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REGULATORY COST-RISK STUDY

April 1983

Prepared for the Argonne National Laboratory

Under Contract No. 31-109-38-6751

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Section 1

INTRODUCTION AND SUMMARY

INTRODUCTION

The practice of nuclear power regulation in the U.S., particularly over the last decade, has resulted in an exponential increase in the number of regulations. Historically, many of the regulations were put in place at a time when insufficient or inadequate data were available to provide a sound basis for such regulations and criteria. The NRC, therefore, in developing the criteria and calling for design approaches, working from such data as they had, as a general principle, interpreted such data very conservatively and in the absence of "absolute proof" tended to set criteria on a worst case basis. This is generally the approach adopted by public health organizations which is consistent with NRC's responsibility for regulation of radiation exposure. However, it is not necessarily the approach that should be taken relative to equipment and system performance, since there is always an opportunity to correct deficiencies when they materialize. Even though convinced that such criteria were excessive in many cases and that a good case could be made for relaxation, utilities were influenced by the enormous economic impacts of delays to go along and accept the criteria, rather than holding the job up for a drawn-out process of experimentation and technical debate.

Considering individually, each of these regulations was put in place by NRC because they were means to improve the safety of some feature of a nuclear plant. However, the cumulative effect of this massive body of regulatory output has been to contribute importantly to the five- or six-fold increase in the average cost of a nuclear plant which has occurred in the last decade. Clearly, there have been other factors which have also been operative, procedural (e.g., the impact of NEPA), and technical (e.g., the general deterioration in labor productivity), which together with the additional physical content of the plant resulting from added regulatory requirements (References 1 through 4), have substantially extended the schedule for design, licensing and construction of nuclear power plants, and at a time when both escalation and money costs were rising sharply. Reform of the procedural processes by which regulatory approvals are obtained in the U.S. is an area to which the current Administration is committed and in which it is planning to introduce appropriate

legislation. It is clearly a critical aspect of the overall problem, but beyond the scope of this assignment.

Another critical aspect is the impact of this massive body of added regulations, which have made the plants much more complex and the operations much more formalized, and has had a deleterious influence on plant performance (i.e., capacity factor), thereby increasing environmental and economic risk to the general public as fossil plants are called upon to replace the lost nuclear power. This aspect is also beyond the scope of this present assignment, but certainly deserves consideration.

Specifically, this analysis deals with the impact of regulatory requirements on plant design aspects as they have affected capital costs. In recent years a substantial amount of new and better data have been developed in many areas. Also, the methods of analysis, particularly the use of Probabilistic Risk Assessment (PRA) has become much more sophisticated and much more developed. This, it is believed that there now exists sufficient background, experience and data to establish a more rational regulatory basis that balances the cost of regulation against the benefits (i.e., reduction in risk) gained. Through the use of PRA methods it is reasonable to design all systems to about the same level of cost-benefit ratio which by definition would result in an optimum use of resources. It is generally perceived that such an optimal situation does not currently exist in the regulation of nuclear power in the U.S. and a quantitative demonstration of this perception is one of the goals of this study.

More importantly, this would seem to be a most appropriate time to establish a more orderly regulatory system. There has been for some years now a hiatus on filing of new construction permit applications. That hiatus is likely to continue for some years ahead. Thus, the regulatory staff should be able to devote some substantial attention to this general problem, and take the opportunity to develop a more systematic approach to regulatory criteria, which would rely more heavily on quantitative expressions of the relationship between the frequency of a postulated event, the consequences of the event, and the costs to reduce the frequency or to mitigate the consequences. In that context, this study is intended to provide some quantitative perspective by selecting certain examples of criteria for which estimates of risks and costs can be obtained, and the balance of the various risks, (i.e., internal versus external risks), can be put into perspective.

Obviously, the existing body of regulations has already been incorporated into the existing plant universe, and is largely irreversible depending on the construction status. Therefore, the main thrust of efforts to revise the regulations would be aimed at making simpler, more cost effective, and better "standard" designs for new plants. It would, as well, provide some basis for determining the requirement for backfits to existing plants, including application to plants not yet completed.

Within the time and budget constraints of this assignment, the analyses performed were based on existing data in the literature, and in fact, the assignment started from a survey of the literature seeking frequency/consequence relationships that had been developed for the examples selected. The examples were chosen for this first phase study from a list of candidates (see Table 1-1), to illustrate different aspects of the overall problem, primarily by examination of equivalent features of older plants. For example, the seismic design of Zion and Indian Point, and the use of ANSI B.31.1 predate current regulations which are far more costly to utilize; however, the Zion and Indian Point designs resulted in plants where risks are not appreciably greater than for plants designed to the current regulations. Another example demonstrates how certain regulations as they now exist (those relating to tornado protection) can be shown to lead to much harder structures than actually required as a result of research and development work carried out in recent years. Another example (protecting containment from a steam line break) illustrates the proliferation of cost which can be involved in implementing a generally desirable criterion in a most complex physical plant.

It is believed that the assignment as reported herein does, in fact, serve the several purposes intended, and that the examples selected do illustrate the potential value of a revised technical approach to regulation. Admittedly the examples were just a few of the many that could have been examined and the results, particularly the cost estimates, are not as definitive as would have been desired. Such matters were intended to be explored in a follow-on effort which would build upon the work reported herein and extend it in the following important ways:

1. To define an appropriate framework within which regulatory decisions can be made using the techniques of Probabilistic Risk Assessments.
2. To develop a pre-conceptual reference design, or elements of such a design, which could improve capital cost for a "standard" nuclear power plant by limiting regulatory requirements to those having a justifiable cost/benefits ratio.

Table 1-1

EXAMPLES FOR COST-RISK EVALUATION

Primary Examples:

1. Tornado-proofing of the spent-fuel building superstructure and LOCA systems is not required.
2. Walls surrounding safe-shutdown systems are thicker than needed to resist credible tornado-generated missiles.
3. Physical separation of redundant trains is not necessary for most systems.
4. Widespread use of ASME III criteria in safety systems is not justified.

Secondary Examples:

1. Many Quality Assurance requirements, especially those involving traceability and documentation, do not materially improve component reliability.
2. Many electrical and control requirements, especially when applied to systems such as reactor protection, do not materially improve the reliability of the system.
3. Radwaste systems are not designed based on the assumption that 1% of the fuel in the reactor will fail, while average failed fuel fractions actually experienced are now nearly two orders of magnitude lower.
4. Sophisticated seismic analyses do not materially improve the designs.
5. Excessive requirements for seismic supports and seismic qualification of components do not significantly improve plant safety.

Table 1-1, (Cont.)

EXAMPLES FOR COST-RISK EVALUATION

6. Ventilation and filtration systems on fuel handling buildings only reduce the small releases that might result from a fuel handling accident and therefore do not significantly affect overall plant risk.
7. Because of the low levels of activity in radwaste and boron recovery systems, requirements for quality assurance and seismic and tornado protection for these systems do not significantly reduce overall plant risk.
8. While design for hot shutdown from outside the control room is obviously desirable, design for cold shutdown from outside is not necessary.
9. For systems other than the reactor coolant system, requirements for removable insulation for in-service inspection do not improve plant safety, since safety system reliability is determined by the active components.
10. Protection of safety systems from the effects of accidents that do not require those systems for their prevention or mitigation (such as protection of containment from steam line breaks) do not improve plant safety.

3. Develop recommendations for ongoing R&D programs to improve the data base for cost/benefit evaluations of various nuclear power risks.
4. it is also believed that important insights into the impact of regulations could be gained by an essentially parallel study which would attempt to quantify the impact of excessive regulation on operating plants. This would include the economic and environmental effects of plant shutdowns or deratings mandated to re-examine or change design features; to comply with technical specification requirements when certain systems are not available; and to explain unexpected events even if they appear to have no serious safety consequences. It would also include the growth in plant staff to meet regulatory requirements.

SUMMARY

The principal purpose of this study was to perform cost-benefit evaluations on a few NRC requirements to illustrate that current requirements result in inconsistent allocation of resources within a nuclear plant and often result in costs per unit of risk reduction which greatly exceed those typical of other industries and activities. The evaluations were primarily based on the information available in EPRI reports, Journals, NUREGs, WASH 1400, and in published Probabilistic Risk Assessments. The following NRC regulations and requirements were investigated in some detail and are described individually in Sections 4 through 11. The areas covered are:

- o Tornado missile protection requirements in general
- o Tornado protection for the spent fuel pool
- o ASME Section III piping versus ANSI B31.1
- o Separation of safety systems trains
- o Seismic requirements
- o Protection of safety systems from effects of accidents which do not require the system to function
- o Regulatory Guide 1.26 impact on Emergency Cooling Injection System pumps.
- o Fuel handling building filtration systems

Several other NRC requirements were reviewed, but were not evaluated in any detail. These are briefly discussed in Section 12.

In order to focus the analysis work, it was decided to concentrate on one reactor type. The PWR was chosen because it is more common and because the most comprehensive recent PRAs (those for Zion and Indian Point) are concerned with PWRs.

The results of the detailed evaluations are summarized in Table 1-2. In all cases, the results are conservative in that the benefits of the regulations are overstated and the costs are understated. Nonetheless, the analyses show that none of the selected NRC requirements are cost-effective. (In some cases, it was not possible within the scope of this study to determine the costs of a regulation. In each of these cases, the costs are clearly greater than the benefits.)

Based on the work performed in this study, the following general conclusions may be drawn:

- o With the exception of a few very specific situations, the risk from PWRs built in the late 1960's is very low (the value of this risk is on the order of two million dollars per plant, or less), so it is clear that most NRC requirements implemented over the past 15 years (which have increased costs far beyond \$2 million) aimed at reducing that risk could not have been cost-effective even if they substantially reduced that risk.
- o The cost-effectiveness of various NRC requirements varies by many orders of magnitude. If it is desired to further reduce nuclear risks below the already low level, ways to do this which are more cost-effective than many current NRC requirements can almost certainly be found.
- o With relatively minor modifications, the designs used in the late 1960's would result in plants which would have very low levels of risk and would meet NRC's proposed safety goals.

Table 1-2

COST-BENEFIT EVALUATION SUMMARY

NRC Requirement	Cost of Implementation (Direct Cost in 1981 Dollars)	Benefit of Implementation (Dollar Value of Reduced Risk)
Concrete Wall Thickness Greater Than 10 inches for Tornado Protection	(1)	\$100
Tornado Missile Protection of Spent Fuel Building	56,000 to \$200,000	\$16,000
Use of ASME Section III for Class 2 and 3 Piping	\$23,000,000	\$17
Physical Separation of Redundant Safety System Trains	(1)	\$1,700
Seismic Design Requirements	(1)	(2)
Protection of Containment from Steam Line Breaks	(1)	\$2,000 to \$60,000
Use of ASME Section III, Class 2, Requirements for ECIS Pumps	(1)	\$35
Fuel Handling Building Filtration System	\$500,000	\$60

NOTES:

- (1) Costs could not be determined within scope of this study due to complex interaction of many factors affecting overall design.
- (2) Benefits are highly specific to plant design. After obvious problems are corrected, benefits from further improvements are on the order of \$10,000 to \$100,000.

Section 2

BACKGROUND

INCREASES IN REGULATORY REQUIREMENTS

There have been many items in the literature which illustrated how regulations have increased over the years. In Reference 5, for example, O'Donnell compared the number of safety systems in plants starting operation in the 1960 to 1978 time period. He showed that (including systems backfitted into older plants) the number of safety systems increased from 17 in plants starting up in 1960 to 50 for plants starting in 1978. Another measure of regulatory growth is in the number of regulations, laws, and other requirements added. Figure 2-1, taken from Reference 6, again graphically illustrates this point. Similar comparisons have been made for increases in building quantities and required documentation (References 7 and 8). Reference 8 shows increases of a factor of 2.6, 1.6, and 1.8 in quantities of concrete, pipe, and wire respectively; all on a per MW basis, over the period from the early 1960's to the late 1970's, the implication being that increased regulations and requirements are the major cause. These increases in quantities caused proportionally increased direct costs and increased construction time. Implementation of NRC procedural (QA) requirements has dramatically increased costs by reducing construction labor productivity, increasing construction times, and increasing costs of manufactured components. Requirements for more detailed analyses have significantly increased engineering costs.

It would seem obvious that, well intentioned or not, it is virtually impossible to introduce the number of regulations that have been put in effect and have the results be consistently cost-effective. This is especially true because, in the past, regulations have been instituted on an individual basis, each concerned with a specific problem or area, with little or no consideration as to overall effects.

It must be recognized that initially there was an inadequate data base available for use in formulating regulations and little experience with methods to measure the benefit of those regulations. The Reactor Safety Study (RSS), WASH 1400 (Reference 9), opened the way for new techniques in assessing the safety of nuclear power plants. Although the RSS had shortcomings and was subject to criticism, its methods are recognized as showing great promise in pointing out the strong and weak points in the

Figure 2- 1

plant design by allowing the various systems and their failure modes to be put into the proper perspective in the context of the overall plant picture. Also these last 15 years have seen dramatic increases in the amount of operating experience and experimental data available to guide the formulation of regulations.

The traditional deterministic methods of specifying the safety design of the plant was through the application of the General Design Criteria, Regulatory Guides, Standard Review Plan, Branch Position Statements, etc. Over the years these requirements and guidelines advanced from rather general statements to more specific requirements, to the expansion and additions of more and more specific requirements and regulations as illustrated in Figure 2-1. Review efforts at that time concentrated on the infrequent design basis accident. New requirements were developed because they were perceived to increase the safety of the plant. They were, however, promulgated without a formal assessment of whether an overall improvement in safety was achieved or whether the degree of improvement justified the cost involved.

The importance of risk assessment in plant evaluations has been recognized by many, and over the years since the RSS, use of risk assessment has increased. Extensive risk assessment studies such as for Zion, Indian Point, Limerick, and Big Rock Point have been commissioned by utilities for their plants, and other studies are underway. The NRC carried out the Reactor Safety Study Methodology Applications Program (RSSMAP) and the Interim Reliability Evaluation Program (IREP). The NRC is becoming more committed to the use of risk assessment/value analysis techniques, particularly since the TMI incident. Reference 10, published in May, 1981, presents the proceedings of a seminar for NRC staff which was held in January, 1980, concerning the future role of risk assessment and reliability engineering in the regulatory process. In the presentation and discussions, the failure of NRC to adopt the techniques prior to TMI was admitted and the intent to incorporate it in the future was expressed. It was stated that the expected usage would be to supplement the Standard Review Plan and traditional procedures rather than replace them.

Further evidence of the NRC's intent to change the way they do business is found in the recent establishment of the Committee to Review Generic Requirements, whose function is to see that all new requirements are necessary and cost effective. How effective this group will be remains to be seen.

All indications point to the intent to use improved techniques, including PRA, in decisions related to new regulations. However, there does not appear to be any systematic effort in place to look at the body of existing regulations to determine whether indeed they are required to produce a "safe" plant or whether they simply add cost with no commensurate reduction in risk.

RISK ASSESSMENT/LITERATURE SURVEY SUMMARY

Before discussing the evaluations done as part of this study, it is appropriate to relate some of the information obtained in the course of a literature search and some overall insights gained from existing risk studies.

Risk Assessments

At the present time, full-scale PRAs have been published for seven PWR designs. The following table shows the number of PWRs which received construction permits in each year during the 1960's and the plant designs for which PRAs are available.

<u>Year</u>	<u>Number of PWR Construction Permits</u>	<u>Plant Designs for Which PRAs Have Been Published</u>
1964	2	
1965	-	
1966	2	Indian Point 2 (Ref 15)
1967	8	Oconee (Reference 12)
1968	17	Zion 1 (Ref 14), Surry (WASH 1400), Crystal River 3 (Ref 13)
1969	6	Indian Point 3 (Ref 15)
1970	6	Sequoyah (Reference 11)

The two 1964 plants, San Onofre 1 and Haddam Neck, are transitional designs while the plants from 1966 onward are those which might be considered "modern" designs. It can be seen from the table that currently available PRAs cover the spectrum of the "early," "modern" PWR design.

WASH 1400 was the first full-scale nuclear plant PRA. The PWR portion of WASH 1400 was based on the Surry design, although accident consequence calculations were based on a composite of a number of nuclear plant sites. The Crystal River 3, Oconee, and Sequoyah PRAs are based on the WASH 1400 methodology (with some

modifications). Probabilities of radioactive releases in various categories are calculated, but off-site consequences are not. The PRAs for Indian Point 2, Indian Point 3, and Zion are not based on WASH 1400 and differ from it most importantly in the following ways:

- o Use of plant specific component failure rates
- o More sophisticated treatment of uncertainties
- o More comprehensive treatment of fires, tornadoes, and earthquakes
- o Use of more recent data on containment failure mechanisms and fission product transport
- o The higher population densities at Zion and Indian Point as compared to other sites

The table below shows the probability of core melt and the average fatalities per year (early plus latent), broken down into internally initiated events (transients, LOCAs) and externally initiated events (seismic, tornado, fire).

PWR RISK ASSESSMENT RESULTS

		CALCULATED PROBABILITY OF MELT (PER YEAR)	FATALITIES PER REACTOR YEAR				
			<u>All Events</u>	<u>Internal Events</u>	<u>Total</u>	<u>External Events</u> <u>Fire</u>	<u>Seismic</u>
Zion 1/2	4×10^{-5}	.02	.001	.02	<.001	.02	-
Indian Point. 2	4.7×10^{-4}	1.4	.007	1.4	.6	.6	.2
Indian Point. 3	2×10^{-4}	.28	.005	.28	.27	.01	.004
WASH 1400	6×10^{-5}	.02(.06)	.02(.06) *	-	-	-	-
Oconee	8×10^{-5}						
Crystal River 3	4×10^{-4}						
Sequoyah	6×10^{-5}						

* WASH 1400 presented results in terms of median probabilities whereas Zion and Indian Point reported mean values. Since mean values are considered to be more representative, the WASH 1400 median probabilities of release were converted to mean values assuming the data presented were lognormally distributed. Fatalities computed from these mean values are shown in parenthesis.

In WASH 1400, external events were evaluated and it was determined that they did not contribute significantly to core melt probability. Therefore, their contribution to risk was not specifically quantified. The Crystal River 3, Oconee, and Sequoyah PRAs did not evaluate external events nor did they calculate the consequences of core melt.

The table clearly shows that there is considerable variability in results among the PRAs. The major reasons for the differences are as follows.

- o In a number of more recent PRAs, the probability of melt is higher than in WASH 1400 due to more extensive and conservative analyses of common-cause failures and operator errors and (in some cases) use of more realistic equipment failure rates.
- o Plant-to-plant differences in core melt probabilities result partly from methodological differences and partly from actual differences in system designs and equipment failure histories.
- o In spite of a higher population density than WASH 1400, the Zion PRA shows a calculated risk comparable to WASH 1400. This results from the use at Zion of more recent data showing that the probability of containment failure following core melt is much lower than assumed in WASH 1400.
- o A few specific design features at Indian Point result in a small probability that wind, earthquake, or fire will result in failure of electrical power and/or control. Since such failures lead to core melt and containment failure and since the population density at Indian Point is the highest of any nuclear plant in the U.S., an atypically high overall risk is calculated.

Considering the typical PWR which received a construction permit in the late 1960's, it appears that the probability of core melt is between 5×10^{-4} and 5×10^{-5} , up to a factor of 8 higher than calculated in WASH 1400. Nonetheless, the WASH 1400 calculated risk appears conservative due to the excessively high probability of containment failure which was used. The calculated Zion risk also appears high because of the above average population density at Zion. Although it cannot now be proven, there is a growing consensus in the field that the fission product release fractions used in all PRAs to date are excessively high and that actual risk may be a factor of 10 or more lower than calculated. Accepting the Zion/WASH 1400 risk estimates as realistic or conservative results in a risk of .02 fatalities per unit-year or 0.8 over the life of one unit. In Section 3 of this report it is shown that a value of \$500,000 per life is higher than used in most regulatory actions and that property damage and other effects of a nuclear accident are generally valued at about three

times the value of lost life. Therefore, the value of the total risk for a late-1960's vintage nuclear unit is on the order of $(1 + 3)(500,000)(0.8) = \1.6 million.

Since the risk for these early plants is so small, it is not surprising that efforts made over the last 15 years to reduce that risk were largely not cost effective. In fact, NRC requirements between 1968 and 1982 have added far more than \$1.6 million to the cost of a new plant and have not eliminated risk. It follows that many (and probably most) of these requirements have not been cost effective.

General Literature Review

In reviewing the literature for information relative to performing value analyses on NRC regulations, several instances were found where other investigators had done relevant work of a general nature. References 16 and 17 covered examples which showed that arbitrary application of the single failure criterion, separation requirements, and IEEE requirements does not necessarily lead to a more reliable system and may result in a less than optimal design. In one instance it was concluded that, at least in the case of a system being considered to prevent overfeeding in a B&W once-through steam generator, there was essentially no difference in the reliability of the system whether or not it was designed to IEEE standards. Another example presented showed that for the particular fluid system under consideration, cross connecting of lines in parallel trains, with no valves in these lines, resulted in a more reliable system than if the parallel trains were completely separate. NRC's deterministic criteria related to single failure would not, however, allow cross connects without valves for isolation. This example is one where compliance with regulations produced a more expensive system (cost of isolation valves and controls, etc.) while reducing safety (albeit by a very small amount).

Reference 4 presents the results of a study on the cost-benefit of Engineered Safety Features. Using the PWR of the RSS, evaluations of the following were made: 1) emergency core-cooling system, 2) containment, including associated heat and fission product removal systems, and 3) emergency on-site alternating current system. In addition, the study considered the benefit associated with adding a hydrogen recombiner system to the RSS plant. In this case it was conservatively assumed that the addition of the hydrogen recombiner completely eliminated the risk of accident sequences that resulted in hydrogen-related containment failure. The results shown in the table below are taken from this paper. For this comparison, WASH 1400 accident

consequences were applied to a hypothetical site with a uniform population distribution of 400 people/mile². The following table shows the results, assuming 10,000 man-rem leads to one statistical fatality.

Engineered Safety Feature	Risk Reduction (Fatality/Yr)	Cost (\$/Yr)	Cost-Benefit Ratio (Millions of Dollars Per Fatality)
ECCS	11.0	1.5×10^6	.14
Containment	.7	3.0×10^6	4.4
Emergency Power System	1.8	2.0×10^6	1.1
Hydrogen Recombiner System	.00001	4.0×10^4	308

Using NRC criteria of \$10 million per life (\$1,000 per man-rem); ECCS, containment, and Emergency Power Systems are cost-effective but a hydrogen recombinder system is not. Using \$2 million per life which is more consistent with other industries (see Section 3) containment as well as the hydrogen recombinder is not cost effective. (Note that this is based on WASH 1400. Containment is now believed to be more effective than assumed in WASH 1400 and might be cost effective.)

It should also be recognized that EPRI has been involved with sponsoring risk assessment/value analysis-related projects since about 1974. EPRI has been working to develop the methodology of risk assessment, including consequence code development, and in the application of the techniques to specific examples such as evaluation of ATWS alternatives and sensitivity studies to determine the value, in terms of dose reduction, of various engineered safety features. A value impact analyses of post TMI alternatives is currently in progress. As an example of EPRI studies, Reference 18 presented analyses which showed that of the alternatives in NUREG 0460 (ATWS) which were: 1) no changes, 2) modifications to electrical systems, 3) alternative 2 plus partial mitigation, and 4) total mitigation; there was no justification to go beyond Alternative 2.

LESSONS LEARNED FROM PRAs TO DATE

As discussed in the preceding pages, PRAs to date show that the risks from PWRs designed in the late 1960's are generally quite low. While most NRC requirements

since 1968 are likely to be noncost-effective, some improvements beyond 1968 designs are clearly possible and may even be at least marginally cost-effective. Review of the major risk contributors at early PRWs provides insights into what these improvements might be.

In its proposed safety goals (Reference 19) NRC calls for a risk of core melt which is less than 10^{-4} unit per year. Most PRAs for earlier PWRs calculate comparable probabilities of core melt. Review of existing PRAs indicate that the risk of melt in early PWR designs could be reduced substantially further by one or more of the following:

1. Improve seismic capability of a few components or structures.
2. Add six to ten inches of concrete around equipment required for safe shutdown where this is not provided.
3. Provide separation and/or fire protection and/or redundant power to permit operation of auxiliary feedwater, reactor coolant pump seal cooling, and reactor makeup in the event of a fire.
4. Reduce the chance for operator error realigning the ECCS following a small LOCA.
5. Add redundancy and diversity to high pressure injection and its associated emergency power.

For a new plant NRC now requires Items 2, 3, and 4. Item 1 is not specifically required although it might be accomplished by the greatly increased conservatism in overall seismic design. Item 5 is currently not required. In the areas of seismic protection, tornado protection, and fire protection, current NRC requirements go far beyond what is necessary (as outlined briefly in Items 1,2, and 3) to limit risk. Because of its deterministic criteria NRC has many burdensome requirements which do not significantly reduce risk.

In incorporating an appropriate mix of items 1-4 into the designs of early plants (such as Surry, Zion, and Indian Point) would generally result in designs which could be built today for far less than the cost of currently licensable designs, and would meet NRC safety goals. If it were determined that further reductions in risk were desirable, these could be accomplished relatively inexpensively by increasing the ruggedness of key components or increasing their redundancy and diversity. A typical PWR high pressure injection can prevent core melt in all situations except ATWS and very large

LOCA and is therefore the most logical candidate for reliability improvement. It appears almost certain that plants can be designed which will be as safe or even safer than now required, at costs well below present-day averages.

Section 3

COST-BENEFIT METHODOLOGY

INTRODUCTION

To carry out a cost-benefit analysis of nuclear regulations, it is necessary to determine the benefit (change in risk) associated with a regulation and compare it to the impact of the regulation on the cost of the plant. In determining the effect of individual regulations on risk, use was made of the results of currently available PWR PRAs, in particular WASH 1400 and the Zion and Indian Point PRAs. Simply stated, the general approach taken in the PRAs is to determine through analysis of event trees and fault trees, the probability of the dominant accident sequences leading to core melt. Analysis of containment response to these accident sequences then determines the probability and timing of containment failure and the extent of fission product releases. Depending on the severity of release, each of the accident sequences is placed in one of a number of fission product release categories for which the consequences are calculated. Combination of the results of these steps leads to the final frequency/consequence relationships. The end results of these studies are presented in terms of frequency with which risk exceeds a given value.

CALCULATION OF BENEFIT

The benefit of a particular NRC requirement was calculated as the dollar value of the reduction in risk which would result from imposing the requirement on a typical earlier-design PWR. The steps involved in this calculation were:

- o Determine the reduction in the probability of core melt.
- o Determine the type of release of radioactivity which could be avoided.
- o Determine the average consequences per release.
- o Determine the monetary value of the consequences.
- o Obtain the product of (change in probability) times (consequences) times (monetary value) times (40 years) to obtain the total monetary value of the risk reduction over the assumed 40-year lifetime of a plant.

Core Melt Probability and Release Magnitude

Determination of the impact of a particular regulation on the probability of release generally required assessment at the system fault tree level. (The details of this step are discussed in the respective sections.) Changes in system or component failure rates or failure modes were made and reflected through the fault trees and event trees to ultimately give changes in the probability of experiencing a particular release category. In general, the NRC requirements considered in this report have a minor effect on the probability of core melt. PRAs focus on the major contributors to risk and usually do not explicitly quantify minor contributors. Therefore, it was often necessary to make a number of assumptions. Conservative assumptions were made throughout and this led to substantial conservatism in most of the risk estimates. For example, if a regulation were perceived to reduce a failure rate, but the amount of reduction could not be clearly determined, the failure was assumed to be completely eliminated. Similarly, if a requirement would reduce the probability of core melt but the fission product release from the melt was not known, the release was assumed to result in an average of 2000 fatalities per occurrence; roughly the average effect of the worst releases in the Zion PRA and equal to the consequences of the most serious WASH 1400 release.

WASH 1400 and the Zion/Indian Point PRAs are quite different in their treatment of uncertainties. In WASH 1400 the probabilities of core melt and various releases are calculated using the median values of the uncertainty distributions of failure rates, etc.; while mean values are used in Zion/Indian Point. Since the mean of a lognormal distribution is always larger than the median, the Zion/Indian Point procedure significantly increases the probabilities of core melt. In the work reported here, failure rates, etc., from both WASH 1400 and the Zion/Indian Point PRAs were used, recognizing that there are a number of inconsistencies. No attempt was made to quantify the error bands on any of the results.

In several instances, errors were discovered in the Zion and Indian Point PRAs and also in some of the references from the literature. If resolution of the error was not obvious, best judgment was used in resolving discrepancies. (It has also been learned that a corrected version of the Zion PRA is planned.)

Consequences of Releases

Given the probability of a core melt occurring and resulting in a release in a particular release category, it is necessary to calculate the consequences of that release in order to evaluate the effect on the public. That is, the effect (risk) is equal to the probability of occurrence of the release category times the consequences of the release category.

In the case of the Zion and Indian Point PRAs, the number of fatalities per release (for each category) was not presented directly, but could be derived through integration of frequency of exceedence tables presented. The procedure utilized and the results of the integration process are found in Appendix 1. In the case of WASH 1400, information on fatalities per release category was presented in the draft report but was not given in the final report. Changes in the computation of consequences between the draft and final report resulted in significantly different consequences, and although the overall change was known, the effect on individual release categories was not. It was not believed that application of the overall factor of change to each category was appropriate. However, Reference 4 contains a tabulation of population dose versus release category which was based on information presented in WASH 1400, adjusted for an assumed population density of 400 persons/square mile. For each release category, the population dose was divided by 4 (to reflect 100 persons/square mile), multiplied by the release category probability from WASH 1400 and divided by 10,000 (assuming 10,000 man-rem results in one statistical latent fatality) to give the latent fatalities per year due to releases in that category. Summing these results for each category resulted in a total of 2×10^{-2} latent fatalities per year due to all releases. Since this is identical to the value presented in the WASH 1400 final report (Table 5-6 of the main report adjusted for one plant and a 30-year period of incidence), it was concluded that the data in Reference 4 is a reasonable representation of the WASH 1400 results. These data are tabulated below. Early fatalities are ignored; however, the error involved is small since, for all release categories, they are a small fraction of the total.

<u>WASH 1400</u> <u>Release Category</u>	<u>Latent Fatalities</u>
1	2000
2	1800
3	1100
4	190
5	48
6	11
7	.3

Value of a Life

One of the recognized difficulties in attempting any sort of cost-benefit analysis associated with protection of the public, whether it be related to nuclear plants or some other area, is the necessity to place a dollar value on a statistical life. In their proposed safety goals for nuclear plants (Reference 19), the NRC has presented a cost-benefit guideline of \$1000 per man-rem averted for use in arriving at decisions related to safety improvements. This guideline is equivalent to \$10 million per life, based on the assumption of a 10,000 man-rem exposure leading to one statistical fatality. As pointed out in the discussion in Reference 19, this value is higher than values calculated for actual or proposed life-saving measures in regulatory actions in nonnuclear areas such as highway or automobile safety or air pollution. In these areas, the costs per life-saved ranged from zero to a few hundred million dollars with most values less than \$200,000 per life. Reference 4 reported, for example, that a value of \$140,000 per life has been used in decision making regarding highway safety programs and that the decision to put air bags in cars (over which there is concern over cost) would result in a cost of about \$300,000 per life.

Based on the preceding, it was determined that \$500,000 per life-saved is somewhat higher than used in most nonnuclear situations. However, reactor accidents can cause damage other than fatalities. WASH 1400 presents average risks of early fatalities; early illnesses, latent fatalities, thyroid nodules, genetic effects, and property damage. The probability of each of these effects (based on one year of operation of 100 reactors) is given in Table 5-6 of WASH 1400 (main report). Table 3-1 shows these

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Table 3-1

CONSEQUENCES OF 40 YEARS OF OPERATION OF ONE NUCLEAR PLANT

<u>Consequence</u>	Frequency ⁽¹⁾ (Yr ⁻¹)	Per 40-Year Plant Life	Dollars Per Occurrence ⁽²⁾	Total Cost Over Plant Life
Early Fatalities	.00003	.0012	500,000	\$ 600
Early Illnesses	.002	.08	50,000	4,000
Latent Fatalities	.021	.84	500,000	420,000
Thyroid Nodules	.21	8.4	10,000	84,000
Genetic Effects	.015	.6	50,000	30,000
<u>Property Damage</u>	\$20,000	\$800,000	1.5 (3)	<u>1,200,000</u>
Total				\$1,738,600

(1) Calculated from WASH 1400, Table 5-6 by dividing by 100 to convert to a single plant. Latent fatalities and thyroid nodules assume 30 years of occurrence and genetic effects assume 150 years, per WASH 1400, pages 73-75.

(2) Rough estimates

(3) To account for inflation since 1975

account for the 30-year period of incidence following an accident and the number for genetic effects has been multiplied by 150 for the same reason (see pages 73-75 in WASH 1400 for further discussion).

It was assumed that an early or latent fatality was valued at \$500,000 (as described above). Working from this number, the rough assumption was made that genetic effects and early illnesses are valued at \$50,000 and thyroid nodules at \$10,000. Using these numbers, it was calculated (see Table 3-1) that the total value of the average risk from one nuclear plant over a 40-year life is \$1.8 million. Two thirds of this represents property damage and only one fourth represents fatalities. Since the average impact of 40 years of plant operation is \$1.74 million and 0.84 fatalities, this suggests a rough conversion factor of $1.74/.84 \approx \$2$ million per life to account for all accident effects.

PLANT COSTS

Cost information presented is based on the DOE EEDB-IV Study (Reference 20). Costs presented are 1981 capital costs and do not include items such as escalation. AFUDC, etc. As such, this represents the cost to build a plant overnight. To the extent that capital cost savings can be realized, greater savings will be realized in total plant cost because escalation, etc., will not operate on these costs.

CONSERVATISMS AND UNCERTAINTIES

As mentioned earlier, a number of conservative assumptions (i.e., assumptions which result in overstated benefits and understated costs) were made in the individual evaluations and these are identified in the applicable discussions. However, there are three significant conservatisms which affect all analyses:

- o In general, the estimates of risk reduction (benefit) due to an NRC requirement are based on a combination of the results of WASH 1400 and the Zion PRA. The estimated total risk in both PRAs is about 0.02 fatalities per year. In the case of WASH 1400 this is likely to be high because the excessively high probabilities of containment failure which were used more than canceled out the somewhat low estimate of core-melt frequency. The Zion risk estimates are higher than would be obtained for a typical plant due to the high population density at Zion. Overall, it appears that the risks (and therefore risk reductions) estimated in this report are higher by a factor of at least 5 than what would be obtained using the Zion PRA methodology for a plant at a typical site.

- o As mentioned previously, there is a growing consensus that fission product source terms used in WASH 1400 and the Zion PRA are unrealistically high. If this is true, all risks and risk reductions discussed in this report are high (perhaps by a factor of 10 or even more).
- o All costs quoted in this report are base costs for a new plant (in roughly 1981 dollars). In general, it would not be correct to add escalation to these costs without also escalating the benefits to be gained. However, indirect costs and the effect of the NRC requirement on AFUDC (both the AFUDC on the money spent on the requirement and the effect of the requirement on construction duration) should be included, and they are not. This results in costs being understated by roughly a factor of 2.

Overall, it can be seen that the true cost-benefit ratios of NRC requirements are likely to be much greater (perhaps by a factor of 100) than calculated in the following sections.

It is often argued that PRA results cannot be used in defining nuclear plant design requirements because of the uncertainties inherent in PRAs and the data bases used in them. However, PRA results represent the best possible estimates of risk given the current state of knowledge. Regulatory requirements beyond what is indicated by the PRAs may occasionally turn out to be cost effective, but they will on the average result in excessive use of resources. Assuming that these resources could have been used more cost effectively in other areas, the net effect will be a reduction in the overall public welfare.

It should be pointed out that PRA results cannot be nonconservative by very large factors. Typical predicted core melt frequencies are on the order of 10^{-4} year⁻¹. Worldwide experience with LWRs is approaching 10^3 years without a core melt, so it is essentially impossible that calculated melt probabilities are low by more than a factor of 10. While it is possible that the consequences of a core melt will be worse than calculated, experimental evidence and the experience at TMI indicates that actual consequences are likely to be lower, and not higher than calculated.

COMPARISON OF NUCLEAR AND NONNUCLEAR RISKS

A comparison of the risk associated with these plants with nonnuclear risk is appropriate. WASH 1400 (Table 6-1 of the main report) states that there are 115,000 early fatalities per year due to nonnuclear accidents in the U.S. The major contributors are motor vehicle accidents (49%) and falls (16%). These are early fatalities whereas the values given above for Zion, Indian Point, and WASH 1400 are

total (early plus latent) fatalities. Typically the number of early fatalities is two to three orders of magnitude lower; however, this factor compensates for the assumption of 1,000 operating reactors in the U.S. Therefore, a direct comparison of the values above with 115,000 is appropriate and shows the risk of these plants to be extremely small in relation to nonnuclear risk. The ratio of nonnuclear to nuclear risk is about 6×10^6 for Zion and WASH 1400, 1.5×10^5 at Indian Point 2, and 4×10^5 at Indian Point 3. Since the total plant nuclear risk is small, the risk associated with individual areas is even smaller, and risk comparisons presented in the individual sections of this report will be limited to nuclear risk.

Section 4

COST-BENEFIT EVALUATION TORNADO-GENERATED MISSILES

INTRODUCTION

Concern over potential damage to safety-related components of a nuclear power plant resulting from tornado-generated missiles has resulted in NRC regulations aimed at reducing this risk. Current NRC positions are contained in Regulatory Guides 1.76 and 1.117 and Sections 3.3.2, 3.5.1.4, 3.5.2, and 3.5.3 of the Standard Review Plan. Among the items specified are tornado wind velocities, a spectrum of acceptable tornado-borne missiles and their impact velocities, and the minimum concrete thickness needed to protect critical components in the plant. Although the current version of the Standard Review Plan has an incorrect reference to Regulatory Guide 1.117 for an acceptable probability of damage, the intent was to state that if the probability of damage by a tornado-generated missile to the plant structures and systems is greater than about 10^{-7} per reactor year, provisions must be included to reduce the probability of unacceptable damage to that level.

PHENOMENOLOGY

It has generally been felt that the design requirements stipulated by the NRC were overly conservative, and that this in turn was contributing to the increased cost of nuclear plants. EPRI has been involved in programs related to tornado missiles for some time. Areas investigated included tornado wind speeds, tornado-borne missile trajectories, response of concrete to missile impact, and development of a missile risk analysis capability. Reference 21 reported on full scale tornado missile impact tests. These results showed that for a 12-inch diameter pipe, which was the most damaging of the penetrating missiles, 18 inches of 3000 psi design strength reinforced concrete prevented backface scabbing (generation of secondary missiles). These results apply to the case where the pipe is traveling at the design velocities stipulated by the NRC for Region I (the region of the country having the most severe tornado winds). Twelve inches of concrete was sufficient to prevent scabbing for Regions II and III. In comparison, Section 3.5.3 of the Standard Review Plan stipulates minimum wall thicknesses for 3000 psi concrete of 23 inches in Region I, 16 inches in Region II, and less than 6 inches in Region III. (NOTE: The Standard Review Plan allows thicknesses

less than those stipulated, provided sufficient justification is presented. Traditionally, utilities follow the NRC requirements rather than risk delay and extra cost in arguing.) Reference 21 was published in 1978 and admittedly reported on a limited number of tests. Reference 22 published in June, 1981, reported on scale model tests which showed basically the same response as the full scale tests. NRC-required concrete thicknesses appear excessive, even when assuming the NRC-specified velocity of impact; however, EPRI work is also indicating that the NRC impact velocities are quite conservative. For example, Reference 25 shows that for a 300 mph tornado and a missile injection height of 33 feet, the mean and 90% quantile value for the horizontal velocity of the 12-inch pipe is 46.9 and 70.1 mph respectively. The NRC design velocity in Region I is 105 mph.

References 23, 24, and 25 report on development of a probabilistic risk assessment methodology for assessing tornado risk to nuclear plants. Models were developed to describe each event in the tornado hazard sequence such as frequency of tornado occurrence, tornado windfield, missile injection and transport, as well as structural damage due to missile impact. Initial results were reported in References 23 and 24 in 1978. Reference 25 reports on updated results obtained from modeling a typical nuclear plant layout. In the plant simulation, a total of 65,550 missiles (assumes a second unit under construction at the site) from the NRC spectrum of missiles were assumed to be at typical locations in the plant vicinity (approximately 8500 of these missiles were assumed to be unrestrained or minimally restrained). The table below presents data, taken from this report, relevant to the mean probability of impact and scabbing damage for all safety-related structures at the site. The impact probability is the probability that at least one of the structures will be impacted by at least one of the missiles. The damage probability is the probability that at least one structure will be damaged, assuming all structures are the thickness stated.

Tornado Region	Impact Probability ⁽³⁾ (Yr ⁻¹)	Scabbing Damage Probability (Yr ⁻¹) Barrier Thickness ⁽¹⁾			
		6 Inches	10 Inches	14 Inches	18 Inches
I	3×10^{-4}	9×10^{-7}	6×10^{-8}	9×10^{-9}	(2)
II	3×10^{-5}	8×10^{-9}	(2)	(2)	(2)
III	2×10^{-5}	4×10^{-9}	(2)	(2)	(2)

(1) Reinforced concrete with nominal 4000 psi compressive strength and conventional steel reinforcement.

(2) No damage obtained.

(3) For all safety-related targets.

This data assumes a maximum tornado wind velocity of 300 mph. Current NRC requirements state a maximum velocity of 360 mph; however, Reference 25 concludes that 300 mph is consistent with best estimates for maximum intensity tornadoes. The increase in risk in assuming 360 mph winds ranged from factors of 3 to 7, depending on wall thickness.

Considering that a probability of unacceptable damage which is less than about 10^{-7} per year is acceptable to the NRC, the above data would indicate that 10-inch walls are adequate for protection in Region I, while 6-inch walls provide protection in Regions II and III. In this case, it has been conservatively assumed that scabbing produces severe damage with a probability of 1. In the Zion PRA, it was estimated that the probability of scabbing causing sufficient damage to internal components to result in core melt was 10^{-2} to 10^{-4} . Even using 10^{-1} for this probability reduces required wall thicknesses to 6 inches in Region I, and less in Regions II and III.

Since an auto is a "soft" missile and affects overall structural response rather than local effects, Reference 25 treats vehicle impact separately. The table below shows the results (mean values) from this report for vehicle impact probability and impact velocity exceedence probability.

Tornado Region	Impact Probability (Yr ⁻¹)	Impact Velocity Exceedence Probability (Yr ⁻¹)		
		$V_i \geq 57.1$ mph	$V_i \geq 82.1$ mph	$V_i \geq 107.1$ mph
I	2×10^{-5}	2×10^{-7}	2×10^{-8}	7×10^{-9}
II	1×10^{-6}	(1)	(1)	(1)
III	6×10^{-7}	(1)	(1)	(1)

(1) None

(2) No vehicle impacts were obtained for $V_i \geq 132$ mph.

(3) Again these data are for a maximum tornado wind velocity of 300 mph. Risk increased by about a factor of 5 if a 360 mph maximum velocity was assumed. Also, for the 360 mph wind, the mean probability of exceeding 132.1 mph was 1×10^{-9} .

These data should be considered in relation to the design vehicle velocities presented in Section 3.5.1.4 of the Standard Review Plan. These velocities are 132 mph, 116 mph, and 92 mph for Regions I, II, and III respectively. This would indicate that to the extent that vehicle impacts affect structural design, the NRC velocity of impact would result in overdesign.

COST BENEFIT ASSESSMENT

Risk Increment

PRAs have shown that the probability of core melt for older nuclear plants is on the order of 5×10^{-4} to 5×10^{-5} /year. A probability of core melt due to tornado missiles of 10^{-8} to 10^{-10} (10^{-6} for scabbing with 6-inch walls, and 10^{-2} to 10^{-4} for scabbing leading to core melt), therefore, contributes negligibly to the risk of melt and is far below the proposed safety goal of 10^{-4} for a probability of melt (Reference 19).

The risk associated with wind and tornado damage at Zion and Indian Point was evaluated in their PRAs. The contribution of tornadoes to risk was significant at Indian Point 2, small at Indian Point 3, and negligible at Zion. The main reason for this is that certain key structures at Indian Point 2 (the control building, diesel generator building, and portions of the auxiliary building) were steel framed as opposed to reinforced concrete. The auxiliary feedpump buildings at Indian Point 2 and 3 are partly steel. It is this absence of any protection in some areas, rather than any inadequacy in the design of concrete structures, which results in the nonnegligible risk contribution at Indian Point.

Current thinking (as typified in the Zion and Indian Point PRAs) is that most core melt sequences will not result in loss of containment function and will therefore result in few, if any, fatalities. The combination of core melt and containment failure requires that several systems fail and is therefore unlikely unless there is a loss of electrical power. Conservatively assuming that all tornado impacts sufficient to cause core melt, also result in loss of electric power (and thereby containment) gives a (conservative) average of about 2000 fatalities per melt (see Section 3). For 6-inch thick walls (in Region I), this yields a risk of:

$$(9 \times 10^{-7} \text{ hits/year})(10^{-2} \text{ melts/hit})(2000 \text{ fatalities/melt})$$
$$1.8 \times 10^{-5} \text{ fatalities/reactor year}$$

For 10-inch thick walls, the risk is:

$$1.2 \times 10^{-6} \text{ fatalities/reactor year}$$

These numbers are likely to be conservative because of the conservative estimate of 10^{-2} melts per hit and the assumption that all melts lead to containment failure. Using \$2 million per life and a 40-year plant life, the risk reduction from increasing wall thicknesses from 6 to 10 inches would justify the expenditure of \$1440, and the value of going from 10 inches to the 20 inches required by NRC (4000 psi concrete), an additional \$96.

Cost Increment

Within the scope of this study, it was not possible to quantify savings resulting from a reduction in the required wall thickness for tornado protection. This was mainly due to the complex interrelationship of many factors such as seismic loads, wind loads, general construction requirements, and building-to-building relationship. Another factor is the desire of the construction groups to maintain standardized wall thicknesses among the various buildings to simplify operations such as form placements. A more detailed assessment--beyond the scope of the present study--would be required to clearly establish any potential cost savings. As a part of this effort, it would be necessary to establish what requirement sets each representative wall thickness and to perform cost-benefit evaluations for these requirements which are dictated by NRC.

CONCLUSIONS

It has been shown that reducing exterior wall thicknesses required for tornado protection produces a negligible effect on the risk of the plant. Fatalities increase by 1.2×10^{-6} per reactor year in going to 10-inch thick walls and 1.8×10^{-5} in going to 6-inch walls compared, for example, to 2×10^{-2} total fatalities per year at Zion or in WASH 1400 where tornadoes were a nonsignificant contributor to risk. It was also shown that the risk-justified expenditure between a 6-inch thickness and current requirements is about \$1540 and between 10-inch and current requirements only \$100. Although it was not possible within the present scope to quantify cost associated with changes in required wall thicknesses, a more detailed evaluation, including reoptimization of other factors influencing wall thickness, would certainly show that costs of implementing current NRC requirements greatly exceed those justified through risk reduction.

PROPOSED FUTURE EFFORTS

It has been demonstrated that the adoption of a probabilistic approach to tornado-missile damage can result in a significant reduction in the required thickness of reinforced concrete protective structures. It appears that EPRI has this work well in hand.

One area in which only limited data seems to be available is the effects of backface scabbing. Some data is available in the EPRI report on full-scale missile tests (Reference 21); however, the data is too limited to draw valid conclusions. If it is assumed that scabbing always leads to core damage, 10-inch walls would probably be required in Region L. Quantification of scabbing effects could reduce the required thickness to 6 inches. The direct benefits of this may well be small since other requirements (such as seismic or shielding) will generally control wall designs at these thicknesses. Nonetheless, data on scabbing effects would better define the conservatism in missile design and would make it easier for NRC to accept reduced wall thicknesses.

It is understood that a PRA approach to tornado damage of the Palo Verde spray ponds is being reviewed for NRC by an outside consultant. Review of the EPRI work is to be included. Should points of disagreement on significant issues develop, funding of additional work may be required.

OTHER RISK CONSIDERATIONS

In the preceding discussion, all plant safety systems were treated the same. A more detailed analysis would certainly show that protection of some systems is more cost-effective than protection of others. Since a tornado is likely to cause loss of off-site power, protection of diesel generators and other portions of the vital electric systems including required support systems, will be necessary to reduce the chance of simultaneous core melt and loss of containment.

Current NRC requirements recognize that the combined probability of the joint occurrence of a tornado and LOCA is sufficiently small that equipment needed solely to deal with the short-term effects of the LOCA need not be protected. Equipment needed for long-term core cooling following a LOCA must, however, be protected. As a practical matter, this means essentially all ECCS equipment is protected. It is felt that a risk assessment on these equipment items would show that tornado protection was not cost effective; however, the savings is also likely to be small.

In most cases, walls around even minimal nuclear island equipment will have to be 6 inches or thicker because of structural, seismic or shielding considerations. If required wall thicknesses can be reduced to 6 or 10 inches, there appears to be little additional benefit to be gained from completely eliminating the requirement for tornado protection of some equipment. Therefore, efforts should continue to be focused on reducing required wall thicknesses.

Section 5

COST-BENEFIT EVALUATION SPENT FUEL POOL PROTECTION FROM TORNADO

INTRODUCTION

In Regulatory Guide 1.117, the NRC requires that nuclear plant spent fuel pools must be protected from tornado missiles to the extent necessary to preclude significant loss of watertight integrity of the storage pool and to prevent missiles from contacting the fuel within the pool.

This has meant that the roof and the walls above the pool (the superstructure) of spent fuel buildings in recent PWRs are thick reinforced concrete. In earlier PWRs, the superstructure consisted of a steel frame and steel siding. To assess the risk impact of current NRC requirements it is assumed that steel structures provide no tornado missile protection while concrete structures provide complete protection.

In the event of a tornado, three mechanisms exist which could potentially lead to damage to the fuel in the spent fuel pool and release of radioactivity.

These are:

- o A tornado-generated missile striking the walls of the pool and causing a leak large enough to result in drainage of the pool.
- o A tornado-generated missile dropping into the pool and striking the fuel.
- o The tornado sucking the water from the pool.

PHENOMENOLOGY

The phenomena of interest here are generally the same as those involved in general tornado protection. These were discussed in Section 4. Questions related to the spent fuel pool involve the trajectories of missiles entering the pool and the effect of missile impact on the stored fuel.

COST-BENEFIT ASSESSMENT

Risk Increment

The following discussion will cover each of the areas mentioned above. Typically the PWR spent fuel pool has dimensions of 25 x 30 feet, the pool walls are approximately 3 feet thick, and the pool has a steel liner. The depth of water above the fuel elements typically is about 15 to 20 feet. The wall thickness and depth of water are dictated by shielding requirements.

Missile Damage To Pool. Missile damage to the walls of the pool itself is not a problem. The 3-foot thick walls provide more than enough protection, even for the most energetic tornado missiles. NRC requirements, as given in Section 3.5.3 of the Standard Review Plan, view 23 inches of 3000 psi concrete to be adequate missile protection in Region I, while EPRI tests indicate that 18 inches is adequate to prevent backface scabbing for the most energetic missiles (see Section 4 of this report). Even if the pool walls were much thinner, the steel liner would not be damaged and would prevent water loss.

Tornado Missile Dropping Into The Pool. WASH 1400 addressed the concern of a heavy object such as a 100-ton spent fuel shipping cask or a turbine missile dropping into the spent fuel pool and resulting in either loss of pool integrity or damage to the fuel itself. It is unlikely that a tornado-generated missile could cause damage to the fuel, even if it entered the pool. First, the missile would have to be large enough to cause damage. Once this large missile is generated it would have to be transported in such a way that it would land in the pool. For above grade pools, this requires the missile to be lifted 20 to 30 feet in addition to the horizontal travel, as opposed to below grade pools where tumbling motion could result in a missile entering the pool. Most pools are above grade. This missile would then be required to pass down through 15 to 20 feet of water and still have sufficient energy to cause damage. Alternatively, it would have to have sufficient weight to cause damage if resting on top of the spent fuel. This suggests that the two tornado missiles of interest would be the 12-inch diameter schedule 40 pipe (the most penetrating missile), and the automobile (the heaviest).

The probability of a 12-inch pipe entering the fuel pool can be estimated from data presented in the literature. Reference 26 suggests that a wind velocity of about 250 feet/second (170 mph) would cause the pipe to begin moving over the ground. (It is conservatively assumed these velocities could also lift the pipe to a sufficient height.)

The simulation in Reference 25 gives the probability of any missile impacting at least one of the structures as about 6×10^{-5} per year in Region I for tornadoes of intensity Level 4 or greater (winds in excess of 170 mph). If it is conservatively assumed that all impacts are on the fuel building and that the number of impacts due to the 12-inch pipe is in the same relationship as the pipe to total missile population assumed (one of five missiles were 12-inch pipes), the probability of impact on the fuel building by the pipe becomes about 1×10^{-5} . Assuming that only 10^{-1} of the impacts on the building would have resulted in the pipe entering the pool if the superstructure were not present and one in ten pipes entering the pool have the required velocity and direction to impact fuel, gives an overall probability of damage of 1×10^{-7} .

Reference 25 also estimates that the probability of an automobile impacting the fuel building is about 2×10^{-6} per year in Region I. If it is again assumed that only 10^{-1} of the impacts would have resulted in the auto entering the pool and that entering the pool in any way results in fuel damage, the resulting probability of damage is about 2×10^{-7} .

Even if a missile strikes the fuel, it is unlikely to do more damage than the rupturing of the cladding of some fuel rods. Most of the noble gasses will have decayed. Damage to fuel cladding will result in release of small fractions of other isotopes and much of this will be absorbed by the pool water before it can escape. There is a small probability that a missile could disrupt fuel to the extent that cooling is blocked and some local melting occurs. Here again, scrubbing action by the water above the fuel would limit releases to a very small fraction of the radioactivity in a reactor core. The only potential for large releases would involve rearrangement of the fuel such that local criticality occurred and boiled the pool dry. This is not physically impossible, but is considered to be very unlikely, especially in reactors which have high density fuel racks containing large amounts of neutron poison in addition to the boron in the pool water.

Water Removal From Pool By Tornado. If the superstructure of the pool were not tornado protected, the possibility would exist that the tornado could suck the water from the pool, thereby causing loss of the ability to adequately cool the fuel. Information presented in Reference 27 was used to predict the degree of water loss. It was concluded that water loss would be limited to about 5 feet. Since approximately 10 feet of water would still cover the fuel rods, it is concluded that this will not result in a release of radioactivity.

Overall Risk Impact. Based on the above discussion, the only potential for release of radioactivity results from a tornado missile entering the fuel pool. The probability of such event was conservatively judged to be about 10^{-7} per year. Since it was not possible to estimate the consequences of fuel damage, the extremely conservative assumption is made that the release would be equivalent to an uncontained core melt. (Note that even if the pool boiled dry and all fuel in the pool melted, actual releases would probably be lower.) Using 2000 fatalities per event, 10^{-7} events per year and a 40-year plant life yields an increase of .008 fatalities per plant. Using this value and \$2 million per life yields a value of providing a concrete superstructure for the spent pool of \$16,000.

WASH 1400 discusses a similar event in that it considers the possibility of a large turbine missile impacting the spent fuel pool. In this case the probability of release was judged to be 4×10^{-7} . While WASH 1400 does not specifically give the consequences of such a release, it was concluded in Section 5.4.5 of the main report that the turbine missile did not represent a significant contribution to the overall risk. Since the probabilities of the tornado event considered here and the turbine missile event are similar, it is reasonable to conclude that removing the tornado-proof fuel storage superstructure would not result in a significant increase in risk at Surry (WASH 1400). In the Zion and the Indian Point PRAs, the consequences of a spent fuel pool release were not presented, and therefore it is not possible to quantify the public risk. Presumably the risk was considered to be negligible, even though the fuel buildings at Indian Point 2 and 3 are not tornado protected. Neither the Crystal River PRA (Reference 13) nor the Oconee or Sequoyah RSSMAP Reports (References 11 and 12) addressed noncore-related accidents.

The two PWRs for which NRC has an SEP Report, Ginna and Palisades, have spent fuel pools whose superstructure is not protected from tornado missiles. At Ginna, this question is being evaluated by the utility as part of an overall structural design review (Reference 28, pages 4-8). At Palisades, however, NRC has concluded that lack of tornado protection results in a small enough increment of risk that backfitting is not required (Reference 29, pages 4-16). No analyses are presented to justify the position, although the Palisades report mentions, but does not reference, a tornado-missile analysis.

The tornado-missile methodology discussed above has been used for the North Anna plant to evaluate the probability of a missile entering the fuel pool and hitting fuel assemblies. North Anna has an above ground pool whose superstructure is sheet metal construction. This information was obtained too late to evaluate; however, the statement made in the FSAR is that for the entire population of postulated missiles, the combined probability of at least one hitting a fuel assembly is 4.3×10^{-8} . While no details on missile population or location were presented, this information would support the proposition that the estimate of 10^{-7} for the probability of serious fuel damage, used in this evaluation, is conservative.

Cost Increments

The cost of providing a tornado-proof superstructure depends on:

- o The required wall thickness (10-inch walls would be less expensive than the current requirements).
- o Whether seismic protection is required (NRC now requires that fuel pool superstructures be seismically designed. If this requirement were removed, the savings from reduced tornado protection requirements would probably be larger.)
- o Whether fuel building filtration is required (NRC now requires filtration in case of a fuel handling accident and use of a concrete superstructure may reduce the required capacity of the filter system. Elimination of this requirement would reduce the incentive for concrete and most likely increase the savings if tornado protection requirements were eliminated.)

For this study, the information contained in EEDB-IV was used to evaluate the cost savings in changing the superstructure from concrete to steel siding. The turbine building was used as a model for the steel siding with appropriate conservative adjustments for structural reduced steel requirements due to differing crane and wind loadings. The resulting unit costs are shown below:

STEEL SIDING STRUCTURE

<u>Material</u>	<u>Cost Item</u>	<u>Unit Cost (\$/Ft²)</u>
Structural Steel	Labor	18.62
	Material	47.90
Steel Siding	Labor	4.55
	Material	6.88
Roofing	Labor	1.67
	Material	1.70

STEEL SIDING STRUCTURE, (Cont.)

CONCRETE STRUCTURE*

Formwork	Labor	38.71
	Material	3.71
Concrete	Labor	9.13
	Material	7.01
Reinforcing Steel	Labor	14.70
	Material	7.02

* Cost of walls and roof, combined and divided by wall area.

The resulting cost savings is conservatively calculated to be \$56,000. This value could possibly be as high as \$200,000 depending on structural steel requirements. This figure does not reflect an adjustment which may be required for increasing the "leak-tightness" to accommodate the fuel handling accident as discussed above. (It is likely that a more detailed evaluation of fuel handling accidents would minimize or eliminate this impact.)

Conclusion

It has been conservatively estimated that the removal of the tornado-protection requirement for the fuel pool superstructure results in an increase of .008 fatalities per plant. This compares to a representative total plant risk at Zion or WASH 1400 of .8 fatalities (.02 per year times 40-year life) and, therefore, represents a negligible increase in risk. The cost of the tornado-proof superstructure was estimated to be between \$56,000 and \$200,000. Based on the risk justified expenditure of \$16,000, the cost-benefit ratio is between 3.5 and 12.5. The true cost-benefit ratio is likely to be considerably higher because of the extreme conservatism incorporated into the risk evaluation.

FUTURE WORK

The TORMIS computer code (Reference 25), developed under EPRI sponsorship, could be used to model several typical plant arrangements and obtain more accurate estimates of the probability of tornado missiles striking the surface of an unprotected fuel pool at various speeds and angles. Depending on the results of these calculations, it might also be necessary to investigate missile trajectories in water and/or the effects of missile impacts on spent fuel racks. There appears to have been little or no work done on these subjects in recent years.

Section 6

COST-BENEFIT EVALUATION USE OF ASME SECTION III CRITERIA FOR SAFETY SYSTEM PIPING

INTRODUCTION

The NRC currently requires that a number of fluid systems which are important to safety be designed to ASME Section III. Section III provides for three different levels of quality, Classes 1, 2, and 3 (with Class 1 being the most stringent). Stated simply, the NRC (in Regulatory Guide 1.26) requires specification of Class 1 for the reactor coolant system, Class 2 for safety systems such as emergency core cooling, and Class 3 for support systems such as service water. Piping in nonsafety systems is normally designed to ANSI B31.1. B31.1 was used for all piping in a number of earlier nuclear plants and is still used in fossil-fired plants. An analysis was performed to determine the cost effectiveness of use of Section III versus B31.1. This analysis was not concerned with piping (such as primary coolant, and main steam and feedwater), the failure of which could initiate an accident, but only with piping in those systems which perform a standby safety function; such as low pressure safety injection, high pressure safety injection, auxiliary feedwater, containment spray, safety-related service water, and safety-related closed cooling water.

TECHNICAL CONSIDERATIONS

The piping in earlier nuclear plants was built to ANSI B31.1 and this is still used for nonnuclear piping in nuclear and fossil power plants. For nuclear plants, B31.1 was supplanted by B31.7 which was later incorporated into ASME Section III. The purpose of all these codes is to give a high level of assurance that piping will not fail under the conditions for which it was designed. Both B31.1 and ASME Section III accomplish these goals very well and failures in power plant piping of all types are infrequent. Data on pipe failures is sparse and difficult to interpret. However, it appears that most failures occur due to imposition of loads well beyond design (caused by water hammer, thermal cycles, flow induced vibration, or excessive restraint). Reference 30 supports this statement. Since the probability of these types of failures is not significantly affected by the pipe code used, it might be expected that failure rates for B31.1 piping and ASME Section III piping would be very similar.

Additional analyses and quality control on ASME Section III piping presumably results in some reduction in failure rate, but this is probably small compared to the total failure rate. It is interesting to note that wall thicknesses may be less for ASME Section III pipe than for B31.1 pipe. This decreased ability of ASME Section III pipe to resist greater-than-design loads will presumably increase its failure probability, although this effect is also likely to be small.

COST-BENEFIT ASSESSMENT

Pipe Failure Rates

WASH 1400 gives the failure rate for piping larger than 3 inches in diameter as 1×10^{-10} per hour per section. A typical safety system train will have 10 to 20 sections, so the train failure rate will be on the order of 1.5×10^{-9} per hour.

In order to estimate the probability that a pipe will not perform its function during an accident, the number of hours that the pipe is "at risk" must be determined.

The following are typical "at risk" times before an accident:

- o If the system is continuously pressurized, failures are self-alarming and the "at risk" time is basically the time to shut down the plant if a failure occurs. A 12-hour period is assumed.
- o If the system is pressurized only during a monthly test, the average "at risk" time is $(0.5)(744) = 372$ hours.
- o If the system is pressurized only during an annual test, the average "at risk" time is $(0.5)(8760) = 4380$ hours.

Therefore, the probability that a pipe in a safety system train is already failed at the time of an accident ranges from $(12)(1.5 \times 10^{-9}) = 1.8 \times 10^{-8}$ to $(4380)(1.5 \times 10^{-9}) = 6.6 \times 10^{-6}$.

Once an accident has occurred, there is some probability that a pipe failure will occur during the time the safety system train is needed. Typical "at risk" times are:

- o If the system is used only for injection, the at risk time ranges from 0.5 hours to as much as 8 hours.
- o If the system is needed for post-accident recirculation, it may be needed for months or longer. However, as time passes, the likelihood increases that operators can bypass broken pipes and/or reestablish flow through

redundant paths. Most risk assessments (e.g., Zion, Indian Point, WASH 1400) assume that the "at risk" time is 24 hours.

Therefore, failure probabilities range from $(4)(1.5 \times 10^{-9}) = 6 \times 10^{-9}$ to $(24)(1.5 \times 10^{-9}) = 3.6 \times 10^{-8}$.

System Failure Rates

In recent plant designs, safety systems typically consist of two completely separate trains. Each train consists of one pump, several manual and remote actuated valves, and piping. Each train is supported by its own separate train of actuation logic, AC Power, DC Power, and cooling water. Analyses in WASH 1400 typically found the probability that one train of a safety system would fail to perform its function was on the order of 10^{-2} . (Appendix 3 summarizes the WASH 1400 analyses of Low Pressure Safety Injection and Core Spray Injection for the PWR.) Because of single failure points and common mode failures, system failure rates were typically on the order of 3×10^{-3} .

Assuming that a safety system is tested monthly and must function for 24 hours after an accident, the probability of pipe failure in a train is $(1.5 \times 10^{-9}) (744/2 + (24)) = 5.9 \times 10^{-7}$. Pipe failure will contribute to system failure only when a pipe in one train fails at the same time the other train fails. The probability of this is $(2)(10^{-2})(5.9 \times 10^{-7}) = 1.2 \times 10^{-8}$.

WASH 1400 (Tables II 2-1 and III 2-2) also gives pipe failure probabilities for post-accident environments. These are 3×10^{-9} /hour for welds and 3×10^{-7} /hour for elbows and flanges. Since the weld failure rates are so much lower, they can be ignored. A detailed analysis of the impact of this on overall system failure rates has not been attempted; however, the following observations can be made:

- o About half the major safety systems (service water, component cooling water, auxiliary feedwater) never see post-accident conditions.
- o Most of the risk from a nuclear plant comes from transients and small LOCAs where post-accident conditions are mild.
- o Even for large LOCAs, severe post-accident environments will typically persist for perhaps 24 to 48 hours. In any case, the concern is with the first 24 hours.

To reflect the potential contribution to failure due to environment, it is assumed that one half the safety systems see a severe environment for 24 hours after 1% of the accidents and that they have 20 elbows per train with a failure rate of 3×10^{-7} /hour. This gives an environmentally related train pipe failure probability of $(3 \times 10^{-7})(.5)(.01)(24)(20) = 7.2 \times 10^{-7}$. The probability of system failure is $(2)(10^{-2})(7.2 \times 10^{-7}) = 1.4 \times 10^{-8}$. Summing the effects of the two types of pipe failure, yields the total contribution of pipe failures to system failure $(1.2 \times 10^{-8}) + (1.4 \times 10^{-8}) = 2.6 \times 10^{-8}$. Since the overall probability of system failure is about 3×10^{-3} , pipe failures are responsible for about 10^{-5} of all system failures.

Risk Increment

For a PWR, WASH 1400 calculated the probability of core melt in each of seven release categories. These are given in Table 6-1. WASH 1400 gives the consequences of all melts but does not give the consequences for each release category. The deaths per release category shown in Table 6-1 are based on Reference 4 as discussed in Section 3.

Most dominant core melt sequences in WASH 1400 involve failure of a single safety-related fluid system (Auxiliary Feedwater, HPSI or LPSI). Exceptions are reactor vessel rupture and interfacing LOCA. Table 6-1 shows the probability of each release category with the effects of these two events subtracted out. It was shown earlier that pipe failures are responsible for about 10^{-5} of the failures of these systems. Therefore, the contribution of pipe failure to core melt probability is the probability of melt times 10^{-5} and this is also shown. Multiplying this incremental risk of melt by the fatalities per melt gives the fatalities per year in Table 6-1. Summing these results gives a total risk of 1.3×10^{-7} fatalities per year.

Some WASH 1400 release categories occur when containment sprays function while others result from similar sequences, but without sprays. Specifically, Category 3 involves some types of melt situations with the sprays working; while Categories 1 and 2 involves similar situations, except sprays do not work. Similarly, Category 4 is Category 5 without sprays and Category 6 is Category 7 without sprays; therefore, the probability of Category 6 is the probability of Category 7 times the probability that sprays will fail in those situations. If pipe failures contribute 10^{-5} to the failure probability of the spray system, then they are responsible for a probability of melt in

Table 6-1

PIPING CONTRIBUTION TO RISK BASED ON WASH 1400

Release Category	Probabilities			Risk	
	WASH 1400	WASH 1400 Minus Vessel Rupture & Interfacing LOCA	Increment due to Pipe Failure	Fatalities Per Release *	Fatalities Per Year
1	9×10^{-7}	5×10^{-7}	5×10^{-12}	2.0×10^3	1×10^{-8}
2	8×10^{-6}	4×10^{-6}	4×10^{-11}	1.8×10^3	7.2×10^{-8}
3	4×10^{-6}	4×10^{-6}	4×10^{-11}	1.1×10^3	4.4×10^{-8}
4	5×10^{-7}	5×10^{-7}	5×10^{-12}	1.9×10^2	9.5×10^{-10}
5	7×10^{-7}	7×10^{-7}	7×10^{-12}	4.8×10^1	3.4×10^{-10}
6	6×10^{-6}	6×10^{-6}	6×10^{-11}	1.1×10^1	6.6×10^{-10}
7	4×10^{-5}	4×10^{-5}	4×10^{-10}	.3	1.2×10^{-10}
8	4×10^{-5}	4×10^{-5}	4×10^{-10}	.4	1.6×10^{-10}
9	4×10^{-4}	4×10^{-4}	4×10^{-9}	.001	4×10^{-12}

* See Section 3

Category 6, which is equal to 10^{-5} times the probability of melt in Category 7. The same would be true for Categories 5 to 4, and 3 to 1 and 2. Summing these yields 7.6×10^{-8} . Adding this to 1.3×10^{-7} gives 2.1×10^{-7} fatalities per year. This is a rough estimate of the total reduction in risk which would result if the probability of pipe failures in safety systems were reduced to zero. Assuming a 40-year plant life and $\$2 \times 10^6$ per life for risk reduction, the value of eliminating all probability of pipe failure at one unit would be $(2 \times 10^6)(40)(2.1 \times 10^{-7}) = \16.80 .

Appendix 2 calculates the risk associated with random pipe failures in nonprimary coolant, safety-related systems at Zion 1 and 2 and Indian Point 2 and 3. The analysis is based on the Zion and Indian Point probabilistic safety studies. These studies noted that the safety-related pipe systems at both stations were designed in accordance with the 1967 edition of ANSI B31.1.

The calculated risk increments due to failure of pipes in safety systems (excluding primary coolant, main steam, and feedwater systems) is:

	<u>Fatalities Per Year</u>
Zion 1 or 2	4.00×10^{-6}
Indian Point 2 *	4.50×10^{-4}
Indian Point 3 *	3.90×10^{-4}

*Indian Point Values estimated based on Zion detailed calculations

At both Zion and Indian Point, the principal contributor to pipe failure-induced risk is the loss of off-site power followed by a service water pipe failure which results in loss of all diesel generators. (Loss of all AC power causes loss of reactor coolant pump seal cooling and leads to a small LOCA. Absence of AC power results in failure of ECCS and containment cooling leading to core melt and containment failure.)

While the risk due to random pipe failures at these stations is low, it is from twenty to over two thousands times greater than the general analysis presented above. At Indian Point, the risk contribution is about two thousand times higher than in the generic analysis. Roughly, the difference is due to the following factors (the multipliers on the generic analysis are given in parentheses).

- o The Indian Point PRA uses the mean rather than the median for pipe failure probability. (x 9)
- o Since the service water system is continuously in operation, the at risk time is 24 hours rather than 15 days (x 1/15)
- o The Indian Point service water systems have a single nuclear header (for each unit) rather than the two header systems in the generic analysis. (x 100)
- o Indian Point has higher consequences per release than WASH 1400. (x 2)
- o Indian Point has a higher probability of loss of off-site power than WASH 1400. (x 10)

At Zion, the risk contribution is about one hundred times lower than at Indian Point for the following reasons.

- o Service water pipe breaks are assumed to be isolated after one hour so the at risk time is one hour rather than 24 hours. (It is noted that Zion has a shared service water system with two headers, one supplied by each unit. The headers are cross connected with pipe rupture in one header failing the other until isolation is achieved). (x 1/24)
- o Probability of loss of off-site power is lower than at Indian Point. (x 1/4)

The preceding illustrates that plant-specific feature and analytical methods can cause substantial variation in calculated risk. Overall, however, the average calculated risk contribution at Indian Point would not differ significantly from the generic analysis results were it not for the fact that Indian Point has a single service water header.

The single header means that a pipe failure will fail the system where the generic analysis assumed the separate headers so that a pipe failure plus a pump failure (failure probability = 10^{-2}) are required to fail the system. As discussed in Section 7, the single header design results in an overall improvement in system reliability but substantially increases the (relatively small) impact of pipe failures on system failure probability.

As a general rule, if all safety systems in a plant were designed with single headers, the worth of the risk reduction which could be achieved by eliminating all pipe failures would be $(\$16.80) \times (100) = \1680 .

Cost Increment

The cost savings associated with changing the required piping code from ASME III to B31.1 in a PWR was evaluated based on data in EEDB-IV. From this data it was possible to separate the piping into two categories: small (2½ inches in diameter and under) and large (greater than 2½ inch diameter). The analysis was also limited to consideration of carbon steel and stainless steel pipe since these pipes represented about 95% of the piping cost. Table 6-2 gives the quantities and costs for the safety class (ASME III) and the nonsafety class (B31.1) piping. Cost savings were calculated by applying the unit cost for nonsafety class (NNS) piping to the quantities of safety class (SC 2, 3) piping. The resulting costs are given in Table 6-3 which shows the cost savings to be about \$23 million. All of this savings is in labor costs which on a unit basis is two to three times higher for ASME III than B31.1. These greater labor costs are the result of increased procedural costs (QA, etc.) involved in the installation of the piping. Table 6-2 shows material costs for the B31.1 option to be lower for small pipe but higher for large pipe. This anomaly results from the inability to break the large pipe into several size groupings since cost trends do not move uniformly for different sizes. It is believed that further breakdown of large piping would show a savings in cost of materials on the order of \$1 million.

It was not possible to separate out the steam/feedwater pipe cost in this estimate. The cost differential, however, should not be greatly impacted because the majority of steam and feedwater piping is already nonnuclear, and also because the cost differential involved in converting that portion which is ASME III to B31.1 is not great. (Large, nonnuclear, high-pressure/high-temperature piping is, in practice, already subjected to essentially the same procedural requirements as ASME III.)

CONCLUSIONS

It has been estimated that the value of eliminating all probability of pipe failure in safety systems in one plant designed to current NRC requirements (dual trains) is \$16.80. This is based on an estimated reduction of 2.1×10^{-7} fatalities per year (a small fraction of the total risk of .02 fatalities per year at either Zion or for WASH 1400). It is also estimated that the incremental cost of using ASME Section III rather than B31.1 for all Class 2 and 3 systems other than steam and feedwater is about $\$23 \times 10^6$. If use of ASME Section III would eliminate all probability of pipe failure, the cost of doing this would still be about 1.3 million times the benefits. Since using Section III does not reduce the failure probability to zero, the actual cost-benefit ratio is undoubtedly higher.

Based on the preceding, it appears that:

- o The reduction in pipe failure rates achieved by using ASME Section III rather than ANSI B31.1 cannot be quantified, but is likely to be small.
- o Even if use of ASME Section III completely eliminates the probability of pipe failures in standby safety systems, its cost would exceed its benefits by a factor of 1.3 million or more.

PROPOSED FUTURE EFFORTS

One can conceive of a research project to demonstrate that ASME Section III and ANSI B31.1 result in essentially the same failure rates. Any laboratory tests would yield dubious results and actual failure rates in the field are so infrequent that it is essentially impossible to gain statistically significant data. One R&D project which might be of some use would be to do risk assessments of pipe failures for a number of specific units. The results presented here are generic or related to only two specific plants (Zion and Indian Point) and might not be sufficient to convince the NRC. The treatment of this subject in many existing risk assessments is superficial, because it is clear that there will not be a sufficient contribution to risk. However, this means that these risk assessments do not accurately quantify the very small risk contribution which does exist. The NRC might wish to see specific detailed quantifications of risk before deciding to relax their requirements.

Table 6-2

PIPING COST ANALYSIS

Material	Safety Class	Quantity	Total Labor Cost		Material Cost	
			Total	Unit	Total	Unit
<u>Small Piping - 2½ inch and under</u>						
Carbon Steel	SC 2,3	19,563	587,777	30.00	39.126	2.0
Stainless Steel	SC 2,3	30,890	2,330,684	75.45	308,900	10.0
Carbon Steel	NNS	109,865	1,335,214	12.00	120,853	1.1
Stainless Steel	NNS	102,778	3,088,007	30.00	539,587	5.25
<u>Large Piping - Larger than 2½ inch</u>						
Carbon Steel	SC 2,3	1,480,740	17,603,366	11.89	3.786,401	2.56
Stainless Steel	SC 2,3	409,280	18,514,512	45.24	3.880,750	9.48
Carbon Steel	NNS	4,551,454	26,889,054	5.91	10,468,345	2.30
Stainless Steel	NNS	307,077	4,613,117	15.02	3,377,847	11.00

Table 6-3

PIPING COST SUMMARY

Material	Quantity	Total Labor Cost ¹		Total Material Cost ^{1,2}		Total Cost ¹	
		ASME III and B31.1	All B31.1	ASME III and B31.1	All B31.1	ASME III and B31.1	All B31.1
<hr/>							
<u>Small Piping - 2½ inch and under</u>							
Carbon Steel	129,428	1,923	1,570	160	142	2,083	1,712
Stainless Steel	133,668	5,419	4,015	848	702	6,267	4,717
<u>Large Piping - Larger than 2½ inch</u>							
Carbon Steel	6,032,194	44,492	35,640	14,254	13,874	58,746	49,514
Stainless Steel	716,357	23,128	10,761	7,259	7,880	30,387	18,641
Totals	7,011,647	74,962	51,986	21,513	21,754	97,483	74,584

1 Thousands of dollars

2 Not including nonpermanent materials used during construction

Section 7

COST-BENEFIT EVALUATION PHYSICAL SEPARATION OF REDUNDANT SAFETY SYSTEM PIPING

INTRODUCTION

Standard Review Plan, Section 3.6.1, describes which systems and components NRC requires to be protected against the effects (pipe whip, jet impingement, etc.) of postulated pipe failures. Basically there are three requirements.

- o For pipe breaks (such as a LOCA) which require safety system action to prevent core melt, the needed safety systems must be protected so there is no loss of redundancy.
- o For pipe breaks in systems required for safe shutdown, the system may suffer a loss of redundancy but not a loss of function (i.e., the redundant train must be protected).
- o For pipe breaks in safety systems not required for safe shutdown, the system may suffer a loss of redundancy but not a loss of function (i.e., the redundant train must be protected).

It is recognized that safety systems must be protected from the physical effects of the accidents which they are intended to mitigate. It is also recognized that leaks from packings, gaskets, etc., are much more likely than pipe ruptures and that electrical equipment in redundant trains must be protected from the effects of such leaks. While written NRC requirements are vague, recent nuclear plants are typically designed so that individual safety system trains (including pipes as well as pumps, valves, etc.) are physically separated from redundant trains by concrete walls. It is this "requirement" which is addressed in this section.

COST-BENEFIT ASSESSMENT

The typical safety system consists of two trains. Possible options for piping of these systems tend to fall into one of three classes:

- I. Common header
- II. Separate trains with minimal physical separation
- III. Separate trains with complete physical separation

Early nuclear plants often contain systems which are a hybrid of Types I and II while systems in more recent plants are almost exclusively Type III. Although NRC presently will not permit designs with a common header, both Types I and III will be considered.

Risk Increment

Separate Train Systems. As shown in Section 6, standby safety system failure probabilities are typically:

One train (nonpiping)	10^{-2}
One train (piping)	6×10^{-7}
System	3×10^{-3}

The purpose of physical separation is to ensure that a pipe failure in one train will not result in failure of the redundant train. Since the majority of piping is in pipe chases, the majority of failures might be expected there. It is very unlikely that a broken pipe could seriously damage another pipe of the same size. Failure of pipes in the vicinity of pumps and valves are much more likely to damage redundant equipment but, even here, failure of the redundant train is not inevitable. Nonetheless, it is conservatively assumed that pipe failure in one train will always fail the redundant train. Thus, the probability of system failure due to pipe failure is $(2)(6 \times 10^{-7}) = 1.2 \times 10^{-6}$. This is $(1.2 \times 10^{-6}) / (3 \times 10^{-3})$ or 1/2500 of the probability of system failure from all causes.

Section 6 showed that changing the probability of system failure by a factor of 10^5 changed the total risk (conservatively) from one plant over 40 years by $(40)(2.1 \times 10^{-7}) = 8.4 \times 10^{-6}$ fatalities. Therefore, the maximum reduction in risk to be gained by requiring physical separation of all redundant safety system trains is $(100,000/2500) \times (8.4 \times 10^{-6}) = 3.4 \times 10^{-4}$ fatalities per plant. There is no doubt that physical separation will essentially eliminate any probability that pipe failures will fail redundant systems. However, actual risk reduction may be considerably lower because the calculated risk contribution is conservative (primarily due to the assumption that all pipe failures will fail the redundant systems).

Common Header Designs. Weaver (Reference 14) has shown that, for some system designs, headers between trains reduce the overall probability of system failure. A simple example will show how this can be true. Consider a system with two trains,

each of which contains one pump and one normally closed motor-operated valve. Typical failure rates are 2×10^{-3} for the pump (with its motor and controls) and 2×10^{-3} for the valve. Overall system failure probability is 3×10^{-3} . A portion of the system failure probability results from failure of the valve in one train and the pump in the other train. This failure contribution, $(2)(2 \times 10^{-3})(2 \times 10^{-3}) = 8 \times 10^{-6}$, would be eliminated in a design with a header connecting the two trains at a point between the pumps and the valves. Considering a system with 15 segments which is tested monthly, the probability of system failure due to header failure would be $(15)(10^{-10})(744/2 + 24) = 5.9 \times 10^{-7}$. Thus, while the improvement in system availability due to adding the header is very small, it is larger than the decrease in availability due to pipe failures with the net effect being a small improvement in safety. As the number of active components in each train increases, the benefit to be gained from headering increases. Weaver also shows that isolation valves in the header make the system less reliable than if valves are omitted, but (at least in some cases) still more reliable than the system with no header at all (the current NRC requirement). Specific benefits of interconnecting safety system headers are addressed in Appendix 4.

Once a header (without valves) has been added to a system, there is no reason for physical separation (except for "splash" protection) because any gross pipe rupture will fail the system, whether or not it is separated. In fact, there is an incentive to minimize the amount of piping in the system in order to reduce the probability of failure. Therefore, rather than having a safety injection system with two pipes, each running from one pump to the reactor in separate pipe chases, there would be a single pipe, carrying the the flow from both pumps to the reactor. This would substantially reduce costs for piping, building space, and walls.

Cost Increment

It is clear that there are savings involved in having single header systems or in running lines in the same pipe chase as opposed to separate chases; however, it was not possible to quantify the savings. This is partly because other things such as electrical separation criteria and fire protection requirements in conjunction with piping separation have been intertwined and have led to optimization of plant layouts around these requirements. True assessment of the cost impact would therefore require reoptimization of plant layout based on new criteria.

CONCLUSIONS

It has been shown that the maximum benefit of physical separation is a risk reduction of 8.6×10^{-4} fatalities per plant. This compares with .8 fatalities per plant (.02 per year times a 40-year life) at Zion or as arrived at in WASH 1400. Using $\$2 \times 10^6$ per life yields the benefit gained from complete physical separation as \$1720.

NRC currently requires that typical safety systems contain two 100% capacity trains which are physically separated and are not connected by headers. Maintaining the two trains, but eliminating the physical separation (except as necessary to prevent damage to electrical components by small leaks), would result in a very small increase in risk and a significant reduction in cost. Providing cross headers between the trains and eliminating redundant piping would further reduce costs and would, in many cases, slightly improve safety over that achieved under current NRC requirements.

Section 8

COST-BENEFIT EVALUATION SEISMIC DESIGN

INTRODUCTION

Based on a fairly simple analysis, WASH 1400 concluded that the probability of seismic-induced melt was about $6 \times 10^{-8} \text{ yr}^{-1}$ for a reactor located on a firm (typical) site and designed for 0.2 g. Given this low core melt frequency, the effect on public risk was not quantified. Many of the PRAs done since WASH 1400 (including those done as part of the SEP and IREP programs) do not include consideration of seismic events. The only PRAs for PWRs which considered seismic and were available at the time of this work were those for Zion and Indian Point.

METHODOLOGY

At Indian Point and Zion, three different methods were used to estimate the relationship between earthquake magnitude and probability. The three results, and the uncertainty distribution on each, were combined to give a probability distribution of earthquake occurrence as a function of magnitude. For each important component and structure in a plant, the median (best estimate) seismic capability was estimated and an associated uncertainty distribution derived. Based on plant design and the median seismic capabilities, those components/structures (and combinations thereof) which were likely to contribute to risk were identified. The probability distribution of frequency of each plant damage state was then determined based on the earthquake magnitude distributions, component load capability distributions, and the plant logic (combinations of failures which will lead to core damage).

COST-BENEFIT ASSESSMENT

Overall Risk Increment

The overall risks at Zion and Indian Point due to seismic events were calculated to be as follows:

MEAN PROBABILITY OF CORE MELT (YR⁻¹)
(SEISMIC EVENTS)

<u>Zion</u>	<u>Indian Point 2</u>	<u>Indian Point 3</u>
5.6 x 10 ⁻⁶	1.4 x 10 ⁻⁴	3.1 x 10 ⁻⁶

FATALITIES (EARLY AND LATENT) PER YEAR
(SEISMIC EVENTS)

	<u>Zion</u>	<u>Indian Point 2</u>	<u>Indian Point 3</u>
Seismic	.016	.61	.011
Other	.001	.81	.275
Total	.017	1.42	.286

Overall, there is little difference between Indian Point 2 and 3 with respect to seismic capabilities, Indian Point 2 being somewhat better in some areas and Indian Point 3 being somewhat better in others. The major difference between units is that Unit 2 shares a control building with (very early design) Unit 1. It is calculated that the Unit 1/2 control building will fail at a median ground acceleration of .27 g, while the next weakest components in Units 2 and 3 fail at a median acceleration of .83 g. Since the probability of an earthquake tends to fall off exponentially with magnitude and since loss of the control building is considered to lead directly to core melt and containment failure, it is not surprising that control building failure dominates seismic-induced risk at Indian Point 2 (omitting control building failures, the probability of seismic-induced melt at Indian Point 2 is about twice that at Indian Point 3 and only slightly higher than that at Zion). Given that Indian Point 2 is otherwise similar to Indian Point 3 and that the control building situation is clearly anomalous, all future discussions will be limited to Zion and Indian Point 3.

It is interesting to note that the probabilities of seismic-induced melt at Zion and Indian Point 3 exceed those in WASH 1400 by factors of 100 and 50 respectively. Because the analyses were so different, it is impossible to quantify the reasons for these differences. However, three factors appear to be significant:

- o WASH 1400 assumed two systems must fail to lead to melt. At Zion and Indian Point there are some systems which alone can lead to melt--given that loss of off-site power is essentially inevitable.
- o The uncertainties in equipment capabilities at Zion and Indian Point are large and appear to be larger than was assumed in WASH 1400.
- o The core melt probabilities given for Zion and Indian Point are mean values while the WASH 1400 probability is a median. The median values for Zion and Indian Point 3 are 2×10^{-6} and 2.4×10^{-8} respectively, considerably closer to the WASH 1400 value of 6×10^{-8} .

Significant Contributors to Risk

The following are the failures and combinations of failures which have significant impact on the probability of seismic-induced core melt at Zion and Indian Point 3.

Zion. The following is a summary of those sequences which are most likely to lead to seismic-induced core melt at Zion. Numbers in parentheses are the median ground accelerations at which the equipment is expected to fail. Note that there is considered to be only a 5% chance that earthquakes larger than .65 g are physically possible at Zion. Components with higher seismic capabilities still have some chance of failure due to uncertainties in the maximum possible earthquake and in the component seismic capabilities.

1. Loss of off-site power (0.2 g) combined with failure of service water pumps (0.63 g). Pump failure leads to diesel failure which results in loss of reactor coolant pump seal cooling and a small LOCA. Absence of AC power results in loss of ECCS and containment cooling and leads to core melt and containment failure.
2. Soil failure beneath the reactor building (0.73 g) leads to differential movement between the reactor building and auxiliary building. This fails many pipes and leads to core melt and loss of containment.
3. Loss of off-site power (0.2 g) combined with failure of an auxiliary building wall (0.86 g) leads to loss of AC power and the same consequences as Sequence 1 above.
4. Failure of the refueling water storage tank (0.73 g) and the condensate storage tank (0.83 g) results in loss of auxiliary feedwater and reactor makeup and causes core melt. Containment remains functional.
5. Loss of off-site power (0.2 g) combined with failure of the service water pump crib house (0.86 g) gives the same results as Sequence 1 above.

The mean probability of core melt from all seismic effects is 5.6×10^{-6} . Essentially all core melts result from Sequences 1, 2, and 3 with Sequence 1 being by far the

largest contributor. Sequences 1, 2, and 3 result in core melt with loss of all containment cooling (Plant State SE). As shown in Appendix 1 (Table A1-6), this essentially always leads to overpressure failure of containment (Release Category 2R) which results in an average of 2765 fatalities per melt. Therefore, the seismic risk at Zion is $(2765)(5.6 \times 10^{-6}) = 1.55 \times 10^{-2}$ fatalities per year.

Additional upgrading of the service water pumps would be very inexpensive and would result in a risk reduction (over an assumed 40-year plant lifetime) valued at $(40)(2,000,000)(1.55 \times 10^{-2}) = \1.2 million.

Soil failure beneath the containment building causes the building to move relative to the auxiliary building. Seismic restraints on pipes which enter the auxiliary building from containment prevent the pipes from accommodating the containment movement and the pipes fail. The Zion PRA shows that pipe restraints provide more than adequate seismic capability and removing some restraints in the auxiliary building might actually improve safety. The benefit of this would be at most, \$120,000 (assuming that Sequence 2 is 10% of the risk of melt), and probably less.

All other combined equipment contribute less than 1% of the probability of core melt and even less of the total risk (since some melts do not lead to containment failure). Therefore, the maximum possible value of any seismic improvements to Zion (other than service water pumps and containment/auxiliary building) is \$12,000.

Indian Point 3. The following is a summary of the sequences which are most likely to lead to core melt at Indian Point 3. The numbers in parentheses are the median estimate ground accelerations at which the equipment will fail. Note that it is estimated that there is only a 5% chance that an earthquake exceeding 0.8 g is physically possible at Indian Point. In all cases it is assumed that off-site power will be lost (0.2 g).

1. Failure of the diesel fuel tanks (1.14 g) result in loss of all AC power which lead to core melt and loss of containment.
2. Failure of the control building (1.20 g) results in loss of all AC power which leads to core melt and loss of containment.
3. Failure of batteries leads to loss of control and loss of power and then to melt and loss of containment. There are batteries in the control building and the diesel building and both must fail, either directly or due to falling walls, to cause melt. Median accelerations for various battery failures are in the range (1.07 g to 1.29 g).

4. Failure of the condensate tank (0.83 g) and city water tank (0.83 g) and refueling water tank (0.83 g) causes loss of feedwater and reactor makeup, and leads to melt. Containment remains functional.
5. Failure of the condensate tank (0.83 g) and city water tank (0.83 g) causes loss of feedwater. PORV (1.48 g) failure incapacitates feed and bleed cooling. The core melts, but containment remains functional.

These are the most significant sequences leading to core melt. However, there are a number of other sequences which taken together are responsible for a few percent of all melts (the exact amount could not be quantified from the information in the PRA). The mean probability of a seismic-induced core melt is 3.1×10^{-6} per year. About 80% of these melts result from Sequences 1, 2, and 3 (and similar ones) which result in containment failure, with an average fatality per release of 4,354 (Release Category 2RW--see Appendix 1). The remaining 20% of melts result primarily from Sequences 4 and 5 (and similar ones) where containment is functional and there is an average of 5.6 fatalities per release (Category 8A--see Appendix 1). Therefore, total risk is approximately $(3.1 \times 10^{-6}) [(0.8)(4354) + (0.2)(5.6)] = 0.01$ fatalities per year.

Using a 40-year plant lifetime and \$2 million per life, the value of eliminating this risk is $(40)(0.011)(2 \times 10^6) = \$880,000$. Approximately 90% of the risk could be eliminated by strengthening the diesel oil tanks, batteries and nearby walls, and control building. Such strengthening might be difficult at an existing plant, but would be relatively inexpensive at a new unit and would result in a risk reduction valued at about $(0.9)(\$880,000) = \$792,000$. Eliminating the remaining risk would require modifications to many components and structures and would result in a risk reduction valued at no more than \$88,000.

EQUIPMENT CONTRIBUTING NEGLIGIBLY TO SEISMIC RISK

Tables 8-1 and 8-2 show median ground acceleration capabilities for a number of structures and components which contribute negligibly to seismic-induced risk at Zion and Indian Point 3.

COST-BENEFIT ANALYSIS

Increases in seismic requirements in the past 15 years have played a major role in increasing nuclear plant costs. Because they permeate all facets of nuclear plant design, it was not possible within the scope of this study to estimate cost increases attributable strictly to seismic requirements. If the Zion and Indian Point risk

Table 8-1

MEDIAN GROUND ACCELERATION CAPACITIES OF COMPONENTS
HAVING NEGLIGIBLE CONTRIBUTION TO SEISMIC RISK AT ZION

<u>DESCRIPTION</u>	<u>MEDIAN GROUND ACCELERATION CAPACITY</u> ⁽¹⁾	<u>COMMENTS</u>
<u>System Components</u>		
Safety Injection Pumps	0.90 g	
Containment Ductwork/Dampers	0.97 g	
Containment Fan Coolers	1.74 g	
Piping	3.35 g	Minimum value for buried piping (containment spray/service water/condensate) is 1.40 g.
Pumps	3.77 g	With the exception of the safety injection and service water pumps.
Active/Passive MOVs	6.93 g	
Heat Exchangers	7.19 g	
<u>Electric Components</u>		
125V AC Distribution Panel	0.60 g	Failure due to relay chatter or breaker trip. Recoverable by manual reset. Permanent damage at double tabulated value.
480/4160V Switchgear	0.72 g	Same as above.
480V Motor Control Centers	0.72 g	Same as above.
4160V Diesel Generators	0.86 g	Same as above.
125V Inverters	2.69 g	
Cable Trays	4.74 g	
<u>Structures</u>		
Reactor/Auxiliary Building Impact	0.78 g	Lower bound value is 0.74 g
Auxiliary Building Roof	1.40 g	
Auxiliary Building Masonry Walls	1.70 g	
Containment Wall Failure	2.40 g	
Reactor Building Foundation Slab	7.30 g	

NOTES:

- (1) Numbers tabulated are for minimum median values of critical systems in plant event tree analysis as specified in Section 7 of the Zion Safety Study.

Table 8-2

MEDIAN GROUND ACCELERATION CAPACITIES OF COMPONENTS
HAVING NEGLIGIBLE CONTRIBUTION TO SEISMIC RISK AT INDIAN POINT 3

<u>DESCRIPTION</u>	<u>ACCELERATION CAPACITY (1) INDIAN POINT 3</u>	<u>COMMENTS</u>
<u>System Components</u>		
Safety Injection Pumps	2.17 g	
Containment Ductwork/Dampers	1.12 g	
Containment Fan Coolers	2.17 g	
Piping	1.84 g	Minimum value for buried piping (containment spray/service water/condensate) is 1.40 g.
Pumps		
Active/Passive MOVs	2.37 g	
Heat Exchangers	6.13 g	
<u>Electric Components</u>		
120V AC Distribution Panel	2.82 g	Failure due to relay chatter or breaker trip. Recoverable by manual reset. Permanent damage at three times tabulated value.
480V Switchgear	1.51 g	Same as above.
480V Motor Control Centers	1.17 g	Same as above.
480V AC Diesel/Gas Turbine Generators	1.30 g	Same as above.
120V AC Inverters	4.03 g	
Cable Trays	2.20 g	
<u>Structures</u>		
Primary Auxiliary Building Walls	0.33 g	
Auxiliary Feed Pump Building	1.70 g	
Fuel Storage Building	0.92 g	

NOTES:

- (1) Numbers tabulated are minimum median values for critical systems in plant event tree analysis as specified in Section 7 of the Indian Point Safety Study.

estimates are accepted, it is clear that the costs of increased seismic design and analysis far greatly exceed the benefits.

It might be argued that seismic risk estimates contain considerable uncertainty, and this actually true. However, the Zion/Indian Point PRAs contain substantial allowances for uncertainty and, in fact, all the calculated risk at each plant is due to uncertainty allowances while the "best estimate" is effectively that there is no seismic risk at all at Zion or Indian Point.

OBSERVATIONS BASED ON PRA RESULTS

The design basis earthquake for Zion was 0.17 g and that for Indian Point 2 and 3 was 0.15 g. The best estimate is that the weakest safety-related features at Zion and Indian Point (except the Indian Point 2 control building) will fail at 0.63 g and .83 g respectively, almost four times the design loadings and roughly equal to the largest earthquake considered possible at each site. If the analyses were based solely on best estimates (deterministic analysis), the risk from seismic events at Zion and Indian Point 3 would be essentially zero. However, the large uncertainties associated with earthquake magnitude and (especially) equipment capabilities lead to calculation of risks which are greater than other risks at the plants, although still lower than most societal risks.

There is a large variability (both within one plant and from plant-to-plant) in the seismic capabilities of safety-related components and structures. Most capabilities fall in the range of .6 to 2.5 g, although some are much higher. This suggests that the design approach which was used at these plants did not result in an optimum allocation of resources.

Large amounts of effort (both analysis and hardware) have been expended in the last decade on seismic design of safety-related piping. Piping at Zion and Indian Point did not have the benefit of most requirements now imposed by NRC. At Indian Point 2, the earliest of these plants (and one of the earliest modern PWRs), pipe supports were located using simple spacing rules without benefit of any of the dynamic analyses, detailed designs of individual hangers, etc., which characterize modern designs. Nonetheless, the best estimate seismic capabilities of safety-related piping (other than buried piping) at these units range from 1.84 g to 3.35 g and seismic-induced piping failures contribute essentially nothing to risk.

Pumps, valves, and electrical components at Zion and Indian Point were not subjected to the rigorous testing (multi-axis shake tests with realistic environmental and operating conditions) now required by NRC, but rather were subjected to simple single-axis shake tests. Nonetheless, the evaluation indicates that they contribute negligibly to risk.

PROPOSED FUTURE EFFORT

There is a great deal of research on seismic design underway in the U.S. and abroad. NRC has a major program to better quantify levels of conservatism in seismic design. Because of the large costs associated with seismic design, research on ways to reduce costs while maintaining adequate levels of safety could be very cost effective.

Section 9

COST-BENEFIT EVALUATION PROTECTION OF SAFETY-RELATED SYSTEMS

INTRODUCTION

As discussed in Section 7, the NRC requires that safety systems remain functional (with loss of redundancy permitted) following pipe breaks where those systems are not required to achieve safe shutdown.

A review of the Zion and Indian Point probabilistic risk analyses was made in order to quantify safety benefits resulting from the protection of safety-related systems from the effects of events which do not directly require those systems for safe shutdown. While these safety systems may not normally be required when other systems function, there is some probability that the normal systems will fail and the protected system will be needed. Thus, protection of normally nonrequired safety systems will result in some improvement in plant safety. Since failure of a steam line does not normally result in a breach of the reactor coolant boundary and the function of containment is to mitigate reactor coolant boundary failures, it was decided to investigate the benefits of protecting the containment building from steam line failures.

COST-BENEFIT ASSESSMENT

Risk Assessment

From the internal initiating event vectors (Φ^I) identified in the Zion and Indian Point PRAs, the mean probability of a steam break inside the containment is:

	<u>Frequency of Occurrence</u>
Zion 1 or 2	9.40×10^{-4}
Indian Point 2	1.95×10^{-3}
Indian Point 3	2.16×10^{-3}

Using the plant matrix (M) and summing the products of the Φ^I/M multiplication, the total mean frequency of plant damage states (involving core melt) due to a steam line break inside containment is:

	<u>Mean Frequency of Plant Damage State</u>
Zion 1 or 2	5.98×10^{-9}
Indian Point 2	1.99×10^{-7}
Indian Point 3	1.56×10^{-7}

Thus, the probability that a steam break will lead to a plant damage state is approximately one in twelve thousand at Indian Point and one in one hundred and sixty thousand at Zion. This difference is apparently due to the plant specific probability and effect of degraded power bus states at the two stations. There are various plant damage categories which may occur, having different conditional probabilities and risk impacts. From the results of Appendix 1, the risk (early and latent fatalities) associated with each damage state is quantified. The total risk due to the possibility of a steam break inside the containment is calculated to be:

	<u>Fatalities Per Year</u> *
Zion 1 or 2	8.08×10^{-8}
Indian Point 2	2.99×10^{-7}
Indian Point 3	2.26×10^{-7}

* Includes early and latent fatalities.

The contribution of steam line failures inside the containment to internally initiated risk and total risk is:

	<u>Steam Break Contribution to Risk</u>	
	<u>Internal</u>	<u>Total</u>
Zion 1 or 2	$8.13 \times 10^{-3}\%$	$4.90 \times 10^{-4}\%$
Indian Point 2	$4.48 \times 10^{-3}\%$	$2.11 \times 10^{-5}\%$
Indian Point 3	$4.52 \times 10^{-3}\%$	$7.90 \times 10^{-5}\%$

Consequently, not only is the risk from potential steam line failures very low, but also the contribution to risk relative to other initiating events. It is noted that the steam break inside containment analysis includes consideration of feedwater breaks as well as nonrupture events, such as failed relief or dump valves or excessive steam demands. Therefore, the risk tabulated above is conservative in quantifying the impact of steam pipe ruptures at the Zion and Indian Point plants.

The risk associated with a steam break inside the containment was reevaluated on the basis that the containment is not protected from the effects of a steam pipe rupture. The following assumptions were made for the analysis:

- o All steam breaks inside containment involve steam line ruptures (as opposed to failed relief valves, etc.).
- o A steam line rupture has a probability of one in failing containment isolation, though containment structural integrity remains intact.
- o Containment isolation failure occurs significantly before core melt.
- o The fan cooler and spray systems remain functional for controlling containment overpressurization. Localized failure to one cooler unit or spray train may occur and will make the systems ineffective in reducing the level of radionuclides escaping into the atmosphere.

With the above assumptions and review of Figure 9-1, which shows a typical plant event tree for a steam break at Zion and Indian Point, it is concluded that the total frequency of plant damage states remains essentially unchanged from the original analysis, although the frequency of individual damage categories will vary. There is a very small probability that primary and secondary cooling may fail, leading to a small loss of coolant accident (LOCA). Under these circumstances it is noted that the containment spray system will impact the availability of the recirculation cooling system as a source for continued injection cooling water after depletion of the refueling water storage tank and that this in turn will influence successful stabilization of the accident sequence. However, with the assumption of continued function of the spray system for containment cooling, the total frequency of the plant damage states remains unchanged. The typical plant arrangement which incorporates single failure criteria precludes the possibility of a single steam line rupture and consequent pipe whip from failing all the containment safeguard systems and makes this assumption a reasonable premise.

Figure 9-1

Typical Plant Event Tree For Steam Line Break Inside Containment At Zion And Indian Point

The very conservative assumption that possible failure of one of the containment safeguard trains prevents effective scrubbing of airborne radionuclides and the assumed prior failure of containment isolation essentially leads to a containment bypass condition without overpressurization. From the Zion and Indian Point containment analyses, this type of sequence was assigned to Release Category 2. Appendix 1 calculated the average number of early and latent fatalities resulting from a Category 2 release at the units as:

	<u>Average Fatalities Per Category 2 Release</u>
Zion 1 or 2	2975
Indian Point 2 or 3	3911

At Zion, a Category 2 release represents the greatest risk among the release categories identified. However, at Indian Point, it is approximately 11% less severe than Release Category 2RW which results from a dried-out debris bed, no containment sprays, and late overpressurization of the containment. Release Category 2RW is not applicable to the accident sequence under consideration due to the initial presence of water and the nondispersive (nonexplosive) manner of the release.

Consequently, the total risk associated with a steam line rupture which fails the containment is:

	<u>Fatalities Per Year</u> [*]
Zion 1 or 2	1.78×10^{-5}
Indian Point 2	7.78×10^{-4}
Indian Point 3	6.10×10^{-4}

* Includes early and latent fatalities.

Comparison with the risk for the original analyses gives the following results:

	Risk Ratio *		
	<u>Accident Event</u>	<u>Internal Events</u>	<u>All Events</u>
Zion 1 or 2	220	1.02	1.001
Indian Point 2	2600	1.12	1.001
Indian Point 3	2700	1.12	1.002

* Ratio of risk (assuming containment failure) divided by risk from original analyses.

With the above example, the relative risk from the accident is substantially increased by not protecting a safety-related system (containment) from the effects of an accident (steam break inside containment) which do not directly require the safety system. In spite of this and the very conservative assumptions leading to this result, the maximum effect is only a modest increase (12%) in risk due to internal events at the units and a very small increase (.20%) to total risk.

One reason the relative increase in internally initiated risk at Zion and Indian Point is moderate is that the total risk is very low. Also, in the Zion and Indian Point Probabilistic Safety Studies, it was seen that core melts generally contribute to risk only when there is a loss of containment function. Since it was conservatively assumed that a steam line failure will cause a total loss of containment isolation, it will contribute somewhat to risk, even though the probability of its occurrence is low.

Cost-Benefit Evaluation

From Section 3, the value of a life and the attendant injuries and property damage is estimated to be \$2 million. Assuming a 40-year plant life, the value of protecting containment from the effects of a steam line break is:

Zion 1 or 2

$$(1.78 \times 10^{-5} - 8.08 \times 10^{-8} \text{ fatalities/yr})(40 \text{ yr})(\$2,000,000/\text{fatality}) = \$1,417$$

Indian Point 2

$$(7.78 \times 10^{-4} - 2.99 \times 10^{-7} \text{ fatalities/yr})(40 \text{ yr})(\$2,000,000/\text{fatality}) = \$62,200$$

Indian Point 3

$$(6.10 \times 10^{-4} - 2.26 \times 10^{-7} \text{ fatalities/yr})(40 \text{ yr})\$2,000,000/\text{fatality} = \$48,800$$

Thus, the benefits of protecting containment vary significantly and are plant specific.

There are many factors which influence layout and design inside containment. Protection from the effects of a steam line break is required not only for containment, but also for the reactor coolant system, feedwater lines, and other safety systems. Within the scope of this study, it was not possible to isolate the costs specifically attributable to the protection of containment. Clearly, the cost of protecting containment will exceed the benefits, at least for plants such as Zion. It should be noted that containment is considered to be very effective in reducing risks in the Zion/Indian Point PRAs. The benefits of protecting other safety systems (such as low pressure injection) from steamline breaks would undoubtedly be much smaller than the benefits of protecting containment.

PROPOSED FUTURE EFFORTS

Significant studies on pipe break effects are in progress. Prior emphasis has been on protection from pipe whip and jet impingement. Protection methods have included numerous and massive shield barriers. This has a significant effect on direct costs and can impact the construction schedule and later, plant maintenance. While some pipe break protection does not appear to be cost effective, as discussed in this section, protection of safety systems from the effects of accidents is generally desirable. As with tornados, it appears that the major problem is not with the requirement of protection, but with the excessive conservatism in the implementation of that requirement.

Programs to reduce the conservatisms currently required in designing to protect against the effects of pipe rupture are underway. A report prepared for EPRI by TVA (Reference 31) summarizes current design practices in such areas as postulated pipe break opening time, jet characteristics, structural response of the pipe, impact of the pipe on structures, and pipe whip restraints. The impact of current practices on plant design is discussed in a qualitative nature along with suggestions regarding further evaluation and study. EPRI has several programs underway to provide the data/tools necessary to reduce some of the present conservatism. Reference 32 covers the status

of pipe break analysis and testing; References 32 and 34 are additional related EPRI reports. Programs are continuing in this area at EPRI and elsewhere. For example, there is an experimental program on pipe whip which is underway in France, and impingement is being addressed in a five-party program now in progress (NRC is a participant).

It appears that programs are in progress to quantify the consequences of pipe whip and jet impingement and determine where protection is not necessary, as discussed in ANSI/ANS 58.3 (Physical Protection for System and Components Important to Safety), and where a reasonable level of protection is appropriate. A detailed study to provide a containment layout based on reasonable levels of protection could be very useful. The costs of excessive protection could then be quantified and specific phenomenological work initiated if they are required and are shown to likely lead to significant cost reductions.

Section 10

COST-BENEFIT EVALUATION NRC REGULATORY GUIDE 1.26 IMPACT ON THE EMERGENCY COOLING INJECTION SYSTEM

INTRODUCTION

NRC Regulatory Guide 1.26 defines the quality group classifications and appropriate standards for water, steam, and radioactive-waste containing components of nuclear power plants. The guide was intended to compliment the requirements of 10 CFR 50 for reactor coolant pressure boundary equipment and provide specific design, fabrication, erection, and quality assurance criteria for the remaining safety-related components. This effectively establishes detailed safety criteria without consideration of plant specific design features and imposes corresponding quality assurance levels and deterministic design and fabrication criteria without consideration of the overall impact on plant safety. Limited nuclear operating experience prevented detailed evaluation of safety equipment performance and led to such deterministic practices. However, the current accumulation of operating data and further developments in risk analysis suggest a more comprehensive and cost effective approach to achieve safety goals.

Using the results of WASH 1400 for PWR plants, an analysis is performed to evaluate the cost benefit of classifying the Emergency Cooling Injection System (ECIS) pumps as ASME Section III, Class 2 components, as directed in Regulatory Guide 1.26, rather than using 'standard' industrial pumps (e.g., ASME Section VIII). Relaxing the specific requirements of Regulatory Guide 1.26, a subsequent analysis reviewed various design options using standard commercial industrial pumps and evaluating the impact on overall safety.

TECHNICAL CONSIDERATIONS

For this analysis, it was desired to identify a system which appears in key potential accident sequences but which does not dominate plant risk. In such systems, changes in risk due to system modifications are large enough to quantify but not so large that they substantially alter the overall risk. Table 10-1 summarizes key accident sequences as identified in WASH 1400 for PWR plants. Review of Table 10-1 and consideration of the above criteria led to selection of the Emergency Coolant Injection System.

Table 10-1
KEY PWR ACCIDENT SEQUENCES AS IDENTIFIED
IN WASH 1400

	RELEASE CATEGORIES						Core Melt		
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-g 1x10 ⁻¹¹ AF-g 1x10 ⁻¹⁰ ACD-g 5x10 ⁻¹¹ AG-g 9x10 ⁻¹¹	AB-y 1x10 ⁻¹⁰ AB-d 4x10 ⁻¹¹ AHF-y 2x10 ⁻¹¹	AD-g 2x10 ⁻⁸ AH-g 1x10 ⁻⁸ AF-d 1x10 ⁻⁸ AG-d 9x10 ⁻⁹	ACD-B 1x10 ⁻¹¹	AD-B 4x10 ⁻⁹ AH-B 3x10 ⁻⁹	AB-C 1x10 ⁻⁹ AHF-C 1x10 ⁻¹⁰ ADP-C 2x10 ⁻¹⁰	AD-C 2x10 ⁻⁶ AH-C 1x10 ⁻⁶	A-B 2x10 ⁻⁷	A 1x10 ⁻⁴
A Probabilities	2x10 ⁻⁹	1x10 ⁻⁸	1x10 ⁻⁷	1x10 ⁻⁸	4x10 ⁻⁸	3x10 ⁻⁷	3x10 ⁻⁶	1x10 ⁻⁵	1x10 ⁻⁴
SMALL LOCA S ₁	S ₁ B-g 3x10 ⁻¹¹ S ₁ CD-g 7x10 ⁻¹¹ S ₁ F-g 3x10 ⁻¹⁰ S ₁ C-g 3x10 ⁻¹⁰	S ₁ B-y 4x10 ⁻¹⁰ S ₁ B-d 1x10 ⁻¹⁰ S ₁ HF-y 6x10 ⁻¹¹	S ₁ D-g 3x10 ⁻⁸ S ₁ H-g 3x10 ⁻⁸ S ₁ F-d 3x10 ⁻⁸ S ₁ G-d 3x10 ⁻⁸	S ₁ CD-B 1x10 ⁻¹¹	S ₁ H-B 5x10 ⁻⁹ S ₁ D-B 6x10 ⁻⁹	S ₁ DF-C 3x10 ⁻¹⁰ S ₁ B-C 2x10 ⁻⁹ S ₁ HF-C 4x10 ⁻¹⁰	S ₁ D-C 3x10 ⁻⁶ S ₁ H-C 3x10 ⁻⁶	S ₁ -B 6x10 ⁻⁷	S ₁ 1x10 ⁻⁴
S ₁ Probabilities	3x10 ⁻⁹	2x10 ⁻⁸	2x10 ⁻⁷	3x10 ⁻⁸	8x10 ⁻⁸	6x10 ⁻⁷	6x10 ⁻⁶	3x10 ⁻⁵	3x10 ⁻⁴
SMALL LOCA S ₂	S ₂ B-g 1x10 ⁻¹⁰ S ₂ F-g 1x10 ⁻⁹ S ₂ CD-g 2x10 ⁻¹⁰ S ₂ C-g 9x10 ⁻¹⁰ S ₂ C-g 2x10 ⁻⁸	S ₂ B-y 1x10 ⁻⁹ S ₂ HF-y 2x10 ⁻¹⁰ S ₂ B-d 4x10 ⁻¹⁰	S ₂ D-g 9x10 ⁻⁸ S ₂ H-g 6x10 ⁻⁸ S ₂ F-d 1x10 ⁻⁷ S ₂ C-d 2x10 ⁻⁶ S ₂ G-d 9x10 ⁻⁶	S ₂ DC-B 1x10 ⁻¹²	S ₂ D-B 2x10 ⁻⁸ S ₂ H-B 1x10 ⁻⁸	S ₂ B-C 8x10 ⁻⁹ S ₂ CD-C 2x10 ⁻⁸ S ₂ HF-C 1x10 ⁻⁹	S ₂ D-C 9x10 ⁻⁶ S ₂ H-C 6x10 ⁻⁶		
S ₂ Probabilities	1x10 ⁻⁷	3x10 ⁻⁷	3x10 ⁻⁶	2x10 ⁻⁷	3x10 ⁻⁷	2x10 ⁻⁶	2x10 ⁻⁵		
REACTOR VESSEL RUPTURE - R	RC-g 2x10 ⁻¹²	RC-y 3x10 ⁻¹¹ RF-d 1x10 ⁻¹¹ RC-d 1x10 ⁻¹²	R-g 1x10 ⁻⁹				R-C 1x10 ⁻⁷		
R Probabilities	2x10 ⁻¹¹	1x10 ⁻¹⁰	1x10 ⁻⁹	2x10 ⁻¹⁰	1x10 ⁻⁹	1x10 ⁻⁸	1x10 ⁻⁷		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4x10 ⁻⁶							
V Probabilities	4x10 ⁻⁷	4x10 ⁻⁶	4x10 ⁻⁷	4x10 ⁻⁸					
TRANSIENT EVENT - T	TKLB'-g 3x10 ⁻⁸	TKLB'-y 7x10 ⁻⁷ TKLB'-d 2x10 ⁻⁶	TKL-g 6x10 ⁻⁸ TKQ-g 3x10 ⁻⁸ TKHQ-g 1x10 ⁻⁸		TKL-B 3x10 ⁻¹⁰ TKQ-B 3x10 ⁻¹⁰	TKLB'-C 6x10 ⁻⁷	TKL-C 6x10 ⁻⁶ TKQ-C 3x10 ⁻⁶ TKHQ-C 1x10 ⁻⁶		
T Probabilities	3x10 ⁻⁷	3x10 ⁻⁶	4x10 ⁻⁷	7x10 ⁻⁸	2x10 ⁻⁷	2x10 ⁻⁶	1x10 ⁻⁵		
(I) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9x10 ⁻⁷	8x10 ⁻⁶	4x10 ⁻⁶	5x10 ⁻⁷	7x10 ⁻⁷	6x10 ⁻⁶	4x10 ⁻⁵	4x10 ⁻⁵	4x10 ⁻⁴
LOWER BOUND (5% VALUE)	9x10 ⁻⁸	8x10 ⁻⁷	6x10 ⁻⁷	9x10 ⁻⁸	2x10 ⁻⁷	2x10 ⁻⁶	1x10 ⁻⁵	4x10 ⁻⁶	4x10 ⁻⁵
UPPER BOUND (95% VALUE)	9x10 ⁻⁶	8x10 ⁻⁵	4x10 ⁻⁵	5x10 ⁻⁶	4x10 ⁻⁶	2x10 ⁻⁵	2x10 ⁻⁴	6x10 ⁻⁴	4x10 ⁻³

Notes: The probabilities for each release category for each event tree and the I for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability (See Section 4.1).

KEY PWR ACCIDENT SEQUENCES AS IDENTIFIED IN WASH-1400

Accident Sequence Symbols:

- A - Intermediate to large LOCA.
- B - Failure of electric power to ESFs.
- B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.
- C - Failure of the containment spray injection system.
- D - Failure of the emergency core cooling injection system.
- F - Failure of the containment spray recirculation system.
- G - Failure of the containment heat removal system.
- H - Failure of the emergency core cooling recirculation system.
- K - Failure of the reactor protection system.
- L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.
- M - Failure of the secondary system steam relief valves and the power conversion system.
- Q - Failure of the primary system safety relief valves to reclose after opening.
- R - Massive rupture of the reactor vessel.
- S₁ - A small LOCA with an equivalent diameter of about 2 to 6 inches.
- S₂ - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.
- T - Transient event.
- V - LPIS check valve failure.
- α - Containment rupture due to a reactor vessel steam explosion.
- β - Containment failure resulting from inadequate isolation of containment openings and penetrations.
- γ - Containment failure due to hydrogen burning.
- δ - Containment failure due to overpressure.
- ϵ - Containment vessel melt-through.

The WASH 1400 analysis is based upon the Surry (788 MW) design with three RCS loops. The Emergency Coolant Injection System is comprised of four subsystems: the accumulator system, the low pressure injection system, the high pressure injection system, and the safety injection control system. Given a LOCA event, the safety injection control system signals the accumulator isolation valves to open in case one is closed and initiates low pressure or high pressure injection, as appropriate. The low pressure injection system has two redundant trains with a common suction and discharge header to the RCS cold legs. Stated pump capacity is 3,000 gpm at a 225 foot head. The high pressure injection system utilizes the three high pressure charging pumps. Each pump has a stated flow capacity of 150 gpm at full head. The pumps are arranged in parallel with common suction and discharge headers.

Success of the Emergency Coolant Injection System was defined as:

- o Discharge of 2 of 3 accumulators in the 2 intact cooling loops and discharge from 1 of 2 low pressure injection pumps for a large LOCA (A).
- o Discharge of 2 of 3 accumulators in the 2 intact cooling loops and discharge from 1 of 3 high pressure injection pumps for a small LOCA (S_1). (LOCA assumed to occur in RCS cold leg)
- o Discharge from 1 of 3 high pressure injection pumps for a very small LOCA (S_2).

COST-BENEFIT ASSESSMENT

Background Considerations

Section 2 discusses the growth of regulatory requirements on the nuclear power industry and the cost implications on plant construction. Regulatory Guide 1.26 was first issued in 1972, subsequent to the design of Surry. It specifies design, fabrication, erection, and quality assurance requirements for general equipment categories. Conformance with the regulatory guide requires that the ECIS pumps be manufactured as ASME Section III, Class 2 components. Prior to this, typical engineering practice was to specify ASME Section VIII or USAS B31.1 for design and inspection requirements and Section IX for welding of pressure retaining parts as done at Surry.

Equipment costs associated with implementation of ASME Section III requirements can be significantly greater than those conforming to more commonly used industry codes (e.g., ASME Section VIII). Impact on overall plant cost is moderate. Even with consideration of seismic criteria, manufacturers have indicated a substantial cost

savings using seismically qualified Section VIII pumps (as used for selected oil drilling applications) over Section III pumps. It should be noted that as application and performance needs become more specialized and motor brake horsepower increases, the impact of code requirements become less dramatic since these other factors have essentially already made the component a specialty item.

Risk Assessment

Quantification of the cost-benefits associated with manufacturing the ECIS pumps to ASME Section III specifications requires evaluation of their impact on pump reliability (e.g., failure modes). To do this, pump failure data was first reviewed. Subsequently, the code requirements are discussed and judgment was made regarding their corresponding impact on reported failures and pump unavailability.

Section 3.3.f. in Appendix III of WASH 1400 summarizes specific safety related pump failures occurring during a one-year period. A total of 24 pump failures of varying severity were reported at 9 of the 17 plants considered in the study. Classification of the pump failures into general categories results in the following breakdown:

<u>Pump Failure Category</u>	<u>Percent of Failures</u>
1. Seal Failures	17%
2. Control and Electrical Failures (e.g., circuit breakers, wiring, relay failures)	42%
3. Mechanical (nonpressure boundary) failures (e.g., lubrication, bearings, lack of performance)	37%
4. Human Error	4%

In order to verify the limited component failure rate data for nuclear power plants when WASH 1400 was prepared, the specific nuclear experience was compared with other industry experience. This other industry experience was derived from various reports and handbooks including United Kingdom System Reliability Studies, nuclear and fossil applications, military and NASA component failure data, and chemical industry operating experience. Using a log normal distribution to describe failure data variability and a subjective weighting factor to address diversity in component application and operating conditions, assessed component failure ranges were established. They were defined for a 90% probability range with the lower and upper bounds specified at the 5% and 95% confidence levels, respectively. Applicable values for

pump unavailability are tabulated below:

	<u>WASH 1400 Nuclear Experience</u>	<u>Industry Minimum Value</u>	<u>Assessed Lower Bound</u>	<u>Assessed Upper Bound</u>	<u>Industry Maximum Value</u>
Pump Fails to Start Per Demand	1.0-3	5.0-5	3.0-4	3.0-3	5.0-3
Pump Fails to Run Per Hour (given start)	3.0-6	1.0-7	3.0-6	3.0-4	1.0-4

It is seen that the specific pump experience at nuclear power plants falls within the industry extremes as well as the assessed ranges. Thus, it can be concluded that the nuclear and industry data are consistent, with no gross differences. Due to the qualitative nature of some of the data and lack of detail presented in Appendix III of WASH 1400 regarding industry component failure rates, little can be said about where the nuclear experience falls within the lower and upper industry bounds.

ASME Section III addresses the material, design, fabrication, examination, testing, and documentation requirements for pump components. Specific design requirements include the pump casing, inlet/outlet connections, auxiliary nozzles, pump covers, seal housings, internal heat exchangers, and mounting. Shafts, impellers, and seal packages are exempted. Thus, the code essentially addresses only the structural/pressure boundary integrity of the pump.

Consideration of the requirements of ASME Section III and review of the reported pump failure experience (failure modes and frequency) suggest that the subject code should have marginal, if any, impact on failure occurrence as compared with standard industry practice (e.g., ASME Section VIII). Using a conservative approach, it was judged that the maximum benefits resulting from specification of ASME Section III would be to reduce pump failure rates to the industry minimum values reported in WASH 1400. These are one to two orders of magnitude better than the reported nuclear experience.

Using the industry minimum failure rate values for pump performance, an evaluation was performed to determine the cost-benefit of classifying the emergency cooling injection pumps as ASME Section III components. As a further conservatism, mean

rather than median pump failure probabilities are used for the non-ASME Section III comparison risk. This follows the current trend in PRA methodology, which considers mean probabilities more representative than median values, and essentially increases the risk as presented in WASH 1400. Thus, pump failure probabilities used in the analysis were:

	Qualified ⁽¹⁾ Commercial Pump	ASME Section III Pump
Pump Fails to Start Per Demand	1.2-3	5.0-5
Pump Fails to Run Per Hour (given start)	8.0-5	1.0-7

-
- (1) Pump failure probabilities correspond to the mean values for nuclear experience (normal environments) as calculated from Table III 4-1 of WASH 1400.

Table 10-1 identifies 18 key PWR accident sequences involving the emergency core cooling injection system. Event frequencies and system unavailabilities involved in these accident sequences are summarized in Table 10-2. Using mean values the frequency and corresponding risk for the 18 accident sequences were calculated as shown in Table 10-3. It is noted that a common mode failure was identified for large LOCA (A) events and Emergency Cooling Injection System Unavailability (D), which increases the frequency of all affected accident sequences. This results from a pipe rupture (large LOCA) on the discharge side of one of the reactor coolant pumps, causing pump overspeed, possible flywheel fracture, and possible failure of the low pressure injection line from the resulting missiles.

The mean risk for WASH 1400 PWR accident sequences involving failure of the emergency cooling injection system totals 3.88×10^{-4} fatalities per year. This reflects the risk associated with ECIS pumps qualified to 'commercial' standards (ASME Section VIII and IX for inspection and welding requirements) as done at Surry.

Emergency coolant injection system unavailability was recalculated using the reduced fault trees presented in Appendix II of WASH 1400 and the estimated ASME Section III, Class 2, pump failure probabilities summarized above. It is noted that successful operation of the charging pumps for high pressure injection requires adequate cooling

Table 10-2

EVENT FREQUENCIES AND SYSTEM UNAVAILABILITIES (1)

Symbol	Description	Lower Bound (5% Value)	Median (50% Value)	Upper Bound (95% Value)	Mean
A	Large LOCA	1.0-5/yr	1.0-4/yr	1.0-3/yr	2.7-4/yr(2)
S ₁	Small LOCA	3.0-5/yr	3.0-4/yr	3.0-3/yr	8.0-4/yr(2)
S ₂	Very Small LOCA	1.0-4/yr	1.0-3/yr	1.0-2/yr	2.7-3/yr(2)
C	Containment Spray Injection	1.0-3	2.4-3	7.8-3	3.4-3 (2)
D	Emergency Coolant Injection				
	Large LOCA (A)				5.94-3 (3)
	Small LOCA (S ₁)				8.93-3 (3)
	Very Small LOCA (S ₂)				7.97-3 (3)
F	Containment Spray Recirculation	2.5-5	1.0-4	9.0-4	2.1-4 (2)
G	Containment Heat Removal	3.0-5	8.5-5	3.0-4	1.2-4 (2)
α	Containment Rupture Due to vessel steam explosion				1.0-2 (4)
β	Containment Leakage				2.0-3 (4)
ε	Containment Rupture By Melthrough				9.9-1 (4)

NOTES:

1. Numbers are in abbreviated scientific notation (e.g., 1.0-5 equals 1.0×10^{-5})
2. Mean values are calculated assuming log normal probability distributions and using the upper and lower bounds as specified in Table II 1-4 and Table III 6-9 of WASH 1400.
3. Mean emergency coolant injection unavailability values were determined by calculating mean event probabilities and substituting into the reduced fault trees presented in Appendix II of WASH-1400. This results in mean estimates which are somewhat different from the values obtained using the method explained in Note 2 and is apparently due to the log normal distribution approximation and specification of individual event, unavailability error factors based on the extreme bound.
4. Point estimate values for accident sequences of interest.

Table 10-3

MEAN RISK CONTRIBUTION OF WASH 1400 PWR ACCIDENT SEQUENCES INVOLVING FAILURE
OF THE EMERGENCY COOLING INJECTION SYSTEM

<u>Accident Sequence(1)</u>	<u>Mean Accident (Release) Frequency Per Year (2)</u>	<u>Release Category</u>	<u>Fatalities Per Release</u>	<u>Mean Fatalities Per Year (2)</u>
ADC- α (3)	1.73-10	1	2000	3.45-7
S ₁ CD- α	2.43-10	1	2000	4.86-7
S ₂ CD- α	7.34-10	1	2000	1.47-6
AD- α (3)	5.08-8	3	1100	5.59-5
S ₁ D- α	7.15-8	3	1100	7.86-5
S ₂ D- α	2.15-7	3	1100	2.37-4
ACD- β (3)	3.46-11	4	190	6.57-9
S ₁ CD- β	4.84-11	4	190	9.20-9
S ₂ DG- β	5.16-12	4	190	9.81-10
AD- β (3)	1.01-8	5	48	4.87-7
S ₁ D- β	1.43-8	5	48	6.86-7
S ₂ D- β	4.30-8	5	48	2.07-6
ADF- ϵ (3)	1.05-9	6	11	1.16-8
S ₁ DF- ϵ	1.49-9	6	11	1.64-8
S ₂ CD- ϵ	7.25-8	6	11	7.97-7
AD- ϵ (3)	5.03-6	7	.3	1.51-6
S ₁ D- ϵ	7.09-6	7	.3	2.13-6
S ₂ D- ϵ	2.13-5	7	.3	6.40-6
Total Risk/Year				3.88-4

NOTES:

- See Table 10-2 for accident sequence symbols.
- Numbers are in abbreviated scientific notation (e.g., 1.73-10 equals 1.73×10^{-10}).
- There is a common mode failure contributing to the probability of A-D events, such that:
 $Q(AD) = P(A) \times P(D) + Q(\text{Common Mode})$
 $Q(\text{Common Mode}) \approx 3.47-6$ (Mean)

for the pump seals and lubrication oil. At Surry, this is provided by two charging pump cooling water pumps and two charging pump service water pumps. Double failures in either set of pumps will fail the charging pumps. Consequently, to be consistent with Regulatory Guide 1.26, this analysis assumes that both sets of support pumps are also fabricated in accordance with ASME Section III, Class 2, requirements with the corresponding failure probabilities given on page 10-7.

The results of the analysis are presented in Table 10-4. Use of ASME Section III, Class 2, pumps is estimated to have a negligible improvement on ECIS unavailability varying from 0.48% to 0.11% (depending on the size of the LOCA event). This corresponds to a reduction in the annual risk of 4.43×10^{-7} fatalities/year or 0.11% of the risk of the subject accident sequences.

Cost-Benefit Evaluation

From Section 3, the value of a life and the attendant injuries and property damage is estimated to be \$2 million. Assuming a 40-year plant life, the value of fabricating three charging pumps, two low pressure injection pumps, two charging pump cooling water pumps, and two charging pump service water pumps in accordance with ASME Section III, Class 2, requirements in lieu of 'commercial' standards is:

$$(4.43 \times 10^{-7} \text{ fatalities/year})(40 \text{ years})(\$2 \text{ million/fatality}) = \$35.44$$

The conclusions are obvious. This comes as no surprise using PRA techniques since unavailability of the emergency cooling injection system is dominated by nonpump related faults. High pump reliability is maintained by redundancy and independence. For low pressure injection, single valve failures and system unavailability due to testing and maintenance are significant compared with double pump failures. Similarly, for high pressure injection, single and double valve failures and double failures associated with maintaining the boric acid injection solution (negative reactivity) at an elevated temperature overshadow charging pump triple failures. Also, use of the charging pumps for high pressure injection further increases the probability of system success. With one normally operating charging pump, high pressure injection does not rely solely on the startup of standby pumps which have a significantly higher probability of failure (approximate 15 times greater). This advantage diminishes for accident sequences involving a loss of power which requires restart of all pumps on emergency power.

Table 10-4

PROJECTED IMPACT OF USING ASME SECTION III, CLASS 2, PUMPS IN THE EMERGENCY
COOLANT INJECTION SYSTEM

Emergency Coolant Injection System Unavailability:

	<u>ECIS Unavailability Using Qualified Commercial Pumps</u>	<u>Reduction in ECIS Unavailability Using Section III Pumps</u>	<u>Percent Improvement in ECIS Unavailability Using Section III Pumps</u>
Large LOCA (A)	5.94-3	2.86-5	.48%
Small LOCA (S ₁)	8.93-3	8.83-6	.10%
Very Small LOCA (S ₂)	7.97-3	8.83-6	.11%

Risk Contribution of WASH-1400 PWR accident sequences involving failure of the Emergency Cooling Injection System:

	<u>Fatalities Using Qualified Commercial Pumps</u>	<u>Reduction in Fatalities Using Section III Pumps</u>	<u>Percent Improvement In Risk Using Section III Pumps</u>
Risk/Year (2)	3.88-4	4.43-7	.11%

NOTES:

1. Numbers are in abbreviation scientific notation (e.g., 5.94-3 equals 5.94×10^{-3})
2. The risk calculated is for those accident sequences identified in Table 10-3.

The inherent reliability of the subject ECIS pump configuration is further realized with the very conservative assumption that specification of ASME Section III requirements eliminates all direct pump failures (but allowing indirect faults such as motor overcurrent trips). The resulting improvement in risk is essentially negligible and totals 4.49×10^{-7} fatalities per year. This has a corresponding cost benefit of only:

$$(4.49 \times 10^{-7} \text{ fatalities/year})(40 \text{ years})(\$2 \text{ million/fatality}) = \$35.92$$

Thus, sole reliance on deterministic criteria without consideration of the specific system design can lead to unnecessary, and often excessive, capital expenditures.

Relaxing the current requirements of Reg. Guide 1.26 and considering alternative ECIS design options which would be more cost effective, it should be kept in mind that the system is highly reliable. If design changes were implemented which could make the emergency coolant injection system 100% available, the maximum cost benefit over the existing design during the life of the plant would be:

$$(3.88 \times 10^{-4} \text{ fatalities/year})(40 \text{ years})(\$2 \text{ million/fatality}) = \$31,040$$

Thus, individual system improvements would have a maximum, potential cost benefit in the range of \$2,500 to \$10,000. While this is not significant, it would be substantially greater (approximately 70 to 280 times) than the benefits derived from ASME Section III pump requirements.

Approximately 85% of the risk from WASH 1400 PWR accident sequences associated with failure of the ECIS involves failure of high pressure injection. Figure 10-1 is a simplified process flow diagram for the system. Double failures dominate system unavailability and are listed below:

	Percentage Contribution to HPIS Unavailability
Failure Description	
o Failure of redundant boron injection tank (BIT) inlet motor operated valves (MOV)	14%
o Failure of redundant BIT outlet MOVs	14%
o Failure of both BIT heaters	24%
o Failure of both sets of BIT piping heat tracing elements	23%

10-13

10-13

Attention was focused on the BIT inlet and outlet valves since their combined contribution to system unavailability is 28% and results primarily from infrequent (yearly) verification of valve operability. These valves are not easily tested without impact on normal plant operation. The addition of manual isolation valves to permit more frequent testing of the MOVs during normal operation was evaluated; however, this was found to be detrimental because of the high probability for human errors of omission (manual valves left closed after testing). Administrative control features in conjunction with control room indication and annunciation for manual valve position would essentially eliminate this concern or an additional (normally open) motor-operated valve could be placed upstream of MOV 1867 A and B and downstream of MOV 1867 C and D for testing purposes. As an alternative, surveillance frequencies could be specified more often without hardware modifications; though this could possibly require a plant shutdown if no planned or unplanned outages occurred during the required test interval.

It was concluded that more frequent surveillance of the BIT inlet and outlet valves could be accomplished with modifications which would not significantly contribute to additional system unavailability. It was further determined that quarterly inspection of the valves resulted in both a practical and optimum test interval. A comparison of valve unavailability due to hardware failures for yearly and quarterly surveillance testing is listed below:

<u>Significant Valve Hardware Failures</u>	<u>Yearly Inspection</u>	<u>Quarterly Inspection</u>
Failure of any 1 of 7 electrical contacts	1.16-2	2.9-3
Failure of either of 2 overload contacts	1.08-2	2.7-3
Valve operator fails on demand	<u>1.2-3</u>	<u>1.2-3</u>
MOV Hardware Unavailability	2.36-2	6.8-3

Fault tree analysis identified two other significant contributors to valve failure. Total valve unavailability is summarized below:

<u>Fault Description</u>	<u>Yearly Inspection</u>	<u>Quarterly Inspection</u>
Safety injection control system contact in MOV fails to close	1.6-3	1.6-3
MOV hardware unavailability	2.36-2	6.8-3
Safety injection control system train fails	<u>7.1-3</u>	<u>7.1-3</u>
Total Mean MOV Unavailability	3.23-2	1.55-2

Thus, the improvement in HPIS unavailability with quarterly surveillance of the BIT inlet and outlet valves becomes:

$$2((3.23-2)^2 - (1.55-2)^2) = 1.61-3$$

System unavailability due to increased surveillance testing is negligible. Consequently, the resulting reduction in risk is 6.48-5 fatalities per year. The corresponding cost benefit is:

$$(6.48-5 \text{ fatalities/yr})(40 \text{ years})(\$2 \text{ million/fatality}) = \$5184$$

which is 144 or more times greater than those using ASME Section III pump requirements.

The NRC requires a boron injection tank (BIT) to prevent criticality following a main steam line break. While a detailed evaluation was not possible in the time available, it appears that the decrease in risk due to the BIT is very small. Analyses (such as in the Indian Point 2 FSAR) indicate that a steam line break without a BIT results in only minor fuel damage, even in the worst case. It is possible that a BIT reduces risk in ways which are not obvious from existing PRAs (such as reducing core damage from a steam line break combined with multiple MSIV failures), but this risk reduction is likely to be very small. On the other hand, elimination of the NRC requirement for the BIT would eliminate approximately 75% of the high pressure injection system unavailability and would consequently reduce the probability of core melt and would reduce public risk by 2.38×10^{-4} fatalities per year. The value of this reduction would be approximately \$19,040 over the 40-year life of a plant.

CONCLUSIONS

The analysis shows that if all ECIS pump failures (a total of nine pumps) could be eliminated, the value of the reduction in risk would be \$35.92. In the limited data considered in WASH 1400, none of the pump failures were of a type, such as casing cracks, which might be eliminated with the use of ASME Section III. A cursory review of license event reports also indicates that only a very small fraction of safety pump failures are related to pressure retaining parts (other than seals and fittings). Therefore, it is probable that the actual value of implementing ASME Section III requirements for ECIS pumps in plants with designs similar to the WASH 1400 PWR (Surry) is less than one dollar! Costs to implement ASME Section III requirements were not estimated, but are obviously many thousands of times greater than the benefits.

If it were desired to reduced ECIS unavailability, and hence to reduce risk, there are many things which could be done (but are not required or are contrary to current requirements) which would be much more cost effective. Adding two normally open motor-operated valves (and associated controls and instrumentation) to permit more frequent surveillance testing would almost certainly not cost as much as ASME Section III pump requirements and would improve high pressure safety injection system reliability (and therefore reduce risk) by at least a factor of 144 times as much. While a more detailed evaluation would be required, it appears that elimination of the requirement for a boron injection tank would simultaneously reduce costs (by eliminating the tank and associated piping, valves, recirculation pumps and tanks, instrumentation, and space) as well as reduce risk. The risk reduction might be as much as 530 times as great as the reduction which could be achieved by eliminating all ECIS pump failures.

PROPOSED FUTURE EFFORTS

The foregoing discussion leads to two general areas for further research. The first would involve evaluation of active component failure rates versus code qualifications. Actual equipment failure rates are frequent enough to permit compilation of statistically significant data. Existing data bases such as SMSC's Operating Plant Evaluation Code (OPEC) are available for this purpose. Thus, components qualified to typical industry standards for early nuclear power plants could be compared with equipment qualified to more stringent code requirements in subsequent plants. Identification of primary failure modes and suggestions for improved component

performance would be developed. A second program would be to quantify the costs associated with QA and code requirements and to identify more cost-effective ways to achieve the goal of improved component reliability.

SECTION 11

COST BENEFIT EVALUATION FUEL HANDLING BUILDING FILTRATION SYSTEM

INTRODUCTION

Concern over releases of radioactive material which could result from a fuel handling accident has led to the requirement for filtration systems on the ventilation system for fuel-handling buildings. The requirements for these systems are based on very conservative assumptions regarding activity release, atmospheric dispersion, frequency of occurrence, etc.

PHENOMENOLOGY

The concern here is basically that of frequency and degree of radioactivity release. There are significant differences in the assumptions used in safety evaluations—those which set design requirements for the systems, and those used in more realistic evaluations, such as for environmental impact reports and PRAs. These conservative assumptions dictate the inclusion of filtration systems which are only effective in reducing the level of iodine (thyroid dose) in the effluent released from the plant.

COST-BENEFIT ASSESSMENT

Risk Assessment

In this evaluation, primary use has been made of the information in NUREG-0612 (Ref. 35) which is the NRC evaluation of the probability and consequences of the drop of a heavy load at a nuclear plant. For this assessment the events of interest are: 1) the drop of a fuel element in the spent fuel pool, and 2) the drop of a heavy load into the spent fuel pool. Since either of these incidents could result in damage to fuel assemblies and the release of radioactivity into the fuel-handling building, the combined probability/consequence of these events is of concern.

NUREG-0612 bases its evaluation of the failure of heavy load handling systems on data from the U.S. Navy regarding crane/hoist failures. The conclusion reached is that the probability of load drop at a nuclear plant is between 10^{-5} and 1.5×10^{-4} per lift. This presumes an improvement in the Navy data by a factor of 0.5 based on improved

procedures at nuclear plants. While it is recognized that fuel-handling cranes are not exactly the same as normal cranes in the manner of operation and grappling, this failure rate is believed to be sufficiently representative for this exercise. (There are considerations which could lead to an increase or decrease in this failure rate which are not definable in the scope of this effort.) Assuming a log-normal distribution, the mean-failure rate would be 5.4×10^{-5} per lift. Approximately 1/3 of the core (about 65 fuel assemblies) will be removed per year during refueling of a PWR. Therefore the mean probability of a fuel assembly being dropped is:

$$(65)(5.4 \times 10^{-5}) = 3.5 \times 10^{-3} \text{ per year}$$

To date a total of about 650 unit-years of experience with large LWRs have been accrued in the U.S. Since there have not been any fuel assembly drops involving major fuel damage, it appears that the actual rate of occurrence is less than 1.5×10^{-3} , a factor of 2 or more lower than the value above.

NUREG-0612 evaluates the probability of a heavy load being dropped into the spent fuel pool as (the probability of operator error in following procedures) x (the probability of failure of interlocks preventing a heavy load from traveling over the fuel pool) x (failure of the handling system) x (the number of lifts per year). Using the failure rate ranges assumed in NUREG-0612 and assuming a log-normal distribution in each case results in the following mean-failure rates:

Operator error	2.5×10^{-2} /year
Interlock failure	4×10^{-3} /year
Handling-system failure	5.4×10^{-4} /lift

The handling system failure rate is a factor of 10 greater than that used in the fuel-handling evaluation; because for the event to occur, it is assumed that procedures had not been followed and that electrical interlocks have failed. These events could have resulted from poorly trained or unqualified operators not following procedures, not checking interlocks, and then not operating the system correctly. This possible common link was assumed to increase the failure probability by a factor of 10. NUREG 0612 assumes the number of lifts per year to be 200. This value considers all lifts in the plant which could result in fuel damage, including such events as reactor vessel head movement. Since this value is clearly conservative with respect to

handling loads over the fuel pool, it has been used. The resulting probability of a heavy load drop into the pool is:

$$(2.5 \times 10^{-2})(4 \times 10^{-3})(5.4 \times 10^{-4})(200) = 1.1 \times 10^{-5}/\text{yr.}$$

The consequences of fuel failures in the fuel-handling building resulting from dropped loads was stated to be a nonsignificant contributor to risk in WASH-1400, and was not evaluated in the Zion or Indian Point PRA's. Consequences can, however, be derived from available information. The filtration systems of the ventilation system are only effective in removing iodine from the released effluent and therefore only significantly impact the thyroid dose. The applicable source term was derived by modification of information in NUREG-0612 to account for the conservative assumptions used (those of Regulatory Guide 1.25 related to Safety Analyses versus those specified in Regulatory Guide 4.2 related to Environmental Reports). Atmospheric dilution was determined via a Gaussian plume model. Site-specific data regarding wind history and population density was taken from the Indian Point PRA. The specifics of the calculations are presented in Appendix A-5. The end result of this evaluation is:

	Thyroid Dose - Man Rem	
	<u>With Filter</u>	<u>Without Filter</u>
Fuel Assembly Drop	2.31	45.36
Load Drop	11.89	103.3

Combining these results with the probability of the events results in the following risk:

	Thyroid Man Rem/Year	
	<u>With Filter</u>	<u>Without Filter</u>
Fuel Assembly Drop	.0081	.16
Load Drop	<u>.00013</u>	<u>.0011</u>
Total	.00823	.1611

The change in risk associated with removal of the filters is therefore $(0.1611 - 0.00823) = 0.153$ man rem (thyroid) per year.

Appendix VI of WASH-1400 states that the number of expected thyroid nodules per 10^6 man-rem is 334 (200 benign and 134 cancerous). It further indicates that 10% of the thyroid cancers result in fatalities. Using these values results in:

$$(0.153)(334 \times 10^{-6}) = 5.1 \times 10^{-5} \text{ thyroid nodules/year}$$

$$(0.153)(134 \times 10^{-6})(.1) = 2.05 \times 10^{-6} \text{ thyroid cancer deaths/year}$$

The "value" of this increased risk can be assessed as that associated with the increase in thyroid nodules or as that associated with the increase in death. Table 3-1 assessed a value for thyroid nodules as \$10,000 per occurrence. The value of a latent fatality was set at \$500,000. (It is assumed that there will be no property damage.)

The value of increased risk can therefore be stated as either:

$$5.1 \times 10^{-5} \frac{\text{nodules}}{\text{year}} \times 40 \text{ years} \times \$10,000 = \$20$$

or

$$2.05 \times 10^{-6} \frac{\text{cancer deaths}}{\text{year}} \times 40 \text{ year} \times \$500,000 = \$41$$

Cost Impact

Based on information in Reference 20, the portion of the filtration system effective in reducing iodine dose (two charcoal filter banks and associated equipment) is estimated to be \$500,000.

CONCLUSIONS

It has been estimated that the value of the reduction in risk from fuel-handling accidents in the fuel-handling building resulting from the inclusion of the required charcoal filter banks is about \$60. The cost of these features is estimated as \$500,000. The cost is therefore about 8,000 times the benefit gained. The actual ratio of cost to benefit is probably a factor of 10 or more greater since the analysis used a rate of fuel assembly drops which appears high and a population density which is much greater than at the average plant, and since the Reference 20 cost estimate is probably low (since it did not include AFUDC).

Section 12

REGULATIONS NOT EVALUATED

INTRODUCTION

All of the regulatory requirements listed in Table 1-1 were investigated and the eight which are presented in Sections 4-11 were selected for detailed evaluations. This section briefly describes findings on the remaining requirements. There is limited information on the frequency consequence relationships in several of the areas. However, a general discussion on each is provided below with recommendations for future R&D work. Those areas perceived as having the greatest potential for reducing cost (most non-cost effective) are addressed first, with succeeding discussions ordered to reflect their decreasing cost potential.

QUALITY ASSURANCE

An area for future R&D which is beyond the scope of this report is quantification of the cost-benefit of quality assurance programs to overall nuclear power plant safety. Of particular interest are those quality assurance activities which do not materially improve component reliability, such as traceability and documentation, but add appreciably in time and direct cost. A recent study (Reference 36) concluded that quality assurance programs currently in practice in the United States add approximately one year to a plant schedule and from 30% to 35% to overall plant cost. The percentage values presumably include interest and escalation expenses during construction. Based upon Reference 35's cost estimates for a 1200 MWe LWR unit with mechanical draft cooling, this yields a total quality assurance cost contribution between \$797 million and \$930 million. If the percentage costs are assigned only to direct and indirect expenses associated with the physical plant and engineering and construction, the cost impact decreases to \$340 million to \$397 million.

From Section 2, the value associated with risk at a typical 1960's vintage PWR is on the order of \$2 million. This represents the maximum risk-justifiable increase in quality assurance costs over original expenditures and conservatively assumes that the additional quality assurance measures would eliminate all risk. From a practical standpoint, it is noted that most of the risk at Zion and Indian Point is dominated by

externally initiated events (earthquakes, fires, high winds). In the case of seismic events, the probability of core melt is governed by one or two component/structure failures in conjunction with loss of off-site power. The probability of component/structure failure is determined by the probability of the seismic event greatly exceeding the safe shutdown earthquake design acceleration. The probability of hazards from fires and high winds is also primarily dependent on the frequency of the event exceeding design conditions and the choice of design conditions. Since quality assurance programs are not intended to qualify components beyond design conditions, they will have negligible impact on risk from external events at Zion and Indian Point. Baseline engineering costs are assumed to include standard quality assurance features such as document control and approval procedures. Thus, it is concluded that the quality assurance programs at Zion and Indian Point primarily impact only the risk from random component failures during internally initiated events. The value associated with internally initiated risk at Zion and Indian Point varies from \$80,000 to \$560,000 and, thus, represents a more reasonable estimate for the maximum justifiable increase over original expenditures due to quality assurance programs.

While certain levels of quality assurance are both necessary and desirable, it appears that considerable change in current practices is appropriate, and that significant cost savings may be possible. A detailed study to identify component and system costs associated with implementation of current quality assurance practices for nuclear power plants at the design, manufacturer, and construction levels is required. It also appears that component and system reliability data as summarized in Appendix III of WASH 1400 for nuclear and conventional applications needs to be updated and expanded to quantify failure rates on a quality assurance basis. Used in conjunction with generalized WASH 1400 PRA techniques and cost-benefit analyses, areas involving excessive or insufficient quality assurance could then be quantified and guidelines for an optimum quality assurance program identified. Because of the very large costs associated with QA, R&D in this area should receive a high priority.

ELECTRICAL AND CONTROL REQUIREMENTS

Another area of potential cost savings is modification or relaxation of electrical and control requirements which are based solely on deterministic design criteria. Of particular interest are those standards impacting the design and single failure requirements for the reactor protection system (IEEE-279), separation of Class 1E equipment (IEEE-384), Class 1E power systems (IEEE-308), and qualification of

Class 1E electrical equipment (IEEE-323). As noted previously in References 16 and 17, strict adherence to these standards does not necessarily lead to a more reliable or optimum system design.

Appendix III of WASH 1400 concluded that there is little difference between nuclear and nonnuclear random component failure rates. The reduced failure rate for nuclear vessels versus conventional design is an exception. Review of the failure rate data for electrical and control equipment (e.g., motors, transformers, switches, instrumentation) indicates that the specific nuclear experience compiled falls midway within the industrial range, with some nuclear components either at the lower or upper bounds.

It is suggested that a risk analysis be performed on reactor protection systems (RPS) which are designed in accordance with current requirements. This will permit comparison with earlier plants such as Zion, Indian Point, and WASH 1400 where the RPS design, while incorporating some of the single failure and separation criteria of current requirements, was completed prior to formal adoption of the design and qualification standards cited above. The mean unavailability per event of the reactor protection systems at these earlier plants is:

Zion 1 or 2	1.78×10^{-4}
Indian Point 2	2.0×10^{-5}
Indian Point 3	3.93×10^{-5}
WASH 1400	$4.4 \times 10^{-5} *$

* Mean value calculated based on upper and lower bounds specified in WASH 1400

The Reactor Protection Systems at these plants are similar with process signals used to deenergize undervoltage devices in two series connected reactor trip breakers. Opening of one of the scram breakers interrupts power to the control rod drive magnetic coils, permitting them to fall by gravity into the core. There are two channels of actuation logic. In the case of Zion and Indian Point, logic errors/failure contribute only 6.3×10^{-11} to train unavailability. Several of the system faults are nonelectrical. At Zion, mechanical reliability of the scram breakers dominates system unavailability. At Indian Point, scram breaker reliability and wiring faults to power and ground are the major contributors. In WASH 1400, mechanical failure of the CRDs

to fully insert into the core, and testing and maintenance coincident with failure of the other train are dominating contributors.

With the increased nuclear operating experience since WASH 1400 and the trend towards using probabilistic risk assessment as a design tool, electrical component failure rate data is being updated and expanded. This information needs to be consolidated and compared with conventional applications to evaluate the cost-benefit of current nuclear regulations.

Electrical requirements have had a major impact on plant costs in recent years and R&D in these areas could be very useful.

RADIOACTIVE WASTE ACCIDENTS

Only limited information was found in the PRAs reviewed related to the frequency/consequence of these events. The Zion and Indian Point PRAs only considered events leading to release of activity from the core. Situations involving radioactive waste system accidents were not discussed. Presumably such events were assessed as being nonsignificant contributors to risk. WASH 1400 (Appendix I, Section 5) deals with potential accidents not involving the core. In this section, estimates of the event frequency and radioactivity released are given. The report concludes that these accidents represent a very small contribution to overall risk, but states that the analyses of these events were rough and largely based on engineering judgment and experience rather than on detailed analysis as performed for other systems. Because of the lack of detail, it was not possible to assess the potential impacts of relaxation of requirements in these areas without involving efforts beyond the scope of this study. A more detailed analysis of the frequency and consequences would be required. It is however felt that because of the low levels of radioactivity involved, a cost-benefit analysis would indicate that some presently required features would be noncost-effective. Since the costs associated with radwaste systems are relatively small, other areas deserve higher priority for R&D.

IN-SERVICE INSPECTION OF NONREACTOR COOLANT SYSTEM (RCS) PASSIVE COMPONENTS

Requirements for removable insulation for in-service inspection (ISI) of non-RCS passive components (e.g., piping, vessels) are perceived as having negligible impact on improving plant safety. This conclusion is based on the observation that active

components such as pumps and valves dominate system reliability. As noted in Section 6, active components are estimated to contribute 10^{-2} to the failure probability of one train in a safety system. This compares with passive pipe failure contributions in the order of 4×10^{-8} . Clearly, active component failures will dominate. Reference 30 indicates that optimization of in-service inspection procedures for vessels should apply to piping systems as well and could result in a factor of improvement in random pipe failures of over three orders of magnitude. While the relative improvement in pipe failure rates is substantial, the improvement in overall plant safety will normally be negligible.

Section 7, which discusses physical separation of redundant safety system piping, concludes that a certain degree of cost reduction and improved system reliability will result with cross connection between headers (trains). This in effect returns the system configuration to a single header design. Since there is a corresponding, nominal increase in risk from pipe failures, this provides a potential opportunity for implementing a beneficial ISI Program.

While ISI Programs for reactor coolant systems are generally considered productive (by reducing the probability of LOCA initiating events) a second look to quantify the cost-benefits may be desirable in light of the potential cost/time impact on refueling schedules. Because costs are generally low, this is not a high priority for R&D.

Appendix 1

FATALITY CALCULATION METHODOLOGY

INTRODUCTION

In this section methods used for calculating average fatalities per plant damage state are discussed. This section uses data from the Probabilistic Risk Assessments (PRAs) of Zion and Indian Point.

We feel that the average number of fatalities due to an event (or sequence of events) leading to release, coupled with the frequency of the event(s), gives the best overall picture of risk to the general public. For this reason, other quantifications of risk in the PRAs (thyroid cancer, injury, man-remS) were not considered.

The plants considered were Zion 1 and 2 and Indian Point 2 and 3. The Zion units were considered identical both in terms of frequencies of occurrences and consequences thereof. The Indian Point plants were analyzed using individually applicable data (i.e., hardware differences, core-power differences).

Both PRAs made the distinction between internal and external initiating events. External events considered included earthquakes, fires, tornadoes, and hurricanes. Internal initiating events include small, medium, and large LOCAs; turbine trips caused by loss of off-site power; loss of RCS flow; and others. Initiating events have conditional probabilities of causing a plant damage state, characterized by combinations of the following abbreviations:

Table A1-1

PLANT DAMAGE STATE ABBREVIATIONS

A	-	Large LOCA Behavior
B	-	Small LOCA Behavior
T	-	Transient Behavior
E	-	Early Core Melt
L	-	Late Core Melt
F	-	Containment Fan Coolers Operable
C	-	Containment Spray Operable

These plant damage states have conditional probabilities of leading to various kinds of releases. To simplify the analysis of risk, we calculate the average number of fatalities per release category. We then calculate the frequency of release categories given a plant damage state. Finally, we calculate the average number of fatalities given a plant damage state. From this we calculate the frequency in average fatalities per reactor-year of initiating events and subsequent failures of engineered safeguard features leading to plant damage states and releases. In this way we can identify the dominant contributors to the risk from the plant. NOTE: In this section, abbreviated scientific notation is used, e.g., 5.216×10^{-12} .

CALCULATION OF AVERAGE FATALITIES PER PLANT STATE AT ZION

Step One - Integration of Exceedence Tables

The goal of this step is to calculate the average number of fatalities (early and latent) given a release category. The latent fatalities exceedence table gives the conditional probability of exceeding "X" fatalities. Referring to Table A1-2, we see that given a Category 5R release, the probability of exceeding one latent fatality is 9.792×10^{-1} . The probability of exceeding two latent fatalities from a Category 5R release is 9.375×10^{-1} . We may express these conditions as follows:

$$\begin{aligned} P(\text{exceeding one fatality}) &= P(\text{one fatality}) + P(\text{two fatalities}) + \dots \\ &= 9.792 \times 10^{-1} \\ P(\text{exceeding two fatalities}) &= P(\text{two fatalities}) + P(\text{three fatalities}) + \dots \\ &= 9.375 \times 10^{-1} \end{aligned}$$

The difference, $P(\text{exceeding one fatality}) - P(\text{exceeding two fatalities})$, gives the probability of exactly one fatality occurring as a result of a Category 5R release.

By the same method, we generate probabilities for various numbers of latent fatalities from a Category 5R release (see Table A1-3). Because the exceedence tables are not continuous (i.e., one, two, three, four——fatalities) we arrive at different terms such as the probability of exactly three or four fatalities, five or six fatalities, etc.

To calculate the average number of fatalities per release, we multiply the conditional probability of fatalities times the number of fatalities and sum the products. In the cases where we have several terms involving fatalities, the mean value is used. As an example, the average number of latent cancer fatalities resulting from a Category 5R release is calculated below.

Table A1-2

CONDITIONAL PROBABILITY OF EXCEEDENCE OF CANCER FATALITY LEVELS (ZION)

Number	Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B	2RV
1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.792E-01	1.000E+00	9.062E-01	1.042E-01	3.750E-01	1.042E-01	9.790E-01
2.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	9.375E-01	1.000E+00	8.125E-01	1.042E-02	2.818E-02	2.083E-02	9.650E-01
3.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	8.958E-01	9.965E-01	7.500E-01	0.	2.188E-01	1.042E-02	9.550E01
5.000E+00	9.896E-01	9.965E-01	9.931E-01	1.000E-01	8.646E-01	9.931E-01	6.354-01	0.	1.667E-01	0.	8.990E-01
7.000E+00	9.792E-01	9.931E-01	9.931E-01	1.000E+00	8.333E-01	9.896E-01	5.937E-01	0.	1.042E-01	0.	8.720E-01
1.000E+01	9.792E-01	9.931E-01	9.931E-01	9.792E-01	7.917E-01	9.896E-01	5.521E-01	0.	7.292E-02	0.	8.370E-01
2.000E+01	9.583E-01	9.861E-01	9.861E-01	9.792E-01	7.187E-01	9.792E-01	3.750E-01	0.	0.	0.	7.260E-01
3.000E+01	9.583E-01	9.826E-01	9.826E-01	9.375E-01	6.875E-01	9.618E-01	2.292E-01	0.	0.	0.	6.600E-01
5.000E+01	9.375E-01	9.583E-01	9.583E-01	8.646E-01	5.729E-01	9.271E-01	1.667E-01	0.	0.	0.	5.660E-01
7.000E+01	9.375E-01	9.514E-01	9.514E-01	8.333E-01	5.000E-01	9.062E-01	9.375E-02	0.	0.	0.	5.170E-01
1.000E+02	9.271E-01	9.236E-01	9.201E-01	8.125E-01	4.062E-01	8.819E-01	3.125E-02	0.	0.	0.	4.030E-01
2.000E+02	8.542E-01	8.646E-01	8.576E-01	7.708E-01	2.604E-01	8.333E-01	0.	0.	0.	0.	2.400E-01
3.000E+02	8.437E-01	8.368E-01	8.299E-01	6.771E-01	1.667E-01	7.708E-01	0.	0.	0.	0.	1.700E-01
5.000E+02	7.917E-01	7.817E-01	7.639E-01	5.937E-01	1.146E-01	6.944E-01	0.	0.	0.	0.	7.640E-02
7.000E+02	7.604E-01	7.222E-01	7.049E-01	5.208E-01	4.167E-02	6.111E-01	0.	0.	0.	0.	4.170E-02
1.000E+03	7.187E-01	6.701E-01	6.458E-01	4.271E-01	2.083E-02	5.417E-01	0.	0.	0.	0.	1.740E-02
2.000E+03	5.625E-01	5.000E-01	4.792E-01	1.979E-01	0.	3.299E-01	0.	0.	0.	0.	0.
3.000E+03	4.167E-01	3.507E-01	3.333E-01	1.354E-01	0.	1.944E-01	0.	0.	0.	0.	0.
5.000E+03	2.500E-01	1.840E-01	1.632E-01	6.250E-02	0.	1.007E-01	0.	0.	0.	0.	0.
7.000E+03	1.875E-01	1.042E-01	9.375E-02	3.125E-02	0.	4.514E-02	0.	0.	0.	0.	0.
1.000E+04	1.146E-01	3.125E-02	3.125E-02	0.	0.	6.944E-03	0.	0.	0.	0.	0.
2.000E+04	2.083E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.000E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.000E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
7.000E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
1.000E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
2.000E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.000E+05	0.	0.	0	0.	0.	0.	0.	0.	0.	0.	0.
5.000E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
7.000E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
1.000E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
2.000E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
3.000E+06	0.	0	0.	0.	0.	0.	0.	0.	0.	0.	0.
5.000E+06	0.										
7.000E+06	0.										
1.000E+07	0.										
2.000E+07	0.										
3.000E+07	0.										

Table A1-3

<u>Number of Fatalities</u>	<u>Mean</u>		<u>Probability</u>		<u>Product</u>
1	1	X	4.17-2	=	4.17-2
2	2		4.17-2		8.34-2
3 or 4	3.5		3.12-2		1.09-1
5 or 6	5.5		3.13-2		1.72-1
7, 8, or 9	8.0		4.14-2		3.31-1
10-19	14.5		7.32-2		1.06-0
20-29	24.5		3.12-2		7.64-1
30-49	39.5		1.15-1		4.53-0
50-69	59.5		7.29-2		4.34-0
70-99	84.5		9.38-2		7.93-0
100-199	149.5		1.46-1		2.18+1
200-299	249.5		9.37-2		2.34+1
300-499	399.5		5.21-2		2.08+1
500-699	599.5		7.29-2		4.37+1
700-999	849.5		2.08-2		1.77+1
1000-1999	1499.5		2.08-2		<u>3.12+1</u>

Average Number of Latent Cancer Fatalities Per 5R Release (Zion) 177.96

Results of similar calculations for the various release categories are summarized in Table A1-4. Notation and definition for release categories are given in Table A1-5.

Table A1-4

AVERAGE NUMBER OF EARLY, LATENT, AND TOTAL FATALITIES PER RELEASE (ZION)

	Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B	2RV
Average Number Of Early Fatalities	60.25	38.61	28.18	66.66	0	16.43	0	0	0	0	0
Average Number Of Latent Fatalities	1619.01	2936.43	2736.04	1415.61	175.24	1929.25	23.94	.021	2.05	.141	164.31
Average Total Fatalities	1679.26	2975.04	2764.61	1482.27	175.24	1945.68	23.84	.021	2.05	.141	164.31

Table A1-5

NOTATION AND DEFINITIONS FOR RELEASE CATEGORIES (ZION)

<u>Release Category</u>	<u>Description</u>
Z-1	Steam Explosion Without Containment Spray
2	Overpressure Failures Without Sprays
2R	Delayed Overpressure Failure Without Sprays
Z-3	Steam Explosion With Sprays
5R	Delayed Overpressure With Sprays
Z-5	Early Overpressure Failure With Sprays
6	Basemat Melt-Through Without Sprays
7	Basemat Melt-Through With Sprays
8A	Containment Intact Without Sprays
8B	Containment Intact With Sprays
2RV	Filtered Vented Release

Step Two - Conversion From Fatalities Per Release To Fatalities Per Plant State

In Step 1 we found the average number of fatalities given a release category. However, because plant event trees lead to plant damage states, it is useful to know the average number of fatalities per plant state in order to quantify risk resulting from postulated scenarios.

Table A1-6 (Containment Matrix) gives the conditional probability of release category given a plant damage state. To calculate the number of fatalities per plant damage state, we sum (over all release categories) the conditional probability of release times the average number of fatalities per release (see Table A1-7).

Table A1-6

CONTAINMENT MATRIX (ZION)

Plant State	Release Category										
	Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B	2RV
SEFC	0.	1.000E-04	0.	1.781E-07	1.309E-04	8.985E-05	0.	1.740E-04	0.	9.996E-01	0.
SEF	1.781E-07	1.899E-04	1.309E-04	0.	0.	0.	1.740E-04	0.	9.996E-01	0.	0.
SEC	0.	1.000E-04	0.	1.781E-07	1.309E-04	8.985E-05	0.	1.740E-04	0.	9.996E-01	0.
SE	2.186E-06	1.999E-04	9.996E-01	0.	0.	0.	1.900E-04	0.	1.000E-04	0.	0.
SLFC	0.	1.000E-04	0.	9.084E-06	1.309E-04	8.985E-05	0.	1.740E-04	0.	9.995E-01	0.
SLF	9.084E-06	1.899E-04	1.309E-04	0.	0.	0.	1.740E-04	0.	9.995E-01	0.	0.
SLC	0.	1.000E-04	0.	9.084E-06	1.309E-04	8.985E-05	0.	1.740E-04	0.	9.995E-01	0.
SL	2.186E-06	1.999E-04	9.996E-01	0.	0.	0.	1.900E-04	0.	1.000E-04	0.	0.
TEFC	0.	1.000E-04	0.	9.797E-08	1.027E-04	9.982E-05	0.	1.899E-04	0.	9.995E-01	0.
TEF	9.797E-08	1.999E-04	1.027E-04	0.	0.	0.	1.899E-04	0.	9.995E-01	0.	0.
TEC	0.	1.000E-04	0.	9.797E-08	1.027E-04	9.982E-05	0.	1.899E-04	0.	9.995E-01	0.
TE	2.186E-06	1.999E-04	9.996E-01	0.	0.	0.	1.900E-04	0.	1.000E-04	0.	0.
AEFC	0.	1.000E-04	0.	1.979E-06	1.499E-04	1.998E-10	0.	1.499E-04	0.	9.996E-01	0.
AEF	1.979E-06	1.000E-04	1.499E-04	0.	0.	0.	1.499E-04	0.	9.996E-01	0.	0.
AEC	0.	1.000E-04	0.	1.979E-06	1.499E-10	1.998E-10	0.	1.499E-04	0.	9.996E-01	0.
AE	1.000E*00	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ALFC	0.	1.000E-04	0.	1.979E-06	9.999E-10	2.204E-10	0.	1.899E-04	0.	9.996E-01	0.
ALF	1.979E-06	1.000E-04	9.999E-05	0.	0.	0.	1.899E-04	0.	9.996E-01	0.	0.
ALC	0.	1.000E-04	0.	1.979E-06	9.999E-05	2.204E-10	0.	1.899E-04	0.	9.996E-01	0.
AL	1.000E*00	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
VE	0.	1.000E*00	0.	0.	0.	0.	0.	0.	0.	0.	0.

Table A1-7

TOTAL AVERAGE FATALITIES PER STATE (ZION)

Plant State	Release Category										Total
	Z-1	2	Z-3	Z-5	6	7	8A	8B	2RV	5R	
SEFC	0.	2.975-1	2.640-4	1.748-1	0.	3.626-6	0.	1.409-1	0.	2.294-2	6.364-1
SEF	2.991-4	5.650-1	0.	0.	4.166-3	0.	2.049-0	0.	0.	0.	2.980-0
SEC	0.	2.975-1	2.640-4	1.784-1	0.	3.626-6	0.	1.409-1	0.	2.294-2	6.364-1
SE	3.671-3	5.947-1	0.	0.	4.549-3	0.	2.05-4	0.	0.	0.	2.764+3
SLFC	0.	2.975-1	1.346-2	1.784-1	0.	3.626-6	0.	1.409-1	0.	2.294-2	6.496-1
SLF	1.525-2	5.650-1	0.	0.	4.166-3	0.	2.049-0	0.	0.	0.	2.995-0
SLC	0.	2.975-1	1.346-2	1.784-1	0.	3.626-6	0.	1.409-1	0.	2.294-2	6.496-1
SL	3.671-3	5.947-1	0.	0.	4.549-3	0.	2.05-4	0.	0.	0.	2.764+3
TEFC	0.	2.975-1	1.452-4	1.942-1	0.	3.958-6	0.	1.409-1	0.	1.800-2	6.507-1
TEF	1.645-4	5.947-1	0.	0.	4.546-3	0.	2.049-0	0.	0.	0.	2.932-0
TEC	0.	2.975-1	1.452-4	1.942-1	0.	3.958-6	0.	1.409-1	0.	1.800-2	6.507-1
TE	3.671-3	5.947-1	0.	0.	4.549-3	0.	2.05-4	0.	0.	0.	2.764+3
AEFC	0.	2.975-1	2.933-3	3.887-7	0.	3.124-6	0.	1.409-1	0.	2.627-2	4.676-1
AEF	3.323-3	2.975-1	0.	0.	3.589-3	0.	2.049-0	0.	0.	0.	2.768-0
AEC	0.	2.975-1	2.933-3	3.887-7	0.	3.124-6	0.	1.409-1	0.	2.627-2	4.676-1
AE	1679.26	0.	0.	0.	0.	0.	0.	0.	0.	0.	1679.26
ALFC	0	2.975-1	2.933-3	4.288-7	0.	3.958-6	0.	1.409-1	0	1.752-2	4.589-1
ALF	3.323-3	2.975-1	0.	0.	4.546-2	0.	2.049-0	0.	0.	0.	2.672-0
ALC	0.	2.975-1	2.933-3	4.288-7	0.	3.956-6	0.	1.409-1	0.	1.752-2	4.589-1
AL	1679.26	0.	0.	0.	0.	0.	0.	0.	0.	0.	1679.26
VE	0.	2975.04	0.	0.	0.	0.	0.	0.	0.	0.	2975.04

NOTE: Because of an error in calculation these values may be up to 1% too low.

CALCULATION OF AVERAGE FATALITIES PER PLANT STATE AT INDIAN POINT

The logic behind these calculations is identical to that of the Zion calculations. However, because the data in the Indian Point PRA is presented slightly differently, the technique is somewhat more involved.

Table A1-8 gives the frequency of occurrence of damage levels (in terms of early fatalities) as a result of release categories due to internal initiating events.

Table A1-9 gives the frequency of occurrence of release categories--also due to internal initiating events. (For definitions of release categories at Indian Point see Table A1-11.)

Division of Table A1-8 entries by the appropriate Table A1-9 (frequency of release) entries yields the conditional probability of occurrence of exceedence of early fatality levels given a release category. (See Tables A1-8, A1-9, A1-10)

Table A1-8

Frequency of occurrence of exceedence of early fatality levels as a result of a release category—internal risk only. (Indian Point 2).

<u>Number</u>	<u>RELEASE CATEGORY</u>					
	<u>Z-1</u>	<u>2</u>	<u>2RW</u>	<u>8A</u>	<u>8B</u>	<u>2RV</u>
1.00E+00	9.651E-10	1.416E-07	0.	0.	0.	0.
2.00E+00	8.660E-10	1.271E-07	0.	0.	0.	0.
3.00E+00	8.660E-10	1.271E-07	0.	0.	0.	0.
5.00E+00	8.660E-10	1.271E-07	0.	0.	0.	0.
7.00E+00	7.989E-10	1.123E-07	0.	0.	0.	0.
1.00E+01	7.670E-10	1.126E-07	0.	0.	0.	0.
2.00E+01	7.318E-10	1.074E-07	0.	0.	0.	0.
3.00E+01	6.647E-10	9.756E-08	0.	0.	0.	0.
5.00E+01	6.008E-10	8.818E-08	0.	0.	0.	0.
7.00E+01	5.337E-10	7.833E-08	0.	0.	0.	0.
1.00E+02	3.324E-10	4.878E-08	0.	0.	0.	0.
2.00E+02	2.662E-10	3.907E-08	0.	0.	0.	0.
3.00E+02	1.917E-10	2.814E-08	0.	0.	0.	0.
5.00E+02	9.587E-11	1.407E-08	0.	0.	0.	0.
7.00E+02	6.391E-11	9.381E-09	0.	0.	0.	0.
1.00E+03	4.794E-11	7.036E-09	0.	0.	0.	0.
2.00E+03	3.196E-11	4.690E-09	0.	0.	0.	0.
3.00E+03	2.557E-11	3.752E-09	0.	0.	0.	0.
5.00E+03	2.237E-11	3.283E-09	0.	0.	0.	0.
7.00E+03	1.822E-12	2.674E-10	0.	0.	0.	0.

Table A1-9

THE FREQUENCIES OF VARIOUS RELEASE CATEGORIES—INTERNAL RISK ONLY
(INDIAN POINT 2)
(Occurrences per reactor year)

<u>Z-1</u>	<u>2</u>	<u>2RW</u>	<u>8A</u>	<u>8B</u>
3.20E-9	4.69E-7	1.14E-6	1.21E-8	8.80E-5

Table A1-10

Conditional probability of exceedence of early fatality levels from release given release categories. (Indian Point 2)

<u>Number of Early Fatalities</u>	<u>Release Z-1</u>	<u>Category 2</u>
1	3.02-1	3.02-1
2	2.71-1	2.71-1
3	2.71-1	2.71-1
5	2.71-1	2.71-1
7	2.50-1	2.50-1
10	2.40-1	2.40-1
20	2.29-1	2.29-1
30	2.08-1	2.08-1
50	1.88-1	1.88-1
70	1.67-1	1.67-1
100	1.04-1	1.04-1
200	8.32-2	8.33-2
300	5.99-2	6.00-2
500	3.00-2	3.00-2
700	2.00-2	2.00-2
1,000	1.50-2	1.50-2
2,000	9.99-3	1.00-2
3,000	7.99-3	8.00-3
5,000	6.99-3	7.00-3
7,000	5.69-4	5.70-4
10,000	0.0	0.0
20,000	0.0	0.0

Table A1-11

NOTATION AND DEFINITIONS FOR RELEASE CATEGORIES
(INDIAN POINT 2)

<u>Release Category</u>	<u>Description</u>
Z-1	Early Overpressure Without Containment Spray and With Source Term Like That From Steam Explosion
Z-1Q	Quick Release (Half Hour) Without Spray Failure of Containment Structure (Seismic Only)
2	Early Overpressure Failure Without Sprays and Without Steam Explosion Source Term
2R *	Delayed Overpressure Failure Without Sprays and Without Vaporization Source Term
Z-3	Steam Explosion With Sprays
5R	Delayed Overpressure With Sprays
Z-5	Early Overpressure Failure With Sprays
6 *	Basemat Melt-Through Without Sprays
7 *	Basemat Melt-Through With Sprays
8A	Containment Intact Without Sprays
8B	Containment Intact With Sprays
2RW	Delayed Overpressure Failure Without Sprays and With Vaporization Source Term
2RV	Filtered Vented Release

* These release categories were determined to have low frequency and were therefore not included in consequence calculations in the PRA. Most, however, were included in containment matrix quantifications (Reference page 5.4-8 Indian Point PRA).

The same operation is performed on Table A1-12 (frequency of occurrence of exceedence of latent cancer fatality levels as a result of a release category).

Table A1-12

Frequency of occurrence of exceedence of latent cancer fatality levels as a result of a release category. (Indian Point 2)

<u>Number</u>	<u>Z-1</u>	<u>2</u>	<u>2RW</u>	<u>8A</u>	<u>8B</u>	<u>2RV</u>
1.00E+00	3.196E-09	4.690E-07	1.141E-06	1.186E-06	2.571E-05	0.
2.00E+00	3.196E-09	4.690E-07	1.141E-06	9.971E-09	8.240E-06	0.
3.00E+00	3.196E-09	4.690E-07	1.141E-06	8.965E-09	1.831E-06	0.
5.00E+00	3.196E-09	4.690E-07	1.141E-06	5.428E-09	0.	0.
7.00E+00	3.196E-09	4.690E-07	1.141E-06	3.404E-09	0.	0.
1.00E+01	3.196E-09	4.690E-07	1.141E-06	1.636E-09	0.	0.
2.00E+01	3.164E-09	4.643E-07	1.141E-06	2.520E-10	0.	0.
3.00E+01	3.164E-09	4.643E-07	1.141E-06	0.	0.	0.
5.00E+01	3.164E-09	4.643E-07	1.141E-06	0.	0.	0.
7.00E+01	3.164E-09	4.643E-07	1.129E-06	0.	0.	0.
1.00E+02	3.129E-09	4.592E-07	1.129E-06	0.	0.	0.
2.00E+02	3.097E-09	4.545E-07	1.105E-06	0.	0.	0.
3.00E+02	3.030E-09	4.446E-07	1.093E-06	0.	0.	0.
5.00E+02	2.868E-09	4.209E-07	1.046E-06	0.	0.	0.
7.00E+02	2.697E-09	3.959E-07	9.866E-07	0.	0.	0.
1.00E+03	2.464E-09	3.616E-07	8.908E-07	0.	0.	0.
2.00E+03	1.581E-09	2.247E-07	6.410E-07	0.	0.	0.
3.00E+03	1.099E-09	1.613E-07	4.391E-07	0.	0.	0.
5.00E+03	6.999E-10	1.027E-07	3.091E-07	0.	0.	0.
7.00E+03	6.008E-10	8.818E-08	2.258E-07	0.	0.	0.
1.00E+04	2.994E-10	4.395E-08	1.540E-07	0.	0.	0.
2.00E+04	3.324E-11	4.878E-04	1.186E-08	0.	0.	0.
3.00E+04	0.	0.	0.	0.	0.	0.

Table A1-13

Conditional probability of exceedence of latent cancer fatality levels given release categories. (Indian Point 2)

Number of Latent Fatalities	RELEASE CATEGORY				
	Z-1	2	2RW	8A	8B
1	9.99-1	1.00	1.00	9.80-1	3.18-1
2	9.99-1	1.00	1.00	8.24-1	1.02-1
3	9.99-1	1.00	1.00	7.41-1	2.27-2
5	9.99-1	1.00	1.00	4.49-1	0.0
7	9.99-1	1.00	1.00	2.81-1	0.0
10	9.99-1	1.00	1.00	1.35-1	0.0
20	9.89-1	9.90-1	1.00	2.08-2	0.0
30	9.89-1	9.90-1	1.00	0.0	0.0
50	9.89-1	9.90-1	1.00	0.0	0.0
70	9.89-1	9.90-1	8.01-1	0.0	0.0
100	9.78-1	9.79-1	8.01-1	0.0	0.0
200	9.68-1	9.69-1	7.84-1	0.0	0.0
300	9.47-1	9.48-1	7.75-1	0.0	0.0
500	8.95-1	8.96-1	7.42-1	0.0	0.0
700	8.43-1	8.44-1	7.00-1	0.0	0.0
1,000	7.70-1	7.71-1	6.32-1	0.0	0.0
2,000	4.78-1	4.79-1	4.55-1	0.0	0.0
3,000	3.43-1	3.45-1	3.11-1	0.0	0.0
5,000	2.19-1	2.19-1	2.19-1	0.0	0.0
7,000	1.88-1	1.88-1	1.60-1	0.0	0.0
10,000	9.36-2	9.37-2	1.09-1	0.0	0.0
20,000	1.04-2	1.04-2	8.41-3	0.0	0.0
30,000	0.0	0.0	0.0	0.0	0.0

Next, Tables A1-10 and A1-13 are manipulated to obtain the average number of early and latent fatalities per release. The entries in these two tables represent the conditional probabilities of exceeding the given number of fatalities, given a release category. For example, referring to Table A1-13 the conditional probabilities of exceeding various latent fatality levels from a Category 8-A release are:

$$\begin{aligned}
 P(\text{exceeding 1 fatality}) &= P(1 \text{ fatality}) + P(2 \text{ fatalities}) + P+\dots \\
 &= 9.80 \times 10^{-1} \\
 P(\text{exceeding 2 fatalities}) &= P(2 \text{ fatalities}) + P(3 \text{ fatalities}) + P+\dots \\
 &= 8.24 \times 10^{-1} \\
 P(\text{exceeding 3 fatalities}) &= P(3 \text{ fatalities}) + P(4 \text{ fatalities}) + \dots \\
 &= 7.41 \times 10^{-1}
 \end{aligned}$$

$$\begin{aligned}
 P(\text{exceeding 5 fatalities}) &= P(5 \text{ fatalities}) + P(6 \text{ fatalities}) + \dots \\
 &= 4.49 \times 10^{-1} \\
 P(\text{exceeding 7 fatalities}) &= P(7 \text{ fatalities}) + P(8 \text{ fatalities}) + \dots \\
 &= 2.81 \times 10^{-1}
 \end{aligned}$$

The difference between $P(\text{exceeding 1 fatality})$ and $P(\text{exceeding 2 fatalities})$ gives conditional probability of exactly one latent fatality from a category 8-A release. Similarly, we obtain the conditional probability of exactly 2 fatalities, 3 or 4 fatalities, 5 or 6 fatalities, and so on.

Now we construct another table, as follows:

Table A1-14

INDIAN POINT 2, EARLY FATALITIES FROM
A CATEGORY 8-A RELEASE

<u>Number of Fatalities</u>	<u>Mean</u>	<u>Probability</u>	<u>Product</u>
1	1	.156	.156
2	2	.083	.166
3 or 4	3.5	.292	1.022
5 or 6	5.5	.168	.924
7 or 8 or 9	8.0	.146	1.168
10-19	14.5	.1142	1.6559
20-29	24.5	.0208	.5096
Total Sum			<u>5.60</u>

Category 8-A releases do not lead to any early fatalities. Therefore, we may say that given a Category 8-A release, we may expect an average of 5.60 total fatalities.

By the same method, the average early, latent, and total fatalities per release were calculated.

Table A1-15

TOTAL FATALITIES PER RELEASE

Release Category/ <u>Plant</u>	<u>Z-1</u>	<u>2</u>	<u>2RW</u>	<u>8A</u>	<u>8B</u>
Indian Point 2	3904.9	3908.8	4352.3	5.59	.417
Indian Point 3	3959.4	3910.6	4354.1	5.60	.497

In the Indian Point PRA, Category Z1-Q (seismic-induced containment failure) was considered separately. By methods similar to those discussed earlier, we calculated an average of 4,503 fatalities per Z1-Q release.

The final step is to convert from average fatalities per release to average fatalities per plant damage state. For this operation we need to know the conditional probability of a release category given a plant damage state. To obtain these probabilities we use Table A1-16 (frequency of occurrence of release categories from plant damage states) and Table A1-17 (frequency of occurrence of plant damage states). Results for Indian Point 2 are summarized below in Table A1-18.

The summation (over all release categories) of the conditional probability of release times the average number of fatalities per release gives the average number of fatalities per plant state. (Table A1-19).

Table A1-16

FREQUENCY OF OCCURRENCE OF RELEASE CATEGORIES FROM PLANT DAMAGE STATES (INDIAN POINT 2)

	Z-1	2	2R	2RW	Z-3	5R	Z-5	6	7	8A	8B
TOTAL	3.196E-09	4.690E-07	8.268E-11	1.141E-06	6.682E-11	8.250E-09	5.281E-10	2.197E-10	2.032E-08	1.212E-08	8.804E-05
SEFC	0.	3.808E-09	0.	0.	6.773E-12	3.421E-09	3.379E-10	0.	9.473E-09	0.	3.801E-05
SEF	6.807E-16	4.162E-10	3.439E-13	6.226E-16	0.	0.	0.	9.521E-13	0.	3.820E-09	0.
SEC	0.	4.102E-10	0.	0.	7.306E-13	3.691E-10	3.644E-11	0.	1.022E-09	4.100E-06	
SE	2.245E-12	2.058E-10	0.	1.027E-06	0.	0.	0.	1.952E-10	0.	1.027E-10	0.
SLFC	0.	1.340E-09	0.	0.	1.181E-12	1.205E-09	1.190E-10	0.	3.337E-09	0.	1.339E-05
SLF	3.771E-16	4.657E-13	3.848E-13	6.654E-16	0.	0.	0.	1.065E-12	0.	4.274E-09	0.
SLC	0.	5.647E-12	0.	0.	4.980E-15	5.081E-12	5.017E-13	0.	1.407E-11	0.	5.645E-08
SL	1.784E-16	1.681E-14	8.156E-11	0.	0.	0.	0.	1.550E-14	0.	8.160E-15	0.
TEFC	0.	3.278E-10	0.	0.	6.487E-13	3.276E-10	3.235E-11	0.	9.070E-10	0.	3.276E-06
TEF	2.784E-16	1.518E-13	1.381E-13	2.499E-16	0.	0.	0.	3.822E-13	0.	1.381E-09	0.
TEC	0.	1.970E-11	0.	0.	3.898E-14	1.969E-11	1.944E-12	0.	5.461E-11	0.	1.969E-07
TE	2.494E-13	2.280E-11	0.	1.140E-07	0.	0.	0.	2.167E-11	0.	1.141E-11	0.
AEFC	0.	7.956E-10	0.	0.	1.574E-11	7.954E-10	1.575E-15	0.	1.511E-09	0.	7.953E-26
AEF	1.501E-16	7.586E-14	7.583E-14	7.661E-18	0.	0.	0.	1.440E-13	0.	7.582E-10	0.
AEC	0.	2.742E-12	0.	0.	5.426E-14	2.741E-12	5.429E-18	0.	5.206E-12	0.	2.741E-04
AE	3.192E-09	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ALFC	0.	2.104E-09	0.	0.	4.164E-11	2.103E-09	4.578E-15	0.	3.996E-09	0.	2.103E-05
ALF	3.500E-15	1.769E-13	1.768E-13	2.658E-17	0.	0.	0.	3.359E-13	0.	1.768E-09	0.
ALC	0.	4.189E-14	0.	0.	8.289E-16	4.187E-14	9.114E-20	0.	7.954E-14	0.	4.187E-10
AL	1.094E-12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
VE	0.	4.600E-07	0.	0.	0.	0.	0.	0.	0.	0.	0.

Table A1-17

THE FREQUENCIES OF PLANT DAMAGE STATES (INDIAN POINT 2)
(OCCURRENCES PER YEAR, MEAN VALUES)

Transposed)
Plant Damage States

SEFC	3.80×10^{-5}
SEF	3.82×10^{-9}
SEC	4.10×10^{-6}
SE	1.03×10^{-6}
SLFC	1.34×10^{-5}
SLF	4.27×10^{-9}
SLC	5.65×10^{-8}
SL	8.17×10^{-11}
TEFC	3.28×10^{-6}
TEF	1.38×10^{-9}
TEC	1.97×10^{-7}
TE	1.14×10^{-7}
AEFC	7.96×10^{-6}
AEF	7.59×10^{-10}
AEC	2.74×10^{-8}
AE	3.19×10^{-9}
ALFC	2.10×10^{-5}
ALF	1.77×10^{-9}
ALC	4.19×10^{-10}
AL	1.09×10^{-12}
V	4.60×10^{-7}

Table A1-18
 CONDITIONAL PROBABILITY OF RELEASE CATEGORIES GIVEN A PLANT DAMAGE STATE
 INDIAN POINT 2

	<u>SEFC</u>	<u>SEF</u>	<u>SEC</u>	<u>SE</u>	<u>SLFC</u>	<u>SLF</u>	<u>SLC</u>	<u>SL</u>	<u>TEFC</u>	<u>TEF</u>	<u>TEC</u>	<u>TE</u>	<u>AEFC</u>	<u>AEF</u>	<u>AEC</u>	<u>AE</u>	<u>ALFC</u>	<u>ALF</u>	<u>ALC</u>	<u>AL</u>	<u>V</u>
Z-1	0.	1.78-7	0.	2.18-6	0.	8.83-8	0.	2.18-6	0.	1.98-7	0.	2.19-6	0.	1.98-6	0.	1.00-0	0.	1.98-6	0.	1.0	0.
2	1.00-4	1.09-4	1.00-4	1.99-4	1.00-4	1.09-4	9.99-5	2.00-4	9.99-5	1.10-4	1.00-4	2.00-4	9.99-5	9.99-5	1.00-4	0.	1.00-4	9.99-5	1.00-4	0.	1.0
2R	0.	9.00-5	0.	0.	0.	9.01-5	0.	9.98-1	0.	1.00-4	0.	0.	0.	9.99-5	0.	0.	0.	9.99-5	0.	0.	0.
2RW	0.	1.63-7	0.	9.97-1	0.	1.56-7	0.	0.	0.	1.81-7	0.	1.00-0	0.	1.01-8	0.	0.	0.	1.50-8	0.	0.	0.
Z-3	1.78-7	0.	1.78-7	0.	8.81-8	0.	8.81-8	0.	1.98-7	0.	1.98-7	0.	1.98-6	0.	1.98-6	0.	1.98-6	0.	1.98-6	0.	0.
5R	9.00-5	0.	9.00-5	0.	8.99-5	0.	8.99-5	0.	9.99-5	0.	9.99-5	0.	9.99-5	0.	1.00-4	0.	1.00-4	0.	9.99-5	0.	0.
Z-5	8.89-6	0.	8.94-6	0.	8.88-6	0.	8.88-6	0.	9.86-6	0.	9.87-6	0.	1.98-10	0.	1.98-10	0.	2.18-10	0.	2.18-10	0.	0.
6	0.	2.49-4	0.	1.90-4	0.	2.49-12	0.	1.90-4	0.	2.77-4	0.	1.90-4	0.	1.90-4	0.	0.	0.	1.90-4	0.	0.	0.
7	2.49-4	0.	2.49-4	0.	2.49-4	0.	2.49-0	0.	2.77-4	0.	2.77-4	0.	1.90-4	0.	1.90-4	0.	1.90-4	0.	1.90-4	0.	0.
8A	0.	1.00-0	0.	9.97-5	0.	1.00-0	0.	9.99-5	0.	1.00-0	0.	1.00-4	0.	9.99-1	0.	0.	0.	9.99-1	0.	0.	0.
8B	1.00-0	0.	1.00-0	0.	9.99-1	0.	9.99-1	0.	9.99-1	0.	9.99-1	0.	9.99-1	0.	1.00-0	0.	1.00-0	0.	9.99-1	0.	0.

Table A1-19

AVERAGE NUMBER OF FATALITIES**

<u>Plant Damage State</u>	<u>Indian Point 2</u>	<u>Indian Point 32</u>
SEFC	8.08-1	8.70-1
SEF	6.02-0	6.00-0
SEC	8.08-1	8.70-1
SE	4.34+3	4.351+3
SLFC	8.07-1	8.71-1
SLF	6.03-0	6.02-0
SLC	8.07-1	2.554+1 *
SL	7.91-1	8.18-1
TEFC	8.07-1	8.63-1
TEF	6.02-0	6.03-0
TEC	8.07-1	8.71-1
TE	4.35+3	4.351+3
AEFC	8.07-1	8.70-1
AEF	5.98-0	6.00-0
AEC	8.08-1	8.70-1
AE	3.9049+3	3.94+3
ALFC	8.08-1	8.69-1
ALF	5.98-0	6.00-0
ALC	8.07-1	8.70-1
AL	3.9049+3	3.957+3
V	3.9088+3	3.936+3

* Suspect typographical error in PRA.

** Due to an error in calculation, these values may be up to 1.% too low.

Appendix 2

SAFETY-RELATED PIPING CONTRIBUTION TO PLANT RISK AT ZION AND INDIAN POINT

INTRODUCTION

The Probabilistic Safety Studies for Zion 1 and 2 and Indian Point 2 and 3 were reviewed to evaluate the contribution of safety-related pipe failures in nonprimary coolant systems to overall plant risk. Safety-related piping failures were identified as contributing to the occurrence of initiating events as well as to the unavailability of plant systems to mitigate the consequences of initiating events.

The risk studies for Zion and Indian Point indicate that pipe failures contribute to the following internal initiating events for nonprimary coolant systems:

1. Steam break inside containment
2. Steam break outside containment
3. Loss of main feedwater

A feedwater break inside the containment is considered a subset of the steam break event since steam generator effects and plant response are similar and can be analyzed within the same plant event tree. Feedwater breaks outside containment are considered separately within the loss of main feedwater event.

The plant analyses evaluated the impact of piping failures on the availability of safety-related systems for accident sequences possibly leading to core melt and serious release. The following fluid systems were reviewed:

1. High Pressure Injection	6. Sodium Hydroxide Addition
2. Low Pressure Injection	7. Containment Fan Coolers
3. Accumulators	8. Component Cooling
4. Recirculation	9. Service Water
5. Containment Spray	10. Auxiliary Feedwater

INTERNAL INITIATING EVENTS

Frequencies of occurrence for internal initiating events involving safety-related pipe failures in nonprimary coolant systems are listed in the following table.

INITIATING EVENT FREQUENCIES PER UNIT YEAR

	<u>Zion 1 or 2</u>	<u>Indian Point 2</u>	<u>Indian Point 3</u>
Steam Break Inside Containment	9.04-4	1.95-3	2.16-3
Steam Break Outside Containment	9.04-4	1.95-3	2.16-3
Loss of Feedwater	5.17	6.70	3.80

NOTE: Numbers are listed in abbreviated scientific notation ($9.04 - 4 = 9.04 \times 10^{-4}$)

The frequency of initiating events used in the safety studies is based upon plant specific event data obtained from plant monthly operating reports and the NRC Operating Units Status Reports. The Zion Station data base was developed for approximately 11 unit-years of operation from commercial startup through December, 1979. The Indian Point Station event data base covers approximately 9 unit-years of operation through December, 1979.

Steam breaks inside or outside the containment have not occurred at either station or at the PWR plants included in the population data base. The tabulated frequency values were determined assuming a lognormal distribution and calculating the mean about the 5th and 95th percentiles considering individual plant specific unit-years of operation. The studies noted that a steam break is synonymous with a loss of steam which not only can be caused by main feedwater or steam line ruptures but also by steam generator relief valves failing open, steam dump valves failing open, or excessive steam demands. Conservatively assuming that all steam breaks result from pipe ruptures gives the maximum contribution of steam line failures as initiating events to degraded core states and release categories. The release frequencies per year with this assumption are tabulated in Table A2-1. Each release category has different risk consequences, depending on the circumstances of core melt and the status of containment safeguards.

Table A2-1

RELEASE FREQUENCY PER YEAR (INTERNAL EVENTS)

<u>Zion 1 or 2</u>	<u>Z-1</u>	<u>2</u>	<u>2R</u>	<u>2RW</u>	<u>Z-3</u>	<u>5R</u>	<u>Z-5</u>	<u>6</u>	<u>7</u>	<u>8A</u>	<u>8B</u>
Steam Break Inside Containment	9.48-17	5.02-13	2.79-11		4.14-14	7.59-13	5.42-13	6.00-15	1.05-12	3.98-12	5.95-9
Steam Break Outside Containment	<u>1.84-16</u>	<u>5.09-13</u>	<u>7.19-11</u>		<u>3.27-14</u>	<u>6.28-13</u>	<u>4.51-13</u>	<u>1.42-14</u>	<u>8.71-13</u>	<u>3.14-12</u>	<u>4.94-9</u>
Sum	2.79-16	10-1-13	9.98-11		7.41-14	13.9-13	9.93-13	2.02-14	1.92-12	7.12-12	10.9-9
Total Frequency	1.14-11	1.09-7	2.41-7		2.16-10	5.15-9	2.74-9	4.74-11	7.41-9	8.51-9	4.18-5
<u>Indian Point 2</u>											
Steam Break Inside Containment	7.45-17	1.99-11	1.31-14	3.18-11	3.77-14	1.97-11	1.94-12	1.31-14	5.45-11	2.58-11	1.98-7
Steam Break Outside Containment	<u>7.45-17</u>	<u>1.99-11</u>	<u>1.31-14</u>	<u>3.18-11</u>	<u>3.77-14</u>	<u>1.97-11</u>	<u>1.94-12</u>	<u>1.31-14</u>	<u>5.45-11</u>	<u>2.58-11</u>	<u>1.98-7</u>
Sum	14.9-17	3.98-11	2.62-14	6.36-11	7.54-14	3.94-11	3.88-12	2.62-14	10.9-11	5.16-11	3.96-7
Total Frequency	3.20-9	4.69-7	8.27-11	1.14-6	6.68-11	8.25-9	5.28-10	2.19-10	2.03-8	1.21-8	8.80-5
<u>Indian Point 3</u>											
Steam Break Inside Containment	4.67-17	1.56-11	1.81-14	2.08-11	1.62-14	1.42-11	1.41-12	6.27-15	3.94-11	8.88-12	1.56-7
Steam Break Outside Containment	<u>4.67-17</u>	<u>1.56-11</u>	<u>1.81-14</u>	<u>2.08-11</u>	<u>1.62-14</u>	<u>1.41-12</u>	<u>6.27-15</u>	<u>3.94-11</u>	<u>8.88-12</u>	<u>1.56-7</u>	
Sum	9.34-17	3.12-11	3.62-14	4.16-11	3.24-14	2.84-11	2.82-12	12.5-15	7.88-11	17.76-12	3.12-7
Total Frequency	2.18-9	4.77-7	8.23-11	6.98-7	7.19-11	1.15-8	8.26-10	1.34-10	2.92-8	5.79-9	1.25-4

From the table it is seen that steam breaks inside and outside containment are very small contributors to serious release. Using the results from Appendix 1, the risk from steam line ruptures can be determined.

RISK (EARLY AND LATENT FATALITIES)

	<u>Per Unit-Year from All Internal Events</u>	<u>Maximum Increment Per Unit-Year Due to Steam Pipe Failures</u>
Zion 1 or 2	9.94-4	2.78-7
Indian Point 2	6.67-3	4.46-7
Indian Point 3	5.00-3	4.58-7

It is concluded that steam break contributions to risk are at least two to four orders of magnitude less than the total risk from other internal initiating events. Consideration of external events further reduces the contribution of steam pipe failures to total risk.

Loss of main feedwater has been experienced a total of 58 times at Zion 1 and 2 during the 11 unit-years of operation included in the PRA data base. This gives a mean value of 5.17 occurrences per year. Feedwater pump problems, control of steam generator level, and steam/feedwater flow mismatch have been the principal contributors. Indian Point 2 and 3 have lost main feedwater a total of 47 times during 8 unit-years of operation. This results in a mean frequency of 6.7 occurrences per year at Indian Point 2 and 3.8 occurrences per year at Indian Point 3. In addition to the main contributors identified at Zion, Indian Point 2 and 3 have experienced feedwater flow regulator valve failures. Indian Point 2 also had one reported leak in the boiler feedpump header at a nipple connection.

The probability of a feedwater line rupture is very small and such an event has not occurred at Zion or Indian Point. If one assigns the probability to main feedwater line rupture as equaling the sum of the probabilities for a steam break inside and outside containment, a conservative estimate for main feedwater pipe failure is obtained since steam piping is a more extensive system. The resultant frequency of losing feedwater due to line ruptures is small as seen below. The consequent risk from feedwater line failures is negligible and was calculated using the same methodology as for steam line failures.

	Maximum Frequency Per Year of Losing Feedwater <u>Due to Pipe Failures</u>	Fatalities Per Unit-Year Due to <u>Feedwater Pipe Failures</u>
Zion 1 or 2	1.88-3	4.53-8
Indian Point 2	3.90-3	4.52-7
Indian Point 3	4.32-3	7.84-9

It is noted that rupture of feedwater piping is more severe than the typical loss of feedwater event, such as pump failure, and will reduce the redundancy provided by the auxiliary feedwater system. Evaluation of the auxiliary feedwater system (AFWS) in the Zion Safety Study included consideration of random pipe failures in the main feedwater lines leading to the steam generators. However, quantification of the reduced reliability in the AFWS due to an initiating event being a feedwater pipe rupture was not directly addressed. Operator action would be required to isolate the failed pipe section. It is estimated that the risk values tabulated above would increase a maximum of one order of magnitude.

FLUID SYSTEMS

The Zion and Indian Point probabilistic safety studies addressed the contribution of pipe failures to key fluid system unavailabilities. Based upon distributions defined in WASH 1400, the mean-failure frequency of a high-quality pipe section was calculated as:

Pipe (less than 3-inch in diameter):	8.60×10^{-9} /hour
Pipe (greater than 3-inch in diameter):	8.60×10^{-10} /hour

The pipe-failure modes include both line ruptures and plugging.

The tabulated component/failure data in the studies does not directly specify the mean-failure probability for 3-inch pipe. In order to be consistent with WASH-1400, the higher mean-failure probability was used.

Table A2-2 summarizes the contribution of pipe failures to fluid system unavailability as evaluated in the probabilistic safety studies for Zion and Indian Point. Where specific values for pipe failures were not given, estimates were made based upon the tabulated system effects of pipe failure in the studies. As seen for the majority of

Table A2-2

PIPING CONTRIBUTION TO SYSTEM UNAVAILABILITY

Fluid System	Zion 1 or 2			Indian Point 2			Indian Point 3		
	Increment Due To Pipe Failure	Percent Contribution To System Failure	Time At Risk	Increment Due To Pipe Failure	Percent Contribution To System Failure	Time At Risk	Increment Due To Pipe Failure	Percent Contribution To System Failure	Time At Risk
	(1) (2)	(3)		(1) (2)	(3)		(1) (2)	(3)	
High Pressure Injection (Medium LOCA)	(9.38-8)	6.84%	24 hr (4)	(4.82-8)	.016%	24 hr (4)	(4.73-8)	.026%	24 hr (4)
Low Pressure Injection	(6.45-8)	.014%	24 hr (4)	(6.79-8)	.008%	24 hr (4)	(4.73-8)	.006%	24 hr (4)
Accumulators	(6.90-13)	$<10^{-7}\%$	25 sec	(6.90-13)	$<10^{-7}\%$	25 sec	(6.90-13)	$<10^{-7}\%$	25 sec
Recirculation (High Head)	(4.3-9)	.001%	24 hr	1.44-7	.029%	1 week (5)	1.44-7	.004%	1 week (5)
Containment Spray	(5.16-8)	.009%	2 hr	6.80-9	.009%	2 hr	6.80-9	.023%	2 hr
Sodium Hydroxide Audition	(5.16-8)	.007%	2 hr	(1.72-8)	.002%	2 hr	(1.72-8)	.002%	2 hr
Containment Fan Coolers	-	-	-	-	-	-	-	-	-
Component Cooling	2.58-8	96.6%	24 hr (4)	3.72-7	3.72%	24 hr	1.46-8	13.3%	24 hr (4)
Service Water	2.15-8	100%	24 hr (4)	6.19-7	.248%	24 hr	3.71-7	.371%	24 hr
Auxilliary Feedwater	-	-	-	-	-	-	-	-	-

NOTES:

1. Numbers are in abbreviated scientific notation (e.g., 9.38-8 equals 9.38×10^{-8})
2. Numbers in parentheses are estimates based upon the tabulation of pipe failure effects on system unavailability in the safety studies.
3. Percent contribution of piping to system failure is computed on the basis of system unavailability with all essential safeguard buses and off-site power available.
4. Operator action is considered after the first hour of an initiating event to isolate failed piping as permitted by system valving.
5. Time at risk is for undetected failures prior to an initiating event.

systems, the increment of system failure due to piping is in the order of magnitude of 10^{-8} and ranges from 10^{-7} to 10^{-13} . This represents percent contributions from as little as less than $10^{-7}\%$ to 100% of system unavailability. It is observed that the pipe systems analyzed are highly reliable and do not significantly impact system unavailability, unless the performance of nonpiping components and human interaction approach or exceed that for piping. Contributing to this high reliability is minimization of the time at risk as justified in the safety studies.

From Table A2-2, pipe failures contribute significantly only to the probability of failure of two fluid systems at Zion:

1.	Component Cooling System	(96.6%)
2.	Service Water System	(100%)

To a smaller degree, piping failures also impact the component cooling system failure probabilities at Indian Point:

1.	Indian Point 2	(3.72%)
2.	Indian Point 3	(13.3%)

While the percent contribution of pipe ruptures to the remaining system unavailabilities is negligible, it is noted that the increment of failure due to piping is approximately the same order of magnitude (10^{-8}) for all the fluid systems. Passive pipe failures in the accumulator systems are an exception where they are significantly lower due to the very short time at risk.

Piping contributions to the containment fan cooler and the auxiliary feedwater system unavailabilities are not included in Table A2-2. This is due to a lack of some details in the safety studies which makes estimation of the values inappropriate. For calculation purposes, it is assumed that both systems have a maximum failure probability from piping of 1×10^{-8} .

In general, the studies show that single failures dominate piping contributions to system unavailability. Pipe ruptures which fail system function are of primary interest. These will require immediate shutdown should they occur in an operating system. Consequently, pipe failures occurring in these systems prior to an initiating event are not included in the failure analysis. For systems which in whole or part are

not functional during normal plant operation, the probability of a pipe failure occurring and remaining undetected prior to an initiating event must be considered. This is determined based upon the frequency of surveillance to verify flow continuity and pressure boundary integrity. Consideration of the probability of prior pipe failures in the high head recirculation system at Indian Point 2 and 3 led to one of the highest increments to system failure (1.44×10^{-7}).

In both safety studies, the plant analysis was limited to a maximum of 24 hours following an initiating event. This was done on the basis that plant conditions would stabilize within a day and that the impact and required response to subsequent system failures would no longer be critical considering the material resources, personnel, and time available for corrective action. Consequently, the maximum exposure time to failure for a given pipe system was 24 hours. In several system analyses, operator response to pipe failures was included after the first hour into the accident sequence. Table A2-2 identifies those systems where this operator action was considered. This has the effect of reducing the time at risk of pipe sections which can be isolated to one hour. In several cases isolation is not available and the failure probability was determined on the basis of 24 hours. The piping analyses also considered those areas where isolation capability was available to save an interfacing system, but not the system experiencing the line failure.

Zion Station

The following discussion quantifies the contribution of random pipe failures in nonprimary coolant systems to overall plant risk. The analysis was performed by reviewing the plant event trees in the Zion Safety Study and calculating the frequency of plant damage states with and without inclusion of system pipe ruptures. The difference in plant damage state frequencies and the results of Appendix 1 permit quantification of the consequent risk from pipe failures. In some plant sequences, operator action is required, with success depending on operator response as well as component/system function. The impact of high pressure safety injection pipe failures on operator action is not included in this report to simplify calculations. After identification of dominating pipe failure sequences, it was verified that these potential pipe ruptures have negligible impact on pipe-induced risk.

The analysis was carried out separately for internally and externally initiated plant accident sequences involving failed, safety-related, fluid systems.

Internal-Initiated Accident Sequences. In calculating the contribution of random pipe failures to internally initiated risk, the following accident sequence categories were reviewed:

1. Large, medium, and small LOCA sequences with all essential power buses available, involving pipe failures in the component cooling water or service water systems.
2. All event tree sequences (with the exception of item 3 below) involving loss of all essential safeguard power buses due to service water system pipe failure.
3. Turbine trip from loss of service water sequences.
4. All dominant plant event tree sequences identified in the Zion PRA involving failure of safety-related fluid systems.

The analysis was limited to these sequences for convenience of calculation (there are over 6500 endpoints) and because most of the significant sequences involving system pipe failures are bracketed. Conservatism in the calculations balance out any instances where contributing sequences may not be included. A major conservatism in the analysis was calculation of the probability of accident sequences, including the dominating ones involving turbine trip from loss of off-site power, without benefit of electric power bus recovery. This gives higher plant damage state frequencies and overstates the resultant risk. Review of the event tree quantification summaries in Section 1.3.4 of the Zion Study indicates an average reduction in plant damage state frequency of 2.9×10^2 when power recovery techniques are employed. It is noted that sequences resulting from a turbine trip due to loss of service water were an exception, where their probabilities were determined using electric power recovery methods.

The following discussions summarize the analysis results of the four sequence categories identified at the top of this page.

Piping failures contribute significantly to component cooling and service water unavailability. The Zion Safety Study was reviewed to determine system failure probabilities when eliminating the effects of pipe ruptures. The results are tabulated below where the system values are seen to be dependent on the availability of the three essential, safeguard, power buses.

<u>System</u>	<u>Power Bus Availability</u>	<u>System Unavailability</u>	
		<u>Including Pipe Failures</u>	<u>Without Pipe Failures</u>
Component Cooling	3 or 2 Buses	2.67-8	8.62-10
	1 bus	4.21-2	4.21-2
	No Buses	1.0	1.0
Service Water	Any Combination	2.2-8	3.0-11
	No Buses	1.0	1.0

NOTE: Numbers are in abbreviated scientific notation (e.g., 2.67-8 equals 2.67×10^{-8}).

From the above tabulation, pipe ruptures impact component cooling system failure only for the cases where either all three, or two, of the essential safeguard power buses are available. Pipe failures contribute significantly to service water unavailability for all power states with the exception of the case where none of the essential buses are available.

The component cooling water and service water systems appear directly only in the LOCA plant event trees. Only those branches with all power buses available are considered, since the majority of significant sequences involving pipe failures will be included. This is because degraded power states are less likely to occur and the probability of pipe failures is independent of electric power conditions. The increase in plant damage state frequencies for LOCA events (with all essential safeguard buses available) due to random pipe ruptures involving component cooling water and service water system failures is summarized in Table A2-3. The resultant risk is 1.40×10^{-9} fatalities per unit-year.

The majority of plant event trees (Number 4 through 14) do not directly include consideration of the component cooling water or service water systems. The event trees do include the impact of the containment fan coolers (CF-2). They are dependent on the service water system and have a conditional probability of failure which is dependent on the unavailability of the recirculation cooling water system (R-3). The recirculation system includes the residual heat removal system, containment and recirculation sumps, the safety injection system, and the containment spray system. It is dependent on both the service water and the component cooling water systems for heat removal.

Table A2-3

ZION INTERNAL RISK FROM RANDOM PIPE FAILURES IN FLUID SYSTEMS

CONTRIBUTING SEQUENCES INVOLVING PIPE FAILURES	PLANT DAMAGE STATE FREQUENCY PER YEAR										FATALITIES PER UNIT-YEAR
	AEFC SERC	AEF SEF	AEC SEC	AE SE	ALFC SLFC	ALF SLF	ALC SLC	AL SL	TEFC TEC	TEF TE	
(1)	3.40-17 3.02-17	1.87-21 1.66-21	3.79-14 4.64-15	2.58-18 1.89-17	4.93-11 9.15-10	2.68-15 2.10-19	4.09-11 7.78-10	2.25-15 9.32-14	0.0 0.0	0.0 0.0	1.40-9
(2)	0.0 0.0	0.0 00	0.0 0.0	1.45-11 1.79-11	0.0 0.0	0.0 0.0	0.0 0.0	0.0 1.76-17	0.0 0.0	0.0 1.09-9	3.06-6
(3)	0.0 0.0	0.0 0.0	0.0 0.0	0.0 0.0	0.0 0.0	0.0 0.0	0.0 1.49-12	0.0 8.19-17	2.41-9 2.41-11	0.0 3.33-10	9.22-7
(4)	1.21-10 3.32-9	9.73-14 3.13-16	0.0 1.10-15	2.22-16 1.55-21	1.88-11 2.20-10	5.02-13 6.75-13	4.14-11 6.61-13	2.21-17 6.08-15	1.10-9 3.46-13	1.19-13 1.31-13	3.44-9

NOTES:

- (1) Large, medium, and small LOCA sequences involving pipe failures in the component cooling water and service water (SW) systems.
- (2) All event tree sequences involving loss of all essential safeguard power buses due to service water system pipe failure.
- (3) Turbine trip from loss of service water sequences due to system pipe failure (Event Tree 11c).
- (4) All event tree dominating sequences as identified in the Zion PRA (with the exception of the above) involving pipe failures in safety-related fluid systems.

Based on the following discussion, it was concluded that pipe failures in the component cooling and service water systems have negligible effect on the unavailability of either the recirculation cooling or the containment fan cooling systems at Zion. On page 1.3-36 of the Zion PRA, the unavailability of the recirculation cooling system (R-3) is defined as:

$$(R-3) = (R-2 \mid \text{Fan Coolers Failed}) (CF-1 \mid \text{SW Succeeds}) \\ + (R-2 \mid \text{Fan Coolers Succeed}) + SW + CC$$

Where: CF-1 = Containment Fan Cooler Unavailability for LOCA Events
 R-2 = High Head Recirculation Unavailability
 SW = Service Water Unavailability
 CC = Component Cooling Unavailability

Review of the results for all essential bus-power states indicates that the probability of service-water failures is a negligible contributor (less than .005%) to recirculation cooling system unavailability. Similarly, the component cooling contribution to recirculation system failure is negligible (less than .007%) with the exception of two degraded essential power-bus states. For both exceptions, component cooling significantly impacts recirculation unavailability. However, under these degraded bus conditions, piping failures no longer dominate the failure probability of the component cooling system. Consequently, it can be concluded that the probability of pipe failures in the component cooling and service-water systems have negligible impact on recirculation cooling system unavailability.

The unavailability of the containment fan-cooling system (CF-2) for plant transients is also defined on page 1.3-36 of the Zion PRA. The expression is:

$$(CF-2) = \{(R-2 \mid \text{Fan Coolers Failed}) + SW\} \{(CF-1 \mid \text{SW Succeeds}) + SW\} / (R-3)$$

Again, service water contribution to containment fan cooling unavailability is small. The impact of the component-cooling system is indirectly included in the recirculation cooling system (R-3). It was determined in the above analysis that failure of the recirculation system is not significantly influenced by pipe ruptures within the

component-cooling system. Consequently, the probability of pipe failures in the component cooling and service water systems have negligible impact on the unavailability of the containment fan cooling system.

Service water supply is required for several other system components, including control room air conditioning, AFWS motor-driven pumps, penetration pressurization air compressors, and safety-related equipment room coolers. However, failure of these components is less severe (and from the foregoing discussion, it is concluded that random pipe ruptures in the component cooling water and service water systems have negligible impact on other system unavailabilities for non-LOCA accident sequences).

The second sequence category identified on page A2-9 addresses the risk from loss of all AC power due to service water pipe failures. The station service water system supplies cooling water to the essential safeguard, diesel generator engine water jackets, and the lube oil coolers. Each diesel engine has two service water supply lines, one from each unit header. The mean service water system failure frequency due to line ruptures in either header was quantified in Section 1.5.2.3.8 of the Zion Study as 2.15×10^{-8} per event. The consequent risk from failure of the electrical system occurs when there is a coincident failure in the normal electrical supply path or transformer system or a concurrent loss of off-site AC power. This leads to a loss of all AC power and fails all dependent, safety-related systems. It is noted that all initiating events considered in the safety analysis involve turbine trips and thus have a potential for leading to a loss of off-site power.

With the exception of the loss of off-site power initiating event, the following expression defines the probability of losing all AC power (PLAAP):

$$\text{PLAAP} = \text{QN} \times \text{PLOP} + \text{QA} \times (1 - \text{PLOP}) \quad (\text{EQ 2A-1})$$

Where: QN = Probability of losing all essential buses when off-site power is not available.

QA = Probability of losing all essential buses when off-site power is available.

PLOP = Probability of losing off-site power to the plant due to turbine trip.

The mean probability of losing off-site power to the switchyard following a turbine trip is 3.38×10^{-4} (Section 1.5.2.2.14.8 of the Zion PRA). From Table 1.5.2.2.1-2C in the Study, the frequency of losing all essential buses when off-site power is available is 5.49×10^{-9} . Thus, Equation 2A-1 becomes:

$$\text{PLAAP} = 3.38 \times 10^{-4} \text{ QN} + 5.49 \times 10^{-9}$$

For the off-site power available boundary condition, the Zion Study does not directly address the contribution of service water pipe failure to the loss of all AC power at the station. It is assumed that the impact of service water pipe failure during a loss of off-site power from turbine trip is substantially more severe and will dominate. The failure probability of service water from piping is 2.15×10^{-8} per event. Thus, the probability of losing all AC power due to service water pipe ruptures for an internal initiating event is:

$$\begin{aligned} \text{PLAAP (Pipe)} &= (3.38 \times 10^{-4}) \times (2.15 \times 10^{-8}) \\ &= 7.27 \times 10^{-12} \end{aligned}$$

The resulting frequency per year in losing all AC power due to service water pipe failures from all internal initiating events (excluding loss of off-site power initiating event) is:

$$(14.02) \times (7.27 \times 10^{-12}) = 1.02 \times 10^{-10}$$

where 14.02 is the sum of the yearly frequencies of all internal initiating events except loss of off-site power.

The frequency per year in losing all AC power due to service water pipe failures during loss of off-site power initiating events is:

$$(5.76 \times 10^{-2}) \times (2.15 \times 10^{-8}) = 1.44 \times 10^{-9}$$

where 5.76×10^{-2} is the yearly frequency of losing off-site power.

Since there is only one order of magnitude separating the two yearly frequencies, all plant event trees were reviewed to quantify the risk associated with a loss of all AC power due to service water pipe failures. The frequency of the resultant plant damage states is summarized in Table A2-3. The consequent risk is 3.08×10^{-6} fatalities per unit-year and is dominated by the loss of off-site power initiating event.

The third sequence category quantifies the risk associated with turbine trips caused by a loss of service water (Event Tree 11c). The Zion Study specifies a yearly frequency of 9.4×10^{-4} for this initiating event. System pipe failures dominate service water failure such that essentially all risk resulting from the initiating event is due to pipe ruptures.

As a check to the initiating event frequency used in the Zion Study, it is noted that the service water system failure probability due to piping is 2.15×10^{-8} per event. This number represents system failure probability during the first hour into an accident sequence. After the first hour, operator-isolation action is permitted, essentially eliminating further exposure to failure. Consequently, 2.15×10^{-8} also represents the hourly frequency of service water pipe failure. Maximum plant operation is an optimistic 8760 hours a year. This gives a yearly pipe rupture frequency of 1.88×10^{-4} and indicates that the value used for the initiating event frequency is conservative by a factor of 5, as a minimum. The resulting plant damage state frequencies are summarized in Table A2-3 with a consequent risk of 9.22×10^{-7} fatalities per unit-year.

The last sequence category evaluates the risk from pipe failures in safety-related fluid systems for all dominating plant event tree sequences identified in the Zion PRA. Sequences addressed in the first three categories are not included. Approximately 165 dominant sequences were identified in the Study. Those involving a fluid system failure were quantified with and without inclusion of the incremental piping contribution to system failure as tabulated in Table A2-2. The difference in plant damage states due to pipe failures is listed in Table A2-3. The attendant risk is 3.44×10^{-9} fatalities per year.

External-Initiated Accident Sequences. The Zion Probabilistic Safety Study quantifies the risk from earthquakes and fires. Other external events such as tornadoes, floods, aircraft, and transportation of hazardous materials are identified as negligible contributors to plant risk.

The seismic safety analysis is discussed in detail in Section 9 of this report. It is noted that, in general, supported piping systems have median horizontal ground acceleration capacities significantly greater than the design basis earthquake (.17 g) as well as the upper median earthquake (.65 g). Buried piping has a median capacity of 1.40 g which

is lower than most supported systems. Buckling failure of the pipe occurs due to relative displacement between plant structures and the soil. With a deterministic point analysis, buried piping does not contribute to risk; however using a probabilistic approach, there is a 25% possibility (based on median fragility curve) of failure at a horizontal ground acceleration of .65 g. The resultant risk is negligible compared with other dominating seismic sequences. Considering the nature of the failure, it is soil/structure interaction, not piping, which governs the failure mode. Similarly, soil failure beneath the reactor building causes failure of interconnecting pipe systems. It is concluded that soil/structure interaction rather than pipe failures contribute to the seismic risk discussed above.

Loss of all AC power due to ceramic insulator failure in conjunction with service water pump failure is the dominating seismic sequence. Random pipe failures during this accident sequence have negligible impact on plant risk.

The fire analysis for the Zion Station is limited to those areas where the most damage can occur and result in the greatest risk. Areas analyzed include the auxiliary electrical equipment room and the cable spreading rooms. The Safety Study concludes that the occurrence of LOCA events due to fires is negligible and that plant transient behavior dominates accident sequences. Simplified event trees with initiating event frequencies and system unavailabilities are shown in Figure 2A-1. The analysis assumes all essential safeguard power buses are initially available. The incremental risk due to random pipe failures during fires was determined by calculating the difference in plant damage states with and without inclusion of pipe failures. The results are:

<u>Plant Damage State</u>	<u>Mean Yearly Frequency Due to Pipe Failures</u>	<u>Early and Latent Fatalities Per Unit-Year</u>
TEFC	4.10-11	2.67-11
TEF	1.47-13	4.31-13
TEC	1.20-11	7.88-12
TE	1.85-13	5.11-10
Total		5.45-10

Figure A2-1
SIMPLIFIED EVENT TREES FOR FIRES

(a) Fire in Cable Spreading Rooms

(b) Fire in Auxiliary Electrical Equipment Room

Notes: (1) Success of an event is up, failure is down

(2) Numbers in parenthesis are maximum contribution to system failure from random pipe ruptures.

Indian Point

While the percent contribution of pipe failures to system unavailabilities is lower at Indian Point, the mean frequency of system failure from piping is somewhat higher than Zion for the recirculation, component cooling, and service-water systems. This is due to differences in allowing operator action to isolate failed pipe sections and inclusion of undetected pipe failures (high-head recirculation) prior to an initiating event.

It is also noted that serious plant damage states (SE, TE, AE, and AL) at Indian Point result in more severe fatality consequences (1½ to 2 times) than Zion. It is estimated that the contribution of system pipe ruptures to these degraded conditions will result in higher values for plant risk as summarized in the next section.

SUMMARY OF RESULTS

From the foregoing analysis, the contribution to risk (early and latent fatalities) from random pipe failures in nonprimary coolant, safety-related systems is:

	<u>FATALITIES PER UNIT YEAR</u>		
	<u>Zion 1 or 2</u>	<u>Indian Point 2</u>	<u>Indian Point 3</u>
Internal Initiating Events	3.23-7	8.98-7	4.65-7
Fluid Systems			
Internal Events	4.00-6	4.50-4 (1)	3.90-4 (1)
External Events	5.45-10	- (2)	- (2)
TOTAL	4.32-6	4.51-4	3.90-4

NOTES:

- (1) At Zion, internal risk from random pipe failures is dominated by turbine trip from loss of off-site power. Indian Point values are estimated by using the ratios of the initiating event frequency, increment of service water failure due to piping, and fatality consequences times the Zion risk.
- (2) External risk from random pipe failures at Indian Point will be greater than the risk at Zion due to the greater frequency of external initiating events and the more severe consequences. However, as with Zion, the contribution is estimated to be negligible compared with internal risk.

Appendix 3

PWR LPIS AND CSIS FAILURE PROBABILITIES FROM WASH 1400

LOW PRESSURE INJECTION SYSTEM

The principal discussion of the Low Pressure Injection System is found on pages II-136 through II-142, and in Figures II 5-31 through II 5-35, and in Table II 5-16. The basic system diagram (Figure II 5-31 from WASH 1400) is reproduced on the following page.

Combining the information on Figure II 5-32 "Reduced Fault Tree" and Table II 5-16 "Event Probabilities Used in the LPIS Fault Tree Evaluation" yields the following significant contributors to the failure of a single LPIS train:

<u>Failures</u>	<u>Probability</u>
Various valve hardware failures	0.3×10^{-3}
Various valves mispositioned (human error)	1.8×10^{-3}
Pump failure	1.0×10^{-3}
Pump electrical failures	1.3×10^{-3}
Train automatic actuation failure	5.2×10^{-3}
Total	9.6×10^{-3}

Major contributors to system failure are as follows:

- Q doubles - This is the probability of random failure in both trains and is equal to $(9.6 \times 10^{-3})^2 = 9.2 \times 10^{-5}$.
- Q maintenance and test - Testing does not affect train availability. Train unavailability due to maintenance is calculated as 4.3×10^{-3} , so system unavailability is $(2)(4.3 \times 10^{-3})(9.6 \times 10^{-3}) = 8.3 \times 10^{-5}$. A single failure, mispositioning of a valve during testing, is included here and has a probability of 8.8×10^{-4} ; so the total probability is 9.6×10^{-4} .

FIGURE II 5-31 Simplified System Diagram

Q singles - This is the probability that one of the normally open valves in the common suction line will be left closed and is calculated to be 3.1×10^{-3} .

Q common mode - This is due to common mode faults in the LPIS automatic actuation system and is 4.5×10^{-5} .

The overall system failure probability is the sum of these four probabilities.

doubles	$.09 \times 10^{-3}$	
maintenance & test	$.96 \times 10^{-3}$	
singles	3.1×10^{-3}	
common mode	$.05 \times 10^{-3}$	
	4.2×10^{-3}	(the Monte Carlo simulation gave 4.7×10^{-3})

The system fault tree shows approximately 20 pipe sections, but does not specify the failure probabilities, simply indicating that they are negligible.

CORE SPRAY INJECTION SYSTEM

The principal discussion of the Containment Spray Inspection System is found on pages II-111 through II-116 in Figures II-5-15 through II-5-19 and in Table II-5-10. The basic system diagram (Figure II-5-15 from WASH 1400) is reproduced on the following page.

Combining the information of Figure II-5-16 "CSIS Reduced Fault Trees" and Table II 5-10 "Event Probabilities Used in the PWR CSIS Fault Tree Evaluation" yields the following significant contributions to the failure of a single train:

<u>Failure</u>	<u>Probability</u>
Manual Valve Left Closed	1.0×10^{-3}
Isolation Valve Closes Spontaneously	0.1×10^{-3}
Test Line Manual Valve Left Open	10.0×10^{-3}
Filter Plugged	0.1×10^{-3}
Motor Control Failures	1.0×10^{-3}
Motor Clutch Failures	0.3×10^{-3}

FIGURE II 5-15 CSIS Simplified Flow Diagram

<u>Failure</u>	<u>Probability</u>
Pump Fails to Start	1.0×10^{-3}
Check Valve Stuck Closed	0.1×10^{-3}
Spray Nozzles Plugged	0.1×10^{-3}
Train Automatic Actuation Fails	4.6×10^{-3}
	<hr/>
	1.8×10^{-2}

Major contributions to system failure as follows:

- Q doubles - This is the probability of random failure of both trains and is equal to $(1.8 \times 10^{-2})^2 = 3.2 \times 10^{-4}$.
- Q maintenance and test - WASH 1400 calculates the probability that one train is unavailable due to maintenance or test as 4.1×10^{-3} . Thus the probability that both trains fails is $(2)(4.1 \times 10^{-3})(1.8 \times 10^{-2}) = 1.5 \times 10^{-4}$.
- Q singles - Since each train has a separate suction line from the Refueling Water Storage Tank, there are no single pipe failures which can defeat the system. The probability of tank failure is considered negligible, so the single failure probability is dominated by plugging of the tank vent and is calculated to be 4.4×10^{-7} .
- Q common mode - WASH 1400 gives the probability that all CSIS actuation instruments are miscalibrated as 1×10^{-3} and the probability that valves in both test lines have been left open as 9×10^{-4} . Thus the probability of common mode failure is 1.9×10^{-3} .

The overall system failure probability is the sum of the four probabilities:

doubles	$.32 \times 10^{-3}$
maintenance & test	$.15 \times 10^{-3}$
singles	$.0004 \times 10^{-3}$
common mode	1.9×10^{-3}
	<hr/>
	3.4×10^{-3}

Pipe failures contribute negligibly to system failure and are not explicitly quantified.

Appendix 4

FLUID SYSTEM UNAVAILABILITY AT ZION AND INDIAN POINT

HISTORICAL BACKGROUND

Zion 1 and 2 are operated by Commonwealth Edison Company and are located 40 miles north of Chicago. Both units are Westinghouse pressurized water reactors (PWR) with Westinghouse supplied turbine-generators having a net rating of 1040 MW. Both units received their construction permits in December, 1968, with commercial operation in October, 1973, for Unit 1 and November, 1973, for Unit 2.

Indian Point 2 is owned and operated by Consolidated Edison Company while Indian Point 3 is operated by the Power Authority of the State of New York. The units are located approximately 25 miles north of New York City. Both PWR units have reactor and turbine-generators supplied by Westinghouse. Unit 2 and 3 have net-capacity ratings of 873 MW and 965 MW respectively. Indian Point 2 received its construction permit in October, 1966, with commercial operation in July, 1974. The construction permit for Unit 3 was obtained in August, 1969, with commercial operation in August, 1976.

FLUID SYSTEM UNAVAILABILITY AND CONFIGURATION

The risk analyses for Zion and Indian Point quantified the failure probability of fluid systems which are critical in determining the outcome of accident sequences. Modeling included simplifications to establish system boundaries, to determine critical components, and to identify the various system functions and conditions leading to failure. It is important to keep in mind that the risk studies do not distinguish between safety-related and nonsafety-related system components to mitigate initiating events, with subsystems analyzed within the limitations of their specific design configuration.

Table A4-1 summarizes system unavailabilities for the emergency core cooling, containment safeguard, auxiliary feedwater, and other support systems at Zion and Indian Point. Unavailability values are tabulated for conditions where off-site power and all essential, safeguard power buses are available. There is a strong system dependency on electric power availability, with a substantial increase in most system failure probabilities at degraded electric power states.

Table A4-1

SYSTEM UNAVAILABILITY WITH ALL ESSENTIAL SAFEGUARD BUSES AVAILABLE

<u>FLUID SYSTEMS</u>	<u>ZION 1 OR 2</u>		<u>INDIAN POINT 2</u>		<u>INDIAN POINT 3</u>	
High Pressure Injection						
Medium LOCA	1.37-6	(M)	4.14-4	(V/P)	1.81-4	(V)
Small LOCA	7.4-9	(V)	1.86-4	(V/P)	1.35-4	(V)
Low Pressure Injection	4.65-4	(H/V)	8.7-4	(V/P)	8.11-4	(V/P)
Accumulators	9.5-4	(V)	1.88-3	(V)	2.22-3	(V)
Recirculation						
Large/Medium LOCA	5.16-3	(H)	5.43-3	(H)	5.32-3	(H)
Small LOCA	3.9-4	(H/M)	4.93-3	(H)	4.05-3	(P)
Containment Spray	5.5-5	(H)	7.5-5	(P/V/H)	3.0-5	(H/P/V)
Sodium Hydroxide Addition	7.5-4	(T)	9.6-4	(V/E/H)	9.6-4	(V/E/H)
Containment Fans	6.1-7	(M)	1.4-6	(V)	6.3-7	(V)
Component Cooling	2.67-8	(PP)	1.0-5	(P/V/T)	1.1-7	(PP/P/V)
Service Water	2.2-8	(PP)	2.5-4	(P/H)	1.0-4	(V/P/H)
Auxiliary Feedwater	4.21-6	(P/M)	1.96-5	(V/H)	1.54-5	(V/H)

NOTES:

- (1) Numbers are in abbreviated scientific notation (e.g., 1.37-6 equals 1.37×10^{-6})
- (2) System unavailabilities are per initiating event.
- (3) System values tabulated are for conditions with off-site power and containment fans available.
- (4) Main contributors to system unavailability are identified with the following abbreviations:

(M) = Maintenance

(V) = Valve

(PP) = Piping

(T) = Tank

(H) = Human Error

(P) = Pumps

(E) = Eductor

(F) = Fan Cooler Unit Trains

From Table A4-1, it is observed that system failure probabilities per event are comparable at all the units with three exceptions. These are high pressure injection, component cooling, and service water. In all cases, system unavailabilities are much lower (better) at Zion than at Indian Point 2 or 3. The reasons are:

High Pressure Injection (HPI)

At Indian Point, high pressure injection is accomplished using one (small LOCA) or two (medium LOCA) of the three HPI pumps, while at Zion one or two of four pumps (two HPI and two charging) are used. The greater level of redundancy at Zion results in improved system availability. It is interesting to note that each unit at Indian Point has a total of six pumps (three HPI and three charging), while each unit at Zion has only four pumps. By utilizing a nonsafety system to back up a safety system, Zion achieves higher levels of safety at what should be significantly reduced costs.

Service Water (SW)

At Zion the service water system is headered, not only between trains but also between the two units. While there is headering between trains at Indian Point, there are no connections between the units which significantly reduces system availability. This is seen with the impact of random pump failures where system failure requires the loss of four out of six pump trains at Zion, but only two out of three pump trains in the nuclear header at Indian Point 2 or 3. This results in an average service water unavailability at Indian Point of almost eight thousand times greater than Zion. If one essential power bus is lost, the ratio in system unavailabilities increases dramatically to almost six hundred thousand. This occurs because the service water system unavailability at Zion essentially remains unchanged (2.2×10^{-8}), while the average unavailability at Indian Point increases to 1.28×10^{-2} . Loss of an individual essential bus at Zion does not contribute significantly to service water failure since the system is designed with each of the six pumps supplied by separate essential buses from both units. Consequently, in this case, it is clear that current NRC perceptions prohibiting cross connection of systems can significantly reduce safety. It should be noted that the Zion analysis utilizes pipe failure rates on the optimistic side of those discussed in Section 6 of this report. However, more pessimistic pipe failure rates will not alter the conclusions.

Component Cooling (CC)

The component cooling water system is also headered between units at Zion but not Indian Point, giving Zion a higher system availability. The shared system at Zion utilizes five pumps versus six at Indian Point (three at each unit) and thus achieves a higher availability at a lower cost. It is noted that system unavailability at Indian Point 2 is almost one hundred times greater than at Indian Point 3. A contributing factor is the two supply header design at Indian Point 3 which reduces the impact of single hardware (piping/valves) failures.

As noted previously, the unavailabilities of the remaining systems are comparable. Reasons for unit differences include:

1. Site specific component failure rates.
2. Differences in surveillance techniques or frequency which impact the risk from undetected failures prior to an initiating event.
3. Design choice within the NRC regulations.

Appendix 5

FUEL HANDLING BUILDING FILTRATION SYSTEM OFF-SITE CONSEQUENCE ASSESSMENT

The offsite consequences due to radioactivity releases from a damaged spent fuel-assembly in the spend fuel pool are evaluated as follows:

$$D_T = \sum_i K_i Q_i (X/Q) \quad \text{Thyroid Dose}$$

Expected Population Dose ————— Thyroid:

$$\begin{aligned} (PD)_T &= \int_{\theta} \int_x f(\theta) D_T(x, \theta) \eta(x, \theta) dx d\theta \\ &= \sum_i K_i Q_i \int_{\theta} \int_x f(\theta) (X/Q) \eta(x, \theta) dx d\theta \end{aligned}$$

Notations used in the above equations are as follows:

K_i	thyroid dose-conversion factor for nuclide i
Q_i	quantity of nuclide i released to environment
(X/Q)	atmospheric dilution factor
x	downwind distance (radial) from the release-point
θ	angular coordinate (sectorial)
$f(\theta)$	probability of "finding" a certain meteorological condition for the sector θ
$\eta(x, \theta)$	population distribution around the release-point

The evaluation of K_i , Q_i can be derived from Table 2.1-1 of NUREG-0612 (Ref. 35) which gives the following information regarding the thyroid dose at the exclusion radius from a damaged fuel assembly (all rods damaged) for a 3000 MW PWR.

<u>Days Subcritical</u>	<u>Dose (Rems)</u>	
	<u>With Filters</u>	<u>Without Filters</u>
4	8.63	173
40	0.39	7.74

These results must be adjusted to account for differences in assumptions used in NUREG-0612 and those appropriate to risk assessments. These differences in assumptions are shown below:

	<u>NUREG 0612</u>	<u>PRA</u>
Filter efficiency	95%	99%
Partition factor for I in water	100	500
Fuel element gap activity available for release	10%	1%
Recipient breathing rate	$3.47 \times 10^{-4} \text{ m}^3/\text{sec}$	$2.32 \times 10^{-4} \text{ m}^3/\text{sec}$
Atmospheric dilution factor (X/Q)	$1 \times 10^{-3} \text{ (EAB)}$	appropriate value

In addition, the following differences are accounted for:

	<u>Damage</u>	<u>Decay Time</u>
Fuel Assembly Drop	One row of rods	7 days
Load Drop	All rods	30 days

Correcting for these effects gives $\sum K_i Q_i$

	<u>With Filter</u>	<u>Without Filter</u>
Fuel Assembly Drop	1.1 rem/sec/m^3	21.6 rem/sec/m^3
Load Drop	5.66 rem/sec/m^3	49.2 rem/sec/m^3

For atmospheric dilution, ground level releases and a Gaussian plume model are assumed. The atmospheric dilution factor is given as:

$$X/Q = \frac{2}{3(2\pi)^{1/2} \sigma_y \sigma_z \bar{u}}$$

Where:

σ_y	=	horizontal standard derivation of the Gaussian plume
σ_z	=	vertical standard derivation of the Gaussian plume
$\bar{u}$	=	mean wind speed

The values of σ_y and σ_z were determined as a function of downwind distance and atmospheric stability class (six stability classes - A through F - are used) from correlations in Appendix VI of WASH 1400. The resultant values are given in Table A5-1.

Indian Point site related data was selected because this site has relatively poor diffusion characteristics and a high surrounding population density. The meteorological data used is that presented in Table 6.2-4 of the Indian Point PRA. The demographic data used is also based on the Indian Point PRA (Table 6.2-5); however, only 16 sectors and 10 radial regions are used (one mile segments from 0 to 5 miles, one segment from 5 to 10 miles, and ten mile segments from 10 to 50 miles). This resultant population data is given in Table A5-2.

Based on the meteorological data, the values of $f(\theta)$ are calculated as functions of the atmospheric stability class. Table A5-3 gives these values - along with the associated mean-wind speeds - as functions of sector and stability class. Using this data and the meteorological data from Indian Point, the values of $f(\theta) (X/Q) (x, \theta)$ can be computed. These values are given in Table A5-4. These results can be combined with the population data in Table A5-3 to form the integral

$$\int_{\theta} \int_{x} f(\theta) (X/Q) (x, \theta) \eta(x, \theta) dx d\theta$$

The steps in this integration are given in Table A5-6 with the result given as 2.1 man - (sec/m³).

Combining this result with $\sum K_i Q_i$ as given above results in the dose in man-rem.

	Dose- Man Rem	
	<u>With Filters</u>	<u>Without Filters</u>
Fuel Assembly Drop	2.31	45.36
Load Drop	11.89	103.3

Table A5-1

<u>Distance</u> <u>(meters)</u>	<u>Calculations of σ_y (meters)</u>					
	Stability Classes					
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>E</u>	<u>F</u>
804.7	1.539 + 02	1.158 + 02	8.791 + 01	6.190 + 01	4.402 + 01	3.038 + 01
2414	4.151 + 02	3.122 + 02	2.371 + 02	1.669 + 02	1.187 + 02	8.194 + 01
4023	6.584 + 02	4.952 + 02	3.760 + 02	2.648 + 02	1.883 + 02	1.300 + 02
5633	8.924 + 02	6.711 + 02	5.096 + 02	3.588 + 02	2.552 + 02	1.761 + 02
7242	1.120 + 03	8.420 + 02	6.394 + 02	4.503 + 02	3.202 + 02	2.210 + 02
12070	1.776 + 03	1.336 + 03	1.014 + 03	7.142 + 02	5.078 + 02	3.505 + 02
24140	3.321 + 03	2.498 + 03	1.897 + 03	1.336 + 03	9.497 + 02	6.555 + 02
40230	5.268 + 03	3.962 + 03	3.008 + 03	2.118 + 03	1.506 + 03	1.040 + 03
56330	7.139 + 03	5.369 + 03	4.077 + 03	2.871 + 03	2.041 + 03	1.409 + 03
72420	8.957 + 03	6.736 + 03	5.115 + 03	3.602 + 03	2.561 + 03	1.768 + 03
804.7	2.819 + 02	8.726 + 01	5.013 + 01	2.678 + 01	1.779 + 01	1.159 + 01
	<u>Calculations of σ_z (meters)</u>					
2414	2.899 + 03	2.869 + 02	1.364 + 02	5.712 + 01	3.840 + 01	2.475 + 01
4023	8.465 + 03	5.011 + 02	2.172 + 02	7.827 + 01	5.061 + 01	3.181 + 01
5633	1.714 + 04	7.242 + 02	2.951 + 02	9.558 + 01	5.975 + 01	3.683 + 01
7242	2.901 + 04	9.537 + 02	3.710 + 02	1.106 + 02	6.722 + 01	4.078 + 01
12070	8.458 + 04	1.670 + 03	5.909 + 02	1.479 + 02	8.429 + 01	4.939 + 01
24140	3.611 + 05	3.572 + 03	1.111 + 03	2.171 + 02	1.121 + 02	6.241 + 01
40230	1.052 + 06	6.256 + 03	1.769 + 03	2.865 + 02	1.368 + 02	7.310 + 01
56330	2.129 + 06	9.053 + 03	2.404 + 03	3.433 + 02	1.552 + 02	8.070 + 01
72420	3.604 + 06	1.193 + 04	3.023 + 03	3.926 + 02	1.703 + 02	8.669 + 01

Table A5-2

INDIAN POINT POPULATION DATA

Sectors	Distances (miles)									
	<u>0-1</u>	<u>1-2</u>	<u>2-3</u>	<u>3-4</u>	<u>4-5</u>	<u>5-10</u>	<u>10-20</u>	<u>20-30</u>	<u>30-40</u>	<u>40-50</u>
N	0	139	1350	16	384	16006	23497	29302	28176	34818
NNE	0	34	116	106	1272	5353	20397	30996	39852	48503
NE	0	4637	11945	4176	5286	13820	19650	28861	41057	19814
ENE	167	4038	2981	2229	2459	25796	31297	49275	54071	151948
E	57	674	0	951	1523	17523	28337	77157	262915	241883
ESE	169	245	196	1078	197	6911	43907	170724	42461	67800
SE	618	1765	897	644	455	30161	74109	104047	344940	410760
SSE	112	1773	1224	600	235	14791	113838	759635	1396320	999180
S	595	740	24	121	4308	33258	141735	1073527	2504891	2287550
SSW	148	1512	1199	3996	12262	34983	176103	314591	571684	856500
SW	0	24	1722	3665	5432	14679	93752	194505	144108	192144
WSW	0	353	238	356	161	2775	67192	156533	41998	39068
W	0	213	285	17	0	2851	6628	19307	38104	21349
WNW	0	104	57	0	1005	7053	23692	52686	13223	9609
NW	0	13	81	141	34	856	25635	20533	15319	16495
NNW	0	61	36	4549	1410	1329	55031	32186	16500	16500

Table A5-3

f(θ) Occurrence Frequency
() Mean Wind Speed, (m/sec)

Sectors	Stability Classes					
	A	B	C	D	E	F
N	2.329-02 (0.894)	3.539-03 (0.849)	4.566-03 (1.16)	3.276-02 (1.16)	3.733-02 (0.805)	8.904-03 (0.358)
NNE	1.553-02 (0.983)	2.283-03 (0.939)	2.740-03 (0.894)	3.995-02 (1.03)	5.582-02 (0.894)	2.158-02 (0.487)
NE	3.881-03 (1.12)	1.712-03 (1.07)	3.539-03 (0.805)	2.694-02 (0.805)	4.441-02 (0.849)	2.466-02 (0.602)
ENE	9.132-04 (0.626)	1.142-04 (0.805)	1.027-03 (0.536)	9.932-03 (0.581)	1.941-02 (0.849)	1.039-02 (0.636)
E	6.849-04 (0.581)	0 (—)	3.425-04 (0.626)	5.137-03 (0.447)	1.005-02 (0.447)	2.626-03 (0.465)
ESE	2.283-04 (0.849)	1.142-04 (0.447)	5.708-04 (0.536)	3.539-03 (0.447)	6.507-03 (0.313)	1.712-03 (0.414)
SE	7.991-04 (0.715)	2.283-04 (0.492)	2.283-04 (0.492)	2.626-04 (0.492)	6.164-03 (0.358)	1.484-03 (0.316)
SSE	2.511-03 (0.492)	1.142-04 (0.447)	4.566-04 (0.447)	7.991-03 (0.447)	1.553-02 (0.358)	2.626-03 (0.315)
S	2.957-02 (0.671)	1.370-03 (0.492)	4.338-03 (0.581)	2.306-02 (0.536)	4.760-02 (0.492)	8.105-03 (0.362)
SSW	3.448-02 (0.715)	2.854-03 (0.715)	5.023-03 (0.581)	2.340-02 (0.626)	4.874-02 (0.581)	5.822-03 (0.311)
SW	1.142-02 (0.715)	2.169-03 (0.760)	2.511-04 (0.805)	1.495-02 (0.626)	2.055-02 (0.536)	2.626-03 (0.358)
WSW	3.767-03 (0.671)	6.849-04 (0.626)	2.283-03 (0.626)	5.708-03 (0.626)	1.336-02 (0.492)	3.425-03 (0.349)

Table A5-3 (cont.)

Sectors	Stability Classes					
	A	B	C	D	E	F
W	5.594-03 (0.671)	9.132-04 (0.849)	1.941-03 (0.760)	1.016-02 (0.805)	2.420-02 (0.671)	3.881-03 (0.344)
WNW	3.767-03 (0.939)	1.265-03 (1.07)	2.169-03 (1.21)	1.370-02 (1.21)	1.906-02 (1.07)	1.827-03 (0.363)
NW	6.507-03 (0.939)	6.849-04 (1.25)	3.196-03 (1.25)	2.386-02 (1.48)	3.151-02 (1.61)	1.027-03 (0.363)
NNW	1.005-02 (0.983)	1.370-03 (0.894)	2.854-03 (1.03)	2.032-02 (1.52)	2.249-02 (1.16)	2.397-03 (0.279)

Table A5-4

Calculation of $f(\theta)(X/Q)(x,\theta)$

Sectors	Distances (meters)									
	804.7	2414	4023	5633	7242	12070	24140	40230	56330	72420
N	3.958-05	6.808-06	3.275-06	2.058-06	1.464-06	7.439-07	3.041-07	1.595-07	1.049-07	7.685-08
NNE	6.126-05	1.058-05	5.104-06	3.215-06	2.291-06	1.169-06	4.802-07	2.358-07	1.666-07	1.224-07
NE	5.442-05	9.408-06	4.540-06	2.861-06	2.040-06	1.041-06	4.284-07	2.258-07	1.489-07	1.094-07
ENE	2.572-05	3.971-06	1.915-06	1.205-06	8.583-07	4.371-07	1.793-07	9.427-08	6.208-08	4.556-08
E	1.379-05	2.378-06	1.140-06	7.141-07	5.066-07	2.557-07	1.035-07	5.385-08	3.521-08	2.569-08
ESE	1.152-05	1.986-06	9.515-07	5.958-07	4.226-07	2.132-07	8.624-08	4.484-08	2.930-08	2.137-08
SE	1.030-05	1.775-06	8.532-07	5.355-07	3.806-07	1.929-07	7.853-08	4.103-08	2.690-08	1.967-08
SSE	2.400-05	4.135-06	1.980-06	1.240-06	8.790-07	4.433-07	1.791-07	9.306-08	6.078-08	4.431-08
S	5.748-05	9.864-06	4.722-06	2.956-06	2.096-06	1.058-06	4.278-07	2.225-07	1.454-07	1.061-07
SSW	4.956-05	8.488-06	4.060-06	2.540-06	1.801-06	9.076-07	3.669-07	1.907-07	1.246-07	9.086-08
SW	2.259-05	3.881-06	1.853-06	1.158-06	8.194-07	4.117-07	1.655-07	8.568-08	5.582-08	4.062-08
WSW	1.839-05	3.161-06	1.519-06	9.542-07	6.789-07	3.443-07	1.405-07	7.359-08	4.832-08	3.538-08
W	2.303-05	3.963-06	1.904-06	1.195-06	8.495-07	4.306-07	1.754-07	1.032-07	6.016-08	4.400-08
WNW	1.184-05	2.035-06	9.734-07	6.089-07	4.317-07	2.175-07	8.789-08	4.567-08	2.984-08	2.177-08
NW	1.159-05	1.987-06	9.443-07	5.875-07	4.146-07	2.069-07	8.243-08	4.237-08	2.748-08	1.993-08
NNW	1.549-05	2.661-06	1.276-06	8.003-07	5.684-07	2.876-07	1.170-07	6.111-08	4.007-08	2.931-08

Distances (miles)

A5-9

REFERENCES

1. 1000 MWe Central Station Power Plant Investment Cost Studies - Part 1 through 6. Six volumes, United Engineers and Constructors, Inc., WASH 1230: U.S. Atomic Energy Commission, June 1972 through December 1973.
2. Power Plant Capital Costs Current Trends and Sensitivity to Economic Parameters. United Engineers and Constructors, Inc., WASH 1345: U.S. Atomic Energy Commission, October 1974.
3. Commercial Electric Power Cost Studies. Volume 8, United Engineers and Constructors, Inc., NUREG-0248, U.S. Nuclear Regulatory Commission and/or COO-2477-12, February 1979.
4. E.P. O'Donnell and J.J. Mauro. A Cost-Benefit Comparison of Nuclear and Non-Nuclear Health and Safety Protective Measures and Regulations. Nuclear Safety, Vol. 20, No. 5, September - October 1979.
5. E.P. O'Donnell. The Need for a Cost-Benefit Perspective in the Nuclear Regulatory Process. Trans. Am. Nuc. Soc., 30:482-484, November 1978.
6. F.C. Olds. Regulatory Growth: Impact on Power Plant Planning and Construction. Power Engineering, Vol. 81, No. 5, May 1977.
7. H. Hollingshaus. Status of Nuclear Power Plant Construction and Regulation in the United States. Trans. Am. Nuc. Soc., Vol. 29, September 1978.
8. Richard M. Collins. Lessons Learned in Design and Construction of Nuclear Power Plants in the United States. Presented to the Institute of Nuclear Engineers, London, England, October 29, 1981.
9. Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. WASH 1400, U.S. Nuclear Regulatory Commission, October 1975.
10. Executive Seminar on the Future Role of Risk Assessment and Reliability Engineering in Nuclear Regulation. NUREG/CP-0017, U.S. Nuclear Regulatory Commission, January 28, 1980.
11. Reactor Safety Study Methodology Applications Program: Sequoyah #1 PWR Power Plant. NUREG/CR-1659, Vol. 1, U.S. Nuclear Regulatory Commission, April 1981.
12. Reactor Safety Study Methodology Applications Program: Oconee #3 PWR Power Plant. Sandia National Laboratories, NUREG/CR-1659, Revision 1, U.S. Department of Energy, May 1981.
13. Crystal River-3 Safety Study Volume I - Main Report and Volume II - Appendices. Science Applications, Inc., NUREG/CR-2515, U.S. Nuclear Regulatory Commission, December 1981.

REFERENCES, (Continued)

14. Zion Probabilistic Safety Study, Commonwealth Edison Company, Initial Issue, 1981.
15. Indian Point Probabilistic Safety Study, Power Authority of the State of New York and the Consolidated Edison Company of New York, Initial Issue, 1982.
16. W.W. Weaver. Deterministic Criteria Versus Probabilistic Analysis: Examining the Single Failure and Separation Criteria. Nuclear Technology, Vol. 49, February 1980.
17. W.W. Weaver. Pitfalls in Current Design Requirements. Nuclear Safety, Vol. 22, No. 3, May - June 1981.
18. Value - Impact Analysis. Science Applications, Inc., EPRI NP-1237, Electric Power Research Institute, November 1979.
19. Safety Goals for Nuclear Power Plants: A Discussion Paper. NUREG-0880, Office of Policy Evaluation, U.S. Nuclear Regulatory Commission, February 1982.
20. Phase IV Final Report and Supplements - Fourth Update of the Energy Economic Data Base Program. C00-5709-1, United Engineers and Constructors for the United States Department of Energy, November 1981.
21. Full-Scale Tornado-Missile Impact Tests. Sandia Laboratories, EPRI NP-440, Electric Power Research Institute, July 1977.
22. Pipe Missile Impact Experiments on Concrete Models. SRI International, EPRI NP-1883, Electric Power Research Institute, June 1981.
23. Tornado Missile Risk Analysis. Carolina Power and Light Company, EPRI NP-768, Electric Power Research Institute, May 1978.
24. Tornado Missile Risk Analysis - Appendices. Carolina Power and Light Company, EPRI NP-769, Electric Power Research Institute, May 1978.
25. Tornado Missile Simulation and Design Methodology Volume 1: Simulation Methodology, Design Applications, and TORMIS Computer Code. Research Triangle Institute, EPRI NP-2005, Electric Power Research Institute, August 1981.
26. A.K. Bhattacharyya, R.C. Burt, P.K. Nuyosi. Characteristics of Tornado Generated Missiles. United Engineers and Constructors, UEC-TR-002-0, October 1975.
27. D.R. Miller, W.A. Williams, A.P. Brog. Tornado Protection for the Spent Fuel Pool. General Electric Company, APED-5696, November 1968.
28. Integrated Plant Safety Assessment - Systematic Evaluation Program - R.E. Ginna Nuclear Power Plant. Rochester Gas and Electric Corporation, NUREG-0821, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, May 1982.

REFERENCES, (Continued)

29. Integrated Plant Safety Assessment - Systematic Evaluation Program - Palisades Plant. Consumers Power Company. NUREG-0820, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, April 1982.
30. S. H. Bush. Reliability of Piping in Light - Water Reactors. Nuclear Safety, Vol. 17, No. 5, September - October 1976.
31. Study of the State of Design for Pipe Whip. Tennessee Valley Authority, EPRI NP 1320, Electric Power Research Institute, January 1980.
32. Analysis of Pipe Whip. Hibbitt and Karlsson, Inc., EPRI NP 1208, Electric Power Research Institute, November 1979.
33. Dynamic Piping Fracture Coupled With System Effects. University of Washington, EPRI NP 1640, Electric Power Research Institute, December 1980.
34. Dynamic Crack Propagation in Pipes. University of Washington, EPRI NP 1742, Electric Power Research Institute, March 1981.
35. Selected Review of Foreign Licensing Practices for Nuclear Power Plants, Structural Mechanics Associates, NUREG/CR-2664, U.S. Nuclear Regulatory Commission, April 1982.