a facility risk equation. For such a human factors issue, the expected improvement can be treated as an improvement in the reliability of the hardware itself. Thus, this "concrete" human factors issue can be analyzed in a manner similar to any other hardware issue.

As an "abstract" example, consider a human factors issue proposing to revise management guidelines for a power plant. Difficulty is foreseen in directly linking this action to parameters in a plant risk equation. One approach might be to assume some small improvement in the portion of the unavailability due to human error in each risk parameter as appropriate. The analysis then could proceed as in a hardware issue, except that many parameters might be affected, thereby complicating the calculations. Studies completed by Samanta et al. (1981, 1989) and Andrews et al. (1985), discussed in Section A.1.1, provide results which can facilitate these types of calculations.

As an alternative, an approach similar to that discussed in Section 5.6.2 may be appropriate. For fairly "nebulous" issues (i.e., ones where the reductions in accident frequency [and/or risk] are difficult to quantify directly via a facility risk equation) expert judgment of the changes in the accident frequency (and/or risk) can be based on the total accident frequency (and/or risk). Employing informal procedures or a formalized one such as the Delphi method (Dalkey and Helmer 1963; Humphress and Lewis 1982), the analyst can obtain a consensus estimate of the percent change in total accident frequency (and/or risk) due to implementation of the proposed action. This may be the best that can be done for the more "abstract" human factors issues.

Several studies have been conducted to address quantification of human error probabilities (HEPs) for nuclear power plant risk analyses. The initial standard for human error analysis, subsequently named the Technique for Human Error Rate Prediction (THERP), was established by the complementary documents NUREG/CR-1278 (Swain and Guttman 1983) and NUREG/CR-2254 (Bell and Swain 1983). Swain and Guttman (1983) developed a handbook of human performance models and procedures for estimating HEPs, including numerical values, for application in nuclear power plant risk

analyses. In its sister document (NUREG/CR-2254), Bell and Swain (1983) detailed a standard procedure to conduct a human reliability analysis for nuclear power plants, emphasizing an event tree approach which utilizes results from NUREG/CR-1278. Swain (1987) supplemented the THERP with a simplified version in NUREG/CR-4772, intended "to enable systems analysts, with minimal support from experts in human reliability analysis, to make estimates of human error probabilities and other performance characteristics which are sufficiently accurate for many probabilistic risk assessments."

Additional studies which can assist the analyst in performing a regulatory analysis, particularly the value-impact portion, for a human factors issue can be grouped into two categories:

1. Documents addressing methods to estimate HEPs, sometimes including numerical results for applying these methods (see Section A.1.2). The previous studies plus a trio by Stillwell et al. (1982), Seaver and Stillwell (1983), and Comer et al. (1984) are examples of these "methods" documents.
2. Documents presenting the results of quantifying the impact of HEPs on a nuclear power plant's overall core-melt frequency and/or public risk (see Section A.1.1). A pair of studies by Samanta et al. (1981, 1989) and one by Andrews et al. (1985) are examples of these "results" documents.

Documents from each group have been reviewed, and summaries are provided in the remainder of this appendix section. We begin with studies from the second group.

A.1.1 Results Documents

In a pair of studies, Samanta et al. (1981, 1989) evaluated the sensitivity of selected risk parameters to changes in HEPs for a pair of representative PWRs. The first study (NUREG/CR-1879 [Samanta et al. 1981]) quantified the effect of changing HEPs for the Surry PWR on the following parameters: system unavailability, accident sequence frequency, core-melt frequency, and release category frequency. The Human Error Sensitivity Assessment of a PWR (HESAP) computer code was developed to model the human errors in fault trees based on the Surry plant as modeled in WASH-1400 (NRC 1975a). HEPs were both increased and decreased by factors of 3, 10, 20, and 30 relative to selected base-case values. Numerous tables and figures give the results of simultaneously varying all HEPs by these factors in terms of the changes in the four risk parameters listed above.

In addition, Samanta et al. (1981) estimated the sensitivity of core-melt and release category frequencies to changes in probabilities for generic classes of human error (e.g., operator error, maintenance error, and errors of omission/commission). Also, individual human errors were ranked relative to one another in terms of their structural importance to core-melt frequency and their reliability importance to core-melt and release category frequencies (Vesely et al. 1983). The results are conveniently presented as tables and figures.

The second study (NUREG/CR-5319 [Samanta et al. 1989]) updated the first using the more recent, and more detailed, Oconee PWR risk assessment performed by EPRI and Duke Power Co. (1984). Only the portion of the Oconee risk assessment pertaining to internal events was employed by Samanta et al. External events were not included. The effect of changing HEPs on the following risk parameters was evaluated: accident sequence frequency, core-melt frequency, and core-melt bin frequency (somewhat analogous to release category frequency). Statistical methods were employed to estimate factors by which HEPs could be both increased and decreased realistically. Factors ranging as high as 26 were calculated, depending upon the type of human error (an additional degree of resolution relative to the first study).