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Optimization of Public and Occupational Radiation Protection at Nuclear Power Plants

A Review of Occupational Dose Assessment Considerations in Current Probabilistic Risk Assessments and Cost-Benefit Analyses

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Science Applications, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

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Until recently decision makers on the Nuclear Regulatory Commission staff have had to evaluate proposals for new maintenance and inspection requirements at nuclear power plants without the benefit of quantitative comparisons between the risk potential averted by the new requirement and the occupational risk created at the same time. While it was fully recognized that the generation of quantitative information of high precision would not be possible, it was also recognized that improved analytical techniques for quantitative comparisons could contribute substantially to the decision making process. Therefore funding was requested for a research project to develop an appropriate technique, to document it, and to provide comprehensive supporting material which would enable users to understand its strenths and weakness and to evaluate the rationale on which it is based. The project was awarded to SAI, Inc., and it has, I believe, been very ably carried out by the SAI staff.

R. E. Alexander

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Abstract

This report reviews current value-impact analysis and probabilistic risk assessment methods, and discusses the manner and degree to which these methods consider occupational radiation exposure that may form a variety of in-plant activities, including: (a) normal operation and maintenance, (b) repair, (c) retrofit, (d) minor incidents and cleanup, (e) major accidents, and (f) decommissioning. Value-impact analysis methods which include occupational exposure as an element of the value-impact equation have been developed, however, no standard approach to such analysis has been adopted. Comparison of the results of value-impact analyses must, therefore, be done with caution because different value-laden assumptions made by the analyst can have strong effects on the outcome. Such assumptions include the monetary equivalent of a person-rem, and the relative value of occupational and public exposure.

Probabilistic methods have been used in value-impact evaluations to quantify incremental or averted occupational exposure from reactor accidents, however, occupational exposure has not been addressed in probabilistic risk assessments (PRAs) of nuclear power plants. Consideration of occupational exposure in a PRA would greatly increase the complexity of the plant model and the benefits from such an analysis are uncertain. In lieu of expanding the scope of PRAs to address occupational risk, the separate, limited-scope probabilistic evaluations developed for value-impact analysis should provide a more practical analytical capability to support the evaluation and optimization of occupational and public radiation exposure.

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1. INTRODUCTION AND BACKGROUND

In the U.S., deterministic limits and guidelines have been established for public and occupational radiation exposure by both the Nuclear Regulatory Commission (Refs. 1 to 3) and the Department of Energy (Ref. 4). In addition, both agencies have adopted policies that public and occupational radiation exposure be maintained as low as reasonably achievable (ALARA), below the deterministic dose limits (Refs. 1,4). The basic objective of an ALARA program is to reduce individual and collective radiation exposure, and hence risk from radiation exposure, to the lowest levels commensurate with practical considerations such as the state of technology and the economics of improvements in relation to: (a) benefits to health and safety and other societal and socioeconomic considerations, and (b) utilization of atomic energy in the public interest.

Implicit in an ALARA program is the need to perform a tradeoff analysis between benefits and detriments in cases that may affect public or occupational exposure. Such tradeoff analyses are known by a variety of names, including cost-benefit analysis, risk-benefit analysis and value-impact analysis. To the extent practical, the term "value-impact analysis" will be used in this report as a generic term which encompasses other similar types of analysis intended for optimizing tradeoffs between benefits and detriments. Ideally, such analyses should be quantitative, however, this goal is complicated when the scope of analysis includes factors with different engineering units (i.e., dollars, rem), and subjective factors that cannot be readily quantified (i.e., perceived risk).

In the nuclear power industry, the use of value-impact analysis is expanding beyond traditional applications in economic analysis and ALARA radiation protection programs (Ref. 5), and is now becoming an integral part of regulatory proceedings which may impose new or revised requirements on licensees. In some recent value-impact analyses, tradeoffs between public and occupational exposures are considered. This approach results in a broader implementation of the ALARA philosophy, and can provide a risk-based justification for making, or not making, specific plant changes.

Referring to Figure 1-1, it can be seen that radiation exposure to workers in nuclear power plants can arise from a variety of activities and events including: (a) activities during normal plant operations (dose D1), (b) repair and retrofit activities during the life of the plant (doses D2 and D3), (c) activities associated with response to minor incidents or accident conditions that may occur (doses D5, D6 and D7), and (d) decommissioning activities at the end of the useful life of a plant (doses D4 and D8).

This report reviews current value-impact analysis and probabilistic risk assessment methods and discusses the approaches used for considering occupational radiation exposures that may arise from the variety of activities shown in Figure 1-1. Based on this review, recommendations are made for improving the quantification of occupational exposure in value-impact analysis and probabilistic risk assessment.

1.1 DETERMINISTIC OCCUPATIONAL EXPOSURE LIMITS

Deterministic limits on occupational exposure at facilities licensed by the Nuclear Regulatory Commission (NRC) are established in 10CFR20 (Ref. 1), which prescribes the quarterly dose limits listed in Table 1-1. Under accident conditions, higher occupational exposures are permitted by General Design Criterion 19 of 10CFR50, Appendix A (Ref. 2) which requires that:

> "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

These dose guidelines are based on actions being taken from the control room to maintain the plant in a safe condition following an accident. Further guidance is not provided on occupational exposure limits during accident or emergency conditions at NRC-licensed facilities.

Department of Energy (DOE) occupational radiation protection standards are specified in Chapter XI of DOE Order 5480.1 (Ref. 4), and are somewhat different than the NRC standards. The quarterly and annual DOE occupational dose standards are summarized in Table 1-2, and exposure guidelines for emergency situations are listed in Table 1-3. As stated in DOE Order 5480.1, the emergency exposure guidelines:

> "...do not establish a rigid upper limit on exposure, but rather leave judgment up to persons in charge of emergency operations to determine the amount of exposure that should be permitted to perform the emergency mission.... The official in charge must carefully examine any proposed action involving further radiation exposure by weighing the risks of radiation insults, actual or potential, against the benefits to be gained."

It is notable that DOE has adopted a value-impact philosophy in setting the emergency exposure guidelines. Specifically, these guidelines permit greater occupational doses under emergency conditions which present a greater risk to health and safety. This approach is compatible with the ALARA philosophy in that it attempts to control radiation exposure, consistent with the immediate objectives of the emergency (i.e., saving human life, recovering deceased victims, protecting health or saving property from unacceptable damage).

1.2 ALARA PROGRAMS

The NRC has established a policy that occupational radiation exposures at nuclear power plants be as low as reasonably achievable (ALARA). Implementation of an ALARA program during planning, design, construction, operation and decommissioning of a light-water reactor is addressed in Regulatory Guide 8.8 (Ref. 6). As stated in Reg. Guide 8.8, the goals of the ALARA program are: (a) to maintain the annual dose to individual station personnel as low as reasonably achievable, and (b) to keep the annual integrated (collective) dose to station personnel (i.e., the sum of annual doses to all station personnel) as low as reasonably achievable. Attaining those goals, and other more detailed objectives, is often based on a case-by-case application of good engineering judgment. It is noted in Reg. Guide 8.8 that:

> "A cost-benefit analysis may be helpful in arriving at the judgment, but it should not be the decisive factor in all cases."

It is further noted that:

> "The favorable cost-benefit ratio for achieving some of these objectives may be obvious without a detailed study. For other objectives, however, a cost-benefit study might be required to determine whether the objectives are reasonably achievable. Doses to station personnel can affect station availability, and this factor should be considered in assessing the cost-benefit ratio."

A formal methodology for performing the suggested cost-benefit analysis is not presented in Reg. Guide 8.8, and, in particular, no dollar-equivalent is assigned to the value of an occupational person-rem.

A method for performing occupational radiation dose assessments during the design of a light water nuclear power plant is described in Regulatory Guide 8.19 (Ref. 7). This assessment supports the identification of significant sources of occupational exposure, and the estimation of individual and collective occupational exposures during plant operation. These dose assessments are intended to permit the early identification of significant contributors to occupational exposure so that practical dose-reducing design changes, innovations or other corrective actions can be incorporated during the early stages of design. It is left to the licensee to develop a "systematic process for considering and evaluating possible dose reducing design changes and associated operating procedure changes."

The Department of Energy also has established a policy that occupational radiation exposures be maintained ALARA. The basic, deterministic requirements for the ALARA program are stated in Chapter XI of DOE Order 5480.1 (Ref. 4), with supplementary guidance, including value-impact analysis guidelines, in DOE/EV/1830-T5 (Ref. 8). A specific cost-benefit methodology is not prescribed by DOE, and the following precautions are offered:

> " ... cost-benefit analysis at best is difficult and fraught with the potential for error. Nonetheless, it can be a useful tool in ALARA programs and analysis if used objectively and in a limited way. ... "

It is noted in DOE/EV/1830-T5 that past attempts to establish a dollar value for a man-rem have yielded values ranging from \$10 to \$980, and adjusting these values for inflation since the early-1970s, when most of the referenced studies were performed, yields a current range from \$20 to \$2000 per person-rem. This approximation is used to establish a suggested lower bound for the value of a person-rem (i.e., \$2000 per person-rem), while an upper bound is set an order of magnitude higher plus an additional increment for conservatism. Regarding these guidelines, the following conclusions are offered:

> "In general, dose reductions that cost less than $2000 per person-rem of dose spared are probably always cost-beneficial, while costs in excess of $60,000 per person rem of dose spared are probably not cost-beneficial. In the absence of sound cost figures, an ALARA program cannot rely on cost-benefit analysis. In such cases, the criterion must be whether or not dose reduction is reasonably achievable, given the limitations of economics and practicality."

1.3 SAFETY GOALS

The NRC is in the process of developing and implementing safety goals for nuclear power plants. These goals have the objective of limiting "to an acceptable level the radiological risk which might be imposed on the public as a result of nuclear power plant operation" (Ref. 9). The proposed goals incorporate quantitative measures of individual risk and societal risk; however, the individuals of interest are members of the public. Occupational exposure is not addressed in the current NRC safety goals.

The Commission has adopted a trial-use guideline of $1,000 per person-rem averted for use in value-impact analysis (Ref. 9). This guideline is in terms of 1983 dollars, and it is expected to be adjusted in the future to account for the effects of inflation.

1.4 SECTION 1 REFERENCES

1. 10CFR20, "Standards for Protection Against Radiation," U.S. Nuclear Regulatory Commission.
2. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.
3. 10CFR100, "Reactor Site Criteria," U.S. Nuclear Regulatory Commission.
4. DOE Order 5480.1, Chapter XI, "Requirements for Radiation Protection," U.S. Department of Energy.
5. Eichholz, G. G., "Cost-Benefit and Risk-Benefit Assessment for Nuclear Power Plants," Nuclear Safety, Vol. 17, No. 5, September-October 1976.
6. USNRC Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low as Reasonably Achievable."
7. USNRC Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Plants - Design Stage Man-Rem Estimates."
8. Kathren, R. L., et al., "A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA)," DOE/EV/1830-T5, Pacific Northwest Laboratory, April 1980.
9. NUREG-0880, Rev. 1, "Safety Goals for Nuclear Power Plant Operation," U.S. Nuclear Regulatory Commission, May 1983.

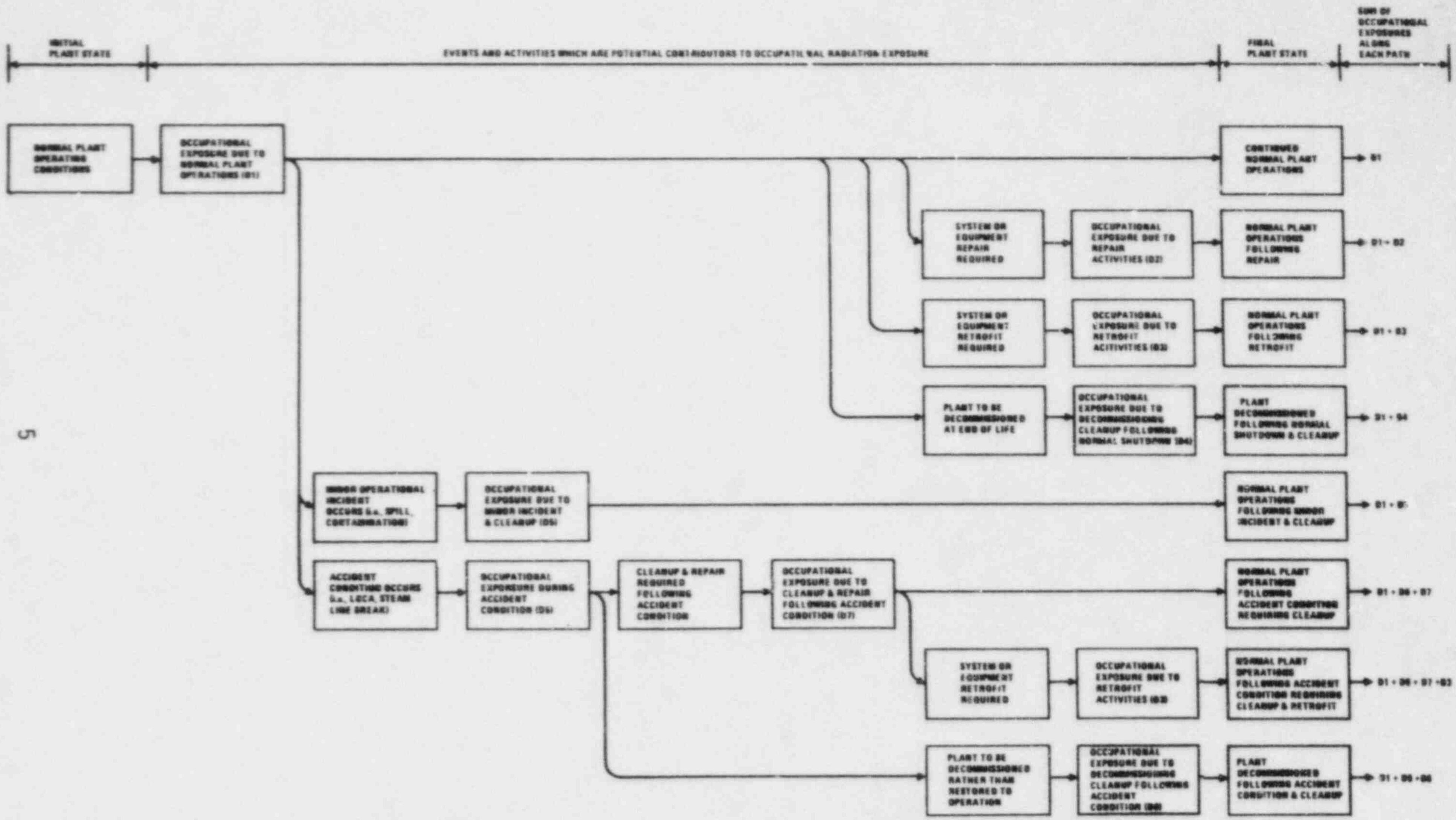


Figure 1-1. Summary of Activities and Events Potentially Contributing to Occupational Exposure at a Nuclear Power Plant.

Table 1-1. NRC Radiation Protection Standards for Occupationally-Related External and Internal Exposures (from 10CFR20).

Type of Exposure	Rems per Calendary Quarter
Whole body; head and trunk; active blood forming organs; lens of eyes; or gonads	1.25 rem[a,b]
Hands and forearms; feet and ankles	18.75 rem
Skin of whole body	7.5 rem

(a) A licensee may permit an individual to receive a total occupational dose to the whole body in excess of 1.25 rem provided:

- Total occupational dose to the whole body shall not exceed 3 rems during any calendar quarter, or 12 rem per year.

- Total dose to the whole body, when added to the accumulated occupational dose to the whole body shall not exceed 5(N-18) rems, where "N" is the individual's age at his last birthday (i.e., the lifetime occupational dose may not exceed an average of 5 rems for each year above the age of 18).

(b) Exposure to radiation that results from radioactive materials taken into the body (internal exposure) is measured, recorded and reported separately from external dose. The internal dose to the whole body does not, at this time, count against the 3 rem per calendar quarter limit. An additional whole-body dose of approximately 5 rems per year is permitted from internal exposure.

Table 1-2. DOE Radiation Protection Standards for Occupationally-Related External and Internal Exposures (from DOE 5480.1, Chapter XI).

Type of Exposure	Exposure Period	Dose Equivalent (Dose or Dose Commitment[1/] rem)
Whole body, head and trunk, gonads, lens of the eye[2/], red bone marrow, active blood-forming organs.	Year Calendar Quarter	5[3/] 3
Unlimited areas of the skin (except hands and forearms). Other organs, tissues, and organ systems (except bone).	Year Calendar Quarter	15 5
Bone.	Year Calendar Quarter	30 10
Forearms[4/]	Year Calendar Quarter	30 10
Hands[4/] and feet.	Year Calendar Quarter	75 25

1/ To meet the above dose commitment standards, operations must be conducted in such a manner that it would be unlikely that an individual would assimilate in a critical organ, by inhalation, ingestion, or absorption, a quantity of radionuclide or mixture of radionuclides that would commit the individual to an organ dose that exceeds the limits specified in the above table.

2/ A beta exposure below a maximum energy of 700 KeV will not penetrate the lens of the eye; therefore, the applicable limit for these energies would be that for the skin (15 rem/year).

3/ In special cases, with the approval of EP-30, a worker may exceed 5 rem/year, provided his or her average exposure per year since age 18 will not exceed 5 rem per year. This does not apply to emergency situations.

4/ All reasonable effort shall be made to keep exposures of forearms and hands to the general limit for the skin.

Table 1-3. DOE Occupational Dose Guidelines Applicable During Emergency Situations (from DOE 5480.1, Chapter XI).

Emergency Situation	Dose Guideline
Actions involving saving human life	Not clearly defined (may approximate 100 rem or more)
Actions involving the recovery of deceased victims	Normal occupational exposure guidelines
- Special circumstances where it is impossible to recover victims without entry of emergency workers into the area	$\leq$ 12 rem total for the year or 5(N-18) rems whichever is more limiting
Actions involving protection of health and property	
- Actions essential to reduce a hazard potential to acceptable levels or to prevent a substantial loss of property	$\leq$ 12 rem total for the year
- Special circumstances, with volunteers	$\leq$ 25 rem
- Actions where the potential risk of radiation hazard is such that life would be in jeopardy, or there would be severe effects on health and safety of the public or loss of property inimical to the public safety	Same as actions involving saving human life (see above)

2. CONSIDERATION OF OCCUPATIONAL EXPOSURE IN VALUE-IMPACT ANALYSIS

Optimization of radiation protection applies to situations where radiation protection can be controlled by protection measures. There are several optimization techniques available, some being more quantitative and some being more qualitative, but all either explicitly or implicitly impose or require the making of value judgments about the possible objectives of optimization. Value-impact (or cost-benefit) analysis is the most commonly applied technique for optimizing radiation protection. Methodologies used by the International Commission on Radiological Protection (ICRP), the NRC and others will be discussed in this chapter.

A common problem with value-impact analysis methods is that the analyst may be required to make some value judgments for which there are few points of reference, and on which administrative and political authorities may hesitate to take a stand (Ref. 1). Such judgments include:

- Factors to be included in the value-impact equation
  - tangible values and impacts
  - intangible values and impacts
- Monetary value of a person-rem (or life, or specific health effects)
- Equitable treatment of benefits and costs that accrue to different populations
  - public vs. occupational
  - present vs. future

Other factors which can have a significant effect on the results of a value-impact analysis include:

- Scope of alternatives considered (i.e. assumptions on presently available or future technological solutions)
- Simplifications and assumptions in models to describe values and impacts
  - dose and health effect models
  - economic models
- Data integrity

A value-impact analysis must, therefore, be carefully considered in the context of all relevant assumptions and constraints.

2.1 ICRP COST-BENEFIT ANALYSIS METHODS

The technique of cost-benefit analysis has been described by the International Commission on Radiological Protection in ICRP Publications 26 and

37 (Refs. 2 and 3). In support of this cost-benefit methodology, the ICRP has assumed that:

> "There is a proportionality between dose and the probability of stochastic effect, within the range of doses encountered in radiation work. A consequence of this assumption is that doses are additive in the sense that equal dose increments increase equally the risk by a value which is independent of the previous accumulated dose. A further consequence of the assumption is that, in principle, radiation risks in a given situation can be reduced as much as is desired by increasing the level of protection, thus decreasing exposure."

The basic ICRP cost-benefit equation in ICRP-37 is the following:

$$B = V - (P + X + Y)$$

where: B is the net benefit of the assumed practice
- V is the gross benefit of the assumed practice accrued to society
- P is the basic production cost of the practice; excluding the cost of radiation protection
- X is the cost of achieving a selected level of radiation protection
- Y is the cost of the detriment resulting from the practice at the selected level of radiation protection (detriment is defined as the mathematical expectation of the amount of harm in the exposed group of people, taking into account both the probability and the severity of the different possible harmful effects).

In this equation, benefits "are taken to include all the benefits accruing to society and not just those received by particular groups or individuals. Costs are considered as comprising the total sum of all negative aspects of an operation, including monetary costs and any damage to human health or to the environment." Occupational exposure is considered in this ICRP equation as a contributor to gross benefit (i.e. a decrease in occupational exposure) or to the cost of the detriment (i.e. an increase in occupational exposure).

Optimization of radiation protection involves maximizing the net benefit from the introduction of a practice. This is accomplished by setting the first derivative with respect to collective dose (S) of the cost-benefit equation equal to zero, as follows:

$$\frac{dV}{dS} - \left(\frac{dP}{dS} + \frac{dX}{dS} + \frac{dY}{dS}\right) = 0$$

The optimization procedure thus can be considered as a differential cost-benefit analysis or a marginal cost-benefit analysis. For a given practice, gross benefit (V) and basic production cost (P) are a constant with respect to collective dose (S). Thus optimization is performed on two variables only: X and Y (cost of radiation protection and cost of the detriment, respectively).

The cost of a radiation protection practice can be estimated in monetary terms using conventional economic methods of costing, discounting, etc.

Estimation of detriment costs are not straightforward and may involve implicit or explicit judgments on the values of life, health and non-health effects. The ICRP does not endorse a particular monetary value for a person-rem for use in cost-benefit analysis, but notes the following in ICRP-37:

> "Without correcting prices to any particular year, the values have ranged from approximately US $1000 per man-sievert ($10 per man-rem) to approximately US $100,000 per man-sievert ($1000 per man-rem). No firm conclusions could be drawn from this range, apart from showing that, over the years, different individuals and organizations had used various methods to produce different values for the unit collective dose... However, it should be mentioned that, in many cases where a wide range of values has been proposed, these values were derived within a conceptual framework clearly different from the one presented in this report (ICRP-37). In the numerical examples in this report, values (cost per unit collective dose) in the range of $10,000 to $20,000 per man-sievert ($100 to $200 per man-rem) have been used."

In the example cost-benefit analyses in ICRP-37, public and occupational exposures are equally weighted.

As a caution regarding the use of quantitative methods of decision-making, the ICRP notes that the results depend "heavily on the quality of judgments and data which went into the analysis. It is, therefore, necessary to evaluate the sensitivity of the solution to variations in some or all of the judgmental inputs and data. Such sensitivity assessment allows the identification of the crucial factors in the decision and helps in making the approach more meaningful, particularly when the problem is complex."

2.2 NRC VALUE-IMPACT ANALYSIS METHODS

The NRC has established requirements for performing value-impact analysis and has implemented, or is considering a variety of methodologies. The NRC Regulatory Analysis Guidelines (Ref. 4) establish broad requirements "to ensure that the NRC regulatory decisions are based on adequate information concerning the need for, and consequences of a proposed regulatory action and to ensure that cost-effective regulatory actions, consistent with providing the necessary protection of the public health and safety and common defense and security, are identified." The Regulatory Analysis Guidelines recommend the use of quantitative methods to estimate costs and benefits of proposed alternatives whenever possible. Some examples of value-impact methods used by, or proposed to the NRC are described in this section.

2.2.1 Basic NRR Value-Impact Analysis Guidelines

The value-impact analysis methods described in Reference 5 are being used by the NRC Office of Nuclear Reactor Regulation in support of the development and justification of significant changes in regulatory requirements. The term "value-impact analysis" is interpreted as follows:

> "It is 'essentially' a technique equivalent to benefit and cost analy-is, or cost and effectiveness analysis. The term value-impact was introduced by the NRC to dispel certain connotations associated with other terms. Benefit-cost analysis, in particular, is often misconceived as a process of reducing all factors to a common dollar form. This, the staff felt, was too restrictive, and therefore value and impact were recommended and designed to include noncommensurables and variables that are nonquantifiable or nonmeasurable."

Impacts are defined as having negative effects (i.e. increase in risk, radiation dose, or environmental damage, expenditure of money, time or other resource), and conversely, values have positive effects (i.e. reduction in risk, radiation dose or environmental damage, etc.)

The NRR value-impact analysis guidelines include basic format and content recommendations which include occupational exposure as an industry-related issue to be addressed. If a proposed action causes occupational exposure to be averted, the dose reduction is a "value", whereas a dose increase would be an "impact." The following general guidance is offered for comparing values and impacts:

> "No particular analytical technique or formal decision methodology is recommended at this time for comparing the values and impacts of alternatives. In most cases, particularly for preliminary statements, the balancing will be done on the basis of professional judgment. When it is possible, meaningful and appropriate, however, values and impacts should be translated into such measures as exposure dose, monetary units, time, risk, etc."

The guidelines recommend that, to the extent practical, risk assessments and cost estimates should be quantitative; however, no dose-to-cost conversion factor is proposed and no guidance is presented on the relative weighting of public and occupational exposure.

Value-impact analyses that have been prepared to these NRR guidelines include analyses in support of the resolution of the following generic safety issues: (a) containment emergency sump performance (NUREG-0869, Ref. 6), (b) water hammer (NUREG-0993, Ref. 7), and (c) steam generator tube degradation and rupture (Ref. 8).

2.2.2 Value-Impact Analysis for Safety Issue Prioritization

2.2.2.1 Overview of the Safety Issue Prioritization Program

The NRC Office of Nuclear Reactor Regulation is conducting a program to priorize generic safety issues. This work is being performed in response to Section IV.E of the TMI Action Plan (Ref. 9) which called for development of a plan and early resolution of significant safety issues. The prioritization program, described in NUREG-0933 (Ref. 10) is intended to aid the timely and efficient allocation of resources to those generic safety issues that have a high potential for reducing risk. The program is also intended to justify removing from further consideration those issues that have little safety significance.

A generic safety issue is assigned a high, medium, low or "drop" priority on the basis of rough approximations of risk and costs and the calculation of a "value-impact score", using the following formula:

$$\text{V-I Score, S} = \frac{\text{Safety Benefit}}{\text{Cost}} = \frac{NFTD}{C}$$

where: N is the number of reactors involved
T is the average remaining life of the affected plants
F is the accident frequency reduction
D is the public dose
C is the total cost of developing and implementing the resolution for all affected plants

The matrix for establishing the priority of a safety issue is shown in Figure 2-1, and the risk thresholds applicable to this table are summarized in Table 2-1.

The value-impact analysis guidelines in NUREG-0993 require the consideration of occupational exposures; however, the value-impact score is initially calculated on the basis of public exposure. Several examples can be found where incremental occupational doses due to implementation, operation and maintenance are considered as well as averted occupational dose due to accident avoidance. Where appropriate, net occupational exposure is compared to net public exposure and the effect on the value-impact score is considered.

A more formalized value-impact methodology has been developed by Pacific Northwest Laboratory (PNL) in support of the NRC program for prioritization of generic safety issues. This methodology is described in NUREG/CR-2800 (Refs. 11 and 12), and is summarized below.

2.2.2.2 PNL Value-Impact Methodology for Safety Issue Prioritization

The scope of the PNL value-impact methodology in NUREG/CR-2800 (Refs. 11 and 12) emphasizes "development of defensible risk, dose and cost estimates at a modest cost." As a result, major simplifications are made to produce an approach that can be implemented within the level-of-effort available for the prioritization process.

The PNL value impact methodology uses the following five step process:

- Obtain information on a safety issue and identify affected plants.
- Obtain or postulate a safety issue resolution (SIR).
- Estimate effect of the SIR on risk equations and then calculate public risk reduction and occupational dose, including uncertainties.
- Estimate cost of the SIR, including engineering cost, projected industry and NRC implementation costs, and operating and maintenance costs, including uncertainties.
- Present results.

This methodology is used to compare estimated changes in public and occupational exposure caused by an SIR, and estimated industry and NRC costs. No dose-to-cost conversion factor is proposed, and no attempt is made to convert all values and impacts to a common monetary unit. Public and occupational exposures are equally weighted.

The PNL methodology examines two components to occupational exposure: (a) an incremental dose from implementation, operation and maintenance of a particular SIR, and (b) an avoided dose from reducing the frequency of an accident (and thus the probability of incurring doses from cleanup, repair and refurbishment activities following the accident). The incremental occupational exposure is defined as follows:

G = occupational dose increase due to implementation, operation, and maintenance of the SIR

$= N(D_0\overline{T} + D)$ in person-rem

where N = number of reactors affected by the SIR
$\overline{T}$ = average remaining operating life of reactors affected (years)
D_0 = annual incremental dose due to operation and maintenance of the SIR (person-rem/reactor-year)
D = incremental dose increase due to implementation of the SIR (person-rem/reactor).

The avoided occupational exposure is defined as follows:

ΔU = change, due to the SIR, in the accident-frequency-weighted occupational dose from cleanup and repair of a reactor following an accident (person-rem)

$= N\overline{T}\,\Delta(FD_R)$

where $\Delta(FD_R)$ = change, due to the SIR, in the product of estimated time frequency of accidents in (reactor-years)$^{-1}$ and occupational dose due to cleanup and repair of the reactor following an accident (person-rem).

The PNL methodology uses a "Public Risk Reduction Work Sheet" to systematize the estimation of change in core melt frequency due to an SIR. An "Occupational Dose Work Sheet" is used to calculate the occupational dose parameters associated with an SIR. The methodology includes standardized approaches for calculating uncertainty in dose estimates, at the 90 percent confidence level. These work sheets are reproduced in Appendix A.

2.2.2.2.1 Occupational Exposure Increase due to Implementation, Operation and Maintenance

The increase in occupational exposure is caused by work in radiation zones during retrofit of equipment in operating plants, and subsequent operation and maintenance of equipment associated with an SIR. This occupational exposure is estimated using existing sources of data on radiation dose rates in various areas of reference reactors and engineering estimates of the labor hours required in radiation zones. Exposure data sources used include

NUREG/CR-0130, and -0672 (Refs. 13 and 14) and plant-specific Safety Analysis Reports.

2.2.2.2.2 Occupational Exposure Decrease Due to Accident Avoidance

Following a serious accident at a reactor plant, a utility has two basic options: (a) cleanup, repair, refurbish and restore the plant to operation, or (b) cleanup and decommission the plant. Occupational exposures associated with the latter option have been evaluated in NUREG/CR-2601 (Ref. 15). The PNL value-impact methodology in NUREG/CR-2800 (Refs. 11 and 12) assumes that the occupational doses associated with the first option (repair and refurbishment) will be about the same as the second option (decommissioning), and aspects of the NUREG/CR-2601 methodology were adopted for estimating occupational dose avoided.

Cleanup doses are estimated in NUREG/CR-2601 for three accident scenarios, which are related to WASH-1400 (Ref. 16) release categories as follows:

WASH-1400 RELEASE CATEGORIES	NUREG/CR-2601 ACCIDENT SCENARIOS
PWR 1 to 7 BWR 1 to 4	3 (Core Melt)
PWR 8, 9 BWR 5	2 (Non-core Melt)
--	1 (Other Non-core Melt)

PWR and BWR plant conditions in the three NUREG/CR-2601 accident scenarios are summarized in Tables 2-2 and 2-3, and the resulting occupational exposures from cleanup activities can be found in Tables 2-4 and 2-5. The WASH-1400 release categories and the corresponding NUREG/CR-2601 accident scenarios are summarized in Appendix B.

It is stated in NUREG/CR-2800 that "for the majority of issues analyzed using the (safety issue prioritization) methodology, only core-melt accidents like Scenario 3 will be considered." The change of frequency of core-melt accidents is estimated using a "Public Risk Reduction Work Sheet," which requires: (a) identifying the events in the dominant event sequences of a Probabilistic Risk Assessment (PRA) that are affected by the safety issue resolution, and (b) re-estimating the probabilities of the affected sequences.

Occupational dose reduction due to accident avoidance is calculated using the previously specified equation, with the following standardized error bounds (at a 90% confidence level):

$$(\Delta U)_{upper} = 6\hat{D}_R \sum_x N_x \bar{T}_x \hat{F}_x$$

$$(\Delta U)_{lower} = 0$$

where

x = plant type x

N_x = number of affected reactors of plant type x

$\bar{T}_x$ = average remaining life of affected plant type x reactors
$\hat{D}_R$ = the best estimate of the occupational dose due to reactor cleanup and repair following an accident
$\hat{F}_x$ = the best estimate of the base-case, affected core-melt frequency for plant-type x

2.2.3 Optimization Methodology in Support of Regulatory Guide 8.10 Update

Pacific Northwest Laboratory has conducted a study for the NRC Office of Nuclear Regulatory Research in support of updating Regulatory Guide 8.10 (Ref. 17) to: (a) implement current recommendations of the ICRP and (b) provide more detailed ALARA guidance for licensees. The results of the PNL study, reported in NUREG/CR-3254 (Ref. 18), include recommendations on optimization of radiation protection using a differential value-impact analysis procedure similar to the ICRP approach described in ICRP Publications 26 and 37 (Refs. 2, 3).

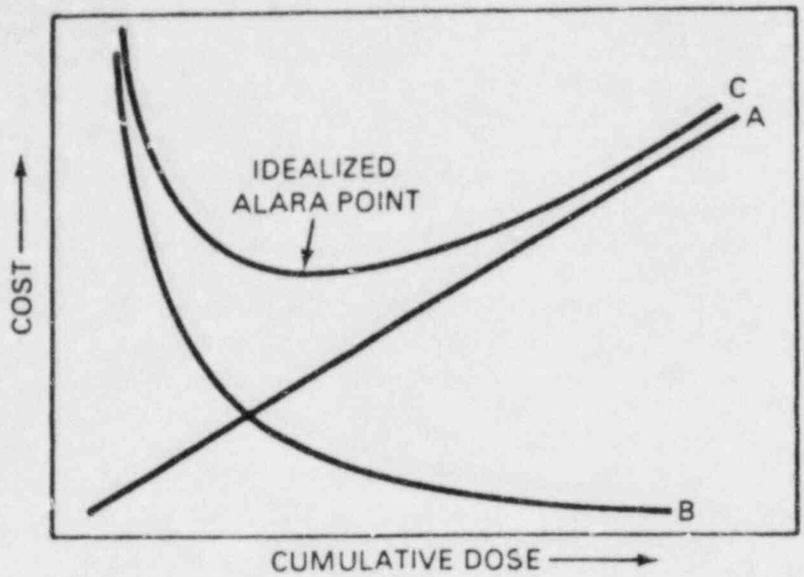
The basic methodology described in NUREG/CR-3254 is referred to as a "cost-effectiveness" analysis, in which the following equation is used to quantify values and impacts:

$$\text{Benefits} \geq \text{Costs}$$

$$B + U \geq M + L + N + O + E + R$$

where B = dose reduction in person-rem achievable if the practice is implemented multiplied by $1,000.
U = intangible benefits multiplied by an estimated value for each benefit.
M = dollar cost of materials required to implement the practice.
L = dollar cost of labor.
N = dollar cost of maintenance of the practice.
O = dollar cost of operation.
E = radiation exposure in person-rem necessary to install and maintain the practice multipled by $1,000.
R = intangible costs of the practice.

As can be seen from the definition of terms, this methodology depends on converting all terms to common units of cost (i.e. dollars). The same value of $1,000 per man-rem (whole body and thyroid) is used for both occupational and public exposure, "until a more definitive value is established." Using the assumption that risk is proportional to dose at all levels of exposure (linear hypothesis as used in ICRP 26 and 37), the ALARA optimization process can be illustrated as follows (from NUREG/CR-3254):



Curve A represents the cost equivalent of doses received, Curve B represents the costs of dose reduction, and Curve C is the sum of A and B. The shape of Curve A (i.e. linear) is established by the linear hypothesis, and the slope is defined by the value assigned to a person-rem. Curve B illustrates the Law of Diminishing Returns in that initial dose reductions may be accomplished at lower costs than future efforts to further reduce doses. Objective and subjective factors can affect the shape of Curve B (i.e. known costs, estimated costs, economic modeling assumptions).

The idealized ALARA point is determined from the optimization process for maximizing the net benefit from the practice being considered. Referring to the preceding illustration of the optimization process, the Curve C is defined as:

$$C = A + B$$

The minimum point of Curve C can be found by setting the first derivative with respect to cumulative dose equal to zero:

$$\frac{dC}{dS} = 0 = \frac{dA}{dS} + \frac{dB}{dS}$$

where A = cost of the detriment involved in the operation (from Curve A)
B = cost of achieving a selected level of protection (from Curve B)
S = cumulative dose

It is, of course, necessary to know the equations of Curves A and B. An example of this optimization process is included in NUREG/CR-3254 (Ref. 18).

2.2.4 Other NRC-Supported Efforts to Establish Procedures for Value-Impact Analysis

2.2.4.1 Improved Cost-Benefit Techniques

In NUREG/CR-3194 (Ref. 26), Pacific Northwest Laboratory (PNL) reviewed: (a) the use of cost-benefit methods by federal agencies, and (b) methods for monetizing nonmarket values and impacts. This report emphasizes the use of cost-benefit techniques which reduce all values and impacts to common monetary units. The planned NRC safety goals in NUREG-0880 (Ref. 24) and the guideline of $1000 per person-rem averted are discussed briefly in this report, however, no specific guidelines are provided for treating occupational exposure in cost-benefit analysis.

2.2.4.2 Handbook for Value-Impact Assessment

PNL has developed NUREG/CR-3568 (Ref. 27) as a handbook which establishes guidelines for performing two types of value-impact analysis: (a) ratio method, and (b) net benefit method. As described in this handbook, the ratio method is used to calculate a value-impact ratio that typically has units of person-rem per million dollars. No attempt is made to establish a monetary equivalent for a person-rem. The net benefit method requires that all attributes be expressed in dollars, and a net benefit is determined by summing all attributes. In both methods, a supplementary evaluation may be needed to describe those effects that are not adequately reflected in the quantitative ratio or net benefit value.

The attributes included in the value impact equation include the following:

- Public health
- Occupational exposure (accidental)
- Occupational exposure (routine)
- Offsite property
- Onsite property
- Regulatory efficiency
- Improvements in knowledge
- Industry implementation
- Industry operation
- NRC development
- NRC implementation
- NRC operation

As in the case of the NUREG/CR-2800 (Refs. 11 and 12) value-impact methodology described previously, this PNL methodology uses a variety of work sheets for systematizing the value-impact analysis process. In fact, the NUREG/CR-2800 methodology is very similar to the ratio method described in the Handbook for Value-Impact Assessment (Ref. 27), augmented by consideration of occupational exposure.

No specific valuation is established for a person-rem. It is noted, however, that "the analyst should use a range of values in the analysis so that the sensitivity of the results to different numerical values can be assessed. One of the values used in the analysis should be $1000 per person-rem." The methodology provides the capability to assign different weights to public and occupational exposure. Although this capability exists, the authors caution that, "justification should be provided for the weights employed".

The equations used in NUREG/CR-3568 (Ref. 27) to define impacts of accidental occupational exposure are the following:

- Ratio method:

$$V^*_{OHA} = NTD_{OA}$$

- Net benefit method:

$$V_{OHA} = NT(D_{OA} \times R)$$

To define impacts of routine occupational expsoure, the following equations are used:

- Ratio method:

$$V^*_{OHR} = N(TD_{ORO} - D_{ORI})$$

- Net benefit method:

$$V_{OHR} = NR(TD_{CRD} - D_{ORI})$$

The terms in these equations are defined as follows:

V^*_{OHA} is the avoided occupational health risk due to accidents (person-rem)
V_{OHA} is the value of avoided occupational health risk due to accidents ($)
V^*_{OHR} is the change in occupational health risk from routine activities (person-rem)
V_{OHR} is the value of the change in occupational health risk from routine activities ($)
N is the number of affected reactors
T is the average remaining lifetime of affected reactors (years)
R is the monetary value of unit dose ($/person rem)
D_{OA} is the avoided occupational dose per reactor-year
D_{ORI} is per-reactor increase in occupational dose required to implement a proposed action
D_{ORO} is the annual per-reactor change in occupational dose to operate following implementation of a proposed action

To illustrate the use of the systematic value-impact methodology recommended in this handbook, an example analysis is provided as an appendix to NUREG/CR-3568. The example is a reworking of the value-impact anlaysis performed in NUREG-0896 (Ref. 6) for Unresolved Safety Issue USI-43, containment emergency sump performance.

The Handbook for Value-Impact Assessment also provides guidelines for scaling the assessment to be compatible with available resources. Three levels of effort are defined: (a) a limited effort, (b) an intermediate effort, and (c) a major effort. For many of the attributes considered in the value-impact equation, guidelines are provided for conducting an assessment at each of the defined levels of effort.

2.3 ATOMIC INDUSTRIAL FORUM (AIF) BENEFIT/COST METHODS

The AIF has sponsored the development of a methodology for estimating potential operational cost savings resulting from incremental reductions in external occupational exposure. The methodology, described in AIR/NESP-010R (Ref. 19), is intended for use as part of an overall ALARA value-impact analysis. Basic assumptions in this methodology include the following:

- There is no incremental operating costs associated with radiation exposure unless occupational exposure approaches applicable administrative limits.

- The consequence of exposures approaching the administrative limits is that more crews are required to complete a task (i.e. because the initial crew was "burned out.")

- Four factors determining the number of crews required to perform a task are:
  - administrative dose limits
  - dose rate in work area
  - on-the-job time to complete task
  - worker's average utilization factor (a measure of dose received while not productively working).

Calculation methods were developed to treat four types of dose reduction actions in detail: (a) dose rate reductions, (b) on-the-job time reductions, (c) training to improve worker's time and dose utilization and (d) reductions in maintenance frequency. A step-by-step procedure is prescribed for using the calculational methods to derive an overall benefit/cost ratio for a proposed dose reduction action. Estimates are provided for occupational cost savings resulting from the dose reduction actions described above.

The AIF methodology is applicable to optimization of external occupational exposure from maintenance and repair activities. It does not address internal occupational exposure (i.e., dose arising from uptake of radioactive material by workers), is limited to the evaluation of a single action, and does not address the potential combined effects of multiple jobs performed by the same person or crew. Public exposure from such activities is beyond the scope of this AIF methodology.

2.4 OTHER VALUE-IMPACT ASSESSMENTS

2.4.1 Z-Plant Cost/Risk/Benefit Analysis

A value-impact analysis on the decontamination and decommissioning (D&D) of the Z-plant (plutonium conversion facility) at DOE's Hanford Engineering Development Laboratory has been performed (Ref. 20). The three elements considered in the evaluation of D&D alternatives were:

- net cost
- occupational exposure to onsite personnel
- potential offsite risk

These elements are the factors in the value-impact equation. Each factor was quantified for each of the alternative D&D endpoints considered in this study. These quantitative results were then input to a subjective evaluation process to identify the preferred D&D alternatives. No attempt was made to reduce the dissimilar factors to a common unit such as cost.

2.4.2 Financial Consequences of Reactor Accidents

A study described in NUREG/CR-2723 (Ref. 21) was performed by Sandia National Laboratories to estimate the financial consequences of reactor accidents (i.e. a part of a value-impact equation). The items of cost included the following:

- Offsite Costs
  - early fatalities
  - early injuries
  - latent cancer fatalities
  - property damage
- Onsite Costs
  - onsite (occupational) health effects
  - replacement power
  - cleanup

In determining the onsite health effects, it was assumed that emergency planning requirements would reduce the onsite population during a major emergency in which a significant release was imminent. It was further assumed that approximately 40 persons would remain onsite in the control room or the support center, both of which are designed to provide some radiological protection for occupants.

The following five accident categories, or groups, developed by the NRC in NUREG-0771 (Ref. 22) were considered in the Sandia study:

- Group 1 - Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment (similar to PWR 2).

- Group 2 - Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g. sprays, suppression pool, fan coolers) operate to reduce release (similar to PWR5).

- Group 3 - Severe core damage. Containment fails by basemat melt-through. All other release mitigation systems have functioned as designed (similar to PWR6).

- Group 4 - Limited to moderate core damage. Containment systems operate but in a somewhat degraded mode (similar to PWR9).

- Group 5 - Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis accidents are assumed. The most severe accident in this group includes substantial core melt, but containment functions as designed (an order of magnitude smaller than PWR9).

The designations PWR2, PWR5, PWR6 and PWR9 refer to WASH-1400 (Ref. 16) release categories which are defined in Appendix B.

The NRC has defined releases or Siting Source Terms (denoted SST-1 to SST-5) for each of the five accident groups listed above (Ref. 23). It is noted in the Sandia study that occupational health effect costs are only determined for SST-1 releases, and other releases are assumed to cause no early effects to reactor personnel. The assumed onsite health effects for an SST-1 release are 10 early fatalities and 30 early injuries.

Health effect "costs" were converted to dollar-equivalents using the following conversions: (a) $1 million per early fatality, and (b) $100,000 per early injury or latent cancer fatality. It is noted that these values are different than what would be obtained using the $1000 per person-rem averted recommended by the NRC in NUREG-0880 (Ref. 24) (i.e. approximately $10 million per latent cancer fatality).

As a result of these conversions, the potential impacts of the three categories of reactor accidents were represented uniformly in terms of dollars and the financial impacts of accidents were estimated for 156 reactor units.

2.4.3 Reactor Decommissioning Value-Impact Analysis

Several value-impact studies have been performed on the alternatives for decommissioning light water reactors. NUREG/CR-0130 and NUREG/CR-0672 (Refs. 13, 14) address the decommissioning of a reactor plant following a normal shutdown at end-of-life. NUREG-2601 (Ref. 15) addresses the decommissioning of a reactor following a major accident. The latter study is the source of the post-accident plant parameters and the estimated doses due to cleanup activities that are used in the NRC safety issue prioritization methodology described in Section 2.2.

In these decommissioning value-impact studies, costs and radiation exposures are calculated and no attempt is made to reduce all impacts to common units. Public and occupational exposures and health effects are tabulated separately.

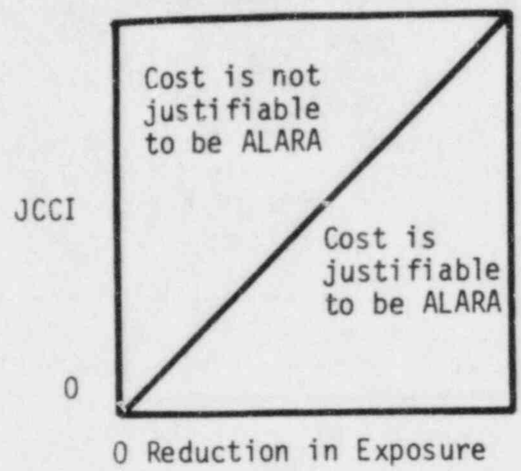
2.4.4 Justifiable Cost of Capital Investment (JCCI) Cost-Benefit Analysis

A method has been developed to perform a simplified value-impact analysis for issues associated with occupational exposure based on the calculation of a "justifiable cost of capital investment" (JCCI, Ref. 25). The following equation is used:

$$JCCI = (d) \times (\text{delta } MR) \times (i)$$

where d = a value chosen as the dollar worth of a man-rem (assumed to be a constant)
delta MR = the yearly man-rem averted by some protective action
i = the present worth factor which is determined from annual interest rate and years of remaining plant life.

This equation describes a line which divides a graph into two regions as shown below:



This is a limited-scope value-impact methodology that only considers occupational exposures, costs, and simple economic factors. The method is highly dependent on the dollar value assigned to an occupational man-rem.

2.5 SUMMARY AND RECOMMENDATIONS FOR CONSIDERATION OF OCCUPATIONAL EXPOSURE IN VALUE-IMPACT ANALYSIS

The value-impact methodologies that have been reviewed in this section can be divided into two basic types: (a) those that reduce values and impacts to a common denominator such as cost, and (b) those that develop ratios to define a relative weighting of values and impacts having different engineering units. Approaches which monetize all values and impacts can support a more rigorous mathematical definition of radiation protection optimization. The "optimum" point depends, however, on the value-laden assumptions made in assigning a dollar worth to non-cost values and impacts such as radiation exposure. Sensitivity studies can be performed to assess the importance of the various assumptions that have been made; however, some assumptions may not be apparent if they appear only as coefficients in the value-impact equation.

When values and impacts are defined in their normal engineering units, a more subjective approach is required for radiation protection optimization. The same value-laden assumptions described above must still be made, if only in the minds of the persons evaluating the results of the value-impact analysis. The final results, therefore, may not be reproduced independently unless a multi-criteria evaluation matrix is defined.

In either case, value-laden assumptions are made in the process of performing value-impact analyses. As a minimum, these assumptions should be clearly specified, and the sensitivity of the results to variations in the assumptions should be determined as part of the value-impact analysis. Without these measures being taken, it is all too likely that the numerical results of a value-impact analysis will be used, perhaps in unintended applications, without remembering their technical basis.

Occupational exposure is one element in a value-impact equation, but it includes two major components: (a) incremental dose from implementation, operation and maintenance of a protective action, and (b) averted dose due to reducing the probability of an accident which results in occupational exposure. Estimating the first element is a relatively straightforward process. There appears, however, to be a variety of approaches for estimating averted occupational exposure in the value-impact methodologies that consider this element. In particular, there is little standardization in the definition of accident categories and associated source terms for use in estimating the occupational dose from an accident. Standardization in these areas would be helpful in making value-impact results comparable.

The quantitative output of a value-impact analysis is available for comparison with specific limits or guidelines. As described in Section 1, safety goals are being developed; however, the scope of the current safety goals in NUREG-0880 (Ref. 24) does not include occupational exposure. The matrix goals shown in Figure 2-1 provide another basis for evaluating the results of value-impact analyses. It would be of great benefit to have a consistent set of goals against which the results of value-impact analyses can be compared, particularly if the goals specify some of the primary value-laden assumptions that are often made in value-impact analyses.

Further considerations for factoring occupational exposure into value-impact analyses are described in volumes 2 and 3 of this report (Refs. 28 and 29).

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Figure 1-Priority Ranking

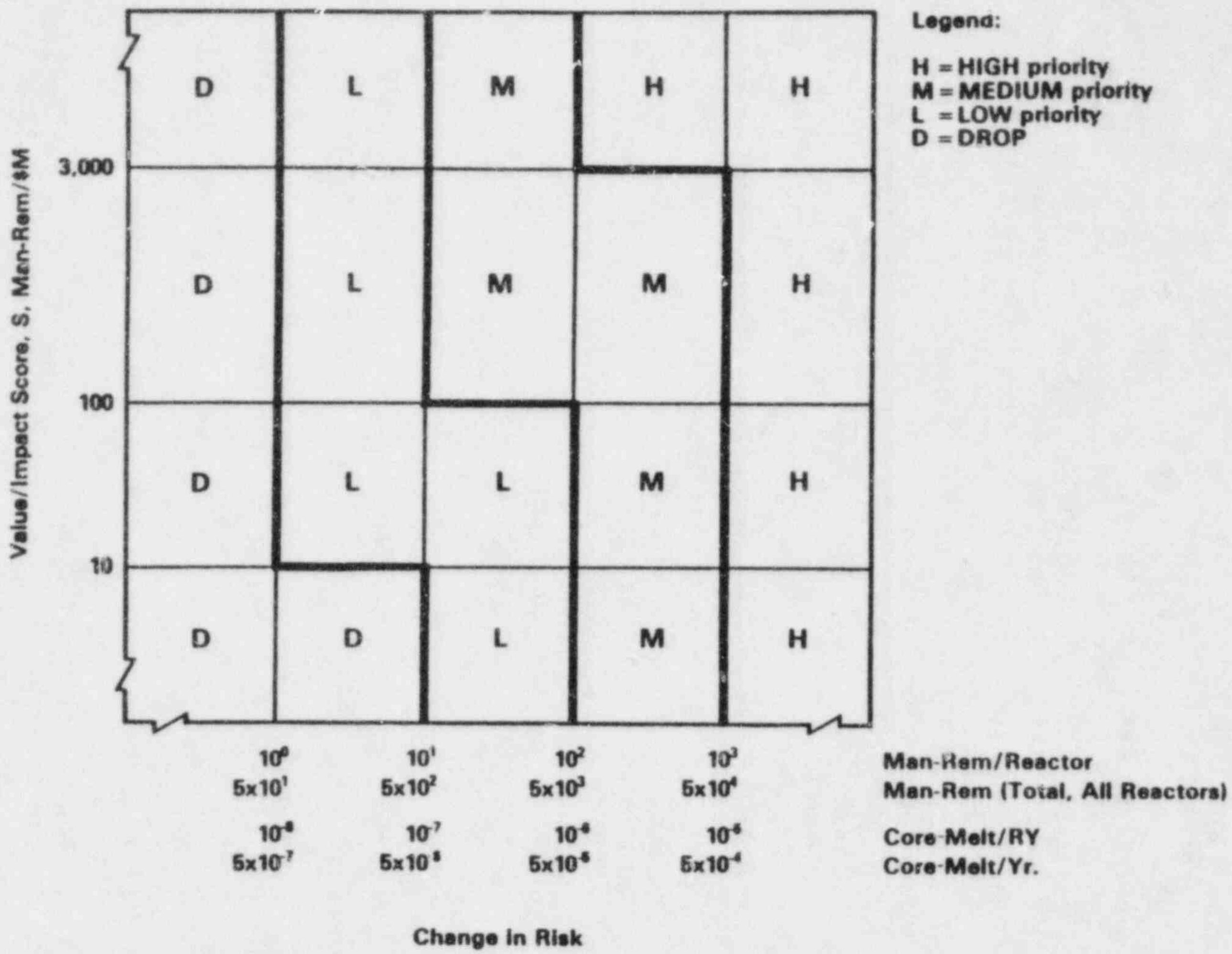

Figure 2-1. Ranking Scheme for Establishing Safety Issue Priority (from NUREG-0933).

Table 2-1. Risk Thresholds for Safety Issue Prioritization (from NUREG-0933).

(a) The priority rank is always HIGH when any of the following risk (or risk-related) thresholds are estimated to be exceeded (or when extraordinary uncertainty suggests that they may well be exceeded):

(1) 1,000 man-rem estimated public dose per remaining reactor lifetime

(2) 50,000 man-rem total estimated for all affected reactors for their remaining lifetime (e.g., 500 man-rem/reactor for 100 reactors)

(3) 10^{-5}/reactor-year large-scale core melt

(4) 5×10^{-4}/year large-scale core melt (total for all affected reactors)

(b) Always at least MEDIUM priority:
10 or more percent of the always-HIGH criteria

(c) Always at least LOW priority:
1 or more percent of the always-HIGH criteria

(d) Never higher than MEDIUM priority:
Less than 10% of the always-HIGH criteria

(e) Never higher than LOW priority:
Less than 1% of the always-HIGH criteria

(f) Always DROP category:
Less than 0.1% of the always-HIGH criteria

Table 2-2. Reference PWR Accident Parameters for Estimating Occupational Exposure From Post-Accident Cleanup Activities (from NUREG/CR-2601.

Parameter	Parameter Value[a] Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Percent of fuel cladding failure	10	50	100
Percent of fuel melting	0	5	50
Volume of sump water (m^3)	200	1000	1600[b]
Depth of sump water (m)	0.2	1.0	1.6
Total fission product radioactivity in sump water (Ci)	2.5×10^4	3.5×10^5	2.5×10^6
Average fission product radioactivity in sump water (Ci/m^3)	125	350	1560
Total fission product radioactivity plated out on building surfaces (Ci)[c]	5	70	500
Average fission product radioactivity on building surfaces (Ci/m^2)			
• Floors	0.001	0.014	0.1
• Walls	0.00001	0.00014	0.001
Average gamma radiation exposure rate at operating floor level (R/hr)			
• Contribution from plateout	0.01	0.15	1.0
• Contribution from sump water	0.015	0.045	0.2
• Total exposure rate	0.025	0.20	1.2
Average gamma radiation exposure rate at lowest entry level (R/hr)			
• Contribution from plateout	0.01	0.15	1.0
• Contribution from sump water	8	30	170
• Total exposure rate	8	30	170
Damage to fuel core	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings on center fuel elements. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout primary coolant system.
Damage to containment building and equipment.	No significant physical damage.	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Polar crane inoperable.	Ventilation ductwork damaged. Doors, catwalks, pipes, and cable conduits dented or ripped away. Loss of electrical and other services. Erosion of concrete and metal surfaces. Polar crane inoperable.
Contamination of auxiliary and fuel buildings	--(d)	Plateout on building surfaces. CVCS contaminated with 20,000 Ci of fission product radioactivity. General area radiation exposure levels about 100 mR/hr.	Plateout on building surfaces. CVCS contaminated with 20,000 Ci of fission product radioactivity. General area radiation exposure levels about 100 mR/hr.

(a) Values refer to conditions inside the containment building approximately 1 year after the postulated accident.
(b) Based on refueling water storage tank volume.
(c) Plateout values are after washdown of the walls by condensing moisture.
(d) Contamination of the auxiliary and fuel buildings is not postulated for the scenario 1 accident.

Table 2-3. Reference BWR Accident Parameters for Estimating Occupational Exposure From Post-Accident Cleanup Activities (from NUREG/CR-2601).

Parameter	Parmeter Value[a]		
	Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Percent of Fuel Cladding Failure	10	50	100
Percent of Fuel Melting	0	5	50
Volume of Suppression Pool Water (m^3)	3160[b]	3160[b]	3160[b]
Total Fission Product Radioactivity in Suppression Pool Water (Ci)	2.5×10^4	3.5×10^5	2.2×10^6
Average Fission Product Radioactivity in Suppression Pool Water (Ci/m^3)	8	110	700
Volume of Reactor Building Sump Water (m^3)	0	0	500
Total Fission Product Radioactivity in Reactor Building Sump Water (Ci)	0	0	3×10^5
Average Fission Product Radioactivity in Reactor Building Sump Water (Ci/m^3)	--	--	700
Total Fission Product Radioactivity Plated Out on Containment Vessel Surfaces (Ci)[c]	5.2	73	460
Average Fission Product Radioactivity on Containment Vessel Surfaces (Ci/m^2)			
• Floors	0.005	0.07	0.44
• Walls	0.00005	0.0007	0.0044
Average Gamma Radiation Exposure Rate at Operating Floor Level Inside Containment (R/hr)			
• Contribution from Plateout	0.052	0.720	4.6
• Contribution from Suppression Pool Water	0.006	0.070	0.5
• Total Exposure Rate	0.058	0.790	5.1
Total Fission Product Radioactivity Plated Out on Reactor Building Surfaces (Ci)	0	10	82
Average Fission Product Radioactivity on Reactor Building Surfaces (Ci/m^2)			
• Floors	-	0.001	0.008
• Walls	--	0.00001	0.00008
Average Gamma Radiation Exposure Rate at Refueling Floor level in Reactor Building (R/hr)[d]			
• Contribution from Plateout	--	0.002	0.020
• Contribution from Sump Water	--	--	0.0
• Total Exposure Rate	--	0.002	0.020

Table 2-3. Reference BWR Accident Parameters for Estimating Occupational Exposure From Post-Accident Cleanup Activities (from NUREG/CR-2601) (Continued).

Parameter	Parameter Value(a) Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Average Gamma Radiation Exposure Rate at Operating Floor Level in Reactor Building(e) (R/hr)			
• Contribution from Plateout	--	0.010	0.083
• Contribution from Sump Water	--	--	0.002
• Total Exposure Rate	--	0.010	0.085
Average Gamma Radiation Exposure Rate at Service Floor Level in Reactor Building(f) (R/hr)			
• Contribution from Plateout	--	0.010	0.083
• Contribution from Sump Water	--	--	30
• Total Exposure Rate	--	0.010	30
Damage to Fuel Core	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings on center fuel elements. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout water recirculation system.
Damage to Containment Vessel and Equipment	No significant physical damage.	Most electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage.	Pipes and cable conduits dented or ripped away. Loss of electrical and other services. Recirculation system pump motors inoperable due to damage to electrical components and corrosion.
Damage to Reactor Building and Equipment	No significant physical damage.	No significant physical damage	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Bridge crane and refueling platform inoperable due to damage to electrical components and corrosion.
Contamination of Radwaste Building	--(g)	--(g)	Plateout on building surfaces. Reactor water cleanup demineralizer system grossly contaminated. General area radiation exposure levels about 50 mR/hr.

(a) Values refer to conditions approximately 1 year after the accident.
(b) Based on maximum water volume specified in Section C.2.1 of Reference 1.
(c) Plateout values are after washdown of walls by condensing moisture.
(d) The refueling floor level is the 185.0-m level. See Figure 16.2-3.
(e) The operating floor level is the 152.7-m level. See Figure 16.2-3.
(f) The service floor level is the 134.4-m level. See Figure 16.2-3.
(g) Contamination of radwaste building is postulated only for the scenario 3 accident.

Table 2-4. Estimated Consequences of PWR Post-Accident Cleanup Activities (from NUREG/CR-2601).

Type of Safety Concern	Source of Safety Concern	Units	Accident Scenario 1	Accident Scenario 2	Accident Scenario 3
	Public Safety[a]				
Radiation Dose	Accident Cleanup Activities[b]	man-rem	6	20	40[c]
	Transportation[d]	man-rem	1.6	4.7	9.6
	Occupational Safety				
Serious Lost-Time Injuries	Accident Cleanup Activities	total no.	0.30	1.3	2.1
	Transportation	total no.	0.17	0.51	1.1
Fatalities	Accident Cleanup Activities	total no.	0.0023	0.0094	0.015
	Transportation	total no.	0.010	0.030	0.066
Radiation Dose	Accident Cleanup Activities	man-rem	670	4 579	12 103
	Transportation	man-rem	17	46	99

(a) Radiation doses from atmospheric releases during normal cleanup activities. Doses resulting from industrial accidents are not included.
(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.
(c) Doses from activities in auxiliary and fuel buildings not calculated, expected to be negligible compared to those shown.
(d) 50-yr committed dose equivalent to the total body, for the population along the transport route.

Table 2-5. Estimated Consequences of BWR Post-Accident Cleanup Activities (from NUREG/CR-2601).

Type of Safety Concern	Source of Safety Concern	Units	Accident Scenario 1	Accident Scenario 2	Accident Scenario 3
	Public Safety[a]				
Radiation Dose	Accident Cleanup Activities[b]	man-rem	6	20	40
	Transportation[c]	man-rem	3	5	11
	Occupational Safety				
Serious Lost-Time Injuries	Accident Cleanup Activities	total no.	0.54	1.0	2.3
	Transportation	total no.	0.31	0.54	1.3
Fatalities	Accident Cleanup Activities	total no.	0.0038	0.0072	0.016
	Transportation	total no.	0.019	0.032	0.076
Radiation Dose	Accident Cleanup Activities	man-rem	1 490	4 170	11 940
	Transportation	man-rem	28	50	120

(a) Radiation doses from atmospheric releases during normal cleanup activities. Doses resulting from industrial accidents are not included.
(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.
(c) 50-yr committed dose equivalent to the total body, for the population along the transport route.

3. CONSIDERATION OF OCCUPATIONAL EXPOSURE IN PROBABILISTIC RISK ASSESSMENT

In the previous section, the use of probabilistic methods in value-impact analysis was described. This use of probabilistic methods is of relatively limited scope in comparison to full-plant probabilistic risk assessments (PRAs) which are addressed in this section.

3.1 SCOPE OF RECENT PROBABILISTIC RISK ASSESSMENTS

Probabilistic risk assessments provide insight into plant response to accident initiating events and establish a systematic framework for estimating the potential consequences and risk from such initiating events. The accidents of interest in a PRA range in severity from design basis accidents (i.e. similar to those evaluated in a licensee Safety Analysis Report) to very severe sequences resulting in core melt and containment failure. The Reactor Safety Study (Ref. 1) was the first comprehensive application of PRA techniques in the analysis of nuclear power plant risk. The specific objective of that study was to "perform a quantitative assessment of risk to the public from reactor accidents." Determining the risk to workers at nuclear power plants from reactor accidents was beyond the scope of the Reactor Safety Study, and has remained beyond the scope of more recent PRA's, including:

- NRC-sponsored PRAs
  - Reactor Safety Study Methodology Applications Program (RSSMAP, Refs. 2 to 5)
  - Interim Reliability Evaluation Program (IREP, Refs. 6 to 9)

- Nuclear Safety Analysis Center (NSAC)-sponsored PRA
  - Oconee Unit 3 (Ref. 10)

- Utility-sponsored PRAs
  - Shoreham (Ref. 11)
  - Oyster Creek (Ref. 12)
  - Limerick (Ref. 13)
  - Indian Point (Ref. 14)

- Foreign PRAs
  - German Risk Study (Ref. 15)

The Risk Reactor Methods Integration and Evaluation Program (RMIEP), which is an NRC-sponsored "full-scope" PRA now being performed by Sandia National Laboratories does not include consideration of risk to workers from reactor accidents.

Recently, procedures guides have been developed as an aid to performing PRAs (Refs. 16, 17). These guides do not address occupational exposure.

In summary, worker risk has been beyond the scope of current PRAs. As will be discussed in the following sections inclusion of worker risk in a PRA could greatly increase the complexity of an already complex undertaking. An alternative is to examine worker risk from reactor accidents, using simplified probabilistic methods in an analytical framework that is seperate from a plant-specific PRA effort (i.e. similar to the value-impact methods described in Section 2).

3.2 OVERVIEW OF PRA METHODS

A PRA includes three major phases: systems analysis, containment analysis and consequence/risk analysis. An event tree is a conventional PRA tool for describing gross plant response to a specific initiating event (or to a class of initiating events). Event trees are usually supplemented by fault tree or other suitable models to describe the detailed response of plant systems and operating personnel. An example of an event tree for a loss of coolant accident at a PWR plant is shown in Figure 3-1 (from Ref. 1). Each branch, or path, in this event tree represents a particular accident sequence for which a probability of occurrence and plant-related consequences can be estimated. Additional steps are then taken in a PRA to: (a) establish groupings (i.e., release categories, or "bins") of accident sequences having similar plant-related consequences, and (b) define an appropriate radioactive material source term for each grouping of accident sequences. Potential consequences can be estimated by modeling the release of the respective source terms, their dispersion, and the resulting internal and/or external dose to the exposed population. An estimate of overall risk then can be calculated (i.e., the sum of the products of group probability times consequences).

The basic steps described above are illustrated in Tables 3-1 to 3-3 using examples from WASH-1400 (Ref. 1). Table 3-1 illustrates how PWR large LOCA accident sequences were grouped into nine "release categories", and also divided into "dominant" or "other" large LOCA accident sequences based on sequence probability. A composite grouping of dominant PWR sequences for large LOCAs, other LOCAs and transients is shown in Table 3-2. The estimated source terms for each PWR release category are included in Table 3-3 (which also includes five BWR release categories). The WASH-1400 release categories are described in Appendix B.

3.3 ANALYSIS OF OCCUPATIONAL EXPOSURE FROM REACTOR ACCIDENTS

A worker may incur occupational exposure either during a reactor accident or during such post-accident recovery activities as cleanup, repair, retrofit or decommissioning (occupational exposures D6, D7, D3 and D8 in Figure 1-1). Occupational exposure related to reactor accidents can be reduced by several measures, including:

- Implementation of retrofits that will correct plant deficiencies and: (a) eliminate or reduce the probability of occurrence of an accident, or (b) reduce the potential consequences of an accident.
- Implementation of effective, integrated control room enhancements and Emergency Operating Procedures to improve the ability of operating personnel to respond to and mitigate accidents that may occur.

- Implementation of an effective Site Emergency Plan to ensure timely protective actions on behalf of site personnel, including evacuation, when necessary.

- Planning of post-accident recovery activities to ensure implementation of ALARA philosophy and optimization of occupational exposure.

Occupational exposures due to retrofit activities have been discussed in Section 2. It is important to note that the occupational exposure detriment from retrofit activities should be balanced against the avoided occupational exposure due to a reduction in accident probability and/or consequences. These considerations have been addressed in some value-impact assessments using probabilistic methods. The other listed items relate to limiting the potential consequences once an initiating event has occurred.

The control room enhancements and the Site Emergency Plan provide means for reducing occupational exposure during the accident response phase; however, use of probabilistic methods to estimate the risk reduction is not required by the respective NRC implementing documents (Refs. 18 to 20). Current PRA technology has only a limited capability to model potential operator interfaces during the course of an accident. Thus, a probabilistic risk assessment of occupational exposure during an accident would be of limited scope, making it difficult to assess the occupational risk benefits of control room enhancements and Emergency Plans.

Recovery activities may begin once a plant has been placed in a stable condition following an accident. Assessments have been made of occupational exposure during the recovery phase following a reactor accident (Ref. 21). Recovery can be a relatively long-term concern that can extend for weeks, months, or as in the case of Three Mile Island, for years after an accident. Radioactive source terms and recovery activities (i.e. cleanup, repair, retrofit, decommissioning) can be relatively well defined during the recovery period, thus conventional ALARA techniques should be applicable to occupational exposure during the recovery period.

3.3.1 Analysis of Occupational Exposure During a Reactor Accident

3.3.1.1 Curent Analysis of Occupational Exposure During a Reactor Accident

Doses to control room personnel during a spectrum of design basis accidents are routinely calculated as part of a licensee Safety Analysis Report (SAR). The purpose of this analysis is to demonstrate compliance with the limit for occupational exposure during accidents, as specified in General Design Criterion 19 of 10CFR50, Appendix A (see Section 1.1). Accidents postulated and analyzed in SARs are grouped by a measure of expected frequency of occurrence into the following frequency groups:

- incidents of moderate frequency
- infrequent incidents
- limiting faults

In spite of this grouping of accidents based on expected frequency of occurrence, SAR accident analysis is not a probabilistic risk assessment.

Risk is not estimated in SAR analysis and the spectrum of postulated accident sequences included in an SAR is relatively limited in comparison to those found in a PRA. The following are the primary constraints which limit the scope of SAR accident analysis (Ref. 22):

- The most adverse conditions within the allowed operating range, as defined in the plant Technical Specifications, are used as the initial plant conditions for accident analysis.

- A single, random failure in a required mitigating system or a single operator error is assumed to occur in addition to the initiating event (and any other events that are a direct consequence of the initiating event.)

From an initial state of high safety system availability, the ultimate course of an accident as described in a Safety Analysis Report, is constrained by the single failure criterion. In contrast, PRAs generate a larger number and more severe accident sequences by virtue of considering multiple system failures and/or operator errors. The SAR design basis accident sequences correspond roughly to the most benign release categories defined in the Reactor Safety Study (PWR 9 and BWR 5, see Appendix B).

Although estimates of potential occupational exposure during accidents are available in Safety Analysis Reports, these may not be representative of potential exposures resulting from the range of degraded plant conditions included in PRAs.

3.3.1.2 Use of PRAs to Estimate Occupational Exposure During Postulated Reactor Accident

To assess the potential occupational exposure during a reactor accident, it is necessary to establish the status of the following:

- relevant accident sequences
- source term
- post-accident operator actions
- spatial and temporal distribution of plant personnel with respect to the source term
- protective features (i.e. shielding, emergency ventilation systems, protective clothing or breathing apparatus)
- protective actions (i.e. evacuation of unnecessary personnel)

As described previously, the accident sequences considered in PRAs usually range in severity from design basis accidents to very severe core-melt sequences. These are the primary contributors to public risk. Minor incidents having negligible impact on public risk are usually not considered in a PRA. As shown in Figure 1-1, minor operational incidents resulting in spills and contamination may have a measurable impact on occupational exposure, and, therefore, should not be excluded from a PRA intended for estimating occupational risk.

Source terms for calculation of offsite exposure are shown in Table 3-3. These are the radionuclides released from the plant to the environment during the course of an accident. The source terms in Tables 2-1 and 2-2 define the radionuclides present in the plant when recovery actions begin after an accident. It may be necessary to define more complex time-dependent source terms to estimate occupational exposure during postulated accidents. The time dependencies of concern involve the magnitude and distribution of the "occupational source term" as radioactive material is released from the core to containment and subsequently: (a) is spatially distributed within containment, and to other buildings, (b) decays, and (c) is released to the environment. The number of occupational "release categories" needed to account for these time-dependencies is unknown.

Post-accident operator actions should be governed by Emergency Operating Procedures (EOPs) and by the ability of automatic safety systems to adequately mitigate an accident. Operating experience has demonstrated, however, that unforseen conditions such as operator response errors, maintenance and testing errors, multiple equipment failures, system design deficiencies and other similar situations can lead to higher probability event sequences, or to accident sequences that may not readily be predicted by PRAs. The Accident Sequence Precursor (ASP) study provides evidence to support this contention (Ref. 23). Therefore, it will be very difficult to predict all possible operator actions and the spatial and temporal distribution of some plant personnel during accidents. Current PRA methods do not have the capability to support the modeling of such a range of detailed operator actions. It, therefore, would be necessary to: (a) develop an improved modeling capability, and (b) determine the availability of adequate data or the need for additional data to permit quantification of the improved model.

The plant model used in current PRAs generally does not include many of the structures, systems and equipment available for protecting plant personnel against radiation and airborne radioactive material (e.g. control room ventilation system, installed shielding, intervening structures, etc.) To estimate occupational exposure in the context of a PRA, it would be necessary to expand the plant model so that the availability and reliability of these relevant protective features can be treated.

Evacuation models have been included in PRAs to assess the impact of such protective actions on public risk. An evacuation model for plant personnel could certainly be developed, and would be an important element in estimating occupational exposure during a reactor accident.

3.3.2 Analysis of Occupational Exposure Following a Reactor Accident

As described previously, a utility has two basic options following a serious reactor accident: (a) cleanup, repair, refurbish and restore the plant to operation, or (b) cleanup and decommission the plant. The occupational exposures associated with the latter option have been evaluated in NUREG/CR-2601 (Ref. 21), and exposures associated with the former option are expected to be similar (Ref. 24). Expected exposures due to cleanup activities following reactor accidents have been discussed in Section 2.

3.4 RECOMMENDATIONS FOR CONSIDERATION OF OCCUPATIONAL EXPOSURE IN PRAs

Current PRAs do not account for occupational risk from reactor accidents. An expanded PRA model incorporating the features to account for occupational risk will likely be more complex than current or anticipated PRA models for estimating public risk from reactor accidents. The Square Law of Computation (Ref. 25) suggests that the amount of computation involved increases at least as fast as the square of the number of equations. Thus the impact of increasing the complexity of a PRA model to account for occupational risk may be surprisingly large. The benefits of developing such a PRA model are not clearly established; therefore, no recommendation is made for developing such a modeling capability. Occupational exposures during and following a reactor accident can be estimated using simpler analytical techniques, such as the value-impact analysis methods described in Section 2.

3.5 SECTION 3 REFERENCES

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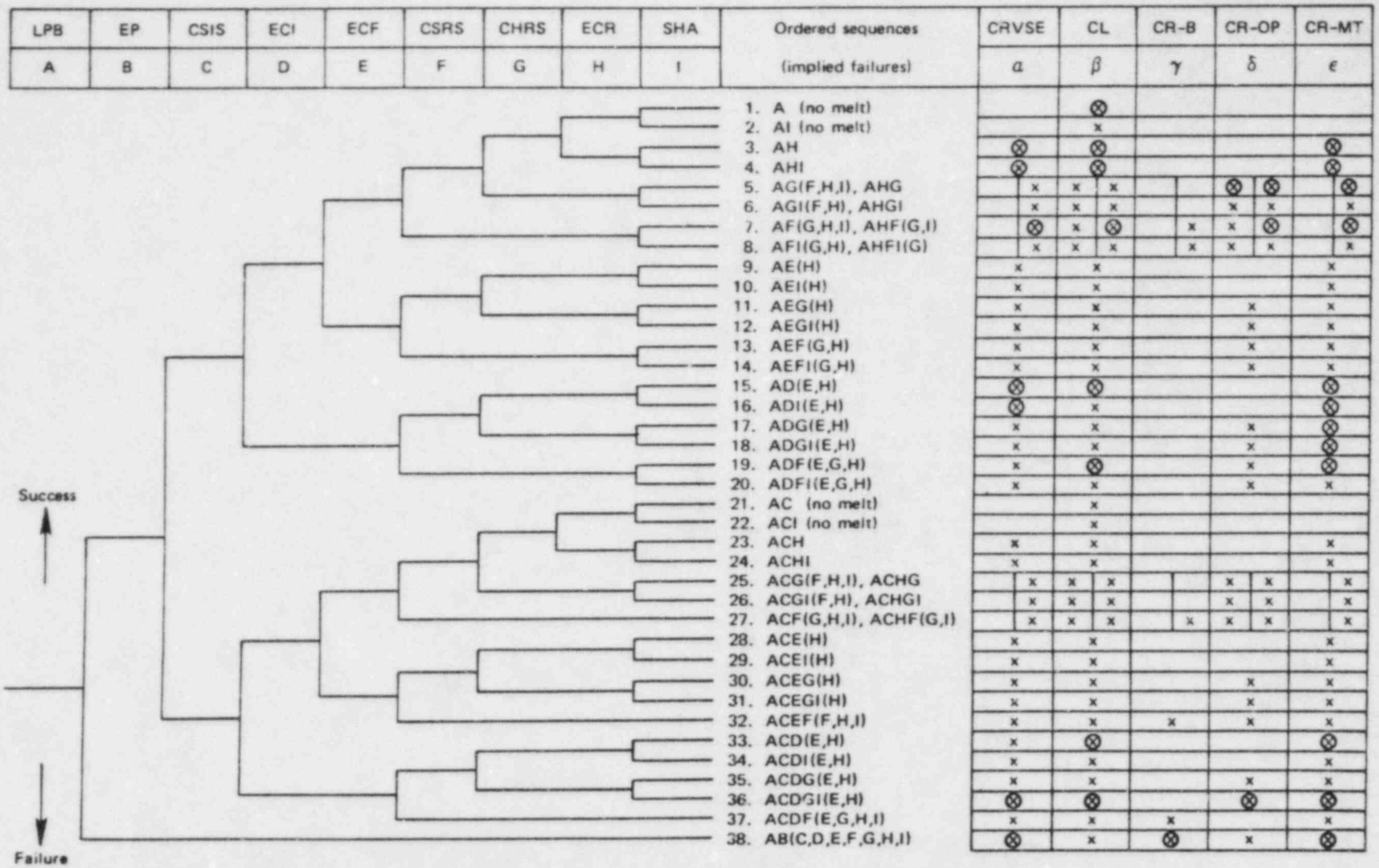

Figure 3-1. PWR Large LOCA Event Tree (from WASH-1400).

Table 3-1. PWR Large LOCA Accident Sequences vs. Release Categories (from WASH-1400).

Core melt | No core melt

Release Categories									
	1	2	3	4	5	6	7	8	9
Dominant Large LOCA Accident Sequences With Point Estimates									
	AB-α $1x10^{-11}$ AF-α $1x10^{-10}$ ACD-α $5x10^{-11}$ AG-α $9x10^{-11}$	AB-γ $1x10^{-10}$ AHF-γ $2x10^{-11}$ AB-δ $4x\ 10^{-11}$	AD-α $2x10^{-8}$ AH-α $1x10^{-8}$ AF-δ $1x10^{-8}$ AG-δ $9x10^{-9}$	ACD-β $1x10^{-11}$	AD-β $4x10^{-9}$ AH-β $3x10^{-9}$	AB-ε $1x10^{-9}$ ADF-ε $2x10^{-10}$ AHF-ε $1x10^{-10}$	AD-ε $2x10^{-6}$ AH-ε $1x10^{-6}$	A-β $2x10^{-7}$	A $1x10^{-4}$
Other Large LOCA Accident Sequences									
	ACDGI-α AHFI-α ACHF-α ACDI-α ACDG-α AGI-α AFI-α ACG-α ACGI-α ACF-α ACDF-α ACEI-α ACEG-α ACEGI-α ACEF-α ACE-α AHF-α	ADF-β AHFI-δ ACHF-δ ACHF-γ ACDF-γ ACF[illegible]-γ AHFI-β ADFI-β ACHF-β ACDF-β AHF-δ AHFI-γ AEF-β AEFI-β ACEF-β AEF-δ AEFI-δ ACEF-δ AB-β AHF-β	AHG-α AHGI-α ADF-α ADFI-α ACH-α ACHI-α ACHG-α ACHGI-α AGI-δ AFI-δ ACG-δ ACGI-δ ACF-δ AHI-α ADGI-α ADI-α ADG-α AE-α AEI-α AEF-α AEFI-α AEG-α AEGI-α	ACDGI-β ADG- β ACDI-β ACDG-β ADGI- β ACE- β ACEI- β ACEG- β ACEGI-β AEG-β AEGI-β	AHI- β AHG- β AHGI-β ADI-β ACH-β ACHI-β ACHG-β AE- β AEI-β	ACHGI- ε AHFI- ε ADFI- ε ACDF- ε ACDGI-ε ACHF-ε AEF- ε AEFI-ε ACEF-ε ACEGI-ε	AHG-δ AHGI- δ AHGI- ε ACH-ε ACHI- ε ACHG- δ ACHG- ε ACHGI- ε ACDI- ε ACDG- δ ACDG- ε ADG- δ ADGI-δ AHG- ε ADI- ε ADG- ε ACD-ε ADGI-ε AHI-ε AE-ε AEI-ε ACE-ε ACEI-ε ACEG-ε ACEG-δ ACEGI-δ ACHGI-δ AEG-δ AEGI-δ AEG-ε AEGI-ε	AI-β AC-β ACI-β	AI AC ACI
$A\Sigma_p^{(a)}$	3×10^{-10}	2×10^{-10}	5×10^{-8}	1×10^{-11}	7×10^{-9}	1×10^{-9}	3×10^{-6}	2×10^{-7}	1×10^{-4}

(a) Σ_p is the arithmetic sum of the probabilities of the accident sequence in each release category.

Table 3-2. PWR Dominant Accident Sequences vs. Release Categories (from WASH-1400).

RELEASE CATEGORIES							Core Melt	No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AHF-γ 2×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ε 1×10^{-9} AHF-ε 1×10^{-10} ADF-ε 2×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 3×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 5×10^{-9} S_1D-β 6×10^{-9}	S_1DF-ε 3×10^{-10} S_1B-ε 2×10^{-9} S_1HF-ε 4×10^{-10}	S_1D-ε 3×10^{-6} S_1H-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-9}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-9} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-8}	S_2G-β 1×10^{-12}	S_2D-β 2×10^{-8} S_2H-β 1×10^{-8}	S_2B-ε 8×10^{-9} S_2CD-ε 2×10^{-8} S_2HF-ε 1×10^{-9}	S_2D-ε 9×10^{-6} S_2H-ε 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-δ 1×10^{-12}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-6}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 3×10^{-10}	TMLB'-ε 6×10^{-7}	TML-ε 6×10^{-6} TKQ-ε 3×10^{-6} TKMQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

Table 3-3. Summary of Accidents Involving Core (from WASH-1400).

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED: Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
PWR 1	$9x10^{-7}$	2.5	0.5	1.0	25	520(d)	0.9	$6x10^{-3}$	0.7	0.4	0.4	0.05	0.4	$3x10^{-3}$
PWR 2	$8x10^{-6}$	2.5	0.5	1.0	0	170	0.9	$7x10^{-3}$	0.7	0.5	0.3	0.06	0.02	$4x10^{-3}$
PWR 3	$4x10^{-6}$	5.0	1.5	2.0	0	6	0.8	$6x10^{-3}$	0.2	0.2	0.3	0.02	0.03	$3x10^{-3}$
PWR 4	$5x10^{-7}$	2.0	3.0	2.0	0	1	0.6	$2x10^{-3}$	0.09	0.04	0.03	$5x10^{-3}$	$3x10^{-3}$	$4x10^{-4}$
PWR 5	$7x10^{-7}$	2.0	4.0	1.0	0	0.3	0.3	$2x10^{-3}$	0.03	$9x10^{-3}$	$5x10^{-3}$	$1x10^{-3}$	$6x10^{-4}$	$7x10^{-5}$
PWR 6	$6x10^{-6}$	12.0	10.0	1.0	0	N/A	0.3	$2x10^{-3}$	$8x10^{-4}$	$8x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-5}$	$1x10^{-5}$
PWR 7	$4x10^{-5}$	10.0	10.0	1.0	0	N/A	$6x10^{-3}$	$2x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$
PWR 8	$4x10^{-5}$	0.5	0.5	N/A	0	N/A	$2x10^{-3}$	$5x10^{-6}$	$1x10^{-4}$	$5x10^{-4}$	$1x10^{-6}$	$1x10^{-8}$	0	0
PWR 9	$4x10^{-4}$	0.5	0.5	N/A	0	N/A	$3x10^{-6}$	$7x10^{-9}$	$1x10^{-7}$	$6x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0
BWR 1	$1x10^{-6}$	2.0	2.0	1.5	25	130	1.0	$7x10^{-3}$	0.40	0.40	0.70	0.05	0.5	$5x10^{-3}$
BWR 2	$6x10^{-6}$	30.0	3.0	2.0	0	30	1.0	$7x10^{-3}$	0.90	0.50	0.30	0.10	0.03	$4x10^{-3}$
BWR 3	$2x10^{-5}$	30.0	3.0	2.0	25	20	1.0	$7x10^{-3}$	0.10	0.10	0.30	0.01	0.02	$3x10^{-3}$
BWR 4	$2x10^{-6}$	5.0	2.0	2.0	25	N/A	0.6	$7x10^{-4}$	$8x10^{-4}$	$5x10^{-3}$	$4x10^{-3}$	$6x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$
BWR 5	$1x10^{-4}$	3.5	5.0	N/A	150	N/A	$5x10^{-4}$	$2x10^{-9}$	$6x10^{-11}$	$4x10^{-9}$	$8x10^{-12}$	$8x10^{-14}$	0	0

APPENDIX A- NUREG/CR-2601 WORK SHEETS FOR ASSESSING CHANGE IN CORE MELT FREQUENCY AND OCCUPATIONAL EXPOSURE

NUREG/CR-2800 Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:
2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):
   (include total number of each plant-type - BWR, PWR)
3. Plants Selected for Analysis:
   (must have known risk equations, e.g., Oconee 3 in Appendix A)
4. Parameters Affected by SIR:
   (from Table A.4 or B.4 in the appendices; document any modifications)
5. Base-Case Values for Affected Parameters:
   (if these differ from those values given in Table A.4 or B.4, document the calculations)
6. Affected Accident Sequences and Base-Case Frequencies:
   (from Table A.3 or B.3 in the appendices; also list the release categories to which they contribute)
7. Affected Release Categories and Base-Case Frequencies:
   (from Table A.1 or B.1 in the appendices)
8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):
9. Base-Case, Affected Public Risk (W):
10. Adjusted-Case, Affected Values for Affected Parameters:
    (document the assumptions and calculations)
11. Affected Accident Sequences and Adjusted-Case Frequencies:
    (relist the sequences and the release categories to which they contribute from step 6, but with the adjusted-case frequencies)
12. Affected Release Categories and Adjusted-Case Frequencies:
    (relist the categories from step 7, but with the adjusted-case frequencies)
13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}^*$):
14. Adjusted-Case, Affected Public Risk (W^*):
15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):
16. Per-Plant Reduction in Public Risk (ΔW):
17. Total Public Risk Reduction, $(\Delta W)_{Total}$:
    (also list the upper and lower bounds)

NUREG/CR-2800 Occupational Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

2. Affected Plants (N):
   (include total number of each plant-type - BWR, PWR. Divide each type into backfit and forward-fit classes)

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

4. Per-Plant Occupational Dose Reduction Due to Accident-Avoidance, $\Delta(\bar{F}D_R)$:

5. Total Occupational Dose Reduction Due to Accident-Avoidance (ΔU):
   (also list upper and lower bounds)

6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:

7. Per-Plant Occupational Dose Increase for SIR Implementation (D):

8. Total Occupational Dose Increase for SIR Implementation (ND):

9. Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (D_o):

11. Total Occupational Dose Increase for SIR Operation and Maintenance ($N\bar{T}D_o$):

12. Total Occupational Dose Increase (G):
    (also list upper and lower bounds)

APPENDIX B - DEFINITION OF WASH-1400 RELEASE CATEGORIES AND NUREG/CR-2601 ACCIDENT SCENARIOS

WASH-1400 PWR Release Categories

PWR 1

This release category can be characterized by a core meltdown followed by a steam explosion on contact of molten fuel with the residual water in the reactor vessel. The containment spray and heat removal systems are also assumed to have failed and, therefore, the containment could be at a pressure above ambient at the time of the steam explosion. It is assumed that the steam explosion would rupture the upper portion of the reactor vessel and breach the containment barrier, with the result that a substantial amount of radioactivity might be released from the containment in a puff over a period of about 10 minutes. Due to the sweeping action of gases generated during containment-vessel meltthrough, the release of radioactive materials would continue at a relatively low rate thereafter. The total release would contain approximately 70% of the iodines and 40% of the alkali metals present in the core at the time of release.[1] Because the containment would contain hot pressurized gases at the time of failure, a relatively high release rate of sensible energy from the containment could be associated with this category. This category also includes certain potential accident sequences that would involve the occurrence of core melting and a steam explosion after containment rupture due to overpressure. In these sequences, the rate of energy release would be lower, although still relatively high.

PWR 2

This category is associated with the failure of core-cooling systems and core melting concurrent with the failure of containment spray and heat-removal systems. Failure of the containment barrier would occur through overpressure, causing a substantial fraction of the containment atmosphere to be released in a puff over a period of about 30 minutes. Due to the sweeping action of gases generated during containment vessel meltthrough, the release of radioactive material would continue at a relatively low rate thereafter. The total release would contain approximately 70% of the iodines and 50% of the alkali metals present in the core at the time of release. As in PWR release category 1, the high temperature and pressure with' ı containment at the time of containment failure would result in a relatively high release rate of sensible energy from the containment.

PWR 3

This category involves an overpressure failure of the containment due to failure of containment heat removal. Containment failure would occur prior to the commencement of core melting. Core melting then would cause radioactive materials to be released through a ruptured containment barrier. Approximately 20% of the iodines and 20% of the alkali metals present in the core at the time of release would be released to the atmosphere. Most of the release would occur over a period of about 1.5 hours. The release of radioactive material from containment would be caused by the sweeping action of gases generated by the reaction of the molten fuel with concrete. Since these gases would be initially heated by contact with the melt, the rate of sensible energy release to the atmosphere would be moderately high.

PWR 4

This category involves failure of the core-cooling system and the containment spray injection system after a loss-of-coolant accident, together with a concurrent failure of the containment system to properly isolate. This would result in the release of 9% of the iodines and 4% of the alkali metals present in the core at the time of release. Most of the release would occur continuously over a period of 2 to 3 hours. Because the containment recirculation spray and heat-removal systems would operate to remove heat from the containment atmosphere during core melting, a relatively low rate of release of sensible energy would be associated with this category.

WASH-1400 PWR Release Categories (continued)

PWR 5

This category involves failure of the core cooling systems and is similar to PWR release category 4, except that the containment spray injection system would operate to further reduce the quantity of airborne radioactive material and to initially suppress containment temperature and pressure. The containment barrier would have a large leakage rate due to a concurrent failure of the containment system to properly isolate, and most of the radioactive material would be released continuously over a period of several hours. Approximately 3% of the iodines and 0.9% of the alkali metals present in the core would be released. Because of the operation of the containment heat-removal systems, the energy release rate would be low.

PWR 6

This category involves a core meltdown due to failure in the core cooling systems. The containment sprays would not operate, but the containment barrier would retain its integrity until the molten core proceeded to melt through the concrete containment base mat. The radioactive materials would be released into the ground, with some leakage to the atmosphere occurring upward through the ground. Direct leakage to the atmosphere would also occur at a low rate prior to containment-vessel meltthrough. Most of the release would occur continuously over a period of about 10 hours. The release would include approximately 0.08% of the iodines and alkali metals present in the core at the time of release. Because leakage from containment to the atmosphere would be low and gases escaping through the ground would be cooled by contact with the soil, the energy release rate would be very low.

PWR 7

This category is similar to PWR release category 6, except that containment sprays would operate to reduce the containment temperature and pressure as well as the amount of airborne radioactivity. The release would involve 0.002% of the iodines and 0.001% of the alkali metals present in the core at the time of release. Most of the release would occur over a period of 10 hours. As in PWR release category 6, the energy release rate would be very low.

PWR 8

This category approximates a PWR design basis accident (large pipe break), except that the containment would fail to isolate properly on demand. The other engineered safeguards are assumed to function properly. The core would not melt. The release would involve approximately 0.01% of the iodines and 0.05% of the alkali metals. Most of the release would occur in the 0.5-hour period during which containment pressure would be above ambient. Because containment sprays would operate and core melting would not occur, the energy release rate would also be low.

PWR 9

This category approximates a PWR design basis accident (large pipe break), in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt. It is assumed that the minimum required engineered safeguards would function satisfactorily to remove heat from the core and containment. The release would occur over the 0.5-hour period during which the containment pressure would be above ambient. Approximately 0.00001% of the iodines and 0.00006% of the alkali metals would be released. As in PWR release category 8, the energy release rate would be very low.

WASH-1400 BWR Release Categories

BWR 1

This release category is representative of a core meltdown followed by a steam explosion in the reactor vessel. The latter would cause the release of a substantial quantity of radioactive material to the atmosphere. The total release would contain approximately 40% of the iodines and alkali metals present in the core at the time of containment failure. Most of the release would occur over a 1/2 hour period. Because of the energy generated in the steam explosion, this category would be characterized by a relatively high rate of energy release to the atmosphere. This category also includes certain sequences that involve overpressure failure of the containment prior to the occurrence of core melting and a steam explosion. In these sequences, the rate of energy release would be somewhat smaller than for those discussed above, although it would still be relatively high.

BWR 2

This release category is representative of a core meltdown resulting from a transient event in which decay-heat-removal systems are assumed to fail. Containment overpressure failure would result, and core melting would follow. Most of the release would occur over a period of about 3 hours. The containment failure would be such that radioactivity would be released directly to the atmosphere without significant retention of fission products. This category involves a relatively high rate of energy release due to the sweeping action of the gases generated by the molten mass. Approximately 90% of the iodines and 50% of the alkali metals present in the core would be released to the atmosphere.

BWR 3

This release category represents a core meltdown caused by a transient event accompanied by a failure to scram or failure to remove decay heat. Containment failure would occur either before core melt or as a result of gases generated during the interaction of the molten fuel with concrete after reactor-vessel meltthrough. Some fission-product retention would occur either in the suppression pool or the reactor building prior to release to the atmosphere. Most of the release would occur over a period of about 3 hours and would involve 10% of the iodines and 10% of the alkali metals. For those sequences in which the containment would fail due to overpressure after core melt, the rate of energy release to the atmosphere would be relatively high. For those sequences in which overpressure failure would occur before core melt, the energy release rate would be somewhat smaller, although still moderately high.

BWR 4

This release category is representative of a core meltdown with enough containment leakage to the reactor building to prevent containment failure by overpressure. The quantity of radioactivity released to the atmosphere would be significantly reduced by normal ventilation paths in the reactor building and potential mitigation by the secondary containment filter systems. Condensation in the containment and the action of the standby gas treatment system on the releases would also lead to a low rate of energy release. The radioactive material would be released from the reactor building or the stack at an elevated level. Most of the release would occur over a 2-hour period and would involve approximately 0.08% of the iodines and 0.5% of the alkali metals.

BWR 5

This category approximates a BWR design basis accident (large pipe break) in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into containment. The core would not melt, and containment leakage would be small. It is assumed that the minimum required engineered safeguards would function satisfactorily. The release would be filtered and pass through the elevated stack. It would occur over a period of about 5 hours while the containment is pressurized above ambient and would involve approximately 6×10^{-9} % of the iodines and 4×10^{-7} % of the alkali metals. Since core melt would not occur and containment heat-removal systems would operate, the release to the atmosphere would involve a negligibly small amount of thermal energy.

NUREG/CR-2800 PWR Accident Scenarios

1. A small LOCA (a small steam line break or the inadvertent opening of a safety or relief valve) in which emergency core cooling functions to cool the core and limit the release of radioactivity. Some fuel cladding rupture is postulated, but no fuel melting. The consequence scenario includes a small amount of contaminated water in sumps and on floors and moderate contamination of the containment building.
2. A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The consequence scenario includes extensive radioactive contamination of the containment building, moderate radioactive contamination of the auxiliary and fuel buildings, and minor physical damage to buildings and equipment.
3. A major LOCA in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The consequence scenario includes severe radioactive contamination of the containment building, moderate radioactive contamination of the auxiliary and fuel buildings, and major physical damage to structures and equipment.

NUREG/CR-2800 BWR ACCIDENT SCENARIOS

1) A small loss-of-coolant accident (LOCA) in which emergency core cooling functions to cool the core and limit the release of radioactivity. The accident is postulated to result in 10% fuel cladding failure, no fuel melting, moderate contamination inside the containment vessel, no significant radioactive contamination in the reactor building, and no significant physical damage.

2) A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The consequence scenario includes extensive radioactive contamination inside the containment vessel, minor contamination in the reactor building, and minor physical damage to equipment inside the containment.

3) A major LOCA in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The consequence scenario includes extensive radioactive contamination inside the containment vessel and in the reactor building, and major physical damage to structures and equipment. The scenario 3 accident is also postulated to result in some contamination in the radwaste building.

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13. ABSTRACT (200 words or less)

This report reviews current value-impact analysis and probabilistic risk assessment methods and discusses the manner and degree to which these methods consider occupational radiation exposure that may form a variety of in-plant activities, including: (a) normal operation and maintenance, (b) repair, (c) retrofit, (d) minor incidents and cleanup, (e) major accidents, and (f) decommissioning. Value-impact analysis methods which include occupational exposure as an element of the value-impact equation have been developed, however, no standard approach to such analysis has been adopted. Comparison of the results of value-impact analyses must, therefore, be done with caution because different value-laden assumptions made by the analyst can have strong effects on the outcome. Such assumptions include the monetary equivalent of a person-rem, and the relative value of occupational and public exposure. Probabilistic methods have been used in value-impact evaluations to quantify incremental or averted occupational exposure from reactor accidents, however, occupational exposure has not been addressed in probabilistic risk assessments (PRAs) of nuclear power plants. Consideration of occupational exposure in a PRA would greatly increase the complexity of the plant model and the benefits from such an analysis are uncertain. In lieu of expanding the scope of PRAs to address occupational risk, the separate, limited-scope probabilistic evaluations developed for value-impact analysis should provide a more practical analytical capability to support the evaluation and optimization of occupational and public radiation exposure.

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AT NUCLEAR POWER PLANTS

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