**Indian Point Nuclear Generating Units 2 and 3**
***Docket Nos. 50-247/ 50-286-LR***

**NRC Staff's Response in Opposition to State of New York's Motion for Partial Summary Disposition of NYS Contention 16/16A**

# Exhibit J

NUREG/BR-0184

United States
Nuclear Regulatory Commission

# Regulatory Analysis Technical Evaluation Handbook

Final Report

Office of Nuclear Regulatory Research

January 1997

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## 5.4 Treatment of Uncertainty

Chapter 4 of the NRC Guidelines requires that uncertainties be addressed in regulatory analyses, both for exposure and cost measures. In addition, NRC's Final Policy Statement on the use of probabilistic risk assessment (PRA) in nuclear regulatory activities (NRC 1995b) states that sensitivity studies, uncertainty analysis, and importance measures should be used in regulatory matters, where practical within the bounds of the state-of-the-art. Uncertainties in exposure measures, especially those related to facility accidents, have traditionally been difficult to estimate. With respect to power reactor facilities, much has been written about uncertainty analysis in risk assessments. The more rigorous assessments typically provide an uncertainty analysis, usually performed via stochastic simulation on a computer. Briefly, the analyst determines probability distributions for as many of his input parameters as deemed necessary and practical. A computer code then samples values from each distribution randomly and propagates these values through the risk equation to yield one result. When repeated a large number of times (at least several hundred), a probability distribution for the result is generated, from which the analyst can extract meaningful statistical values (e.g., mean, standard deviation, median, and upper and lower bounds for given confidence levels).

Risk assessments for non-reactor facilities often identify best estimates only. Some have provided uncertainty ranges (see Appendix C), but their development has generally been less rigorous than that for reactor facilities. On the positive side, accident scenarios for non-reactor facilities are much less complex than for power reactors, facilitating uncertainty estimation, at least from a calculational perspective.

This Handbook is not intended to provide basic information on probability and statistics, and therefore does not attempt to describe the details of uncertainty analysis techniques. The analyst needing information on these topics is referred to textbooks on probability and statistics, as well as the following references: Seiler (1987), Iman and Helton (1988), Morgan and Henrion (1990), and DOE (1996). Instead, this Handbook presents a general discussion of the types of uncertainty that will be encountered in a regulatory analysis, primarily the value-impact portion, and outlines some of the more recent approaches to deal with them.

### 5.4.1 Types of Uncertainty

Vesely and Rasmuson (1984) identified seven categories of uncertainties in PRA, the majority of which, if treated at all, have only recently begun to receive attention. The seven categories are uncertainties in data, analyst assumptions, modeling, scenario completeness, accident frequencies, accident consequences, and interpretation. These seven categories, going from first to last, represent a progression from uncertainties in the PRA input to higher-level uncertainties with the PRA results. Vesely and Rasmuson considered these categories to be generally applicable to any modeling exercise, not just a PRA. Thus, they would also apply to the cost analysis portion of the regulatory analysis.

The first category, data uncertainty, is the most familiar and most often treated. It can be divided into four groups: population variation, imprecision in values, vagueness in values, and indefiniteness in applicability. Population variation refers to parameter changes from scenario to scenario, usually due to physical causes. The variations occur among the random variables which, when treated as constants, give a false impression of the stability of the results. Parameter imprecision and vagueness refer to separate concepts. Imprecision occurs when only limited measurements are available from which to estimate parameter values. Vagueness occurs when definitive values or intervals cannot be assigned to parameters. Indefinite applicability deals with the extrapolation of parameter values to situations different from those for which they were derived (e.g., extrapolating component failure data for normal environments to accident conditions).

The second category, analyst uncertainty, refers to variations in modeling and quantification which arise when different analysts perform different portions of the analysis. Often included with data uncertainty, analyst uncertainty provides its own separate contribution. Modeling uncertainty, the third category, arises from the indefiniteness in how comprehensive

and how well characterized are the numerous models in the analysis. Do the models account for all significant variables? How well do the models represent the phenomena? Is the dependence between two phenomena accurately modeled? Similar to modeling uncertainty is completeness uncertainty, the fourth category. It differs only in that it occurs at the initial, identification stage in the analysis. When the analytic "boundaries" are drawn at the start of the analysis, how can one be sure that all "important" items have been included (e.g., the Three-Mile Island core-damage scenario was not specifically identified in PRAs until it had occurred)? Even if the important items have been included, are their interrelationships adequately defined (if even known)?

The last three uncertainty categories—those for accident frequencies and consequences, and interpretation—deal with the analytic output and results. Accident frequency uncertainties arise from two sources: variations between accidents of the same type and limited knowledge of the data, models, and completeness. Accident consequence uncertainties parallel those in accident frequency, except that they involve consequence modeling rather than frequency estimation. Interpretation uncertainty arises from the combination of all previous uncertainties plus the difficulty in conveying the information to the decision-maker. Even the most precise uncertainty analysis can be wasted if the meaning cannot be transferred to the decision-maker. Often, this results from difficulty in the way the results are presented. Ernst (1984) provides insight on reducing the uncertainty in interpretation of results.

### 5.4.2 Uncertainty Versus Sensitivity Analysis

As defined by Vesely and Rasmuson, uncertainty and sensitivity analyses are similar in that both strive to evaluate the variation in results arising from the variations in the assumptions, models, and data. However, they differ in approach, scope, and the information they provide.

Uncertainty analysis attempts to describe the likelihood for different size variations and tends to be more formalized than sensitivity analysis. An uncertainty analysis explicitly quantifies the uncertainties and their relative magnitudes, but requires probability distributions for each of the random variables. The assignment of these distributions often involves as much uncertainty as that to be quantified.

Sensitivity analysis is generally more straightforward than uncertainty analysis, requiring only the separate (simpler) or simultaneous (more complex) changing of one or more of the inputs. Expert judgment is involved to the extent that the analyst decides which inputs to change, and how much to change them. This process can be streamlined if the analyst knows which variables have the greatest effect upon the results. Variation of inputs one at a time is preferred, unless multiple parameters are affected when one is changed. In this latter case, simultaneous variation is required. Hamby (1993) provides a detailed description of the most common techniques employed in sensitivity analysis.

Vesely and Rasmuson identify which of the seven types of uncertainties encountered in PRAs are best handled by uncertainty versus sensitivity analysis. They are as follows:

1. Data Uncertainty: Use uncertainty analysis for population variation and value imprecision, sensitivity analysis for value vagueness and indefiniteness in applicability.

2. Analyst Uncertainty: Use sensitivity analysis.

3. Modeling Uncertainty: Use sensitivity analysis.

4. Completeness Uncertainty: Use sensitivity analysis.

5. Frequency Uncertainty: Use uncertainty analysis for variation from one accident to another, sensitivity analysis for the limited knowledge of the data, models, and completeness.

6. Consequence Uncertainty: Use uncertainty analysis for variation from one accident to another, sensitivity analysis for the limited knowledge of the data, models, and completeness.

7. Interpretation Uncertainty: Use sensitivity analysis.

### 5.4.3 Uncertainty/Sensitivity Analyses

Three major NRC studies involving detailed uncertainty/sensitivity analyses were NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants* (NRC 1991); NUREG/CR-5381, *Economic Risk of Contamination Cleanup Costs Resulting from Large Non-Reactor Nuclear Material Licensee Operations* (Philbin et al. 1990); and NUREG/CR-4832, *Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)* (Payne 1992). The first and third studies address reactor facilities, the second non-reactor facilities. The approach used in each study is summarized below.

#### 5.4.3.1 NUREG-1150

"An important characteristic of the PRAs conducted in support of this report [NUREG-1150] is that they have explicitly included an estimation of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena." With this introduction, NUREG-1150 identified four steps in the performance of its uncertainty/sensitivity analysis:

1. Define the Scope. The total number of parameters that could be varied to produce uncertainty estimates was quite large and limited by computer capacity. Thus, only the most important sources were included, these sources being identified from previous PRAs, discussion with phenomenologists, and limited sensitivity analyses. For those parameters important to risk and having large uncertainties and limited, if any, data, subjective probability distributions were generated by expert panels.

2. Define Specific Uncertainties. Each section of the risk assessment was conducted at a slightly different level of detail, none of which to the degree involved in a mechanistic analysis. This resulted in the uncertain input parameters being "high level" or summary parameters, for which their relationships with their fundamental physical counterpart parameters were not always clear. This resulted in Vesely and Rasmuson's "modeling uncertainties." In addition, "data uncertainties" arose from limited knowledge of some important physical or chemical parameters. NUREG-1150 included both types of uncertainty, with no consistent effort to distinguish between them.

3. Define Probability Distributions. Probability distributions were developed by several methods, paramount among these being "expert elicitation" (discussed below). "Standard" distributions employed in previous risk assessments were used when the experts' estimation was not needed.

4. Combination of Uncertainties. The Latin hypercube method, a specialized form of stochastic simulation, was employed to sample from the various probability distributions. The sampled values were propagated through the constituent analyses to produce probability distributions for core damage frequency and risk. Results were presented graphically as histograms and complementary cumulative distribution functions showing the mean, median, and two-sided 90% confidence intervals.

A major innovation of the NUREG-1150 project was the development of a formal method for elicitation of expert judgment. Nine steps were involved:

1. Selection of Issues. The initial list of issues was identified from the important uncertain parameters specified by each plant analyst.

2. Selection of Experts. Seven expert panels were assembled to address issues in accident frequency (two panels), accident progression and containment loading (three panels), containment structural response (one panel), and source terms (one panel). Selection was based on recognized expertise in the nuclear industry, the NRC and its contractors, and academia. Each panel contained 3-10 experts.

3. Elicitation Training. Decision analysis specialists trained both the experts and analysis team members in elicitation methods, including the psychological aspects of probability estimation. The experts perfected their estimation techniques by conjuring probabilities for items for which "true" values were known.

4. Presentation and Review of Issues. The analysis staff formally presented the relevant issues to each panel over the course of several days. Interactive discussions ensued.

5. Preparation of Expert Analyses. Over a periods ranging from one to four months, each panel deliberated on its issues. However, each panel member arrived at his/her own quantitative results.

6. Expert Review and Discussion. At a final meeting, each expert presented his/her analysis which, in some cases, resulted in members modifying their preliminary results subsequent to the meeting.

7. Elicitation of Experts. Two analysis staff members, one trained in elicitation techniques, the other familiar with the technical subject, interviewed each expert privately. The expert's final quantitative results were documented.

8. Aggregation of Judgments. From each expert's results, the analysis staff composed probability distributions which were then aggregated to produce a single composite for each issue. Each expert was equally weighted in the composite.

9. Review by Experts. Each expert's probability distribution, as developed by the analysis staff from the expert's interview, was reviewed privately with that expert to correct any misconceptions that may have arisen. The probability distribution was then finalized, as was the composite.

#### 5.4.3.2 NUREG/CR-5381

In NUREG/CR-5381, Philbin et al. took advantage of some of the convenient combinatorial properties of the lognormal distribution to facilitate a straightforward uncertainty analysis. NUREG/CR-5381 assessed the economic risk of cleanup costs resulting from non-reactor NRC licensee contamination incidents (see Section C.4). The calculational procedure involved three steps: estimating the frequency and cleanup cost of each accident scenario, taking their product to yield the "cleanup risk" (probabilistically-weighted cleanup cost) per scenario, and summing the scenario risks to yield the total facility risk. The uncertainty analysis paralleled these three steps.

For both the accident frequency and cleanup cost, probability distributions were selected from the available data, if possible, or by expert judgment. When using historical data to obtain frequency estimates, the assumption was made that the number of incidents for a specified scenario followed the Poisson distribution. This was deemed reasonable in light of the small number of incidents over a relatively large number of operating years and the absence of any obvious trends. The

Poisson point estimate incident rate was taken to be the historical rate, with two-sided 80% confidence bounds derived from the properties of the Poisson distribution.

When a calculational model was used to estimate the frequency, the uncertainty was based on expert judgment. Unless deemed inappropriate, the frequency distribution was taken to be lognormal with an error factor of 10. If previous analyses provided only a frequency range, the distribution was again assumed to be lognormal, with the upper and lower bounds taken as the endpoints of this range. Thus, the point estimate (median, in this case) became their geometric mean. For the cleanup costs, the point estimates were derived from historical data of calculational models. These costs were assumed to be lognormally distributed with error factors of 1.25.

Philbin et al. defended their choice of the lognormal as a "generically" representative probability distribution for several reasons. The lognormal has a minimum value of zero, a realistic limit on the minimum frequency and cost, and is skewed in a way which yields relatively wider error bounds on the upper than lower side. Thus, it produces an uncertainty band which is conservative. Also, the lognormal has two convenient combinatorial properties. The product of two lognormally distributed variables is lognormally distributed, while the sum can be approximated by another lognormal provided one variable dominates the other.

The economic risk per accident scenario was estimated by propagating the frequency and cost uncertainties through their product. When both frequency and cost were lognormally distributed, this product was also lognormal. When the frequency distribution was Poisson, it was approximated by a lognormal to simplify the calculation. Each scenario thus resulted in an economic risk which was lognormally distributed. These were summed to yield the total economic risk per facility. The individual variances were summed and the resultant total economic risk was assumed to be approximately lognormal, a reasonable assumption if it was dominated by one scenario risk. Referring to Tables C.4-C.8 in Section C.4, one can see that this assumption was generally valid for three of the five facilities (i.e., one scenario risk contributed over 50% to the total facility risk). The final results were reported as two-sided 80% confidence bounds.

#### 5.4.3.3 NUREG/CR-4832

In NUREG/CR-4832, Payne generally followed an uncertainty/sensitivity calculational procedure similar to that employed in NUREG-1150. The major contribution was the development of a new computer code, TEMAC (Iman and Shortencarier 1986) to perform the final quantification of the accident sequence uncertainties via the Latin hypercube sampling method. The TEMAC code also calculated various risk importance measures (Vesely et al. 1983) and ranked the basic events by their contribution to mean core damage frequency.

Three importance measures were estimated in NUREG/CR-4832. The first, risk reduction importance, calculates the decrease in the total core damage frequency which could result if a single basic event's probability were set to zero (i.e., the component could not fail or the event could not occur). The second, risk increase importance, calculates the increase in the core damage frequency which could result if a single basic event's probability were set to one (i.e., the component would always fail or the event would always occur). The third, uncertainty importance, estimates the extent to which the uncertainty in the total core damage frequency depends upon the underlying uncertainty in a common contributor to a set of related basic events (e.g., a failure to actuate in all motor-operated valves). These importance measures represent a combination of sensitivity with uncertainty analyses which feature some of the better aspects of each.

### 5.4.4 Suggested Approach

The value-impact portion of a regulatory analysis will often require use of an existing risk assessment for the estimation of some of the attributes. If the risk assessment has an uncertainty/sensitivity analysis accompanying it, the analyst should

try to adapt it for use in the value-impact analysis. Unfortunately, this is often impractical for the standard analysis since the analyst does not have access to the computer code and numerous data and assumptions necessary to generate the resultant probability distributions.

When a detailed uncertainty/sensitivity analysis is not possible or practical, the following approach is suggested for the standard analysis. The standard analysis should attempt to include an uncertainty/sensitivity analysis approaching the level of that conducted by Philbin et al. in NUREG/CR-5381 (see Section 5.4.3.2). This analysis can be done with varying degrees of formality and rigor. First, a systematic attempt should be made to identify all of the pertinent factors (assumptions, data, models) that could affect the results. Since the number of such factors is usually very large, not all of them can be treated in detail. Nevertheless, it is useful to make a systematic effort at least to identify them. As a second step, the list of factors should be screened to select a subset for detailed examination. The screening process should concentrate on eliminating unimportant factors (for example, those that are known to contribute little to the overall uncertainty or those that have minimal effect on the bottom line results) and reducing the list to manageable size. Typically, the screening will be done on the basis of judgment and experience, but more formal methods and calculations may be appropriate in some circumstances (e.g., an abridged form of the "expert elicitation" procedure in NUREG-1150 [see Section 5.4.3.1]). The third step is to define a set of cases to be evaluated. The most common approach is to define a best estimate, establish a range of interest for each factor, and then systematically vary the factors, one or more at a time. The results are then expressed as a range (low value, best estimate, high value) which indicates the effect on the output of variations in the factors, and thus provides some insight concerning uncertainties and their effects.

Uncertainty/sensitivity analysis for the cost measures is generally simpler than that for exposures. Complex accident scenarios are not involved. Moreover, the analyst usually has a better "feel" for cost-related measures (e.g., labor rates, interest rates, and equipment costs) than for risk-related ones. Thus, such analyses require no more than the straightforward variation of interest rates, labor hours, contingency factors, etc. However, the analyst is cautioned that, while the calculational techniques may be simple, wide ranges can still result.

To assist the analyst in performing uncertainty/sensitivity analyses for the standard analysis, this Handbook provides high and low values for selected best estimates in the evaluation of certain attributes (see, for example, Section 5.7.3.1). Should the analyst have access to better estimates, they should be used. In the cases where the analyst has access to a computerized assessment, the uncertainty/sensitivity analysis results obtainable via computer can be incorporated into the standard analysis. However, it is felt that more formal uncertainty/sensitivity analyses will only be practical for regulatory analyses requiring major efforts.

Finally, automated uncertainty calculations using default distributions are a feature of the FORECAST computer code for regulatory effects cost analysis (Lopez and Sciacca 1996). Uniform, lognormal, and several user-specified probability distributions are options.

## 5.5 Identification of Attributes

For every value-impact analysis to be performed, those attributes that could be affected by the proposed action must be identified. Once identified, the attributes may be quantified using the techniques presented in Sections 5.6 and 5.7. Note that the subsections of this section and Section 5.7 are numbered so as to correspond to one another in their discussions of the attributes. This section introduces the most commonly used attributes. Most of the attributes presented may be quantified in monetary terms, either directly or through use of a radiation exposure-to-money conversion factor (see Section 5.7.1.2). The remaining attributes are not readily quantifiable and are treated in a more qualitative manner. However, the analyst should attempt quantitative estimation whenever possible, relying on qualitative descriptions when no quantification is feasible.

## 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-rems 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-rems) 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:

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

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 | | | | 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 | 76 |
| BWR | RPB1 | CFatVB | LOSP | Early/Late | Sm/None | Dry | DWMth | 5.25E+6 | 80 |
| | RPB2 | | ATWS | | | | | 5.32E+6 | |
| | RPB3 | CFdurCD | | None | Large | | WWvent | 3.26E+6 | 84 |
| | RPB4 | Late CF | LOSP | Early/Late | | | DWrup | 1.13E+6 | 92 |
| | RPB5 | No CF | | 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 | 82 |
| | RLAS6 | | | | Large | Dry | CF-Ped | 4.18E+6 | 73 |

**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 | | No CCI | | 2.70E+6 | 93 |
| | RGG5 | No CF | | None | | | No CF | 1.18E+2 | 59 |

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
- WWaw-lk = 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:

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

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:

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

or

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

**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

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

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-rems 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)

$G_{PRO}$ = 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):

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

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

$$V_{PHR} = \sum_i N_i \left(H_{PRI_i} + H_{PRO_i}\right)$$

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-rems) 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.

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

$$V_{OHA} = \sum_i N\left(W_{IO_i} + W_{LTO_i}\right)$$

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 $D_{IO}$ (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 | 1,230 | 3,060 | 7,660 |
| 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.

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

where $\Delta F$ = 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 $W_{IO}$ = 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)
$Z_{IO}$ = 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

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

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 $W_{IO}$ in the undiscounted case.

The quantity $W_{IO}$ 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 $Z_{IO}$); 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 $Z_{IO}$ 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

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

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[11]
$r$ = 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

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

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-rems to dollars via monetary evaluation of health effects (see Section 5.7.4.2):

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

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):

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

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

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

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

$\Delta F$ = 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

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

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, $C = t_f = t_i$ (or, $C = t_f$ 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.

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
$\Delta F$ = 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:

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):

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

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 $\Delta F$ 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.

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

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:

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

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 $\Delta F$ 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

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:

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

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

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 ($\Delta F$) is 1.0E-6 and the number of reactors affected (N) is 111. The total averted onsite damage costs would be

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

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