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Review of Light Water Reactor Regulatory Requirements

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

- Reactor Containment Leakage Rates
- Main Steam Isolation Valve Leakage Control Systems
- Fuel Design Safety Reviews

Prepared by M. F. Mullen, W. J. Bailey, C. E. Beyer,
G. J. Konzek, P. J. Pelto, W. B. Scott

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

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Prepared by
M. F. Mullen, W. J. Bailey, C. E. Beyer,
G. J. Konzek, P. J. Pelto, W. B. Scott

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Risk Analysis and Operations
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
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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 streamlining regulatory requirements in the areas of reactor containment leakage rate, main steam isolation valve leakage control systems in boiling water reactors (BWRs), and NRC fuel system safety reviews. 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 streamlined regulations were evaluated in terms of such factors as population dose, individual dose, prompt fatalities and injuries, and costs to industry and NRC. The results indicate that streamlining the regulatory requirements in all three areas would have little impact on public risk. Substantial savings in operating costs may be realized in the areas of containment leakage rates and leakage control systems for BWR main steam isolation valves. The cost analysis indicates that only marginal benefits may be gained by streamlining NRC's safety review of fuel system designs.

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 Department of Energy (DOE) by Battelle Memorial Institute, is conducting a series of studies in support of this NRC program. This report covers a portion of PNL's work. The purpose of the report is to present information on the risks, costs and benefits of streamlining regulatory requirements in three areas:

- reactor containment leakage rates
- main steam isolation valve (MSIV) leakage control systems (LCS)
- NRC licensing review of fuel design information.

These three areas of regulation were selected by NRC staff for analysis in the initial (pilot) phase of the regulatory review program.

CONCLUSIONS

Analyses were performed to assess the effects of streamlining regulatory requirements in the three selected 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 streamlining the requirements. Various measures of risk were examined including population dose, expected early fatalities and injuries, and individual dose. Sensitivity studies were also performed to explore the effects of such factors as accident source terms.

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

- The three areas of regulation cover a range of different types of regulatory requirements, and the analyses considered a range of different degrees of regulatory relaxation. In the case of the containment leakage rate limit, the regulatory modification considered was to relax the numerical leakage rate limit. In the case of main steam isolation valve leakage control systems, the regulatory modification considered was complete elimination of the requirement, and disabling of the systems currently in place. In the case of fuel system safety reviews, the modification considered was the selective elimination of items in the current review procedure that may have marginal risk significance.

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

	Area of Regulation		
	Reactor Containment Leakage Rate[(a)]	Main Steam Isolation Valve Leakage Control System[(b)]	Fuel System Safety Reviews[(c)]
Effect on public risk if requirements were streamlined[(d)]	Marginal (On the order of a few percent, or less, of overall risk)	Marginal (Less than one percent of overall risk)	Marginal (Not quantified)
Benefits of streamlining requirements[(e)]	Greater than $\$10^7$	Greater than $\$10^6$	Marginal (Not quantified)
Benefit-Risk comparison, if requirements were streamlined (dollars saved per person-rem of risk)	In the range of $\$10^3$-$\$10^4$ per person-rem	Greater than $\$10^4$ per person-rem	NA[(f)]

(a) Increase allowable leakage rate to 10% per day. Currently, typical allowable leakage rates are 0.1% for PWRs and 1% for BWRs.
(b) Eliminate the requirement for MSIV leakage control systems and disable the systems in plants that currently have (or will have) them.
(c) Eliminate selected items from the current procedures for fuel system safety reviews, which are set forth in Section 4.2 of the Standard Review Plan, NUREG-0800 (NRC 1981a).
(d) Various measures of public risk were considered, including population and individual dose, early fatalities and injuries, and latent cancers. For each of these measures the effect of streamlining the requirements was marginal.
(e) Costs and cost savings in this table are summed over the remaining lifetimes 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 1984a).
(f) Not applicable. It is assumed that the benefit-risk comparison is not of interest when the benefits are marginal.

- In all three cases, judiciously streamlining the existing regulatory requirements is 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.

- The benefits of streamlining the existing regulatory requirements vary. In the case of reactor containment leakage rate limits, the estimated benefits are on the order of several tens of millions of dollars. In the case of fuel system safety reviews, the benefits are insignificant (no dollar estimate was computed). The case of the MSIV leakage control system occupies a middle ground, with benefits of several million dollars.

- Comparisons of benefits and risks also vary. In the case of the containment leakage rate limit, the ratio of dollars saved to risks incurred (dollars per person-rem) is estimated to be slightly higher than the benchmark of $1000 per person-rem that has been used in some other contexts (e.g., the proposed safety goals, and 10 CFR 50, Appendix I). In the case of the MSIV leakage control system, the ratio is estimated to be considerably higher than the $1000 per person-rem benchmark. For the case of the fuel system safety reviews, no ratio was estimated; it was assumed that comparison of benefits to risks is not of interest in this context because the benefits are insignificant.

- 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 et al. (1983) in the Handbook for Value-Impact Assessment, the real strengths of quantitative analysis are the discipline that it provides and its display 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 four sections. Section 1 covers the background, objectives, and scope of this study. Section 2 presents the analysis of containment leakage rate testing. Section 3 covers the MSIV leakage control system. Section 4 covers fuel system safety reviews. Two appendices contain supporting information on the containment leakage rate analysis.

OVERVIEWS OF THE THREE ANALYSES

The remainder of the summary section gives an overview of the analyses that were performed in each of the three areas of regulation selected by the NRC staff for examination in this study.

Reactor Containment Leakage Testing

Reactor containments constitute one of the principal lines of defense in the defense-in-depth design philosophy embodied in the current generation of light water power reactors. Because of their importance in mitigating the consequences of accidents, containments are subject to a variety of regulatory requirements covering design, operation, inspection, and testing. One element of the containment regulatory requirements that has received considerable attention is the requirement for leakage testing set forth in 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J specifies in some detail requirements for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment. These tests are designed to assure that leakage through the containment will not exceed allowable leakage rate values, which are defined in each plant's technical specifications.

The allowable containment leakage rate is determined on a plant-specific basis to meet the dose limits in 10 CFR 100, assuming a hypothetical major accident. In practice, a value lower than that required to meet the 10 CFR 100 limits is written into the plant's technical specifications. Typical allowable leakage rates are 0.1% per day for a PWR and 1% per day for a BWR.

Probabilistic risk assessments, beginning with the Reactor Safety Study, WASH-1400 (NRC 1975), have consistently shown that containment leakage is a relatively minor contributor to overall plant risk. The dominant containment-related contributions to risk stem from accidents in which the containment ruptures (due to steam explosions, overpressure, hydrogen combustion, etc.) or the containment isolation function fails or is bypassed (e.g., an interfacing systems LOCA with resulting direct release outside containment). In these dominant scenarios, containment leakage plays no significant role.

While the risk contribution due to containment leakage may be small, the cost impact of containment leakage rate testing is substantial. The primary reason for this is that integrated leak rate tests (ILRTs) of the entire containment (called Type A tests in Appendix J) are generally on the reactor outage critical path. These tests typically cause three to five days of incremental plant downtime at an estimated cost of $1.3 to $2.6 million. If this downtime could be reduced by modifying the existing regulatory requirements without compromising public health and safety, the cost savings might be substantial.

Objective of the Containment Leakage Rate 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 if the current requirements for testing containment leakage rates were modified to reduce regulatory burdens without compromising public health and safety. For purposes of this analysis, the option under consideration is the following:

- Increase the allowable leakage rate for both PWRs and BWRs to 10% per day. Sensitivity studies to show the effect of varying this numerical value are included in the analysis. The test frequency is not changed.

The underlying hypothesis is that increasing the allowable leakage rates might reduce regulatory burdens, perhaps substantially, without causing any significant adverse impact on safety.

Alternatives to Modifying Containment Leakage Rate Requirements

Current regulatory requirements pertaining to containments are complex. Furthermore, a host of technical issues involving containments and their role in reactor safety have been identified and are currently being studied in research programs worldwide. Numerous alternatives for modifying containment requirements exist and are being considered. For example, a major revision and updating of 10 CFR 50, Appendix J is pending at the NRC. A comprehensive reassessment of containment requirements, based on recent research on severe accidents, is planned. Other examples can be cited.

Although many of these other alternatives may lead to reduced regulatory burdens without adversely affecting safety, it is not possible within the scope of the present study to consider all of them. A much more extensive effort, with a different emphasis, would be needed to fully explore all of the options for modifying regulatory requirements for containment.

Consequences of Increasing the Allowable Containment Leakage Rate

A series of risk sensitivity calculations was performed under a range of conditions and assumptions to explore the effects on public health and safety of increasing the allowable leakage rate. Consideration was given to the following factors, among others:

- population dose in person-rem
- early fatalities and early injuries
- individual dose impacts (both whole body and thyroid doses) as a function of distance from the plant
- the effects of alternative source term assumptions.

Full details are contained in the body of the report. Only the most salient points will be highlighted in this summary section.

Table S.2 shows the estimated sensitivity of risk (population dose in person-rem per plant year) to leakage rate for four different cases. In each case, the risk is not very sensitive to changes in leakage rates; increasing the leakage rate to 10% would increase the calculated risk by a few person-rem per plant year. These calculations were based on the following information and assumptions:

- Accident frequencies were obtained from the Reactor Safety Study (Surry 1 and Peach Bottom 2, NRC 1975) and two probabilistic risk assessments (Oconee 3 and Grand Gulf 1, NRC 1981b) performed as part of the Reactor Safety Study Methodology Applications Program (RSSMAP).
- Risk sensitivity values were obtained from a study by Oak Ridge National Laboratory (Hermann and Burns 1984). Their analysis of containment leakage rate sensitivity used a set of generic source terms and frequencies of occurrence developed as representative of the range of LWR accidents.
- Population doses were calculated by the CRAC2 (Ritchie et al. 1983 and 1984) program using a set of standard assumptions, including a uniform population density of 340 persons per square mile within 50 miles of the plant, which represents an average population density for all US plants. Site-specific consequence analyses were not performed. The standard assumptions were those of the Handbook for Value-Impact Assessment (Heaberlin et al. 1983).

TABLE S.2. Sensitivity of Risk to Containment Leakage Rate for Four Cases

PWR Leak Rate %/day	Expected Population Dose, person-rem/reactor-year		BWR Leak Rate %/day	Expected Population Dose, person-rem/reactor-year	
	Surry 1	Oconee 3		Peach Bottom 2	Grand Gulf 1
1.0	71	207	0.5	151	250
10.0	72	210	5.0	153	254
100.0	82	238	50.0	174	288

An assessment was made of the cost impacts to the NRC and industry of increasing the allowable leakage rate. Impacts on occupational exposure were also considered. A summary of the estimated cost impacts is presented in

Table S.3. The overall impact is an estimated net cost savings of $40 million to $74 million.[a] The largest component of this is a reduction in plant downtime. With an increase in the allowable leakage rate, plants would be less likely to fail their Type A Integrated Leak Rate Tests, (current failure rates are in the neighborhood of forty to fifty percent), and the additional downtime due to test failures would be avoided, with resulting cost savings. It should be noted that this cost savings is subject to considerable uncertainty. The estimate is heavily dependent on the assumed Type A test failure rate after the allowable leakage rate is increased. This failure rate cannot be predicted precisely. Hence, the uncertainty range on the estimate is large. Other key assumptions are discussed in detail in the main report.

Calculations were also done to explore the sensitivity of individual dose, early fatalities, and early injuries to increases in leakage rate. Again, the effect was found to be small.

TABLE S.3. Summary of Cost Impacts of Increasing the Allowable Containment Leakage Rate -- Total for All Plants

Cost Category[a]	Qualitative Effect	Estimated Cost Impact,[b] Thousands of Dollars
Industry		
Implementation Costs	Cost Increase	800
Operation Costs	Cost Savings	42,000 to 76,000
NRC		
Implementation Costs	Cost Increase	1,000
Operation Costs	Cost Savings	7 to 13
Total	Cost Savings	40,000 to 74,000

(a) Implementation costs are the one-time initial costs of implementing the change. Operation costs are the recurring costs (or cost savings) over the remaining life of the plants.
(b) Costs shown are not discounted. Discounting at a 10% real discount rate would reduce the total net cost savings by approximately a factor of 3. Discounting at 5% would reduce them by approximately a factor of 2.

(a) It should be noted that these estimates are not discounted. Discounting at a 10% real discount rate would reduce them by approximately a factor of 3. Discounting at 5% would reduce them by approximately a factor of 2.

Variations in assumed source terms and other parameters were also considered. Although the details of the calculations are, of course, affected by these variations, the basic conclusion is not altered.

Conclusions -- Containment Leakage Rate Requirements

If the effects of increasing the allowable leakage rate are expressed on a dollars per person-rem basis, the ratio is on the order of several thousand dollars saved per person-rem of public exposure. This is determined as follows. A cost savings of $40 million to $74 million, discounted at a 10% real rate as suggested by the Regulatory Analysis Guidelines (NUREG/BR-0058) yields a present value of roughly $13 million to $24 million. An increase of a few person-rem per plant year (range: 1 to 5) times 120 plants (operating and planned) times 30 years (nominal average remaining plant lifetime) yields 3,600 to 18,000 person-rem. The resulting ratio could range from about $700 per person-rem (i.e., $13 million/18,000 person-rem) to about $7,000 per person-rem (i.e., $25 million/3,600 person-rem).

These ratios can be compared to the benefit-cost guideline of $1000 per person-rem that has been used in certain other contexts (i.e., the proposed safety goals and 10 CFR 50, Appendix I). However, it should be stressed that quantitative calculations of this nature, even if they are assumed applicable in this instance, are never the sole or even the principal basis for regulatory decisions. Other regulatory considerations, such as defense-in-depth, must be factored into the process. Moreover, the numerical values are highly uncertain and should be interpreted cautiously.

Main Steam Isolation Valve (MSIV) Leakage Control System

Most of the boiling water reactors (BWRs) that are currently operating and soon to be operating have been required to install leakage control systems (LCS) to control leakage past the main steam isolation valves (MSIVs) in the event of an accident. The purpose of the LCS is to collect and process (filter) any leakage of fission products past the MSIVs and thereby ensure that the radiological effects of certain postulated accidents do not exceed the numerical limits set forth in 10 CFR 100, "Reactor Site Criteria." The NRC staff's regulatory position on the MSIV leakage control system is spelled out in some detail in NRC Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." The rationale supporting the current requirements for leakage control systems is essentially deterministic. The systems are designed to ensure that the offsite dose limits in 10 CFR 100 are not exceeded under the following conditions:

- design basis loss-of-coolant accident (LOCA)
- missiles, dynamic effects (e.g., pipe whip) and environmental conditions (pressure, temperature, steam) resulting from a design basis LOCA
- an assumed single active failure concurrent with the LOCA (e.g., failure of one of the MSIVs to close)

- an assumed single failure in the leakage control system itself
- loss of all offsite power coincident with the LOCA
- occurrence of a safe shutdown earthquake coincident with the LOCA (i.e., the leakage control system is designed to seismic category I)
- fission product source term as defined in TID-14844, "Calculations and Distance Factors for Power and Test Reactor Sites," (DiNunno et al. 1962.)

Substantial elements of conservatism inherent in this deterministic approach have been recognized for a long time. In recent years, improvements in the data and methods available for measuring risks have provided additional insights into the benefits of MSIV leakage control systems, both in an absolute sense and relative to other systems designed to protect the health and safety of the public. Estimates of the benefits of MSIV leakage control systems, based on probabilistic risk assessment techniques, indicate that the benefits are marginal at best, and that implementation of such systems is difficult to justify from the standpoint of cost-effectiveness. The basis for these conclusions is documented in the body of the report. The next sections of this summary highlight the more salient points.

Objectives of the MSIV Leakage Control Systems 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 MSIV leakage control systems. For purposes of this analysis, the option under consideration is to:

- eliminate the requirement for MSIV leakage control systems (i.e., eliminate NRC Regulatory Guide 1.96, Standard Review Plan Section 6.7 (NRC 1981a), and make conforming changes in other regulatory documents such as technical specifications and 10 CFR 50, Appendix J)
- disable the leakage control systems in plants that currently have them (or will have them).

Alternatives to Modifying MSIV Leakage Control System Requirements

There are a number of complex technical issues surrounding the requirements for MSIV leakage control systems. Each of the technical issues, in turn, gives rise to a number of regulatory alternatives. A comprehensive examination of the full range of issues has been conducted as part of a large-scale, multi-year effort to resolve Generic Safety Issue C-8, "MSIV Leakage and LCS Failure." A report on the resolution of Generic Issue C-8 has been prepared, NUREG-1169 (Ridgely and Wohl 1986). Among the issues considered in NUREG-1169 are:

- methods for reducing MSIV leakage (thus reducing the need for leakage control systems)
- allowable MSIV leakage rates
- alternative methods/pathways for mitigating the consequences of MSIV leakage
- analytical methods for more accurately calculating the consequences of MSIV leakage (taking account of such factors as fission product deposition and decay).

The scope of the present report is more limited, since the intent is to provide information on the effects of eliminating or relaxing current requirements.

Consequences of Eliminating Requirements for MSIV Leakage Control Systems

Risk sensitivity calculations were performed to estimate the change in risk that could result if MSIV leakage control systems were eliminated. With the Grand Gulf 1 BWR as the reference case (Hatch et al. 1981) event trees were constructed to model fission product leakage scenarios following core-melt accidents. Conservative (i.e., optimistic) assumptions were made for the effectiveness of the leakage control system vis-a-vis the alternative (no system). Even with these optimistic assumptions, the risk reduction attributable to the MSIV leakage control system was estimated as 0.3 person-rem per reactor year.

Two qualitative insights should be noted in order to place this estimate in perspective. First, even in the absence of a leakage control system, MSIV leakage is a small contributor to overall plant risk (on the order of 2 person-rem per reactor year in the calculations presented in this report); so there is only limited risk reduction to be achieved even by a highly effective leakage control system. Second, the MSIV leakage control system is effective only to a limited degree. It eliminates about 15% of the risk contribution due to MSIV leakage (0.3 person-rem/2 person-rem). The reason for this limited effectiveness is that the LCS is effective only when the leakage is less than about 100 standard cubic feet per hour (SCFH); the system is not effective when large leakages (on the order of 1000 SCFH) occur. These large leakage scenarios, although they have low probability, have relatively large consequences, and are the dominant contributor to the risk due to MSIV leakage.

These findings concerning the effectiveness of the LCS are consistent with the results of the recently completed study by Ridgely and Wohl (1986) of Generic Issue C-8 (NUREG-1169). Among other conclusions, the key findings of the C-8 report are:

- At most plants there are alternative MSIV leakage paths that do not depend on the availability of offsite power and that are at least as effective as the LCS systems presently required.

- Alternative pathways for MSIV leakage control that take advantage of the condenser holdup volume are extremely effective in mitigating the offsite radiological consequences of an MSIV failure to close; this is true even if offsite power is lost.

- In the attempt to meet the current strict MSIV leakage requirements, utilities have sometimes performed excessive maintenance on valves. In some cases, this maintenance has damaged the valves (e.g., seat refurbishment in situ has resulted in out-of-round seats) without providing any substantial safety benefit.

- From the PRA analyses examined, the requirement for a safety-grade LCS could not be defended on a value-impact basis using a value of $1000/person-rem saved.

On the cost side of the ledger, estimates were obtained for the industry and NRC cost impacts that could result if MSIV leakage control systems were eliminated. Because the risk reduction due to the LCS was found to be small, it was not necessary to quantify the costs with great precision. For industry, the cost to procure and install an LCS was estimated as $500,000 per plant (initial cost). Operating costs for maintenance and surveillance were estimated at $20,000 per reactor year. Therefore, if the requirement for the LCS were eliminated, an operating cost savings of $20,000 per reactor year could be achieved. The $500,000 initial system cost, on the other hand, is a sunk cost and would not be affected in any way unless some BWRs still under construction have not yet acquired the systems.

Implementing the change in regulatory requirements would entail some additional costs both for industry and for the NRC. Physically disabling the LCS could cost several thousand dollars (nominal estimate: one man-week of effort or $2,000). Changing plant technical specifications and other documentation is estimated to cost industry about $10,000 per plant. This would be a one-time cost. NRC costs for the technical specification changes would be about the same (nominal estimate: $11,000 per plant). The cost for the NRC to develop and implement the revised staff position (i.e., eliminate Regulatory Guide 1.96, and conforming changes in other documents, such as the Standard Review Plan and Appendix J) was estimated at up to $500,000, although with the imminent completion of staff work on Generic Issue C-8, an estimate of 1/2 man-year, or $50,000, to prepare recommendations, a regulatory analysis and supporting documentation may be more realistic.

Conclusions -- MSIV Leakage Control Systems

If the MSIV leakage control system were a new requirement to be evaluated under current procedures and policies such as the revised backfit rule or the proposed safety goals, it could not be justified on the basis of a quantitative benefit-cost guideline of $1000 per person-rem; conservatively calculated, the ratio for the LCS is on the order of $100,000 per person-rem.

If one considers eliminating the systems, the calculations are slightly different. First, the large initial cost of the system is now a sunk cost for the current generation of plants. Second, there would be a non-trivial initial cost to effect the regulatory changes: 25 plants times $12,000 per plant for the industry (i.e., $300,000); for the NRC, $50,000 for revising the staff position plus $250,000 for technical specification changes (i.e., $300,000). After this initial outlay of $600,000, savings would then accrue to industry at the rate of $500,000 per year (25 plants times $20,000 per reactor year in operating savings) over the remaining life of the current plants. Adding all the costs and cost savings, and discounting future cost savings at a 10% real discount rate (as suggested in NUREG/BR-0058, NRC 1984a) over 30 years, the net monetary benefit is

$$\$500{,}000 \times 9.43 - \$600{,}000 = \$4.1 \text{ million.}$$

The increase in risk to the public is estimated at 7.5 person-rem per year (0.3 person-rem per reactor year times 25 reactors). This works out to $18,000 saved per person-rem of dose increase.

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.

Fuel System Safety Reviews

A fundamental concept in the design of nuclear power plants is the provision of multiple fission product barriers to protect the health and safety of the public from releases of radioactive material during normal operations and under accident conditions. The first of these multiple barriers is provided by the fuel cladding in the fuel system. Fuel system components include the fuel rods (including pellets, cladding, springs, end plugs, fill gas, etc.), burnable poison rods, control rods and various associated hardware such as spacer grids, springs, end plates and channel boxes. Because of its role as the first line of defense in the defense-in-depth design philosophy, the licensee's fuel system design is carefully reviewed by the NRC staff to ensure compliance with applicable regulatory requirements.

Procedures for fuel system safety reviews by NRC staff are set forth primarily in Section 4.2 of the Standard Review Plan, NUREG-0800 (NRC 1981a). The reviews address a large number of complex technical issues that have been identified over the years. An overview of these review procedures is given in the body of this report.

With about one thousand reactor operating years of experience in the United States, and on the order of four thousand reactor operating years worldwide, the technology of fuel design is mature. Given the accumulated experience and the sophistication of current analytical models and design practices, it is conceivable that the NRC's reviews of fuel system design information submitted by licensees could now be streamlined and simplified without adversely affecting safety. To test this hypothesis, this report considers the consequences of eliminating some steps from the current review procedures.

Objective of the Fuel System Safety Reviews 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 could result if current procedures for fuel system safety reviews were streamlined. Since complete elimination of all NRC fuel system safety reviews is not thought to be a reasonable option, this analysis postulates a graded approach, i.e., elimination of certain items with marginal significance to risk and continued careful review of items that are risk-significant.

Alternatives to Modifying Fuel System Safety Review Requirements

It should be recognized that the NRC staff already follows a graded approach to some degree in its fuel system safety reviews. For example, only designs or portions of analytical methods that have changed from previously approved designs or analytical methods are reviewed. Similarly, the level of effort spent in reviewing particular items is informally graded so that the time spent is commensurate with the item's importance to fuel system safety. Essentially, the option being considered in this report is primarily a formalization and to a lesser degree an extension of current staff practices.

Consequences of Streamlining Fuel System Safety Review Procedures

Fuel system safety reviews are structured according to twenty-one fuel damage and fuel failure mechanisms that can contribute to fuel system damage, fuel rod failure or loss of fuel coolability. The acceptance criteria for these damage and failure mechanisms are referred to as Specified Acceptable Fuel Design Limits (SAFDLs). The question addressed in this report is the following: What would be the consequence, in terms of risks, costs and benefits, if some SAFDLs (and the corresponding evaluation methods) were eliminated from the fuel system safety review process?

To assess the potential safety consequences, a qualitative analysis was performed, with some limited quantitative analysis provided for additional perspective. Each SAFDL was assigned to one of three categories graded according to their importance to safety. The categorization was based on the engineering judgment of PNL staff members who have extensive background and experience in fuel design and performance and who also have experience with the NRC fuel safety review procedures set forth in Section 4.2 of the Standard Review Plan (NRC 1981a). From the results of the categorization, it appears that some of the SAFDLs have relatively minor safety significance. Eliminating them from the current fuel system safety review process would be expected to have marginal effect on public health and safety.

The industry and NRC cost impacts of eliminating some SAFDLs were also examined in a semi-quantitative manner. For the industry costs, discussions were held with a number of utilities and industry groups, including fuel vendors. The consensus of those contacted was that streamlining the review process by eliminating certain SAFDLs would have minimal effect on industry costs. In other words, no significant cost savings would be expected, although some small cost savings might be achieved in certain situations. The basic reason for this conclusion is that industry would continue to design fuel in the same way. Current fuel design practices have developed and matured over the years and they work reasonably well, meeting the needs of the vendor, the utility, and the NRC. In the absence of some significant improvement in economics or performance, there is little incentive for industry to change current practices.

On the NRC side, the cost savings from streamlining the reviews would depend on the number of SAFDLs eliminated from the review process. Rough estimates were made of the percentage cost savings that might be achieved in various circumstances and they range from negligible to small or moderate, depending on the situation. It is difficult to convert these percentages to an overall estimate of dollar savings because of the variability and uncertainty in the time and resources devoted to reviews. However, in relation to the overall NRC review process, the savings would be very small.

Conclusions -- Fuel System Safety Reviews

Based on the information presented in this report, it appears that some steps in the current fuel system safety review procedures could be eliminated without compromising public health and safety. However, the benefits of doing so would be marginal. In relation to the overall costs associated with the current fuel design process, the cost savings for both industry and NRC would be very small.

As noted earlier, the NRC staff already follows a graded approach to some degree in its fuel system safety reviews. Although there may be some opportunities to achieve additional efficiencies by formally eliminating some SAFDLs, there do not seem to be any strong incentives to depart from the current, informal graded approach.

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FIGURES

TABLES

LIST OF ACRONYMS

ANS	American Nuclear Society
ANSI	American National Standards Institute
ATWS	anticipated transient without scram
BV	bypass valve
BWR	boiling water reactor
CRAC	calculations of reactor accident consequences
CV	control valve
ECCS	emergency core cooling system
EPA	Environmental Protection Agency
FSAR	final safety analysis report
ILRT	integrated leak rate tests
LCS	leakage control system
LOCA	loss of coolant accident
LWR	light water reactor
MSIV	main steam isolation valve
MSIV-LCS	main steam isolation valve - leakage control system
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OL	operating license
OR	operating reactors
PNL	Pacific Northwest Laboratory
PPG	Policy and Planning Guidance
PRA	Probabilistic Risk Assessment
PWR	pressurized water reactor

RPV	reactor pressure vessel
RSSMAP	Reactor Safety Study Methodology Applications Program
SAFDL	specified acceptable fuel design limits
SBGT	standby gas treatment
SCFH	standard cubic feet per hour
SEA	Science and Engineering Associates
SRP	Standard Review Plan
SRV	safety relief valve
SWGTS	steam and waste gas treatment system
SV	stop valve
TMI	Three Mile Island (Nuclear Plant)

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) has 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. In a Federal Register notice 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 describes the results of a study conducted by Pacific Northwest Laboratory (PNL) to assist the NRC in executing the first phase of the program. The report presents information on the costs and benefits of streamlining current requirements in three areas:

- reactor containment leakage rates
- Main steam isolation valve (MSIV) leakage control systems in boiling water reactors (BWRs)
- NRC licensing review of fuel design information.

These areas of regulation were selected by NRC staff for examination in the first phase of the regulatory review program. The objective of these analyses is twofold. First, the technical information will be considered by the NRC staff in formulating recommendations on whether the three areas of regulation should be modified, and if so, how they should be modified. Second, the analyses are intended also to demonstrate the technical methods and tools needed to reexamine existing regulatory requirements and determine whether they might be eliminated or streamlined to reduce regulatory burden without adversely affecting the public health and safety.

1.1 BACKGROUND

On October 3, 1984, the NRC published a notice in the Federal Register (Vol. 49, No. 193, p. 39066) announcing a new program to review the effectiveness of existing light water reactor regulatory requirements in limiting risk. The program was initiated in response to guidance received from the Commission in its Policy and Planning Guidance (PPG) for 1984, NUREG-0885 (NRC 1984b) and specific programmatic direction from the Executive Director for Operations. The Commission, in the section of its Policy and Planning Guidance entitled "Improving Regulation of the Nuclear Industry," stated:

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

The PPG for 1985, NUREG-0885 (NRC 1985), reiterated this objective:

> "Existing regulatory requirements should be reviewed to see if some could be eliminated without compromising safety."

As part of the program guidance developed in support of the Commission's PPG, the Executive Director for Operations called for a three-pronged effort to systematically review existing regulations. The 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, Pacific Northwest Laboratory (PNL) was asked to provide technical information and analyses to support the NRC staff in its work. PNL's work in FY 1985 consisted of two principal tasks:

1. Identification of regulatory requirements that might be relaxed or eliminated to reduce regulatory burdens without compromising the public health and safety. The purpose of this task was to screen the existing regulatory requirements and guidance associated with 10 CFR 50 and identify a set of candidates for further detailed study.

2. Detailed analysis of several regulatory requirements selected by NRC staff. The purpose of this task was twofold. First, the task was to produce technical information for the NRC staff to consider in deciding whether the selected requirements could be eliminated or relaxed to reduce regulatory burdens without compromising safety. Cost-benefit assessments of the consequences of changing or eliminating the requirements are an important part of this technical information. Second, the task was also intended to demonstrate the assessment methods and tools needed to provide a technical information base for NRC regulatory decisions concerning the effectiveness of existing regulatory requirements in limiting risk.

This report presents the results of Task 2. A companion report covers Task 1 (NUREG/CR-4330, Volume 1, Mullen et al. 1986).

The idea of reviewing existing regulatory requirements to assess their efficacy and continued importance is not new, of course, nor is it unique to the nuclear regulatory context. Such reassessments are a natural consequence of:

- industry operating experience

- new information and methods for the measurement of risks, costs and benefits

- improvements in knowledge of the regulated technology.

Such reassessments, sometimes referred to as "sunset reviews," played a prominent role in legislative proposals for regulatory reform in the 1970s and have been pursued by the Environmental Protection Agency (EPA), for example. In the nuclear field, many individuals and groups have made suggestions along these lines. Probabilistic risk assessment (PRA) techniques have been prominently featured in most of the proposals, inasmuch as PRA provides a systematic quantitative approach for appraising the benefits (in terms of risk reduction) of regulations. A systematic risk-based review of the regulations has the potential to both strengthen and streamline the regulatory structure. The risk-based approach to reviewing existing regulatory requirements is considered in this report. The costs and benefits of selected modifications to the existing requirements are discussed and estimates based on PRAs are presented.

1.2 TECHNICAL APPROACH

This report covers PNL's work on the second of the two tasks mentioned above, namely the analysis of three 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 1984a) and the Handbook for Value-Impact Assessment NUREG/CR-3568 (Heaberlin et al. 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 contents 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.

1.3 CONTENTS OF THIS REPORT

Following this introductory chapter, Chapter 2 presents PNL's analysis of the risks, costs, and benefits of relaxing current regulatory requirements for containment leakage rate. The primary focus is the allowable leakage rate limit. For purposes of the analysis, an increase of this limit to 10% per day is postulated. Chapter 3 covers MSIV leakage control systems for BWRs. The option considered is elimination of the requirement and disabling the leakage control systems in reactors that currently have them. Chapter 4 addresses the current NRC procedures for fuel system safety reviews, which are defined in Section 4.2 of the Standard Review Plan, NUREG-0800 (NRC 1981a). The option considered in the analysis is to selectively eliminate from the current review process certain items that may have marginal risk importance. Two appendixes to the report provide supporting information for the analyses in Chapter 2.

2.0 RISK AND COST IMPACTS FOR NUCLEAR REACTOR CONTAINMENT LEAKTIGHTNESS

Reactor containments constitute one of the principal lines of defense in the defense-in depth design philosophy embodied in the current generation of light water power reactors. Several mechanisms can result in releases from containment. These include gross failure of containment due to the pressure forces resulting from an accident, containment base-mat melt-through, failure of containment isolation systems, and releases as a result of containment leakage. Probabilistic risk assessments, beginning with the Reactor Safety Study, WASH-1400 (NRC 1975), have shown that containment leakage (at, or slightly above the design leakage rate) is a relatively minor contributor to overall nuclear reactor risk. The dominant containment-related contributions to risk stem from accidents in which the containment ruptures (due to steam explosions, overpressure, hydrogen combustion, etc.) or the containment isolation function fails or is bypassed (e.g., an interfacing systems LOCA with resulting direct release outside containment). While the risk contribution due to containment leakage may be small, the cost impact of containment leakage testing is substantial.

2.1 CURRENT REGULATORY REQUIREMENTS

Because of their importance in mitigating the consequences of accidents, containments are subject to a variety of regulatory requirements covering design, operation, inspection and testing. One element of the containment regulatory requirements that has received considerable attention is 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors". Appendix J specifies in some detail the requirements for preoperational and periodic operational tests to verify the leaktight integrity of the primary reactor containment. These tests are designed to assure that leakage through the containment will not exceed the allowable leakage rate values defined in each reactor's technical specifications.

The allowable leakage for a reactor primary containment is determined on a plant specific basis to meet 10 CFR 100 dose limits after a specified design basis accident. In practice, a value lower than that required to meet the 10 CFR 100 limits is written into the plant's technical specifications. Typical allowable leakage rates are 0.1% per day for a PWR and 1% per day for a BWR.

2.2 ALTERNATIVES TO REGULATORY REQUIREMENTS

Current regulatory requirements pertaining to containments are complex and a host of technical issues involving containments and their role in reactor safety have been identified and are currently being studied in research programs worldwide. Numerous alternatives for modifying containment requirements are being considered. Existing requirements on containment leakage testing in 10 CFR 50, Appendix J are undergoing revision. Containment performance under severe accident conditions is also under investigation by the NRC.

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, cost and benefits that would result from modifications to current containment regulatory requirements. As discussed above, many potential alternatives exist. The option considered in this analysis is increasing the allowable leakage rates for both PWRs and BWRs. A range of numerical values are considered.

The option considered in this study is thought to be an appropriate illustration of the potential risks, costs and benefits that could result if certain requirements are judiciously modified or eliminated. A much more extensive effort, with a different emphasis, would be needed to fully explore all of the options for modifying containment regulatory requirements. The remainder of this section examines on a preliminary basis the risk and cost impacts associated with increasing the allowable containment leakage rates.

2.3 RISK IMPACTS

Two basic approaches are used in this study to examine the risk impacts associated with increasing the allowable containment leakage. The first takes advantage of several existing probabilistic risk assessments (PRAs) and calculates the incremental risk due to increasing the allowable containment leakage rates. The risk measure used in this first approach is expected person-rem (i.e., the probability of an accident multiplied by its consequences in terms of person-rem to the surrounding population). The second approach examines selected accident sequences and considers several additional measures including individual radiation exposures and early health effects.

2.3.1 Risk Impacts of Containment Leakage Rates Using Existing PRAs

As discussed earlier, several mechanisms can result in releases from containment. These include gross failure of containment from pressure forces resulting from the accident, failure of the containment isolation systems, and release as a result of leakage. Accident sequences identified in a probabilistic risk assessment are typically grouped into sets determined by the magnitude of their associated radioactive release. These groupings are the release categories. Each category is often distinguished by the containment failure mode.

Most PRAs use the release categories defined by the Reactor Safety Study, WASH-1400. For convenience, short descriptions of each of these categories are provided in Appendix A. As seen from these descriptions, categories PWR-1 through PWR-3 and BWR-1 through BWR-3 include containment failure from overpressure; categories PWR-4, PWR-5, PWR-8 and BWR-4 include containment failure to isolate; and categories PWR-6, PWR-7, PWR-9 and BWR-5 include design leakage rate. (For PWRs, a 1% per day leakage rate was typically used, while for BWRs, the corresponding value was typically 0.5% per day.)

The approach used in this analysis is to examine the release categories and their associated frequencies identified by selected PRAs and determine the impact of increasing the design leakage rate. The following gives the results of this analysis. Key assumptions and uncertainties are examined in a subsequent section.

2.3.1.1 PRA Results

Four reactors for which probabilistic risk assessments have been performed are used as the basis of this analysis. These include:

- Surry 1 Subatmospheric Containment (WASH-1400)
- Peach Bottom 2 Mark I Containment (WASH-1400)
- Oconee 3 Large Dry Containment (NUREG/CR-1659)
- Grand Gulf Mark III Containment (NUREG/CR-1659)

To minimize the effect of differences in such site characteristics as meteorology and population, the public dose consequence factors developed by Andrews et al. (1983) and Heaberlin et al. (1983) for the WASH-1400 release categories are used to estimate population doses. The computer program CRAC2 was applied to a typical midwest site (Braidwood). The calculations used the following assumptions and parameters:

- Dose consequences are represented by the whole body population dose commitment (person-rem) received within 50 miles of the site. The use of a 50-mile radius is consistent with assumptions incorporated in the proposed safety goals.[a]
- An exclusion area of 1/2 mile is assumed with uniform population density of 340 persons per square mile beyond 1/2 mile. (The assumed population density represents an average for all US plants).
- Evacuation of people is not considered. (Sensitivity of the results to this assumption is examined in section 2.3.2.1).
- All exposure pathways are included for non-core-melt sequences (PWR-8 and 9, and BWR-5). For core-melt sequences all exposure pathways except ingestion pathways are included.

(a) A number of studies (e.g., NUREG/CR-2239, Aldrich et al. 1982) indicate that total latent cancer fatalities (i.e., including those beyond the 50-mile radius) would differ from the latent cancer fatalities within the 50-mile radius by a factor of two or less for the population distribution surrounding the Indian Point Plant. If uniform population densities are assumed, the total latent cancer fatalities may be 2 to 5 times greater than those expected within the 50-mile radius.

- Farmland usage parameters for the state of Illinois are used for non-core-melt ingestion pathway calculations.

- Meteorological data is taken from the U.S. National Weather Service station at Moline, Illinois. CRAC2 uses weighted values of wind speed and direction, stability class, precipitation, etc. pertaining to the selected weather station. There may be a large stochastic variation in results associated with the actual meteorology at the time of a radiological release.

- The core inventory at the time of the accident is assumed to be represented by a 3412 MWt (1120 MWe) PWR.

The resulting public dose consequence factors are given in Table 2.1. Appendix A provides a brief description of each release category.

TABLE 2.1. Population Doses for WASH-1400 Release Categories[a]

Release Category	Population Dose person-rem
PWR-1	5.4E6
PWR-2	4.8E6
PWR-3	5.4E6
PWR-4	2.7E6
PWR-5	1.0E6
PWR-6	1.5E5
PWR-7	2.3E3
PWR-8	7.5E4
PWR-9	1.2E2
BWR-1	5.4E6
BWR-2	7.1E6
BWR-3	5.1E6
BWR-4	6.1E5
BWR-5	2.0E1

(a) These baseline calculations assume a leakage rate of 1%/day for PWRs and 0.5%/day for BWRs.

Each of the probabilistic risk assessments was reviewed to obtain information on release frequencies and release categories. This information was then combined with the public dose consequence factors discussed above to obtain risk estimates. All of these estimates are based on WASH-1400 source terms. The release category, frequency, population dose, and expected population dose (risk) information for the four plants described are given in Tables 2.2 and 2.3.

TABLE 2.2. Surry 1 and Peach Bottom 2 Risk Information Summary[(a)]

Release Category	Frequency, per year	Population Dose, person-rem	Expected Dose (Risk), person-rem per year
		Surry 1	
PWR-1	9E-7	5.4E6	4.86
PWR-2	8E-6	4.8E6	38.40
PWR-3	4E-6	5.4E6	21.60
PWR-4	5E-7	2.7E6	1.35
PWR-5	7E-7	1.0E6	0.70
PWR-6	6E-6	1.5E5	0.90
PWR-7	4E-5	2.3E3	0.09
PWR-8	4E-5	7.5E4	3.00
PWR-9	4E-4	1.2E2	0.05
			71.0 Total
		Peach Bottom 2	
BWR-1	1E-6	5.4E6	5.40
BWR-2	6E-6	7.1E6	42.60
BWR-3	2E-5	5.1E6	102.00
BWR-4	2E-6	6.1E5	1.22
BWR-5	1E-4	2.0E1	0.002
			151 Total

(a) These baseline calculations assume a leakage rate of 1%/day for PWRs and 0.5%/day for BWRs.

TABLE 2.3. Oconee 3 and Grand Gulf 1 Risk Information Summary[a]

Release Category	Frequency, per year	Population Dose, person-rem	Expected Dose (Risk), person-rem per year
		Oconee 3	
PWR-1	1.1E-7	5.4E6	0.59
PWR-2	1.0E-5	4.8E6	48.0
PWR-3	2.9E-5	5.4E6	156.6
PWR-4	9.7E-8	2.7E6	0.26
PWR-5	4.6E-7	1.0E6	0.46
PWR-6	7.3E-6	1.5E5	1.1
PWR-7	3.5E-5	2.3E3	0.08
			207 Total
		Grand Gulf 1	
BWR-1	1.1E-7	5.4E6	0.59
BWR-2	3.4E-5	7.1E6	241.4
BWR-3	1.4E-6	6.1E5	7.14
BWR-4	1.6E-6	6.1E5	0.98
			250 Total

(a) These baseline calculations assume a leakage rate of 1%/day for PWRs and 0.5%/day for BWRs.

Hermann and Burns (1984) examined the risks from LWR accidents as a function of containment leakage rates. Their analysis used a set of generic source terms and frequencies of occurrence developed as representative of the range of LWR accidents. An accident-spectrum-weighted impact fraction was formulated as the sum of fractional increases in consequences due to containment leakage for each type of accident weighted by its frequency of occurrence. A value of 1.5E-3 fractional increase in risk per percent/day containment leakage rate was obtained. This value is applied to the above PRA information and the results are presented in Table 2.4.

TABLE 2.4. Sensitivity of Overall Plant Risk to Containment Leakage Rate

PWR Leak Rate %/day	Expected Population Dose, person-rem/reactor-year		BWR Leak Rate %/day	Expected Population Dose, person-rem/reactor-year	
	Surry 1	Oconee 3		Peach Bottom 2	Grand Gulf 1
1.0	71	207	0.5	151	250
10.0	72	210	5.0	153	254
100.0	82	238	50.0	174	288

Figure 2.1 graphically indicates that the overall plant risk is not very sensitive to changes in containment leakage rates. A key assumption in the above analyses is that the preexisting leakage does not influence the accident sequence propagation (i.e., it does not influence the pressure/temperature conditions or result in equipment failures). For leakages in the range of 100% per day or larger this assumption may not be valid. These scenarios involving larger leakages will be similar to the failure-to-isolate scenarios. The effect of these larger leakages on pressure/temperature conditions and/or equipment failures must be modeled to determine the risk impact.

The above results indicate that the risk effects of containment leakage are small when compared with the risk of a large release due to other containment failure modes. To examine these effects further, a simple analysis is performed using the Surry 1 and Oconee 3 PRA results. Design containment leakage primarily contributes only to the PWR-6, PWR-7, and PWR-9 release categories. For Surry 1 these categories were estimated to contribute 1.04 expected person-rem out of a total of 71. For Oconee 3 these categories contribute 1.18 expected person-rem out a total of 207. The releases calculated for PWR-6 and PWR-7 release categories assume a 1%/day leakage rate. Conservatively assuming that the resulting radiation dose scales linearly with the leakage rate one obtains the values given in Table 2.5.

TABLE 2.5. Sensitivity of Risk to Leakage Using Scaling Method

PWR Leak Rate %/day	Expected Population Dose, person-rem per reactor year	
	Surry 1	Oconee 3
1.0	71	207
10.0	81	219

These values represent upper limit estimates and include some double counting due to the puff release when the containment basemat fails for accident sequences in the PWR-6 and PWR-7 release categories. The values in Table 2.4 are felt to be more representative.

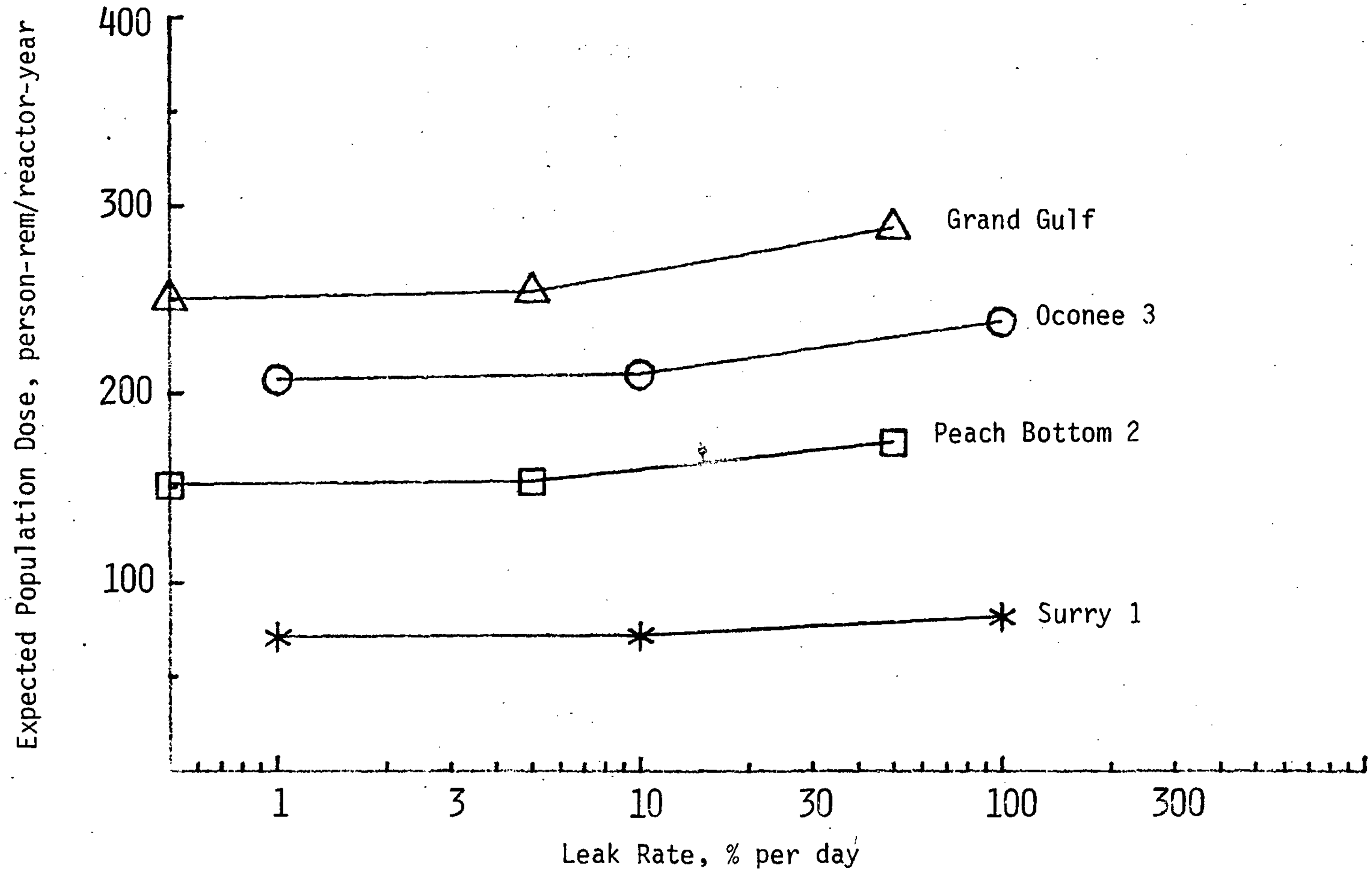


FIGURE 2.1. Sensitivity of Risk to Containment Leakage

The results presented in this section show that LWR accident risk is relatively insensitive to the containment leakage rate because the risk of accidents which cause containment to fail from overpressure and/or involve failure to isolate containment dominate those in which containment does not fail but leaks. Containment leakage rates in the range of 10% per day appear to have a relatively small risk impact. This risk impact is in the range of a few person-rem per year for the four plants examined using the Oak Ridge National Laboratory results.

2.3.1.2 Discussion of Assumptions and Uncertainties

The above results are based on the WASH-1400 source term. To examine the sensitivity of these results to the source term used, the Battelle Columbus study of the Surry facility was reviewed (Gieseke et al. 1984). That study only examined a limited number of scenarios and concluded that for many important accident sequences the source term in WASH-1400 is overestimated by approximately an order of magnitude. One major exception is the interfacing LOCA (or V) sequence for which considerable uncertainty exists regarding the release pathway (i.e., whether or not it will be through a pool of water). Two cases are examined using these results: 1) an across the board decrease in release of a factor of 10 for each WASH-1400 release category and 2) a decrease in release of a factor of 10 for each release category except the interfacing LOCA (or V) sequence, which remains unchanged. For the first case the values provided for Surry 1 in Table 2.2 would simply be divided by a factor of 10 giving a total risk of 7.1 person-rem per year. Table 2.6 presents the results for no reduction in the V sequence source term. Note that the V sequence is in PWR-2 and the frequency of this sequence is 4E-6 per year. The frequency of PWR-2 in Table 2.6 has been revised to this value.

Using the same approach as before, design leakage rate primarily contributes to the PWR-6, PWR-7 and PWR-9 release categories. For the revised source term these categories contribute 0.10 expected person-rem out of 7.1 for the factor of 10 reduction in the WASH-1400 source term and 0.104 expected person-rem out of 22.5 for the factor of 10 reduction except for the V sequence. The releases calculated for PWR-6 and PWR-7 release categories assume a 1% per day leakage rate. Conservatively assuming that the resulting radiation dose scales linearly with the leakage rate, the risk impact of a 10% leakage rate is found to be 1 person-rem per year for both cases. This results in an increase from 7.1 to 8.1 person-rem per year for the first case and an increase from 22.5 to 23.5 for the V sequence case. Unless a plant has specific sequences in the release categories where containment leakage is important (e.g., PWR-6, PWR-7 and PWR-9) and whose source term is not significantly reduced, the decrease in the risk impact of containment leakage with a lower source term would appear to be generally true.

The analyses in this section use expected person-rem (probability of an accident times its consequence expressed in person-rem) as a risk measure. This risk measure can be directly related to latent health effects by conversion factors. Potential early health effects cannot be directly related to a person-rem population dose. However, if the risk in terms of person-rem

TABLE 2.6. Surry 1 (Factor of 10 Decrease In Source Term Except V Sequence)[a]

Release Category	Frequency, per year	Population Dose, person-rem	Expected Dose (Risk), person-rem per year
PWR-1	9E-7	5.4E5	.49
PWR-2	4E-6	4.8E6	19.2
PWR-3	4E-6	5.4E5	2.16
PWR-4	5E-7	2.7E5	.14
PWR-5	7E-7	1.0E5	.07
PWR-6	6E-6	1.5E4	.09
PWR-7	4E-5	2.3E2	.009
PWR-8	4E-5	7.5E3	.30
PWR-9	4E-4	1.2E1	.005
			22.5 Total

(a) These calculations assume a leakage rate of 1%/day for PWRs.

is small then the risk in terms of early health effects is also expected to be small. The use of other risk measures is examined in Section 2.3.2.

The consequence modeling assumptions used in this section include a uniform population distribution out to 50-miles and no evacuation. Changes in these assumptions may change the absolute values of the risk but will have little impact on relative comparisons. Since a person-rem risk measure was used, consideration of evacuation will have minimal impact on the risk values. Because evacuation may be important in the consideration of early health effects, these are examined in Section 2.3.2.

The risk impact of leakage is sensitive to the failure thresholds assumed for the containment structure. The four risk assessments reviewed in this study used relatively conservative containment failure pressures based upon the approach in WASH-1400. Recent work being conducted by Sandia and other organizations indicates that reactor containments may fail at higher pressures than assumed in existing risk assessments. The time and funding constraints of this study did not permit a detailed examination of the sensitivity of the risk impact of leakage to containment failure thresholds. Given the potential for containment bypass events and containment isolation failures, it is believed that the risk impact of containment leakage rates will not be significant even if higher containment failure thresholds are assumed.

The above results show that LWR accident risk is relatively insensitive to the containment leakage rate because the risk is dominated by accident sequences which result in failure of containment from overpressure. The incremental risk from leakage in the range of 1 to 10% per day is small. The current leakage rate requirements of many plants are 0.1% per day. The results of this preliminary study indicate that incentives exist to re-evaluate these requirements as to their risk significance. However, accident risk is only one of the considerations relating to containment integrity. Among others is the effect of containment leakage on the consequences of routine airborne effluents released during reactor operations (e.g., leaks from reactor coolant pump seals, leaks from vents). Requirements in this area may dictate lower leakage rates. Another consideration is the importance of containment in the defense in depth philosophy. Still another is the effect of containment leakage on maximum individual exposure. This last factor is considered in more detail in the following section.

2.3.2 Impacts of Leakage on Selected Accident Scenarios

This section analyzes two PWR and two BWR accident scenarios from WASH-1400 and a hypothetical scenario related to the Three Mile Island (TMI) accident to indicate the impacts of various containment leakage rates for the selected accident scenarios.

2.3.2.1 PWR Scenarios

Two PWR scenarios are selected from WASH-1400 and used to examine the potential risk impacts of containment leakage. These scenarios include AHϵ and AHFϵ where:

A = Large Pipe Break
H = Emergency Core Cooling Recirculation Failure
F = Containment Spray Recirculation System Failure
ϵ = Containment Failure by Basemat Meltthrough

The primary difference between AHϵ and AHFϵ is the fact that the containment spray system does not operate for AHFϵ. AHϵ is a PWR-7 release and AHFϵ is a PWR-6 release. As discussed in the previous section, the design leakage is expected to have a significant impact on these release categories. The core-melt timing, the release to containment, and the containment leakage values are taken from the CORRAL results presented in Appendix V of WASH-1400. The bulk of the release occurs during the period of 1.5 to 5.5 hours. Table 2.7 summarizes these results for this four-hour release period.

The releases from Table 2.7 were input to CRAC2 using the assumptions described in the previous section. The CORRAL results assumed a 1% per day leakage rate and the CORRAL results for this leakage rate were linearly scaled to obtain values for 10% per day and 100% per day leakage rates. Table 2.8 presents the public consequences from sequences AHϵ and AHFϵ as a function of the design leakage rate.

TABLE 2.7. Release from Containment for AHε and AHFε Sequences

Sequence	Leakage Rate, % per day	Accident Timing	4-Hour Release, fraction of core	
AHε	1	100 min. Core Melt Starts	Xe-Kr	2E-4
		210 min. Vessel Meltthrough	I-Br	2E-6
		230 min. Cont. Meltthrough starts	Cs-Rb	7E-6
		1290 min. Cont. Meltthrough ends	Te	5E-6
			Ba-Sr	7E-7
			Ru	4E-7
			La	5E-8
AHFε	1	Same As Above	Xe-Kr	1E-3
			I-Br	6E-4
			Cs-Rb	4E-4
			Te	3E-4
			Ba-Sr	5E-5
			Ru	3E-5
			La	4E-6

TABLE 2.8. Consequences to the Public from AHε and AHFε Versus Leakage

Sequence	Leakage Rate, % per day	Frequency per year	Consequences, person-rem	Early Fatalities	Early Injuries
AHε	1.0	1E-6	1E3	0	0
	10.0	1E-6	1E4	0	0
	100.0	1E-6	1E5	0	0
AHFε	1.0	1E-10	7E4	0	0
	10.0	1E-10	5E5	0	1
	100.0	1E-10	3E6	31	93

As mentioned earlier, AHε is in PWR-7 and AHFε is in PWR-6. The consequences in Table 2.8 can be compared with those of PWR-6 and PWR-7. A PWR-6 release results in 100,000 person-rem and no early fatalities or injuries. A PWR-7 release results in 2,000 person-rem and no early fatalities or injuries. As expected, these values are similar for the 1% per day leakage values given in the table. As noted from the table for sequence AHε, increasing the leakage to 10% and 100% per day increases the population dose (and consequently latent cancers) but does not result in any early injuries or fatalities. For sequence AHFϵ increasing the leakage rate to 100%/day results in early fatalities and injuries. However, it must be remembered that this sequence has a very low probability (1E-10) and is not significant on a risk basis.

Sensitivity of Results to Evacuation

In order to provide perspective on the effects of evacuation, CRAC2 was used to calculate the consequences of a PWR-1 release with and without evacuation. The evacuation model assumed that all persons within 10 miles of the plant were evacuated to 15 miles at a rate of 10 miles/hour. The evacuation was assumed to begin after a delay of either 1, 3, or 5 hours, with delay time weights of 30%, 40%, and 30%, respectively. The PWR-1 release without evacuation resulted in 5.4E6 person-rem of exposure, 1080 early fatalities and 2250 early injuries. The same release with evacuation results in public exposure of 5.3E6 person-rem, 226 early fatalities and 1830 early injuries. As expected, for the PWR-1 release, evacuation has a small effect on overall exposure, but reduces substantially the early health effects. Caution must be exercised in the interpretation of this evacuation sensitivity calculation because the ability of evacuation to substantially reduce early health effects is highly dependent on the accident scenario. For example, accident sequences that result in longer times to containment failure generally allow more time for evacuation and result in more effective protection of the public. Evacuation is less effective in scenarios where containment failure occurs early in an accident sequence.

Table 2.9 presents individual dose commitments from sequences AHϵ and AHFϵ as a function of containment leakage rate.

TABLE 2.9. Consequences to the Individual for AHϵ and AHFϵ Versus Leakage

Sequence	Leakage, % per day	Frequency, per year	Distance, meters	Whole Body Dose, rem Ave.	Whole Body Dose, rem Max.	Thyroid Dose, rem Ave.	Thyroid Dose, rem Max.
AHϵ	1	1E-6	402	1.4E-1	5.4E-1	1.9E0	8.0E0
			2012	2.6E-2	1.0E-1	3.1E-1	1.3E0
			5230	6.7E-3	2.5E-2	6.9E-2	2.6E-1
			14886	1.2E-3	1.1E-2	1.1E-2	1.1E-1
	10	multiply above values by 10					
	100	multiply above values by 100					
AHFϵ	1	1E-10	402	1.2E1	4.7E1	4.5E2	2.0E3
			2012	2.1E0	7.5E0	7.3E1	3.0E2
			5230	4.6E-1	1.6E0	1.6E1	6.0E1
			14886	7.2E-2	6.5E-1	2.6E0	2.6E1
	10	multiply above values by 10					
	100	multiply above values by 100					

The calculation of individual doses used the following parameters and assumptions:

- Dose consequences are represented by the 50-year individual dose commitment.
- Evacuation and/or sheltering of people is not considered.
- The core inventory at the time of the accident is assumed to be represented by a 3412 MWt (1120 MWe) PWR.
- Weather conditions for computing average and maximum individual doses are based on a 1-year sampling of local weather conditions at Moline, Illinois, including more than 100 sets of conditions. The average individual dose is calculated by averaging the doses computed for all sets of weather conditions. The maximum individual dose is obtained by selecting the largest dose value from among the doses computed for all sets of weather conditions.
- The breathing rate is assumed to be $2.66 \times 10^{-4} m^3$/sec. This represents a daily average that considers periods of activity and rest.

The whole body and thyroid individual dose results for the AHε sequence are presented graphically in Figures 2.2 and 2.3, respectively. The results are presented in terms of an average individual dose and a maximum individual dose (based on the worst weather conditions for the particular CRAC2 run) at specific distances.

As seen from Table 2.9, the leakage rate has a direct effect on the individual doses. These doses can be compared approximately to the numerical values in 10 CFR 100. The Part 100 values suggest a 25-rem whole body and a 300-rem thyroid dose limit for an individual located at the boundary of the exclusion area for two hours immediately following the onset of a postulated release from a hypothetical maximum credible accident. Although Table 2.9 is based on different source terms, assuming the 402-meter distance corresponds to the exclusion area boundary, using the average dose and conservatively assuming one-half of the four hour release applies (the four-hour release is not linear with time), both the AHε release and the AHFε release are less than these 10 CFR 100 numerical values for a 1% per day leakage rate. The AHε dose is also smaller for a leakage of 10% and 100% per day. The AHFε dose is greater than the 10 CFR 100 numerical value for leakages of 10% and 100% per day. Again it should be noted that the frequency of this release is low.

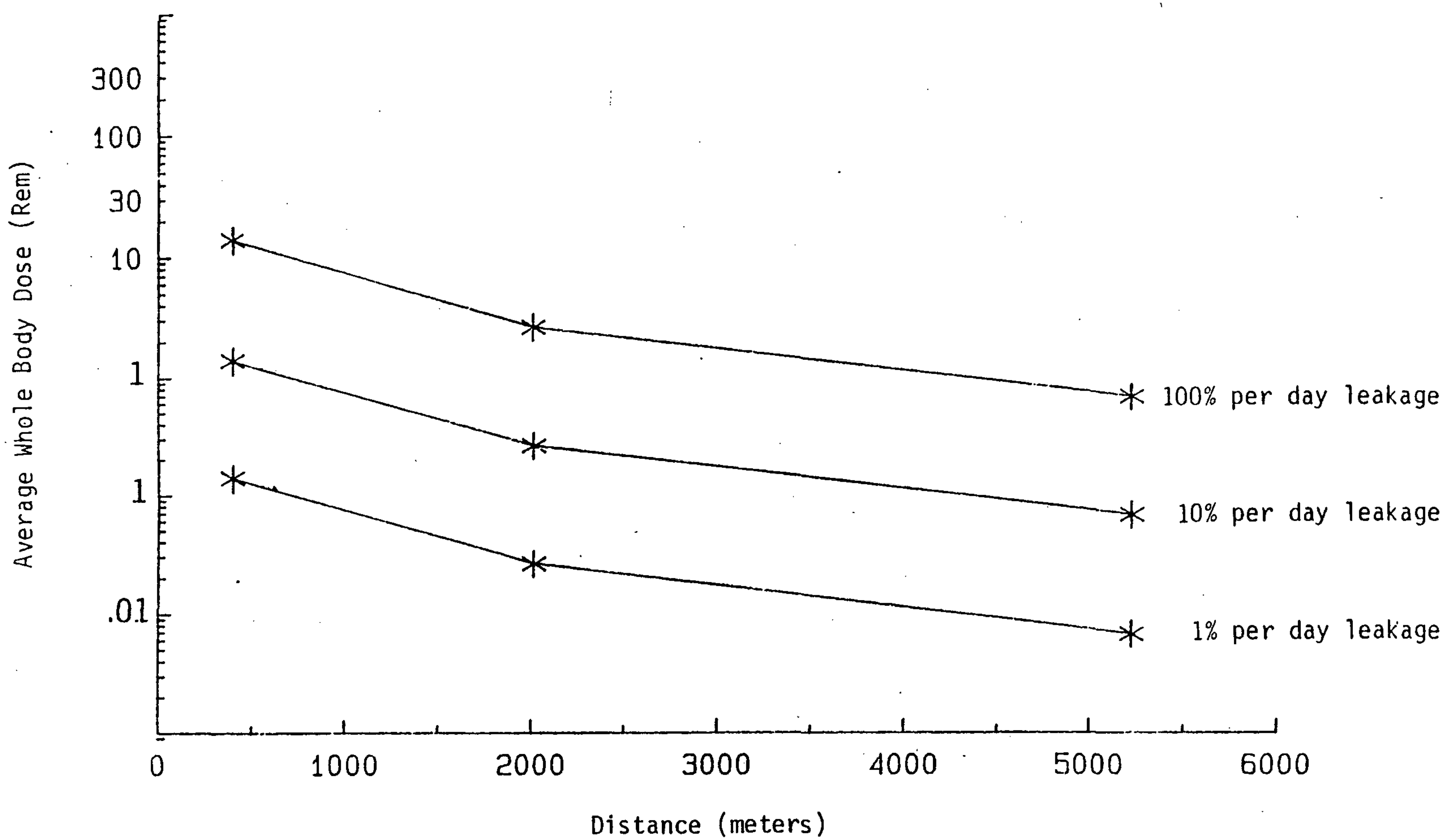


FIGURE 2.2. Whole Body Dose to Individual for AHε Sequence

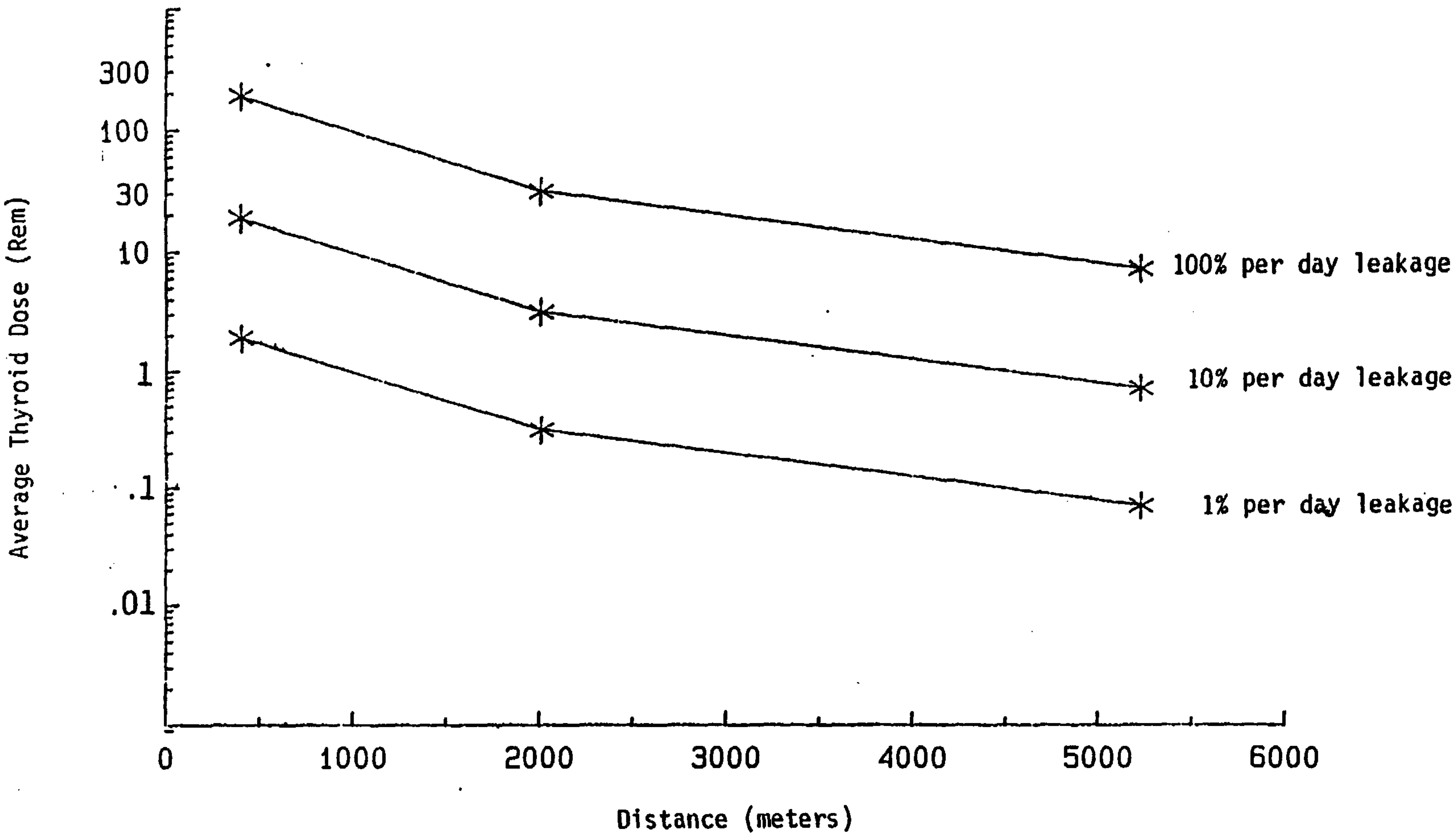


Figure 2.3. Thyroid Dose to Individual for AHε Sequence

2.3.2.2 BWR Scenarios

Two BWR scenarios are selected from WASH-1400 and used to examine the potential risk impacts of containment leakage. These scenarios include A and AFγ where:

A = Large Pipe Break
F = Emergency Core Cooling Function Failure
γ = Containment Failure by Overpressure

A is a large pipe break with no core melt or gross containment failure. AFγ is a core-melt sequence with containment failure by overpressure. A is a BWR-5 release and AFγ is a BWR-3 release. For sequence AFγ, only the period of leakage before failure by overpressure is considered. Overpressure occurs after approximately four hours and this time period is used for these release sequences. The core-melt timing, the release to containment, and the containment leakage values are taken from the CORRAL results presented in Appendix V of WASH-1400. Table 2.10 summarizes these values for this four-hour release period.

TABLE 2.10. Release from Containment for A and AFγ Sequences

Sequence	Leakage Rate, % per day	Accident Timing	4-Hour Release (fraction of core)	
A	0.5	1 min. Gap Release	Xe-Kr	6E-6
			I-Br	2E-11
			Cs-Rb	1E-9
			Te	3E-12
			Ba-Sr	3E-14
			Ru	0
			La	0
AFγ	0.5	20 min. Core Melt Begins	Xe-Kr	1E-4
		210 min. Vessel Melting	I-Br	1E-9
		290 min. Cont. Failure	Cs-Rb	2E-
			Te	1E-8
			Ba-Sr	2E-9
			Ru	1E-9
			La	2E-10

The releases from Table 2.10 were input to CRAC2 using the assumptions described in the previous section. The CORRAL results assumed a .5% per day leakage rate and the CORRAL results for this leakage rate were linearly scaled to obtain values for 5% per day and 50% per day leakage rates. Table 2.11 presents the public consequences from sequences A and AFγ as a function of the containment leakage rate.

As mentioned earlier, A is in BWR-5 and AFγ is in BWR-3. The consequences in Table 2.11 can be compared with those of BWR-3 and BWR-5. A BWR-3 release results in 5 million person-rem, 23 early fatalities and 463 injuries. A BWR-5 release results in 20 person-rem, and no early fatalities or injuries. As seen from the table, the consequences of the A sequence are very small. Increasing the leakage rate increases this very small consequence slightly. The incremental consequences of AFγ due to increasing the containment leakage rate are insignificant compared to the consequences from failure by overpressure. The impact of increasing containment leakage rates on these two scenarios is small. Due to the relatively small size of BWR containments, the containment typically is assumed to fail as a result of a core-melt accident or is assumed to leak at a rate much larger than the containment leakage rate if containment is intact.

TABLE 2.11. Consequences to the Public from A and AFγ Versus Leakage

Sequence	Leakage, % per day	Frequency, per year	Person-rem	Early Fatalities	Early Injuries
A	0.5	1E-4	7E-1	0	0
	5.0	1E-4	7E0	0	0
	50.0	1E-4	7E1	0	0
AFγ	0.5	1E-8	1E1	0	0
	5.0	1E-8	1E2	0	0
	50.0	1E-8	1E3	0	0

Table 2.12 presents individual dose commitments from sequences A and AFγ as a function of containment leakage rate. The results are presented in terms of an average individual dose and a maximum individual dose (based on the worst weather conditions for the particular CRAC2 run) at specific distances. The A scenario has very small consequences and does not exceed the 10 CFR 100 siting guidelines as discussed in the previous section for a leakage rate of 50% per day. The incremental consequences of the AFγ scenario up to the point of containment failure are small and do not exceed the 10 CFR 100 siting guidelines for a leakage rate of 50% per day. The consequences of the AFγ sequence once containment ruptures are very large.

TABLE 2.12. Consequences to the Individual for A and AFγ Versus Leakage

Sequence	Leakage, % per day	Frequency, per year	Distance, meters	Whole Body Dose, rem		Thyroid Dose, rem	
				Ave.	Max.	Ave.	Max.
A	0.5	1E-4	402	5.2E-4	2.0E-3	5.1E-4	2.0E-3
			2012	1.9E-4	8.3E-4	1.8E-4	7.9E-4
			5230	6.7E-5	3.3E-4	6.4E-5	3.2E-4
			14886	1.7E-5	1.2E-4	1.6E-5	1.2E-4
	5.0	multiply above values by 10					
	50.0	multiply above values by 100					
AFγ	0.5	1E-8	402	9.0E-3	3.4E-2	9.9E-3	3.9E-2
			2012	3.2E-3	1.4E-2	3.2E-3	1.4E-2
			5230	1.2E-3	5.5E-3	1.2E-3	5.4E-3
			14886	2.8E-4	2.1E-3	2.7E-4	2.0E-3
	5.0	multiply above values by 10					
	50.0	multiply above values by 100					

2.3.2.3 Hypothetical TMI-Related Scenario

Accidents similar to the Three Mile Island 2 accident can be used to provide some additional risk perspective on the effects of changes in containment leakage rates. The TMI release consisted of noble gases and iodine. The primary release path was through the auxiliary building via the letdown line of the makeup and purification system and the reactor coolant drain tank rather than leakage from the primary containment system. However, for many accidents of this type, the release pathway may be through primary containment and a functioning primary containment system served to limit any further consequences at TMI.

To examine the effects of containment leakage on core damage accidents similar to TMI, an arbitrary source term of all the noble gases and 1% of the iodine in the core is assumed to be released to the containment atmosphere two hours after shutdown. The probability of such a release is assumed to be 1E-3 per year. The computer program CRAC2 is used to calculate the consequences of this release for leakages of 0.1%, 1%, 10%, and 100% per day. The same site characteristics and modeling assumptions for the CRAC2 calculations discussed in Section 2.3.1.1 are used. It is conservatively assumed that no radioactive decay occurs after the start of release two hours after reactor shutdown. It is assumed that the integrity of primary containment is not affected by the accident and the release is controlled by the above leakages. No plateout of the 1% release of iodine to the containment atmosphere is assumed. A release will occur as long as the containment pressure is above atmospheric pressure and radioactive material is present. Two release periods of 2 hours and 10 hours are selected. Since no decay is assumed the results are proportional to the length of the release period. The risk is expressed in terms of expected person-rem, expected early fatalities and expected early injuries. Table 2.13 and Table 2.14 provide these results. Consistent with the other analyses in this section, the risk impact of a 1% or 10% per day leakage rate does not appear to be large. Also, as seen from the above results, no early fatalities result from leakages up to 100% per day for this example and the risk of early injury is small.

TABLE 2.13. Risk from Two-Hour Release of Noble Gas and Iodine Source Term

Leakage, % per day	Expected Person-rem/yr	Expected Early Fatalities/yr	Expected Early Injuries/yr
1.0	1.17E-1	0	0
10.0	1.15	0	0
100.0	9.76	0	2.0E-4

TABLE 2.14. Risk from Ten-Hour Release of Noble Gas and Iodine Source Term

Leakage, % per day	Expected Person-rem/yr	Expected Early Fatalities/yr	Expected Early Injuries/yr
1.0	5.86E-1	0	0
10.0	5.75	0	0
100.0	4.88E1	0	5.6E-4

Table 2.15 presents the individual exposures as a result of this hypothetical two-hour release of noble gas and iodine. As seen from Table 2.15 the leakage has a direct effect on the individual doses as a result of this release. This release is briefly examined using the 10 CFR 100 siting guidelines. These guidelines provide a 25-rem whole body and a 300-rem thyroid dose limit for an individual located at the boundary of the exclusion area for two hours immediately following the onset of a postulated release from a hypothetical maximum credible accident. Assuming the 402-meter distance corresponds to the exclusion area boundary, leakage rates up to 10% per day meet the above limits and a leakage rate of 100% per day fails to meet the limit for thyroid dose. This postulated release assumes a 1% release of iodine to the containment and no plateout. Again it should be noted that the frequency of this release is low. The expected doses per year are small, as is the product of the individual doses given in the table and the release frequency of 1E-3 per year.

TABLE 2.15. Individual Exposure from Two-Hour Release of Noble Gas and Iodine Source Term

Leakage, % per day	Distance, meters	Whole Body Dose, rem Ave.	Whole Body Dose, rem Max.	Thyroid Dose, rem Ave.	Thyroid Dose, rem Max.
1.0	402	1.20E-1	4.50E-1	4.72E0	2.06E1
	2012	3.10E-2	1.27E-1	7.76E-1	3.22E0
	5230	9.54E-3	4.24E-2	1.75E-1	6.54E-1
	14886	2.22E-3	1.65E-2	2.92E-2	2.80E-1
10.0		multiply above values by 10			
100.0		multiply above values by 100			

The results in this section have reinforced the conclusions made at the end of the previous section on population dose impacts. The effect of containment leakages is small for accident sequences which result in failure of containment from overpressure. The effect of containment leakages is approximately linear for those accidents in which containment integrity remains intact. On an expected individual dose basis, the effect of containment leakage is small. These results indicate that incentives exist to reevaluate requirements for containment leakage rates in relation to their risk significance.

2.4 COST IMPACTS

This section presents the results of a preliminary analysis of the cost impacts associated with the postulated increase in the allowable limits for containment leakage rates. Limited information is also presented on occupational exposure impacts. This assessment uses methods suggested by Heaberlin et al. (1983) and data developed in SEA (1985), which addresses a cost analysis of proposed revisions to 10 CFR Part 50, Appendix J. The assessment relies heavily upon existing NRC reports and upon contacts made with selected industry testing services and utilities for the estimates presented herein. A summary of the cost impact factors examined in this assessment are shown in Table 2.16.

The study results are presented in summary form in the section which follows. The bases and assumptions for the results are presented in Section 2.4.2. A brief discussion of the supporting information used in this study is presented in Sections 2.4.3 through 2.4.6. Detailed supporting information is provided in Appendix B of this report. In addition, several contacts were made with testing services and utilities for the purpose of confirming cost and occupational exposure information concerning Type A, B, and C testing and/or obtaining new information. Summaries of the discussions with these contacts also are included in Appendix B for completeness.

TABLE 2.16. Cost Impact Assessment Factors

	Impacts	
	Causes Quantified Change	Causes[a] Unquantified Change
Occupational Exposure (routine)		X
Industry Implementation	X	
Industry Operation	X	
NRC Implementation	X	
NRC Operation	X	

(a) Unquantified means not readily estimated in dollars.

2.4.1 Summary of Cost Impacts

A summary of the estimated cost impacts examined in this assessment is presented in Table 2.17. It can be seen from the table that the postulated increase in the allowable containment leak rate limits is estimated to result in a total cost savings in the range of about $40 million to about $74 million (all plants, all remaining life).

It appears that all 90 operating plants could be affected by the postulated increase in containment leakage rate limits. Currently, 40 to 50% of Type A Integrated Leak Rate Tests (ILRTs) result in failure according to SEA (1985). Improvement in this failure rate should result in decreased

TABLE 2.17. Summary of Estimated Cost Impacts - Total for All Plants

Assessment Factor	Estimated Cost Impact,[a] Thousands of dollars
• Industry	
Implementation	819
Operation	-41,900 to -75,500
• NRC	
Implementation	1,000
Operation	-7 to -13
Estimated Total Quantified Impact	-40,000 to -74,000

(a) Impacts are defined as the costs incurred as a result of the postulated action. Negative impacts indicate cost savings. No discount rate has been applied.

costs and radiation exposure. Type A tests are generally on the reactor outage critical path. Thus, the largest cost impact resulting from this preliminary analysis is anticipated to be the incremental downtime costs that could potentially be reduced by the postulated change in leak rate limit. This savings is anticipated because the change appears to reduce the likelihood that plants will fail their Type A ILRT.

To quantify the impact of the postulated increase in leak rate limit, it was necessary to estimate the potential reduction in Type A test failures. Although the extent of the potential reduction in test failures cannot be quantified with a high degree of confidence, it is assumed (for purposes of illustration of potential cost savings) that an improvement in the range of from 50 to 90% occurs as a result of the postulated change. In turn, this results in a new range of average values for the likelihood of Type A test failures from about 5% to about 23%. Note that increasing the leakage rate may not necessarily reduce the test failures if a utility chooses to do only minimal maintenance on the containment and its penetrations.

The aggregate reduction in downtime costs is estimated to result in total industry savings (in constant 1985 dollars) in the range of about $42 million to about $76 million. This range of costs is based on the difference in estimated costs of retesting using the current average failure value (45%) and the lower range of failure values (5% to 23%) which are assumed to occur due to the postulated change in the leakage rate limit. The supporting calculations can be found in Appendix B (Section B.1.2.4).

One-time costs associated with the postulated increase in leakage rate requirements would involve changes to existing plant licenses (i.e., changes to the plant technical specifications). Based on an estimated cost of $9,100 per plant, the total industry cost would be $819,000. The review and approval of the technical specification changes by the NRC is estimated to cost about $11,100 per plant for a total cost of $1 million. If 10 CFR Part 50, Appendix J is subsequently revised, the cost associated with that effort also would be incurred by the NRC.

Along with the previously mentioned reduction in the number of ILRT failures would be an accompanying reduction in the number of licensee testing schedules required to be submitted to the NRC for review and approval whenever the licensee fails a Type A test. This reduction in work for the NRC is estimated to result in a total cost savings in the range of $7,200 to about $13,000 over all the remaining reactor lifetimes.

In addition, the following conclusions are drawn concerning cost impacts based on the information examined in this study:

- For the most part, Type A tests are on the reactor outage critical path. Type B and C tests are not generally on the reactor outage critical path.
- Very little detail was readily available concerning cost, occupational radiation exposure, and/or time specifically associated with valve and

component leak testing activities (as differentiated from maintenance tasks on those same elements).

- The Type A ILRT pressurization and depressurization steps are being done at optimum rates. This is not surprising, considering the cost of downtime at nuclear power stations.

- Specialty contractors of one kind or another are being used by some utilities for selected tasks associated with leakage rate test programs.

2.4.2 Bases and Assumptions

The cost and occupational exposure estimates used in this analysis are intended to provide information useful to the NRC. The analysis bases and assumptions can have major impacts on the costs associated with the postulated increase in the allowable leakage rate limits addressed in this assessment. The bases and assumptions used in this analysis must therefore be carefully examined before the results can be applied to specific LWRs.

The analysis bases are:

- The analysis must yield realistic and up-to-date results based on current nuclear plant designs.

- The analysis is conducted within the framework of existing regulations and regulatory guidance.

- To the extent possible, the analysis evaluates leak rate testing of existing nuclear reactor facilities for the purpose of developing useful cost data.

- Generic replacement power costs, ranging from \$13.5K/hour to \$18.97K/hour, per Type A ILRT were taken from SEA (1985) as was the representative cost of \$200,000 per ILRT when the test is done by an external contractor.

- The costs are in early-1985 dollars unless stated otherwise.

The principal analysis assumptions are:

- The Type A ILRT utlized in the reference case for this assessment is on the critical path and includes work done by an external contractor.

- No downtime penalty is incurred for Type B and C testing as delineated in SEA (1985).

- No license amendment costs are incurred for planned plants since it is assumed that the postulated increase in leakage rate limits will become a normal part of their licensing documentation.

- Because of the high costs of downtime at nuclear power plants, it is assumed that the pressurization and depressurization steps integral to

Type A testing have already been optimized in line with existing plant designs.

2.4.3 Cost and Occupational Radiation Exposure Related to Type A ILRTs

The principal parameters associated with Type A ILRTs under the current leak rate regulation (10 CFR Part 50, Appendix J) and guideline (ANSI/ANS 56.8-1981) are shown below. This is for a reference case, which assumes the ILRT is on the critical path.

Cost[(a)]	$1.3 to 2.6 million
Incremental Plant Downtime	3 to 5 days
Occupational dose, avg.	0.8 man-rem

The foregoing values were extracted from SEA (1985). A Breakdown of the estimated costs for the reference case Type A ILRT is presented in Table 2.18.

TABLE 2.18. Summary of Estimated Costs Associated With Industry Operation for the Reference Case Type A ILRT

Cost Category	Range of Estimated Costs, Thousands of Dollars	Remarks
Replacement Power	1,200 to 2,500	Based on information in SEA (1985), Section 5.0.
Compressors/Dryers	35 to 50	Rental option only.
Instruments, Test	approx. 35	Rental option only.
Outside Team	30 to 60	Specialty Contractor
Misc. Equipment & Materials	Insignificant	Based on information in SEA (1985), Section 6.0.
Estimated Total, Type A ILRT Costs	1,300 to 2,610	

(a) The major cost element is replacement power costs ranging from $13.5K/hour to $18.97K/hour as reported in SEA (1985).

In addition, a reference case Type A ILRT time frame was developed (see Table 2.19) based on information contained in Frank et al. (1982) and discussions with selected utility contacts (see Appendix B for details). The data shown in Table 2.19, for the most part, are calculated averages. The pressurization, stabilization, measurement, verification, and depressurization activities are based on three actual ILRTs, including two tests as reported by Frank et al. (1982) and a third test as reported by an eastern utility power plant contact (see Appendix B for details). All the tests were conducted since 1980. The maximum rate of depressurization (5 psi/hour) occurred at the eastern utility's plant. The 10 hours for depressurization shown in the table reflects the eastern utility plant's rate and time. Actual test measurement times ranged from 12 hours to about 27 hours; therefore, the "average" used in the table was adjusted to 24 hours to more closely reflect the guidance found in Appendix J of 10 CFR 50.

For purposes of this analysis, it is convenient to express all test costs in terms of costs per hour. For example, using an average total Type A ILRT cost from Table 2.18 (i.e., $1,955,000) divided by the estimated total hours for the ILRT presented in Table 2.19 yields an "average" cost of about $20,400 per hour for the reference case Type A ILRT.

2.4.4 Cost and Occupational Radiation Exposure Related to Local Leakage Rate Testing (Type B and C Tests)

Type B and C leak testing is a labor-intensive activity. Therefore, labor cost data must be carefully selected and applied. With the data provided in SEA (1985), a representative cost for Type B and C tests is roughly estimated at about $15,400, or an average aggregate cost of about $41 per hour. This hourly cost also includes a nominal top-level management cost component (see Appendix B for details).

TABLE 2.19. Reference Case Containment ILRT Time Frame Used in this Analysis

ILRT Activity	Time, hours
Pretest Preparations	15
Pressurization	13
Stabilization	14
Integrated Leak Rate Measurement	24
Verification Test	6
Depressurization	10
Post Test Restoration	14
Total Hours	96

For Type B and C tests the number of valves and penetrations requiring repairs is highly plant-specific. And, according to information contained in SEA (1985), depending upon plant age, equipment supplier, and general housekeeping practice at the utility, occupational exposures vary widely for these tests. They can nominally range from 0.7 to 12.5 man-rem. Couching their estimate carefully (because of the limited data available), the authors of SEA (1985) derived a generic, weighted average leak test exposure for all LWRs of four man-rem for Type B and C tests. This estimate assumed a mix of one-third BWRs and two-thirds PWRs. At a generic LWR, SEA (1985) further estimated that local Type B and C leak rate testing can be accomplished in about 275 man-hours in radiation zones during the course of a refueling outage (typically six to ten weeks) by a three to five man crew.

A decrease in Type B and C repairs and maintenance could be expected with the higher leakage rate limits. However, sufficient information is not available at this time for an accurate quantification of the potential amount of the accompanying reduction in occupational exposure.

2.4.5 Industry Cost

Results of the analyses for the cost impacts associated with industry implementation and operation are briefly described below, with detailed supporting information presented in Appendix B.

2.4.5.1 Industry Implementation

It appears that all existing plants could be affected by this postulated change. Once the leakage limits are developed the plant technical specifications (tech specs) must be amended with appropriate approvals from plant management. These technical specification changes must then be submitted to the NRC for their review and approval. The overall effort for each plant affected is assumed to be a "typical uncomplicated" technical specification change. However, it is recognized that this assumption probably results in a lower bound cost estimate since the engineering effort could range from a modest effort to a fairly complex safety analysis depending on the number of plant-specific factors. The technical specification modification is assumed to involve modest manpower requirements and requires no plant downtime or other industry cost impacts. The industry effort per plant is estimated at 4 man-wks at a cost of about $9100.

In addition, Dougan (1984) theorizes that to maintain flexibility, reference to ANSI/ANS-56.8-1981 probably will not be made directly in the proposed revision to 10 CFR 50, Appendix J. More likely, reference will be made in a regulatory guide that can be more easily revised than Appendix J in case it becomes necessary to reference a new industry standard should one be developed to replace the ANSI standard. Also, Dougan (1984) states that exceptions taken by the NRC to the guidelines in ANSI/ANS-56.8-1981 can be dealt with in a regulatory guide, as can areas of direct conflict between the ANSI standard and Appendix J. Therefore, it seems reasonable that the ANSI/ANS-56.8-1981 would undergo review if the leakage rate limit postulated in this analysis were adopted. However, no estimate of cost for this latter effort was developed in this assessment.

2.4.5.2 Industry Operation

When the ILRT is on the critical path of an outage each of the following five sequential steps of the test itself lies on the critical path:

- Pressurization
- Stabilization
- Integrated Leak Rate Measurement
- Verification Test
- Depressurization

Correspondingly, the time it takes to perform each of the five steps can impact costs. The potential effect on each of the five steps resulting from the increased leakage rate limit postulated in this analysis is summarized in Table 2.20 (see Appendix B for detailed supporting information). In addition, pretest preparations and post test restoration activities take varying amounts of time; however, these activities may or may not be considered by the utility to be on the critical path.

TABLE 2.20. Summary of Estimated Impacts on the Five Steps of the Type A ILRT

Step	Potential Effect[a]
Pressurization	Reactor-specific analysis required[b]
Stabilization	No substantive change anticipated
Integrated Leak Rate Measurement	No substantive change anticipated
Verification Test	Greater time may be required[c]
Depressurization	Reactor-specific analysis required[b]

(a) Based on the assumption that the current containment vessel pressure testing requirements delineated in 10 CFR 50, Appendix J remain in effect.
(b) Because of the high costs of downtime at nuclear power plants, it is assumed that the licensees have already optimized the pressurization and depressurization steps based on their existing plant's design. For example, the installation of larger diameter pipes (a major and costly design change) could be required at some nuclear power plants to effect a change in their present pressurization and/or depressurization rates.
(c) As a result of the postulated increased leakage allowance, it is possible that this step could involve the movement of a larger metered mass of air, thus requiring more time to move the mass of air.

It can be seen from Table 2.20 that, barring major design changes, the potential effect on each procedural step of the ILRT is estimated to be relatively minor. Overall, the largest cost impact resulting from the postulated increase in leak rate limits is anticipated to be the incremental downtime costs that could potentially be saved because the change appears to reduce the likelihood that nuclear plants will fail their Type A ILRT. The aggregate reduction in downtime costs is estimated to result in total industry savings (in constant 1985 dollars) in the range of about $42 million to about $76 million. The supporting calculations can be found in Appendix B.

2.4.6 NRC Cost

Results of the analyses for the cost impacts associated with NRC implementation and operation follow.

2.4.6.1 NRC Implementation

The cost increase associated with the review and approval of plant technical specifications amendments previously mentioned is estimated at about $11,000 per plant as shown in Table 2.21. This cost includes an estimated four man-weeks of technical staff time, two man-weeks of management and legal review, and $800 for Federal Register notices. In addition, this cost only affects operating plants since it is assumed that planned plants will incorporate such changes into their license documents. Virtually all of these one-time administrative costs will ultimately be passed on to the utilities under the NRC's fee recovery program. Potentially, the modification of the technical specifications could be handled by generic letter or other regulatory means, resulting in a reduction in costs per plant from those shown in the table.

TABLE 2.21. Estimated Per Plant Cost of Technical Specification Amendment

(4 man-wks)(40 hrs/wk x $41/hr)	=	$6,560
(2 man-wks)(40 hrs/wk x $47/hr)	=	$3,760
Federal Register Notice(s)	=	$ 800
Total/plant		$11,120

The above estimate is based on cost estimates provided by NRC's Division of Budget and Analyses and on information contained in the Cost Estimate Basis presented in Section 6.0 of SEA (1985).

2.4.6.2 NRC Operation

The existing Appendix J of 10 CFR 50 specifies the NRC must review and approve the testing schedules subsequent to the failure of a Type A ILRT. This requirement is not anticipated to change with the postulated change in leakage rate limit. However, as previously mentioned, it is anticipated that the number of ILRT failures will be reduced due to the postulated increase in the allowable leakage rate limits. Therefore, the number of testing schedules submitted by licensees to the NRC for review also will be reduced, resulting in a reduction of NRC resources in this area. The total cost savings over all remaining reactor lifetimes is estimated to be in the range of $7,200 to $12,960 (see Appendix B for details).

3.0 RISK AND COST IMPACTS FOR MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS

Boiling Water Reactors (BWRs) draw steam directly from the reactor vessel to the turbine via main steam lines. The main steam lines installed in these plants contain dual quick-closing main steam isolation valves (MSIVs). These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. The current technical specification limit for MSIV leakage is typically 11.5 standard cubic feet per hour (SCFH). Operating experience has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs, and the specified low leakage has not always been maintained.

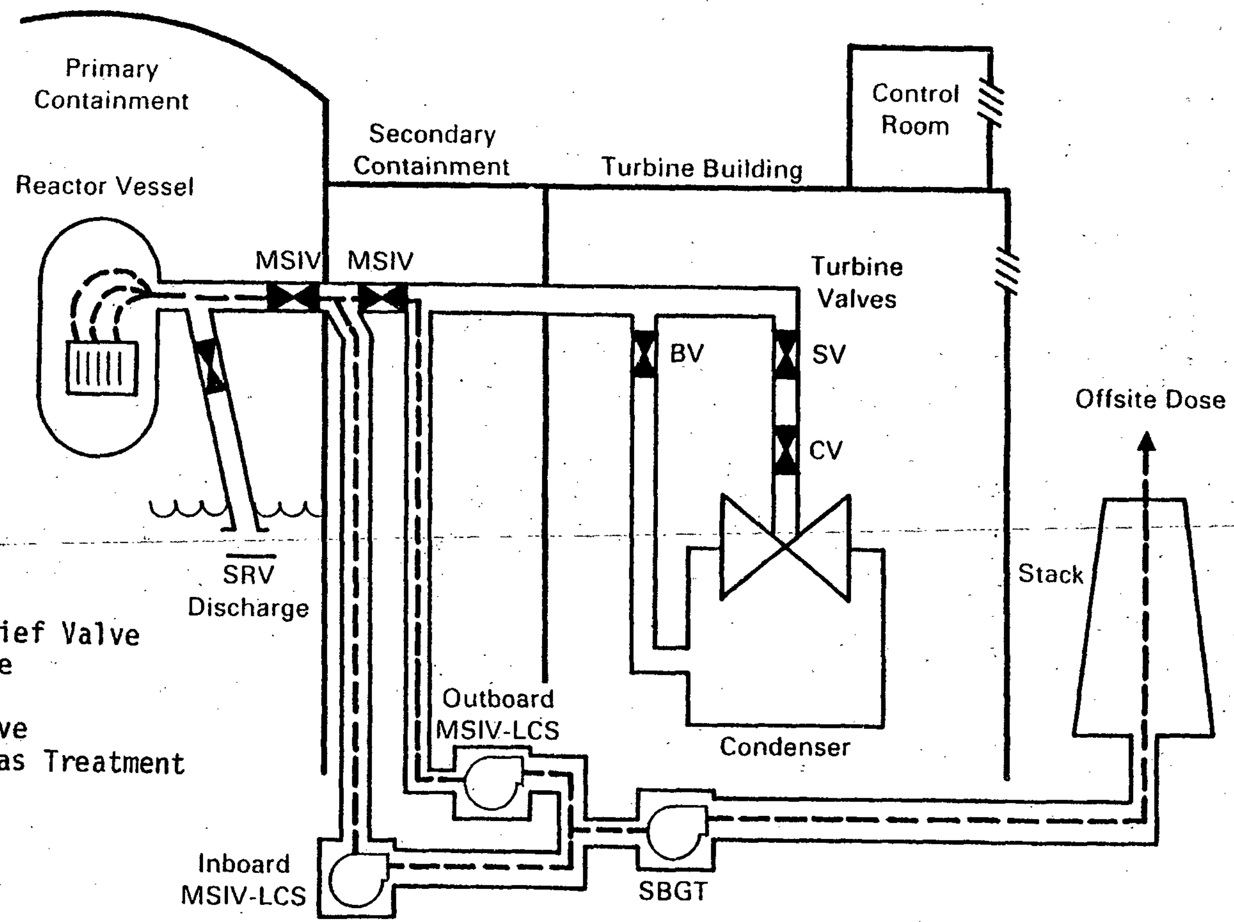
3.1 CURRENT REGULATORY REQUIREMENTS

Due to recurring problems with excessive leakage of MSIVs, NRC Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Nuclear Power Plants" recommended the installation of a supplemental leakage control system (LCS) to ensure the isolation function of the MSIVs complies with the specified limits.

Figure 3.1 provides an illustration of a typical MSIV leakage control system. The figure shows that both an inboard and an outboard MSIV-LCS are used. Each system is typically designed for a maximum flow of 100 SCFH. The normal flow path is through the inboard system for leakages less than 100 SCFH. Leakage from the inboard MSIV in excess of 100 SCFH generally results in isolation of the inboard MSIV-LCS. Any resultant leakage past the outboard MSIV is collected by the outboard MSIV-LCS downstream of the outboard MSIV. The MSIV-LCS routes the leakage through the standby gas treatment system (SBGT) for holdup, treatment, and release through the stack.

3.2 ALTERNATIVES TO REGULATORY REQUIREMENTS

The desirability of the MSIV leakage control system requirement has recently undergone review by the NRC. The LCS may not be effective for MSIV leakage rates greatly in excess of technical specification limits due to limitations in its design. In addition, alternate MSIV leakage treatment methods, which make use of the holdup volume in the main steam lines and condenser, may be superior to the LCS in reducing offsite consequences. Generic Issue C-8, "MSIV Leakage and LCS Failure" was established to address these concerns. The remainder of this section examines, on a preliminary basis, the risk and cost impacts associated with eliminating the requirements for MSIV leakage control systems in BWRs.



SRV = Safety Relief Valve
BV = Bypass Valve
SV = Stop Valve
CV = Control Valve
SBGT = Standby Gas Treatment System

FIGURE 3.1. MSIV Leakage Control System

3.3 RISK IMPACTS

Two situations are considered: 1) a representative BWR with a leakage control system which will be used as required, and 2) a representative BWR without a leakage control system. The following analysis is based upon the approach and data used in NUREG-0933, The Prioritization of Safety Issues (U.S. NRC 1983).

The Grand Gulf PRA (Hatch et al. 1981) is assumed to represent a typical BWR. Two potential release pathways for MSIV leakage are considered: 1) Through the LCS; and 2) through the steam lines, the condenser, and the steam and waste gas treatment system (SWGTS). The first pathway is illustrated in Figure 3.1 and was discussed in the previous section. Figure 3.2 provides an illustration of a typical path for MSIV leakage for a plant without an MSIV-LCS. The MSIV leakage passes through the steam lines, condenser, and SWGTS. This path provides a mechanism for long holdup times, cold trapping of iodine and volatiles, and treatment in the SWGTS.

3.3.1 Risk Impact for BWR Without Leakage Control System

For a BWR without a LCS, the following parameters are analyzed:

- Core-melt frequency
- MSIV leakage
- Steam line, condenser, SWGTS path availability
- Off-site release.

The following key data/assumptions are used:

1. Frequency of core melt is the same as the Grand Gulf PRA (3.8E-5 per reactor year). Approximately 26% of the core-melt scenarios involve loss of offsite power.

2. The SWGTS does not function with loss of offsite power.

3. The condenser and the SWGTS are conservatively assumed to function together, ie., when the SWGTS is unavailable, so is the condenser.

4. The unavailability of the SWGTS is 0.05 per demand. That is, the SWGTS will not be available to prevent leakage to the environment 5% of the time.

5. The partitioning efficiency for the SWGTS is 99.9%. This is an approximate factor which accounts for plateout, filtering and holdup for this pathway.

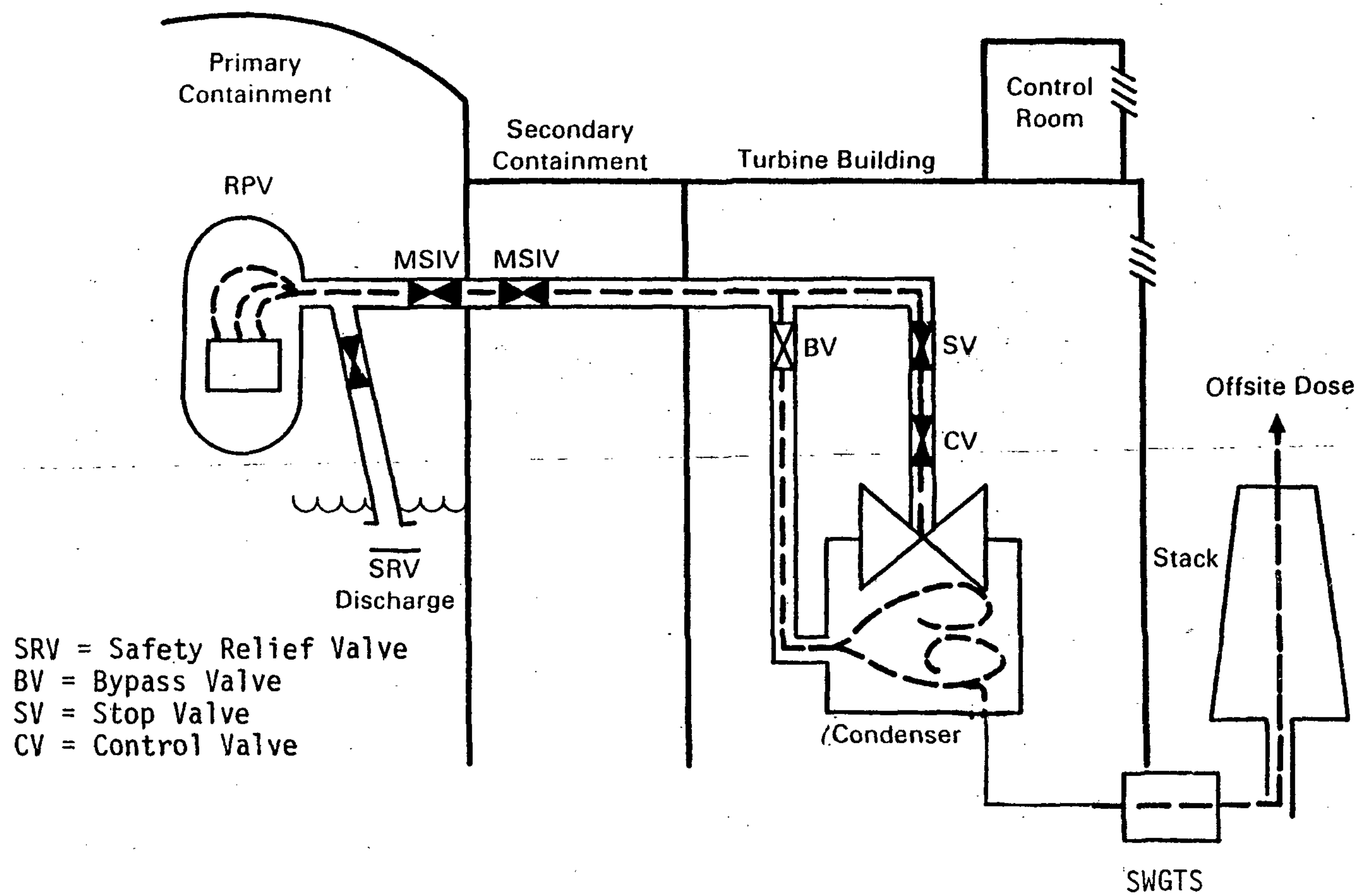


FIGURE 3.2. Steam Line, Condenser, SWGTS Release Path

6. MSIV leak rate data is taken from NUREG-0933 (NRC 1983). These results are based upon a survey performed in 1982 and are shown in Table 3.1. Independent failure or leakage for two MSIVs in series is assumed. The leakage rate from both valves is assumed to be the smaller of the values for each valve.

7. The radiological consequences calculated in NUREG-0933 for a leakage of 3000 SCFH (5.2E6 person-rem) are used. These consequences are based on a WASH-1400 BWR-2 release category source term. A two-hour delay before fission product release begins and a 0.27 hour delay in the main steam lines were used. A nominal low-energy ground level release to the atmosphere at the turbine stop valve was assumed. Consequences of other MSIV leakages are estimated by taking a ratio of the leakages to the above value and using the appropriate partitioning factors for the SWGTS. These radiological consequences represent a conservative upper bound in that containment is not assumed to fail as a result of the core-melt event and the release pathway through the MSIVs is the dominant pathway. For the majority of the BWR core-melt events, containment is likely to fail due to overpressure and alternative, larger release pathways will exist.

TABLE 3.1. Mean MSIV Frequency of Leakage for a Single Valve

Leakage, SCFH	Relative Frequency
11.5	.58
55.0	.17
1500.0	.25

Figure 3.3 presents the event tree for this case. Table 3.2 provides the details of the quantification of the event tree. The sequence (number 27) shown in Table 3.3 dominates the results. Other sequences involving successful operation of the SWGTS or mechanical failure of the SWGTS and MSIV leakage rates of 11.5 and 55 SCFH contribute an additional 0.53 man-rem per year resulting in a total risk of 2.14 man-rem per year. Table 3.2 provides the detailed results for these sequences.

3.3.2 Risk Impacts for BWRs With Leakage Control System

For a BWR with a leakage control system the following parameters are analyzed:

- Core-melt frequency
- MSIV leakage
- LCS availability (including SBGT)
- SWGTS availability
- Offsite release.

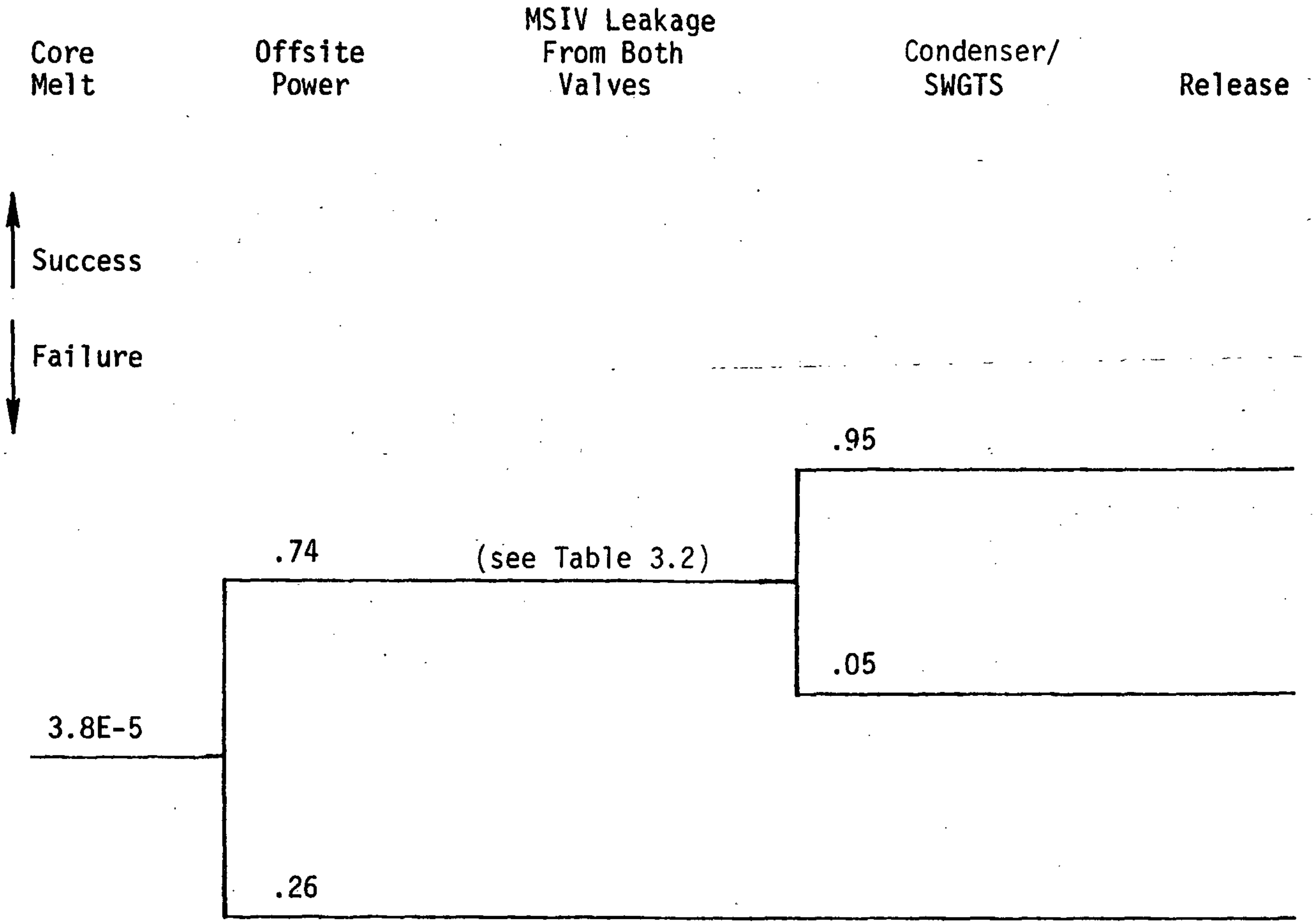


FIGURE 3.3. Event Tree for BWR Without MSIV-LCS

TABLE 3.2. Quantification of Event Tree for BWR Without MSIV-LCS

Sequence Number	Core Melt Frequency, per year	Offsite Power Avail.	Offsite Power Fails	Inboard MSIV Leak Frequency	Outboard MSIV Leak Frequency	Condenser/SWGTS Avail.	Condenser/SWGTS Fails	Release Frequency, per year	Consequence, person-rem	Risk, person-rem/year
1	3.8E-5	0.74		0.58 (11.5 SCFH)	.58 (11.5 SCFH)	0.95		8.99E-06	1.99E+01	1.79E-04
2	3.8E-5	0.74		0.58 (11.5 SCFH)	.17 (55 SCFH)	0.95		2.63E-06	1.99E+01	5.24E-05
3	3.8E-5	0.74		0.58 (11.5 SCFH)	.25 (1500 SCFH)	0.95		3.87E-06	1.99E+01	7.71E-05
4	3.8E-5	0.74		.17 (55)	.58 (11.5)	0.95		2.63E-06	1.99E+01	5.24E-05
5	3.8E-5	0.74		.17 (55)	.17 (55)	0.95		7.72E-07	9.53E+01	7.36E-05
6	3.8E-5	0.74		.17 (55)	.25 (1500)	0.95		1.14E-06	9.53E+01	1.09E-04
7	3.8E-5	0.74		.25 (1500)	.58 (11.5)	0.95		3,87E-06	1.99E+01	7.71E-05
8	3.8E-5	0.74		.25 (1500)	.17 (55)	0.95		1.14E-06	9.53E+01	1.09E-04
9	3.8E-5	0.74		.25 (1500)	.25 (1500)	0.95		1.76E-06	2.60E+03	4.58E-03
10	3.8E-5	0.74		0.58 (11.5)	.58 (11.5)		0.05	4.73E-07	1.99E+04	9.43E-03
11	3.8E-5	0.74		0.58 (11.5)	.17 (55)		0.05	1.39E-07	1.99E+04	2.77E-03
12	3.8E-5	0.74		0.58 (11.5)	.25 (1500)		0.05	2.04E-07	1.99E+04	4.07E-03
13	3.8E-5	0.74		0.17 (55)	.58 (11.5)		0.05	1.39E-07	1.99E+04	2.77E-03
14	3.8E-5	0.74		0.17 (55)	.17 (55)		0.05	4.06E-08	9.53E+01	8.09E-04
15	3.8E-5	0.74		0.17 (55)	.25 (1500)		0.05	5.98E-08	9.53E+01	1.19E-03
16	3.8E-5	0.74		.25 (1500)	.58 (11.5)		0.05	2.04E-07	1.99E+04	4.07E-03
17	3.8E-5	0.74		.25 (1500)	.17 (55)		0.05	5.98E-08	9.53E+04	1.19E-03
18	3.8E-5	0.74		.25 (1500)	.25 (1500)		0.05	8.79E-08	2.60E+06	2.29E-01
19	3.8E-5		0.26	.58 (11.5)	.58 (11.5)		1.0	3.32E-06	1.99E+04	6.62E-02
20	3.8E-5		0.26	.58 (11.5)	.17 (55)		1.0	9.74E-07	1.99E+04	1.94E-02
21	3.8E-5		0.26	.58 (11.5)	.25 (1500)		1.0	1.43E-06	1.99E+04	2.85E-02
22	3.8E-5		0.26	.17 (55)	.58 (11.5)		1.0	9.74E-07	1.99E+04	1.94E-02
23	3.8E-5		0.26	.17 (55)	.17 (55)		1.0	2.86E-07	9.53E+04	2.73E-02
24	3.8E-5		0.26	.17 (55)	.25 (1500)		1.0	4.20E-07	9.53E+04	4.00E-02
25	3.8E-5		0.26	0.25 (1500)	.58 (11.5)		1.0	1.43E-06	1.99E+04	2.85E-02
26	3.8E-5		0.26	0.25 (1500)	.17 (55)		1.0	4.20E-07	9.53E+04	4.00E-02
27	3.8E-5		0.26	0.25 (1500)	0.25 (1500)		1.0	6.18E-07	2.60E+06	1.61
										2.14

TABLE 3.3. Dominant Sequence for BWR Without MSIV-LCS

Probability of core melt involving loss of power	9.9E-6 per year
Fraction of MSIV leakages resulting in 1500 SCFH	0.0625
Failure of SWGTS given loss of power	1.0
Partitioning factor for SWGTS given failure	1.0
Ratio of 1500 SCFH and 3000 SCFH source term	0.5
Reference source term	5.2E6 man-rem
Risk equals the product of the above terms	1.61 man-rem per year

The following key data/assumptions are used along with those given earlier:

1. The unavailability of both the inboard and outboard LCS is 0.05 per demand.
2. The inboard and outboard LCS function during loss of off-site power.
3. The inboard and outboard LCS fail at leakage rates greater than 100 SCFH.
4. The partitioning efficiency for the LCS is 99%.

The leakage scenarios are initiated by a core melt event. The MSIVs close in response to this event. The inboard valve is assumed to leak either at 11.5, 55, or 1500 SCFH, with the relative frequencies provided earlier in Table 3.1. The leakage will be processed by the inboard LCS if the leakage is below 100 SCFH. If the leakage is greater than 100 SCFH the inboard system isolates and leakage is directed to the outboard MSIV. The outboard MSIV is assumed to leak at 11.5, 55, or 1500 SCFH, and the overall leakage past both MSIVs is assumed to be the smaller of the two leakage rates. That is, no more can leak from the outboard valve than what was originally leaked from the inboard valve. The leakage will be processed by the outboard LCS if it is less than 100 SCFH. If the leakage is greater than 100 SCFH, it is assumed to be routed through the steam line, condenser and SWGTS pathway.

Figure 3.4 presents the event tree for this case. Table 3.4 provides the details of the quantification of the event tree. A comparison of Tables 3.2 and 3.4 indicates that the leakage control system reduces MSIV leakage risks by 0.29 person-rem per year. Interestingly, the dominant MSIV leakage sequence for the BWR with an MSIV-LCS (see Table 3.5, sequence 47) is essentially the same as for BWRs without MSIV-LCS (see Table 3.3, sequence 27). This occurs because the LCS is automatically isolated when MSIV leakage exceeds 100 SCFH (as indicated by an MSIV-LCS failure rate of 1.0). The steamline, condenser and SWGTS pathway must accommodate the leakage. This

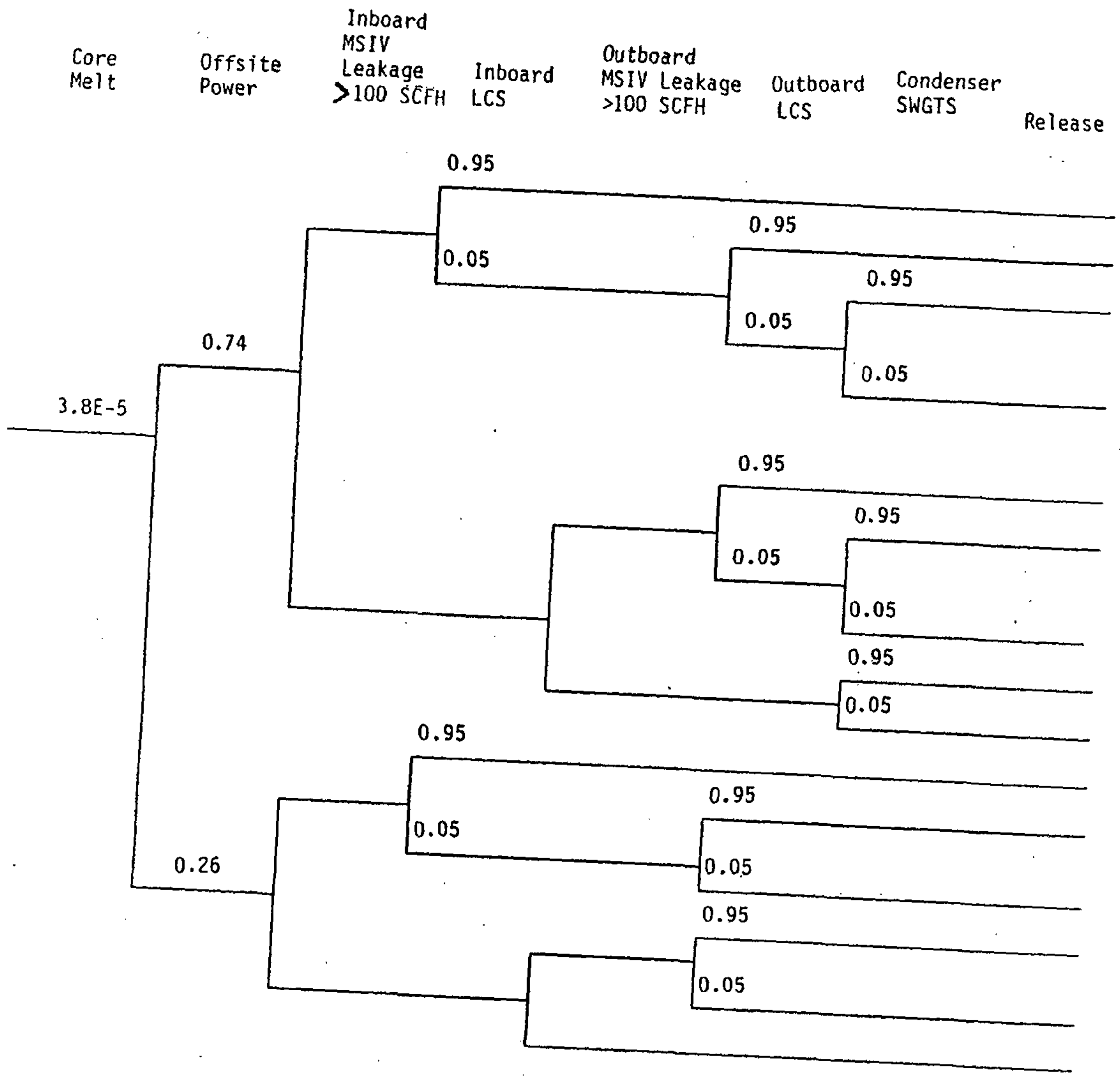


FIGURE 3.4. Event Tree for BWR With MSIV-LCS

TABLE 3.4. Quantification of Event Tree for BWR With MSIV-LCS

Sequence Number	Core Melt Frequency	Offsite Power Avail.	Offsite Power Fails	Inboard MSIV Leak Frequency	Inboard LCS Avail.	Inboard LCS Fails	Outboard MSIV Leak Frequency	Outboard LCS Avail.	Outboard LCS Fails	Condenser/ SWGTS Avail.	Condenser/ SWGTS Fails	Release Frequency per year	Consequence, person-rem	Risk, person-rem/ year
1	3.8E-5	0.74		0.58 (11.5 SCFH)	0.95							1.55E-05	1.99E+02	3.09E-03
2	3.8E-5	0.74		0.17 (55 SCFH)	0.95							4.54E-06	9.53E+02	4.33E-03
3	3.8E-5	0.74		0.58 (11.5)		0.05	0.58 (11.5)	0.95				4.49E-07	1.99E+02	8.95E-05
4	3.8E-5	0.74		0.58 (11.5)		0.05	0.17 (55)	0.95				1.32E-07	1.99E+02	2.63E-05
5	3.8E-5	0.74		0.58 (11.5)		0.05	0.25 (1500)	0.95				1.94E-07	1.99E+02	3.87E-05
6	3.8E-5	0.74		0.58 (11.5)		0.05	0.58 (11.5)		0.05	0.95		2.25E-08	1.99E+01	4.48E-07
7	3.8E-5	0.74		0.58 (11.5)		0.05	0.17 (55)		0.05	0.95		6.59E-09	1.99E+01	1.31E-07
8	3.8E-5	0.74		0.58 (11.5)		0.05	0.25 (1500)		0.05	0.95		9.68E-09	1.99E+01	1.93E-07
9	3.8E-5	0.74		0.58 (11.5)		0.05	0.58 (11.5)		0.05		0.05	1.18E-09	1.99E+04	2.35E-05
10	3.8E-5	0.74		0.58 (11.5)		0.05	0.17 (55)		0.05		0.05	3.47E-10	1.99E+04	6.92E-06
11	3.8E-5	0.74		0.58 (11.5)		0.05	0.25 (1500)		0.05		0.05	5.10E-10	1.99E+04	1.02E-05
12	3.8E-5	0.74		0.17 (55)		0.05	0.58 (11.5)	0.95				2.27E-07	1.99E+02	4.52E-05
13	3.8E-5	0.74		0.17 (55)		0.05	0.17 (55)	0.95				3.86E-08	9.53E+02	3.68E-05
14	3.8E-5	0.74		0.17 (55)		0.05	0.25 (1500)	0.95				5.68E-08	9.53E+02	5.41E-05
15	3.8E-5	0.74		0.17 (55)		0.05	0.58 (11.5)		0.05	0.95		6.59E-09	1.99E+01	1.31E-07
16	3.8E-5	0.74		0.17 (55)		0.05	0.17 (55)		0.05	0.95		1.93E-09	9.53E+01	1.84E-07
17	3.8E-5	0.74		0.17 (55)		0.05	0.25 (1500)		0.05	0.95		1.14E-08	9.53E+01	1.09E-06
18	3.8E-5	0.74		0.17 (55)		0.05	0.58 (11.5)		0.05		0.05	3.47E-10	1.99E+04	6.92E-06
19	3.8E-5	0.74		0.17 (55)		0.05	0.17 (55)		0.05		0.05	1.02E-10	9.53E+04	9.72E-06
20	3.8E-5	0.74		0.17 (55)		0.05	0.25 (1500)		0.05		0.05	1.49E-10	9.53E+04	1.42E-05
21	3.8E-5	0.74		0.25 (1500)		1.0	0.58 (11.5)	0.95				3.87E-06	1.99E+02	7.71E-04
22	3.8E-5	0.74		0.25 (1500)		1.0	0.17 (55)	0.95				1.14E-06	9.53E+02	1.09E-03
23	3.8E-5	0.74		0.25 (1500)		1.0	0.58 (11.5)		0.05	0.95		1.94E-07	1.99E+01	3.87E-06
24	3.8E-5	0.74		0.25 (1500)		1.0	0.17 (55)		0.05	0.95		5.68E-08	9.53E+01	5.41E-06
25	3.8E-5	0.74		0.25 (1500)		1.0	0.58 (11.5)		0.05		0.05	1.02E-08	1.99E+04	2.03E-04
26	3.8E-5	0.74		0.25 (1500)		1.0	0.17 (55)		0.05		0.05	2.99E-09	9.53E+04	2.85E-04
27	3.8E-5	0.74		0.25 (1500)		1.0	0.25 (1500)		1.0	0.95		1.67E-06	2.60E+03	4.34E-03
28	3.8E-5	0.74		0.25 (1500)		1.0	0.25 (1500)		1.0		0.05	8.79E-08	2.60E+06	2.29E-01
29	3.8E-5		0.26	0.58 (11.5)	0.95							5.44E-06	1.99E+02	1.08E-03
30	3.8E-5		0.26	0.17 (55)	0.95							1.60E-06	9.53E+02	1.53E-03
31	3.8E-5		0.26	0.58 (11.5)		0.05	0.58 (11.5)	0.95				1.58E-07	1.99E+02	3.15E-05
32	3.8E-5		0.26	0.58 (11.5)		0.05	0.17 (55)	0.95				4.63E-08	1.99E+02	9.21E-06
33	3.8E-5		0.26	0.58 (11.5)		0.05	0.25 (1500)	0.95				6.80E-08	1.99E+02	1.35E-05
34	3.8E-5		0.26	0.58 (11.5)		0.05	0.58 (11.5)		0.05		1.0	8.31E-09	1.99E+04	1.66E-04
35	3.8E-5		0.26	0.58 (11.5)		0.05	0.17 (55)		0.05		1.0	2.44E-09	1.99E+04	4.86E-05
36	3.8E-5		0.26	0.58 (11.5)		0.05	0.25 (1500)		0.05		1.0	3.58E-09	1.99E+04	7.14E-05
37	3.8E-5		0.26	0.17 (55)		0.05	0.58 (11.5)	0.95				4.63E-08	1.99E+02	9.23E-06
38	3.8E-5		0.26	0.17 (55)		0.05	0.17 (55)	0.95				1.36E-08	9.53E+02	1.30E-05
39	3.8E-5		0.26	0.17 (55)		0.05	0.25 (1500)	0.95				1.99E-08	9.53E+02	1.90E-05
40	3.8E-5		0.26	0.17 (55)		0.05	0.58 (11.5)		0.05		1.0	1.43E-08	1.99E+04	2.85E-04
41	3.8E-5		0.26	0.17 (55)		0.05	0.17 (55)		0.05		1.0	7.14E-10	9.53E+04	6.81E-05
42	3.8E-5		0.26	0.17 (55)		0.05	0.25 (1500)		0.05		1.0	1.05E-09	9.53E+04	1.00E-04
43	3.8E-5		0.26	0.25 (1500)		1.0	0.58 (11.5)	0.95				1.36E-09	1.99E+02	2.71E-04
44	3.8E-5		0.26	0.25 (1500)		1.0	0.17 (55)	0.95				3.99E-07	9.53E+04	3.80E-04
45	3.8E-5		0.26	0.25 (1500)		1.0	0.58 (11.5)		0.05		1.0	7.16E-08	1.99E+04	1.43E-03
46	3.8E-5		0.26	0.25 (1500)		1.0	0.17 (55)		0.05		1.0	2.10E-08	9.53E+04	2.00E-03
47	3.8E-5		0.26	0.25 (1500)		1.0	0.25 (1500)		1.0		1.0	6.18E-07	2.60E+06	1.61E-00

1.85 person-rem

TABLE 3.5. Dominant Sequence for BWR With MSIV-LCS

Probability of core melt involving loss of power	9.9E-6 per year
Fraction of MSIV leakages resulting in 1500 SCFH	0.0625
Failure of both inboard and outboard MSIV-LCS[a]	1.0
Failure of SWGTS given loss of power	1.0
Partitioning factor for SWGTS given failure	1.0
Ratio of 1500 SCFH and 3000 SCFH source term	0.5
Reference source term	5.2E6 person-rem
Risk equals the product of the above terms	1.61 person-rem per year

(a) MSIV-LCS is automatically isolated for MSIV leakages greater than 100 SFCH.

result indicates that the largest contributors to public risk are those scenarios where MSIV leakage is in excess of the leakage control system capacity. In other words, the LCS has no effect on the dominant risk sequence associated with MSIV leakage.

3.3.3 Comparison of BWRs With and Without LCS

As seen from Tables 3.2 and 3.4, the primary benefit of the LCS in reducing risk is the result of its availability during an event such as the loss of offsite power for handling leakage rates of 11.5 and 55 SCFH. The fact that the outboard LCS can act as a backup to the inboard LCS also results in a slight risk reduction when 11.5 and 55 SCFH leakage rates are compared for cases with and without a LCS. However, since the risk is dominated by MSIV leakage rates for which the LCS is ineffective (greater than 100 SCFH), the actual benefit of a LCS is small.

Results of work by Ridgely and Wohl (1986) on Generic Issue C-8 (NUREG-1169) also support this conclusion. The effectiveness of alternative release pathways to the LCS was quantified. The potential effect of current source term modeling work was considered for alternative MSIV leakage pathways. Among other conclusions, the key findings of the C-8 report are:

- At most plants there are alternative MSIV leakage paths that do not depend on the availability of offsite power and are at least as effective as the LCS systems presently required.
- Alternative pathways for MSIV leakage control that take advantage of the condenser holdup volume are extremely effective in mitigating the

offsite radiological consequences of an MSIV failure to close; this is true even if offsite power is lost.

- In the attempt to meet the current strict MSIV leakage requirements, utilities have sometimes performed excessive maintenance on valves. In some cases, the maintenance has damaged the valves (e.g., seat refurbishment in situ has resulted in out-of-round seats) without providing any substantial safety benefit.

- From the PRA analyses examined, the requirement for a safety-grade LCS could not be defended on a value-impact basis using the value of $1000/person-rem saved.

- The probable ultimate impact of the current work on source terms will be the refinement of some release assumptions used in the C-8 study. The net effect will be to reduce some of the uncertainties and conservatisms of the study, without altering the major conclusions.

When comparing the case for a BWR without a leakage control system with that of a BWR with a leakage control system, a risk benefit of 0.3 person-rem per reactor year is obtained for the leakage control system. Twenty-five plants are assumed to have, or will have, an LCS. Assuming an average remaining lifetime of 30 years, a total risk reduction of 225 person-rem is obtained. The assumptions on the operation of the LCS used in this analysis tended to maximize its risk benefit. The risk benefit of a leakage control system appears to be small. Based on this preliminary analysis, incentives exist to re-evaluate the requirements for these systems based upon their small benefit.

3.4 COST IMPACTS

The cost information given in this section is primarily derived from NUREG-0933 (NRC 1983). These costs were reviewed by PNL staff with experience in nuclear power plant maintenance and felt to be conservative. Costs are discussed in terms of industry and NRC cost for the options of requiring leakage control systems in all BWRs and deleting this requirement. No discounting has been applied to the cost estimates in this section.

3.4.1 Industry Costs

1. The cost to procure and install a LCS is estimated to be $500,000 per plant. An estimated 10 person-weeks per year is required for maintenance and surveillance. At $2,000 per person-week this results in a cost of $20,000 per reactor year for maintenance and surveillance.

2. The one-time cost to disable an LCS in a plant is estimated to be $2000. This is based on the assumption of disabling operations requiring one man-week of effort.

3. If any changes to plant technical specifications are required as a result of disabling the MSIV-LCS, a cost of about $9100 per plant is

estimated. This is based on the assumption of an uncomplicated technical specification change requiring four person-weeks.

4. Assuming 25 plants have, or will have, an LCS and assuming an average remaining lifetime of 30 years, a total industry cost estimate is:

$$(25)(\$12{,}000) + 25(30)(-\$20{,}000) = -\$14.7M \text{ (cost savings)}$$

3.4.2 NRC Costs

1. A cost of $500,000 is estimated in U.S. NRC (1983) to perform necessary tradeoff studies, develop and justify recommended new requirements, review and approve the requirements, and implement the requirements for MSIV leakage control systems. Discussions held with PNL staff involved in ongoing work on Issue C-8, Main Steam Line Leakage Control Systems, support these preliminary cost estimates.

2. If any changes to plant technical specifications are required, an NRC cost of $11,000 per plant is estimated. This estimate is based on cost estimates provided by NRC's Division of Budget and Analysis and includes four man-weeks of technical staff time, two person-weeks of management and legal review and $800 for Federal Register notices.

4.0 FUEL SYSTEM SAFETY REVIEWS

A fundamental concept in the design of nuclear power plants is the provision of multiple fission product barriers to protect the health and safety of the public from releases of radioactive material during normal operations and under accident conditions. The first of these multiple barriers is provided by the fuel cladding in the fuel system. Other fuel system components include the fuel rods (including pellets, cladding, springs, end plugs, fill gas, etc.), burnable poison rods, control rods and various associated hardware such as spacer grids, springs, end plates and channel boxes. Because of its role as the first line of defense in the defense-in-depth design philosophy, the licensee's fuel system design is carefully reviewed by the NRC staff to ensure compliance with applicable regulatory requirements.

After roughly a thousand reactor years of operating experience in the United States, and on the order of four thousand reactor years worldwide, the technology of fuel design and fuel safety review is fairly mature. This accumulated experience and the sophistication of current analytical models and design practices suggests that NRC's reviews of fuel system design information submitted by licensees might be candidates for potential regulatory relaxation without adverse effects on public health and safety. To assist the NRC in evaluating this possibility, this chapter provides information on the risks, costs and benefits that could result if the current regulatory requirements were relaxed.

4.1 OVERVIEW OF REGULATORY REQUIREMENTS AND GUIDANCE

Requirements and guidance for fuel design safety and fuel design reviews are contained in 10 CFR 50, Appendix A, (General Design Criteria 10, 27, and 35); 10 CFR 50.46; 10 CFR 50, Appendix K; Standard Review Plan (SRP) Section 4.2 (NUREG-0800, NRC 1981a); and Regulatory Guides 1.25, 1.3, 1.4, and 1.77. The scope and content of each of these regulatory requirements are discussed briefly.

The general design criteria of 10 CFR 50, Appendix A define the general requirements for fuel safety and the specific requirements for maintaining control rod insertability and core coolability.

Because the initial condition of the fuel at the onset of a loss of coolant accident (LOCA) can be a significant factor in accident progression, 10 CFR 50.46 and Appendix K to 10 CFR 50 identify fuel safety requirements associated with fuel performance during a LOCA. These regulations require the use of a fuel stored energy model that determines "the highest calculated cladding temperature (or, optionally, the highest stored energy)" of the fuel. The fuel stored energy is the thermal driving force of a LOCA.

Section 4.2 of the SRP identifies those fuel performance areas that need to be addressed in calculating the fuel stored energy for LOCAs and other accident sequences. Additionally, it defines the Specified Acceptable Fuel

Design Limits (SAFDLs) that are used to assure that fuel system safety is maintained as required in criterion 10 of Appendix A and 10 CFR 50.46. NRC uses these SAFDLs as the basis for their review of fuel systems.

The regulatory guides contain guidance for fuel design and fuel safety analysis. These include the acceptable analytical methods for analysis of accidents such as LOCAs, fuel handling accidents, and control rod ejection or rod drop accidents. If the methods and assumptions presented in the regulatory guides are not used, the NRC must review and approve the alternative methods and assumptions.

The remainder of this chapter describes the objectives of evaluating the safety impacts of the fuel design review process, summarizes the current procedures used by NRC's Office of Nuclear Reactor Regulation (NRR) in the review of licensing submittals, discusses the relative importance of the SAFDLs, discusses the risk effects and cost reductions of eliminating certain SAFDLs from the review process, and presents the advantages and disadvantages of eliminating certain SAFDLs from the review process.

4.2 OBJECTIVES OF STUDY

As a part of NRC's program to review regulatory requirements that might be relaxed or eliminated to reduce regulatory burdens without compromising public health and safety, PNL conducted a review of the current fuel system safety review requirements. The purpose of the review was to provide information to NRC staff that could be used to improve the efficiency and effectiveness of the fuel review process. Various aspects of the fuel review process were evaluated by PNL to qualitatively assess the safety and cost impact of specific review items. This study does not make recommendations to the NRC on which areas (if any) of the fuel review process should be modified; rather, it provides information that the NRC can consider in making these decisions.

4.3 CURRENT FUEL SYSTEM SAFETY REVIEW PROCEDURES

As part of the licensing process for light water reactors, applicants are required to include information on fuel system design, safety criteria and analytical methods in their Safety Analysis Reports (SARs). The safety criteria are referred to as Specified Acceptable Fuel Design Limits (SAFDLs). Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", provides guidance on the information needed by the NRC staff to perform the necessary safety evaluations, and fuel system design constitutes one of the categories of information that is needed. In reviewing the applicant's SAR, the NRC staff is guided by the SRP. Section 4.2 of the SRP addresses the fuel system review process, and spells out in some detail the review procedures that are followed.

There are several distinct situations in which fuel system safety reviews take place:

- Operating license (OL) submittals. This situation applies to new reactors. These reviews concentrate on those fuel design items that are reactor specific, e.g., fuel surveillance, creep collapse times and differences from the standard design. The effort spent on these reviews is minimal.

- Reload submittals. This situation applies to operating reactors (ORs) under certain conditions when they refuel. In particular, a reload fuel design that requires a change to the Final Safety Analysis Report (FSAR) or the technical specifications, or involves an unreviewed safety question requires a reload submittal and subsequent review. The effort spent on these reviews is minimal.

- Generic design submittals in the form of topical reports. These reports are prepared by fuel vendors when major changes are made in their fuel designs. These design changes are usually made to provide utilities with increased fuel performance. These reports are submitted to NRC for review and approval.

- Submittals containing new analytical methods and/or safety criteria in the form of topical reports. The changes in analytical methods and/or safety criteria are usually made because new data or technology has become available that allows a reduction in the margin of conservatism without reducing fuel safety. This also results in increased fuel performance for the utilities. Again, these reports are, in most cases, prepared by the fuel vendors and submitted to the NRC for review and approval.

The objectives of the fuel system safety review, as defined in the SRP, are to provide assurance that 1) the fuel system is not damaged[a] as a result of normal operation or anticipated operational occurrences, 2) fuel system damage is never so severe as to prevent control rod insertion, when required, 3) the number of fuel rod failures[b] is not underestimated for postulated accidents, and 4) core coolability is always maintained. The general philosophy of the SRP is that SAFDLs be defined in order to attain these objectives. In defining SAFDLs, the fuel vendor or licensee may either follow the guidance in the SRP or propose alternatives that are specific to their design. Once the SAFDLs are defined and approved by the NRC, evaluations are required to show that they have been satisfied by the vendor's fuel design and thus the fuel system is safe for operation. The evaluation methods, if different from those in the regulatory guides, are also submitted to the NRC for approval.

(a) The phrase "not damaged" is defined in the SRP as follows: Fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis.

(b) As defined in the SRP, fuel rod failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached.

The following discussion of the fuel system review process parallels the major areas identified in the Standard Review Plan and is organized into the following topics: design bases (including both criteria and evaluation methods); fuel system description and design drawings; testing, inspection and surveillance plans; and evaluation findings.

4.3.1 Design Bases - Criteria and Evaluation Methods

Section 4.2 of the SRP categorizes fuel failure mechanisms under three major headings that represent varying levels of failure severity: 1) fuel system damage, 2) fuel rod failure, and 3) fuel coolability. The failure mechanisms identified under the first two categories will not, in most cases, result in severe core damage if fuel damage or failure is experienced. The failure mechanisms identified in the fuel coolability category are potentially more serious in terms of possible consequences.

Fuel design review procedures are subdivided further into criteria and evaluations, with the former intended to limit the extent of fuel damage and the latter to provide assurances that the criteria will not be exceeded during normal operation, anticipated transients, off-normal postulated accidents, and design basis accidents. Not all of the failure mechanisms have criteria spelled out in the SRP; rather, it is stated that the applicant must address the subjects and/or include their effects in safety analyses in the form of SAFDLs. New SAFDLs are usually submitted in topical reports to support a new fuel design or a new analytical method. If the new fuel design has not changed the SAFDLs used in previous designs, the reviewer confirms that NRC-approved analytical methods have been used to show the new design meets each SAFDL.

As indicated above and in Section 4.2.II.C of the SRP, the evaluation methods (including operating experience, prototype testing, and analytical predictions) used by the applicants in demonstrating compliance with the SAFDLs are reviewed. Many of the evaluation methods are presented generically in topical reports and are incorporated in the analyses for OL, OR, and fuel design topical report submittals. The review process for new analytical methods determines whether the new method provides adequate assurance that the SAFDL has been met with an adequate margin of safety. In the OL, OR, and fuel review process, a check is made to determine that each of the SAFDLs has been met using NRC-approved methods. A check is also made to see if these NRC-approved models, codes, etc., are used and referenced and whether the results are presented in the OL, OR, and fuel design submittals.

In those cases where items are questioned in the review process, they become confirmatory or open issues that require formal responses from the applicant.

4.3.2 Fuel System Description and Design Drawings

The NRC review verifies that the fuel system description and design drawings presented in Section 4.2.2 of the FSAR furnish a complete and accurate representation of the design and that the information needed for audit evaluations is provided. Completeness is a matter of judgment, but some guidelines are listed in Section 4.2.II.B of the SRP. In general, the applicants typically do not provide data on every parameter listed in Section 4.2.II.B of the SRP but do furnish an acceptable amount of information in sufficient detail. This section and the following are not applicable to the review of analytical methods.

4.3.3 Testing, Inspection, and Surveillance Plans

Testing and Inspection of New Fuel. As indicated in Section 4.2.II.D.1 of the SRP, the testing and inspection programs for the fabrication and onsite receipt of new fuel and control assemblies should be documented, referenced, and summarized in the FSAR. These programs should include verification of significant fuel design characteristics such as cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Currently, reviews of typical submittals indicate that the applicants provide adequate information on this matter and that the documents have been previously approved by the NRC.

On-Line Fuel System Monitoring. Section 4.2.II.D.2 of the SRP indicates that both the sensitivity of the on-line monitoring system and the applicant's commitment to use this system should be evaluated. It should be noted, however, that no guidelines are presented as to the required sensitivity of the on-line monitoring system.

This section also states that surveillance is needed to assure that B_4C control rods are not losing reactivity because of their susceptibility to leaching in the event of a cladding defect. This surveillance is accomplished through reactivity worth tests.

Postirradiation Surveillance. Section 4.2.II.D.3 of the SRP requires that the postirradiation fuel surveillance program to detect anomalies or confirm expected fuel performance be described in the FSAR for each plant. How extensive the program needs to be depends on the fuel design. Using the guidelines in Section 4.2.II.D.3 of the SRP, a less extensive program is needed if the fuel design is similar to the design in use at other operating plants and a more extensive program is needed if the fuel design involves new features. The programs have to include a commitment to perform additional surveillance if unusual fuel behavior is observed or detected. The programs must also address the disposition of failed fuel. In the review for a given plant, the adequacy of the postirradiation surveillance program is evaluated on the basis of the fuel design (e.g., its previous performance, if available).

4.3.4 Evaluation Findings

The review involves verifying that sufficient information has been provided by the applicant to satisfy the requirements of Section 4.2 of the SRP and that the evaluation supports the conclusions stated in Section 4.2.IV of the SRP. As part of the review process, the confirmatory and open issues that need to be resolved before NRC approval is granted are listed here.

These issues require formal responses from the applicants which are, in turn, reviewed by the NRC leading ultimately to resolution of the issues.

4.4 CONSEQUENCES OF CHANGES TO FUEL REGULATORY REVIEW PROCESS

As noted earlier, the purpose of this analysis is to provide information that the NRC staff can consider in deciding what modifications, if any, to make to the current fuel system safety review process. As explained in Chapter 1, the potential for such modifications is being studied by the NRC as part of a comprehensive program to review existing light water reactor regulatory requirements to see if some could be relaxed or eliminated to reduce regulatory burdens without compromising public health and safety. Fuel system safety reviews constitute one of the three areas of regulation selected by NRC staff for study in the initial (pilot) phase of the program.

To perform the analysis, a three-step process is followed. First, it is necessary to postulate a set of possible regulatory changes for consideration in the analysis; i.e., some alternatives to the current regulatory review process are identified as potential candidates. Second, to develop information on the advantages and disadvantages of these potential alternatives, an assessment is made of the effects that each alternative might have on public health and safety, industry costs, and NRC costs, if it were implemented; where appropriate, other effects such as occupational radiation exposure are also considered. Third, after the potential effects of each alternative have been assessed, a concise summary is prepared for consideration by NRC staff. The decision about what changes, if any, should be made is left up to NRC staff.

This section of the report presents PNL's assessment of the effects that could result if the NRC chooses to implement certain modifications of the current review process for fuel systems. The modifications postulated here, for purposes of analysis, are the selective elimination of certain items from the review process. Specifically, the question addressed in this section is the following: What would be the consequences, in terms of public health and safety, industry costs, and NRC costs, if some SAFDLs and their corresponding evaluation methods were eliminated from the fuel system safety review process?

The assessment is organized into three sections. Section 4.4.1 presents a qualitative categorization of each SAFDL in terms of its importance to public health and safety. Section 4.4.2 provides a quantitative perspective on the potential public risk effects of modifying the fuel system safety review process. Section 4.4.3 discusses the NRC and industry cost effects. The results of the assessment are then concisely summarized in Section 4.5.

4.4.1 Fuel Safety Issues in Categories of Importance to Plant and Public Safety

Table 4.1 lists the SAFDLs addressed in Section 4.2 of the Standard Review Plan along with a qualitative categorization of their importance to plant and public safety. As noted from this table, operating license (OL) and operating reactor (OR) reviews typically require very little effort on the part of the reviewer. These reviews concentrate only on reactor-specific issues, such as fuel surveillance, cladding creep collapse times, seismic criteria, and any differences from the standard fuel design.

The process of placing SAFDLs in the categories shown in Table 4.1 is primarily qualitative in nature. It is based on the engineering judgment of PNL staff members who have extensive background and experience in fuel design and performance and who also have experience with the NRC fuel safety review procedures in Section 4.2 of the SRP.

The three categories of significance to plant and public safety are defined as follows:

- Category I-- A small number of fuel failures (less than 1% of the fuel in the core) might occur.[a] Occupational exposures to plant workers would remain generally in the small-to-moderate range. Offsite doses would be insignificant. This assumes that plant filters work as designed and do not allow significant activities to be released to the environment.
- Category II-- A moderate to large number of fuel failures (greater than 1% of the fuel in the core) might occur,[a] but the core and safety systems would remain intact and would perform acceptably. Occupational exposures to plant workers might be relatively high but offsite doses would be small-to-moderate.
- Category III-- A large number of fuel failures might occur. Coolable core geometry (i.e., fuel coolability) might be lost. High exposures to plant workers might result. Significant offsite doses might occur. These types of consequences might, for example, occur as the result of an accident involving a lack of reactivity control (control rod failure to insert) or a loss of primary coolant to the core (LOCA) followed by ECCS failures.

(a) The one percent criterion is based on the fact that many plant filter systems are designed to handle up to one percent fuel failures without any significant release to the environment. As noted earlier, "fuel failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached.

TABLE 4.1. A Tentative Categorization of Fuel Damage and Failure Mechanisms

	Category of Safety Significance[1]: Operating License Reviews (New Reactors) and Operating Reactor Reviews (Reloads)	Generic Topical Report Reviews: New Design	Generic Topical Report Reviews: Analytical Methods and Criteria
1. FUEL SYSTEM DAMAGE			
(a) Design Stress	*	I	II
(b) Design Strain	*	I	II
(c) Strain Fatigue	*	I	II
(d) Fretting Wear:			
BWR Fuel Rods and Channel Boxes	*	I	II
PWR Fuel Rods and Guide Tubes	**	II	III
(e) Oxidation, Hydriding, and Corrosion Product (Crud) Buildup	*	I	II
(f) Rod Bowing	*	II	II
(g) Axial Growth	*	I	II
(h) Rod Pressure	*	II	II
(i) Assembly Liftoff	*	I	I
(j) Control Material Leaching	*	I	II
2. FUEL ROD FAILURE			
(a) Hydriding	*	I	I
(b) Cladding Collapse	*	I	II
(c) Overheating of Cladding	*	II	II
(d) Overheating of Fuel Pellets	*	II	II
(e) Pellet/Cladding Interaction	*	I	I
(f) Cladding Rupture	*	III	III
(g) Mechanical Fracturing	III	III	III
3. FUEL COOLABILITY			
(a) Fragmentation of Embrittled Cladding	*	III	III
(b) Violent Expulsion of Fuel Material	*	III	III
(c) Cladding Ballooning and Flow Blockage	*	III	III
(d) Structural Damage from External Forces	III	III	III

*Not typically reviewed for these submittals.
**Reviewed for some specific designs.

(1) Categories of safety significance:

I - Small number of fuel failures (less than or equal to 1%) might occur; exposure of plant workers on the average would be small to moderate; offsite doses would be insignificant.

II - Moderate to large number of fuel failures (greater than 1%) might occur; core and safety systems would remain intact; might lead to high exposures to plant workers but offsite dose would be small to moderate.

III - Large number of fuel failures might occur; core and safety systems might not remain intact; coolable fuel geometry might be lost; high exposures to plant workers and significant offsite doses might result.

In establishing the three categories, the intent was to estimate what consequences might occur if SAFDLs were eliminated from the review process. It should not be assumed that these consequences would necessarily occur. In order for the adverse consequences to actually materialize, elimination of the SAFDLs would have to be accompanied by several other occurrences such as significant errors in the fuel design process and failure of the fuel vendor or utility to detect these errors before they result in the adverse consequences. In some cases (e.g., Category III), an independent initiating event such as a loss of coolant accident (LOCA) or a transient, would also have to occur. Thus, the categories reflect bounding estimates of the potential consequences of eliminating SAFDLs, without any implication about the probabilities associated with those consequences.

As can be seen in Table 4.1, different safety categories are sometimes assigned to design reviews and reviews of analytical methods and safety criteria even though the safety issues are the same. The reason for these differences is a greater chance for larger, more severe errors in the development of analytical methods and safety criteria than for errors in design, because analytical methods and safety criteria are complex and somewhat abstract. Therefore, errors or lack of conservatism in these areas are more difficult to recognize and less likely to be discovered by the vendor or licensee.

The SRP sections on fuel system description and design drawings and testing, inspection and surveillance plans were not included in Table 4.1. These sections will be discussed briefly here to establish their importance in the regulatory review process. The fuel design reviews use the fuel system description and design drawings section as reference material to evaluate the safety issues defined in Table 4.1. Consequently, there is no formal review of this section per se; however, it is valuable in the review of safety issues. The testing, inspection and surveillance plans section of the SRP requires very little review effort and the safety benefits are large. Section 4.2 of the SRP contains three subsections on testing and inspection of new fuel, on-line fuel monitoring system, and postirradiation fuel surveillance. The safety benefits of testing and inspection of new fuel are large in relation to the effort, because testing and inspection help identify fabrication errors that can lead to fuel failures. The on-line fuel monitoring system helps to detect fuel failures by monitoring in-reactor activities. It allows early detection and an estimate of the level of fuel failures. The loss of B_4C from control rods and thus loss of reactivity control is also addressed in this subsection. The loss of B_4C control rod reactivity must be monitored and this is accomplished by a surveillance program. This surveillance program is important for assuring the ability to shut down the reactor when necessary. The postirradiation surveillance section addresses the need for postirradiation fuel surveillance so that adverse in-reactor fuel behavior can be detected before it becomes a significant safety issue. This particular subsection is valuable in monitoring new fuel designs. Consequently, plants with new fuel designs are required to have a more extensive fuel surveillance program than older designs where the requirements are minimal.

4.4.2 Quantitative Risk Perspective

The purpose of this section is to lend some quantitative support to the essentially qualitative information in the preceding section. Table 4.1 and the accompanying discussion were intended to provide a qualitative perspective on the potential consequences that might result if certain SAFDLs were eliminated from the review process. In this section, an attempt is made to give a quantitative perspective on those potential consequences.

It is important to stress at the outset that the estimates presented here represent the consequences that might result if SAFDLs in the various categories were eliminated from the review process. It should not be assumed that these consequences would necessarily occur. In fact, for the adverse consequences to materialize, elimination of the SAFDLs would have to be accompanied by several other occurrences such as significant errors in the fuel design process and failure of the fuel vendor or utility to detect these errors before they result in adverse consequences. In some cases (e.g., severe accidents), an independent initiating event such as a LOCA or a transient, would also have to occur. Thus, as with the case in the preceding section, the information presented in this section represents bounding estimates of the potential consequences of eliminating SAFDLs, without any implication about the probabilities associated with those consequences. (In order to assess the probabilities, it would be necessary to estimate such quantities as the probability of design errors assuming certain SAFDLs were eliminated from the review process. Any such estimate would be highly speculative.)

The approach taken in this section is to select from existing probabilistic risk assessments (PRAs) accident scenarios that may be considered representative of the three categories of safety significance defined in the preceding section. Quantitative risk information on the selected scenarios is then calculated. These quantitative estimates are the basis for discussing the risk significance of the three categories. The estimates are intended to be illustrative, rather than rigorous assessments of the risk that could result if SAFDLs were eliminated.

4.4.2.1 Risks Associated with Category I

According to the definition of Category I, failure to satisfy a Category I SAFDL might lead to a small number of fuel failures (less than 1%), but the failures would remain localized and coolable core geometry would be maintained. Since plant systems are already designed to accommodate up to 1% fuel failures without significant public health and safety consequences, the risks associated with eliminating Category I SAFDLs would be minimal. In existing probabilistic risk assessments, risks associated with the small releases that occur during normal operations are negligible, and for this reason they are not explicitly analyzed. In summary, eliminating a Category I SAFDL from the review process would not be expected to result in any significant adverse effect on public health and safety.

4.4.2.2 Risks Associated with Category II

According to the definition of Category II, failure to satisfy a Category II SAFDL might result in a moderate to large number of fuel failures (greater than 1%) but the core would remain coolable and intact. At the low end of the range of fuel failures (close to 1%), these consequences would be similar to those discussed above for Category I; they would be negligible in terms of their impact on public health and safety. At the high end of the range (near 100%), the consequences can be roughly bounded by considering the case of a mitigated LOCA, which is modeled in existing PRAs. In a mitigated LOCA, as modeled in the Reactor Safety Study (WASH-1400), 100% of the radioactivity in the gap between fuel pellet and cladding is instantaneously released into containment; it is assumed that 100% of the fuel rods fail. The PWR 9 and BWR 5 release categories in WASH-1400 correspond to mitigated LOCAs in which all safety features function properly. In the Handbook for Value-Impact Assessment, NUREG/CR-3568 (Heaberlin et al. 1983), generic population doses were calculated for the WASH-1400 release categories. (These are presented in this report in Table 2.1). The population dose for PWR-9 is estimated as 120 person-rem; for BWR-5, it is estimated as 20 person-rem. These doses represent the consequences to the public for the specific accident sequence, and do not consider the release frequency.

These estimates give some insight (as conservative upper bounds) into the potential consequences of eliminating a Category II SAFDLs from the review process. In interpreting these numbers, it should be remembered that mitigated LOCAs assume instantaneous release of the gap activity from 100% of the fuel rods, and that eliminating SAFDLs from the review process could result in a moderate to large number of fuel failures only if 1) errors in design occurred, and 2) these errors were not detected by the vendor or utility before they led to adverse consequences. Thus, the actual risk impacts would probably be a small fraction of these conservative upper bounds.

4.4.2.3 Risks Associated with Category III

According to the definition of Category III, failure to satisfy a Category III SAFDL might result in a large number of fuel failures and might conceivably lead to a situation in which the core and safety systems might not remain intact and able to function properly, and coolable geometry might be lost. It is clear that the public health and safety consequences of such a situation could be significant. To illustrate the magnitude of the risk potential, a simple risk sensitivity calculation is presented.

In existing PRAs such as WASH-1400, large break LOCAs are generally very minor contributors to risk. This is because of the relatively low probability of the initiating event (i.e., large break) and the effectiveness of engineered safety features (ESFs) in responding to these events if they occur. Eliminating Category III SAFDLs from the review process would not be expected to affect the probability of a large break. However, the concern is that the capability for reactor shutdown and cooling could be degraded. For example, fuel design errors might occur that could compromise the ability to scram or that could impede the ability to cool the core. Thus, fuel design errors could increase the probability of a core melt.

In an attempt to bound this risk, it is assumed that in 10% of all large break LOCAs, fuel design errors result in failure of the ECCS to effectively remove heat from the core, ultimately leading to core melt (e.g., PWR sequence AD, BWR sequence AE, using the terminology of WASH-1400). The results of the risk calculations are shown in Table 4.2. They indicate that loss of core coolability can have a significant impact on risk, particularly for the BWR case; as indicated in Chapter 2, Table 2.2, the baseline risk levels for the BWR case are on the order of 150 person-rem per reactor year (ry), and the 54 person-rem increment calculated for the BWR case is a sizable fraction of this baseline. This calculation is not to be interpreted as a rigorous measure of the risk impact of eliminating Category III SAFDLs from the review process. Rather, it is an indication that there is a potential for such an impact. To fully measure the impact, the potential effects of fuel design errors on the full spectrum of accident scenarios would have to be determined, and the probabilities of such errors would also have to be estimated.

TABLE 4.2. Potential Risk Impacts from Loss of Core Coolability

Release Category	Containment Failure Probability	Release Frequency, releases/ry	Release Consequence, person rem/release	Risk Contribution, person-rem/ry
PWR-3	0.01	4E-7	5.4E+6	2.2
PWR-5	0.002	8E-8	1.0E+6	0.08
PWR-7	1.0	4E-5	2.3E+3	0.09
PWR Total		4E-5		2.4
BWR-1	0.02	2E-7	5.4E+6	1.1
BWR-2	0.1	1E-6	7.1E+6	7.1
BWR-3	0.9	9E-6	5.1E+6	46.
BWR Total		1E-5		54.

To summarize the risk considerations, fuel system safety reviews appear to be particularly important to risk if the SAFDL in question contributes to the prevention of fuel design errors that affect either the initiation or progression of accidents leading to severe core damage (i.e., Category III above). Those aspects of the reviews that contribute to controlling fuel failures, but are not related to coolability (i.e., Categories I and II), are less important.

In order for any of these potential adverse consequences to materialize, elimination of NRC reviews would have to be accompanied by several other occurrences such as significant errors in the fuel design process and failure of the fuel vendor or utility to detect these errors before they result in the adverse consequences. In the case of the severe accidents, an initiating event, such as a LOCA or transient, would also have to occur.

4.4.3 Cost Considerations

Potential cost savings and considerations as a result of streamlining fuel system safety reviews are discussed in this section. Cost considerations for the NRC and for industry are treated separately.

4.4.3.1 NRC Cost Considerations

The NRC resources currently devoted to fuel design reviews are discussed below. The objective is to provide a rough estimate of potential NRC cost savings if some steps in the current review process are eliminated.

The range of NRC staff time, in percent, spent in the review process in each of the major categories in the SRP that relate to fuel system design is provided in Table 4.3. This illustrates the large degree of variability between different reviews. Several examples of this variability will be discussed here to provide insight into the difficulty of arriving at a single estimate of the time spent on any particular item in the SRP.

One illustration is the case of a fuel vendor who submits a new fuel design for review. Suppose that the only changes from previously approved designs are increases in the plenum volume and the heat rating limit. The review for this submittal would likely concentrate on design stress and strain, corrosion, rod pressures, cladding collapse, overheating of cladding, LOCA-related analyses (e.g., cladding rupture and ballooning) and fuel surveillance. In like manner, an OL review for a new plant would likely concentrate on those design areas that have changed from previously approved plants. The OL review typically only concentrates on reactor-specific issues such as fuel surveillance, cladding creep collapse times, and seismic criteria that impact fuel safety issues. Similarly, OR reviews only concentrate on changes in fuel operating limits or design changes from the previous fuel design. However, in the case of the Extended Burnup Topical Reports submitted by the fuel vendors, all criteria and SAFDLs are reviewed for their applicability to the extended burnup range requested by each vendor. In many cases, the unresolved issues between the applicant and the NRC can consume up to 60% of the NRC's time in the review process. For example, if an issue arose on the LOCA analysis methods used for a specific design (or in the generic LOCA methods) this could take a considerable amount of time to resolve because of the complexity of the question.

Bearing these caveats in mind, the information in the table can be used to roughly estimate or bound the potential NRC cost savings that might result from streamlining fuel design reviews. If all items falling in Category I were eliminated, this might constitute a savings of only 5 to 15% for OL and OR reviews, because typically very few Category I SAFDLs are addressed in these reviews. A savings of 10 to 30% might be experienced for new design reviews and only 5 to 10% for the review of analytical methods and criteria. In terms of dollars, the savings would be small.

TABLE 4.3. Tentative Estimate of the Range of Cost Significance of Fuel System Design Reviews to the NRC

	Estimated Range of Cost Significance to NRC, % of Total Effort[a]
DESIGN BASES AND EVALUATION	60-95
1. FUEL SYSTEM DAMAGE	10-70
(a) Design Stress (or Loading Limit)	
(b) Design Strain (or Loading Limit)	
(c) Strain Fatigue	
(d) Fretting Wear:	
BWR Fuel Rods and Channel Boxes	
PWR Fuel Rods and Guide Tubes	
(e) Oxidation, Hydriding, and Corrosion Product (Crud) Building	
(f) Rod Bowing	
(g) Axial Growth	
(h) Rod Pressure	
(i) Assembly Liftoff	
(j) Control Material Leaching	
2. FUEL ROD FAILURE	10-70
(a) Hydriding	
(b) Cladding Collapse	
(c) Overheating of Cladding	
(d) Overheating of Fuel Pellets	
(e) Pellet/Cladding Interaction	
(f) Cladding Rupture	
(g) Mechanical Fracturing	
3. FUEL COOLABILITY	10-70
(a) Fragmentation of Embrittled Cladding	
(b) Violent Expulsion of Fuel Material	
(c) Cladding Ballooning and Flow Blockage	
(d) Structural Damage from External Forces	
DESCRIPTION AND DESIGN DRAWINGS	0-10
TESTING, INSPECTION, AND SURVEILLANCE PLANS	0-40
1. Testing and Inspection of New Fuel	
2. On-line Fuel System Monitoring	
3. Postirradiation Surveillance	

(a) Percent of total NRC staff effort expended on fuel system design reviews.

Eliminating both Category I and II SAFDLs might yield 5 to 20% savings for OL and OR reviews, a 20 to 35% savings for new design reviews and a 20 to 40% savings for the review of analytical methods and criteria. Note, however, that this estimate is subject to wide variability depending on the circumstances. It is difficult to convert this estimate into actual dollar savings because of the variability and uncertainty in the time and resources devoted to reviews. It is clear, however, that the total savings would be small.

4.4.3.2 Industry Cost Considerations

On the industry side, only qualitative statements about potential cost savings are available. Discussions were held with a number of utilities and industry groups, including fuel vendors. The consensus of those contacted was that streamlining the review process by eliminating certain steps would have minimal effect on industry costs. In other words, no significant cost savings would be expected, although some small cost savings might be achieved in certain situations. The basic reason for this conclusion is that industry would continue to design fuel in the same way. Current fuel design practices have developed and matured over the years and they work reasonably well, meeting the needs of the vendor, the utility, and the NRC. In the absence of some significant improvement in economics or performance, there is little incentive for industry to change current practices. Eliminating some steps in the review process was not judged to offer any significant improvement in economics or performance.

Some small incremental savings would probably accrue, however. For example, the number of open and confirmatory issues between the NRC and licensees would probably decline. Since resolving each such issue consumes industry resources, reducing the number of issues would result in a cost savings; however, this was judged to be small in relation to the overall costs associated with fuel design. No estimate of the absolute magnitude of these cost savings is available. Similarly, the volume of paperwork associated with the fuel design process might be reduced slightly, with some small savings as a result; again, this was judged to be insignificant in relation to the overall costs of fuel design.

In conclusion, eliminating some steps in current fuel design review procedures could be accomplished. The benefits of doing so would be marginal. The savings to the NRC for the fuel system safety reviews would be small to moderate depending on the number of SAFDLs eliminated from the review process; in relation to the overall NRC review process the savings would be very small. For the industry, the savings would be very small in absolute terms and in relation to the overall costs associated with the fuel design.

4.5 CONCLUSIONS

This analysis has not recommended which fuel safety issues (SAFDLs) be eliminated (if any) from the regulatory review process. This decision is left to the NRC. The paragraphs that follow concisely present some of the advantages and disadvantages of eliminating some SAFDLs from NRC review.

4.5.1 Advantages of Eliminating Some SAFDLs From NRC Review

- Reducing the number of SAFDLs in the regulatory review process will reduce the amount of time the NRC staff and industry spend on fuel safety reviews.

- Reducing the number of safety issues in the review process could help the inexperienced licensee or NRC reviewer concentrate on those fuel issues of greatest safety importance.

- Based on industry experience, the degree of severity for some fuel safety issues is not great for particular reviews (see Table 4.1). Fuel failures are currently below approximately 0.01% and thus have minimal impact on public safety.

- It should also be noted that the adverse consequences of fuel failure in each of the three categories listed in Section 4.4.1 are upper limits. In order for these consequences to actually occur two or more of the following things must happen in addition to the changes in the review process:

  - errors or lack of conservatism in the applicant's analysis methods or safety criteria which reduce the safety margin of the fuel

  - errors in the applicant's design and safety calculations which reduce the safety margin of the fuel

  - the above errors must not be detected by the vendor's or licensee's quality assurance program

  - the on-line fuel monitoring system fails to detect fuel failures prior to significant damage

  - for some fuel issues, an additional event such as an accident (e.g., a LOCA) would have to occur first, before the assumed consequences could occur.

4.5.2 Disadvantages of Eliminating Some SAFDLs From NRC Review

- The primary disadvantage of eliminating Category I and/or Category II SAFDLs from the review process is that there would be no independent oversight of the fuel vendors and utilities to verify that new fuel designs, criteria and analytical models are developed and applied correctly. The vendors and utilities are already required to assure the correctness of their safety analyses (through QA procedures). Eliminating or reducing NRC reviews of new designs, criteria and analytical methods might increase the probability for error, particularly for criteria and analytical methods that are difficult to verify with QA programs. This is of particular concern, because utilities are urging the vendors to provide improved fuel performance. The utilities typically do not check the vendor's criteria and analytical models and many times do not check their safety analyses in any detail, because they generally lack the technical expertise to perform these functions.

- Elimination of Category I SAFDLs may have an insignificant effect on general public safety; however, it may have a small but measurable impact on plant occupational exposures.

- In like manner, elimination of Category II SAFDLs may have a small to moderate effect on the general public exposures; however, the effect on occupational exposures may be large.

- The NRC staff already simplifies the NRC review process in an informal manner by addressing only those design aspects or portions of the analytical methods that diverge from previously approved designs or analytical methods. The level of effort spent on these items is commensurate with their importance to fuel system safety. Consequently, the advantages of formalizing this simplified approach further are not great, while the disadvantages are that it administratively reduces the flexibility of the review process.

REFERENCES

Aldrich, D. C., et al. 1982. Technical Guidance for Siting Criteria Guidance Development. NUREG/CR-2239, SAND-81-1549, Sandia National Laboratory, Albuquerque, New Mexico.

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800, Pacific Northwest Laboratory, Richland, Washington.

ANSI/ANS 56.8-1981. Containment System Leakage Testing Requirements. American Nuclear Society through American National Standards Institute.

Code of Federal Regulations, Title 10, Part 50 (10 CFR 50). "Domestic Licensing of Production and Utilization Facilities." 1986. Appendices: A, I, J and K.

Code of Federal Regulations, Title 10, Part 100 (10 CFR 100). "Reactor Site Criteria." 1986. Government Printing Office, Washington, D.C.

Cranston, G. V. 1972. Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants. BN-TOP-1, Bechtel Corporation, San Francisco, California.

DiNunno, et al. 1962. Calculations and Distance Factors for Power and Test Reactor Sites. TID-14844, U. S. Atomic Energy Commission, Washington, D.C.

Dougan, J. R. 1984. Evaluation of Containment Leak Rate Testing Criteria. NUREG/CR-3549, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Frank, S., et al. 1982. Containment Integrated Leak-Rate Testing Improvements. EPRI-NP-2726, Stone & Webster Engineering Corporation, Boston, Massachusetts.

Gieseke, J. A., et al. 1984. Radionuclide Release Under Specific Accident Conditions. BMI-2104, Battelle Columbus Laboratories, Columbus, Ohio.

Hatch, S., et al. 1981. Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant. NUREG/CR-1659/4, Sandia National Laboratories, Albuquerque, New Mexico.

Heaberlin, S. W., et al. 1983. A Handbook for Value-Impact Assessment. NUREG/CR-3568, Pacific Northwest Laboratory, Richland, Washington.

Hermann, O. W. and T. J. Burns. 1984. Impact of Containment Building Leakage on LWR Accident Risk. NUREG/CR-3539, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Manion, W. J., et al. 1980. Decommissioning Handbook. DOE/EV/10128-1, U. S. Department of Energy, Washington, D. C.

Mullen, M. F., et al. 1986. Identification of Regulatory Requirements that May Have Marginal Importance to Risk. NUREG/CR-4330, PNL-5809, Volume 1, Pacific Northwest Laboratory, Richland, Washington.

Nuclear News. 1985. Word List of Nuclear Power Plants as of June 30, 1985.

Ridgely, J. N., and M. L. Wohl. 1986. Resolution of Generic Issue C-8: An Evaluation of Boiling Water Reactor Main Steam Isolation Valve Leakage and the Effectiveness of Leakage Treatment Methods. NUREG-1169, U. S. Nuclear Regulator Commission, Washington, D.C.

Ritchie, L. T., et al. 1983. Calculations of Reactor Accident Consequences Version 2 CRAC2: Computer Code User's Guide. NUREG/CR-2326, Sandia National Laboratories, Albuquerque, New Mexico.

Ritchie, L. T., et al. 1984. CRAC2 Model Description. NUREG/CR-2552, Sandia National Laboratories, Albuquerque, New Mexico.

Science and Engineering Associates, Inc. 1985. Cost Analysis of Revisions to 10 CFR Part 50, Appendix J, Leak Tests for Primary and Secondary Containments of Light-Water-Cooled Nuclear Power Plants. Contract NRC 33-84-407, Washington, D. C.

U.S. Nuclear Regulatory Commission. "Public Notice of Availability of Program Plant to Review Effectiveness of LWR Regulatory Requirements in Limiting Risk." 49 Fed. Reg. 39066 (October 3, 1984).

U.S. Nuclear Regulatory Commission. 1975. Reactor Safety Study. WASH-1400, NUREG/75/014, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1981a. Standard Review Plant for the Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1981b. Reactor Safety Study Methodology Applications Program: Oconee 3 and Grand Gulf I. NUREG/CR-1659, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1983. Prioritization of Safety Issues. NUREG-0933, Washington, D. C.

U.S. Nuclear Regulatory Commission. 1984a. Regulatory Analysis Guidelines of the U. S. Nuclear Regulatory Commission. NUREG/BR-0058, Revision 1, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1984b. Policy and Planning Guidance for 1984. NUREG-0885, Issue 3, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1985. Policy and Planning Guidance for 1985. NUREG-0885, Issue 4, Washington, D.C.

U.S. NRC Regulatory Guide 1.3. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." Washington, D.C.

U.S. NRC Regulatory Guide 1.4. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." Washington, D.C.

U.S. NRC Regulatory Guide 1.25. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." Washington, D.C.

U.S. NRC Regulatory Guide 1.70. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plant." Washington, D.C.

U.S. NRC Regulatory Guide 1.77. "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." Washington, D.C.

U.S. NRC Regulatory Guide 1.96. "Design of Main Steam Isolation Valve leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." Washington, D.C.

APPENDIX A

WASH-1400 RELEASE CATEGORIES

PWR 1

This release category can be characterized by a core meltdown followed by a steam explosion resulting from contact of molten fuel with the residual water in the reactor vessel. It is assumed that the steam explosion would rupture the upper portion of the reactor vessel and breach the containment barrier, with the result that a substantial amount of radioactivity might be released from the containment in a puff over a period of about 10 minutes. If the containment is at an elevated pressure at the time of the steam explosion, containment failure will be accompanied by a very high sensible energy release. With a low containment pressure, as would be the case if the containment safety features are available, a lower sensible energy release would still occur due to the steam generated by the steam explosion itself. The sweeping action of gases generated following reactor vessel melt-through and during containment-vessel melt-through, would continue but at a relatively lower rate. The total release was estimated to contain approximately 70% of the iodines and 40% of the alkali metals present in the core at the time of release. This category also includes certain potential accident sequences that would involve the occurrence of core melting and a steam explosion after containment rupture due to overpressure. In these sequences, the rate of energy release at the time of the steam explosion would be somewhat lower, although still relatively high.

PWR 2

This category is representative of accident sequences in which containment failure takes place relatively shortly after core melting, implying failure of core-cooling systems concurrent with the failure of containment spray and heat-removal systems. Failure of the containment barrier would occur through overpressure either by excessive steam generation or due to hydrogen burning, causing a substantial fraction of the containment atmosphere to be released in a puff over a period of about 30 minutes. Due to the sweeping action of gases generated by containment melt-through, some release of radioactive material would continue but at a relatively lower rate thereafter. The total release would contain approximately 70% of the iodines and 50% of the alkali metals present in the core at the time of release. The high temperature and pressure within containment at the time of containment failure would result in a relatively high release rate of sensible energy from the containment. This category is also intended to cover core melting sequences that may be initiated by system ruptures located outside containment. In such sequences the core is predicted to melt with the releases essentially bypassing the containment and containment mitigating systems.

PWR 3

This category involves an overpressure failure of the containment due to failure of containment heat removal which in turn interacts with and fails core cooling systems. Containment failure would occur prior to the commencement of core melting. Core melting would then cause radioactive materials to be released through a ruptured containment barrier. It is estimated that approximately 20% of the iodines and 20% of the alkali metals present in the core at the time of release would be released to the atmosphere. Most of the release could occur over a period of about 1-1/2 hours. The driving forces for the release of radioactive materials from containment would be the subsequent meltdown processes and the sweeping action of gases generated by the reaction of the molten fuel with concrete. Since these gases would be initially heated by contact with the melt, the rate of sensible energy release to the atmosphere would be moderately high.

PWR 4

This category involves failure of the core-cooling system and the containment spray system after a loss-of-coolant accident, together with a concurrent failure of the containment system to properly isolate. This would result in an estimated release of almost 9% of the iodines and 4% of the alkali metals present in the core at the time of release. Most of the release would occur continuously over a period of 2 to 3 hours. Due to the restricted leak rate and extended period of release, a relatively low rate of release of sensible energy would be associated with this category.

PWR 5

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

PWR 6

This category involves a core meltdown due to failure in the core cooling systems after a LOCA or transient initiating event. The containment sprays are not available for mitigating the radioactive material released into the containment, but the containment barrier is predicted to retain its integrity until the molten core proceeded to melt through the concrete containment base mat. The containment pressure would remain relatively high but below the estimated failure pressure. The radioactive materials would thus be released into the ground, with some leakage to the atmosphere occurring upward through the ground with most of the atmospheric release being noble gases. Direct leakage to the atmosphere would also occur at a low rate prior to pressure relief following containment-vessel melt-through. It was also assumed that this direct leakage occurred at a volumetric rate of ~1%/day. Most of the release would occur continuously over a period of about 10 hours. The release would include approximately 0.08% of the iodines and alkali metals present in the core at the time of release. Because leakage from containment to the atmosphere would be low and gases escaping through the ground would be cooled by contact with the soil, the energy release rate would be very low.

PWR 7

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

PWR 8

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

PWR 9

This category approximates a PWR design basis accident (large pipe break), but with about a ten fold deterioration in the Containment design leakage rate, in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt. It is assumed that the minimum required engineered safeguards would function satisfactorily to remove heat from the core and containment. Because of the subatmospheric features of the containment the release would occur over about a 0.5-hour period during which the containment pressure would be above ambient. Thereafter, there would be no additional leakage. Approximately 0.00001% of the iodines and 0.0006% of the alkali metals would be released. As in PWR release category 8, the energy release rate would also be very low.

BWR 1

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

BWR 2

This release category is made up of sequences in which core meltdown is caused by failure to remove decay heat or failure of the emergency cooling system. Containment overpressure failure occurs either before core melt or during the core meltdown process. The key characteristic is a containment failure location such that radioactivity would be released directly to the atmosphere without significant retention of fission products within the confinement building which surrounds most of the BWR primary containment. Most of the release would occur over a period of about 3 hours. This category involves a relatively high rate of energy release due to the sweeping action of the gases generated during core meltdown process. Approximately 90% of the iodines and 50% of the alkali metals present in the core are estimated to be released to the atmosphere. The most probable sequence in this category is a transient event with failure of the decay heat removal system.

BWR 3

This release category represents core meltdown sequences caused by transient events accompanied by a failure to scram or a failure to remove decay heat, as well as loss-of-coolant accidents with failure to remove decay heat or failure of emergency coolant injection. It is similar to Category 2 except some fission-product retention would occur either in the suppression pool or the reactor building prior to release to the atmosphere. Most of the release was assumed to occur over a period of about 3 hours and is estimated to involve 10% of the alkali metals. For those sequences (e.g., loss of decay heat removal) in which the containment would fail due to overpressure before core melt, the rate of energy release to the atmosphere during the subsequent core melt, would be somewhat smaller, although still moderately high.

BWR 4

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

BWR 5

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

APPENDIX B

COST IMPACT SUPPORTING INFORMATION

Section B.1 of this appendix contains specific cost impact supporting information for the estimates previously presented in Section 2.4 of this report. General cost impact supporting information obtained via telephone from selected contacts made during the course of this study is presented in Section B.2.

B.1 SPECIFIC SUPPORTING INFORMATION

The supporting information for the estimates previously presented in Section 2.4 of this report follow.

B.1.1 Supporting Information for Section 2.4.4 - Cost and Occupational Radiation Exposure Related to Local Leakage Rate Testing (Type B and C Tests)

The representative cost for Type B and C tests is roughly estimated as follows. Recognizing that work output in radiation areas can be severely reduced (e.g., by as much as 50%) from theoretical just due to protective clothing and other radiation control measures (Manion et al. 1980) a 25% reduction is assumed for purposes of this analysis. This latter assumption equates to a 75% efficiency for the 275 man-hours in the radiation field for the conduct of Type B and C tests at a generic LWR as reported in SEA (1985); this yields a calculated proportional value of 367 hours for actual total labor time associated with these tests. Again, drawing on SEA, Section 6.0 Cost Estimate Basis (1985) for industry labor rates, a labor rate of $38/hr is calculated based on an average of the labor rates given for support engineers and mechanics. However, since SEA (1985) reported these labor rates in 1984 dollars, they are adjusted by a factor of 1.05 (representing a one-year inflation rate at 5%) to bring them in line with the early-1985 costs used in this study.

The base wage bill (BWB) then becomes:

367 hours ($38/hr x 1.05) = $14,700

Inclusion of top-level management is considered a necessary cost component since containment tests are considered critical activities that command their attention. According to SEA (1985), the average industry labor rate for top-level management is $67/hr; adjusting this for inflation yields:

$67/hr x 1.05 = $70/hr

Their time is reported in SEA (1985) to be about 5% of the BWB, or:

$$\frac{\$14{,}700 \text{ BWB} \times 0.05}{\$70/\text{hr}} = 10 \text{ hrs}$$

Summing, \$14,700 + (10 hrs @ \$70/hr) yields an estimated total labor cost for Type B and C leak testing of about \$15,400, or an average aggregate cost of about \$41/hr.

B.1.2 Supporting Information for Section 2.4.5.2 - Industry Operation

When the ILRT is on the critical path of an outage each of the following five sequential steps of the test itself lies on the critical path:

- Pressurization
- Stabilization
- Integrated Leak Rate Measurement
- Verification Test
- Depressurization.

Correspondingly, the time it takes to perform each of the five steps can impact costs. The potential effect on each of the five steps resulting from the postulated increase in the allowable containment leakage rate limits is discussed in this section. In addition, a discussion of the overall quantification of impacts due to the postulated accompanying reduction in failure rate of Type A tests is presented.

B.1.2.1 Pressurization, Stabilization and Leak Rate Measurement

The optimal pressurization rate and corresponding air flow rate are a function of ILRT pressure, the containment free air volume, the replacement power cost (assuming that the ILRT is a critical path operation), and the cost of the pressurization system. In the equation (PV=nRT) that determines the gaseous mass within the containment, the total pressure P is equal to the sum of the following two components: partial pressure from air (which is affected by temperature) and partial pressure from water vapor. Since water molecules could be added to (by evaporation) or depleted from (by condensation) the gaseous mass within the containment, such change in gaseous mass should be corrected before the actual leakage rate can be accurately determined. These components must be stabilized before the leak rate measurement test and be stable during the test. Appendix J to 10 CFR Part 50 requires a 24-hr leakage test of the containment following a 4-hr stabilization period of the containment atmosphere after pressurization.

Leakage rate from a containment is dependent upon the pressure differential between inside and outside the containment. For any leakage pathway with a fixed-size hole, the leakage rate will increase as the pressure differential increases. It should be noted that no changes in ILRT pressure requirements are postulated in this analysis. In addition, it should be recognized that for other types of leakage pathways, such as certain penetrations and valves, the leakage rate may decrease as the pressure differential increases because a seal in the penetration or a seat in the valve could close tighter under the higher pressure differential. In the postulated case where more leakage is acceptable, the push to repair and maintain valves/components might lessen, (with a corresponding lessening in costs). The overall effect appears to be a reduction in the likelihood that plants might fail their leak rate measurement test.

B.1.2.2 Verification Test

According to information contained in SEA (1985), the verification test should be by means of a mass step change (MSC) of approximately the daily allowable leakage. The MSC method of verification test is performed by removing from or pumping into (pumpback) the containment a metered mass of air during a short time interval. This metered mass of air should be approximately the amount which would leak from the containment in one day if the containment were leaking at the maximum allowable rate of L_a when pressurized to P_a. SEA (1985) recommends a short time interval of one hour for this step, which provides a sufficiently slow rate of removal or injection to allow the atmosphere to stabilize. It is possible that the MSC could involve a larger metered mass of air being involved due to the postulated increase in allowable leakage; therefore, a time interval of greater than the above suggested one hour could be anticipated in this step.

B.1.2.3 Depressurization

Frank et al. (1982) suggests that the depressurization system design should allow depressurization to proceed at an average rate of 10 psi/hour to avoid most of the following potential problems associated with more rapid containment depressurization:

- excessive noise
- damage to purge system ducting
- damage to purge system valves
- damage to purge system fans by windmilling
- blistering of containment paint
- separation of containment liner from containment wall
- damage to metal cans around thermal insulation
- excessive rate of release of radioactivity to the environment
- damage to instrumentation and electronics equipment inside containment.

It should be recognized that the installation of larger diameter pipes (a major and costly design change) could be required at some nuclear plants in order to meet the aforementioned suggested depressurization rate. In addition, Frank et al. (1982) points out that "if, for some unexpected reason, the radioactivity level of a PWR containment atmosphere is too high to permit an unfiltered release to the environment, then rapid depressurization may not be possible because of pressure limits in the ductwork connecting the containment penetration with the purge exhaust filters." Such changes as the aforementioned would have to be determined on a reactor-specific basis; therefore, they are not estimated in this analysis.

B.1.2.4 Quantification of Impacts Due to the Postulated Reduction in Failure Rate for Type A Tests

Forty to fifty percent of Type A tests result in failure according to SEA (1985). Improvement in this failure rate should result in decreased costs and radiation exposure. The overall impact resulting from this preliminary analysis is that costs could potentially be reduced by this postulated change in leak rate limit because it appears to reduce the likelihood that plants will fail their Type A ILRT.The present Appendix J requirement specifies that leakage rate not exceed 0.75 L_a. It is assumed that plants would have a greater chance of passing a Type A test with the postulated increase in the L_a limit, since the 0.75 L_a limit would rise a proportional amount. In turn, the attendant incremental downtime associated with retesting could be avoided, resulting in a cost savings. The largest anticipated cost savings associated with this change would result from the postulated reduction in the number of Type A retests and their associated plant downtime costs.

To quantify the impact of the postulated increase in leak rate limit, it was necessary to estimate the potential reduction in Type A test failures. As mentioned previously, the likelihood of Type A test failures is currently estimated in the range of 0.4 to 0.5. An average value of 0.45 is assumed for the reference case in this analysis. Although the extent of the potential reduction in test failures cannot be quantified with a high degree of confidence, it is assumed (for purposes of illustration of potential cost savings) that an improvement in the range of from 50 to 90% occurs as a result of the postulated change. In turn, this results in a new range of average values for the likelihood of Type A test failures from about 5% to about 23%.

A clear definition of causes of "failure" is not provided in SEA (1985); therefore, for purposes of this analysis, it is interpreted to mean test events resulting in either 1) those plants where the entire ILRT had to be repeated in order to pass the test or 2) those plants whose "failure" was corrected by the identification and correction of a leak pathway during the performance of the test itself. It is postulated that the former event is not considered to be as likely as the latter event. Therefore, about 10% of the plants postulated to require a retest are assumed to require a complete retest, while about 90% of the plants are assumed to require about 3 hours (an arbitrary time value selected for estimating potential costs) for identification and correction of a leak pathway for successful completion of the

Type A test. The uncertainty surrounding these assumptions could probably be clarified by detailed examination of plant-specific ILRT records.

Considering the approximate nature of the aforementioned assumptions, the industry-wide cost savings is calculated as follows. First, drawing on the base-line information provided in SEA (1985), two plant categories are considered: those 90 units currently in operation, which are assumed to have an average remaining lifetime of 26 years; and 30 new plants, which are assumed to have a plant lifetime of 30 years. Complete retests are estimated to follow closely on the heels of the Type A test failures so that remobilization of testing resources is not a factor in this analysis. Therefore, only the 67 hours associated with pressurization, stabilization, measurement, verification, and depressurization (as presented previously in Table 2.19 of the report) are needed to successfully complete the test when required. The cost per retest is the product of hours times the average cost of about $20,400 per hour (previously estimated for the reference case Type A ILRT) or about $1,367,000 per retest. Similarly, the cost for retests of 3 hours duration results in a cost of about $61,000 per retest.

The total industry savings (in constant 1985 dollars) is estimated to be in the range of about $42 million to about $76 million. This range of costs is based on the difference in estimated costs between the current average failure rate (0.45) and the lower range of failure rates (0.05 to 0.23) which are assumed to occur due to the postulated increase in the allowable containment leakage rate limits. The supporting calculations follow.

Using Equation B.1, the estimated total retest cost for the reference case is calculated to be $83.9 million. The equation utilizes the current average failure rate of 45% and further assumes that only 10% of the plants that fail will require complete retesting as discussed previously.

$$C_t = F_t \ (P_o \ R_o + P_p \ R_p) \ (F_r \ T_r \ C_c) + F_t \ (P_o \ R_o + P_p \ R_p) \ (F_r \ P_r \ C_p) \qquad \text{(B.1)}$$

where C_t = the estimated total retest costs

F_t = 1 test/3.33 years, the current ILRT test frequency

P_o = 90 plants currently operating

P_p = 30 planned plants

R_o = 26 years, the remaining lifetime of each operating plant

R_p = 30 years, the remaining lifetime of each planned plant

F_r = 45%, the current failure rate

T_r = 10%, the percentage of F_r assumed to require a complete retest

P_r = 90%, the percentage of F_r assumed to require a partial (3-hr) retest

C_c = \$1.37E+06, estimated cost of a complete retest

C_p = \$6.1E+04, estimated cost of partial (3-hr) retest.

As shown in Equations B.2 and B.3 below, the product of Equation B.1 times the assumed range of 50% and 90% improvements in the average failure rate previously discussed yields a total cost savings in the range of \$41.9 million to about \$75.5 million over all remaining reactor lifetimes.

$$(\$83.9\text{E+06})\ (0.50) = \$41.9\text{E+06} \tag{B.2}$$

$$(\$83.9\text{E+06})\ (0.90) = \$75.5\text{E+06} \tag{B.3}$$

B.1.3 Supporting Information for Section 2.4.6.2 - NRC Operation

The existing Appendix J of 10 CFR Part 50 specifies the NRC must review and approve the testing schedules subsequent to the failure of a Type A test. This requirement is not anticipated to change with the postulated increase in leakage rate limits. However, as previously mentioned, it is anticipated that the number of Type A test failures will be reduced due to the postulated increase in leakage rate limits. Therefore, the number of testing schedules submitted by licensees to the NRC for review also will be reduced, resulting in a reduction of NRC resources in this area. The total cost savings over all remaining reactor lifetimes is estimated to be in the range of \$7,200 to about \$13,000. The supporting calculations follow.

Using Equation B.4 (i.e., the Reference Case), the estimated total number of tests for all plants under current regulatory conditions is calculated to be 972.

$$A_t = F_t\ (P_o\ R_o + P_p\ R_p) \tag{B.4}$$

where A_t = the total number of Type A tests remaining

F_t = 1 test/3.33 years, the current ILRT test frequency

P_o = 90 plants currently operating

P_p = 30 planned plants

R_o = 26 years, the remaining lifetime of each operating plant

R_p = 30 years, the remaining lifetime of each planned plant

As shown in Equation B.5, the product of the average failure rate, 0.45, times 10% of the plants assumed to require a complete retest (as previously discussed in Section B.1.2.4) times the result calculated in Equation B.4 yields the estimated current number of testing schedules that will probably require review by the NRC.

$$(0.45 \times 0.10)\ (972) = 44 \qquad (B.5)$$

Assuming 8 hours of NRC time for review of each of the estimated 44 testing schedules calculated in Equation B.5 times a cost of $41/hour (as previously shown in Section 2.4.6.1 of the main report) yields an estimated cost, under current regulatory conditions, of about $14,400, as shown in Equation B.6.

$$(8\ \text{hrs} \times \$41/\text{hr})\ (44\ \text{reviews}) = \$14,400 \qquad (B.6)$$

As shown in Equations B.7 and B.8 which follow, the product of Equation B.3 times the assumed range of 50% to 90% improvements in the average failure rate previously discussed yields a total cost savings in the range of $7,200 to about $13,000 over all remaining reactor lifetimes.

$$(\$14,400)\ (0.50) = \$7,200 \qquad (B.7)$$

$$(\$14,400)\ (0.90) = \$12,960 \qquad (B.8)$$

B.2 SUPPORTING INFORMATION OBTAINED FROM SELECTED STUDY CONTACTS

The methodology used in this study for the collection of data was based on 1) acquiring background material found in the literature, including existing NRC reports and 2) telephone contacts with persons familiar with conducting Type A, B, and C leakage tests at commercial light water reactors. The latter contacts included personnel at two companies providing testing services to nuclear utilities and staff at three selected nuclear utilities. The results of those telephone conversations follow.

B.2.1 Testing Services Contacts

Two testing services contacts were made. Both firms have conducted more than 150 ILRTs on both domestic and foreign LWRs over the past four years. The majority of those ILRTs were conducted on PWRs although ILRTs on all three types of BWR containments - Mark I, II, and III - have been conducted by these companies. One company is considered a primary contractor for ILRTs and uses their own computer(s) to do the leak rate calculations. The other company provides subcontractor services to and in support of them, including set-up and data collection.

Essentially, both company representatives indicated that they gear-up to meet the customer's demands according to current requirements. They develop concise, reactor-specific contractual packages for the conduct of the ILRTs. They indicated further that they would not anticipate significant, if any, changes in costs to occur in those contractual packages as a result of the change in leakage rate limit postulated in this analysis. In the opinion of these sources, the reason for this is that the postulated change in leakage rate limit would probably not change the basic instrumentation or the techniques utilized to conduct the integrated leak rate measurement. Staying within the instrument accuracy limits imposed by the regulations and specifications while maintaining traceability to the National Bureau of Standards involves a large amount of work and this probably would not change much, if at all.

B.2.2 Selected Nuclear Utility Contacts

Three selected nuclear utility contacts were made by telephone. Originally, the three reactors were selected based on 1) their genre (two PWRs and one BWR, roughly representing the current ratio of the total population), 2) all of them had been in commercial operation for at least 10 years; thus, they should have been at that point in time for their 10 year inspections, including their ILRTs, and 3) their locations -- one plant is located on the east coast, one in the midwest, and one in the south. However, the utility located in the south provided information on two reactors on their site (see Section B.2.2.3 for details) for a total of two reactors in the south.

B.2.2.1 Eastern Utility

The eastern utility contact indicated that time, cost, and occupational exposure breakdowns associated with their leak testing schedule would be extremely difficult to reconstruct. This is especially true with their Type B and C testing. The principal reason for this was that many of the valves and components on their regular maintenance schedule are the same Type B and C tests valves and components upon which repairs and testing often occur coincidentally. Therefore, they are closely interrelated activities. In response to a question on whether or not all their Type A tests are considered to be on the critical path, the contact indicated that this is usually the case. They utilize only one external contractor for Type A, B, and C leak testing activities and that is primarily for preparing the final test report.

To ensure availability of air compressors when they are needed for the Type A test, this utility rents the air compressor units for the entire month during which the leakage test is scheduled (even though the compressors only may be needed for less than a week). This rental contract costs them about $350,000 or about their equivalent cost of one day's replacement power cost.

Since oil-free, dry air is an essential requirement for them, they purchased a dryer unit to remove oil and water vapor for about $15,000. This unit is skid-mounted and portable. It is utilized by the utility at other nuclear power plants within their system on an as-needed basis. The contact also indicated they were essentially at their optimum rates for pressurization and depressurization. In the pressurization mode this was based primarily on consideration for temperature stabilization. In the depressurization mode, it was based on meeting the controlled release limits for airborne activity and on pipe size constraints. In addition, for one of their PWRs, the contact confirmed that their most recent (1985) Type A test times for pressurization, stabilization, measurement, verification, and depressurization were similar to those reported by Frank et al. (1982) for a Type A test conducted at this same plant in May, 1980.

B.2.2.2 Midwestern Utility

The midwestern utility's reactor is an 800 MWe BWR that maintains a 12-month refueling cycle. Licensing staff at headquarters and station engineering staff at the plant were contacted. Similar to the eastern utility's staff, they could offer relatively few details concerning their Type A, B, and C testing manpower requirements, cost, schedules, occupational dose, etc. without time-consuming and detailed examination of their actual test records. A summary of the information which they did provide follows:

- Type A tests are almost always on the critical path. One test was done about two years ago and the most recent Type A test was done about two weeks ago. The latter test was unscheduled (putting it on the critical path) and resulted from changes made to a containment vessel penetration.

- Their ILRTs are conducted at their optimum rates for pressurization (3 psig/hour) and depressurization (slightly less than 3 psig/hour due to physical constraints).

- They rent only one backup air compressor at a cost of $100 per day unused or $350 per day when they actually press it into service.

- They use a specialty contractor for data taking, test evaluation, and troubleshooting. The latest charge for these services was about $60,000.

The approximate time frame for their 1985 ILRT is presented in Table B.1. In addition, preparations were reported to take two days and post-test restoration activities took about three days. It is not exactly known what percentage of time associated with these activities is considered to be on the critical path.

B.2.2.3 Southern Utility

The southern utility that was contacted has two PWRs on the same site. Both units are in the 850 MWe range and are on 18-month refueling cycles. Licensing staff at headquarters and station engineering staff at the site were contacted concerning testing activities associated with both nuclear power plants. The last ILRT at Reactor A was conducted in December 1984, while the last ILRT at Reactor B was conducted in April 1985. Little or no detail was available on Type B and C testing at either unit simply because it was considered very difficult to separate hours for maintenance from those hours specifically used for testing without detailed examination of plant records. However, the site contact indicated that for the most part

TABLE B.1. Approximate 1985 ILRT Schedule Reported for a Selected Mid-western BWR

ILRT Activity	Time, hours
Pressurization	12
Stabilization	4
Integrated Leak Rate Measurement	8
Verification Test	4
Depressurization	9
Total Hours	37

the Type B and C testing was not on the critical path. The site contact had been involved directly with two ILRTs and indirectly with two others while at the site. A summary of the information that was provided by this contact follows:

- Long-term test preparations (e.g., development of test procedures, checking prints for as-built and up-to-date, etc.) require about two man-months.
- Occupational dose would not be directly retrievable for the ILRTs since valve alignment activities by operators are not recorded in a manner directly relatable to specific ILRT valving as such.
- They rent six air compressors (8,000 scfm capacity) and two refrigerated air dryers for about $45,000 per month, together with one equipment operator. About 3 to 4 days is allocated for service connections setup time. The refrigerated units keep the humidity down thus allowing for faster temperature stabilization.
- A specialty contractor writes their summary technical report at a reported cost of $42,230 (1985 dollars).
- They conduct their ILRTs at optimum rates for pressurization and depressurization.
- After first resetting the blades, they keep their reactor building coolers on during the ILRT for purposes of temperature stabilization.
- To date, all ILRTs have been on the critical path.

The approximate time frames for the latest ILRTs at their nuclear power stations are presented in Table B.2. Full-pressure tests are always conducted on both units. The reason for the shorter time period shown for the measurement at Reactor A in the table is because they use the Bechtel topical report (BN-TOP-1) criteria to successfully run the test in less than 24 hours.

TABLE B.2. Approximate ILRT Schedules Reported for Reactor A (1984) and Reactor B (1985) by the Southern Utility Contact

ILRT Activity	Time, hours Reactor A	Time, hours Reactor B
Pressurization	17	17
Stabilization	5	5
Integrated Leak Rate Measurement	8	24
Verification Test	5	5
Depressurization	20	20
Total Hours	55	71

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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 reactor containment leakage rate, main steam isolation valve leakage control systems in boiling water reactors (BWRs), and NRC fuel system safety reviews. 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 streamlined regulations were evaluated in terms of such factors as population dose, individual dose, prompt fatalities and injuries, and costs to industry and NRC. The results indicate that streamlining the regulatory requirements in all three areas would have little impact on public risk. Substantial savings in operating costs may be realized in the areas of containment leakage rates and leakage control systems for BWR main steam isolation valves. The cost analysis indicates that only marginal benefits may be gained by streamlining NRC's safety review of fuel system designs.

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