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Review of Light Water Reactor Regulatory Requirements

Assessment of Selected Regulatory Requirements That May Have Marginal Importance To Risk

- Postaccident Sampling System
- Turbine Missiles
- Combustible Gas Control
- Charcoal Filters

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ABSTRACT

In a study commissioned by the Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory (PNL) evaluated the costs and benefits of modifying regulatory requirements in the areas of the postaccident sampling system, turbine rotor design reviews and inspections, combustible gas control for inerted Boiling Water Reactor (BWR) containments, and impregnated charcoal filters in certain plant ventilation systems. The basic framework for the analyses was that presented in the Regulatory Analysis Guidelines (NUREG/BR-0058) and in the Handbook for Value-Impact Assessment (NUREG/CR-3568). The effects of selected modifications to regulations were evaluated in terms of such factors as public risk and costs to industry and NRC. The results indicate that potential modifications of the regulatory requirements in three of the four areas would have little impact on public risk. In the fourth area, impregnated charcoal filters in building ventilation systems do appear to limit risks to the public and plant staff. Revisions in the severe accident source term assumptions, however, may reduce the theoretical value of charcoal filters. The cost analysis indicated that substantial savings in operating costs may be realized by changing the interval of turbine rotor inspections. Small to moderate operating cost savings may be realized through postulated modifications to the postaccident sampling system requirements and to the requirements for combustible gas control in inerted BWR containments. Finally, the use of impregnated charcoal filters in ventilation systems appears to be the most cost-effective method of reducing radioiodine concentrations.

SUMMARY

BACKGROUND AND OBJECTIVE

The U.S. Nuclear Regulatory Commission (NRC) has initiated a program to review current light water reactor (LWR) regulatory requirements to see if some could be relaxed or eliminated to reduce regulatory burdens without compromising public health and safety (Federal Register, October 3, 1984). Pacific Northwest Laboratory (PNL), which is operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute, is conducting a series of studies in support of this NRC program. The purpose of this report is to present information on the risks, costs and benefits of possible modifications to regulatory requirements in four areas:

- postaccident sampling system (PASS)
- turbine rotor design reviews and inspections
- control of combustible gas in reactor containment buildings
- use of impregnated charcoal filters in selected plant ventilation systems.

These four areas of regulation were selected by NRC staff for analysis in the second year of the regulatory review program. Previous analyses are described in Volumes 1 and 2 of this series (Mullen 1986a and 1986b).

CONCLUSIONS

Analyses were performed to assess the effects of selected modifications to regulatory requirements in the four areas. The basic framework for the analyses was that presented in the Regulatory Analysis Guidelines, NUREG/BR-0058 (NRC 1984a) and in the Handbook for Value-Impact Assessment, NUREG/CR-3568 (Heaberlin et al. 1983). Probabilistic risk assessment, supplemented by other considerations where appropriate, was used to evaluate the risk significance of modifying the requirements.

The results of the analyses are summarized in Table S.1. Several comments and observations concerning the table are provided below.

- The four areas of regulation cover a range of different types of regulatory requirements, and the analyses considered a variety of alternative regulatory considerations. For PASS, the regulatory modifications examined were to relax the time required in which to obtain certain samples and the elimination of several samples. For turbine rotor design reviews and inservice inspections, the regulatory modification considered was the complete elimination of the NRC design reviews and inservice inspection requirements. For combustible gas control in reactor containments, the modification

TABLE S.1. Summary of Risk Impacts, Benefits and Benefit-Risk Comparisons--Total for All Affected Reactors

Area of Regulation	Effect on Public Risk[a] If Requirements Were Modified	Benefits[b] of Modifying Requirements	Benefit-Risk Comparison If Requirements Were Modified (dollars saved per person-rem of risk)
Postaccident Sampling System[c]	Marginal	Marginal	NA[d]
Turbine Missiles[e]	Marginal	Marginal[f]	NA[d]
Combustible Gas Control[g]	Marginal	Greater than $\$10^7$	NA[d]
Impregnated Charcoal Filters[h]	11,000 person-rem	Marginal	NA[d]

(a) Public risk was assessed as the product of the change in core melt frequency and the estimated consequences in terms of person-rem of public exposure.
(b) Cost and cost savings in this table are summed over the remaining lifetime of all affected plants and discounted to present value at a 10% real discount rate as suggested by the Regulatory Analysis Guidelines, NUREG/BR-0058 (NRC 1984b).
(c) Relaxing time requirement for obtaining and analyzing all samples except reactor coolant system boron concentration, eliminating requirement for hydrogen analysis of containment atmosphere sample, eliminating requirement for radiological analysis of samples, eliminating analysis of reactor coolant sample for dissolved gases, and eliminating guidance to minimize radioiodine plateout on containment atmosphere sample lines.
(d) Not applicable. It is assumed that the benefit-risk comparison is not of interest when the benefits are marginal or the effect on public risk is marginal.
(e) Eliminating requirements for NRC reviews of turbine missile analysis and requirement for implementing turbine vendor rotor inspection requirements.
(f) Assumes the licensees will not change turbine inspection intervals even if the requirement to implement turbine vendor recommendations were eliminated (see Section 3.0). If the inspection interval were doubled, the study shows that industry savings may exceed $700 million. Furthermore, the study indicates that if inspections were eliminated altogether, an industry savings in excess of $1 billion might result.
(g) Eliminate requirement for recombiners in inerted BWR Mark I and II containments.
(h) Eliminate the impregnated charcoal filter banks in building ventilation systems with no other changes in system operation. A second alternative, "bottling-up" the buildings, results in no significant economic benefits to the utilities.

considered was the elimination of recombiners in the inerted boiling water reactor (BWR) Mark I and II containments. For impregnated charcoal filters in building ventilation systems, two regulatory modifications were considered: 1) removing the charcoal filters from the ventilation systems, essentially allowing the unfiltered release of radioiodines to the environment, and 2) isolating ventilation systems in the accident response mode, containing the majority of any releases within the plant buildings with leakage occurring only through natural building leakage paths.

- In three of the four cases (PASS, turbine rotors, and recombiners in inerted BWR containments), judiciously streamlining the existing regulatory requirements was estimated to have marginal effect on public health and safety. Marginal, in this context, means that the effect is relatively small, on the order of a few percent (or less) of overall plant risk.

- In the fourth case (impregnated charcoal filters), given the existing severe accident source term, the impact on public risk is not marginal. Merely removing the impregnated charcoal filters from the ventilation system of Three Mile Island-2 (TMI-2), for example, would have resulted in a population thyroid dose of 11,000 person-rem.

- The benefits of the possible modifications to the existing regulatory requirements vary. For example, the estimated benefits of modifications to PASS requirements are rather insignificant because installation of the system is complete at all plants and the potential operational cost savings are small. However, the benefits of modifying turbine rotor inspection may be significant if changes in vendor-recommended inspection frequencies result from eliminating the regulatory requirement. In the case of combustible gas control in inerted BWR containments, moderate benefits result from eliminating recombiners. Finally, use of any other media than impregnated charcoal filters to remove radioiodines results in substantially larger capital and operating expenses to the utilities. The regulatory modification to isolate the ventilation system to "bottle up" a potential release results in a small operating cost savings at the expense of inaccessibility and more widely spread plant contamination in the event of an accident.

- The quantitative analyses on which these conclusions are based are highly uncertain and should be interpreted cautiously. For this reason, the results in the table are reported in terms of ranges and orders of magnitude. All of the usual caveats and uncertainties surrounding the use of quantitative risk-cost-benefit analysis apply. Specific areas of uncertainty and possible areas of conservatism in the analyses are discussed in the main report.

- It should be stressed that analyses of this kind, and especially the quantitative portions of such analyses, are not the sole, or even the principal, basis for regulatory decisions. Rather, they are one of a number of inputs. As noted by Heaberlin (1983) in the Handbook for Value-Impact Assessment, the real strengths of quantitative analysis are the discipline it provides and the displays of key information and assumptions in understandable form so that they can be scrutinized and, if appropriate, challenged by interested parties.

STRUCTURE OF THIS DOCUMENT

The main report, which follows this summary, consists of five sections. Section 1 covers the background, objectives, and scope of this study. Section 2 presents the analysis of PASS. Section 3 covers the turbine rotor design reviews and inservice inspections. Section 4 covers combustible gas control in reactor containment buildings, and Section 5 presents the analysis of impregnated charcoal filters in certain plant ventilation systems.

OVERVIEWS OF THE FOUR ANALYSES

An overview is provided below of the analyses performed in each of the four regulation areas selected by the NRC staff for examination in this phase of the project.

Postaccident Sampling System

All of the reactors that are currently operating and soon to be operating are required to have the capability to sample the reactor coolant and containment atmosphere under accident conditions without incurring excessive individual radiation exposures. Licensees have complied with this requirement by installing Postaccident Sampling Systems (PASS) to meet the criteria established in NUREG-0737 (NRC 1980a) and its precursor correspondence (NRC 1979b and c). The purpose of PASS is to allow reactor operators to sample the reactor coolant and the containment atmosphere to obtain information on the condition of the core and the amount of radioactive material and the concentration of combustible gases present in the containment atmosphere.

Because PASS provides information to reactor operators following the initiation of an accident, some utilities contacted in an earlier phase of this project (Mullen 1986a) questioned the contribution of PASS to reduce public risk from reactor accidents.

Objectives of the PASS Analysis

Consistent with the overall objectives of the NRC program to review the effectiveness of current LWR regulatory requirements in limiting risk, the purpose of this analysis is to provide information on the risks, costs, and benefits that would result from elimination or modification of current requirements for PASS. Several options for modifying the requirements and guidance associated with PASS were considered:

- Relaxing the time requirement for collection and analysis of reactor coolant and containment atmosphere samples

- Eliminating the requirement to conduct radiological analysis of samples

- Eliminating the requirement for hydrogen analysis of the containment atmosphere sample

- Eliminating the requirement for dissolved gas analysis of the reactor coolant sample

- Eliminating the guidance to minimize plateout of samples in the sample lines.

<u>Alternatives to Modifying PASS Requirements</u>

There are a number of complex issues surrounding the requirements for PASS. These issues were highlighted during the TMI-2 accident when the amount of core damage, the combustible potential of the containment atmosphere, the potential accident source term, and the behavior of a severely damaged core were not well understood. In addition, emergency planning was not as well-developed as it is today. A comprehensive consideration of all of these factors may lead to further regulatory alternatives to PASS requirements and guidance. A much more extensive analysis, with a broader scope, would be needed to explore all of the options for modifying the regulatory requirements for PASS.

<u>Consequences of Modifying PASS Regulatory Requirements and Guidance</u>

PASS requirements and guidance were evaluated for three phases of an accident. These phases of an accident are:

- <u>Accident Management</u>. This phase of an accident is defined as the period of time immediately following initiation of a transient during which reactor operators need information on the status of the reactor in order to take actions to arrest and mitigate the consequences of the transient. This, in most accident scenarios, is a relatively short period of time lasting from several hours to about 2 days, during which the reactor is placed in a shutdown condition.

- <u>Emergency Response</u>. This phase of an accident often occurs concurrently with the accident management phase, however, the purpose of actions taken and decisions made is primarily the protection of the public through communicating plant status and recommending protective actions to the public from an emergency response team consisting of utility staff and federal, state, and local government officials. Information required by decisionmakers in this phase includes details about the potential for and estimated concentration of radioactive releases, the direction such a release is anticipated to travel, and the estimated doses that members of the public might receive as a result of exposure to a radioactive release.

- Accident Recovery. This phase of an accident refers to the actions required to deal with the damaged reactor and the contamination of plant and public property resulting from an accident. Information that is useful in this phase is the degree of core damage that may have occurred, the amount of radioactive material contaminating the containment and other plant buildings, and the potential damage done to the reactor components as a result of the accident. This information is not needed during the transient. At TMI-2, for example, it was many days before the reactor was brought to a cold shutdown condition when passive cooling mechanisms were sufficient to remove heat. It is at this point that information is needed regarding plant and reactor damage for recovery.

The risks and benefits associated with each of the alternatives to PASS requirements and guidance were evaluated in the context of the three phases of an accident described above. An evaluation of the cost impacts of the alternatives was also conducted. The results of the risk and benefit evaluations are summarized below; more detailed discussions are contained in Section 2 of the report.

Relaxation of the time requirement for collection and analysis of samples. The regulatory requirement regarding the time to obtain and analyze certain samples is three hours overall. Evaluations of emergency operating procedures, emergency response procedures and expected recovery actions revealed one area where prompt sample analysis results could aid reactor operators in the mitigation or arrest of an accident: the analysis of the reactor coolant sample for boron concentration is needed to verify boron concentration estimates made by the operators based on mixing ratios. The timing of other PASS sample results was found to have marginal or negligible effects on public risks due to reactor accidents. The cost evaluation indicated, however, that because all plants have already installed PASS, very little operational savings can be realized by relaxation of the time to obtain the samples.

Elimination of the requirement to conduct radiological analysis of samples. The purpose of the requirement to conduct radiological analysis of reactor coolant samples is to establish the extent of core damage. The purpose of the analysis of containment atmosphere samples is to determine the potential for hydrogen combustion inside containment and to determine the source term of a radioactive release if one were to occur. Reviews of the use of this information in the three phases of an accident indicated that the information may not be available in time for accident management and emergency response decisionmaking and therefore eliminating the requirements would have marginal effect on the risk of reactor accidents to the public. This result is due, in part, to the fact that other indicators of core damage, containment hydrogen concentration, and potential source term are more immediately available to reactor operators and emergency response decisionmakers, while PASS results may lag actual plant conditions by up to three hours. PASS information was found to be useful in planning plant recovery actions following an accident; however, this is primarily an economic benefit from reducing the costs and occupational exposure of recovery actions rather than one of protecting the public from the consequences of an accident. The cost evaluation of this alternative resulted in very small savings to utilities because PASS is already installed in all operating plants and the costs to operate and maintain the system are small.

Elimination of the requirement for hydrogen analysis of the containment atmosphere sample. The risk impact of eliminating this requirement was found to be negligible because most plants have redundant safety-grade containment hydrogen monitors that serve as the primary source of information concerning hydrogen concentrations in containment. The safety-grade hydrogen monitors provide real time indication of hydrogen levels in containment and these results are available much sooner than PASS sample results. However, the cost evaluation of this alternative indicated minimal savings to the utilities since most utilities rely on PASS samples as a backup to the safety-grade containment hydrogen monitors.

Elimination of the requirement for dissolved gas analysis of the reactor coolant sample. This analysis provides information that can be used to determine the corrosive potential of the coolant and to infer the potential for in-vessel gas bubbles. The risk evaluation concluded that information regarding the corrosive potential of the core is used for making decisions during the recovery phase of the accident and has no impact on the protection of the public health and safety. There is a risk associated with the formation of noncondensible bubbles in the reactor vessel because of their potential for uncovering the core and decreasing the heat removal capability of the engineered safety feature (ESF) systems. However, other NRC regulations establish requirements for 1) a reactor vessel level indication system to detect the presence of a bubble and core uncovery, and 2) a head vent system to remove noncondensible gases from the high points of the reactor coolant system. These systems adequately remove the potential for noncondensible gases to interfere with core cooling. The PASS sample requirement is redundant with these requirements.

The cost evaluation indicated that some savings in operating costs and reduced occupational radiation exposure could result from the elimination of the requirement to obtain and conduct dissolved gas sample analysis. Dissolved gases require pressurized samples to be taken, which significantly complicates the design and operation of PASS. One utility estimated that the number of valves in PASS could be reduced from 40 or 50 to 5 or 6 if pressurized samples were not required. Operating, maintenance, and occupational exposure savings result from reduced maintenance on the valves that are not needed (even though they might not be removed from the system) and the reduction in operator time necessary to align and operate the system to take pressurized samples.

Elimination of the guidance to minimize plateout of samples in the sample lines. Plateout in containment sample lines is a problem because radioiodine tends to precipitate onto the cooler surface of the sample line tubing or pipe, thereby artificially reducing the measured radioiodine concentration. Sample plateout is reduced by heat tracing the sample lines. In order to eliminate the cost of installing and maintaining the heat tracing system on the sample lines, complete elimination of the containment atmosphere sample was considered. The risk evaluation of this alternative indicated that the effect of eliminating the containment atmosphere sample and the guidance to minimize sample plateout would affect public risk only marginally. The potential 3-hour delay in obtaining containment atmosphere sample results causes reactor operators and emergency response decisionmakers to take action and make decisions based on other indications of the potential source term that might be released if containment were to fail. However, the operational savings associated with eliminating the heat tracing of the containment sample lines are limited to the power consumed by the heat tracing system and the maintenance of that system. These costs are estimated to be insignificant and were not quantified.

Conclusions--Postaccident Sampling System

The risk impact of implementing these five potential regulatory alternatives is marginal. Likewise, the benefits in terms of cost savings to utilities with operating plants is small. However, the benefits of implementing these potential alternatives increases substantially for new plants that have not designed, procured or installed PASS. The cost evaluation of the five alternatives considered in this study indicated that the major costs associated with PASS are the capital and installation costs, which can be as large as $2 million per system.

Turbine Missile Design Reviews and Turbine Inspection Requirements

General Design Criterion 4, "Environmental and Missile Design Bases," of 10 CFR 50, Appendix A, establishes the requirements to protect the reactor and its safety systems from damage due to missiles. With regard to missiles that may be generated by the low pressure turbine rotors in the main turbines of nuclear plants, NRC requires licensees to submit a safety analysis to demonstrate that the probability of plant damage due to turbine missiles meets specified criteria. Additionally, licensees are required to periodically inspect turbine rotors for flaws. The frequency of this inspection is usually the frequency recommended by the turbine vendor. Recent improvements in turbine rotor design and materials have reduced turbine failures that could generate missiles.

Procedures and guidance for evaluating the public health and safety effects of turbine failures are contained in Regulatory Guide 1.115 (NRC 1977a) and several sections of the Standard Review Plan (NRC 1977b).

Objective of the Turbine Missile Requirements Analysis

Consistent with the overall objectives of the NRC program to review the effectiveness of current LWR regulatory requirements in limiting risk, this analysis provided information on the risks, costs, and benefits that could result if current procedures for turbine missile reviews and turbine rotor inspections were eliminated. Specifically, the analysis divides this overall objective into three alternatives:

- Eliminating the turbine missile design reviews and inservice rotor inspection requirements

- Eliminating the turbine missile design reviews and relaxing the turbine inspection frequency

- Eliminating the turbine missile design reviews with no change in the inservice rotor inspection frequency.

These three alternatives represent a spectrum of inspection relaxations. The option that involves no change in the inspection interval is a potential result of the relaxation of the NRC review requirements without a relaxation of the requirement to implement turbine vendor inspection recommendations. The other extreme, complete elimination of the inspection requirements allowing inspection intervals to be set by the utilities, results from completely eliminating the NRC requirements for design reviews and the NRC requirement to implement the turbine vendor recommendations. In this case, it is assumed that utilities would choose inspection intervals based on the financial risk of a turbine failure versus the inspection costs.

Alternatives to Modifying Turbine Missile Requirements

There are several complex technical issues associated with the requirements for protection against turbine missiles. These involve the probabilities associated with the generation of a missile, the penetration of the turbine casing, the trajectory of the missile, the striking of a safety-related component, the damage of the struck component, and the potential for core melt given the damaged component. These issues are complicated by the proprietary nature of the turbine design information. A thorough review of this proprietary information may reveal additional alternatives to those considered in this study.

Consequences of Modifying Turbine Missile Requirements

To assess the potential consequences of the three regulatory alternatives examined in this study, the effects of eliminating the requirement for NRC staff review of turbine missile licensee submittals was evaluated and risk analyses conducted to quantify the changes in public risk associated with changing the turbine rotor inspection intervals.

The result of the qualitative analysis of the risks associated with the potential elimination of NRC reviews of licensee turbine missile evaluations indicated that the level of risk would be unaffected by this alternative. Additionally, the cost savings that would result from the elimination of this review are too small to be quantified.

The quantitative risk analysis of increasing the turbine rotor inspection interval from a present industry average of 4.5 years to the alternative intervals of 9 years and 30 years was accomplished by examining accident sequences in representative PWRs and BWRs that involve power conversion system transients and assumed concurrent failures of essential reactor equipment. Data on the probability of turbine missile generation were obtained from historical failures of turbines in both nuclear and fossil applications and from estimates based on fracture mechanics models developed by the turbine vendors.

The results of the risk analyses indicate that increasing the turbine rotor inspection interval from a nominal 4.5-year average to a 9-year or a 30-year average would have a marginal impact on public risk. The changes in core melt frequency and public risk as a result of lengthening the inspection intervals are shown in Table S.2 for both PWRs and BWRs.

The cost savings associated with lengthened turbine inspection intervals were also evaluated. The analyses indicated that significant industry savings could be realized if the inspection interval were longer. Changes in the turbine inspection intervals, if implemented, would result in an overall industry savings of $715 million for the 9-year inspection interval or $1.2 billion for the 30-year inspection interval.

Finally, the financial risk to the utility industry was calculated for the extended inspection intervals considered in this study. Financial risk represents the increased probability that a turbine missile will cause damage to the plant and the corresponding repair costs will be incurred. The financial risk is the product of the frequency of missile generation that causes plant damage and the cost to repair the damage. The calculations indicate that the financial risk for the utility industry as a whole for the 9-year interval is $250,000/year. For the 30-year interval, the financial risks to the utility industry are $1.27 million/year. The financial risk of turbine failure due to longer inspection intervals is small compared to the annual savings.

TABLE S.2. Results of Turbine Missile Risk Analyses

Plant Type/Inspection Interval	Increase in Core Melt Frequency (per plant-year)	Increase in Public Risk (person-rem/plant-year)
PWR/9-Year Interval	6.0E-09	1.6E-02
PWR/30-Year Interval	3.0E-08	8.0E-02
BWR/9-Year Interval	1.2E-08	8.8E-02
BWR/30-Year Interval	6.2E-08	4.4E-01

NOTE: Increases in core melt frequency and public risk are relative to core melt frequency and public risk calculated for a 4.5-year inspection interval.

Conclusions--Turbine Missile Requirements

Based on the information presented in this report, it appears that NRC reviews of turbine missile safety analyses could be eliminated without compromising public health and safety. The risk of implementing this alternative is marginal and so are the cost savings to the utilities. The savings were too small to be quantified.

This study also indicates that the turbine inspection interval could be extended significantly with marginal impact on public risk. In this case, the potential benefits to the utilities are significant. Benefit-cost ratios for extending the turbine inspection intervals range from $100 million to $1 billion saved per person-rem of dose increase. These ratios can be compared to the guideline of $1000 per person-rem that has been used in certain other contexts (e.g., 10 CFR 50, Appendix I). These quantitative calculations are provided for perspective. It should be stressed that quantitative analyses of this nature are not the sole or even the principal basis for regulatory decisions. Moreover, the numerical values are highly uncertain, and should be interpreted cautiously.

Combustible Gas Control Requirements

The control of combustible gases in reactor containment buildings following an accident in which quantities of hydrogen gas may be generated through reactions between the reactor coolant and the zircaloy fuel cladding has been recognized as a key element in preserving the integrity of the reactor containment. Regulations (10 CFR 50.44) establish requirements for controlling combustible gases in the various types of containments. In particular, recombiner capability (either internal or external to the containment) to control the relative concentrations of hydrogen and oxygen are specified. For all types of containments, the criteria for sizing the recombiners are based on a metal-water reaction involving approximately 5% of the active fuel cladding. These recombiner sizing requirements are based on the amount of hydrogen that would be generated in a design basis accident and therefore the capacity of the recombiners is too small to process the potentially large amounts of hydrogen that might be generated in an accident involving larger portions of the core. Because of the increased vulnerability of Mark I and II BWR containments resulting from their smaller volumes, additional requirements for inerting the containments were imposed to protect them following a severe accident. Since the overall public risk of reactor accidents is dominated by the more severe core damage accidents that usually involve containment failure, several utilities indicated (Mullen 1986a) that recombiners are not effective in reducing this dominant contributor to public risk and therefore, the costs of installing, maintaining and operating the recombiners are burdensome.

Objectives of the Combustible Gas Control Study

This analysis provided information on the risks, costs, and benefits that would result from elimination or modification of current requirements for recombiners in inerted BWR Mark I and II containments. The option considered for recombiners is to:

- eliminate the requirement for recombiner capability and to disable or remove the recombiner capacity from the plants.

The study also considered an alternative related to the time that BWR Mark I and II containments are inerted during initial startup testing. Current requirements specify that these containments must be inerted during operation beginning six months after initial criticality. Many owners of these plants have had to apply to NRC for an exemption to this requirement because startup testing could not be completed within six months of initial criticality. The regulatory option considered in this area was:

- delay the time that BWR Mark I and II containments must be inerted until the initial startup tests are completed.

In the course of the study, the recombiner options were focused on the issue of recombiners in inerted containments. The NRC is currently sponsoring research related to the control of hydrogen following severe accidents in large, dry PWR containments.

Alternatives to Modifying Combustible Gas Control Requirements

There are a number of complex technical issues surrounding the requirements for combustible gas control in containments. Each of these technical issues gives rise to a number of additional regulatory alternatives. A comprehensive examination of these issues has been under way at NRC for some time and is included in the Severe Accident Research program.

Consequences of Modifying Combustible Gas Control Requirements

To assess the potential safety consequences of eliminating the requirement for recombiner capability in inerted BWR Mark I and II containments, a qualitative analysis was performed to identify the avenues of risk that might be affected by eliminating the recombiner capability in these containments. This analysis relied in part on the qualitative arguments presented by Northeast Utilities (NU) in their Combustible Gas Control Evaluation Report to NRC on hydrogen control issues in their Millstone Unit No. 1, a BWR with a Mark I containment (NU 1982) and the probabilistic risk assessment for Philadelphia Electric Company's Limerick Units 1 and 2 (Philadelphia Electric Company 1982). The Limerick units are BWR/4s with Mark II containments.

Based on the review of the Limerick PRA and the NU report, the risks due to hydrogen combustion/detonation in inerted BWR containments are primarily due to operating the reactor with the containment only partially inerted. This occurs during routine plant startup and prior to routine plant shutdown, or by failure of the containment inerting system. In these circumstances, elimination of the recombiner capability requirement would have a negligible effect on the level of public risk because of the relatively small capacity of the recombiners and the large amounts of hydrogen that could be generated in a degraded core accident.

The cost evaluation indicated that for existing plants, the implementation of the option to eliminate recombiners in BWR Mark I and II containments would save the industry $16.4 million and the NRC $1.8 million over the remaining lives of the existing plants.

The risk evaluation of the option to delay initial inerting of BWR Mark I and II containments during startup testing indicated that the impact on public risk would be marginal. The cost evaluation showed that no appreciable benefit could be realized for the plants remaining to be started. The benefits of delaying initial containment inerting lie in saving the costs of operating and maintaining the units, and the costs of preparing and negotiating an exemption request with the NRC staff.

Conclusions--Combustible Gas Control Requirements

Based on the risk evaluation, it appears that the risks associated with the option of eliminating the requirement for recombiner capability in inerted BWR Mark I and II containments are marginal. The cost analysis indicated that the utilities and the NRC could save costs associated with the maintenance, operation and inspection of recombiners if the requirement were eliminated.

The analysis also concluded that the cost savings associated with delaying the time to initially inert these containments until the completion of startup testing are negligible for the remaining plants to be started. The risks of such delays are marginal.

Impregnated Charcoal Filter Analysis

NRC regulations currently require operating nuclear power plants to have filtered ventilation systems serving the structures that house various parts of the plant. In general, these filtered ventilation systems serve to restrict uncontrolled release of radioactive gases and particulates from process areas, and to protect plant operations personnel, who must remain on station during an emergency, from the effects of radioactive releases arising from that emergency.

The filtered ventilation systems now in use employ both high efficiency particulate (HEPA) filters and activated, impregnated charcoal filter media to remove radioactive contaminants. The requirement for use of an absorbing medium such as charcoal is based on the existence of large inventories of radioiodine isotopes in reactor cores during and after power operation. The radioiodines, if released to the atmosphere and inhaled, would concentrate in the thyroid gland, possibly causing radiation exposure to that organ in excess of allowable limits.

Recent work on reactor accident source term definition had raised doubts about the validity of past assumptions regarding the chemical and physical nature of the fission product mixture that would be released during a reactor accident. This, in turn, led to reassessment of the value of forced, filtered exhaust ventilation in reducing accident consequences.

Objectives of the Impregnated Charcoal Filter Study

The purpose of this analysis is to provide information on the risks, costs, and benefits that would result from elimination or modification of current requirements for impregnated charcoal filters in forced, filtered ventilation systems. For the purposes of this study, the regulatory options considered are:

- eliminating the requirement for iodine removal from forced, filtered building ventilation systems and isolating the ventilation system in the event of an accident, effectively "confining" the release inside plant buildings

- eliminating the requirement for hygroscopic impregnation media in the charcoal beds, thereby eliminating the need for installing heaters in the filter beds or incoming air stream to reduce filter degradation.

Alternatives to Modifying Impregnated Charcoal Filter Requirements

Alternatives to removing the activated, impregnated charcoal filters from plant ventilation systems include other iodine removal methods, such as scrubbers. While such alternatives are feasible, the study indicates that such options are far more costly to install, operate and maintain.

Consequences of Modifying Impregnated Charcoal Filter Requirements

The proposed alternative involving the confinement of postulated radioactive releases has several complicated consequences that preclude quantitative analysis. The behavior of the release, i.e., particulates and gases, as they pass through various rooms in plant buildings, the agglomeration of aerosols and plateout on plant surfaces, and the leak rate of the building confining the postulated release are examples of factors that could not be quantified in this study.

A qualitative investigation of the possible consequences of this alternative revealed several undesirable results. Engineered safety feature systems (ESF) generally rely on building ventilation systems to maintain the environment within limits that permit equipment operability in the event of an accident. If a release of hot primary system water and steam were confined in a vault containing an ESF pump, the pump might fail without the ventilation system to remove moisture and heat. A release confined in a plant building could concentrate airborne radioactive contamination to a level that precludes personnel entry to operate equipment, fight fires, or repair equipment. Based on the results of the qualitative analysis, the confinement alternative was found undesirable from the standpoint of plant operation during an accident or transient.

An evaluation of the second alternative, elimination of the hygroscopic adsorption media from the charcoal beds, resulted in no change in risk to the public because the regulation currently permits the use of any media that meets the specified filter efficiency. However, the costs of other methods of iodine removal are considerably more expensive than the total cost of the impregnated charcoal beds using hygroscopic media.

The effects of the TMI-2 accident were evaluated to quantify the risk averted by the use of impregnated charcoal filters in the ventilation systems of a PWR auxiliary or fuel handling building. This evaluation compared the calculated TMI-2 release of radioiodine with the estimated release of radioiodine if the impregnated charcoal filter were removed from the filter bank. The results indicated that the releases of radioiodine from TMI-2 without impregnated charcoal filter banks would have increased the population thyroid dose from an estimated 11 person-rem with charcoal filters to about 11,000 person-rem for filters without impregnated charcoal media.

The cost analysis of these alternatives indicated that impregnated charcoal filters are the most cost-effective method of removing radioiodines from a ventilation system stream.

Conclusions--Impregnated Charcoal Filter Requirements

Based on the results of this study and the current severe accident source term, the risk impact of removing impregnated charcoal filters from forced, filtered ventilation systems at nuclear power plants is not marginal. Depending on the option, either the population thyroid dose increases significantly, or the operation of the plant is hindered tremendously. Furthermore, the use of charcoal beds impregnated with hygroscopic media is cost-effective compared with alternatives for removal of radioiodines from postulated releases.

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1.0 INTRODUCTION

In 1984, the Nuclear Regulatory Commission (NRC) initiated a program to review existing light water reactor (LWR) regulatory requirements to see if some could be relaxed or eliminated to reduce regulatory burdens without compromising public health and safety. This volume presents the analyses performed by Pacific Northwest Laboratory (PNL) in the second year of this program. In a Federal Register notice (NRC 1984a) announcing the program, the NRC stated that the objective of the program is:

> "to identify current regulatory requirements which, if deleted or appropriately modified, would improve the efficiency or effectiveness of the NRC regulatory program without adversely affecting safety."

This report presents information on the costs and benefits of potential modifications to current requirements in four areas:

- Postaccident Sampling System (PASS)
- Turbine Missiles
- Recombiners in reactor containment buildings that are part of the Combustible Gas Control System
- Impregnated charcoal filters in building ventilation systems.

These areas of regulation were selected by NRC staff for examination in 1986. The objective of these analyses is to provide technical information that the NRC staff may consider in formulating recommendations on whether these areas of regulation should be modified, and if so, in what way.

1.1 BACKGROUND

On October 3, 1984, the NRC published a notice in the Federal Register (NRC 1984a) announcing a new program to review the effectiveness of existing LWR regulatory requirements in limiting risk. The program was initiated in response to guidance received in NRC's Policy and Planning Guidance (PPG) for 1984, NUREG-0885 (NRC 1984b), and in response to specific programmatic direction from the Executive Director for Operations. In the section of the PPG entitled "Improving Regulation of the Nuclear Industry," the NRC stated:

> "Existing regulatory requirements that have a marginal importance to safety should be eliminated."

Subsequent editions of the PPG have reiterated and expanded this guidance. The PPG for 1986 (NRC 1986), for example, states:

"Existing regulatory requirements should be reviewed to see if some could be eliminated without compromising safety, safeguards or environmental protection."

The 1986 PPG also states in the "Policy" section that:

"NRC regulations should be changed when research shows them to be either too stringent or not stringent enough to adequately protect the public health and safety."

As part of the program guidance developed in support of the Commission's 1984 PPG, the Executive Director for Operations called for a three-pronged effort to systematically review existing regulations. This effort was to address the following distinct aspects of the existing regulatory structure:

1. Existing operating reactor licensing actions

2. Technical specifications

3. Rules and the associated regulatory guidance, with the initial emphasis on 10 CFR 50.

Programs have been initiated in each of these three areas. The work discussed in this report is part of the program formulated to address the third area, i.e., the regulatory requirements of 10 CFR 50.

As part of the overall program, PNL has provided technical information and analyses to support the NRC staff in their work. In 1986, PNL performed detailed analyses of four regulatory requirements selected by NRC staff; these analysis included cost-benefit assessments of the consequences of changing or eliminating the requirements. As the third volume in a series, this document presents the results of those analyses. Volumes 1 and 2 (Mullen 1986a and b) present the results of the 1985 evaluations.

1.2 TECHNICAL APPROACH

This report describes PNL's analysis of four areas of regulation selected by the NRC staff. The analysis follows the guidance and procedures contained in the Regulatory Analysis Guidelines, NUREG/BR-0058 (NRC 1984c), and the Handbook for Value-Impact Assessment, NUREG/CR-3568 (Heaberlin 1983). These two NUREG reports describe a set of systematic procedures accepted by the NRC for providing information to support regulatory decisions. The Regulatory Analysis Guidelines give the basic structure and content of the regulatory analyses currently required by NRC management for a broad range of regulatory decisions. The Handbook contains more detailed descriptions of the methods and data that can be useful in evaluating the values and impacts of potential alternatives.

1.3 CONTENTS OF THIS REPORT

Following this introductory chapter, Chapter 2 presents an analysis of current Postaccident Sampling System (PASS) regulatory requirements. The primary focus of the evaluation is to assess the consequences of eliminating selected PASS requirements; the assessment considers the importance of information provided by PASS in mitigating the progression of an accident, to make emergency response decisions, and in planning and conducting recovery actions following an accident. Chapter 3 addresses the regulatory requirements for low pressure turbine rotor design reviews and inservice inspections conducted to prevent the generation of turbine missiles and evaluates the option of completely eliminating NRC reviews and inservice inspection requirements. Chapter 4 presents an evaluation of selected aspects of the current NRC requirements for combustible gas control systems in the various types of reactor containment buildings. The main focus of this evaluation is the current requirement to provide a recombiner capability for Boiling Water Reactors (BWRs) with inerted Mark I and II containments. Finally, Chapter 5 focuses on the requirement for certain building ventilation systems to be provided with impregnated charcoal filters, and addresses alternative strategies for processing gases through filtered ventilation systems.

2.0 RISK AND COST IMPACTS OF POSTACCIDENT SAMPLING SYSTEMS

The NRC requires nuclear reactors to have the capability of obtaining and analyzing postaccident samples of the reactor coolant system and the containment atmosphere for certain radionuclides, chemicals, and gases. All plants have installed a Postaccident Sampling System (PASS) to meet these requirements.[a] This system allows operators to take samples from the reactor coolant system and the containment atmosphere during and after severe core damage accidents without incurring excessive radiation doses. The results obtained from analysis of the reactor coolant and the containment atmosphere samples provide information on the condition of the core, the contaminants present in the reactor coolant system, the containment combustible gas concentration, and the radionuclides that could possibly be released to the environment, if a release were to occur. Because of the high cost of installing these systems, the impact on plant operation costs, and the belief of some utilities that the systems provide limited safety benefits, several utilities identified the regulatory requirement for PASS as burdensome and of marginal benefit to limiting public risk (Mullen 1986a).

The PASS requirements were developed because during the TMI-2 accident, high radiation levels in the area of the primary system sample sink prevented plant personnel from obtaining timely reactor coolant and containment atmosphere samples. These samples were needed to determine whether significant core damage had occurred. The samples were finally collected several days following the accident after taking extensive radiation protection precautions. Even after the samples were obtained they could not be counted on plant counting equipment due to high background radiation levels. The samples had to be packaged and flown to a Department of Energy laboratory for analysis.

Many different types of postaccident sampling systems have been installed in nuclear plants in response to the PASS requirement. A 1984 survey of NRC Region III plants revealed six different manufacturers and several in-house designed systems being used at the plants. The manufacturers included Sentry, NUS, Bechtel, General Electric, and Science Applications International Corporation (SAIC). Most systems rely primarily on obtaining grab samples of reactor coolant and containment atmosphere. Several plants have more sophisticated in-line systems which provide real-time isotopic results and other analytical results. One in-line PASS system reviewed has real-time capabilities for the chemical analyses (i.e., boron, hydrogen, oxygen) and a grab sample capability for radiological analysis of the reactor coolant and containment atmosphere samples.

A block diagram of a typical PASS is shown in Figure 2.1. Reactor coolant and containment atmosphere samples are drawn from containment via sample lines to a PASS sample panel. The sampling points in containment were identified by each plant as representative locations during an accident. Plants should have

(a) In this study, regulatory guidance related to PASS (i.e., NUREG-0737, Item II.B.3) will be referred to as requirements (NRC 1980a).

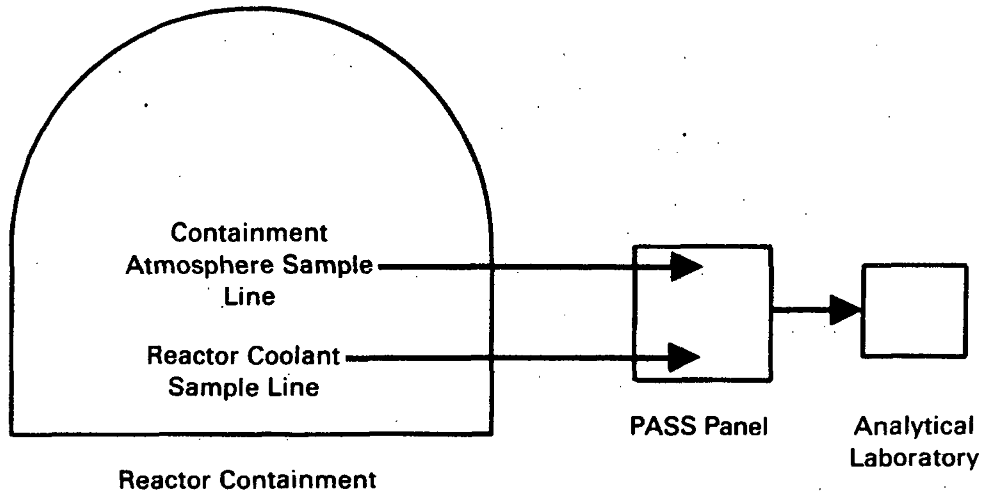

FIGURE 2.1. Block Diagram of a Typical Postaccident Sampling System

the capability to collect reactor coolant samples from the containment sump, ECCS pump room sumps, and other auxiliary building sumps. At the sample panel, trained personnel align the many system valves and perform the necessary line flushes to enable the collection of reactor coolant and containment atmosphere samples. At some plants, the valves are aligned at a remote panel away from high exposure rates. For a grab-sample system, the samples collected at the sample panel are transported to the plant's analytical laboratory for radiological/chemical analysis. An in-line system has the analytical equipment incorporated into the PASS sample panel, making transport to the laboratory unnecessary.

Sample results obtained via a grab sampling system would generally not be available to decisionmakers for two to three hours after the decision to take a PASS sample is made. This time can be shortened to less than two hours if decision-makers anticipate the need for a sample and start system activation (e.g., flush the system) prior to the decision to take the sample. In-line postaccident sampling systems have the advantage of providing sample results to the decisionmakers more quickly; sample results can be available in less than one hour after the sampling decision is made. As with a grab sample system, the time required to get sample results from an in-line system can be reduced if decisionmakers anticipate the need for sampling and begin system activation.

2.1 CURRENT REGULATORY REQUIREMENTS

The first NRC guidance on postaccident sampling capability was contained in NUREG-0578, Section 2.1.8a (NRC 1979a), which stated that: "Chemical and radiological analysis of reactor coolant liquid and gas samples can provide substantial information regarding core damage and coolant characteristics. Analysis of containment atmosphere samples can determine if there is any prospect of a hydrogen reaction in containment, as well as provide core damage information." NUREG-0578 required utilities to perform a design and operational review to determine whether:

- reactor coolant and containment atmosphere samples could be obtained under accident conditions without incurring individual radiation exposures in excess of 3 rem to the whole body or 18.75 rem to the extremities

- samples could be analyzed promptly to quantify certain radioisotopes indicative of the degree of core damage.

If the exposure limits or the analytical capabilities could not be met, NUREG-0578 required the utilities to upgrade their systems to meet these criteria.

Following NUREG-0578, additional clarification on postaccident sampling was provided by the NRC in letters dated September 13, 1979 (NRC 1979b) and October 30, 1979 (NRC 1979c), in NUREG-0660 (NRC 1980b), and in Regulatory Guide 1.97, Rev. 2 (NRC 1980d). Regulatory Guide 1.97 references Criterion 64 of Appendix A to 10 CFR Part 50 ("Monitoring Radioactivity Releases") which includes a requirement that means be provided to monitor the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents (CFR 1986). This is the regulatory basis for PASS.

The current NRC requirements for postaccident sampling capability are found in NUREG-0737, Item II.B.3 (NRC 1980a) and Regulatory Guide 1.97 Rev. 3 (NRC 1983). The NRC issued one clarification letter to NUREG-0737 requirements in June 1982 (NRC 1982). A listing of the 11 PASS requirements in NUREG-0737 and the purpose of each requirement is presented in Table 2.1.

2.2 ALTERNATIVES TO CURRENT PASS REQUIREMENTS

Two approaches to the regulatory requirements for PASS were considered. First, all PASS requirements could be completely eliminated. However, early in the project this was rejected because of the importance of PASS in determining boron levels in the reactor coolant as well as its importance in estimating the amount of core damage.

Second, the modification of individual requirements contained within the PASS requirements was considered in detail. Proposed modifications to individual requirements that were given detailed consideration are:

- relaxation of the 3-hr time requirement for all samples except boron

- elimination of the radiological analysis of samples

TABLE 2.1. Purpose of Postaccident Sampling System Requirements

NUREG-0737 Requirement	Purpose of Requirement
1. Promptly obtain and analyze reactor coolant samples and containment atmosphere samples within 3 hr from the time a decision is made to take a sample	Assure the postaccident sampling system can collect and analyze samples within 3 hrs
2. Radiological and chemical analysis	
a. Radiological analysis	Quantify certain radionuclides in the reactor coolant and containment atmosphere samples that are indicators of the degree of core damage
b. Hydrogen in containment atmosphere	Provide information on the potential for a hydrogen reaction in containment
c. Dissolved gases (e.g., H_2,O_2) and chlorides	Provide information on the degree of core damage and corrosion potential of the reactor coolant
d. In-line monitoring capabilities	Alternatively, provides the option of installing an in-line system to perform the analyses in 2a, 2b, and 2c
3. PASS sampling shall not require an auxiliary system to be placed in operation	Assure PASS operation does not require an isolated auxiliary system (e.g., letdown system) to be placed in operation and that PASS valves not accessible after an accident are environmentally qualified for the conditions in which they must operate
4. Pressurized reactor coolant samples are not required if dissolved gases can be quantified from unpressurized samples	Provides option of not collecting a pressurized reactor coolant sample if dissolved gases can be quantified from an unpressurized sample. Measurements for either total dissolved gases or H_2 gas is adequate; O_2 analysis is optional
5. Chloride Analysis Time of 24 hr or 4 days	Sets alternative time requirements for chloride analyses. Samples must be analyzed within 24 hr if the plant's coolant water is seawater or brackish water and if there is only a single barrier between it and the primary coolant systems. The rector coolant samples must be analyzed within 4 days for all other plants

TABLE 2.1. (contd)

NUREG-0737 Requirement	Purpose of Requirement
6. Radiation Exposure Limits	Limits occupational radiation exposures to meet GDC 19 (5 rem whole body, 75 rem extremities)
7. Boron analysis required for PWRs and BWRs	Provide information to reactor operators concerning criticality of the core and potential for boron plugging of ESF system lines
8. Backup grab sampling capability if in-line monitoring system used.	Assure a backup capability exists for obtaining and analyzing samples should the optional in-line system be inoperable
9. Radiological and chemical sample analysis capability	
a. Sensitivity of radiological analysis	Assure that licensees have the capability to identify and quantify radionuclides at levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4
b. Restrict background levels in analytical area	Assure background radiation levels in the radiological and chemical analysis facility are restricted (analytical error should be within a factor of two)
10. Sample analysis accuracy	Provides accuracy levels for radiological and chemical analyses described in 2a, 2b, 2c, and 7
11. Provisions for:	
a. Heat tracing of sample lines to reduce radioiodine plateout	Assure that licensees have provisions for purging sample lines, for reducing plateout in sample lines, and for minimizing sample loss or distortion, for preventing blockage of sample lines by loose materials, or appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample lines
b. Ventilation exhaust filtered	Assure the ventilation exhaust from the system is filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters to limit the offsite dose

- elimination of the hydrogen analysis of the containment atmosphere sample

- elimination of the dissolved gas samples

- elimination of the guidance to minimize plateout in sample lines.

The proposed alternatives were developed after reviewing how PASS information would be used during an emergency. Appendix A presents a discussion on how PASS information would be used in the three phases of an accident described below:

- Accident Management. This phase of an accident is defined as the period of time immediately following the initiation of a transient, during which reactor operators need information on the status of the reactor in order to take actions to arrest and mitigate the consequences of the transient. In most accident scenarios, this phase is relatively short, lasting from several hours to about two days, during which time the reactor is placed in a shutdown condition.

- Emergency Response. This phase of an accident often occurs concurrently with the accident management phase; however, the purpose of actions taken and decisions made is primarily the protection of the public through communicating plant status and recommending protective actions to the public from an emergency response team consisting of utility staff, and state and local government officials. Information required by decisionmakers in this phase involves details about the potential for and estimated concentration of radioactive releases, the anticipated direction of such a release, and the estimated doses that members of the public might receive as a result of exposure to the radioactive release.

- Accident Recovery. This phase refers to the actions required to deal with the damaged reactor and with the contamination of plant resulting from an accident. Information that is useful in this phase includes the degree of core damage that may have occurred, the amount of radioactive material contaminating the containment and other plant buildings, and the potential damage to the reactor components as a result of the accident. This information is typically not needed during the progression of the transient. Using the example of the TMI-2 accident, it was many days before the reactor was brought to a cold shutdown condition when passive cooling mechanisms were sufficient to remove heat. It is only after the plant has been stabilized that recovery planning begins.

Table 2.2 presents a summary on the information from each of the eleven NUREG-0737 requirements and guidance that is used by decisionmakers during the three accident phases. Also presented is information on whether the requirement or guidance should be retained or if there is a reasonable basis for relaxing the requirement. Justifications for the proposed alternatives are presented below:

TABLE 2.2. Use of Postaccident Sampling System Information During and After an Emergency

NUREG-0737 Requirements[a]	Phase 1: Accident Management	Phase 2: Emergency Response Decisionmaking	Phase 3: Recovery	Possible Modification(s) to Requirement
1. Promptly obtain and analyze reactor coolant samples and containment atmosphere sampling within 3 hrs from the time a decision is made to take a sample	No	No	Yes; however prompt sample not required	Redefine time requirement to be consistent with recovery phase requirements
2. Radiological and chemical analysis				
a. Radiological analysis	No	No	Yes	Maintain requirements for recovery phase; PASS provides information regarding core damage assessment
b. Hydrogen in containment atmosphere	No	No	Yes	Eliminate the requirements; since all licensees should have a safety-related containment hydrogen monitor providing a real-time indication of containment hydrogen levels to meet NUREG-0737 item II.F.1.6
c. Dissolved gases (H_2,O_2) and chloride; chloride analysis time 24 hr or 4 days[b]	No	No	Yes	Maintain requirement for recovery phase
d. In-line monitoring capabilities	Optional	Optional	Optional	Maintain requirement
3. PASS sampling shall not require an auxiliary system to be placed in operation	(c)	(c)	(c)	Maintain requirement for system operation
4. Pressurized reactor coolant samples are not required if dissolved gases can be quantified from unpressurized samples	(c)	(c)	(c)	Maintain requirement for system operation
5. Chloride Analysis Time (see 2c.)	No	No	Yes	Combine all chloride analysis requirements; see item 2c.

TABLE 2.2. (contd)

NUREG-0737 Requirements[a]	Phase 1: Accident Management	Phase 2: Emergency Response Decisionmaking	Phase 3: Recovery	Possible Modification(s) to Requirement
6. Radiation Exposure Limits	(c)	(c)	(c)	Maintain requirement for system operation
7. Boron analysis required for PWRs and BWRs	Yes	No	Yes	Maintain requirement and establish time frame for sample results
8. Backup grab sampling capabilities for in-line monitoring system	(c)	(c)	(c)	Maintain requirement
9. Radiological and chemical sample analysis capability				
a. Sensitivity of radiological analysis	(c)	(c)	(c)	Maintain requirement for core damage assessment
b. Restrict background levels in analytical area	(c)	(c)	(c)	Maintain requirement for system operation
10. Sample analysis accuracy	(c)	(c)	(c)	Maintain requirement for system operation
11. Provisions for:				
a. Heat tracing of sample lines to reduce radioiodine plateout	No	No	Yes	Eliminate if radioiodine data results are not being used for core damage estimate
b. Ventilation exhaust filtered	(c)	(c)	(c)	Maintain requirement for system operation

(a) Criteria were taken from NUREG-0737, Item II.B.3 entitled "Postaccident Sampling Capability" (NRC 1980a).
(b) Chemical analysis for boron is discussed under Requirement 7.
(c) Maintain requirement for safe and reliable PASS operations.

- Relaxation of Time Requirement. Criterion 1 of NUREG-0737, Item II.B.3 (NRC 1980a) requires that reactor coolant and containment atmosphere samples be collected and analyzed within three hours. Shortening the three-hour time requirement so PASS information could be more readily considered in Accident Management and Emergency Response Decisionmaking is not technically feasible unless all utilities are required to have in-line systems. Boron analysis of the reactor coolant sample is required in the initial accident management phase; therefore, increasing the time to collect a reactor coolant sample would not be feasible. However, during the recovery phase, when timeliness of results is not as critical, the time allotted for collecting and analyzing samples could be increased (e.g., to six hours).

- Elimination of the Radiological Analysis of Samples. Requirements for radiological analysis of postaccident samples are found in Item 2a of NUREG-0737, II.B.3. These include the ability to monitor short and long-lived volatile and nonvolatile radionuclides and to estimate the extent of core damage based on radionuclide concentrations. Reviews of the use of this information in the three phases of an accident indicated that the information may not be available in time for accident management and emergency response decisionmaking. This information was, however, found to be useful in planning the recovery actions for the plant following an accident.

- Elimination of Hydrogen Analysis of Containment Atmosphere Sample. Item 2b of NUREG-0737 requires the containment atmosphere sample to be analyzed for hydrogen. This appears to be a duplicate requirement since the containment hydrogen monitor provides a real-time indication of containment hydrogen levels. This is a safety-grade monitoring system and is expected to function with a high degree of reliability. Therefore, this requirement could be eliminated without losing any critical data.

- Elimination of the Dissolved Gas Samples. Item 2c of NUREG-0737 requires analysis of the reactor coolant sample for dissolved gases, in particular, hydrogen. This provides information to determine the corrosive potential of the coolant and to infer the potential for in-vessel gas bubbles. The formation of noncondensible bubbles in the reactor vessel has the potential to uncover the core and decrease the heat removal capability of the engineered safety feature (ESF) systems. However, other plant systems adequately remove the potential for noncondensible gases to interfere with core cooling. The PASS sample requirement is redundant with these requirements and therefore is considered a candidate for elimination.

- Elimination of Guidance to Minimize Plateout in Sample Line. The intent of Item 11a of NUREG-0737 is to reduce radioiodine plateout in containment atmosphere sample lines. Some plants have installed heat tracing equipment as part of their sample lines to reduce plateout of radioiodine and particulates. The potential 3-hour delay in obtaining containment atmosphere sample results causes reactor operators and emergency response decisionmakers to take action and make decisions based on other indications of the potential source term that might be released if containment were to fail.

2.3 RISK IMPACTS OF IMPLEMENTING ALTERNATIVES

Risk has two components: the probability of occurrence of an event that could result in harm to the public and the consequences that might occur as a result of that event. To understand and evaluate the risk impacts of PASS and of the proposed modifications to current PASS requirements identified in the previous section, it is important to note that PASS, as the name implies, is used after an accident has been initiated to provide sample results that indicate the degree of the core damage and the content of the containment atmosphere. As such, PASS is not intended to, nor does it have, any influence on the frequency of reactor accidents. As a result, it is not meaningful to evaluate the benefits of PASS, or changes to PASS, in terms of accident frequency.

With respect to the second component of risk, the ability of PASS to reduce an accident's consequences through more timely actions to protect the public has been debated. Arguments have been formulated that PASS results can be used to guide operator response during an event, that PASS results can influence the decisions to protect the public in the event of radioactive releases, and that PASS results can provide data for assessing the degree of core damage that may have occurred during an accident.

The following subsections discuss the risk impacts of implementing the five alternatives defined in Section 2.2. Appendix A provides the basis for much of this discussion.

2.3.1 Relaxation of Time Requirement

The regulatory requirement regarding the time to obtain and analyze samples is three hours overall. Evaluations of several plant-specific emergency operating procedures, emergency response procedures and expected recovery actions indicated only one area where prompt sample analysis results could aid reactor operators in the mitigation or arrest of an accident. The analysis of the reactor coolant sample for boron concentration is needed to verify the operators' estimates of boron concentration that are based on mixing ratios. Upon detection of plant parameters that indicate the plant is experiencing a serious transient, the reactor is immediately isolated and engineered safety feature systems are started. Part of this action is the automatic isolation of reactor coolant system letdown which is intended to prevent the loss of reactor coolant. Since the source of primary coolant for normal boron sample analysis is in the letdown system, isolation of the system prevents the operators from obtaining boron analyses of the primary coolant throughout the transient. The level of boron in the reactor coolant system is essential to prevent criticality in a degraded core accident.

The timing of other PASS sample results was found to have marginal or negligible effects on public risk due to reactor accidents. Other PASS results (i.e., radionuclide composition of reactor coolant and containment atmosphere, dissolved hydrogen and oxygen levels in coolant, chloride concentrations in coolant, and pH of coolant) provide information on the amount of core degradation and the corrosive potential of the coolant. This information is not generally used to terminate or limit the progression of an accident to core melt, but is used primarily in recovery operations.

2.3.2 Elimination of the Radiological Analysis of Samples

The purpose of the requirement to obtain radiological analysis of reactor coolant samples is to establish the extent of core damage. The purpose of the analysis of containment atmosphere samples is to determine the potential for hydrogen combustion inside containment and to determine the source term of a radioactive release if one were to occur.

Reviews of the use of this information in the three phases of an accident indicated that the information may not be available in time for accident management and emergency response decisionmaking. Therefore the elimination of this requirement would have a marginal effect on the risk of reactor accidents to the public. This conclusion is due, in part, to the fact that other indicators of core damage, containment hydrogen concentration, and source term are more immediately available to reactor operators and emergency response decisionmakers, while PASS results may lag actual plant conditions by up to three hours. Other indicators of core damage include high-range containment radiation monitor readings, core-exit thermocouple readings, the reactor vessel level, and containment atmosphere hydrogen monitor readings.

Radiological analysis of reactor coolant and containment atmosphere samples was found to be useful in planning recovery actions. Containment atmosphere sample results provide an estimate of the radioactive source term in containment that may be related to potential occupational exposures during any reentrys or to potential offsite exposures should containment be vented.

2.3.3 Elimination of the Containment Hydrogen Sample and Analysis

The risk impact of eliminating this requirement was found to be negligible because plants have safety-grade containment hydrogen monitors that serve as a primary source of information concerning hydrogen concentrations in containment. The safety-grade hydrogen monitor is required by NUREG-0737, Item II.F.1(6) and provides real-time indication of hydrogen levels in containment. Hydrogen concentrations determined through PASS samples may be delayed up to three hours and would typically be used as a backup to the hydrogen monitor.

2.3.4 Elimination of the Dissolved Gas Samples

This dissolved gas sample analysis provides information that can be used to determine the corrosive potential of the coolant and to infer the potential for in-vessel gas bubbles. There is a risk associated with the formation of noncondensible bubbles in the reactor vessel because of their potential for uncovering the core and decreasing the heat removal capability of the ESF systems. Other NRC regulations establish requirements for 1) a reactor vessel level indication system to detect the presence of a bubble and core uncovery, and 2) a head vent system to remove noncondensible gases from the high points of the reactor coolant system. These systems adequately remove the potential for noncondensible gases to interfere with core cooling. The PASS sample requirement is redundant with these requirements. Therefore, dissolved gas information would not be used in the accident management or emergency response decisionmaking phases of an accident.

The information from the dissolved gas samples would be used primarily in the recovery phase as an indication of the corrosive potential of the reactor coolant. Information regarding the corrosive potential of the core would have no impact on the outcome of the accident and therefore, would not affect the protection of the public health and safety.

2.3.5 Elimination of Guidance to Minimize Plateout in Sample Lines

Plateout is a problem in the containment atmosphere sample lines, where radioiodine tends to precipitate onto the cooler surface of the sampling line tubing or pipe (particularly at line bends). Many utilities reduce this plateout by heat tracing the sample lines. Without heat tracing the results of the radiological analysis of the containment atmosphere sample are likely to be inaccurate: the noble gas concentration should be accurate but radioiodine and particulate concentrations could be significantly lower than actually present in containment. If the utility uses this information to estimate the source term in containment or the amount of core damage, the accident severity could be underestimated. As discussed in Appendix A, the potential 3-hour delay in obtaining containment atmosphere sample results causes reactor operators and emergency response decisionmakers to take action and make decisions based on other indications of the potential source term that might be released if containment were to fail. Therefore, the effect of eliminating the containment atmosphere sample and the guidance to minimize sample plateout affects public risk only marginally.

2.4 COST IMPACTS OF IMPLEMENTING ALTERNATIVES

This section discusses the cost impacts of implementing the following five alternatives to PASS:

- relaxation of the 3-hr time requirement for all samples except boron
- elimination of the radiological analysis of samples
- elimination of the hydrogen analysis of the containment atmosphere sample
- elimination of the dissolved gas samples
- elimination of the guidance to minimize plateout in sample lines.

Several utilities were contacted to estimate the potential impact of the above alternatives. Reactors designed by all major reactor manufacturers (i.e., General Electric, Westinghouse, Combustion Engineering, and Babcock & Wilcox) were represented.

As discussed before there are many types of PASS installed in nuclear plants. A 1984 survey of NRC Region III plants revealed six different manufacturers and several systems designed in-house. The manufacturers included Sentry, NUS, Bechtel, General Electric, and SAIC. Most systems rely primarily on obtaining grab samples of reactor coolant and containment atmosphere. Several plants have more sophisticated in-line systems which provide real-time isotopic results and other analytical results.

Utilities indicated the purchase and installation of postaccident sampling systems is an expensive undertaking. The evaluation indicated that PASS systems cost between $350,000 to $11,000,000. These expenditures are sunk cost. Much of the initial expense was due to installation costs and debugging the system. Some vendor-supplied systems did not initially function as designed and required major modifications by the utilities. One utility, for example, spent approximately $11 million installing an in-line system in one of their units and later installed the same system in another unit for a cost of about $2 million. The utility had worked out all the design and installation problems in the first unit. Annual maintenance costs for PASS ranged from $30,000 to $75,000/unit in this evaluation.

Regarding the five alternatives to PASS requirements, the utilities were questioned on the potential impact on:

- staff training requirements
- financial costs
- occupational radiation exposure
- information available for making decisions during an emergency.

The responses of the utilities to the above questions are presented below.

2.4.1 Relaxation of Time Requirement

Discussions with utility representatives indicate that all plants currently meet the three-hour time requirement. They indicate that short extensions of the time requirement (up to six hours, for example) would result in minimal cost savings.

At each plant, chemistry technicians are generally trained to collect and analyze PASS samples. At the plants contacted, the number of plant staff trained in the operation of PASS ranged from 7 to 30 people. The upper end of this range (30) corresponds to a multi-unit site. On back shifts there are a minimum of one to three chemistry personnel qualified to operate PASS. All utilities contacted indicated they would not decrease the number of staff trained if the time requirement were extended, for example, to six hours. Thus, there would be no cost savings associated with training.

2.4.2 Elimination of Radiological Analysis of Samples

Based on discussion with utilities and knowledgeable PNL staff, the capability for radiological analysis of samples already exists in plants. These systems are used for analysis and counting of the samples during routine and postaccident operations. Therefore, eliminating the radiological analysis of PASS samples would result in minimal cost savings.

2.4.3 Elimination of Hydrogen Analysis of Containment Atmosphere Samples

Based on discussions with selected utilities, elimination of the requirement for hydrogen analysis of containment atmosphere would result in minimal cost savings. Economically, there would be some savings resulting from:

- the elimination of training associated with the analytical procedures
- the elimination of annual or biannual review of the analytical procedures
- the elimination of calibration of equipment associated with the analysis
- the elimination of the need for spare parts for analytical equipment.

While the utilities did not quantify these savings, it is estimated by one utility that the annual savings would be less than $5000 per plant. This estimate is based on the number of man-hours required for training and procedure review.

2.4.4 Elimination of Dissolved Gas Sampling

In contrast, eliminating the requirement for analyzing dissolved gases in the reactor coolant sample could result in significant cost savings for the utilities. A pressurized sample of reactor coolant is generally required for analyzing the dissolved gas concentration in the coolant. The need for a pressurized reactor coolant sample results in a more complex PASS. For example, one utility estimated that the liquid PASS panel would only need 4-5 valves instead of the current 40-50 valves if a pressurized sample were not required. The additional valves result in greater operations and maintenance costs. One utility estimated that PASS maintenance time could be cut by 50% (equivalent to about $35,000/plant/yr for each system) if the number of valves in the system were reduced by a factor of 10.

Occupational radiation exposure could also be reduced if the number of valves were reduced because there would be fewer valves to maintain and a corresponding reduction in exposure. One utility estimated a reduction of up to two person-rem/yr.

2.4.5 Elimination of Guidance to Minimize Plateout in Sample Lines

Finally, eliminating heat tracing of containment atmosphere sample lines would not result in significant cost savings to the utilities because the heat tracing equipment is already installed. The utilities contacted indicated that maintenance cost and occupational radiation exposures associated with maintenance are minimal. For a new plant, a significant savings could be realized because of the initial cost of heat tracing equipment. One utility estimated this cost at $20,000/unit.

3.0 RISK AND COST IMPACTS OF TURBINE MISSILE REQUIREMENTS

The high speeds associated with the larger diameter, low-pressure turbine rotors in the main turbines of nuclear power plants constitute a significant source of energy. If this energy is released through the failure of a turbine rotor or blade, substantial damage may occur to the turbine and the surrounding equipment and buildings. Should a missile generated by a turbine failure penetrate the turbine casing and other structures and strike a safety-related structure or component with sufficient energy to cause damage, the safety of the plant may be degraded.

Early turbine rotor designs were susceptible to stress corrosion cracking in the keyway as a result of the rotor material and the shrink fit on the shaft. Recent developments in turbine rotor design, include improving the metallurgy of the rotors to reduce their susceptibility to stress corrosion cracking and changing to forged rotors, have reduced stresses in the rotor near the shaft.

Several studies (Bush 1978; NRC 1975; Twisdale 1983; Mullen 1986a) have suggested that the risk of a large radioactive release caused by a turbine missile does not constitute a significant contribution to the overall risk from reactor accidents. Present requirements to protect plant safety-related equipment from possible damage from turbine-generated missiles involve detailed NRC reviews of the turbine design and implementation of a turbine inspection program. These inspections are costly to utilities and, especially with the recent improvements in turbine designs, it may be possible to relax the frequency of these inspections without significantly changing the overall risk from reactor accidents.

3.1 CURRENT REGULATORY REQUIREMENTS

Current design criteria for nuclear power reactors require safety-related reactor structures, systems and components to be adequately protected against the effects of potential missiles that could result from equipment failures. General Design Criterion 4, "Environmental and Missile Design Bases," of 10 CFR 50, Appendix A, requires in part, "that structures, systems and components important to safety . . . shall be appropriately protected against the dynamic effects of missiles that might result from such failures" (CFR 1986). With regard to past construction permit (CP) and operating license (OL) applications, NRC evaluation of the effects of turbine failures on public health and safety followed Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles" (NRC 1977b), and Standard Review Plan (SRP) Sections 10.2, "Turbine Generator" (NRC 1981a), 10.2.3, "Turbine Disk Integrity" (NRC 1981b) and 3.5.1.3, "Turbine Missiles" (NRC 1977a).

Applicants or licensees who accept the turbine manufacturer's recommendations, in conjunction with the guidelines supplied in SRP 3.5.1.3, develop specific maintenance and inspection programs that include a curve (or curves) of missile generation probability (P_1) versus inspection interval for their specific rotors. These programs and the bases used to justify them are then transmitted to the NRC for review and approval.

According to NRC guidelines stated in Section 2.2.3 of the SRP and in Regulatory Guide 1.115, the probability of unacceptable plant damage from high energy missiles from the low-pressure stages of a large steam turbine (P_4) should be less than or equal to about one chance in ten million per reactor (ry) year for an individual reactor, i.e., $P_4 \leq$ 1.0E-07/ry. The probability of unacceptable damage resulting from turbine missiles is generally expressed as the following:

$$P_4 = P_1 * P_2 * P_3$$

where P_1 = the probability of turbine missile generation and penetration of the turbine casing
P_2 = the probability that a turbine missile strikes a critical reactor target, given generation and casing penetration
P_3 = the probability that a critical safety-related target is damaged, given a missile strike
P_4 = the overall probability of reactor damage due to turbine failures.

Turbine vendors have a long-established policy of recommending periodic inservice inspection of turbine components. In the case of nuclear low-pressure (LP) turbines, it has been the recommendation of turbine manufacturers that all components, including wheels, be thoroughly inspected at approximately six-year intervals. In some cases, inspection intervals shorter than six years have been recommended, depending on the condition of the turbine. The inspection interval is re-evaluated according to guidance provided by the turbine manufacturer each time an in-service inspection is conducted. For operating reactors, the inspection intervals recommended by the turbine manufacturers range from 1.5 to 6.0 years for current rotor designs.

3.2 ALTERNATIVES TO REGULATORY REQUIREMENTS

In an earlier phase of this project (Mullen 1986a), it was suggested that the improvements in turbine materials, designs and inspection methods merit a re-evaluation of the requirements for protection against turbine missiles. In addition, the analyses of turbine missile hazards may be conservative. Specifically, the probability of missile generation and penetration of the turbine casing may actually be lower than the value typically postulated and the damage probability may be conservative. The hazard analysis also presumes that reactor damage is synonymous with core melt.

Three alternatives to the current requirements were identified:

- Elimination of the requirements involving NRC reviews of licensee submittals on turbine missile protection, but maintaining the inspection intervals as they currently exist

- Partial relaxation of the turbine inspection requirements from an average interval of 4.5 years to a 9-year inspection interval while maintaining NRC review of licensee submittals

- Complete elimination of all review and inspection requirements, which is equivalent to a 30-year inspection interval.

Because the NRC requires utilities to implement the inspection recommendations of the turbine manufacturers, changes to or elimination of the turbine inspection requirements may depend on the response of the turbine manufacturers to the proposed alternatives. Contacts with the major turbine vendors indicate that, at least presently, they do not anticipate any changes in their recommended inspection frequencies. It must be recognized that vendors are interested in limiting all potential liability of failing, therefore, they have no financial motivation to reduce recommended inspection frequencies.

The principal alternative examined in this study was the complete elimination of turbine missile review and inspection requirements based on previous work that indicates turbine missiles are not a significant hazard to be included in the missile protection requirements of General Design Criterion 4 of 10 CFR 50, Appendix A. The following sections analyze the risks and costs associated with a possible relaxation or elimination of regulatory requirements involving turbine missiles for the three potential alternatives defined above.

3.3 RISK IMPACTS

Presently, turbine missile analyses assume the probability of missile generation and penetration of the turbine casing (P_1) to be approximately 1.0E-04/ry, based on historical failure rates. The probability of a turbine missile striking a safety-related component, given generation and casing penetration (P_2), is estimated on the basis of postulated missile sizes, shapes, and energies and on available reactor-specific information, such as turbine placement, number and type of intervening barriers, target geometry, and potential missile trajectories. In general, the conditional strike probability is on the order of 1.0E-04 to 1.0E-02. The conditional damage probability (P_3) is conservatively assumed to be 1.0.

Risk studies on the topic of turbine missiles are not new. Two earlier studies (Bush 1978 and NRC 1975) were based on historical data from both fossil and nuclear applications. The conclusions of both studies indicated that the probability of turbine missile damage to safety-related systems is considered to be small.

As a result of the importance of turbine missile requirements to reactor design, and in view of the need for new experimental and analytical research to provide improved data bases and mathematical models for prediction of turbine missile risks, a research program was initiated by the Electric Power Research Institute (EPRI) in 1976. This program included experimental and analytical turbine casing impact analysis, full and sub-scale reinforced concrete wall impact tests, turbine failure and missile generation analysis, and the development of a probabilistic analysis methodology. The principal findings of the study (Twisdale 1983) include the following:

- Given impact by a turbine-generated missile, the probability of scabbing and/or perforating a reinforced concrete barrier is much lower than previously suspected.

- The effects of turbine orientation on reactor impact and damage probabilities may not be as significant as previously suggested. A factor of four reduction in reactor perforation probability resulted when the turbine was reoriented to a peninsular design.

- Using the best estimate turbine failure rates of 1.64E-04/ry at the design operating speed, the damage probabilities for hitting and perforating external surfaces were estimated to be 7.0E-05/ry and 5.0E-06/ry, respectively. However, these results are reactor-specific.

In addition, reports concerning the probability of turbine failure at normal design speed and overspeed have been prepared by turbine manufacturers and are often incorporated by reference into utility safety analysis reports for licensing purposes. These turbine manufacturer reports are proprietary and are not discussed here other than to note that the missile generation probability (P_1) at the current inspection interval is much lower than the previously suggested value of 1.0E-05/ry.

Recent turbine experience also includes a fire caused by a turbine failure in the Maanshan Plant in Southern Taiwan. In brief, a turbine rotor failed and threw several blades that did not penetrate the turbine casing. The resulting rotor-imbalance, however, bent the shaft and caused the failure of the generator shaft seals. Hydrogen used to cool the generator leaked through the failed seals and ignited. Fires of this nature are not thought to be a concern in this analysis of turbine missile risks because of the extensive fire protection requirements contained in Appendix R of 10 CFR 50. Any safety-related equipment that may be in the turbine buildings of U.S. plants is required to be protected by fire barriers and fire suppression systems.

In summary, the major findings of risks due to turbine missiles are the following:

- Turbine missile risk is at least three orders of magnitude lower than the quantitative design objective for core melt (i.e., 1.0E-06/ry) that was considered in the Safety Goal Policy Statement proposed in August 1986. (The Safety Goal Policy Statement published in 1986 did not include a quantitative design objective for core melt.)

- P_1, the missile generation probability, may actually be lower than the value typically postulated (i.e., 1.0E-05/ry).

- P_2, the conditional strike probability (given generation), may be conservative.

- P_3, the conditional damage probability (given strike), may be conservative.

- Plant damage, represented by P_4, is not synonymous with core melt.

For better understanding, diagrams of turbine components (the shaft, disc, and blades) and the relative position of a turbine with respect to the reactor for favorable and an unfavorable orientation are shown in Figures 3.1, 3.2, and 3.3, respectively. The favorable orientation is defined as a peninsular arrangement of the turbine with respect to safety-related targets (i.e., the containment building). An unfavorable orientation is defined as a non-peninsular of the turbine with respect to safety-related targets.

3.3.1 Sensitivity of Risks to Changes in Regulatory Requirements

Because the utilities commit to the NRC to implement the inspection recommendations of the turbine manufacturers, any changes in the requirements regarding turbine missiles may depend on the response of the turbine manufacturers to the proposed alternatives. Contacts with the major turbine vendors indicate that, at least initially, no change in vendor recommendations for inspection frequencies would result. Turbine vendor warranties last for three years following turbine synchronization to the grid for commercial operation. At this point, there is no reason to assume that utilities would disregard the manufacturer's recommendations for inspections. Therefore, even if the NRC requirements were completely or partially eliminated, it is not expected that utilities would change their current inspection practices.

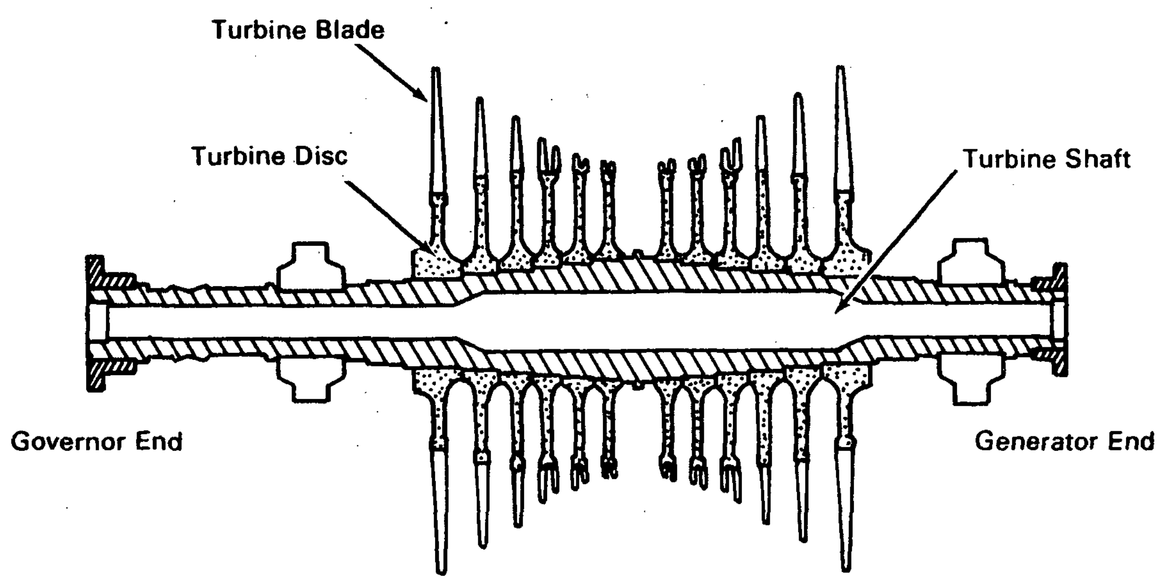

FIGURE 3.1. Schematic Drawing of Westinghouse Model X-1 Low-Pressure Rotor

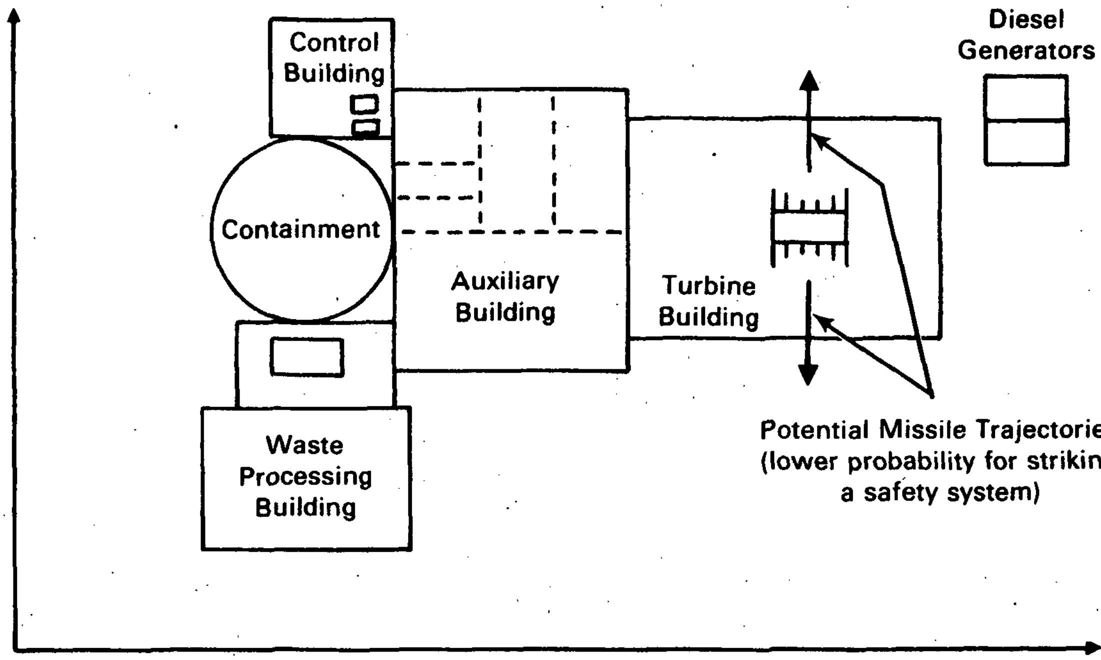

FIGURE 3.2. Favorable Orientation of Turbine with Respect to the Reactor (Peninsular)

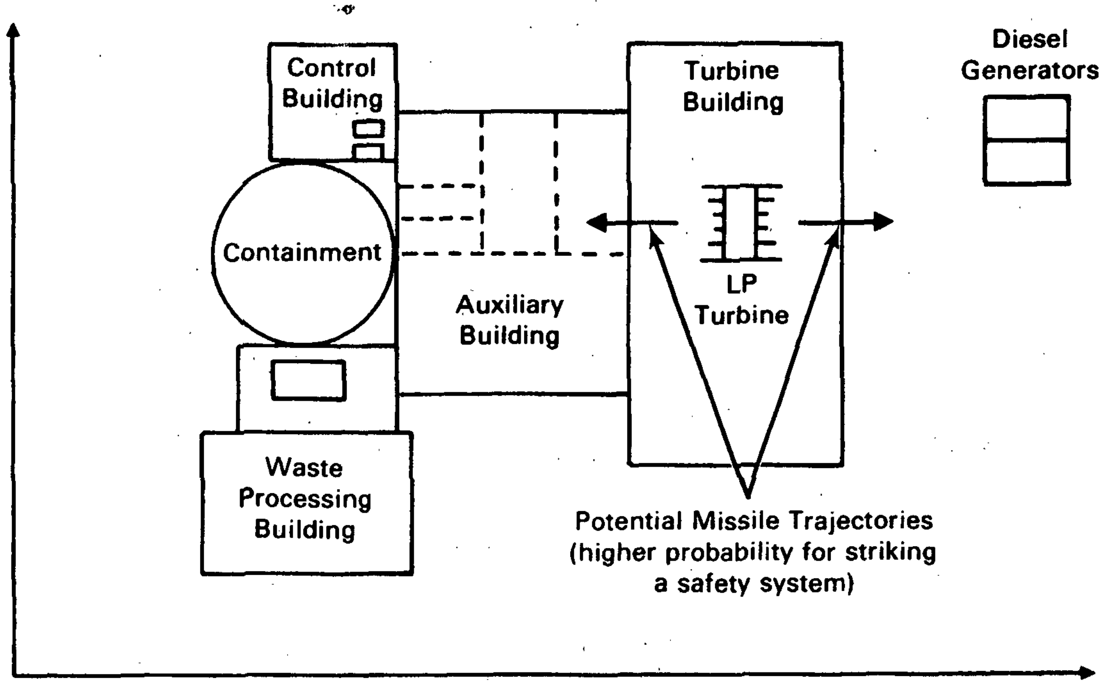

FIGURE 3.3. Unfavorable Orientation of Turbine with Respect to the Reactor (Non-Peninsular)

The first alternative considered in this study maintains the inspection intervals as they currently exist but eliminates the requirements involving NRC review of licensee submittals in the area of all turbine missiles. In light of today's design and technology, there should be no changes in the level of risk associated with the elimination of the NRC reviews and no changes in inspection frequency.

The remainder of this section focuses on the second and third alternatives, i.e., a sensitivity study of the risk impacts associated with the potential relaxation and elimination of inspection interval requirements. Generic techniques for estimating the uncertainties (upper and lower bounds) on parameters related to the public risk were described by Andrews (1983). A summary of the selected considerations utilized in this analysis is presented in Table 3.1.

A risk analysis was performed to assess the effect of decreased turbine rotor inspection intervals in all nuclear power reactors. This was accomplished by examining accident sequences in representative PWR and BWR reactors that involve power conversion system transients and by assuming concurrent failures in essential reactor equipment. As a check, WASH-1400 was reviewed for applicable accident sequences and compared to the representative plant analysis (NRC 1975).

Assumptions

The present inspection intervals range from 1.5 to 6 years, with an average of 4.5 years (base case). BWR and PWR reactors with existing cracks are required to have the entire turbine impacted every 1.5 years (corresponding to each refueling outage). The inspection intervals for the sensitivity study are assumed to range from 4.5 to 30 years, with an average of 9 years (adjusted case for alternative 2) and 30 years (adjusted case for alternative 3).

TABLE 3.1. Reactor Characteristics Assumed for Sensitivity Study

Consideration	Affected Reactor PWR	Affected Reactor BWR
Number of Reactors	83	40
Remaining Life (yr)	28.8	27.4
Inspection Interval (yr)		
• Present (base case)	4.5	4.5
• Assumed (adjusted cases)	9.0	9.0[a]
	30.0	30.0[b]

(a) Alternative 2.
(b) Alternative 3.

The remainder of this section presents data on turbine missile probabilities and the results of PWR and BWR risk analyses performed using a PNL-developed probabilistic risk assessment computer program.

Turbine Missile Data

Two major sources of data on the probability of turbine failures exist: 1) analyses of historical turbine failures in fossil and nuclear applications, and 2) estimates based on fracture mechanics models developed by turbine manufacturers. These two sources of information were used to estimate the probability of turbine failures. Not all turbine failures result in high energy turbine fragments that penetrate the turbine casing.

Historical data for the conditional probability of a turbine missile occurring given generation of turbine missile and penetration of the turbine casing were obtained from an EPRI study presented to a conference on "Turbine Missile Effects in Nuclear Power Reactors" on October 25-26, 1982 (Shaffer 1982). The results of the study are summarized in Table 3.2.

TABLE 3.2. Conditional Probability of Turbine Missiles Occurring Given a Turbine Failure

High Estimate	Best Estimate	Low Estimate
0.95	0.76	0.44

In addition to historical failure data, turbine manufacturers also use fracture mechanics models as a basis for estimating reactor-specific turbine failure rates and setting reactor-specific inspection schedules. This information is proprietary and is not reproduced in this report other than to note that the missile generation and casing penetration probabilities (P_1) at the 4.5-, 9-, and 30-year inspection intervals are 1.0E-05/ry, 4.0E-05/ry, and 1.6E-04/ry, respectively. From these data, the risk due to turbine missiles can be estimated. Alternatively, the length of the inspection interval corresponding to a predetermined missile generation probability can also be determined.

The final piece of information needed to calculate the frequency of significant damage due to turbine missiles (P_4) is an estimate of the conditional probability of a missile striking and damaging safety-related equipment given missile generation (i.e., the product of P_2 and P_3). While several industry studies have attempted to estimate these probabilities using sophisticated modeling techniques, their results are reactor-specific and complex. For the purpose of this analysis, the NRC-developed conditional probability (P_2) of the missile striking safety-related equipment was used. Values range from 1.0E-02 to 1.0E-04, and the largest value was used. The conditional damage probability (P_3) given a missile strike is conservatively assumed to be 1.0.

PWR Risk Impacts

To calculate the increase in risk due to lengthening the turbine inspection interval in PWRs, it was first necessary to determine the safety significance of damage to safety systems caused by turbine missiles. For this analysis, it was assumed that the turbine missile would strike and damage the most important single safety system modeled by the risk equations. Results of the Oconee-3 Reactor Safety Study Methodology Applications Program (RSSMAP) study used in prioritizing generic safety issues (Andrews 1983) were assumed to be representative of all PWRs.

Turbine failure was assumed to always cause loss of the power conversion system (PCS), thereby initiating a PCS (T_2) transient sequence. Only those parameters related to the susceptible systems were assumed to be affected in T_2 sequences. Thus, the only accident sequences in the Oconee study that could be affected by a turbine missile are T_2MLU, T_2MQH, T_2MQFH, T_2MLUO, T_2KMU, and T_2MQD. The system or component failures associated with a T_2 transient are given in Appendix A. Descriptions of these sequences are as follows:

- T_2MLU - Failure of high pressure injection system following emergency feedwater system failure, high head auxiliary feedwater system failure, and failure of normal PCS.
- T_2MQH - Failure of emergency coolant recirculation system after pressurizer safety/relief valves reclosure failure, and failure of normal PCS.
- T_2MQFH - Failure of emergency coolant recirculation system following containment spray recirculation system failure, pressurizer safety/relief valves reclosure failure, and failure of normal PCS.
- T_2MLUO - Failure of reactor building cooling system following high pressure injection system failure, emergency feedwater system failure, high head auxiliary feedwater system failure, and failure of normal PCS.
- T_2KMU - Failure of high pressure injection system following normal PCS and reactor protection system failures.
- T_2MQD - Failure of emergency coolant injection system following pressurizer safety/relief valves reclosure failure, and failure of normal PCS.

The frequency assumed for T_2 is 1.0E-05/ry for the base case. For the adjusted cases, T_2 was assumed to be 4.0E-05/ry and 1.6E-04/ry for the 9- and 30-year inspection intervals, respectively.

By performing a sensitivity analysis for each of the parameters associated with a T_2 transient, it was determined that risk was most sensitive to variations in F1, the failure of a pump in Train B of the low-pressure service water system. The value of F1 in the affected cases was assumed to be the product of the conditional probability of turbine missile occurrence (0.76) and the probability of safety system damage (the maximum product of strike, P_2, and damage probabilities, P_3, is assumed, i.e., 1.0E-02/ry), added to the original F1 (0.0014/ry, the failure rate in the RSSMAP report), yielding an affected F1 of 0.0090/ry. The conditional probability of nonrecovery of the PCS within 30 minutes following a T_2 transient (PCSNR) was conservatively set to 1.0/ry. Results of the calculations are shown in Table 3.3. Additional information from the computer calculations is contained in Appendix A.

BWR Risk Impacts

The BWR risk and core-melt frequency increases were calculated in the same manner as the PWR calculations. In this case, the representative reactor was assumed to be Grand Gulf 1. Results of the Grand Gulf 1 RSSMAP study used in prioritizing generic safety issues (Andrews 1983) were assumed to be representative of all BWRs.

Turbine failure was assumed to always cause loss of the power conversion system, thereby initiating a PCS (T_{23}) transient sequence. Only those parameters related to the susceptible systems were assumed to be affected in T_{23} sequences. Thus, the only accident sequences in the Grand Gulf study that could be affected by turbine missiles are T_{23}PQI, T_{23}PQE, T_{23}QW, and T_{23}C. As in the PWR analysis, the system, component or functional failures associated with a T_{23} transient in BWRs are given in Appendix A. Descriptions of these sequences are as follows:

- T_{23}PQI - Failure of residual heat removal system after a loss-of-coolant accident (LOCA) following failure of safety/relief valves to reseat and PCS failure.
- T_{23}PQE - Failure of emergency core cooling system following failure of safety/relief valves to reseat and PCS failure.

TABLE 3.3. PWR Risk Increase from Lengthening Inspection Intervals

Changing Inspection Intervals (years)	Best Estimate	High Estimate	Low Estimate
	Increase in Core Melt Frequency (event/ry)		
4.5 to 9	6.04E-09	3.02E-08	0
4.5 to 30	3.02E-08	1.51E-07	0
	Increase in Public Risk (person-rem/ry)		
4.5 to 9	1.61E-02	4.83E-01	0
4.5 to 30	8.05E-02	2.4E+00	0

- $T_{23}QW$ - Failure of residual heat removal systems following PCS failure.

- $T_{23}C$ - Failure to render the reactor subcritical following a reactor transient.

As in the PWR analysis, potentially affected parameters in the above accident sequences were identified. Studies were performed to determine the sensitivity of public risk to changes in the parameter values. The frequency assumed for T_{23} was 1.0E-05/ry for the base case, and 4.0E-05/ry and 1.6E-04/ry for the 9- and 30-year inspection intervals, respectively. No credit was taken for recovery of the PCS (i.e., parameter Q1 was conservatively set to 1.0/ry).

By performing a sensitivity analysis for each of the parameters associated with a T_{23} transient, it was determined that risk was most sensitive to variations in the loss of flow path into (defined as SSA) and through (defined as SSB) pumps A or B of the standby service water system, including the pump A or B oil coolers. The value of SSA in the affected cases was assumed to be the product of the conditional probability of turbine missile occurrence (0.76) and the probability of safety system damage (maximum product of P_2 and P_3 is assumed, i.e., 1.0E-02/ry) added to the original SSA (0.021/ry), the failure rate in the RSSMAP report, yielding an affected SSA of 0.0286/ry. Results of the risk calculations for the BWR are shown in Table 3.4. Additional information from the computer calculation is contained in Appendix A.

TABLE 3.4. BWR Risk Increase from Lengthening Inspection Intervals

Changing Inspection Intervals (years)	Best Estimate	High Estimate	Low Estimate
	Increase in Core Melt Frequency (event/ry)		
4.5 to 9	1.24E-08	6.18E-08	0
4.5 to 30	6.18E-08	3.09E-07	0
	Increase in Public Risk (person-rem/ry)		
4.5 to 9	8.77E-02	2.63E+00	0
4.5 to 30	4.38E-01	1.32E+01	0

Summary of Risk Impacts

To estimate the public benefit from increasing the inspection interval from 4.5 to 9 years, or from 4.5 to 30 years, the risk results for a representative reactor must be summed over the remaining lives of the affected reactors, (for reactor characteristics, see Table 3.1). The calculations of public risk increase for each potential alternative are:

Risk Increase in Going from a 4.5 to a 9-Year Inspection Interval

PWR: (1.61E-02 person-rem/ry) * (83 reactors) * (28.8 years) = 38.49 person-rem.

BWR: (8.77E-02 person-rem/ry) * (40 reactors) * (27.4 years) = 96.12 person-rem.

The total public risk increase for a 9-year inspection interval is approximately 135 person-rem with high and low estimates of 4.04E+03 and 0 person-rem, respectively.

Risk Increase in Going from a 4.5 to a 30-Year Inspection Interval

PWR: (8.05E-02 person-rem/ry) * (83 reactors) * (28.8 years) = 192.4 person-rem.

BWR: (4.38E-01 person-rem/ry) * (40 reactors) * (27.4 years) = 480.6 person-rem.

The total public risk increase for a 30-year inspection interval is approximately 673 person-rem with high and low estimates of 2.02E+04 and 0 person-rem, respectively.

3.3.2 Occupational Exposure

There are two components of occupational exposure to be considered: the exposure received as a result of a postulated accident and subsequent recovery, and the exposure received during routine reactor operation and maintenance. This section evaluates the potential changes in these two components of occupational exposure resulting from changes in the inspection intervals.

Occupational Exposure Changes During Accidents

The change in occupational exposure from accidents was estimated as the product of the change in total core-melt frequency and the occupational exposure likely to occur in the event of a major accident. The occupational exposure in the event of a major accident has two parts. The first is the "immediate" exposure to the personnel during the time it takes to control the accident. The second is the longer-term exposure associated with the cleanup and recovery from the accident. Doses associated with cleanup after turbine failures that do not result in a core melt in BWRs were considered negligible. For this analysis, the total occupational exposure is assumed to be 1.99E+04 person-rem/event.

The total avoided occupational exposure is calculated as follows:

4.5 to 9-Year Inspection Interval

PWR: (1.99E+04 person-rem/event) * (6.04E-09 event/ry) = 1.20E-04 person-rem/ry

BWR: (1.99E+04 person-rem/event) * (1.24E-08 event/ry) = 2.46E-04 person-rem/ry.

For the remaining life of all reactors, the estimated results for a 9-year inspection interval are:

Best Estimate = 5.47E-01 person-rem
High Estimate = 2.78E+00 person-rem
Lower Estimate = 0.0 person-rem.

4.5 to 30-Year Inspection Interval

PWR: (1.99E+04 person-rem/event) * (3.02E-08 event/ry) = 6.01E-04 person-rem/ry

BWR: (1.99E+04 person-rem/event) * (6.18E-08 event/ry) = 1.2E-03 person-rem/ry.

For the remaining life of all reactors, the estimated results of a 30-year inspection interval are:

Best Estimate = 2.78E-00 person-rem
High Estimate = 1.39E+01 person-rem
Lower Estimate = 0.0 person-rem.

Occupational Exposure Changes in Routine Operations

An essentially negligible occupational radiation dose due to turbine inspections occurs at PWRs. Based on this information, it is assumed, barring gross cross-contamination between the primary and secondary systems in all PWRs considered in this analysis, that only negligible occupational radiation dose from this source could be anticipated over the remaining PWR reactor lifetimes.

Because primary coolant is circulated through the turbine in BWRs, the turbine is contaminated and routine occupational exposure is incurred in turbine operation and maintenance. For the purposes of this portion of the analysis, the "selected" BWR was represented by a composite developed from information obtained from previous PNL work.

The occupational radiation dose of 3.25 person-rem given in Table 3.5 for the selected BWR during 1981 represents about 1/3 of the total dose anticipated

TABLE 3.5. Summary of Estimated Occupational Radiation Doses for a Selected BWR Turbine Generator System's Inspection/Maintenance Program Through the First Cycle[(a)]

Year	Estimated Length of Outage Dedicated to TGS Inspection/ Maintenance Program	Estimated Occupational Radiation Dose (person-rem)
1981	8 weeks	3.25
1982	8 weeks	4.40

(a) There are two BWRs at the selected reactor site. Each reactor's turbine-generator system (TGS) includes a total of 6 rotors: 3 low-pressure (LP) rotors, 1 high-pressure (HP) rotor, 1 generator rotor and 1 alterex/exciter rotor. Each unit's TGS inspection is completed on a combination 3- and 5-year schedule (i.e., each unit's turbine rotors are inspected on a 3-year cycle, while their respective generators are inspected once every 5 years).

for one complete inspection interval or cycle (i.e., for 1 HP and 3 LP rotors) or about 9.75 person-rem per cycle for the three LP rotors.

The 3.25 person-rem is considered a more reasonable representation of a typical dose associated with this inspection work than the 4.4 person-rem given for the 1982 work, because some undefined amount of additional occupational dose associated with moisture separator reheater work is known to be included in the latter dose estimate. There is essentially no occupational dose associated with inspection work on the generator portion of the turbine-generator system at the selected site.

Assuming that the same total amount of work (about 18 weeks for the three LP rotors) could be rescheduled for inspection intervals of 4.5, 9, or 30 years, then the occupational exposure for the inspection/maintenance tasks could be expected to remain virtually unchanged for these varied cycle lengths. Based on this reasoning, the comparison of the occupational radiation doses associated with two hypothetical cases is shown in Table 3.6.

From Table 3.6, the total occupational dose reduction in routine operations for all reactors is:

Doses Reduction in Changing from a 4.5- to 9-Year Inspection Interval

PWR: 0 person-rem.
BWR: (59.37 person-rem - 29.68 person-rem) * (40 reactors) = 1,188 person-rem.

Doses Reduction in Changing from a 4.5- to a 30-Year Inspection Interval

PWR: 0 person-rem.
BWR: (59.37 person-rem - 8.91 person-rem) * (40 reactors) = 2,018 person-rem.

TABLE 3.6. Comparison of Estimated Reactor Lifetime Inspection Dose (ERLID) for Three BWR Turbine Low-Pressure Rotor Inspections for Three Inspection Interval Cases

For a 4.5-year average inspection interval:

$$\text{ERLID} = \frac{\text{27.4 yr operating lifetime}}{\text{4.5-yr/cycle}} * \frac{\text{9.75 person-rem}}{\text{cycle}} = \text{59.37 person-rem}$$

For a 9.0-year average inspection interval:

$$\text{ERLID} = \frac{\text{27.4 yr operating lifetime}}{\text{9.0-yr cycle}} * \frac{\text{9.75 person-rem}}{\text{cycle}} = \text{29.68 person-rem}$$

For a 30.0-year average inspection interval:

$$\text{ERLID} = \frac{\text{27.4 yr operating lifetime}}{\text{30-yr cycle}} * \frac{\text{9.75 person-rem}}{\text{cycle}} = \text{8.91 person-rem}$$

3.4 COST IMPACTS

Because no change is anticipated in inspection frequency by the turbine manufacturers and the utilities, there should not be any significant cost savings associated with the elimination of the regulatory requirements regarding turbine missiles. This section focuses on a sensitivity study of the cost impacts associated with the assumed elimination of regulatory reviews and extension of inspection intervals. Again, the generic techniques for estimating the uncertainties (upper and lower bounds) of parameters related to the industry and NRC costs were obtained from Andrews (1983). The following subsections present the results of the sensitivity study.

3.4.1 Industry Implementation Costs

The costs to the nuclear industry of implementing the review and analysis actions are composed of management and licensing review costs for all reactors. This is estimated to require one person-month, or $8,333/reactor. A low estimate of $5,833/reactor reflects the view that substantial reductions from the one person-month effort are unlikely. The high estimate is taken to be three person-months, or $25,000/reactor. For 123 reactors, this yields a total industry implementation cost of $1.02 million, with lower and upper bounds of $0.72 million and $3.08 million, respectively.

The costs to the nuclear industry of implementing a 9- or 30-year turbine inspection interval involve assigning utility (or consultant) staff to process the planned changes through the approval chain, plan the implementation effort, train the staff, and make the necessary changes in reactor procedures. It is estimated that three person-months of staff time would be required (estimated cost of $25,000/reactor, with a range from $15,000 to $33,000/reactor). For 123 reactors, this yields a total industry implementation cost of $3.08 million, with lower and upper bounds of $1.84 million and $4.06 million, respectively.

3.4.2 Industry Inspection and Maintenance Costs

Cost Savings Due to Elimination of Regulatory Reviews

The cost savings due to streamlining regulatory requirements and eliminating some reviews would stem from savings in management review, QA control, licensing review, and engineering for all reactors. This saving is estimated to equal one person-year, or $100,000/reactor, with a range from $70,000 to $300,000/reactor. For 123 reactors, this yields a total cost savings of $12.3 million with lower and upper bounds of $8.6 million and $36.9 million, respectively.

Cost Savings Due to Relaxation/Elimination of Inspection Intervals

Information on sectionalized turbine and turbine-generator set inspection and maintenance work typically performed at selected PWRs and BWRs during refueling outages was obtained for this analysis. In both cases, all inspection tasks were assumed to be coordinated with the reactor refueling outages so that replacement power was not a factor. Thus, expensive downtime due to turbine inspection tasks was avoided. Because of the differences in costs, the following subsections present the costs of PWR and BWR inspection and maintenance separately.

PWR. Information on a typical PWR turbine-generator system inspection and maintenance program is presented in Table 3.7. The estimated average cost (in 1986 dollars) for each low-pressure rotor inspection is about $0.47 million and the time required is about six to eight weeks. Thus, the total estimated cost of one turbine system inspection cycle for three LP rotors for the selected PWR is about $1.41 million (3 LP rotors, $0.47 M each) as shown below, while the estimated cost of the entire turbine inspection cycle is about $3.35 million.

	$ Million (1986 Dollars)
3 LP Rotors (0.47M/each)	1.41
1 HP Rotor ($1.33M average)	1.33
Generator Rotor and Alterex/Exciter Rotor	0.61
TOTAL	3.35

It should be recognized that this adjusted total cost (in 1986 dollars) for the three LP rotor inspections remains virtually unchanged for varied cycle lengths, assuming there are no changes in inspection time per rotor. The reason is that the same total amount of effort, requiring about 16 to 24 work weeks, will be done, regardless of whether those weeks are scheduled over 4.5, 9 or 30 years of outages. Therefore, the estimated total cost per cycle for the three LP rotor inspections can be utilized to determine the impact over the lifetime of the reactor resulting from revised NRC requirements for all nuclear

TABLE 3.7. Summary of Estimated Costs of a Selected PWR Turbine-Generator System's Inspection/ Maintenance Program Through the First Cycle, Including the Burn-In Inspection

Year	Rotor Inspection Task[a]	Estimated Length of Outage Dedicated to TGS Inspection/ Maintenance Program	Primary Manpower Utilized		Estimated Cost $M[b]	Adjusted Costs, Millions (1983 Dollars)	Adjusted Costs, Millions (1986 Dollars)[c]
			Reactor Forces	Contractor(s)			
1977	Burn-in, tear-down inspection[d]	6 months	X	X	0.5 to 0.6	0.77 to 0.92	0.89 to 1.06
1978	LP "B" Rotor	8 weeks		X	0.28	0.41	0.47
1979	N/A[e]	--		--	--	--	--
1980	HP Turbine Rotor	6 to 8 weeks		X	0.9 to 1.0	1.1 to 1.22	1.27 to 1.41
1981	LP "C" Rotor	6 to 8 weeks		X	0.35 to 0.4	0.39 to 0.44	0.45 to 0.51
1982	Generator Rotor and Alterex/Exciter Rotor	10 weeks	X	X	0.5	0.53	0.61

(a) The selected reactor's Turbine-Generator System (TGS) includes a total of 6 rotors: 3 Low-Pressure (LP) rotors, 1 High-Pressure (HP) rotor, 1 Generator rotor, and 1 Alterex/Exciter Rotor.
(b) It should be recognized that other TGS tasks (e.g., valve inspections and maintenance) also were accomplished during the outage according to the selected reactor's Turbine-Generator Inspection Schedule. For the most part the costs given are for vendor(s) via contracts and do not generally include normal outage overhead costs (e.g., planning, engineering, security, etc.).
(c) Based on a standard cost escalation factor.
(d) Burn-in inspection includes complete tear-down inspection after the first year. Only the LP "B" rotor was not inspected.
(e) N/A means not applicable, since other priority work (non-TGS related) took precedence in CY-1979.

reactors. Based on this reasoning, the costs for the three cases corresponding to 4.5-, 9- and 30-year inspection intervals are compared in Table 3.8.

BWR Costs. Information on a typical BWR turbine generator system inspection and maintenance program is presented in Table 3.9. The estimated average cost (in 1986 dollars) for each low-pressure rotor inspection is about $0.93 million. Thus, the total estimated cost of one inspection cycle for three LP rotors at the selected BWR is about $2.8 million (3 LP rotors, $0.93 million each). Assuming that the same total amount of work, about 24 weeks' worth, could be scheduled over 4.5-, 9- or 30-year inspection intervals, then the adjusted combined total cost for inspecting the three LP rotors remains virtually unchanged for these varied cycle lengths except for minor adjustments due to inflation. The costs for inspecting the three LP rotors associated with the two inspection interval cases are compared in Table 3.10.

TABLE 3.8. Comparison of Estimated Reactor Lifetime Inspection Cost (EPLIC) of Three PWR Turbine Low-Pressure Rotor Inspections for Three Inspection Interval Cases (1986 dollars)

Case 1

For a 4.5-year average inspection interval:

$$\text{ERLIC} = \frac{\text{28.8 yr operating lifetime}}{\text{4.5 yr avg}} * \frac{\$1.41\text{M}}{\text{cycle}} = \$9.02 \text{ million}$$

Case 2

A 9.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{28.8 yr operating lifetime}}{\text{9.0 yr avg}} * \frac{\$1.41\text{M}}{\text{cycle}} = \$4.51 \text{ million}$$

Case 3

A 30.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{28.8 yr operating lifetime}}{\text{30.0 yr avg}} * \frac{\$1.41\text{M}}{\text{cycle}} = \$1.35 \text{ million}$$

TABLE 3.9. Summary of Estimated Costs of a Selected BWR Turbine-Generator System's Inspection/Maintenance Program Through the First Cycle

Year	Rotor Inspection Task[a]	Estimated Length of Outage Dedicated to TGS Inspection/ Maintenance Program	Primary Manpower Utilized: Reactor Forces	Primary Manpower Utilized: Contractor(s)	Estimated Cost $M[b]	Adjusted Costs, Millions (1983 Dollars)	Adjusted Costs, Millions (1986 Dollars)[c]
1980	--	8 weeks		X[d]	1.1	1.34	1.54
1981	--	8 weeks		X[d]	0.73	0.81	0.93
1982	--	8 weeks		X[d]	0.76	0.80	0.92
--[e]	--	10 weeks		X	0.5	0.53	0.61

(a) There are two BWRs at the selected reactor site. Each reactor's Turbine-Generator System (TGS) includes a total of 6 rotors: 3 Low-Pressure (LP) rotors, 1 High-Pressure (HP) rotor, 1 Generator rotor, and 1 Alterex/Exciter Rotor. Each unit's TGS inspection is completed on a combination 3 and 5 year schedule (i.e., each unit's turbine rotors are inspected on a 3-year cycle, while their respective generators are done once every 5 years). In addition, similar inspections for each unit take place on alternate years. Since no inspection data were obtainable prior to CY-1981, data from both units are synthesized to represent the equivalent of the required inspections for a single BWR for purposes of this analysis.

(b) It should be recognized that other TGS tasks (e.g., valve inspections and maintenance) also were accomplished during the outage according to the selected reactor's Turbine Generator Inspection Schedule. For the most part the costs given are for vendor(s) via contracts and do not generally include normal outage overhead costs (e.g., planning, engineering, security, etc.).

(c) Based on a standard cost escalation factor.

(d) Essentially 99% plus of the work is done by fixed bid contract.

(e) These data were unobtainable, therefore, for purposes of this analysis, they are assumed to be the same as these given for the generator rotor work of the selected PWR presented in Table 3.7.

TABLE 3.10. Comparison of Estimated Reactor Lifetime Inspection Cost (ERLIC) of Three BWR Turbine Low-Pressure Rotor Inspections for Three Inspection Interval Cases (1986 Dollars)

Case 1

For a 4.5-year average inspection interval:

$$\text{ERLIC} = \frac{\text{27.4 yr operating lifetime}}{\text{4.5 yr avg}} * \frac{\$2.8\text{M}}{\text{cycle}} = \$17.0 \text{ million}$$

Case 2

A 9.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{27.4 yr operating lifetime}}{\text{9.0 yr avg}} * \frac{\$2.8\text{M}}{\text{cycle}} = \$8.52 \text{ million}$$

Case 3

A 30.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{27.4 yr operating lifetime}}{\text{30.0 yr avg}} * \frac{\$2.8\text{M}}{\text{cycle}} = \$2.56 \text{ million}$$

Summary of Industry Operating Costs. All reactors considered in Table 3.1 are assumed to be affected by the proposed actions. Tables 3.8 and 3.10 present the lifetime costs for PWRs and BWRs, respectively.

The total industry operating cost savings in 1986 dollars for changing inspection intervals are:

- Changing inspection intervals from 4.5 to 9 years

  83 * (9.02M - 4.51M) + 40 * (17.0M - 8.52M) = $714 million, with estimated bounds of $358 million and $1,073 million.

- Changing inspection intervals from 4.5 to 30 years

  83 * (9.02M - 1.35M) + 40 * (17.0M - 2.56M) = $1,214 million, with estimated bounds of $608 million and $1,824 million.

3.4.3 Change in Turbine Failure Costs

To estimate the costs due to turbine failure, two cases were considered: 1) turbine failures that result in relatively minor damage (for example, failures in which the turbine casing is not penetrated and damage is confined to the turbine itself), and 2) turbine failures that result in major plant damage (for example, failures that produce one or more large, energetic missiles that penetrate the casing and cause damage beyond the turbine itself). Table 3.11

TABLE 3.11. Change in Turbine Missile Frequency (per reactor year)

Change in Inspection Interval	Major Damage Frequency	Minor Damage Frequency
4.5 yr to 9 yr	3.0E-05	7.20E-06
4.5 yr to 30 yr	1.5E-04	3.60E-05

gives the changes in probabilities for these two cases as a fraction of changes in the inspection interval. These changes in probabilities are then multiplied by the estimated costs shown in Table 3.12. (The cost estimates for repairing damaged equipment and structures are shown in Table 3.13.) The result is the estimated change in turbine failure costs due to lengthening the inspection interval. The resulting cost increases are summarized in Table 3.14.

3.4.4 NRC Development Costs

NRC development costs are those costs incurred in establishing the need for the change in regulations and the costs involved in confirming the decision to implement a change. For all cases, this activity is estimated to require two person-months of professional staff time, or $16,667/reactor. For the entire industry (123 reactors), this is estimated to be $2.05 million with estimated bounds of $1.64 million and $2.50 million.

3.4.5 NRC Implementation Costs

NRC costs incurred to implement the changes in regulations would include NRC staff time to prepare rulemaking packages, Federal Register Notices, responses to comments, etc. For all cases, it is estimated that six person-months of staff time would be required to implement the rule change, equivalent to $50,000, and could range from $40,000 to $60,000.

3.4.6 NRC Operation Costs

The change in NRC operation costs associated with this proposed change in regulatory requirements stems from reduced NRC staff effort as a result of eliminating the turbine review process conducted in support of reactor licensing. This is estimated to require one person-month, or $8,333/reactor, with a range from $6,700 to $10,400/reactor. For 123 reactors, this yields a total cost savings of $1.02 million with a lower and upper bounds of $0.82 million and $1.28 million, respectively.

Additionally, less NRC inspection effort would be required at each reactor to observe, monitor and evaluate the results of turbine inspections. For all cases, this reduction in NRC staff efforts is estimated to be $16,667 (two person-months) per reactor year.

TABLE 3.12. Estimated Generic Forward-Fit LWR Low-Pressure Turbine Disc Failure Costs for Two Types of Hypothetical Failures[a]

	Estimated Costs, Millions (1986 dollars)	
Category	Missile Penetrates Turbine LP Casing	Missile Does Not Penetrate Turbine LP Casing
1. Repair Activities		
- Rebuild Damaged Rotor[b]	5.75	5.75
- Rebuild Upper Casing	0.35 to 1.15	0.23 to 0.58
- Miscellaneous[c]	0.58	0.58
- Repair Other Damaged Equipment/Structures[d]	3.50 to 16.50	0
2. Outage Costs[e]	29.0 to 116.0	29.2 to 58.0
3. Replacement Activities	13.8	13.8
- Rotor Replacement Turbine Overhaul (excludes normal maintenance)		
TOTALS	53 to 154	50 to 79
Best Estimate[g]	70	50

(a) Selected forward-fit reactors are given in Oconee 3 and Grand Gulf 1.
(b) Includes costs of transportation to the vendor's factory.
(c) Includes costs of vendor's engineers onsite and onsite machine shop work.
(d) Estimate is based on cost breakdown given in Table 3.13.
(e) Outage costs (assumed to be primarily replacement power costs) were taken from Andrews (1983). Outage time is based on information provided by the Rancho Seco Nuclear Plant, Clay Station, California. Rancho Seco used 12 weeks for their repair. An upper bound of 1 year for missile generating events and 6 months for nonmissile generating events was used. The lower bound was assumed equal to the Best Estimate for both cases.
(f) Low pressure rotor replacement costs estimated at $10.5 million based on information taken from EPRI Journal (1982) and escalated to 1986 dollars; $1.73 million estimated for turbine overhaul.
(g) The uncertainty bound given in the table reflects a 50% spread, which was an estimate of the uncertainty level.

Similar to the previous section (i.e., industry operational cost), the comparison of the NRC operational costs associated with the two hypothetical cases for PWRs and BWRs are shown in Tables 3.15 and 3.16, respectively.

TABLE 3.13. Estimated Costs of Repairs for Equipment and Structures Damaged in the Hypothetical LWR LP Turbine Missile Generating Event

Category	Estimated Costs Millions (1986 dollars)
Prerepair Engineering	0.12 to 0.58
Mobilization/Demobilization	0.17
Supplies and Equipment	0.08 to 0.58
Decontamination:	
Not Required (if PWR)	0
Required (if BWR)	1.20
Equipment and Structure Repairs	1.20 to 12
Subtotal	2.77 to 14.53
Contingency (25%)	0.69 to 3.63
Total	3.46 to 18.16

TABLE 3.14. Change in Turbine Failure Costs

Accident Type	Change in Inspection Interval (yr)	Change in Turbine Failure Costs Estimate ($M/yr)
Major Plant Damage	4.5 to 9	0.26
	4.5 to 30	1.30
Minor Damage (confined to the turbine)	4.5 to 9	0.05
	4.5 to 30	0.22

The total NRC operating cost savings for changing inspection intervals are:

- Changing inspection intervals from 4.5 to 9 years:

  83 * ($0.107M-$0.053M) + 40 * ($0.102M-$0.0051M) = $6.50 million, with estimated bounds of $3.25 million and $9.75 million.

- Changing inspection intervals from 4.5 to 30 years:

  83 * ($0.107M-$0.0016M) + 40 * ($0.120M-$0.0015M) = $12.8 million, with estimated bounds of $6.40 million and $19.2 million.

TABLE 3.15. Comparison of Estimated Reactor Lifetime Inspection Cost (ERLIC) of Three PWR Turbine Low-Pressure Rotor Inspections for Three Inspection Interval Cases

Case 1

For a 4.5-year average inspection interval:

$$\text{ERLIC} = \frac{\text{28.8 yr operating lifetime}}{\text{4.5 year avg}} * \frac{\$16,667}{\text{cycle}} = \$0.107 \text{ million}$$

Case 2

For a 9.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{28.8 yr operating lifetime}}{\text{9.0 year average}} * \frac{\$16,667}{\text{cycle}} = \$0.053 \text{ million}$$

Case 3

For a 30.0-year inspection interval:

$$\text{ERLIC} = \frac{\text{28.8 yr operating lifetime}}{\text{30.0 year average}} * \frac{\$16,667}{\text{cycle}} = \$0.0016 \text{ million}$$

3.5 CONCLUSIONS

In summary, this sensitivity study to compare the risks and costs of 4.5-, 9- and 30-year inspection intervals demonstrates that the increase in public risk from reactor accidents is minor for the 9- and 30-year inspection intervals compared to the cost savings of the industry. Table 3.15 summarizes these results.

These data reconfirm the early assessments of turbine missile risks by Bush (1978) and the Reactor Safety Study (NRC 1975) that concluded, "the probability that a large radioactive release could be caused by a turbine missile does not represent a significant contribution to the overall risks from reactor accidents." The analysis further demonstrates that this regulatory requirement exacts a high price from industry for compliance. Therefore, if the regulatory requirements were eliminated and the utilities extended the turbine inspection intervals, the benefits could be significant and yet have a minimal impact on public safety.

It should be recognized that turbine inspections and overhauls are closely related. Turbine overhauls are currently completed during the required turbine rotor inspections. A turbine overhaul is needed periodically to check, maintain, and repair turbine parts and functions. This maintenance work is needed at intervals ranging from five to seven years in length. In addition,

TABLE 3.16. Comparison of Estimated Reactor Lifetime Inspection Cost (ERLIC) of Three BWR Turbine Low-Pressure Rotor Inspections for Three Inspection Interval Cases

Case 1

For a 4.5-year inspection interval:

$$\text{ERLIC} = \frac{\text{27.4 yr operating lifetime}}{\text{4.5 year average}} * \frac{\$16{,}667}{\text{cycle}} = \$0.102 \text{ million}$$

Case 2

For a 9.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{27.4 yr operating lifetime}}{\text{9.0 year average}} * \frac{\$16{,}667}{\text{cycle}} = \$0.051 \text{ million}$$

Case 3

For a 30.0-year average inspection interval:

$$\text{ERLIC} = \frac{\text{27.4 yr operating lifetime}}{\text{30.0 year average}} * \frac{\$16{,}667}{\text{cycle}} = \$0.0015 \text{ million}$$

inspection and maintenance in accordance with manufacturer's recommended intervals may be needed to satisfy the conditions for the manufacturer's warranty of the turbine, in the very early years of turbine operation.

The reduction in either the risk/dose or the cost (Table 3.17) may not actually be realized because the utilities may continue to inspect and maintain the turbines at the present intervals regardless of regulatory relaxation. Therefore, if the regulatory requirement regarding the frequency of inspections were eliminated or relaxed, the benefit may be minimal; the utilities' operating flexibility may be enhanced but the actual monetary savings may be insignificant.

TABLE 3.17. Summary of Risks and Costs Associated with Turbine Missiles

Affected Reactors PWR (Planned and Operating) = 83
BWR (Planned and Operating) = 40
Assumed Inspection Interval = 9.0 and 30 years

	Best Estimate		
Risk/Dose Impact (person-rem)	Alternative 1[a]	Alternative 2[b]	Alternative 3[c]
Public Risk	0	-135.00	673.00
Occupational Exposure:			
- Accident	0	-0.55	-2.78
- Operational	0	-1187.60	-2018.40
Cost Results ($Millions)			
Industry Cost Impact:			
- Implementation	-1.02	-3.08	-3.08
- Operational	12.30	713.53	1214.21
Industry Costs Due to Turbine Failure:[d]			
- Major Damage	0	-0.26	-1.30
- Minor Damage	0	-0.05	-0.22
NRC Cost Impact:			
- Development	-2.05	-2.05	-2.05
- Implementation	-0.05	-0.05	-0.05
- Operational	1.02	6.50	12.80

(a) Alternative 1 - Eliminate the requirements involving NRC reviews of licensee submittals on turbine missile protection, but maintain the inspection intervals as they currently exist.
(b) Alternative 2 - Partial relaxation of the turbine inspection requirements from an average interval of 4.5 years to a 9-year inspection interval while maintaining NRC review of licensee submittals.
(c) Alternative 3 - Complete elimination of all review and inspection requirements, which is equivalent to a 30-year inspection interval.
(d) These costs are represented by the industry's analyzed financial risk due to lengthening the inspection interval. Major damage is defined as failures that produce one or more large energetic missiles that penetrate the casing and cause damage beyond the turbine itself. Minor damage is defined as failures in which the turbine casing is not penetrated and damage is confined to the turbine itself.

NOTE: Favorable or beneficial consequences of a proposed action have a positive sign. Unfavorable or adverse consequences have a negative sign. For instances, an increase in industry or NRC implementing costs would be considered an unfavorable consequence and should be entered in the table with a negative sign.

The reduction in either the risk/dose or the cost may not actually be realized because the utilities may continue to inspect/maintain at the present intervals (e.g., periodic turbine inspection/overhaul to satisfy the conditions for the manufacturer's warranty) regardless of regulatory relaxation.

4.0 RISK AND COST IMPACTS OF SELECTED COMBUSTIBLE GAS CONTROL REGULATORY REQUIREMENTS

Regulations have been implemented to require PWR AND BWR licensees to include in their plants the "... means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA), by 1) metal-water reaction involving the fuel cladding and the reactor coolant, 2) radiolytic decomposition of the reactor coolant, and 3) corrosion of metals" (10 CFR 50.44). If a LOCA were to occur, this gas would collect in the containment building and, if the relative concentrations of hydrogen and oxygen were to reach certain levels, the mixture could burn or explode. A hydrogen explosion could potentially damage safety-related equipment inside containment that may be needed to establish and maintain control of the reactor, and could rupture the containment building as well, allowing radioactivity to escape into the environment.

In an earlier phase of this project (Mullen 1986a), some licensees identified these requirements as being unnecessarily burdensome, having only marginal effectiveness in limiting public risk. Two aspects of the requirements were cited as examples. The first concerned hydrogen recombiners. Utilities contacted in that study indicated that hydrogen recombiners do not significantly decrease public risk since the recombiners are generally incapable of processing hydrogen at the rate generated under the hypothetically extreme conditions of severe core damage. The second example concerned inerting during startup of BWRs with Mark I and II containments. Several of the utilities contacted in the study that had recently started up plants with Mark I or Mark II containments indicated that the requirement to inert containment within six months after initial criticality constituted an unjustified regulatory burden because startup testing was not complete. The burden of this requirement has been partially lifted due to specific inerting exemptions granted by NRC staff. However, these utilities still had to expend resources to apply for and justify their exemption requests.

Based on these considerations, the purpose of this chapter is to evaluate the risks, costs, and benefits of selected alternatives to combustible gas control regulatory requirements. The study was limited to three aspects of the requirements: 1) the requirement to provide recombiner capability for the inerted BWR Mark I and II containments, 2) the requirement to inert BWR Mark I and II containments within six months after initial criticality, and 3) the requirement for recombiner capability in large, dry containments.

Section 4.1 summarizes the regulatory requirements in each of these areas, Section 4.2 describes the tentative alternatives to the requirements identified and evaluated in this study, Section 4.3 describes the risk effects of the alternatives and Section 4.4 describes their costs and cost savings.

4.1 CURRENT REGULATORY REQUIREMENTS

The current standards for combustible gas control are contained in 10 CFR 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors." These standards apply to both boiling and pressurized light water nuclear power reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding. In general, the regulation requires the means to control hydrogen gas, the capability to measure hydrogen gas concentrations in the containment, the means to insure that the containment atmosphere is well mixed, and the means to control combustible gas concentrations in the containment following a postulated LOCA. The specific requirements with regard to the amount of fuel cladding involved in a reaction with the coolant depend on the type of containment at the plant.

Each plant must demonstrate that during the time period following a postulated LOCA but prior to the operation of the combustible gas control system that an uncontrolled hydrogen-oxygen reaction would not occur in containment or that the plant could withstand the consequences of an uncontrolled hydrogen-oxygen reaction without the loss of safety functions. If any of these conditions cannot be satisfied, then the containment must be inerted.

Because of the differences among containment types (volume, design pressures, etc.) at plants, the requirement for combustible gas control has different effects. In the following sections, the implications of the requirements are summarized for three distinct cases: requirements applicable to all types of containments, additional requirements applicable to BWR Mark III and PWR ice condenser containments, and requirements for BWR Mark I and II containments.

4.1.1 Requirements for All Types of Containments

All plants must be provided with either an internal or external recombiner or the capability to install an external recombiner following the start of an accident. Plants with construction permits issued after November 5, 1970, are required to have either internal or external built-in recombiners. Earlier plants are only required to have the capability of installing a recombiner following an accident. The required design capacity of the recombiners is the larger of the following values:

- Five times the hydrogen generation calculated in demonstrating compliance with the emergency core cooling system (ECCS) performance requirements of 10 CFR 50.46(b)(3), which is required to be less than 0.01 times the hypothetical amount of hydrogen generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. In the most conservative case, this translates to a recombiner capable of processing the hydrogen generated in a metal-water reaction involving less than 5% of the cladding in the active fuel region of the core.

- The amount of hydrogen generation that would result from the reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 in. In other words, between about 1% and 5% of the active fuel cladding must be assumed to react within approximately 2 minutes.

If neither of these evaluations has been conducted, the recombiners shall be designed for a metal-water reaction involving 5% of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.

4.1.2 Additional Requirements for BWR Mark III and PWR Ice Condenser Containments

In addition to the requirements applicable to all plants, plants with these types of containments must be provided with a hydrogen control system capable of processing an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume, without loss of containment structural integrity. Furthermore, systems and components needed to establish and maintain safe shutdown and maintain containment integrity must be able to function during and after a hydrogen burn, if the containment is not inerted. A metal-water reaction involving 75% of the active fuel cladding was selected as a limiting case of degraded core accidents, beyond which a core melt is expected to occur under all circumstances.

4.1.3 Requirements for BWR Mark I and II Type Containments

In addition to the requirement that these containments be provided with recombiners having the capacity described in Section 4.1.1, the regulations require that all BWR Mark I and II containments be inerted.

4.2 ALTERNATIVES TO REGULATORY REQUIREMENTS

Because the scope of the regulations surrounding combustible gas control is so broad, three areas of concentration were initially selected for this project. The three areas in which alternatives to the current requirements were identified and evaluated are:

- recombiners in inerted BWR Mark I and II containments
- time to initially inert BWR Mark I and II containments after initial criticality
- recombiners in large, dry containments.

The results of a preliminary scoping study conducted to evaluate these initial alternatives are described in Sections 4.2.1 through 4.2.3. From these results, one of the alternatives was selected for more detailed evaluation: the requirement for recombiner capability in inerted BWR Mark I and II containments. The results are presented in Sections 4.3 and 4.4.

The other two alternatives were not considered beyond the preliminary scoping study. In the case of the time to inert BWR Mark I and II containments after initial criticality, only marginal benefits were identified in the scoping study. Because other ongoing NRC research projects are addressing the need for additional hydrogen control measures in large, dry containments for degraded core accidents involving about 75% of the active fuel cladding, further evaluation of recombiner requirements in large, dry containments was deferred until the results of the degraded core research are available. The outcome of that program may affect the control of hydrogen generated in design basis accidents involving about 5% of the active fuel cladding.

4.2.1 Recombiners in Inerted BWR Mark I and II Containments

BWR Mark I and II containments are required to be inerted and provided with a recombiner capability. Several utilities operating BWRs with Mark I and II containments have been granted exemptions by NRC from the requirement to provide recombiner capability. NRC is empowered by 10 CFR 50.12 to grant exemptions because the Commission believes that "it is not possible for regulations to predict and accommodate every conceivable circumstance" (NRC 1985). In order to be responsive to the interpretations of the courts with respect to the granting of exemptions (NRC 1985), NRC has recently revised 10 CFR 50.12 to clarify the standards applied when it considers whether to grant exemptions from the regulatory requirements. Currently, the rule requires that an exemption can only be granted if special circumstances are present, and defines several categories of special circumstances. The following special circumstances may have influenced granting exemptions to the requirement for recombiner capability in BWR Mark I and II containments:

- Where compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated [10 CFR 50.12(a)(2)(iii)].

- Where any other material circumstance is present that was not considered when the regulation was adopted for which an exemption would be in the public interest [10 CFR 50.12(a)(2)(v)].

The NRC also acknowledged in the Notice of Final Rulemaking for 10 CFR 50.12 that it "will exercise its discretion to limit exemptions in any particular area if the 'exceptions' to the rule threaten to erode the rule itself." The NRC recognizes, however, that exemptions may be an indicator that a rule may need to be revised. The exemption requests can serve to supplement the traditional evaluation process for determining areas of the regulations in need of revision.

Because the exemptions from recombiner capability requirements requested by plants with inerted Mark I and II containments may be an indicator of an area of the regulation needing revision, this area was selected for further evaluation in this study.

4.2.2 Time to Initially Inert BWR Mark I and II Containments Following Initial Criticality

BWR Mark I and II containments are required to be inerted within six months after initial criticality. Usually, as a Mark I or II plant is approaching the six-month mark after initial criticality in the startup testing program, the utility formally requests an exemption to the inerting requirement. The exemption request may be based on two of the special circumstances of 10 CFR 50.12, namely:

- Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated [10 CFR 50.12(a)(2)(iii)].
- The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation [10 CFR 50.12(a)(2)(v)].

Industry costs associated with inerting and deinerting the containment during the latter phases of startup testing (i.e., for startup testing programs that last longer than six months after initial criticality) constitute the rationale used for requesting the exemption on the basis of financial hardship.

During the preliminary scoping phase of this project, it was determined that there are only three plants with Mark I and II containments remaining to be started up. Some approximate scoping calculations were done to estimate the magnitude of the potential benefits that could be gained by delaying inerting. One utility that recently started up a BWR with a Mark I containment indicated that without receiving an exemption to the requirement they would have had to reinert containment about nine times before completing the startup testing program. With a containment volume of approximately 300,000 cubic feet, a minimum of 2.7 million cubic feet of nitrogen would have been consumed. The total cost of the nitrogen alone (at $0.80/100 cubic feet) would have been $21,600/plant. Additional costs of time to inert and deinert, the costs of industrial safety procedures to ensure complete deinerting prior to personnel entry, etc., could add an additional $10,000/plant to the cost of the startup operations. For the three BWRs with inerted containments remaining to be started, the total cost savings associated with not inerting the containment within six months after initial criticality would be about $94,800. Although these costs are not insignificant, the primary burden resulting from the regulation is the potential for delaying the plant startup date. Even small delays can create substantial economic penalties.

Because these utilities have elected to request an exemption to the inerting requirement rather than incur the costs and delays of inerting containment during the latter part of the startup testing program, the actual costs that plant owners might save from revising the requirement for inerting these containments within six months after initial criticality would be limited to those costs associated with obtaining an exemption. In other words, the exemptions are already achieving the benefits of delaying inerting until the completion of startup testing. Effectively, the choice is between delaying inerting via exemptions and delaying inerting by changing the requirement. A utility spends and estimated total of one staff-month preparing and negotiating the exemption request with NRC staff. Elimination of the need to request an exemption, therefore, would save the utilities about $8,300/plant. With three BWRs remaining to be started, the total industry savings would equal $24,900. Because the overall benefit to the industry is thus fairly small, this revision to the rule was not evaluated further in this study. However, two other factors should be noted. First, NRC costs to revise the regulation to allow for the completion of startup testing prior to inerting the containment could be quite small if this revision were coordinated with other, more major changes in the regulation (e.g., in connection with large, dry containments, see Section 4.1.1). Another factor is the timeliness of changing the rule. A rulemaking could take several years, which might be too late for the three remaining plants. In this case, the utilities would still have to apply for exemptions.

4.2.3 Recombiners in Large, Dry Containments

As discussed in Section 4.1.1, all containments are required to have the capability of processing (with recombiners) an amount of hydrogen equivalent to the amount that would be generated in a metal-water reaction involving up to 5% of the cladding volume surrounding the active fuel region; the cladding surrounding the plenum volume is excluded. The utilities surveyed in the earlier phase of this project suggested complete elimination of the recombiner requirement because the costs associated with the recombiners are large and because the majority of the risk to the public from reactor accidents results from accident sequences involving metal-water reactions that greatly exceed the required capacity of the recombiners. The utilities identified the following costs: the high initial cost of installing the recombiners, the costs of new containment penetrations, if needed for external recombiners, and the costs of maintaining the equipment once it was installed.

The NRC compiled data on the cost of recombiners in PWR containments in the 1976 version of Regulatory Guide 1.110 (NRC 1976a). The direct costs associated with installing a single recombiner unit in a PWR were estimated to be $566,200 in 1975 dollars, or $1.1 million in 1986 dollars. Annual operation and maintenance costs, in 1986 dollars, were estimated to be $27,400/recombiner unit. Elimination of the requirement for built-in recombiners would save existing plants $509,000/plant (assuming 2 recombiners/plant, an average remaining life of 27.7 years, and a discount rate of 10%) in operating and maintenance costs. New plants would save the direct costs of recombiner purchase and installation, in addition to the annual operation and maintenance cost savings, or about $1.6 million/plant. A smaller savings would result from the elimination of the requirement to maintain the capability to install external recombiners after the initiation of an accident. The smaller savings would result from the reduced capital costs (due to shared recombiners between several sites) and elimination of the maintenance costs.

In December 1984, NRC staff recommended in SECY-83-357B (NRC 1984d) that rulemaking with regard to hydrogen control for LWRs with large, dry containments be deferred due to the greater inherent capability of these containments to accommodate large quantities of hydrogen. The NRC is conducting several programs to address this issue: experimental work at the Nevada Test Site, evaluation of hydrogen burning during the TMI-2 accident, and hydrogen burn experiments at Sandia National Laboratory. These and other programs are being conducted to support a final recommendation on whether safe shutdown equipment and containment integrity are likely to survive an accident involving a hydrogen burn.

In the preliminary phase of this study, the scope of other NRC programs addressing hydrogen control issues in large, dry containments was reviewed. It was determined that this study would not evaluate recombiner requirements in large, dry containments due to the potential impact resulting from these other studies; the means for controlling larger concentrations of hydrogen may be effective in the control of low concentrations of hydrogen. Because of this, no further analysis of this aspect of hydrogen control requirements was undertaken in this study.

4.3 RISK IMPACTS

The risk to the public from a reactor accident involving a hydrogen reaction in the containment building is due to the potential damage to containment and safety-related equipment that could be caused by the reaction and the subsequent release of radioactivity to the environment. Traditionally, PRAs have assumed that containment failure is highly probable if a hydrogen explosion occurs. During the TMI-2 accident, which involved about a 45% metal-water reaction, a mixture of about 7 to 8% hydrogen reacted with the oxygen in containment to produce a deflagration. Containment was not breached; damage inside containment was essentially limited to plastics and other low melting point materials (i.e., telephone cases and the crane operator's seat were damaged).

As discussed in Section 4.2, the focus of this study is the requirement for recombiner capability in the inerted containments of the BWR Mark I and II plants. The evaluation of the risks associated with eliminating the requirement is based primarily on the qualitative arguments presented by Northeast Utilities in their Combustible Gas Control Evaluation Report to NRC on hydrogen control issues in their Millstone Unit No. 1, a BWR with a Mark I containment (Northeast Utilities 1982).

There is significant work currently under way that affects the subject of this study, particularly the rebaselining of risks associated with the source term research. Small portions of this work are beginning to be published; however, the results are too recent to be included here. This new work may have a significant effect on quantitative risk calculations regarding combustible gas control. Nevertheless, a scoping quantitative evaluation of the risks associated with removal of the recombiners was attempted in this study using the Limerick PRA (Philadelphia Electric Company 1982). Philadelphia Electric

Company's Limerick Units 1 and 2 are BWR/4s with Mark II containments. This quantitative evaluation serves as supportive evidence to the qualitative discussion.

4.3.1 Qualitative Evaluation of Recombiners in BWR Mark I and II Containments

The Combustible Gas Control Evaluation Report submitted to NRC by Northeast Utilities (NU) for its Millstone I plant lays out in detail the reasons that NU believes recombiners are not required in the Millstone I containment to protect the public from the risk of reactor accidents involving containment failures initiated by hydrogen reactions (Northeast Utilities 1982). NU stated, and the NRC staff evaluation agreed, that:

- Preinerting with excess nitrogen is so effective that hydrogen generation via the metal-water reaction alone cannot yield flammable mixtures unless an additional source of oxygen is present.

- With the elimination of oxygen inleakage through the main steam isolation valve and safety-related valve control air lines, radiolysis is the only credible source of oxygen present in the post-accident containment environment.

- Radiolytic production of oxygen is limited by natural phenomena to values substantially below those predicted in Regulatory Guide 1.7 (NRC 1978a) due to the fact that boiling in the core region ceases within a relatively short time and liquid phase recombination of hydrogen and oxygen in a radiation field is significant.

- Excess hydrogen gas from large metal-water reactions causes a recombination of hydrogen and oxygen that consequently reduces the amount of oxygen produced in the radiolytic decomposition of water.

- Hydrogen over-pressure of the containment, in concentrations below the flammability limit, ensures dissolved hydrogen concentrations in the reactor coolant that act to stabilize radiolysis. As a result, the use of recombiners that reduce the over-pressure of hydrogen and oxygen on the reactor coolant actually decreases the solubility of hydrogen and oxygen in the reactor coolant. This process allows more hydrogen and oxygen to be released from the coolant into the containment.

- The hydrogen gas added to the containment atmosphere from the accident is a diluent of the relative oxygen concentration in containment (containment oxygen concentrations are limited to about 4% by plant technical specifications).

A key link in this argument is the elimination of oxygen inleakage. In order to justify the removal of the requirement for recombiners in BWR Mark I and II containments, assurance must be provided that the plants are capable of eliminating the ingress of oxygen into containment. Oxygen inleakage is effectively limited by controlling two potential paths of oxygen into the containment:

- ingress of oxygen via leakage from the containment instrument air system
- inleakage via normal containment leakage paths.

These inleakage paths can be eliminated by: 1) converting the instrument air system in containment to one that uses inert gas, and 2) maintaining a positive containment pressure.

Typically, instrument air systems that have been converted to nitrogen include an air compressor back up in the event the nitrogen supply fails. Because the backup air compressor uses outside air containing oxygen, unless the system is isolated from containment following an accident, oxygen could be introduced into containment if the nitrogen system fails. The containment isolation valves in the instrument air system, shown in Figure 4.1, isolate the instrument air system from containment upon receipt of a containment isolation signal. The figure shows two isolation valves in series: a motor-operated globe valve outside containment that has position indication and manual control in the control room should the automatic closure feature fail, and a check valve located inside the drywell. These containment isolation valves preclude the nitrogen system backup from charging oxygen into containment.

With respect to the inleakage of oxygen into containment, the generation of hydrogen and the input of heat energy into the containment atmosphere initially pressurize the containment to a positive value, thereby preventing the inleakage of oxygen into containment through normal leakage paths.

Because the hydrogen gas generated in the accident is mixed with noble gases and other accident products, the system removing gases from the containment must be capable of handling the hydrogen without causing combustion or detonation that might compromise the system and allow radioactivity to be released. Gases are removed from the containment using the purge/repressurization system, shown in Figure 4.2, and processed through the standby gas treatment system to remove radionuclides. The purge/repressurization system maintains a positive containment pressure that forces leakage from the dry well into the reactor building where it is processed through the standby gas treatment system. Most standby gas treatment systems use recombiners that process hydrogen gas to remove the potential for combustion.

4.3.2 PRA Perspective on Recombiners in BWR Mark I and II Containments

To supplement the qualitative discussion of the previous section, several plant-specific PRAs modeling Mark I and II containment performance during reactor accidents were reviewed to assess the public risk due to hydrogen burning and explosions.

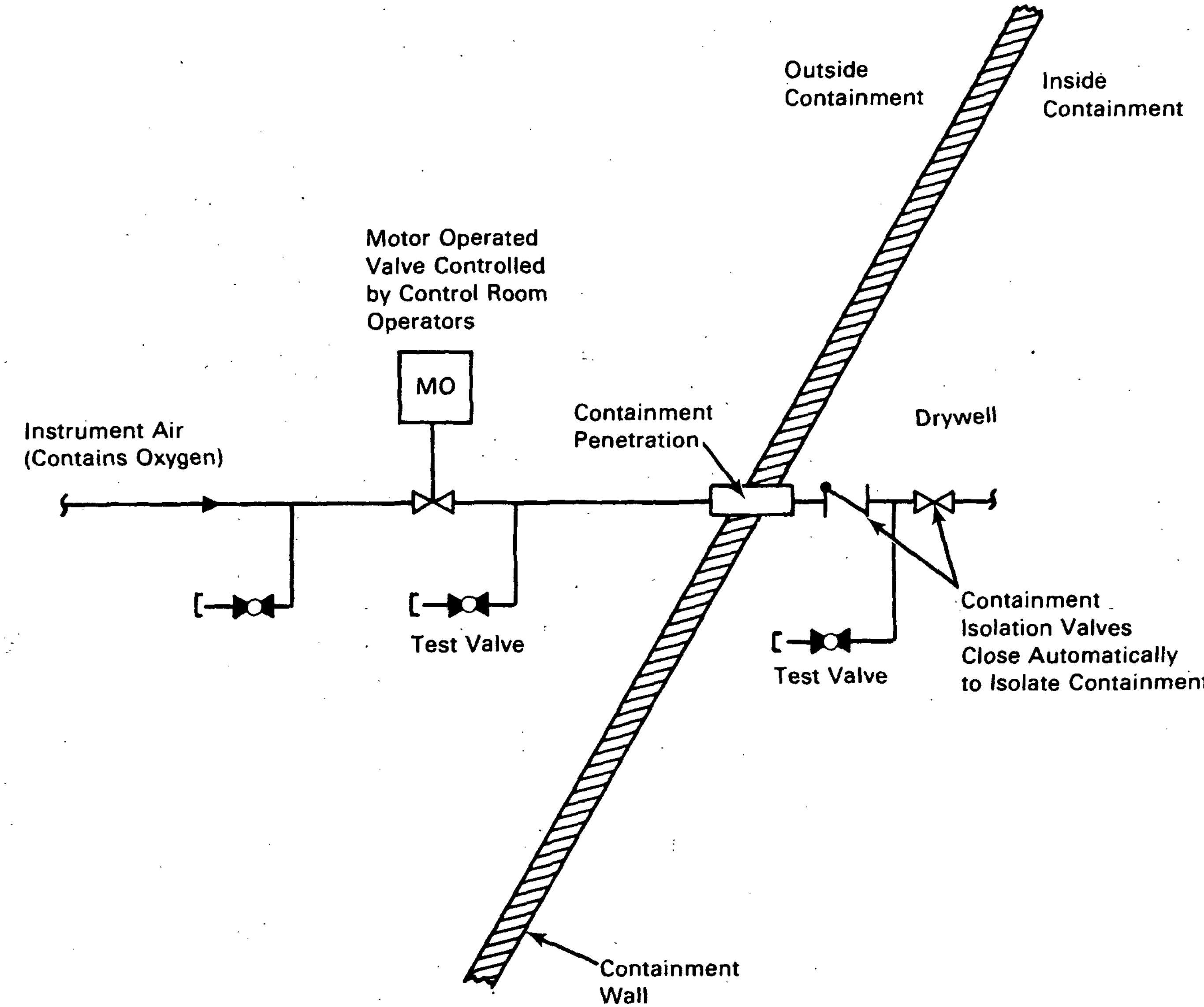

FIGURE 4.1. Containment Instrument Air System Isolation Features

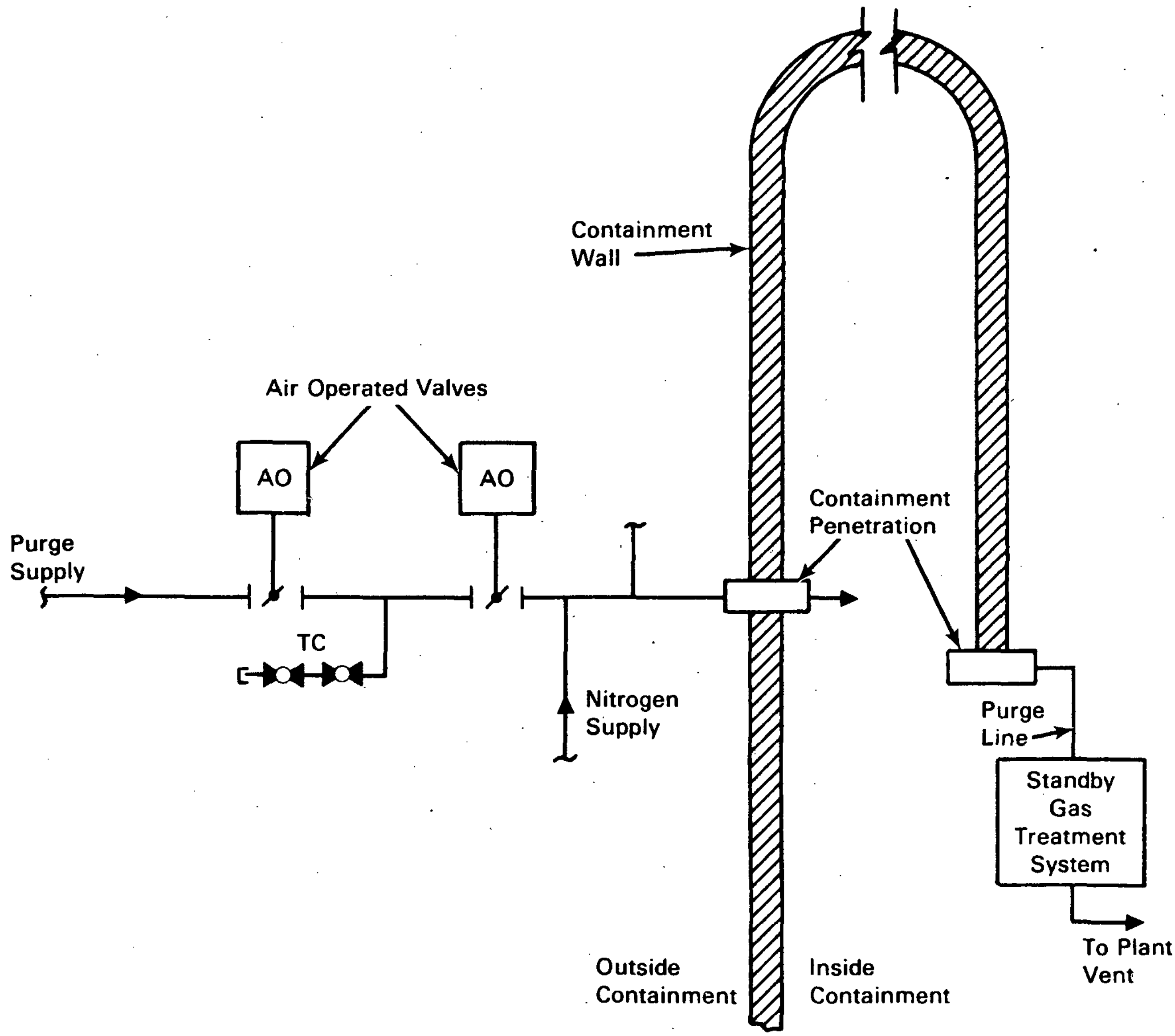

FIGURE 4.2. Purge/Repressurization System

Of all the Mark I and II PRAs reviewed, only the PRA for Limerick Units 1 and 2 addressed containment failure due to hydrogen detonation or burn. According to the Limerick PRA, the possibility of a hydrogen detonation appears quite remote for the inerted containment. However, hydrogen combustion is of concern if any of the following conditions exist:

- an accident occurs during a period when containment is deinerted
- a containment inerting system failure goes undetected and sufficient oxygen accumulates in containment to allow possible combustion during a core melt

- subsequent to core melt, a containment failure occurs that would result in oxygen inflow into the primary containment

- oxygen generated by radiolysis accumulates slowly to form a combustible mixture.

The Limerick PRA defines two levels of hydrogen reaction that might occur if one of the above conditions were met: hydrogen combustion and hydrogen detonation. The PRA conservatively assumes the conditional probability of containment failure induced by hydrogen burning, given a core melt, to be 0.01. Of these failures, the event tree indicates that 90% are due to hydrogen combustion, with the remaining 10% resulting from hydrogen detonation. With the estimated core-melt frequency of 3.0E-05 per year for Limerick (Joksimovich 1984), the contribution of hydrogen events to this core melt frequency is 1%, or 3.0E-07 per year.

To evaluate the effectiveness of recombiners in reducing public risk, the effects of recombiners on each of the conditions noted in the Limerick PRA that could lead to hydrogen combustion/detonation were evaluated. In the first condition (i.e., an accident occurs during a period when the containment is deinerted), the PRA states that "any reduction of the hydrogen concentration by means of the hydrogen recombiners was not assumed due to the large amounts of hydrogen released during a core melt and the relatively small capacity of the recombiners." Because the recombiners are ineffective in reducing the concentrations of combustible gas mixtures in this instance, no significant reductions in public risk are derived from recombiners during accidents that may occur during startup as the containment is being inerted or during shutdown as the inert atmosphere is being removed.

In the event that a containment inerting system failure goes undetected, allowing sufficient oxygen to accumulate so that the containment is essentially deinerted, the recombiners would be of no benefit if a core melt occurred for the same reason as the previous case, i.e., the small capacity of the recombiners cannot keep up with the large amounts of hydrogen that may be generated. As a result, the elimination of recombiners from the containment would have a negligible effect on public risk due to core melt accidents occurring after the containment inerting system has failed and the containment has become deinerted.

Based on the NU report, the possibility of oxygen inflow into the containment as a result of containment isolation system failures can be eliminated from further consideration. Conversion of the containment instrument air system to an inert gas system in conjunction with the system isolation valves reduces the probability of oxygen inflow through this pathway to negligible levels. Because containment pressure will be well above atmospheric pressure following an accident that generates hydrogen, any other failures in the containment isolation system will result in outflow of the containment atmosphere rather than inflow of outside air containing oxygen. Therefore, the inflow of oxygen into containment following a severe accident does not require further consideration.

Finally, the PRA states that the last condition (i.e., accumulation of hydrogen through radiolysis) could only occur if the hydrogen recombiners were inoperable. On the basis of the NU report discussed in Section 4.3, even with the recombiners inoperable, natural phenomena would control the buildup of combustible gas mixtures. Therefore, accumulation of hydrogen and oxygen as a result of radiolysis can be eliminated from further consideration.

Based on the above discussion of the Limerick PRA and the NU report, it is concluded that the risks due to hydrogen combustion/detonation in inerted BWR containments are primarily due to operation of the reactor with the containment only partially inerted. This could occur during plant startup and prior to plant shutdown, or by failure of the containment inerting system. In these circumstances, elimination of the recombiner capability requirement would have a negligible effect on the level of public risk because of the relatively small capacity of the recombiners and the large amounts of hydrogen that could be generated in a degraded core accident.

4.4 COST IMPACTS

The analysis of the costs associated with the elimination of the recombiner requirement in inerted BWR Mark I and II containments is addressed in this section. This analysis used the methodology of the Handbook for Value-Impact Assessment (Hauberlin 1983) modified for this specific application. In most plants, the recombiner systems have already been installed. Therefore, the capital cost of purchasing and installing the systems is a sunk cost and has no effect on the cost-benefit results for operating or near-term operating plants. The costs of recombiner systems are given in this section, however, to indicate the savings to new plants if the recombiner requirement were eliminated.

4.4.1 Industry Costs

Industry costs are those costs incurred by the utility in eliminating the recombiner requirements and the associated changes in operating and maintenance costs compared to the present costs. While initial recombiner purchase and installation costs are considered sunk costs for plants with recombiners already installed, a new plant would benefit by eliminating the recombiner requirement. Estimates of purchase and installation costs for a single, catalytic-type recombiner unit are about $800,000 in 1975 dollars (NRC 1976b), or $1.5 million in 1986 dollars, accounting for inflation. This is $3 million/plant, assuming 2 recombiner units. The estimate accounts for the purchase of the process equipment, building space occupied by the system and its ancillaries, associated piping systems, instrument and control systems, electrical service installation, and system spare parts. The overall direct cost of about $1.5 million for each recombiner unit was corroborated by one utility that spent roughly $2 million to install two recombiner units in its BWR plant.

For plants with recombiner systems already installed, eliminating the recombiner requirement would require each utility to submit an amendment request to the NRC for each affected plant to remove the recombiner requirements from the technical specifications. An uncomplicated amendment request that results in a "no significant hazards" finding (i.e., does not require public hearings) is estimated (NRC 1986) to take 8 staff-weeks of utility personnel time to prepare and receive NRC approval. This is about $16,000/plant.

Estimates for annual maintenance and operation costs of the recombiner units are $23,300 per year per recombiner unit (NRC 1976) in 1975 dollars, or about $45,000 in 1986 dollars. These estimates include labor, supervision, overhead, materials, and consumables (e.g., catalysts).

Actual experience in annual operations and maintenance costs of one utility were reviewed. The utility performs annual inspection and maintenance on each recombiner unit during the annual refueling outage. The level of effort estimated for this work is 2 person-months per recombiner unit, or about $33,600 per plant per year. Additionally, the utility performs a semiannual surveillance test of the recombiners that consists of calibrations, channel checks, etc. This semiannual surveillance testing requires 2 person-weeks of effort every 6 months on each recombiner unit, amounting to an annual cost of $16,000 per plant per year. Total annual operation and maintenance costs for 2 recombiner units at a plant are $49,600/plant, or a total of $460,600/plant over the remaining life of the plant (assumed average remaining life for a BWR is 27 years and a 10% discount rate).

To summarize, the annual cost savings for eliminating the recombiner requirement in inerted BWR containments would be $49,600/year for plants with 2 recombiner units already installed. For new plants, the potential savings, including the purchase and installation costs of about $3 million for 2 recombiner units, plus the annual operations and maintenance savings amounting to $485,000 over a plant lifetime of 40 years, would be about $3.5 million in 1986 dollars.

4.4.2 NRC Costs

The costs to the NRC of eliminating the requirements for recombiners in inerted BWR containments can be divided into two areas: the costs of implementing a rule change, and the costs (or savings) of NRC operations with the changed rule compared with current operating costs.

NRC implementation costs are the costs associated with developing NRC staff positions, conducting reviews, selecting the regulatory alternative, preparing the Notice of Proposed Rulemaking, responding to the comments on the notice, preparing a Federal Register Notice announcing the final rule, providing interpretations for licensees as needed, and processing license amendments to change each plant's technical specifications.

Implementation costs for changing the regulation are estimated to require about 1 year of NRC staff time, or about $100,000; i.e., 6 staff-months for the Notice of Proposed Rulemaking and 6 staff-months for the final rule. After implementation of the final rule, each licensee would submit a license amendment request to implement the change into each plant's technical specifications. The NRC staff time required to process license amendments is estimated to be a one-time cost of 3 to 6 staff weeks per type of amendment request (NRC 1986). Therefore, NRC staff would spend $13,000 to $25,000, depending on the difficulty of the issue. Because of the sensitivities regarding hydrogen control requirements, the actual expenditures for this issue are expected to be on the high end of the range. New plants would not require license amendments if the regulation were changed before the licensing process begins. The implementation costs are estimated to be about $125,000, considering that NRC would spend $100,000 preparing, publishing and interpreting the rule, plus the one-time license amendment cost of $25,000. Both of these costs are one-time costs incurred during the implementation of the regulatory alternative. The cost of publishing notices may be further reduced if NRC coordinates this change with those anticipated for the large, dry PWR containments.

The changes in NRC operating costs arise from the differences in effort that the NRC must devote to inspection, review, oversight, investigation, or enforcement activities as a result of implementing the alternative. In this case, the effort that NRC might apply to inspection of maintenance and surveillance conducted by the licensees on the recombiner systems would be saved if the recombiners were eliminated.

Reduced NRC operating costs are also expected to result in the same areas in which the licensees save operation and maintenance costs; i.e., reduced inspections, maintenance, and surveillance on the recombiners frees the NRC Resident Inspectors from observing, tracking and monitoring this work. While the utilities have been spending 2 person-months per year conducting inspections and maintenance on each recombiner unit, NRC oversight of this activity is estimated to be on the order of 3 to 5 days per year per recombiner unit. Additionally, NRC involvement in the semiannual surveillance tests of the recombiners is estimated to be 1 to 2 days per recombiner. Overall, the savings in NRC time would amount to about 10 to 18 days annually for a plant with 2 recombiner units. This is a savings of $4000 to $7200 per plant year.

4.4.3 Cost Summary

The costs and savings to the industry and NRC are summarized for existing and new plants in Table 4.1. Because the exact number of BWRs with Mark I and II containments that have received an exemption to the recombiner requirement is not known, the total industry costs are based on a total of 37 BWR plants (Millstone I is excluded from this total). If NRC acts to eliminate the requirement for recombiners in inerted BWR Mark I and II containments, the savings to the utilities operating plants would be about $16.4 million for all plants. If new plants are started, the utilities will save an additional $3.5 million/plant by not having to install recombiners.

As discussed in Section 4.4.2, NRC costs are composed of a one-time cost of $125,000 to implement the rule change with an annual cost savings averaging $5600 per plant year, or $1.9 million for the remaining life of the BWRs with Mark I and II containments, assuming a 10% discount rate.

TABLE 4.1. Total Industry and NRC Costs and Savings from Eliminating Recombiner Requirement

	Costs[a] for Existing Plants, $1000	Costs[a] for New Plants, $1000
Industry Implementation	600.0	-3,000.0/plant
Industry Operation	-17,000.0	-485.0/plant
Industry Totals	-16,400.0	-3,485.0/plant
NRC Implementation	125.0	100.0
NRC Operation	-1,900.0	-55.0/plant
NRC Totals	-1,775.0	100.0 plus $55.00/savings per plant
Industry and NRC Net Effect	-18,175.0	100.0 plus $3,540 savings per plant

(a) Negative numbers indicate cost savings.

The costs associated with eliminating the recombiner capability requirements in existing plants are small compared with the benefits for both the utilities and the NRC. These cost savings are a result of the annual cost savings that can be achieved by not having to maintain, inspect and conduct surveillance tests on recombiners. The industry cost savings for new plants are higher than the savings due to avoided annual inspection, maintenance, and surveillance because of saving the direct costs of purchasing and installing the recombiners in the plant.

The overall benefit to industry and the NRC for the existing plants is shown in Table 4.1 to be a cost savings of over $18 million.

4.5 SUMMARY

The results of this study are summarized in Table 4.2. As discussed in Section 4.3, the risk increase due to eliminating recombiners in BWR Mark I and II containments could be marginal primarily because of the effectiveness of the inert atmosphere in limiting the potential for hydrogen combustion events, the natural phenomena that control and limit the production of combustible mixtures, and the small capacity of the recombiners relative to the quantity of hydrogen that would be generated in a degraded core accident. Review of a PRA for a BWR/4 plant with a Mark II containment suggests that the risk due to hydrogen combustion/detonation results from plant operation with the containment only partially inerted, not from any contribution of the recombiners to the reduction of the concentration of the combustible gases. While quantitative risk calculations may be affected by the rebaselining of risks associated

TABLE 4.2. Summary of Risks and Costs of Eliminating Recombiner Capability Requirements from Mark I and II Containments

Public Risk Increase	Marginal
Industry Cost Savings	$16.4 million
NRC Cost Savings	$1.8 million

with the source term research, the results of this study confirm, at least qualitatively, that the risk of eliminating the requirements for recombiners in inerted BWR Mark I and II containments is marginal.

The cost estimates presented in Table 4.2 are based on a plant population of 37 (i.e., all BWRs with Mark I and II containments, except Millstone I), an average remaining plant lifetime of 27 years, and a discount rate of 10%. While these costs are not extremely large compared with the initial cost of a plant, these costs appear to be unnecessary given the marginal risk reduction associated with the recombiners in inerted BWR containments.

The other alternatives originally identified for inclusion in this study were not pursued beyond the scoping study because of marginal benefits in the case of the time to inert BWR Mark I and II containments following initial criticality, and the possible impacts of the results of the NRC degraded core research programs on the control of low concentrations of hydrogen in large, dry containments.

5.0 IMPREGNATED CHARCOAL FILTERS

The NRC regulations currently require operating nuclear power plants to have filtered ventilation systems serving the structures housing various parts of the plant. In general, these filtered systems serve to restrict uncontrolled release of radioactive gases and particulates from process areas, and to protect plant operations personnel who must remain on station during an emergency from the effects of radioactive releases arising from that emergency.

The filtered ventilation systems now in use employ both high-efficiency particulate (HEPA) and activated, impregnated charcoal filter media to remove radioactive contaminants. The requirement for use of an adsorbing medium such as charcoal is based on the large inventories of radioiodine isotopes in reactor cores during and after power operation. The radioiodines, if released to the atmosphere and inhaled or ingested, would be concentrated in the thyroid gland, possibly causing radiation exposure to that organ in excess of allowable limits. Radioiodines that are chemically bound to other elements in form of particulate solids can be removed by HEPA filters. Free (elemental iodine) and organic iodine compounds formed in reactions with plant materials are not appreciably removed by conventional fiber filter media; they can, however, be largely removed by the adsorptive and isotopic exchange processes on the surfaces of activated charcoal granules.

Recent work on reactor accident source term definition has raised doubts about the validity of past assumptions regarding the chemical and physical nature of the fission product mixture that would be released during a reactor accident. This led to reassessment of whether, in view of the new source term research results, a positive dose benefit from having forced, filtered exhaust ventilation with charcoal filters could be shown. However, when the evaluation of the dose benefit issue was outlined, it was determined that the revised source term characteristics were not sufficiently defined to support regulatory analysis. Therefore, the principal question of charcoal filters versus no charcoal filters was not treatable within the time constraints of this project. A secondary issue, that of the filter medium to be used for iodine removal, was evaluated and is documented in this report.

5.1 CURRENT REGULATORY REQUIREMENTS

General Design Criteria 41, 42, and 43 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," require that containment atmosphere cleanup systems be provided as necessary to reduce the amount of radioactive material released to the environment following a postulated design basis accident (DBA) (CFR 1986). The criteria also require that these systems be designed to permit appropriate periodic inspection and testing to ensure their integrity, capability, and operability.

General Design Criterion 61 of Appendix A to Part 50 requires that fuel storage and handling systems, radioactive waste systems, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions and that they be designed with appropriate containment, confinement, and filtering systems. General Design Criterion 19 requires that adequate radiation protection be provided to permit access to and occupancy of the control room under accident conditions and for the duration of the accident without personnel radiation exposures in excess of 5 rems to the whole body.

Regulatory Guide 1.52 presents methods acceptable to the NRC staff for implementing the regulations with regard to design, testing, and maintenance criteria for air filtration and adsorption units of engineered-safety-feature (ESF) atmosphere cleanup systems in light-water-cooled nuclear power plants (NRC 1978b). The guide applies only to postaccident engineered-safety feature atmosphere cleanup systems designed to mitigate the consequences of postulated accidents.

The Regulatory Guide invokes ANSI Standards N509-1976 and N510-1975 and other references for standards and criteria. The Guide sets forth specific guidance in the following six areas:

1. Environmental Design Criteria
2. System Design Criteria
3. Component Design Criteria and Qualification Testing
4. Maintenance
5. In-Place Testing Criteria
6. Laboratory Testing Criteria for Activated Carbon.

The guidance in Regulatory Guide 1.52 applies to all ESF atmosphere cleanup systems, regardless of whether they are recirculating or exhausting (once-through) systems. No distinction is drawn between the two types of systems regarding the need for adsorption units to remove gaseous and elemental iodines. On the subject of adsorber media, the Regulatory Guide states that the adsorber section "may contain any adsorbent material demonstrated to remove gaseous iodine (elemental iodine and organic iodides) from air at the required efficiency. Since impregnated activated carbon is commonly used, only this adsorbent is discussed in this guide."

Thus, the requirements of the General Design Criteria have resulted in:

1. filtered atmosphere cleanup systems on containments, fuel storage/ handling buildings and other buildings that house equipment or systems containing radioactivity

2. control Room ventilation systems that will protect operators from the effects of a design basis LOCA.

The basic principles of ventilation system design have, in general, resulted in the wide use of exhausting (or once-through) ventilation systems for structures other than containments. Cost and maintenance considerations have resulted in activated impregnated charcoal being the overwhelming choice as an adsorber material.

5.2 ALTERNATIVE TO CURRENT REQUIREMENTS

5.2.1 Scoping Study

The resolution of the ventilation issue requires quantification of cost and risk implications of using activated, impregnated charcoal filtration systems in venting nuclear plant buildings. Because charcoal filters are used in a number of different nuclear plant ventilation systems, a preliminary scoping study was conducted to identify a set of representative systems to use as models for quantitative assessment of dose consequences and costs. Systems using charcoal filters in typical BWR and PWR plant designs were evaluated. Selection of specific systems was based on three factors:

1. subjective estimates of value-impact attributes, i.e., whether a change in system operating mode or filter media is likely to have a quantifiable impact on occupational exposure, offsite releases, accident management or costs to the licensee or NRC

2. the ability to analyze the system adequately within the time and funding constraints

3. whether the system was "typical" or representative of a class of such systems in general use at plants of a given type.

Table 5.1 presents a summary of the scoping analysis. Based on this qualitative assessment, three candidate system alternatives were identified for detailed analysis. These were:

1. control room ventilation systems - remove charcoal filters and isolate makeup air in emergencies

2. auxiliary building exhaust - remove charcoal filters, secure ventilation and isolate building to minimize releases

3. fuel handling building exhaust - remove charcoal filters and isolate building to minimize releases.

5.2.2 Selection of Alternatives

The original statement of this issue involved the trade-off of whole body dose from noble gases against the thyroid dose from radioiodines. If the requirement for charcoal-filtered exhaust ventilation systems is based on an expected large iodine component of the radioactive release from any DBA, a substantial reduction in the radioiodine release might mean that the noble gases, exhausted directly to the environment, would dominate the offsite doses. This statement of the issue implied an alternative strategy of confinement. Using this implied strategy, any accident release would simply be contained or confined within the structure to allow maximum time for decay of the radionuclides, rather than exhausting the mixture rapidly to the environment. The source term basis for investigating this alternative strategy, i.e., a much-reduced estimate of gaseous and elemental iodine releases, was not available

TABLE 5.1. Qualitative Assessment of Alternatives

Plant Type	Application	Intended Use	Alternative(s)	Effects of Alternative Versus Status Quo: Occupational Exposure	Offsite Exposure	Operations/ Accident Management	Industry Costs Imp./Op.[(a)]	NRC Costs Imp./Op.[(a)]	Analysis Requirements: Extent of Modeling Effort Required	Adequacy of Approximations and Assumptions	Other Factors
BWR/PWR	CR Ventilation System	Emergency	Eliminate filters and isolate makeup air	?[(b)]	None	?	Low/Positive	Low/?	Low	Good	Impractical for extended accident conditions
PWR	Containment Air Cleanup	Normal Operations	Eliminate system	?	No effect	Reduced	?/?	Low/?	Low	Good	
PWR	Containment Air Cleanup	Normal Operations	Eliminate filters	?	No effect	Reduced	?/?	Low/?	Low	Good	
BWR	Standby Gas Treatment System	Emergency	Isolate system	Higher	?	Much reduced	Low/?	Low/?	High	Fair-poor	
BWR	Standby Gas Treatment System	Emergency	Eliminate filters	Lower	Higher	No effect	Low/Positive	Low/?	Low	Good	Severe negative public relations
PWR	Emergency Gas Treatment System	Emergency	Isolate system	Higher	?	Reduced	Low/?	Low/?	High	Fair-poor	
PWR	Emergency Gas Treatment System	Emergency	Eliminate filters	Lower	Higher	No effect	Low/Positive	Low/?	Low	Good	Public relations
BWR/PWR	Containment Purge Exhaust	Normal Operations	Isolate (no purge)	Higher	Slightly lower	Reduced	Low/Negative	Low/?	High	Poor	ALARA (in plant) considerations
BWR/PWR	Containment Purge Exhaust	Normal Operations	Eliminate filters	Slightly lower	Higher	None	Low/Positive	Low/?	Low	Good	ALARA (offsite) considerations
PWR	Auxiliary Building Exhaust	Emergency	Isolate system	Higher	?	Much reduced	Low/?	Low/?	High	Fair-poor	
PWR	Auxiliary Building Exhaust	Emergency	Eliminate filters	Lower	Higher	No effect	Low/Positive	Low/?	Low	Good	Severe negative public relations
PWR	Fuel Handling Area Exhaust	Emergency	Isolate system	Higher	?	Much reduced	Low/?	Low/?	Moderate-high	Fair-poor	
PWR	Fuel Handling Area Exhaust	Emergency	Eliminate filters	Lower	Higher	No effect	Low/Positive	Low/?	Low	Good	
PWR	Waste Packaging Area Exhaust	Emergency	Isolate system	Higher	?	Much reduced	Low/?	Low/?	Moderate-high	Fair-poor	
PWR	Waste Packaging	Emergency	Eliminate	Lower	Higher	No effect	Low/Positive	Low/?	Low	Good	

(a) Imp./Op. is defined as the industry (or NRC) costs to implement the alternative, and impact (positive or negative on subsequent costs of routine operations.
(b) ? indicates that trend or direction of effect could not be determined without substantial analysis.

during this project. Hence, the consequences of implementing the confinement alternative with the "new" source term could not be quantitatively evaluated.

When it was recognized that quantitative evaluation of the confinement alternative would not be feasible, a second alternative was identified that involves the charcoal filter medium and additives used to enhance iodine removal. The more effective additives are hygroscopic and tend to absorb moisture from the air, which degrades the charcoal's iodine removal efficiency. To prevent this degradation, electric heaters are installed in conjunction with the charcoal beds either to keep the beds continuously warm and dry, or to warm the incoming air stream when the system blower is activated. Warming the incoming air to reduce its relative humidity to 70% or less ensures that the charcoal will retain its iodine adsorption capability.

Therefore, the only alternative evaluated was elimination of hygroscopic impregnation materials from the charcoal filter media, thereby reducing the requirements for electric heating of the charcoal beds and the air steam.

5.3 RISK IMPACT OF NONHYGROSCOPIC ADSORPTION MEDIA

In order to quantify the risk impact from the use of alternative (nonhygroscopic) adsorption media, the difference between the performance of those media and activated impregnated charcoal must be quantified. Two factors preclude this comparison. First, no "nonhygroscopic" iodine adsorption media suitable for standby use in ventilation systems are generally available. Thus, there are no performance data or standards with which to compare the iodine removal performance of impregnated charcoal.

Second, Regulatory Guide 1.52 explicitly states that adsorber media other than impregnated charcoal may be used, so long as the specified efficiency is achieved. Thus, the performance of the adsorber, and hence, the risk from radioiodine releases is limited not by the type of adsorber, but by the performance specification. Therefore, risk is not an issue for this alternative as stated.

5.4 COST IMPACT OF NONHYGROSCOPIC ADSORPTION MEDIA

The alternative evaluated here is the use of nonhygroscopic iodine removal media in place of impregnated activated charcoal, thereby reducing or eliminating the electrical heating requirements for the filter beds.

Impregnated activated charcoal is commonly used because of economic and maintainability factors, e.g., other available adsorption media or systems of comparable efficiency are substantially more costly to install and maintain, or have characteristics which make them unsuitable for standby operation. Since licensees have always had the option of using other materials and technologies to comply with the regulations, the fact that they have not widely done so reflects favorably on the economics of impregnated activated charcoal, even with its electric heating requirements. This is not surprising, considering that the operating cost penalty for a typical adsorption system with 2 kW of installed heaters to maintain bed temperature is only a few hundred dollars per year. The cost of operating another typical system for 10 hours a month, as prescribed in Regulatory Guide 1.52, to dry out the adsorbers and HEPA filters (40 kW of air preheaters, 25 kW blower) would be similar.

To determine if other, less hygroscopic, filter media might be available at competitive cost, industry representatives and literature were surveyed. The survey established several significant facts:

- The activated, impregnated charcoal market is very competitive, with prices decreasing by about 50% over the last 5 to 10 years to around $2 to $3 per pound. Impregnation materials and technology have also improved.

- TEDA- (Tri-ethylene Diamine) impregnated filters are less hygroscopic than those impregnated with iodide salts, and are generally shown to perform better on the most demanding (low-temperature) efficiency tests.

- There is no other filter medium that is in a position to challenge charcoal as the medium of choice for meeting the Regulatory Guide 1.52 guidance. Specifically:

  - silver zeolite is about 20 to 30 times <u>more</u> expensive than activated impregnated charcoal

  - silver zeolite performance is more sensitive to the effects of water than is charcoal

  - nonimpregnated charcoal is less efficient as an iodine adsorber under nearly all operating conditions

  - nonimpregnated charcoal performance is also degraded by high-humidity conditions

  - under some conditions, methyl iodide, which is less readily adsorbed by charcoal, can actually be formed from elemental iodine in nonimpregnated charcoal beds.

- Impregnated, activated charcoal filters are generally viewed by the users as being inexpensive, long-lasting (unless contaminated with organic solvents or similar vapors) and fairly trouble-free.

- The need for continuous heaters to maintain the efficiency of a charcoal filter bed is not generally regarded as expensive (~1% to 2% of the capital cost of a typical filtration system) or as onerous otherwise. Similarly, the need for periodic (monthly) operation of systems with in-line heaters is not a source of apparent concern to licensees, since all plant safety equipment receives periodic surveillance and operability testing. The heater operating costs (electric power, \$200 to \$800 per year per system) are negligible.

- HEPA filters, used in conjunction with charcoal in ESF ventilation systems, are also adversely affected by moisture. Thus, it is generally believed that the heaters installed to maintain charcoal adsorber efficiency have a beneficial effect on HEPA performance and life.

Table 5.2 presents a summary of costs associated with operation of a typical ESF ventilation system that might be affected by the hygroscopic nature of the adsorber medium.

TABLE 5.2. Iodine Adsorber Summary Cost Comparison[a]

	With Activated Impregnated Charcoal	With Nonhygroscopic Adsorber
Installation Cost	\$ 3750/train[b]	Unknown - no good candidate materials
Service Life	3 to 10 years	Unknown - no good candidate materials
Continuous Heaters	\$700/train[c]	0[d]
10 hr/month Run[e]	\$312/train[e]	Unknown[f]
Total Operating Cost	\$1012/train	Unknown[f]

(a) Based on typical BWR Standby Gas Treatment System design with 1500 pounds of charcoal.
(b) Impregnated charcoal at \$2.50/lb.
(c) Based on 2 kW of installed heaters X 8760 hr/yr X \$0.04/kWh.
(d) Assumed ideal adsorber material requiring no standby heating.
(e) 120 hr/year X (40 kW air preheaters + 25 kW Blower) X \$0.04/kWh.
(f) Air preheaters would still be needed in some systems to reduce relative humidity for protection of HEPA filters. Ten hr/month run time might not be needed if adsorber medium is nonhygroscopic.

5.5 DISCUSSION OF THE ROLE OF CHARCOAL-FILTERED VENTILATION IN THE THREE MILE ISLAND ACCIDENT

The effects of the TMI-2 accident were evaluated to illustrate the value of typical charcoal filtered exhaust systems in the PWR auxiliary building and fuel handling building.

5.5.1 Radioiodine Release and Population Thyroid Dose

According to the Rogovin (1980) report, 15 curies of Iodine-131 (^{131}I) were released to the surrounding countryside during the accident. The release was primarily from two plant structures: the auxiliary building and the fuel handling building. The main release came from the fuel handling building, primarily because the charcoal filters for that building had previously been degraded (Rogovin 1980).

Neither the Rogovin nor Kemeny et al. (1979) reports give a value for population thyroid dose. However, the thyroid dose (person-rem) to the surrounding population within a 50-mile radius, as estimated by Pickard, Lowe, & Garrick, Inc. (1979) was:

By Inhalation:	180	person-rem thyroid
By Consumption of Milk:	1,100	person-rem thyroid
TOTAL	1,280	person-rem thyroid.

The calculation was made using an estimated release of 14.1 curies of ^{131}I and 2.6 curies of ^{133}I, which is fairly close to Rogovin's best estimate above.

If all filters had been up to specification, the dose would have been substantially lower. Regulatory Guide 1.52 (NRC 1978b) requires the charcoal used in nuclear reactor iodine filters critical to safety systems to meet ANSI/ASME N509 (Table 5.1) for "each original or replacement batch of impregnated activated carbon used in the adsorber section."

Assuming that all iodine filters had met Regulatory Guide 1.52 specifications for two-inch filters operating outside of primary containment at a relative humidity of 70% (Table 2 of the Regulatory Guide), the assigned decontamination efficiencies would be 95% for elemental iodine and organic iodides. Table II-4 of the Rogovin report gives the iodine activity captured on the filters as 112 curies. Adding this value to the activity released results in approximately 127 curies of iodine presented to the filters. Assuming an efficiency of 95%, 127 x .05 or 6.35 curies would have been released. The factor of decrease would be 6.35/15 = 0.42, which would have produced a population thyroid dose of 1280 x 0.42 or 540 person-rem.

Although the charcoal filters are typically credited with only 95% efficiency for iodine gas and organic iodides other than methyl iodide, the filters may have a higher efficiency in an actual accident situation, as shown in Table II-4 of the Rogovin report. If it is assumed that an actual efficiency of 99.9% for all iodine species could be achieved, then the iodine release would have been 127 x 0.001 = 0.127 curies and thus the release would have decreased to 0.127/15 or 0.0085 times the best estimate value. The resulting population thyroid dose for this release would have been 1280 x 0.0085 or 11 person-rem.

If no charcoal filters had been in place, the only filters available would have been the HEPA filters. These might have removed any cesium iodide present, as well as the iodine adhering to particles, but would have had little effect on the gaseous species. The distribution of the radioiodines in the TMI-2 release between particles and gaseous species is not known, thus no credit can be taken for any iodine removal by the HEPAs. Therefore, all of the 127 curies of ^{131}I challenging the filters are assumed to have been released. This would increase the population thyroid dose by a factor of 127/15 or 8.5, which would have produced a population thyroid dose of 1280 x 8.5 or 11,000 person-rem.

In summary, the population thyroid doses (person-rem) for the four cases are:

Case	^{131}I Thyroid Dose (person-rem)	Filter Efficiency
Optimistic case	11	Efficiency of 99.9%
Filters to specification	540	Efficiency of 95%
Actual best estimate	1,280	Estimate by Pickard, Lowe, and Garrick (1979)
No iodine filters	11,000	No credit for HEPAs, i.e., filter efficiency of 0%

5.5.2 Impact of "Confinement" Alternative

The filtered ventilation systems serving the auxiliary building and fuel handling building were the main gaseous release pathways for radioactive materials. Continued operation of the letdown system transferred primary coolant from the reactor to the components in the auxiliary building. There, due to seal leakage and coolant degassing, pressure buildup in the letdown system components caused gaseous leakage to the auxiliary building, fuel handling building and to the environment. Figure 5.1 illustrates the release pathways. Releases of particulate radioactive material to the environment were negligible because of the two banks of HEPA filters in each vent system.

The operation of the filtered vent systems is estimated to have released a total of 2.37E+06 curies of noble gases (largely ^{133}Xe) to the environment during the release of the previously-discussed 15 curies of radioiodines. The estimates of population whole body dose from this noble gas release range from 300 to 3500 person-rem, the most likely value being about 2000.

The positive and negative consequences that may have resulted from using the "confinement" option during the TMI-2 accident are listed below:

Positive Consequences	Negative Consequences
Reduction of offsite releases (and resultant population dose commitment) through holdup and decay.	Higher airborne activity and direct radiation levels in Unit 2 Auxiliary Building during and after the accident.
	Uncontrolled spread of airborne contamination to other areas of the plant.
	Releases to the environs from multiple, unmonitored, ground-level release points.

These consequences are discussed in detail below.

Reduction of Offsite Release and Population Dose

Rigorous evaluation of the potential population dose reduction from use of the "confinement" option is extremely complex and will not be attempted here. In general terms, however, the best estimates of cumulative population dose (about 2000 person-rem) and maximum offsite individual dose (70 millirem) are both far below the levels at which any health effects would be expected. Less than one additional cancer death in the population within 50 miles of the plant is projected as a result of the release. This is to be compared with the nearly 541,000 cancers from all causes expected in the same population over its remaining lifetime. Thus, the maximum physical health benefit which could possibly be achieved with a different effluent management strategy (confinement) would, necessarily, be trivial. The effects on the mental health of the people living in the region, concluded by the President's commission to be the "major health effect of the accident," (Kemeny et al. 1979) would not have been influenced by use of a different release strategy.

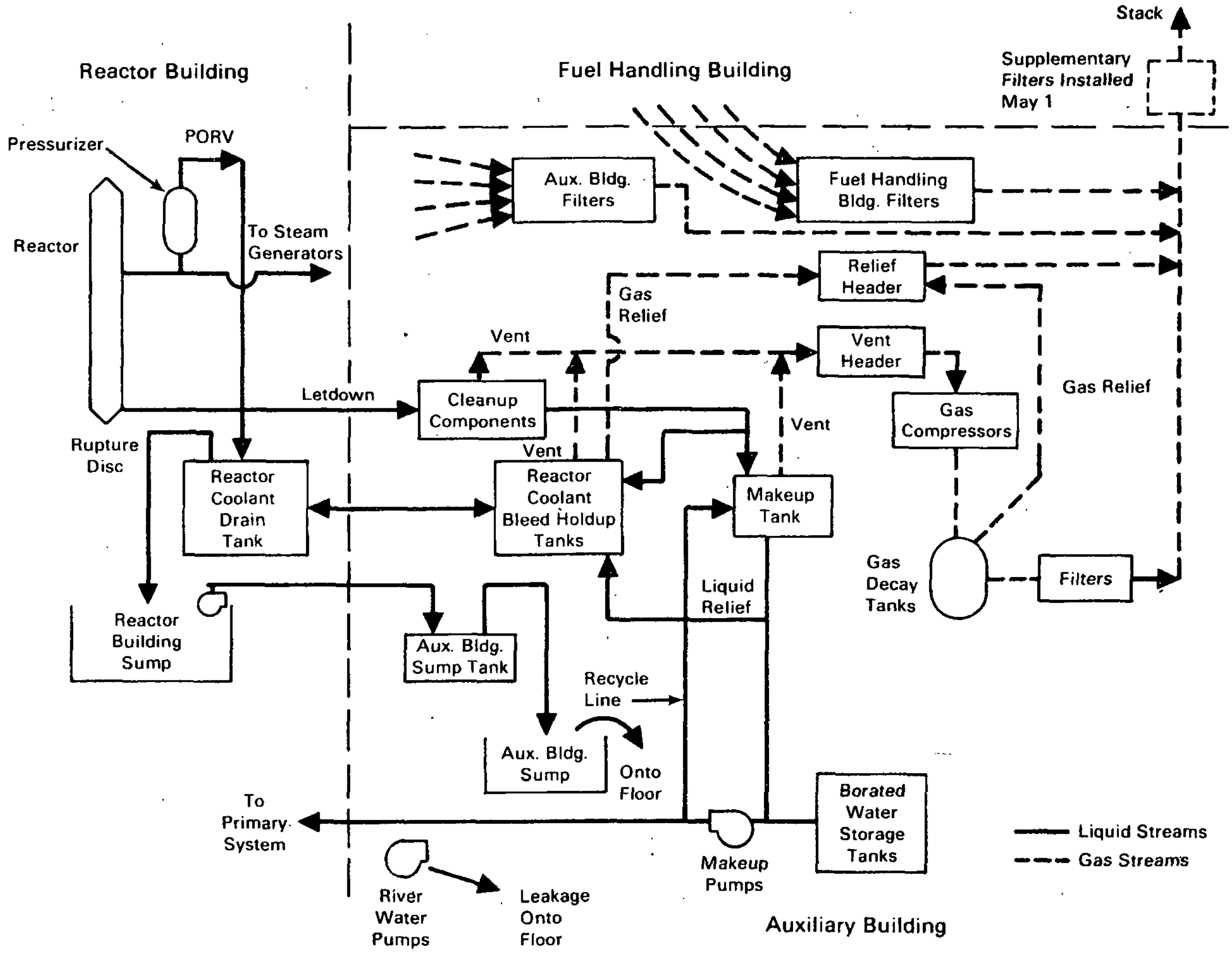

FIGURE 5.1. Release Pathways (Rogovin 1980)

Higher Airborne Activity and Direct Radiation Levels in Unit 2 Auxiliary Building

Shutting down the exhaust fans to minimize the noble gas release was, in fact, attempted several times in the first two days of the event. Securing the fuel handling building and auxiliary building ventilation systems caused exposure rates to increase significantly in the Unit 2 auxiliary building, hampering emergency activities. Personnel entries into the auxiliary building during the first few days of the accident were accomplished with respiratory protection (self-contained and filter-type) and protective clothing. Radiation levels were in the hundreds of mrem/hr range in the hallways with tens to hundreds of rem/hr in and around the rooms housing the letdown system piping and components. Had the ventilation exhaust fans been left off, the following consequences could have been expected:

- Increasing direct radiation in all parts of the auxiliary building due to buildup of the noble gases being released from leaking primary coolant. These increases of 1 to 5 rem/hr would not have been significant in the very hot areas containing letdown components and primary coolant, but would have been very significant in the hallways and other passage areas, resulting in much higher exposures to persons entering for inspection, repair and corrective actions. The already-short personnel stay times would have had to be reduced, and with it, the effectiveness of repair efforts.

- Airborne activity levels in the auxiliary building would have exceeded the levels at which use of filter-type respirators would be authorized. Since the plant personnel were seriously short of self-contained breathing apparatus (SCBA units) and air bottles, emergency workers would likely have received internal radionuclide deposition, or else recovery and survey teams might have been prevented from entering the auxiliary building because of high airborne activity levels.

Uncontrolled Spread of Airborne Contamination to Other Areas of the Plant

When the Unit 2 auxiliary building and fuel handling building exhaust fans were secured, it was noted that the Unit-2 control room airborne activity levels started increasing. Because of the need to ensure habitability of the control room and to keep dose rates in the Unit-2 auxiliary building as low as possible to facilitate emergency activities, the exhaust fans were subsequently kept in operation. Gaseous activity in Unit 1 was also noted to increase significantly whenever Unit 2 Ventilation Systems were turned off. Since the Unit 1 structures were being used as staging areas for re-entry and corrective action in Unit 2, the recovery effort would have been severely hampered if such areas as the Unit 1 Chemistry/Health Physics control point, with its protective clothing supplies, survey instruments and equipment, had been made uninhabitable by high airborne activity.

If the exhaust fans had been left off, airborne activity levels would probably have continued to rise throughout the plant. Because of the early shortage of SCBAs and air bottles, reentry teams were sometimes forced to make do with partially-filled air bottles or filter respirators. If high airborne activity levels had forced moving the reentry staging area to some more distant location, the increase in recovery team transit time and distance would have severely reduced the teams' effectiveness. Similarly, if conditions in the Unit-2 control room had required continued use of respiratory protection, the number of personnel in the control room would have had to be limited and their ability to communicate verbally among themselves, with offsite authorities and with other in-plant staff would have been hampered. The impairment of communications alone could have been disastrous.

Release to the Environment from Multiple, Unmonitored, Ground-Level Release Points

Because of the placement and type of the TMI-2 Vent Stack Monitors, they were of limited use in quantifying the release. Therefore, the benefit of a monitored (i.e., being able to more accurately calculate offsite dose potential) versus unmonitored release was only partly realized. However, by keeping the exhaust fans running, the staff could be assured that the release was coming from a single, known point rather than from many different doors and building penetrations. This made planning the movement of personnel around the site much simpler.

The fact that the vent stack is somewhat elevated may have improved atmospheric dispersion of the release. The enhanced dispersion would have had little effect on offsite doses but significantly aided in planning site access and onsite movement of personnel.

Summary

Use of the "confinement" option was attempted during the TMI-2 accident and rejected. The overriding concern was to keep the rest of the plant, particularly the Unit-2 control room, habitable in order to carry out actions necessary to mitigate the accident. Any offsite population dose that could have been averted by staying with the "confinement" option would have been at the expense of higher exposures to plant staff and reduced accident management capability. The ability to ventilate the auxiliary building, exhausting through charcoal filters, made it possible to manage the accident response much more effectively while minimizing the offsite dose due to radioiodine.

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APPENDIX A

ASSESSMENT OF PASS IN AN EMERGENCY

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ASSESSMENT OF PASS IN AN EMERGENCY

Three pertinent phases of a severe core damage accident were defined to categorize PASS uses and benefits:

- Accident Management. This phase of an accident is defined as the period of time immediately following initiation of a transient during which reactor operators need information on the status of the reactor in order to take actions to arrest and mitigate the consequences of the transient. This, in most accident scenarios, is a relatively short period of time, lasting from several hours to about 2 days, during which the reactor is placed in a shutdown condition.

- Emergency Response. This phase of an accident often occurs concurrently with the accident management phase; however, the purpose of actions and decisions made is primarily the protection of the public through the communication of plant status and recommended protective actions to the public from an emergency response team consisting of utility staff and local governmental officials. Information required by decisionmakers in this phase involves details about the potential for and estimated concentration of radioactive releases, the direction such a release is anticipated to travel, and the estimated doses that members of the public might receive as a result of exposure to a radioactive release.

- Accident Recovery. This phase of an accident refers to the actions required to deal with the damaged reactor and the contamination of plant and public property resulting from an accident. Information that is useful in this phase is the degree of core damage that may have occurred, the amount of radioactive material contaminating the containment and other plant buildings, and the potential damage done to the reactor components as a result of the accident. This information is not needed during the transient. Using the example of TMI-2, it was many days before the reactor was brought to a cold shutdown condition when passive cooling mechanisms were sufficient to remove heat. It is at this point that information is needed regarding plant and reactor damage for recovery.

In each of these three areas, the ability of PASS to affect the overall risk to the public was evaluated. Each of these areas is discussed in detail in the following sections.

A.1 ACCIDENT MANAGEMENT

PASS provides information on the radionuclide composition of the reactor coolant and containment atmosphere; dissolved hydrogen and, optionally, oxygen levels in reactor coolant; chloride and boron concentrations in reactor coolant; and the pH of the reactor coolant. Of this information, only a knowledge of boron concentration in reactor coolant and hydrogen concentration in the containment atmosphere is considered useful in responding to a transient in the accident management phase.

Upon detection of indicators that the plant is experiencing a serious transient, the reactor is immediately isolated and engineered safety feature systems are initiated. Part of this action is the automatic isolation of reactor coolant system letdown, which is intended to prevent the loss of reactor coolant. Since the source of primary coolant for normal boron analysis is in the letdown system, isolation of this system prevents the operators from obtaining samples of the primary coolant for boron analyses throughout the transient. The level of boron in the reactor coolant system is essential to prevent recriticality in a degraded core accident.

Indication of the hydrogen concentrations in the containment atmosphere would provide plant operators with information regarding potential combustible mixtures in containment. This would allow operators to take corrective actions that could potentially prevent damage to the containment. The primary method for determining hydrogen concentrations in containment atmosphere, however, is the safety-grade containment hydrogen monitor specified by a separate requirement in NUREG-0737, Item II.F.1(6) (NRC 1980a). This monitor provides a real-time indication of hydrogen concentration in the containment. Hydrogen analysis results obtained from PASS may be delayed 2 to 3 hours if a grab sampling system is used because of the time required for sample collection and analysis. Therefore, PASS is used mainly as a backup for obtaining containment hydrogen levels.

Other PASS results (i.e., radionuclide composition of reactor coolant and containment atmosphere; dissolved hydrogen and, optionally, oxygen levels in reactor coolant; chloride concentrations in reactor coolant; and pH of the reactor coolant) provide information on the extent of core degradation and the corrosive potential of the coolant. This information would not generally be used to terminate or limit the progression of a serious accident. PASS results would instead be used in recovery operations.

A.2 EMERGENCY RESPONSE DECISIONMAKING

To assess the role of PASS in emergency response decisionmaking, plant-specific emergency plans and emergency plan implementing procedures (EPIPs) of selected licensees, as well as NRC guidance documents, were reviewed. In this area of response to an accident, the decisions to be made include emergency classification, offsite protective action recommendations, and core damage assessments.

The utility documents and NRC guidance reviewed all contain provisions for using PASS results to define the emergency classification of an event, to recommend protective actions, and to assess the amount of core damage. However, in all cases, it is apparent that PASS sample results will lag other indicators of plant status that drive the emergency classification decisions, the recommendation of protective actions, and the assessment of core damage. Only in the most slowly developing accidents will PASS results be timely enough to influence the initial decisions, recommendations, and assessments. Grab-sampling types of PASS will provide information to confirm the results of other plant instruments and indicators, but only after a time lag of several hours in most cases. In-line PASS systems will provide information in a more timely manner and could be a primary source of information for emergency response decisionmaking.

A.2.1 Emergency Classification

NRC guidance on emergency classification is found in Appendix 1 to NUREG-0654 (NRC 1980c). Four emergency classes are identified: Unusual Event, Alert, Site Area Emergency, and General Emergency. Example initiating conditions are provided for each of these four emergency classes. Licensees are required to determine specific Emergency Action Levels (EALs), which when exceeded, cause each initiating condition to be classified at the appropriate level. EALs involving PASS results are:

- Unusual Event - high reactor coolant activity exceeding technical specifications indicating fuel damage
- Alert - very high reactor coolant sample activity (>300 μCi/cc dose equivalent of ^{131}I) indicating the severe loss of fuel cladding
- Site Area Emergency - NUREG-0654 specifies coolant activity and containment radioactivity levels as two possible indicators of a degraded core with possible loss of coolable geometry
- General Emergency - (no reactor coolant or containment atmosphere activity levels are mentioned).

The emergency classification procedures from ten plants were reviewed to establish the manner in which PASS sample results are used at plants. The EALs from the ten plants are shown in Table A.1. The conclusions that can be drawn from this data are:

- All ten plants have an EAL based on reactor coolant sample activity for determining emergency class.
- An Unusual Event is generally declared when the reactor coolant activity exceeds the technical specification limits for dose equivalent ^{131}I concentration. This sampling is done routinely at all plants using the primary sampling system.

TABLE A.1. Emergency Action Levels Based on Reactor Coolant Sample Data

Plant	EAL Category	RCS Activity for Each Emergency Class (μCi/cc ^{131}I Equivalent) Unusual Event	Alert	Site Area Emergency	General Emergency
A	Degraded Core	70-350	350-1770	>1180	>1180
B	Fuel Damage	>4	>300	>300	(a)
C	Fission Product Barrier Degradation	(a)	>300	(a)	>300
D	Fuel Cladding Degradation	>2	>300	>1000	>1000
E	Fuel Cladding Degradation	>1	>300	(a)	(a)
F	Radiological Controls	≥4-<300	≥300-<1430	≥1430	(a)
G	Fuel Element Failure	>Tech Specs	>300	>300	(a)
H	Fuel Damage	>1.1	>300	(a)	(a)
I	Degraded Fuel Integrity	>1.2	>300	(a)	(a)
J	Loss of Fission Product Barriers	(a)	>300	>300	>300

(a) None specified.

- Nine out of the ten plant EALs used the NUREG-0654 value of 300 μCi/cc dose equivalent of ^{131}I for the Alert-Level EAL.
- Four out of ten Site Area Emergency EALs and six out of ten General Emergency EALs do not specify reactor coolant activities. Four out of the six plants that specify RCS activity for these two classes indicate >300 μCi/cc dose equivalent ^{131}I.

Most of the reactor coolant activity EALs sets reviewed either 1) require another condition to be met prior to declaring the emergency class or 2) contain a list of conditions, any one of which would result in declaration of the emergency class. These other conditions may include subcooled margin, containment pressure readings, containment temperature readings, high-range containment radiation monitor readings, main steam line high radiation trip annunciator, and/or core exit thermocouple readings. These are illustrated in Figure A.1 which is an excerpt from a plant's emergency classification procedure. Three EAL events are listed, any one of which can initiate the declaration of an Alert: the first is high radiation level in the offgas system, the second is a high coolant sample activity, and the third is high radiation in the main steam lines. Only the high reactor coolant activity EAL may come from PASS. It is reasonable to expect that the other two EALs will cause the declaration of an alert before the primary coolant sample analysis is completed. Some classification procedures require indication of high coolant activity for escalating to higher emergency classes. In Figure A.1, the escalation of the event to a Site Area Emergency based on the core fuel damage category requires both a high coolant activity (RCS >300 μCi/cc dose equivalent ^{131}I) and an indication of reactor water level at the top of the active fuel core. For this particular EAL regarding fuel damage, PASS provides information on the

Category	Initiating Condition	Emergency Action Level Events	Emergency Classification
		ANY of the following	
Core Fuel Damage	Severe Loss of Fuel Cladding	1. Offgas Pretreatment Monitor reading greater than 5 Ci/sec	Alert
		OR	
		2. Coolant sample analysis indicates 300 μCi/ml equivalent ^{131}I or greater	
		OR	
		3. Main Steam Line Radiation Monitor exceeds trip set point	
		BOTH of the following:	Site Area Emergency
	Degraded Core with Possible Loss of Coolant Geometry	1. Reactor Water Level at top of active fuel core height as indicated on fuel zone level indicator (-167" Fuel Zone)	
		AND	
		2. High Coolant activity indicated by analysis of sample greater than 300 μCi/ml equivalent ^{131}I	

FIGURE A.1. Excerpt from Emergency Classification Scheme for Plant B

condition of the core when the vessel water level is lowered. In the unlikely event that no other EAL triggers the declaration of a Site Area Emergency, then the time-delayed PASS results might contribute to emergency classification.

Only one of the ten EAL sets (Plant F) reviewed used any information from the containment atmosphere sample in the classification procedures (Figure A.2). If containment atmosphere samples are not available from PASS, an emergency classification could still be made based on the area radiation monitor reading.

The review of plant procedures revealed no EALs based on the required chemical analysis of reactor coolant and containment atmosphere samples (e.g., hydrogen concentrations, dissolved gases, chloride or boron).

Radiological analyses of containment atmosphere samples provide an estimate of the source term in containment that can be used in projecting offsite doses, assuming a design-basis leak rate or a breach in containment. This dose projection can be used in emergency classification, since all plants have an EAL based on projected dose rates or doses at the site boundary. However, in the judgment of PNL staff involved in emergency plan reviews and drills, it is

	Unusual Event	Alert	Site Area Emergency	General Emergency
1.0 RADIOLOGICAL CONTROLS				
1.1.D (Liquid Release)	RM-L-12 $\geq$200 cpm (High Alarm) OR (TWO) RM-L-7 $\geq$1000 cpm (High Alarm)	NA	NA	NA
1.1.E (Containment Building)	NA	Containment Post Accident Sampling (CAT-PAS) $\geq$50 μCi/cc but <250 μCi/cc Total Noble Concentration OR $\geq$0.1 μCi/cc but <0.5 μCi/cc Total Radioiodine Concentration OR (TWO) Containment Post Accident Sample (CAT-PAS) results are NOT available with; RM G-8 $\geq$2E5 mrem/hr but <1E6 mrem/hr as read on the meter face	Containment Post Accident (CAT-PAS) indicates: $\geq$250 μCi/cc Total Noble Gas Concentration OR $\geq$0.5 μCi/cc Total Radioiodine Concentration OR (TWO) If the Containment Post Accident Sample (CAT PAS) results are NOT available with; RM G-8 $\geq$1E6 mrem/hr as read on the meter face	NA

FIGURE A.2. Emergency Action Levels for Plant F likely that EALs based on deteriorating plant conditions will trigger an appropriate emergency classification before dose projections based on the results of the containment air sample are available.

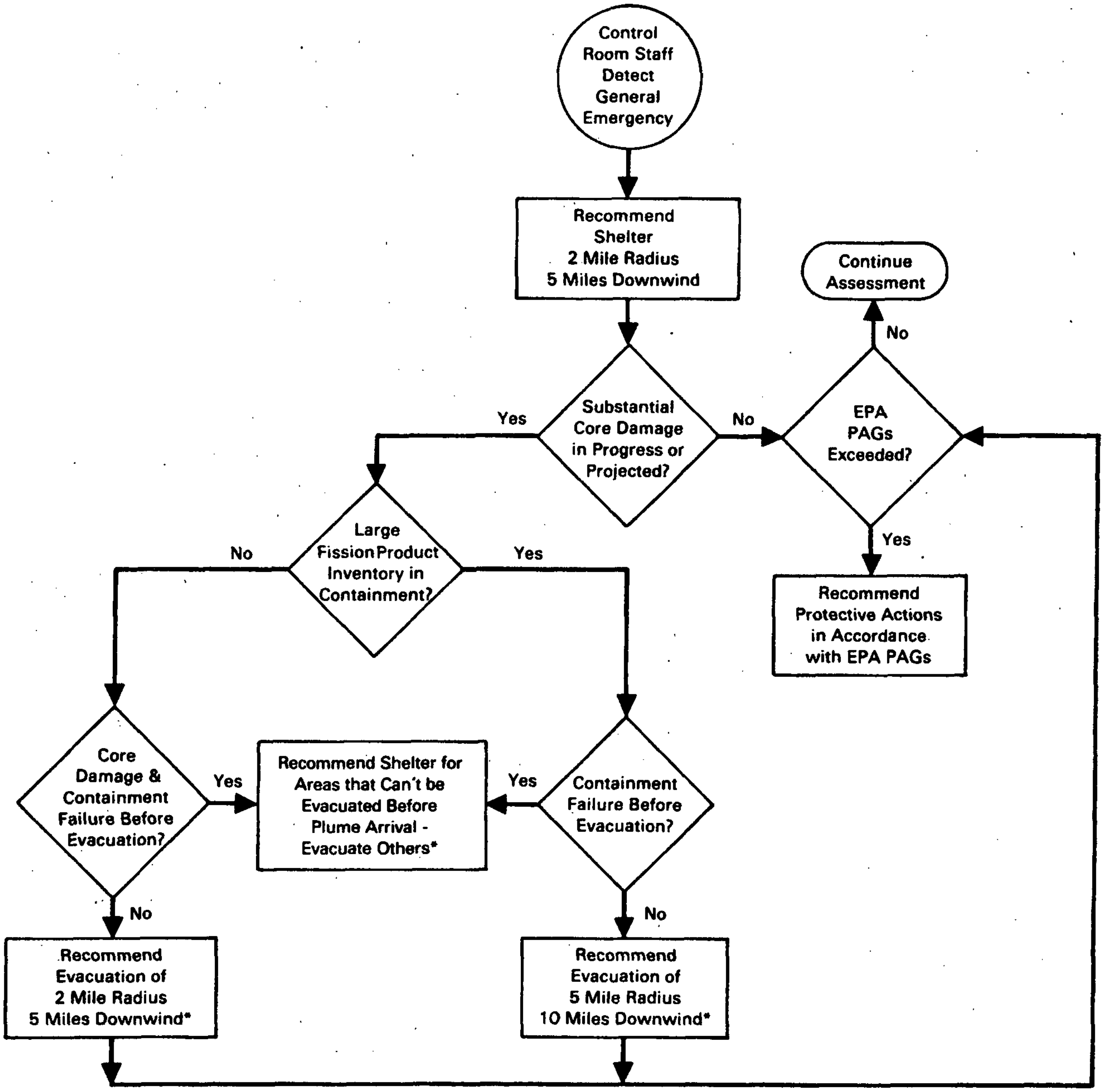
The importance of PASS data in emergency classification may depend on how quickly the accident progresses to core melt. Since it will take some plants two to three hours to obtain PASS sample results, it is likely that some plants would have to rely on other indicators (e.g., high-range containment monitor, incore thermocouples) to classify an emergency if core conditions deteriorate substantially in less than three hours. In slowly developing accidents, PASS information may be a more timely indicator of core damage; however, even in this situation other indicators would be available two to three hours sooner. It must be noted that plants which have an in-line PASS capability should not have this delay time and should be able to more readily use PASS information in emergency classification. Plants with grab sampling capabilities could reduce the time to collect and analyze samples if the need for the sample is anticipated so system activation (i.e., flushing sample lines) can be initiated.

A.2.2 Offsite Protective Action Recommendations

NRC guidance on offsite protective action recommendations (PARs) is found in NUREG-0654 (NRC 1980c). An inspection and Enforcement Information Notice presented the decision flowchart based on NUREG-0654 information shown in Figure A.3.

Four plant protective action procedures were reviewed. Flowcharts developed by the plants closely follow NRC guidance. A typical flowchart is shown in Figure A.4. This figure indicates that upon the declaration of a General Emergency, sheltering in a two-mile radius and to five miles downward is automatically recommended by the utility to local agencies. The first decision point is based on the potential loss of the fission product barrier or the potential loss of physical control of the facility. Information on the potential loss of the fission product barrier is a prediction of accident progression based on the course of the accident up to the point of declaring the General Emergency. The next decision point is based on the projected or actual degree of core damage. In this case, the utility can obtain this information from several sources. PASS sample results may be available, particularly for plants that have in-line systems. The utility may also make estimates on the degree of core damage based on reactor vessel water level, incore thermocouple readings, the availability of emergency core cooling system (ECCS) equipment, the containment hydrogen monitor, or the readings of containment radiation monitors.

At this point, the decision tree splits to consider the level of projected doses and the integrity of containment. Dose projections may be aided by PASS results. If a decision must be made before PASS or other indicators of projected doses are available, many plants rely on the dose projections for the accidents described in the Final Safety Analysis Report (FSAR). The FSAR doses are conservative and would tend to extend the protective action recommendations to greater distances. The last decision point in the flowchart may use PASS results to establish the fission product inventory in containment. In lieu of



Source: Appendix 1. NUREG-0654/FEMA-REP-1. Rev. 1

* For all evacuations, shelter the remainder of the plume EPZ and relocate the population affected by any ground contamination promptly following plume passage.

FIGURE A.3. NRC Guidance on Offsite Protective Decisions (NRC 1980c)

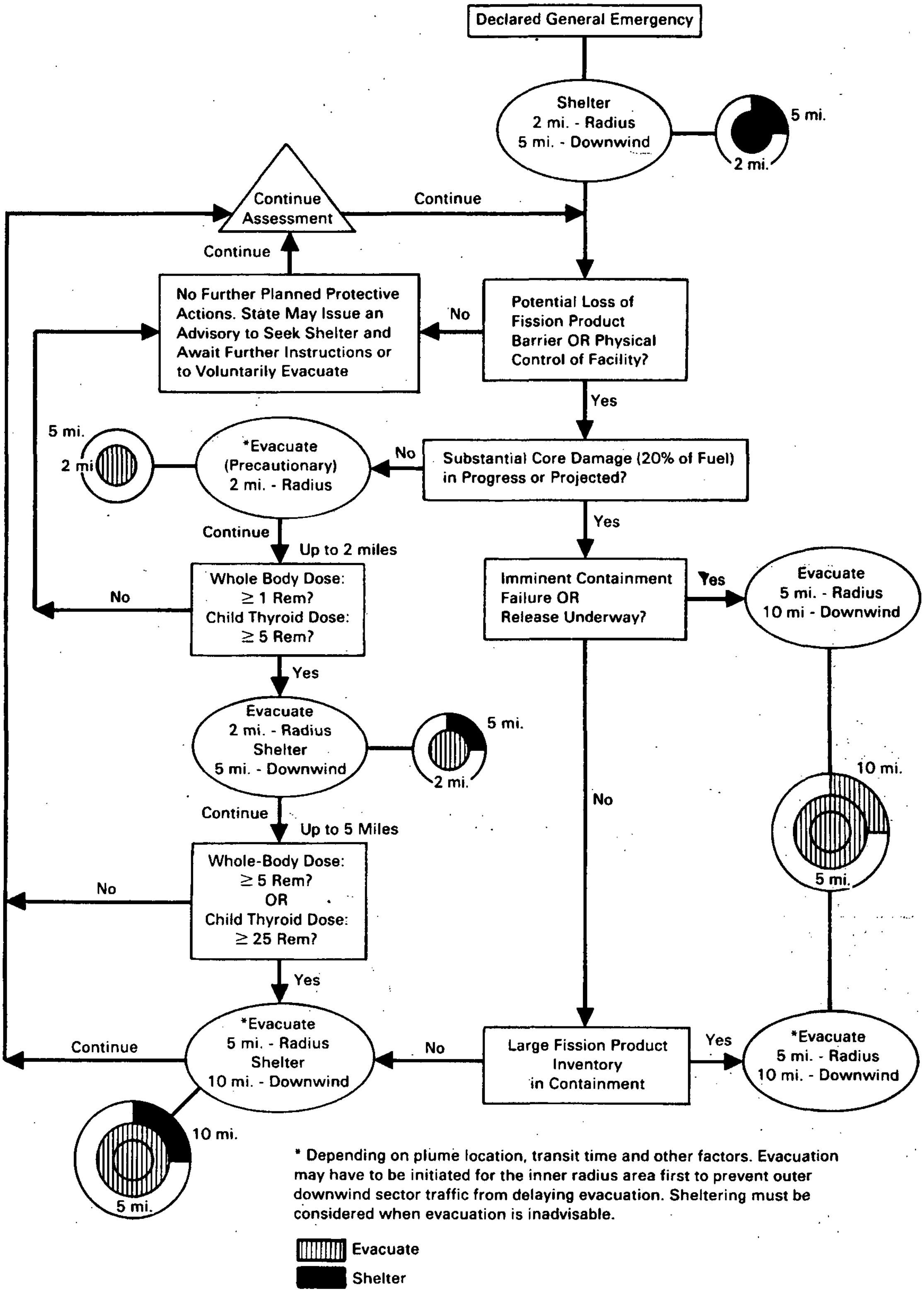


FIGURE A.4. Protective Action Recommendations Flowchart for Plant B a PASS sample to indicate the fission product inventory in containment, the utility can use the containment radiation monitor, the rate of primary to secondary leakage, or the level of the containment sump. This decision point is used to determine whether evacuation in the downwind sectors to 10 miles is necessary or whether sheltering of the population between 5 and 10 miles is adequate.

In all of this discussion of the PAR flowchart shown in Figure A.4, it is important to emphasize that these are recommendations to the local and state agencies who make the final decisions regarding the protective actions to be implemented. These state and local agencies have their own emergency response plans and procedures and have participated in extensive training on appropriate protective actions and the hazards of reactor accidents. They will factor the recommendations of the utility into their decisionmaking process.

One plant's offsite protective action procedure provides guidance on how to determine whether substantial core damage has occurred and whether substantial fission products are found in containment (Table A.3). Seven indicators of substantial core damage are listed, only two of which may be obtained from PASS. One of the five indicators of substantial fission product release to containment is taken from PASS results. The protective action flowchart for another plant uses the high-range containment monitor as the only indicator of core damage.

Dose projections can be based on PASS containment atmosphere sample results. The dose projection can be used to recommend protective actions in accordance with EPA Protective Action Guidelines as shown in Figure A.3. However, with a large fission product inventory in containment, use of NRC guidance would result in immediate evacuation of the population within a five-mile radius of the plant and 10 miles downwind of the plant. Sheltering is indicated for those areas that cannot be evacuated before plume arrival. This recommendation would be made based on plant conditions. Any recommendations based on projected doses from PASS data may come two to three hours later if plants have a grab-sample type of PASS and may not be responsive to information needs of public agencies and NRC guidance regarding timely protective action recommendations. However, PASS information would serve to confirm the correctness of the evacuation decision.

The results of this review indicate that PASS information obtained via grab sampling may not be used in making PARs because of the two to three hour delay in obtaining sample results. Although one licensee lists PASS as one indicator to consider in determining core damage, decisionmakers would likely use real-time indicators since PARs must be made in a timely manner. For example, when a General Emergency is declared, the licensee has to provide a PAR to the state/local agencies with the initial notification (within 15 minutes). In this situation, decisionmakers would not have time to wait for PASS results obtained via a grab-sample system.

TABLE A.3. Offsite Protective Action Decisionmaking Information

1. Actual or Imminent Substantial Core Damage

   A.[a] Comparison of RCS sample with pre-accident data; and/or

   B. Core known to be uncovered (RVLIS indication and/or Subcooling Monitor indicates superheat); and/or

   C. Five (5) incore thermocouples indicate $\geq$1100°F; and/or

   D. Actual or imminent loss of all ECCS equipment; and/or

   E.[a] Positive hydrogen concentration in containment atmosphere; and/or

   F. RM-RM-219 A/B or RM-RM-201 increasing; and/or

   G. Other indicators as may be deemed appropriate.

2. Actual or Imminent Substantial Fission Product Release to Containment

   A.[a] Comparison of containment atmosphere samples with pre-accident data; and/or

   B. RM-RM-202 increasing and RM-RM-219 A/B or RM-RM-201 increasing; and/or

   C. RCS leakage to containment > Technical Specification limit; and/or

   D. Containment safeguards sump level > pre-accident level; and/or

   E. Other indicators as may be deemed appropriate.

(a) Post-accident sampling system can provide this information.

NOTE: Assessment personnel are to consider these conditions as general symptoms rather than definite diagnosis. They must be considered as additional input to the assessment process and shall not individually be taken as sole evidence of the existence of a significant hazard.

A.2.3 Core Damage Assessment

The NRC guidance on reactor core damage assessment includes a core damage index consisting of 10 categories. The major classes of fuel degradation include: no fuel damage, clad failure, fuel overheat, and fuel melt. The latter three classes are each divided further into three levels of severity (minor, intermediate, and major).

Based on the NRC guidance and guidance from reactor owners groups (e.g., General Electric, Westinghouse, Babcock and Wilcox) in the form of generic core damage procedures, licensees have developed their own site-specific core damage procedures. These procedures are reviewed and approved by the NRC for all operating plants.

Six utility core damage assessment procedures were reviewed. They all use classification schemes identical or similar to the NRC core damage index. For example, one plant has modified its scheme to include only seven classifications; no fuel damage, <50% cladding failure, >50% cladding failure, <50% fuel over temperature, >50% fuel over temperature, <50% fuel melt, and >50% fuel melt.

PASS can provide information to aid in the assessment of potential fuel damage by indicating the types of fission products released to reactor coolant. The presence of certain radionuclides will give indication of the degree of core damage. NUREG-0737 states that the presence of iodine and cesium isotopes is indicative of high fuel temperatures, whereas the presence of nonvolatile isotopes is indicative of fuel melting (NRC 1980a).

PASS results are not the only information used in estimating core damage. Core damage assessment procedures require the evaluation of a variety of parameters to arrive at a core damage estimate. Other parameters that are evaluated include the following:

- high-range containment radiation monitor readings
- core exit thermocouple readings
- reactor vessel water level
- containment atmosphere hydrogen monitor readings.

Most of these parameters are displayed in real time on direct readouts in the control room and are immediately available to decisionmakers. PASS information may or may not be immediately available to decisionmakers dependent on system type (grab versus in-line). Therefore, decisionmakers may make an initial core damage assessment based on available parameters and update it later based on PASS results. Table A.4 taken from a licensee's procedures shows how core damage categories are determined based on specific parameter readings. Two of the six parameters in Table A.4 are determined from PASS information. These are "Percent and Type of Fission Products Released" and "Fission Product Ratio." Each parameter analyzed will result in it own indication of core damage. It is unlikely that all parameters will indicate the same core damage class. Therefore, a best estimate must be made based on available data.

TABLE A.4. Characteristics of Categories of Fuel Damage

Core Damage Category	Percent and Type of Fission Products Released[a]	Fission Product Ratio[a]	Containment High Range Area Monitor R/hr[b]	Core Exit Thermocouple Readings (DEG F)	Core Uncovered (Yes/No)	Hydrogen Monitor 1(2) AI PS 343 or 344 (Vol% H_2)
No Clad Damage	Kr-87<1x10-3 Xe-133<1x10-3 1-131<1x10-3 1-133<1x10-3	Kr-87=0.022 1-133=0.71	--	<750	No Uncovery	Negligible
0-50% Clad Damage	Kr-87 10^{-3}-0.1 Xe-133 10^{-3}-0.1 1-131 10^{-3}-0.3 1-133 10^{-3}-0.1	Kr-87=0.022 1-333=0.71	0-97	750-1300	Core Uncovery	<6%
50-100% Clad Damage	Kr-87 0.1-0.2 Xe-133 0.1-0.2 1-131 0.3-0.5 1-133 0.1-0.2	Kr-87=0.022 1-133=0.71	97-194	1300-1650	Core Uncovery	6%-11%
0-50% Fuel Pellet Overtemperature	Xe, Kr, Cs, I 1-20 Sr, Ba 0-0.4	Kr-87=0.22 1.33=2.1	194-25,000	>1,650	Core Uncovery	6%-11%
50-100% Fuel Pellet Overtemperature	Xe, Kr, Cs, I 20-40 Sr, Ba 0.4-0.8	Kr-87=0.22 1-133=2.1	2.5E4-5.0E4	>1,650	Core Uncovery	6%-11%
0-50% Fuel Melt	Xe, Kr, Cs, I 40-70 Sr, Ba 0.2-0.8 Pr, Rb, 0.1-0.8	Kr-87=0.22 1-33=2.1	5.0E4-8.5E4	>1,650	Core Uncovery	6%-11%
50-100% Fuel Melt	Xe, Kr, Cs, I, Te >70 Sr, Ba >24 Pr, Rb >0.8	Kr-87=0.22 1-133=2.1	>8.5E4	>1,650	Core Uncovery	6%-11%

(a) Can be determined using PASS information.
(b) *10 hours after shutdown.

Some of the limitations associated with core damage assessment parameters are:

- Reactor coolant and containment atmosphere samples must be sampled from the appropriate location (e.g., drywell, suppression pool) for the accident type.

- PASS sample results and high-range containment monitor readings may not be representative of current core conditions if there has not been adequate time for mixing. PASS samples taken during rapidly changing core conditions should not be heavily weighted in the core damage assessment. Ideally, multiple PASS samples should be obtained over an extended time period to better determine whether results are representative.

- Use of core exit thermocouple readings to assess core damage is limited by the maximum temperature readings of the thermocouples. For example, one plant's Core Damage Assessment Procedure indicates the maximum core exit thermocouple reading to be about 2300°F. At this temperature, most of the fuel rods would have cladding rupture but little other structural damage would have occurred. Therefore, information on degrees of fuel overheat or fuel melt cannot be determined from core exit thermocouple readings.

Results of containment atmosphere samples obtained through PASS are typically analyzed for hydrogen with the results used in core damage assessment. The safety-grade containment hydrogen monitor will provide a real-time indication of containment hydrogen levels. Therefore, containment hydrogen levels as determined by PASS would be used as a backup in core damage assessment.

Core damage assessments can be made without PASS results by using other parameters (e.g., high-range containment monitor readings, incore thermocouple readings). If the licensee has a grab-sample type of PASS, this assessment can most likely be made two to three hours before PASS sample results are available. It is likely that decisionmakers would use PASS results to refine or confirm core damage assessment based on parameters obtained earlier in this situation. The collection of a PASS reactor coolant sample allows the physical removal of coolant from the core for subsequent analysis. Other systems, such as high-range containment monitors and reactor vessel water level indicators, provide only indirect or partial information on core damage. If PASS results indicate greater core degradation than the other parameters, decisionmakers may re-examine previous decisions to determine if any additional precautions are warranted.

A.3 RECOVERY

NRC requirements for recovery actions are found in 10 CFR 50.47(b)(13) and 10 CFR 50 Appendix E (H). Licensees are required to develop general plans for long-term recovery and re-entry after termination of an accident or release.

Recovery procedures from 10 licensees were reviewed. Most plans were very general in nature and did not contain specific parameters that would be evaluated in making recovery decisions. One procedure did indicate that primary coolant and containment atmosphere sample results should be trended during recovery.

The recovery phase is a logical time to use PASS for the following reasons:

- Plant conditions have been stabilized and PASS samples should be representative of actual core and containment atmosphere conditions.
- The delay time for obtaining sample results if a grab sampling system is used will not be a critical factor since time is not critical once the accident has been terminated. There is also sufficient time in the recovery phase to take multiple samples for better estimates of the extent of core damage.
- Radiological analysis of a containment atmosphere sample would provide information on the source term in containment, which could be used in projecting offsite doses that may result from planned venting during the recovery phase.
- PASS samples should be useful in the historical documentation of the core damage condition to enable reconstruction of the extent of the accident.

A.4 SUMMARY OF PASS ASSESSMENT IN LICENSEE DECISIONMAKING PROCESS

A limitation of using PASS information in accident management and emergency response decisionmaking is the two to three hour time delay before sample analysis results are available if a grab-sampling type of PASS is used. An in-line PASS system would not have this limitation. The time delay is not as critical when considering long-term recovery. Other real-time indicators and procedures are available for use in emergency classification decisions, core damage assessments, and PARs. Some of these are listed below:

- For classification schemes that use PASS information, many EALs are constructed so that if PASS results are not available, other conditions will trigger the emergency classification (e.g., subcooled margin, containment pressure readings, containment temperature readings, high-range containment monitor readings, and core exit thermocouple readings).
- Some licensees have a EAL category that uses PASS information (e.g., rector coolant sample analysis to determine the degree of fuel damage) for emergency classification. However, each licensee has other EAL categories (e.g., abnormal primary coolant leak rate, steam system leaks, radiological effluent releases) that result in earlier

classifications than waiting two to three hours while PASS samples obtained by a grab sampling system are being collected and analyzed.

- Initial core damage assessments can be based on real-time indicators, including high-range containment radiation monitor readings, core exit thermocouple readings, reactor vessel water level, and containment atmosphere hydrogen monitor readings.

- Protective action recommendations require estimates of core damage and fission product inventory in containment. Real-time indicators of these two conditions include high-range containment radiation monitor readings, core exit thermocouple readings, reactor vessel water level, containment atmosphere hydrogen monitor readings, subcooling monitor, and factors such as loss of all ECCS equipment, and reactor coolant system leakage to containment.

Although of marginal use in making emergency classifications, initial core damage assessments, or PARS, PASS information would provide several beneficial functions during an emergency. These are:

- The boron analysis of the PASS reactor coolant sample can provide valuable information to reactor operators regarding criticality of the core and could be used in the accident management phase.

- The isotopic analysis of the PASS reactor coolant sample could be used to confirm or modify the initial core damage assessment made from earlier real-time indicators. Since PASS involves the physical removal of coolant from the core, the results should be direct indicators of core damage. These analyses will also serve to document the status of the core at specific times throughout the emergency, which will be useful in later reconstruction of the accident.

- As discussed in Section 4.2.1, a PASS containment atmosphere sample will provide an estimate of the source term in containment. The source term can then be used in making offsite dose projections assuming a design-basis leak rate or a breach in containment. If no release is occurring, this would be one of the primary methods for making dose projections. These dose projections will give decisionmakers an estimate of worst case offsite consequences. This is useful information even if emergency classification and offsite PARs have already been made based on other indicators.

- PASS data can be used in the recovery phase. The collection and analysis of multiple reactor coolant samples and containment atmosphere samples will give more accurate information on extent of core damage. This information can be used in planning recovery operations.

APPENDIX B

PWR AND BWR TURBINE MISSILE RISK CALCULATIONS

APPENDIX B

PWR AND BWR TURBINE MISSILE RISK CALCULATIONS

PWR RISK CALCULATIONS

For this analysis, it was assumed that the turbine missile would strike and damage the most important single safety system modeled by the risk equations (assuming that this system is susceptible to a missile strike). Results of the Oconee 3 Reactor Safety Study Methodology Applications Program (RSSMAP) study used in prioritization of generic safety issues (Andrews et al. 1983) were assumed to be representative of all PWRs.

Of the systems that contribute to the core melt frequency at Oconee 3, as indicated in the RSSMAP study, the following were assumed to be susceptible to potential damage from a missile strike:

- low-pressure/containment-spray injection system (LP/CSIS)--train A or B
- low-pressure/containment-spray recirculation system (LP/CSRS)--train A or B
- engineered safeguards protective system (ESPS)--channels 1, 2, 3, or 4
- emergency feedwater system (EFWS)--electric pump trains or turbine pump train
- high pressure injection system (HPIS)--train A or B, or train C
- low pressure service water system (LPSWS)--train A or B.

Common cause failure of multiple trains/channels of a system was not considered to be credible for this analysis.

Turbine failure was assumed to always cause loss of the power conversion system (PCS), thereby initiating a power conversion system (T_2) transient sequence. Only those parameters related to the susceptible systems were assumed to be affected in T_2 sequences. Thus, the only accident sequences in the Oconee study that could be affected by turbine missiles are: T_2MLU, T_2MQH, T_2MQFH, T_2MLUO, T_2KMU and T_2MQD. The system or component failures associated with a T_2 transient are given in Table B.1.

Parameters in these accident sequences that are potentially affected by turbine missiles are listed in Table B.2 along with their probability values

TABLE B.1. System or Component Failures Associated with a T_2 Transient

Designation	Description
M	Power conversion system (normal operation).
L	Emergency feedwater system, recovery of power conversion system and high head auxiliary feedwater system.
U	High pressure injection system.
H	Emergency coolant recirculation system.
F	Containment spray recirculation system.
O	Reactor building cooling system.
K	Reactor protection system.
D	Emergency coolant injection system.
Q	Reclosure of pressurizer safety/relief valves.

from the Oconee study (Andrews et al. 1983) and the probability values that were used for the computer calculations of risk.

Studies were first performed to determine the sensitivity of the public risk to changes in individual missile-affected parameter values. With the probability of nonrecovery of the power conversion system (PCSNR) set to 1.0 per reactor year and appropriate values for T_2 based in the 4.5, 9, and 30-year turbine inspection intervals, each of the remaining parameters and credible combinations (e.g., CH1, CH2, etc.) was increased by 0.01 per reactor year in separate calculations to simulate simultaneous failure of the turbine and one train/channel of a susceptible system. Again, these failure parameters were assumed to be affected only in the T_2 sequences. This assumed that the time following a turbine missile event was short enough to preclude other independent accident initiation. Results of this analysis indicate that the public risk is most sensitive to parameter F1: failure of a pump in train B of the low-pressure service water system.

The final computer run to calculate the risk values included in this appendix was performed using values for only T_2, PCSNR, and F1 that were different than the RSSMAP study described in Andrews (1983). T_2 was chosen so that the base case assumes a 4.5 year inspection interval, and the adjusted cases represent 9- or 30-year inspection intervals, respectively. This resulted in T_2 values of 1.0E-05/ry for the base case and 4.0E-05/ry or 1.60E-04/ry for the two adjusted cases, respectively. The value of F1 is assumed to be the product of the conditional probability of turbine missile occurrence (0.76) and the probability of safety system damage (the product of P_2 and P_3 is

TABLE B.2. Potential PWR Missile-Affected Parameters

Parameters	Probability/ry	Description
T_2	(a)	Loss of power conversion system transient caused by other than a loss of offsite power.
PCSNR	1.0[b]	Failure to restore the PCS within 30 min. following a T_2 transient.
F1	0.0090[c]	Failure of pump in train B of the low pressure service water system (LPSWS).
CH1	5.00E-03	Failure of logic channel 1 of the engineered safeguards protective system (ESPS).
CH2	5.00E-03	Failure of logic channel 2 of the ESPS.
CH3	5.00E-03	Failure of logic channel 3 of the ESPS
CH4	5.00E-03	Failure of logic channel 4 of the ESPS.
G1	0.014	Failure of a pump in train A of the LPSWS.
CONST1	2.10E-04	Failure of emergency feed water system (EFWS).
A1	9.80E-03	Failure of pump discharge valve in the discharge line common to both backup pumps (A and B) of the high pressure injection system (HPIS).
C1	9.80E-03	Failure of pump suction valve in the suction line common to both backup pumps (A and B) of the HPIS.
B1	0.035	Failure of component in the main line of the HPIS downstream from the borated water storage tank (BWST) isolation valve.
D.E	4.90E-04	Failure of both trains A and B of low pressure containment spray recirculation system (LP/CSRS).
D.X	2.10E-04	Failure of both a pump discharge valve in Train A of low pressure/containment spray injection system (LP/CSIS) and the containment sump suction valve in train B of the LP/CSRS.
E.W	2.10E-04	Failure of both a pump discharge valve in Train B of LP/CSIS and the containment sump suction valve in train A of the LP/CSRS.

TABLE B.2. (contd)

Parameters	Probability/ry	Description
W.X	8.80E-05	Failure of both containment sump suction valves in the LP/CSRS.
B.W	2.70E-05	Failure of both a pump suction valve in train B of the LP/CSIS and the containment sump suction valve in train A of the LP/CSRS.
C.X	2.70E-05	Failure of both a pump suction valve in train A of the LP/CSIS and the containment sump suction valve in train B of the LP/CSRS.

(a) T_2 = 1.0E-05/ry for an inspection interval of 4.5 years.
T_2 = 4.0E-05/ry and 1.60E-04/ry for 9- and 30-year inspection intervals, respectively.
(b) PCSNR, although set to 1.0/ry for this study, was set to 0.1/ry in the Andrews et al. (1983) study.
(c) F1, which was set to 0.0090 for the final risk analysis case as detailed in the text below, was set to 0.0014 in the Andrews et al. (1983) study.

assumed, i.e., 1.0E-02/ry) added to the original F1 (0.0014/ry) the failure rate in the RSSMAP report, yielding an affected F1 of 0.0090/ry. PCSNR, the failure to restore the PCS within 30 minutes following a T_2 transcent, was set to 1.0/ry. Tables B.3 and B.4 list the best estimate frequencies of the affected accident sequence values obtained from computer calculations for the base case and the adjusted cases for turbine missiles.

Each of the accident frequencies was then multiplied by its original containment mode probability to obtain frequencies for the PWR release categories. Results of this evaluation are shown in Tables B.5 and B.6 for the changes in inspection intervals from 4.5 years to either 9 or to 30 years, respectively.

The final step in the risk calculation was to multiply each of the changes in release category frequency from the base to the adjusted case (from Tables B.5 and B.6) by the dose factor for each release category. These products were then summed to yield the total public risk increase for this action at PWRs. The dose factors used in this analysis assume a population density of 340 people per square mile at a Midwestern site (Andrews et al. 1983). These factors are shown in Table B.7.

TABLE B.3. Affected Oconee Accident Sequence Frequencies for Changing the Turbine Inspection Interval from 4.5 to 9 Years

Affected Accident Sequence	Containment Failure Mode	Mode Probability	Affected Frequency (event/ry)	
			Base	Adjusted
T_2MLU	γ(PWR-3)	0.5	6.44E-11	2.58E-10
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2MQH	γ(PWR-3)	0.5	3.67E-11	1.47E-10
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2MQFH	γ(PWR-2)	0.5	1.67E-11	6.67E-11
	β(PWR-4)	0.0073		
	ε(PWR-6)	0.5		
T_2MLUO	γ(PWR-3)	0.5	1.74E-09	6.94E-09
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2KMU	γ(PWR-3)	0.5	2.60E-11	1.04E-10
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2MQD	γ(PWR-3)	0.5	5.00E-12	2.00E-11
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		

Results of these calculations indicate that the best estimates for risk and core melt frequency increases are as follows:

	Change in Inspection Interval	
	4.5 to 9 years	4.5 to 30 years
Core melt Frequency Increase (event/ry)	6.04E-09	3.02E-08
Public Risk Increase (person-rem/ry)	1.61E-02	8.05E-02

BWR RISK CALCULATIONS

The BWR calculations for risk and core-melt frequency reduction were done in the same manner as the PWR calculations. In this case, the representative reactor was assumed to be Grand Gulf 1. Results of the Grand Gulf 1 RSSMAP study used in the prioritization of generic safety issues (Andrews et al. 1983) were assumed to be representative of all BWRs.

TABLE B.4. Affected Oconee Accident Sequence Frequencies for Changing the Turbine Inspection Interval from 4.5 to 30 Years

Affected Accident Sequence	Containment Failure Mode	Mode Probability	Affected Frequency (event/ry) Base	Affected Frequency (event/ry) Adjusted
T_2MLU	γ(PWR-3)	0.5	6.44E-11	1.03E-09
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2MQH	γ(PWR-3)	0.5	3.67E-11	5.87E-10
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2MQFH	γ(PWR-2)	0.5	1.67E-11	2.67E-10
	β(PWR-4)	0.0073		
	ε(PWR-6)	0.5		
T_2MLUO	γ(PWR-3)	0.5	1.74E-09	2.78E-08
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2KMU	γ(PWR-3)	0.5	2.60E-11	4.16E-10
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		
T_2MQD	γ(PWR-3)	0.5	5.00E-12	8.00E-11
	β(PWR-5)	0.0073		
	ε(PWR-7)	0.5		

TABLE B.5. Affected PWR Release Category Frequencies for Changing the Turbine Inspection Interval from 4.5 to 9 Years

Release Category	Release Category Frequency (event/ry) Base	Release Category Frequency (event/ry) Adjusted	Change in Release Category Frequency
PWR 1	0	0	0
PWR 2	9.69E-12	3.88E-11	2.91E-11
PWR 3	9.82E-10	3.93E-09	2.94E-09
PWR 4	1.54E-13	6.16E-13	4.62E-13
PWR 5	1.42E-11	5.69E-11	4.27E-11
PWR 6	1.08E-11	4.31E-11	3.24E-11
PWR 7	9.96E-10	3.98E-09	2.99E-09
Total	2.01E-09	8.05E-09	6.04E-09

TABLE B.6. Affected PWR Release Category Frequencies for Changing the Turbine Inspection Interval from 4.5 to 30 Years

Release Category	Release Category Frequency (event/ry) Base	Adjusted	Change in Release Category Frequency
PWR 1	0	0	0
PWR 2	9.69E-12	1.55E-10	1.45E-10
PWR 3	9.82E-10	1.57E-08	1.47E-08
PWR 4	1.54E-13	2.46E-12	2.31E-12
PWR 5	1.42E-11	2.28E-10	2.13E-10
PWR 6	1.08E-11	1.73E-10	1.62E-10
PWR 7	9.96E-10	1.59E-08	1.49E-08
Total	2.01E-09	3.22E-08	3.02E-08

TABLE B.7. PWR Dose Conversion Factors

Release Category	Base Factor (person-rem)
PWR 1	5.40E06
PWR 2	4.80E06
PWR 3	5.40E06
PWR 4	2.70E06
PWR 5	1.00E06
PWR 6	1.50E05
PWR 7	2.30E03

Of the systems that contribute to the core melt frequency at Grand Gulf 1, as indicated in the RSSMAP study, the following were assumed to be susceptible to potential damage from a missile strike.

- high-pressure core spray system (HPCSS)
- low-pressure core spray system (LPCSS)
- reactor core isolation cooling system (RCICS)--electric-motor-driven pump train or turbine-driven pump train
- residual heat removal system (RHRS)--train A or B
- low-pressure coolant injection system (LPCIS)--train A, B, or C

- suppression pool makeup system (SPMS)--train A or B
- standby service water system (SSWS)--train A, B, or C.

Common cause failure of multiple trains of a system was not considered to be credible for this analysis.

Turbine failure generation was assumed to always cause loss of the power conversion system (PCS), thereby initiating a power conversion system (T_{23}) transient sequence. Only those parameters related to the susceptible systems were assumed to be affected in T_{23} sequences. Thus, the only accident sequences in the Grand Gulf study that could be affected by turbine missiles are: T_{23}PQI, T_{23}PQE, T_{23}QW, and T_{23}C. The system, component, or function failures associated with a T_{23} transient are given in Table B.8.

As in the PWR analysis, parameters in these accident sequences potentially affected by turbine missiles are listed in Table B.9 along with their probability value from the Grand Gulf study (Andrews et al. 1983) and the probability values that were used for computer calculations of risk in this turbine missile-affected study of BWRs. Studies were first performed to determine the sensitivity of the public risk to changes in individual missile-affected parameter values. No credit was taken for recovery of the PCS (i.e., parameter Q1 was set to 1.0/ry). Each of the affected parameters and credible combinations other than T_2 and Q1 that had been fixed for this study was then individually increased by 0.01/ry in separate calculations to simulate simultaneous failure of the turbine and one train of a susceptible system. Again, this was done only for the T_{23} sequences. These calculations identified the sensitivity of the public risk to changes in each individual parameter value.

TABLE B.8. System, Component, or Functional Failures Associated with a T_{23} Transient

System Parameters	Description
P	Failure of a safety/relief valve to reseat.
Q	Failure of the power conversion system.
I	Failure of the residual heat removal systems after a LOCA.
E	Failure of the emergency core cooling system.
W	Failure of the residential heat removal systems after a transient.
C	Failure to render the reactor subcritical.

TABLE B.9. Potential BWR Missile-Affected Parameters

Parameters	Probability/ry	Description
T_{23}	(a)	Transient other than loss of offsite power that requires a reactor shutdown.
P	0.1	Failure of a safety/relief valve to reseat.
Q1	1.0[b]	Failure of the power conversion system to remove decay heat in approximately 30 hours.
VGA2	0.024	Failure of a valve in train A of the residual heat removal system (RHRS).
VGB2	0.024	Failure of a valve in train B of the RHRS.
SSA	0.0286[c]	Loss of flow path into and through pump A of the standby service water system (SSWS), including the pump A oil cooler.
SSB	0.021	Loss of flow path into and through pump B of the SSWS, including the pump B oil cooler.
VGB1	0.015	Failure of a valve in the inlet/outlet piping of the RHRS of the SSWS for RHRS heat exchanger B.
LB2	0.014	Loss of flow path from the suppression pool through the pump in train B of the low-pressure coolant injection system (LPCIS).
LA2	0.014	Loss of flow path from the suppression pool through the pump in train A of the LPCIS.
SBC	0.0012	Failure of the actuation and control circuitry for train B of SSWS.
SAC	0.0012	Failure of the actuation and control circuitry for train A of the SSWS.
Q	1.0	Failure of power conversion system to provide makeup water.

(a) T_{23} = 1.0E-05/ry for an inspection interval of 4.5 years.
T_{23} = 4.0E-05/ry and 1.60E-04/ry for 9- and 30-year inspection intervals, respectively.

(b) Q1, although set to 1.0/ry for this study, was set to 0.0070/ry in the Andrews (1983) study.

(c) SSA, which was set to 0.0286 for the final risk analysis case as detailed in the text below, was set to 0.0210 in the Andrews (1983) study.

Results indicate that the public risk is most sensitive to missile-induced failure of either train of the standby service water system (SSWS). Affected parameters for this system include SSA and SSB (loss of flow path into and through pump A and B, respectively, of the SSWS). Train A was selected as the target of the turbine missile in this analysis.

The final computer run to calculate the risk values included in this appendix was performed using values for only T_{23}, Q1, and SSA that were different from the RSSMAP study described in Andrews et al. (1983). This calculation set T_{23} to the values shown in the footnote to Table B.9 and Q1 (failure of the PCS to remove decay heat is ~30 hours) was set to 1.0/ry. SSA values were calculated by adding the product of the turbine missile conditional probability (0.76) and the probability of safety system damage (1.0E-02/ry) to the original SSA (0.021/ry), the failure rate in the RSSMAP report, yielding an affected SSA of 0.0286/ry. Tables B.10 and B.11 list the best estimate

TABLE B.10. Affected Grand Gulf Accident Sequence Frequencies for Changing the Turbine Inspection Intervals from 4.5 to 9 Years

Sequence	Containment Failure Mode	Mode Probability	Affected Frequency (event/ry) Base	Adjusted
T_{23}PQI	α (BWR-1) δ (BWR-2)	0.01 1.0	8.90E-10	3.56E-09
T_{23}PQE	γ (BWR-3) δ (BWR-4)	0.5 0.5	7.71E-13	3.09E-12
T_{23}QW	δ (BWR-2)	1.0	3.15E-09	1.26E-08
T_{23}C	δ (BWR-2)	1.0	7.71E-12	3.09E-11

TABLE B.11. Affected Grand Gulf Accident Sequence Frequencies for Changing the Turbine Inspection Intervals from 4.5 to 30 Years

Sequence	Containment Failure Mode	Mode Probability	Affected Frequency (event/ry) Base	Adjusted
T_{23}PQI	α (BWR-1) δ (BWR-2)	0.01 1.0	8.90E-10	1.42E-08
T_{23}PQE	γ (BWR-3) δ (BWR-4)	0.5 0.5	7.71E-13	1.23E-11
T_{23}QW	δ (BWR-2)	1.0	3.15E-09	5.04E-08
T_{23}C	δ (BWR-2)	1.0	7.71E-12	1.23E-10

frequencies of the affected accident sequence value obtained from computer calculations for the base case and the adjusted cases for the turbine missiles.

Each of the accident frequencies was then multiplied by its original containment mode probability to obtain frequencies for the BWR release categories. Results for these calculations are shown in Tables B.12 and B.13 for the change in inspection intervals from 4.5 years to either 9 or to 30 years, respectively.

The difference between the base and adjusted case frequencies (from Tables B.12 and B.13) was multiplied by the dose conversion factor for each release category. These products were then summed to yield the total public risk increase for this action at BWRs. Dose factors used in this analysis were developed for the same environment as for the representative PWR reactor (Andrews et al. 1983). They are shown in Table B.14.

TABLE B.12. Affected Release Category Frequencies for Changing the Turbine Inspection Interval from 4.5 to 9 Years

Release Category	Release Category Frequency (event/ry) Base	Release Category Frequency (event/ry) Adjusted	Change in Release Category Frequency
BWR 1	9.89E-12	3.96E-11	2.97E-11
BWR 2	4.11E-09	1.64E-08	1.23E-08
BWR 3	4.04E-13	1.62E-12	1.21E-12
BWR 4	4.62E-13	1.85E-12	1.39E-12
Total	4.12E-09	1.65E-08	1.24E-08

TABLE B.13. Affected Release Category Frequencies for Changing the Turbine Inspection Interval from 4.5 to 30 Years

Release Category	Release Category Frequency (event/ry) Base	Release Category Frequency (event/ry) Adjusted	Change in Release Category Frequency
BWR 1	9.89E-12	1.58E-10	1.48E-10
BWR 2	4.11E-09	6.58E-08	6.16E-08
BWR 3	4.04E-13	6.47E-12	6.07E-12
BWR 4	4.62E-13	7.40E-12	6.93E-12
Total	4.12E-09	6.59E-08	6.18E-08

TABLE B.14. BWR Dose Conversion Factors

Release Category	Dose Conversion Factor (person-rem)
BWR 1	5.40E+06
BWR 2	7.10E+06
BWR 3	5.10E+06
BWR 4	6.10E+05

Results of these calculations indicate that the best estimates for risk and core melt frequency increases are as follows:

	Change in Inspection Interval	
	4.5 to 9 years	4.5 to 30 years
Core melt Frequency Increase (event/ry)	1.24E-08	6.18E-08
Public Risk Increase (person-rem/ry)	8.77E-02	4.38E-01

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In a study commissioned by the Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory (PNL) evaluated the costs and benefits of streamlining regulatory requirements in the areas of postaccident sampling systems, turbine missiles, combustible gas control, and impregnated charcoal filters. The basic framework of the analyses was that presented in the Regulatory Analysis Guidelines (NUREG/BR-0058) and in the Handbook for Value-Impact Assessment (NUREG/CR-3568). The effects of streamlined regulations were evaluated in terms of such factors as population dose and costs to industry and NRC. The results indicate that streamlining the regulatory requirements in three of the four areas, i.e., postaccident sampling systems, turbine missiles, and combustible gas control, would have little impact on public risk. Streamlined regulatory requirements in the fourth area, impregnated charcoal filters, might increase public risk. Cost evaluations indicate substantial savings by lengthening the inspection interval for low-pressure turbine rotors. Small-to-moderate savings may be realized through postulated modifications to the postaccident sampling system requirements and to the combustible gas control requirements for hydrogen recombiners in inerted BWR Mark I and II containments. The results indicate that the use of impregnated charcoal filters is the most cost-effective method of radioiodine removal from building ventilation systems.

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